Facility: **Browns Ferry NPP** Scenario No.: **NRC - 1** Op-Test No.: **1108**

Examiners:______________ Operators: **SRO:**______________

______________ **ATC:**______________

______________ **BOP:**______________

Initial Conditions: SLC pump 2B and EECW Pump A3 out of service. HPCI surveillance testing has just been completed and Torus cooling is to be secured. Reactor Power is 76%.

Turnover: Secure RHR Pump 2A from Torus cooling. Commence a power increase to 100%.

Event No.	Malf. No.	Event Type*	Event Description
1		N-BOP N-SRO	Secure Torus Cooling lineup IAW 2-OI-74 Section 8.6
2	SW3j	C-BOP TS-SRO	RHR/SW pump C3 trip
3		R-ATC R-SRO	Commence a power increase with rods
4	fic-85-11 0-100(L)	C-ATC C-SRO	CRD Controller Failure
5	Batch file	C-BOP C-SRO	Steam Packing Exhauster 2A trip and failure 2B discharge valve
6	CU04	I-ATC TS-SRO	RWCU Leak with failure to Auto isolate
7	PC14	M-ALL	Non-isolable leak on torus
8	IOR	C	HPCI minimum flow valve will not open
9	IOR	C	All SRVs except 3 fail to open for Emergency Depressurization

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Events

1. BOP shutdowns RHR Loop 1 from suppression pool cooling, IAW 2-OI-74 RHR System section 8.6

2. EECW Pump C3 trip, BOP will align RHRSW Pump C1 for EECW and start C1 Pump to restore EECW flow to the south header, IAW ARPs and 0-OI-67 EECW System section 8.3. The SRO will evaluate Technical Specification 3.7.2 and Condition A. When the C1 RHRSW Pump is aligned for EECW, then evaluate Technical Specification 3.7.1 and Condition A.

3. ATC will commence to raise power with control rods

4. CRD Controller fails, ATC takes manual control of controller and restores CRD parameters

5. Steam Packing Exhauster will trip and the STBY Exhauster will Start but the discharge damper will fail to open. The BOP will open the Steam Packing Exhauster discharge damper and restore Steam Packing Exhauster operation IAW with ARPs.

6. The ATC will respond to RWCU alarms indicating a leak and RWCU will fail to isolate. The ATC will isolate RWCU and take actions IAW 2-AOI-64-2A. The SRO will evaluate Technical Specification 3.6.1.3 Condition A is required. The SRO will evaluate TRM 3.4.1 and direct Chemistry to sample in order to satisfy TSR 3.4.1.1.

7. An unisolable Torus leak will commence. Suppression Pool level will start to lower and continue to lower. The SRO will enter EOI-3 on flood alarms and eventually EOI-2 on Suppression Pool Level. The crew will place HPCI in pull to lock prior to Torus level lowering to less than 12.75 feet.
   The SRO will determine that Suppression Pool level cannot be maintained above 11.5 feet and enter EOI-1 to scram the reactor and then transition to Emergency Depressurize. SRO will evaluate Technical Specification 3.6.2.2 Condition A

8. 2-FCV-73-30, HPCI MIN FLOW VALVE will fail to open. The crew will open the RCIC CST SUCTION VALVE and RCIC PUMP MIN FLOW VALVE to establish makeup to the Torus.

9. 11 SRVs fail on ED, with less than 4 MSRVs open the crew will try to rapidly depressurize the RPV with systems listed in C2-12 of 2-EOI-2-C-2, Emergency RPV Depressurization.

Terminate the scenario when the following conditions are satisfied or upon request of Lead Examiner:

Control Rods are inserted

Emergency Depressurization complete

Reactor Level is restored and maintained

Critical Tasks - Three

CT#1-When Suppression Pool level cannot be maintained above 11.5 feet the US directs the Reactor scrammed and either 1) anticipates Emergency Depressurization and depressurizes using bypass valves or 2) directs Emergency Depressurization before suppression pool level lowers to 11.5 feet.

1. Safety Significance:
   Precludes failure of Containment

2. Cues:
   Procedural compliance
   Suppression Pool level trend

3. Measured by:
   Observation - US determines (indicated by announcement or observable transition to C-2) that Emergency Depressurization is required before Suppression Pool level drops below 11.5 feet.
   AND
   Observation - RO opens at least 6 SRV's during performance of Emergency Depressurization actions.

4. Feedback:
   RPV pressure trend
   Suppression Pool temperature trend
   SRV status indication

CT#2-When Suppression Pool Level cannot be maintained above 12.75 feet HPCI secured to prevent damage prior to reaching 12.75 feet.

1. Safety Significance:
   Prevent failure of Primary Containment from pressurization of the Suppression Chamber

2. Cues:
   Procedural compliance
   Suppression Pool Level indication

3. Measured by:
   Observation – HPCI Auxiliary Pump placed in Pull to Lock

4. Feedback:
   HPCI does not Auto initiate
   No RPM indication on HPCI

CT#3-With a primary system discharging into the secondary containment, take action to manually isolate the break.

1. Safety Significance:
   Isolating high energy sources can preclude failure of secondary containment and subsequent radiation release to the public.

2. Cues:
   Procedural compliance
   Area temperature indication

3. Measured by:
   With the reactor at pressure and a primary system discharging into the secondary containment, operator takes action to manually isolate the break.

4. Feedback:
   Valve position indication

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER: 1

7 Total Malfunctions Inserted: List (4-8)

2 Malfunctions that occur after EOI entry: List (1-4)

4 Abnormal Events: List (1-3)

1 Major Transients: List (1-2)

3 EOI's used: List (1-3)

1 EOI Contingencies used: List (0-3)

75 Validation Time (minutes)

3 Crew Critical Tasks: (2-5)

YES Technical Specifications Exercised (Yes/No)

Scenario Tasks

TASK NUMBER	K/A	RO	SRO
Shutdown Suppression Pool Cooling			
RO U-92B-NO-05	219000A4.01	3.8	3.7
EECW Pump Trip			
RO U-067-NO-12	400000A2.01	3.3	3.4
Raise Power with Control Rods			
RO U-085-NO-07 SRO S-000-AD-31	2.2.2	4.6	4.1
CRD Controller Failure			
RO U-085-AB-03	201001A3.01	3.0	3.0
Steam Packing Exhauster Trip			
RO U-47C-AL-2 SRO S-047-AB-3	271000A1.01	3.3	3.2
RWCU Leak with Failure to Auto Isolate			
RO U-069-AL-10 SRO S-000-EM-12	223002A2.03	3.0	3.3
Torus Leak			
RO U-000-EM-7 RO U-000-EM-17 RO U-000-EM-83 SRO S-000-EM-07 SRO S-000-EM-15	295030EA2.01	4.1	4.2

Simulator Instructor

1. Setup IC-90

110801 Preference File
F3 bat NRC/110801
F4 imf sw03j
F5 mrf sw06 close
F6 trg! E11
F7 bat NRC/110202-1
F8 imf pc14 100 360 10

110801 Batch File

#RWCU seal leak no auto iso
imf cu06
imf cu04 (e1 0) 100 300 50

10 SRV overrides
Trg e3 NRC/singleelement
Trg e3 = bat NRC/110202-4
Ior zdihs7330a close
Oir ypobkrrhrswpa3 fail_ccoil
Ior zlohs2385a[1] off
ior zlohs6635a[1] on
ior ypomtrspea (e11 0) fail_control_power
ior ypovfcv6635 (e11 0) fail_power_now
trg 10 NRC/spe
trg 10 = bat NRC/110801-1

#Steam packing blower trip 110801-1
Dor ior ypovfcv6635
Dor ior zlohs6635a

TRG SPE
Zdihs6635a[3]. Eq. 1

Facility: **Browns Ferry NPP** Scenario No.: **NRC - 1** Op-Test No.: **1108**

Examiners:____________________ Operators: **SRO:**____________________

____________________ **ATC:**____________________

____________________ **BOP:**____________________

Initial Conditions: SLC pump 2B and EECW Pump A3 out of service. HPCI surveillance testing has just been completed and Torus cooling is to be secured. Reactor Power is 76%.

Turnover: Secure RHR Pump 2A from Torus cooling. Commence a power increase to 100%.

Event No.	Malf. No.	Event Type*	Event Description
1		N-BOP N-SRO	Secure Torus Cooling lineup IAW 2-OI-74 Section 8.6
2	SW3j	C-BOP TS-SRO	RHR/SW pump C3 trip
3		R-ATC R-SRO	Commence a power increase with rods
4	fic-85-11 0-100(L)	C-ATC C-SRO	CRD Controller Failure
5	Batch file	C-BOP C-SRO	Steam Packing Exhauster 2A trip and failure 2B discharge valve
6	CU04	I-ATC TS-SRO	RWCU Leak with failure to Auto isolate
7	PC14	M-ALL	Non-isolable leak on torus
8	IOR	C	HPCI minimum flow valve will not open
9	IOR	C	All SRVs except 3 fail to open for Emergency Depressurization

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Critical Tasks - Three

CT#1-When Suppression Pool level cannot be maintained above 11.5 feet the US directs the Reactor scrammed and either 1) anticipates Emergency Depressurization and depressurizes using bypass valves or 2) directs Emergency Depressurization before suppression pool level lowers to 11.5 feet.

1. Safety Significance:
   Precludes failure of Containment

2. Cues:
   Procedural compliance
   Suppression Pool level trend

3. Measured by:
   Observation - US determines (indicated by announcement or observable transition to C-2) that Emergency Depressurization is required before Suppression Pool level drops below 11.5 feet.
   AND
   Observation - RO opens at least 6 SRV's during performance of Emergency Depressurization actions.

4. Feedback:
   RPV pressure trend
   Suppression Pool temperature trend
   SRV status indication

CT#2-When Suppression Pool Level cannot be maintained above 12.75 feet HPCI secured to prevent damage prior to reaching 12.75 feet.

1. Safety Significance:
   Prevent failure of Primary Containment from pressurization of the Suppression Chamber

2. Cues:
   Procedural compliance
   Suppression Pool Level indication

3. Measured by:
   Observation – HPCI Auxiliary Pump placed in Pull to Lock

4. Feedback:
   HPCI does not Auto initiate
   No RPM indication on HPCI

CT#3-With a primary system discharging into the secondary containment, take action to manually isolate the break.

1. Safety Significance:
   Isolating high energy sources can preclude failure of secondary containment and subsequent radiation release to the public.

2. Cues:
   Procedural compliance
   Area temperature indication

3. Measured by:
   With the reactor at pressure and a primary system discharging into the secondary containment, operator takes action to manually isolate the break.

4. Feedback:
   Valve position indication

Events

1. BOP shutdowns RHR Loop 1 from suppression pool cooling, IAW 2-OI-74 RHR System section 8.6

2. EECW Pump C3 trip, BOP will align RHRSW Pump C1 for EECW and start C1 Pump to restore EECW flow to the south header, IAW ARPs and 0-OI-67 EECW System section 8.3. The SRO will evaluate Technical Specification 3.7.2 and Condition A. When the C1 RHRSW Pump is aligned for EECW, then evaluate Technical Specification 3.7.1 and Condition A.

3. ATC will commence to raise power with control rods

4. CRD Controller fails, ATC takes manual control of controller and restores CRD parameters

5. Steam Packing Exhauster will trip and the STBY Exhauster will Start but the discharge damper will fail to open. The BOP will open the Steam Packing Exhauster discharge damper and restore Steam Packing Exhauster operation IAW with ARPs.

6. The ATC will respond to RWCU alarms indicating a leak and RWCU will fail to isolate. The ATC will isolate RWCU and take actions IAW 2-AOI-64-2A. The SRO will evaluate Technical Specification 3.6.1.3 Condition A is required. The SRO will evaluate TRM 3.4.1 and direct Chemistry to sample in order to satisfy TSR 3.4.1.1.

7. An unisolable Torus leak will commence. Suppression Pool level will start to lower and continue to lower. The SRO will enter EOI-3 on flood alarms and eventually EOI-2 on Suppression Pool Level. The crew will place HPCI in pull to lock prior to Torus level lowering to less than 12.75 feet.
   The SRO will determine that Suppression Pool level cannot be maintained above 11.5 feet and enter EOI-1 to scram the reactor and then transition to Emergency Depressurize. SRO will evaluate Technical Specification 3.6.2.2 Condition A

8. 2-FCV-73-30, HPCI MIN FLOW VALVE will fail to open. The crew will open the RCIC CST SUCTION VALVE and RCIC PUMP MIN FLOW VALVE to establish makeup to the Torus.

9. 11 SRVs fail on ED, with less than 4 MSRVs open the crew will try to rapidly depressurize the RPV with systems listed in C2-12 of 2-EOI-2-C-2, Emergency RPV Depressurization.

Terminate the scenario when the following conditions are satisfied or upon request of Lead Examiner:

Control Rods are inserted

Emergency Depressurization complete

Reactor Level is restored and maintained

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER: 1

7 Total Malfunctions Inserted: List (4-8)

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4 Abnormal Events: List (1-3)

1 Major Transients: List (1-2)

3 EOI's used: List (1-3)

1 EOI Contingencies used: List (0-3)

75 Validation Time (minutes)

3 Crew Critical Tasks: (2-5)

YES Technical Specifications Exercised (Yes/No)

Scenario Tasks

TASK NUMBER	K/A	RO	SRO
Shutdown Suppression Pool Cooling			
RO U-92B-NO-05	219000A4.01	3.8	3.7
EECW Pump Trip			
RO U-067-NO-12	400000A2.01	3.3	3.4
Raise Power with Control Rods			
RO U-085-NO-07 SRO S-000-AD-31	2.2.2	4.6	4.1
CRD Controller Failure			
RO U-085-AB-03	201001A3.01	3.0	3.0
Steam Packing Exhauster Trip			
RO U-47C-AL-2 SRO S-047-AB-3	271000A1.01	3.3	3.2
RWCU Leak with Failure to Auto Isolate			
RO U-069-AL-10 SRO S-000-EM-12	223002A2.03	3.0	3.3
Torus Leak			
RO U-000-EM-7 RO U-000-EM-17 RO U-000-EM-83 SRO S-000-EM-07 SRO S-000-EM-15	295030EA2.01	4.1	4.2

Procedures Used/Referenced:

Procedure Number	Procedure Title	Procedure Revision
2-OI-74	Residual Heat Removal System	Rev 156
2-ARP-9-20A, W35	EECW South HDR DG Section Pressure Low	Rev 25
0-OI-67	Emergency Equipment Cooling Water System	Rev 91
TS 3.7.2	Emergency Equipment Cooling Water (EECW) System and Ultimate Heat Sink (UHS)	Amd 254
TS 3.7.1	Residual Heat Removal Service Water (RHRSW) System and Ultimate Heat Sink (UHS)	Amd 254
2-GOI-100-12	Power Maneuvering	Rev 12
2-OI-85	Control Rod Drive System	Rev 128
2-ARP-9-5A, W10	CRD Accumulator Charging Water Header Pressure Hi	Rev 48
2-ARP-9-7A, W12	Steam Packing Exhauster Vacuum Low	Rev 27
2-OI-47C	Seal Steam System	Rev 24
2-ARP-9-3D, W17	RWCU Leak Detection Temperature High	Rev 28
2-AOI-64-2A	Group 3 RWCU Isolation	Rev 25
TS 3.6.1.3	Primary Containment Isolation Valves	Amd 212
TRM 3.4.1	Coolant Chemistry	Rev 21
2-ARP-9-3B, W15	Suppression Chamber Water Level Abnormal	Rev 28
2-EOI-3	Secondary Containment Control	Rev 12
2-EOI-2	Primary Containment Control	Rev 12
2-EOI-App-18	Suppression Pool Water Inventory Removal and Makeup	Rev 8
2-EOI-1	RPV Control Flowchart	Rev 12
2-AOI-100-1	Reactor Scram	Rev 95
2-EOI-App-5A	Injection Systems Lineup Condensate/Feedwater	Rev 9
2-EOI-App-6A	Injection Subsystems Lineup Condensate	Rev 4
2-EOI-2-C-2	Emergency RPV Depressurization	Revision 6
2-EOI-App-11H	Alternate RPV Pressure Control Systems Main Condenser	Rev 6
EPIP-1	Emergency Classification Procedure	Revision 46
EPIP-4	Site Area Emergency	Revision 32

Console Operator Instructions

A. Scenario File Summary

110801 Preference File
F3 bat NRC/110801
F4 imf sw03j
F5 mrf sw06 close
F6 trg! E11
F7 bat NRC/110202-1
F8 imf pc14 100 360 10

110801 Batch File

#RWCU seal leak no auto iso
imf cu06
imf cu04 100 300 50

10 SRV overrides
Trg e3 NRC/singleelement
Trg e3 = bat NRC/110202-4
Ior zdihs7330a close
Oir ypobkrrhrswpa3 fail_ccoil
Ior zlohs2385a[1] off
ior zlohs6635a[1] on
ior ypomtrspea (e11 0) fail_control_power
ior ypovfcv6635 (e11 0) fail_power_now
trg 10 NRC/spe
trg 10 = bat NRC/110801-1

#Steam packing blower trip 110801-1
Dor ior ypovfcv6635
Dor ior zlohs6635a

TRG SPE
Zdihs6635a[3]. Eq. 1

Scenario 1

		DESCRIPTION/ACTION
Simulator Setup	manual	Reset to IC 90
Simulator Setup	**Load Batch**	RestorePref NRC/110801
Simulator Setup	manual	F3
Simulator Setup	manual	Tag SLC pump B and EECW pump A3
Simulator Setup		Verify file loaded

RCP required (76% - 100% with control rods and flow) and RCP for Urgent Load Reduction
Provide marked up copy of 2-GOI-100-12

Simulator Event Guide:

Event 1 Normal: Secure Torus Cooling lineup

	SRO	Directs securing Torus cooling lineup IAW 2-OI-74, section 8.6
	BOP	Secures Torus cooling lineup
		8.6 Shutdown of Loop I(II) Suppression Pool Cooling **NOTE** 1) All operations are performed at Panel 2-9-3 unless otherwise noted. 2) RHR flow should be monitored while in operation with multiple flow paths (e.g., LPCI and Suppression Pool Cooling together, etc.). During any evolution, total system flow as indicated on RHR SYSTEM I(II) FLOW, 2-FI-74-50(64), should remain between 7,000 to 10,000 gpm for 1 pump operation or between 10,000 and 20,000 gpm for 2-pump operation. [1] **VERIFY** Suppression Pool Cooling in operation. **REFER TO** Section 8.5. [2] **REVIEW** the precautions and limitations in Section 3.0. [3] **NOTIFY** Radiation Protection of Suppression Pool Cooling loop removed from service. **RECORD** name and time of Radiation Protection representative notified in NOMS narrative log.
	Driver	As Radiation Protection, acknowledge removing Suppression Pool Cooling from service
	BOP	**CAUTIONS** 1) To prevent draining an RHR Loop, at least one of the RHR System test valves must be closed before stopping RHR Pumps in the associated loop. 2) To prevent excessive vibration, RHR pumps should not be allowed to operate for more than 3 minutes at minimum flow. 3) When closing throttle valve RHR SYS I(II) SUPPR POOL CLG/TEST VLV, 2-FCV-74-59 and 2-FCV-74-73 from the control room, the handswitch should be held in the close position for approximately 6 seconds after the red light extinguishes. Failure to completely close these valves could provide a leak path to the suppression pool from the RHR discharge piping. [4] **IF** both RHR Pumps in Loop I(II) are in operation **AND** one pump is to be removed from service due to reduced heat load, **THEN**: [4.1] **THROTTLE** RHR SYS I(II) SUPPR POOL CLG/TEST VLV, 2-FCV-74-59(73), to obtain a flow of between 7,000 to 10,000 gpm and Blue light illuminated as indicated on RHR SYS I(II) FLOW, 2-FI-74-50(64). [4.2] **STOP** RHR PUMP 2A(2B) or 2C(2D) using 2-HS-74-5A(28A) or 16A(39A).

Simulator Event Guide:

Event 1 Normal: Secure Torus Cooling lineup

	BOP	[4.3] **CLOSE** associated RHR HX 2A(2B) or 2C(2D) RHRSW OUTLET VALVE, 2-FCV-23-34(46) or 40(52). [4.4] **IF** RHRSW for the Heat Exchanger removed from service is not required to support other unit operations, **THEN STOP** RHRSW pump for the Heat Exchanger removed from service.
	Driver	When contacted as other unit, RHRSW HX is not required
	BOP	[5] **CLOSE** RHR SYS I(II) SUPPR POOL CLG/TEST VLV, 2-FCV-74-59(73). [6] **WHEN** RHR SYS I(II) SUPPR POOL CLG/TEST VLV, 2-FCV-74-59(73) is CLOSED, **THEN STOP** RHR PUMPS 2A(2B) or 2C(2D) using 2-HS-74-5A(28A) and/or 16A(39A). [7] **CLOSE** RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV, 2-FCV-74-57(71). [8] **CLOSE** RHR HX(s) 2A(2B) and 2C(2D) RHRSW OUTLET VLV(s), 2-FCV-23-34(46) and 40(52). [9] **IF** RHRSW for RHR Heat Exchanger(s) A(B) and C(D) is not required to support other unit operations, **THEN STOP** RHRSW Pump(s) for the Heat Exchanger(s) removed from service. [10] **CHECK** RHR System discharge header pressure is greater than TRM 3.5.4 limit as indicated on 2-PI-74-51(65), RHR SYS I(II) DISCH PRESS.
	Driver	When contacted as other unit, RHRSW HX is not required
	Driver	At NRC direction, insert F4 (imf sw03j), EECW pump C3 trip

Simulator Event Guide:

Event 2 Component: EECW pump C3 trip

	BOP	Respond to alarm 20A-35.
		20A-35 EECW SOUTH HDR DG SECTION PRESS LOW B. **CHECK** Panel 2-9-3 for status of North header pump(s) breaker lights and pump motor amps normal. C. **NOTIFY** UNIT SUPERVISOR, Unit 1 and Unit 3. D. **START** standby EECW Pump for affected header, if available. H. **IF** pump failure is cause of alarm, **THEN REFER TO** Tech Spec 3.7.2.
	DRIVER	If contacted, as Unit 3 Operator, inform that 4KV SD BD 3EB received a Motor Overload or Trip alarm If contacted as Unit 1 operator, you did not secure the C3 EECW Pump
		8.3 Operation of RHRSW Pump C1 (for EECW in place of C3) **CAUTION** Only one RHRSW pump in a given RHRSW pump room may be counted toward meeting Technical Specification 3.7.2 requirements for EECW pump operability. **NOTES** 1) RHRSW Pump C1 may be aligned for service by this section when: • It is used to meet the minimum number of Tech. Spec. operable pumps; or • At the discretion of the Unit Supervisor, it is needed to replace another pump's operation; or • At the discretion of the Unit Supervisor, it is needed to assist in supplying header flow/pressure demand. 2) If used to meet EECW requirements, RHRSW pump C1 must be aligned to EECW, the pump started, and should remain running. RHRSW Pump C1 does **NOT** have the same auto start signals as RHRSW Pump C3. 3) The RHRSW pump control switches and amp meters are located at Control Room Panel 9-3, Unit 1, 2, and 3. 4) When RHRSW Pump C1 is aligned for EECW, its RHRSW function required by the Safe Shutdown Program (Appendix R) is inoperable. Appendix R program equipment operability requirements of FPR-Volume 1 shall be addressed. [1] To line up RHRSW Pump C1 for EECW System operation, **PERFORM** the following: [1.1] **VERIFY** EECW System is in prestartup/standby readiness alignment in accordance with Section 4.0. [1.2] **REVIEW** all precautions and limitations in Section 3.0. [1.3] **VERIFY** RHRSW Pump C1 is in standby readiness in accordance with 0-OI-23.

Simulator Event Guide:

Event 2 Component: EECW pump C3 trip

		8.3 Operation of RHRSW Pump C1 (for EECW in place of C3) (contd) [1.4] **VERIFY** RHRSW Pump C1 upper and lower motor bearing oil level is in the normal operating range. [1.5] **UNLOCK** and **CLOSE** RHRSW PMP C1 & C2 CROSSTIE, 0-23-544 at RHRSW C Room. [1.6] **OPEN** RHRSW PMP C1 CROSSTIE TO EECW, 0-FCV-67-49 using one of the following: • RHRSW PMP C1 CROSSTIE TO EECW, 0-HS-67-49A/1 on Unit 1 • RHRSW PUMP C1 SUPPLY TO EECW, 0-HS-67-49A/2 on Unit 2 • RHRSW PUMP C1 SUPPLY TO EECW, 0-HS-67-49A/3 on Unit 3 [1.7] **REQUEST** a caution order be issued to tag RHRSW Pump C1 and its associated crosstie valves to inform Operations personnel that it is aligned for EECW system operation and that the C1 pump should remain running to be operable for EECW. [2] To start RHRSW (EECW) Pump C1, **PERFORM** the following: [2.1] **START** RHRSW Pump C1 using one of the following: • RHRSW PUMP C1, 0-HS-23-8A/1 on Unit 1 • RHRSW PUMP C1, 0-HS-23-8A/2 on Unit 2 • RHRSW PUMP C1, 0-HS-23-8A/3 on Unit 3 [2.2] **VERIFY** RHRSW Pump C1 running current is less than 53 amps using one the following: • RHRSW PUMP C1 AMPS, 0-EI-23-8/1 on Unit 1 • RHRSW PUMP C1 AMPS, 0-EI-23-8/2 on Unit 2 • RHRSW PUMP C1 AMPS, 0-EI-23-8/3 on Unit 3 [2.3] **VERIFY** locally, RHR SERVICE WATER PUMP C1 breaker charging spring recharged by observing amber breaker spring charged light is on and closing spring target indicates charged. [2.4] **VERIFY** RHRSW Pump C1 upper and lower motor bearing oil level is in the normal operating range.
	Driver	If dispatched to check C3 EECW pump breaker, report breaker tripped on overload and breaker smells burnt but no visible smoke or flames (3EB 4kv SD BD)

Simulator Event Guide:

Event 2 Component: EECW pump C3 trip

		8.3 Operation of RHRSW Pump C1 (for EECW in place of C3) (contd) [2.5] **NOTIFY** Chemistry of running RHRSW (EECW) pump(s). [2.6] **VERIFY** a caution order has been issued to tag RHRSW Pump C1 and its associated crosstie valves to inform Operations personnel that it is aligned for EECW system operation and that the C1 pump should remain running to be operable for EECW.
	Driver	When chemistry contacted, acknowledge report When contacted as Work Control for Caution Order, acknowledge direction and inform will begin working on a Caution Order When dispatched as intake AUO to check Oil Levels and close 0-23-544 valve wait 2 minutes and insert F5 (mrf sw06 close), then report oil levels are normal and the 0-23-544 valve is closed When contacted to check breaker charging spring recharged for the C1 EECW pump, wait 2 minutes and inform amber breaker spring charged light is on and closing spring target indicates charged. When contacted as Intake AUO for second Oil Level check, report Oil Levels are normal
	SRO	Evaluate Technical Specification 3.7.2 before the C1 EECW Pump is aligned
		Condition A: One of three (3) ~~Two~~ required EECW pumps inoperable. (A3 and C3) Required Action A.1 ~~A.2~~: ~~Be in Mode 3 in 12 hrs, Mode 4 in 36 hrs~~ Restore in 7 days
	SRO	Evaluate Technical Specification 3.7.1 after the C1 EECW Pump is aligned
		Condition A: One required RHRSW pump inoperable Required Action A.1: Restore required RHRSW pump to OPERABLE status. Completion Time: 30 days

BK

Simulator Event Guide:

Event 3 Reactivity: Raise Power with Control Rods

	SRO	Notify ODS of power increase.
		Direct Power increase using control rods per 2-GOI-100-12. [21] **WHEN** desired to restore Reactor power to 100%, **THEN PERFORM** the following as directed by Unit Supervisor and recommended by the Reactor Engineer: • **RAISE** power using control rods or core flow changes. **REFER TO** 2-SR-3.3.5(A) and 2-OI-68. • **MONITOR** Core thermal limits using ICS, and/or 0-TI-248
	ATC	Raise Power with Control Rods per 2-OI-85, section 6.6. Control Rods: 22-31…00 to 12 30-31…00 to 48 14-31…00 to 16 22-39…00 to 16 30-39…00 to 12 30-15…00 to 16 22-23…00 to 16 38-21…00 to 12 46.31…00 to 16 38.23…00 to 16 30-23…00 to 12 30-47…00 to 16 38-39…00 to 16
		NOTES Continuous control rod withdrawal may be used when a control rod is to be withdrawn greater than three notches. When continuously withdrawing a control rod to a position other than position 48, the CRD Notch Override Switch is held in the Override position and then the CRD Control Switch is held in the Rod Out Notch position. • Both switches should be released when the control rod reaches two notches prior to its intended position.
		6.6.1 Initial Conditions Prior to Withdrawing Control Rods [2] **VERIFY** the following prior to control rod movement: • CRD POWER, 2-HS-85-46 in ON. • Rod Worth Minimizer is operable and LATCHED into the correct ROD GROUP when Rod Worth Minimizer is enforcing (not required with no fuel in RPV). **6.6.2 Actions Required During and Following Control Rod Withdrawal** [4] **OBSERVE** the following during control rod repositioning: • Control rod reed switch position indicators (four rod display) agree with the indication on the Full Core Display. • Nuclear Instrumentation responds as control rods move through the core. (This ensures control rod is following drive during Control Rod movement.) [5] **ATTEMPT** to minimize automatic RBM Rod Block as follows: • **STOP** Control Rod withdrawal (if possible) prior to reaching any RBM Rod Block using the RBM displays on Panel 2-9-5 and **PERFORM** Step 6.6.2[6].

Simulator Event Guide:

Event 3 Reactivity: Raise Power with Control Rods

		[6] **IF** Control Rod movement was stopped to keep from exceeding a RBM setpoint or was caused by a RBM Rod Block, **THEN** **PERFORM** the following at the Unit Supervisor's discretion to "REINITIALIZE" the RBM: [6.1] **PLACE** CRD POWER, 2-HS-85-46 in the OFF position to deselect the Control Rod. [6.2] **PLACE** CRD POWER, 2-HS-85-46, in the ON position.
	ATC	**6.6.3 Control Rod Notch Withdrawal** [1] **SELECT** the desired control rod by depressing the appropriate CRD ROD SELECT pushbutton, 2-XS-85-40. [2] **OBSERVE** the following for the selected control rod: • CRD ROD SELECT pushbutton is brightly ILLUMINATED. • White light on the Full Core Display ILLUMINATED. • Rod Out Permit light ILLUMINATED. [3] **VERIFY** Rod Worth Minimizer is operable and LATCHED into the correct ROD GROUP when the Rod Worth Minimizer is enforcing. [4] **PLACE** CRD CONTROL SWITCH, 2-HS-85-48, in ROD OUT NOTCH, and **RELEASE**. [5] **OBSERVE** the control rod settles into the desired position and the ROD SETTLE light extinguishes.
	Driver	At NRC direction, Manually enter CRDH controller failure fic-85-11 0-100(L)

Simulator Event Guide:

Event 3 Reactivity: Raise Power with Control Rods

	ATC	**6.6.4 Continuous Rod Withdrawal** [1] **SELECT** desired Control Rod by depressing appropriate CRD ROD SELECT, 2-XS-85-40. [2] **OBSERVE** the following for the selected control rod: • CRD ROD SELECT pushbutton is brightly ILLUMINATED. • White light on the Full Core Display ILLUMINATED. • Rod Out Permit light ILLUMINATED. [3] **VERIFY** Rod Worth Minimizer operable and LATCHED into correct ROD GROUP when the Rod Worth Minimizer is enforcing. [4] **VERIFY** Control Rod is being withdrawn to a position greater than three notches. [5] **IF** withdrawing the control rod to a position other than "48", **THEN** **PERFORM** the following: (Otherwise N/A) [5.1] **PLACE AND HOLD** CRD NOTCH OVERRIDE, 2-HS-85-47, in NOTCH OVERRRIDE. [5.2] **PLACE AND HOLD** CRD CONTROL SWITCH, 2-HS-85-48, in ROD OUT NOTCH. [5.3] **WHEN** control rod reaches two notches prior to the intended notch, **THEN** **RELEASE** CRD NOTCH OVERRIDE, 2-HS-85-47 and CRD CONTROL SWITCH, 2-HS-85-48. [5.4] **IF** control rod settles at notch before intended notch, **THEN** **PLACE** CRD CONTROL SWITCH, 2-HS-85-48, in ROD OUT NOTCH and **RELEASE**.

Simulator Event Guide:

Event 3 Reactivity: Raise Power with Control Rods

	ATC	**6.6.4 Continuous Rod Withdrawal (Continued)** [5.5] **WHEN** control rod settles into the intended notch, **THEN** **CHECK** the following. • Four rod display digital readout and the full core display digital readout and background light remain illuminated. • CONTROL ROD OVERTRAVEL annunciator, 2-XA-55-5A, Window 14, does **NOT** alarm. [5.6] **CHECK** the control rod settles at intended position and ROD SETTLE light extinguishes. [6] **IF** continuously withdrawing the control rod to position 48 and performing the control rod coupling integrity check in conjunction with withdrawal, **THEN** **PERFORM** the following: (Otherwise N/A) [6.1] **PLACE** and **HOLD** CRD NOTCH OVERRIDE, 2-HS-85-47, in NOTCH OVERRRIDE. [6.2] **PLACE** and **HOLD** CRD CONTROL SWITCH, 2-HS-85-48, in ROD OUT NOTCH. [6.3] **MAINTAIN** the CRD Notch Override Switch in the Override position and the CRD Control Switch in the Rod Out Notch position, with the control rod at position 48. [6.4] **CHECK** control rod coupled by observing the following: • Four rod display digital readout and the full core display digital readout and background light remain illuminated. • CONTROL ROD OVERTRAVEL annunciator, 2-XA-55-5A, Window 14, does not alarm. [6.5] **RELEASE** both CRD NOTCH OVERRIDE, 2-HS-85-47, and CRD CONTROL SWITCH, 2-HS-85-48.

Simulator Event Guide:

Event 3 Reactivity: Raise Power with Control Rods

	ATC	[6.6] **CHECK** control rod settles into position 48 and ROD SETTLE light extinguishes. [6.7] **IF** control rod coupling integrity check fails, **THEN REFER TO** 2-AOI-85-2.
	ATC	**6.6.5 Return to Normal After Completion of Control Rod Withdrawal** [1] **WHEN** control rod movement is no longer desired **AND** deselecting control rods is desired, **THEN**: [1.1] **PLACE** CRD POWER, 2-HS-85-46, in OFF. [1.2] **PLACE** CRD POWER, 2-HS-85-46, in ON.
	Driver	At NRC direction, Manually enter CRDH controller failure fic-85-11 0-100(L)

Simulator Event Guide:

Event 4 Component: CRDH Controller Failure

	ATC	Report Alarm 5A-10 CRD ACCUM CHG WTR HDR PRESS HIGH
		A. **VERIFY** pressure high on CRD ACCUM CHG WTR HDR 2-PI-85-13A, B. **CHECK** 2-FCV-85-11A (B) in service. C. **IF** in-service controller has failed, **THEN REFER TO 2**-OI-85. D. **IF** pressure is still greater than 1510 psig after verifying proper controller operation, **THEN THROTTLE** PUMP DISCH THROTTLING, 2-THV-085-0527, to maintain between 1475 and 1500 psig.
	ATC	Report CRD controller has failed in Automatic, takes manual control and restores CRD Parameters
	ATC	Continues to withdraw control rods
	Driver	At NRC direction, insert F6 (trigger 11) to enter Steam Packing Exhauster Failure

Simulator Event Guide:

Event 5 Component: Steam Packing Exhauster 2A trip and failure 2B discharge valve

	BOP	Responds to Alarm 7A-12, Steam Packing Exhauster Vacuum Low.
		7A-12, Steam Packing Exhauster Vacuum Low Automatic Action: Alternate SPE fan starts and discharge damper opens, and the running fans trips.
		A. **CHECKS** the following: 1. Alternate STEAM PACKING EXHR BLOWER 2B, 2-HS-66-51A started. 2. 2B DISCHARGE VLV, 2-HS-66-35A opens.
	BOP	Determines that Alternate Blower started, but discharge damper fails to open.
		Opens 2B DISCHARGE VLV, 2-HS-66-35A to restore SPE Vacuum.
	NRC	NOTE: SPE B Blower indication will have "Red and Green" lights. In order for "Red" light only indication, the crew would have to stop the A SPE. IAW 2-OI-47C
	Driver	When dispatched, wait 5 minutes and report no obvious problems at SPE or Breaker.
	NRC	Ensure that the ATC gets the next event, BOP may need to be distracted.
	Driver	When directed by NRC, insert for RWCU Leak with failure to Auto isolate (imf cu06) (imf cu04 100 300 50)

Simulator Event Guide:

Event 6 Instrument: RWCU leak with failure to auto isolate

	ATC	Report alarm RWCU LEAK DETECTION TEMP HIGH (2-9-3D Window 17) A. **IF** this alarm is received in conjunction with RWCU ISOL LOGIC CHANNEL A TEMP HIGH [2-XA-55-5B, window 32] and RWCU ISOL LOGIC CHANNEL B TEMP HIGH [2-XA-55-5B, window 33], **THEN EXIT** this procedure and **GO TO** 2-ARP-9-5B. Otherwise, **CONTINUE** in this procedure.
	ATC	Report alarms RWCU ISOL LOGIC CHANNEL A TEMP HIGH, RWCU ISOL LOGIC CHANNEL B TEMP HIGH
		A. **VERIFY** alarm by checking: 1. ATUs on Panel 2-9-83 and 2-9-85. 3. Area temperature indications on LEAK DETECTION SYSTEM TEMPERATURE, 2-TI-69-29, on Panel 2-9-21.
		B. **IF** leak is suspected, **THEN MANUALLY ISOLATE** RWCU or if RWCU automatically isolates, **REFER TO** 2-AOI-64-2A. C. **IF** TIS-69-835A(C) indicates greater than 131°F, **THEN ENTER** 2-EOI-3.
	ATC	Reports RWCU Valve 69-1 failed to isolate
T#3	ATC	Closes 69-1 to stop RWCU Leak
CT#3	SRO	Directs Penetration Isolated or concurs with the closure of 69-1
	SRO	Enter EOI-3 and 2-AOI-64-2A
	ATC	**4.1 Immediate Actions** [1] **VERIFY** automatic actions occur. [2] **PERFORM** any automatic actions which failed to occur.
	Driver	Acknowledge Notifications, when dispatched to ATUs report high temperatures in RWCU HX room and temperature lowering.
		4.2 Subsequent Actions [5] **CHECK** the following monitors for a rise in activity: • AREA RADIATION, 3-RR-90-1, Points 9, 13, and 14 (Panel 3-9-2) • AIR PARTICULATE MONITOR CONSOLE, 3-MON-90-50, 3-RM-90-55 and 57 (Panel 3-9-2) • RB, TB, and Refuel Zone Exhaust Rad on CHEMISTRY CAM, MONITOR CONTROLLER, 0-MON-90-361 (Panel 1-9-2)
		[6] **IF** it has been determined that leakage is the cause of the isolation, **THEN NOTIFY** RADCON of RWCU status.
		[7] **NOTIFY** Chemistry that RWCU has been removed from service for the following evaluations: • The need to begin sampling Reactor Water • The need to remove the Durability Monitor from service

Simulator Event Guide:

Event 6 Instrument: RWCU leak with failure to auto isolate

	ATC/BOP	[8] **IF** the isolation cannot be reset, **THEN** [9] **EVALUATE** Technical Requirements Manual Section 3.4.1, Coolant Chemistry, for limiting conditions for operation.
	ATC/BOP	Inserts substitute data per 2-OI-69
		7.0 SYSTEM SHUTDOWN 7.1 ICS Temperature Point Subsitution for Heat Balance [1] IF removing Reactor Water Cleanup System from service when operating at power, THEN PERFORM RWCU ICS Temperature Point Substitution for Heat Balance adjustments: **NOTE** The following values are to be substituted for RWCU Inlet and Outlet temperatures so RWCU parameters provide conservative input to the Integrated Computer System (ICS) thermal power calculation. • 525 degrees F for 69-6A, RWCU LOOP INLET TEMP. • 420 degrees F for 69-6D, RWCU LOOP OUTLET TEMP.
	SRO	Evaluate Technical Specification 3.6.1.3 and determine Condition A required and TRM 3.4.1. Notifies Chemistry that continuous monitoring is no longer available and to commence sampling per TRM Surveillance 3.4.1.1
		3.6 CONTAINMENT SYSTEMS 3.6.1.3 Primary Containment Isolation Valves (PCIVs) LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE. APPLICABILITY: MODES 1, 2, and 3, When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

Simulator Event Guide:

Event 6 Instrument: RWCU leak with failure to auto isolate

<table>
<tr><td rowspan="3"></td><td rowspan="3">SRO</td><td>CONDITION</td><td>REQUIRED ACTION</td><td>COMPLETION TIME</td></tr>
<tr><td>A. --------------NOTE------------
Only applicable to penetration flow paths with two PCIVs.

One or more penetration flow paths with one PCIV inoperable except due to MSIV leakage not within limits.</td><td>A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

<u>AND</u></td><td>4 hours except for main steam line

<u>AND</u>

8 hours for main steam line

(continued)</td></tr>
</table>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 ------------NOTE------------ Isolation devices in high radiation areas may be verified by use of administrative means. --------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside primary containment

Simulator Event Guide:

Event 6 Instrument: RWCU leak with failure to auto isolate

	SRO	TR 3.4 REACTOR COOLANT SYSTEM TR 3.4.1 Coolant Chemistry LCO 3.4.1 Reactor coolant chemistry shall be maintained within the limits of Table 3.4.1-1. APPLICABILITY: According to Table 3.4.1-1
		TECHNICAL SURVEILLANCE REQUIREMENTS

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.1.1	------------------------NOTE------------------------ Not required when there is no fuel in the reactor vessel. --- Monitor reactor coolant conductivity.	Continuously OR 4 hours when the continuous conductivity monitor is inoperable and the reactor is not in MODE 4 or 5 OR 8 hours when the continuous conductivity monitor is inoperable and the reactor is in MODE 4 or 5

Simulator Event Guide:

Event 6 Instrument: RWCU leak with failure to auto isolate

		Enters EOI-3 on High Secondary Containment Temperature. **Secondary Containment Temperature** Monitor and Control Secondary Containment Temperature. Operate available ventilation, per Appendix 8F. Answers **YES** to: Is Any Area Temp Above Max Normal? Isolate all systems that are discharging into the area Verifies RWCU Isolated **Secondary Containment Radiation** Monitor and Control Secondary Containment Radiation Levels. Answers **NO** to: Is Any Area Radiation Level above Max Normal? **Secondary Containment Level** Monitor and Control Secondary Containment Water Levels. Answers **NO** to: Is Any Floor Drain Sump Above 66 inches? AND Answers **NO** to: Is Any Area Water Level Above 2 inches?
	DRIVER	When directed by the NRC, insert F8 (imf pc14 100 360 10) to cause a unisolable leak from the torus.

Simulator Event Guide:

Event 7 Major: Non-Isolable Leak on Torus

	ATC/BOP	Respond to alarm multiple Pump Room Flood Level alarms and **SUPPR CHAMBER WATER LEVEL ABNORMAL**
	ATC/BOP	Reports lowering suppression pool water level. 9-3B W15
		A. **CHECK** level using multiple indications. B. **IF** level is low, **THEN DISPATCH** personnel to check for leaks. C. **IF** level is high, **THEN** D. **REFER TO** 2-OI-74, Sections 8.2, 8.3, and 8.4. E. **REFER TO** Tech Spec Section 3.6.2.2. F. **IF** level is above (-) 1" or below (-) 6.25 inches, **THEN ENTER** 2-EOI-2 Flowchart.
	Driver	When dispatched, wait 4 minutes and report, "Water level is 4 inches and rising in the Southeast Quad. Water is flowing in from the Torus Area. Unable to determine source of the leak."
	SRO	Enters EOI-3 on Flood Alarms
		EOI-3 Secondary Containment Temp Monitor and Control Secondary CNTMT Temp Answers **No** to Is Any Area Temp Above Max Normal **EOI-3 Secondary Containment Radiation** Monitor and Control Secondary CNTMT Radiation Levels Answers **No** to Is Any Area Radiation Level Above Max Normal

Simulator Event Guide:

Event 7 Major: Non-Isolable leak on Torus

	SRO	**Enters EOI-3 on Flood Alarms** **EOI-3 Secondary Containment Level** Monitor and Control Secondary CNTMT Water Level Answers **Yes** to Is Any Floor Drain Sump Above 66 inches Answers **Yes** to Is Any Area Water Level Above 2 inches Restore and Maintain Water Levels using all available sump pumps Answers **No** to Can All Water Levels be Restore and Maintained Below Isolate all systems that are discharging into the area except systems required to: • Be operated by EOIs OR • Suppress a Fire Answers **No** to Will Emergency Depressurization Reduce Discharge Into Secondary Containment.
	SRO	Enters EOI-2 on Low Suppression Pool Level
	SRO	**Enter EOI-2 on Low Suppression Pool Level** Monitor and Control Suppression Pool Level Between (-) 1 inch and (-) 6 inches. (Appendix 18) Answers **NO** to: Can Suppression Pool Level Be Maintained Above (-) 6 inches? Answers **YES** to: Can Suppression Pool Level Be Maintained Below (-) 1 inch?
CT #2		When Suppression Pool Level cannot be maintained above 12.75 feet, HPCI secured to prevent damage.
	SRO	Sets a Value for HPCI to place in Pull to Lock, prior to 12.75 feet.
	ATC/BOP	Places HPCI in Pull to Lock, before Suppression Level lowers to 12.75 feet.

Simulator Event Guide:

Event 8 C: HPCI minimum flow valve will not open

	SRO	Directs Appendix 18
	BOP	**Appendix 18** 6. IF Directed by SRO to add water to suppression pool, THEN **MAKEUP** water to Suppression Pool as follows: a. **VERIFY OPEN** 2-FCV-73-40, HPCI CST SUCTION VALVE. b. **OPEN** 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE c. IF HPCI is NOT available for Suppression Pool makeup, THEN **MAKEUP** water to Suppression Pool using RCIC as follows: 1) **VERIFY OPEN** 2-FCV-71-19, RCIC CST SUCTION VALVE. 2) **OPEN** 2-FCV-71-34, RCIC PUMP MIN FLOW VALVE.
	BOP	Attempts to makeup water to the Suppression Pool using HPCI; 2-FCV-73-30 will not open. Utilizes RCIC to makeup water to the Suppression Pool and dispatches personnel to investigate 2-FCV-73-30.
	Driver	2-FCV-73-30 fails closed when the Torus leak is inserted, crew will dispatch personnel to investigate. Acknowledge investigation and provide no further information.
CT #1	SRO	Determines a trigger value for inserting a Reactor Scram on lowering Suppression Pool Water Level and enters EOI-1, Scrams Reactor before Suppression Pool level reaches 11.5 feet.
	SRO	Determines that Emergency Makeup to the Suppression Pool using Standby Coolant is required and directs BOP to line up Standby Coolant to the Suppression Pool per Appendix 18.
	BOP	**Appendix 18** 5. IF Directed by SRO to Emergency Makeup to the Suppression Pool from Standby Coolant, THEN **CONTINUE** in this procedure at Step 9. 9. IF Directed by SRO to Emergency Makeup to the Suppression Pool using Standby Coolant Supply, THEN **MAKEUP** water to the Suppression Pool as follows: a. **VERIFY CLOSED** the following valves: • 2-FCV-74-61, RHR SYS I DW SPRAY INBD VALVE • 2-FCV-74-60, RHR SYS I DW SPRAY OUTBD VALVE • 2-FCV-74-58, RHR SYS I SUPPR CHBR SPRAY VALVE • 2-FCV-74-52, RHR SYS I LPCI OUTBD INJ VALVE • 2-FCV-74-59, RHR SYS I SUPPR POOL CLG/TEST VALVE • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VALVE

Simulator Event Guide:

Event 7 Major: Non-Isolable leak on Torus

	BOP	**Appendix 18 (continued)** b. **PLACE VERIFY** RHR Pumps 2A and 2C are NOT running. c. **START** RHRSW Pumps D1 and D2. NOTE: 2-BKR-074-0100, RHR SYS I U-1 DISCH XTIE Breaker compartment is maintained in the OPEN position as an Appendix R requirement d. **NOTIFY** Unit 1 Operator to perform the following 1) **VERIFY CLOSED** 1-FCV-23-52, RHR HEAT EXCHANGER D COOL WATER OUTLET VLV (Unit 1, Panel 1-9-3). 2) **OPEN** 1-FCV-23-57, STANDBY COOLANT VALVE FROM RHRSW (Unit 1, Panel 1-9-3). 3) **DISPATCH** personnel to place 2-BKR-074-0100, RHR SYS I U-1 DIXCH XTIE in ON (480V RMOV BD 1B, Compartment 19A).
	Driver	When personnel dispatched to close 2-BKR-074-0100, wait 6 minutes then close breaker and report, delete override for breaker control power. When requested 1-FCV-23-52 is closed. When requested to open 1-FCV-23-57 insert remote function **sw09 open** and report
	BOP	**Appendix 18 (continued)** e. **NOTIFY** Unit 3 Operator to **VERIFY CLOSED** 3-FCV-23-52, RHR HX 3D RHRSW OUTLET VLV (Unit 3, Panel 3-9-3).
	Driver	When requested 3-FCV-23-52 is closed

Simulator Event Guide:

Event 7 Major: Non-Isolable leak on Torus

	BOP	**Appendix 18 (continued)** f. **INJECT** Standby Coolant into the Suppression Pool as follows: 1) **OPEN** 2-FCV-74-100, RHR SYS I U-1 DISCH XTIE. 2) **OPEN** 2-FCV-74-57, RHR SYS I SUPPR CHMBR/POOL ISOL VLV. 3) **THROTTLE OPEN** 2-FCV-74-59, RHR SYS I SUPPR POOL CLG/TEST VLV to control injection.
	SRO	Enters EOI-1 at pre-determined trigger value and directs Core Flow Runback and Reactor Scram based on EOI-2 step SP/L-7.
	SRO	**Enters EOI-1 from EOI-2 step SP/L-7** Verify Reactor Scram **EOI-1 RC/L** Monitor and Control RPV Water Level Verify as Required: • PCIS Isolations (Groups 1,2 and 3) • ECCS • RCIC Restore and maintain RPV water level +2 to +51 inches using Condensate and Feedwater in accordance with App 5A **EOI-1 RC/Q** Monitor and Control Reactor Power • Crew will exit RC/Q and enter 2-AOI-100-1 based on RC/Q-2.
	SRO	SRO expected to lower pressure band to commence cooldown prior to anticipating Emergency Depressurization.
CT #1	SRO	May Anticipate Emergency Depressurization and Rapidly Depressurize using Bypass valves based on EOI-1 step RC/P-3
	BOP	Verifies and reports PCIS isolations and, if directed, opens all Bypass Valves to Rapidly Depressurize RPV irrespective of cooldown rate. Maintains Reactor Water Level +2 to +51 inches using Condensate and Feedwater per App 5A
	ATC	Initiates Core Flow Runback and Manual Reactor Scram and performs Immediate Actions of 2-AOI-100-1
	SRO	**EOI-1 RC/P** Monitor and Control RPV pressure When Emergency Depressurization is required Exits RC/P and enters C-2, Emergency RPV Depressurization, based on Override step RC/P-4.

Simulator Event Guide:

Event 7 Major: Non-Isolable leak on Torus

	ATC	**2-AOI-100-1 Immediate Actions** [1] DEPRESS REACTOR SCRAM A and B, 2-HS-99-5A/S3A and 2-HS-99-5A/S3B, on Panel 2-9-5. [2] IF scram is due to a loss of RPS, THEN PAUSE in START & HOT STBY mode for approximately 5 seconds before going to REFUEL. (Otherwise N/A) [3] REFUEL MODE ONE ROD PERMISSIVE light check: [3.1] PLACE REACTOR MODE SWITCH, 2-HS-99-5A-S1, in REFUEL. [3.2] CHECK REFUEL MODE ONE ROD PERMISSIVE light, 2-XI-85-46, illuminates. [3.3] IF REFUEL MODE ONE ROD PERMISSIVE light, 2-XI-85-46, is not illuminated, THEN CHECK all control rod positions at Full-In Overtravel, or Full-In. (Otherwise N/A) [4] PLACE REACTOR MODE SWITCH, 2-HS-99-5A-S1, in SHUTDOWN position. [5] IF all control rods CAN NOT be verified fully inserted, THEN INITIATE ARI by Arming and Depressing, (Otherwise N/A) • ARI Manual Initiate, 2-HS-68-119A OR • ARI Manual Initiate, 2-HS-68-119B [6] REPORT the following status to the US: • Reactor Scram • Mode Switch is in Shutdown • "All rods in" or "rods out " • Reactor Level and trend (recovering or lowering). • Reactor pressure and trend • MSIV position (Open or Closed) • Power level [7] US REPEAT back status to UO, eye contact is not necessary.
	BOP	Performs necessary actions of 2-EOI-App-5A to maintain RPV water level in band
		2-EOI-App-5A 13. **ADJUST** RFPT speed as necessary to control injection using the methods of step 12. 14. WHEN RPV level is approximately equal to desired level AND automatic level control is desired, THEN **PLACE** 2-LIC-46-5, REACTOR WATER LEVEL CONTROL, in AUTO with individual 2-SIC-46-8(9)(10), RFPT 2A(2B)(2C) SPEED CONTROL in AUTO.

Simulator Event Guide:

Event 7 Major: Non-Isolable leak on Torus

	SRO	When RPV pressure has decreased to approximately Condensate Injection Pressure directs ATC to maintain RPV Water Level +2 to +51 inches per App 6A
	ATC	Maintains RPV Water Level in band with 2-EOI-App-6A
		2-EOI-App-6A 1. **VERIFY CLOSED** the following feedwater heater return valves: 2-FCV-3-71, HP HTR 2A1 LONG CYCLE TO CNDR 2-FCV-3-72, HP HTR 2B1 LONG CYCLE TO CNDR 2-FCV-3-73, HP HTR 2C1 LONG CYCLE TO CNDR. 2. **VERIFY CLOSED** the following RFP discharge valves: • 2-FCV-3-19, RFP 2A DISCHARGE VALVE • 2-FCV-3-12, RFP 2B DISCHARGE VALVE • 2-FCV-3-5, RFP 2C DISCHARGE VALVE. 3. **VERIFY OPEN** the following drain cooler inlet valves: • 2-FCV-2-72, DRAIN COOLER 2A5 CNDS INLET ISOL VLV • 2-FCV-2-84, DRAIN COOLER 2B5 CNDS INLET ISOL VLV • 2-FCV-2-96, DRAIN COOLER 2C5 CNDS INLET ISOL VLV. 4. **VERIFY OPEN** the following heater outlet valves: • 2-FCV-2-124, LP HEATER 2A3 CNDS OUTL ISOL VLV • 2-FCV-2-125, LP HEATER 2B3 CNDS OUTL ISOL VLV • 2-FCV-2-126, LP HEATER 2C3 CNDS OUTL ISOL VLV. 5. **VERIFY OPEN** the following heater isolation valves: • 2-FCV-3-38, HP HTR 2A2 FW INLET ISOL VLV • 2-FCV-3-31, HP HTR 2B2 FW INLET ISOL VLV • 2-FCV-3-24, HP HTR 2C2 FW INLET ISOL VLV • 2-FCV-3-75, HP HTR 2A1 FW OUTLET ISOL VLV • 2-FCV-3-76, HP HTR 2B1 FW OUTLET ISOL VLV • 2-FCV-3-77, HP HTR 2C1 FW OUTLET ISOL VLV. 6. **VERIFY OPEN** the following RFP suction valves: • 2-FCV-2-83, RFP 2A SUCTION VALVE • 2-FCV-2-95, RFP 2B SUCTION VALVE • 2-FCV-2-108, RFP 2C SUCTION VALVE. 7. **VERIFY** at least one condensate pump running. 8. **VERIFY** at least one condensate booster pump running. 9. **ADJUST** 2-LIC-3-53, RFW START-UP LEVEL CONTROL, to control injection (Panel 2-9-5). 10. **VERIFY** RFW flow to RPV.

Simulator Event Guide:

Event 9 Component: All SRVs except 3 fail to open for Emergency Depressurization

CT #1	SRO	When Suppression Pool level cannot be maintained above 11.5 feet the US determines that Emergency Depressurization is required; RO initiates Emergency Depressurization as directed by US.
	SRO	When Emergency Depressurization is required exits RC/P and enters C-2, Emergency RPV Depressurization
		Determines Emergency Depressurization is required and enters C-2 Answers **Yes** to will the reactor remain subcritical under all conditions. Answers **No** to is DW pressure above 2.4 psig Answers **Yes** to is Suppression Pool Level above 5.5 ft Directs All ADS Valves opened Answers **No** to can Six ADS Valves be opened Directs BOP to open additional MSRVs as necessary to establish 6 MSRVs open Answers **No** to are at least 4 MSRVs open Answers **Yes** to is RPV pressure 80 psi or more above Suppression Chamber Pressure **Directs** BOP to Rapidly Depressurize the RPV to less than 80 psi above Suppression Chamber pressure with one or more of the systems listed on C2-12
CT #1	BOP	Opens ADS Valves Opens additional MSRVs as necessary in an attempt to establish 6 MSRVs open Identifies only 3 valves open and informs SRO

Simulator Event Guide:

Event 7 Major: Non-Isolable leak on Torus

CT #1	SRO	**Directs** BOP to Rapidly Depressurize the RPV to less than 80 psi above Suppression Chamber pressure utilizing App 11H
CT #1	BOP	**2-EOI-App-11H** 2. **VERIFY** Main Condenser Off-Gas is aligned to the stack as follows: b. **VERIFY OPEN** 2-FCV-66-28, OFFGAS SYSTEM ISOLATION VALVE (Panel 9-53). 3. **VERIFY** SJAE 2A or 2B in service and aligned to Main Condenser (Panel 9-7). 5. IF ANY Main Steam Line is NOT isolated, THEN **CONTINUE** in this procedure at Step 12. **CAUTION** Offsite release rate limits may be exceeded. 12. **OPEN** Turbine Bypass valves as necessary to rapidly depressurize RPV.
	SRO	Classify the Event
		Event Classification is 2.1-S

SHIFT TURNOVER SHEET

Equipment Out of Service/LCO's:

SLC pump 2B and EECW pump A3 out of service.

Operations/Maintenance for the Shift:

HPCI surveillance testing has just been completed and Torus cooling is to be secured. Reactor Power is 76%. Secure RHR Loop II from Torus cooling. Commence a power increase to 100%.

Units 1 and 3 are at 100% power

Unusual Conditions/Problem Areas:

None

ATTACHMENT 2
(Page 1 of 2)

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	22-31	12	00		
N/A	30-39	12	00		
N/A	38-31	12	00		
N/A	30-23	12	00		
N/A	30-31	48	00		
N/A	30-15	16	00		
N/A	46-31	16	00		
N/A	30-47	16	00		
N/A	14-31	16	00		
N/A	22-23	16	00		
N/A	38-23	16	00		
N/A	38-39	16	00		
N/A	22-39	16	00		
N/A	14-23	48	00		
N/A	14-39	48	00		
N/A	46-39	48	00		
N/A	46-23	48	00		

REMARKS[4]: Emergency Shove Sheet – Loadline reduction or Unit Shutdown Insert Rods Continuously to 00. Insertion may stop after completion of any group.

NOTES:

(1) RWM Group may be marked “N/A” if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position “48”, this signoff verifies coupling integrity was checked in accordance with 2-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by: ____________________ / __________ Issued by ____________________ / __________
Unit Supervisor Date Reactor Engineer Date

ATTACHMENT 2
(Page 2 of 2)

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	22-47	48	00		
N/A	38-47	48	00		
N/A	38-15	48	00		
N/A	22-15	48	00		
N/A	14-47	48	00		
N/A	46-47	48	00		
N/A	46-15	48	00		
N/A	14-15	48	00		
N/A	06-31	48	00		
N/A	30-55	48	00		
N/A	54-31	48	00		
N/A	30-07	48	00		
N/A	06-39	48	00		
N/A	54-39	48	00		
N/A	54-23	48	00		
N/A	06-23	48	00		

REMARKS[4]: Emergency Shove Sheet – Loadline reduction or Unit Shutdown Insert Rods Continuously to 00. Insertion may stop after completion of any group.

NOTES:

(1) RWM Group may be marked "N/A" if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position "48", this signoff verifies coupling integrity was checked in accordance with 2-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by: ____________ / ______ Issued by ____________ / ______

Unit Supervisor Date Reactor Engineer Date

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

Reactivity Maneuver Plan U2 NRC Exam 1

Unit 2 Rod Pattern Adjustment

Attachment 7
(Page 1 of 2)

Reactivity Control Plan Form

BFN Unit: 2 Valid Date(s): 8/7/11 – 8/19/11 Reactivity Control Plan #: **U2 NRC Exam 1**

Are Multiple Activations Allowed: No (If yes, US may make additional copies)

Prepared by: ____________ / ______
Reactor Engineer Date

Reviewed by: ____________ / ______
Qualified Reactor Engineer Date

Approved by: ____________ / ______
RE Supervisor Date

Concurrence: ____________ / ______
WCC/Risk/US SRO Date

Approved by: ____________ / ______
Ops Manager or Supt. Date

Authorized by: ____________ / ______
Shift Manager Date

RCP Activated: ____________ / ______
Unit Supervisor Date

RCP Terminated: ____________ / ______
Unit Supervisor Date

Title of Evolution: Unit 2 Rod Pattern Adjustment

Purpose/Overview of Evolution: Adjust Rod Pattern for 100% power operation

Maneuver Steps

1. ***Withdraw Control rods IAW Attachment 2 provided by Reactor Engineer.***
2. ***Increase flow to 100% power.(No Ramp Rate limits)***

Attachment 7
(Page 2 of 2)

Reactivity Control Plan Form

Operating Experience and General Issues: U2 NRC Exam 1

Previously known control rod issues:

4	172292	05/28/2009	Control Rod 46-27 double notched during the performance of the Unit 2 sequence exchange, 00 to 04.
4	150002	08/10/2008	During power ascension activities, control rod 46-27 double notched from position 00 to 04.
4	149981	08/09/2008	Control Rod 38-35 double notched during control rod withdrawal from 00 to 04.
4	148263	07/12/2008	While pulling control rods during U2 startup, CR 38-03 double-notched twice. 10 to 14 and 14 to 18

Cautions/Error Likely Situations/Special Monitoring Requirements/Contingencies:

- Rod Out Notch Override is authorized, for Rod Out Notch Override follow the guidance in 2-OI-85 section 6.6.4.
- This plan is NOT valid if the unit is operating with a suspected or known fuel leaker and is not to be used. Contact Reactor Engineering if there are indications of a fuel leak.

BFN	Reactivity Control Plan	

Attachment 8

Reactivity Maneuver Instructions

STEP 1 of 2 **Reactivity Maneuver Plan # U2 NRC Exam 1**

Description of Step: ***Withdraw Control rods IAW Attachment 2 provided by Reactor Engineer.***

Conditions : To be recorded at the Completion of Step Recorded: ___________/______
(by RO) (Date)

QRE presence required in the Control Room? Yes _____ No _X_ (check)

	Predicted (may be ranges)	Actual		Predicted (may be ranges)	Actual
MW Electric	850-1025		MFLCPR	.80 - .85	
MW Thermal	2600-3100		MAPRAT	.45 - .55	
Core Flow	78-82mlbm/hr		MFDLRX	.65 - .70	
Loadline	102-104				
Core Power	85-90%		Other		

Critical Parameters: To be recorded DURING Step. **IF** parameters are outside of the predictions, **THEN** discuss with the RE **AND** record conclusions in the Comments / Notes section.

Description including frequency, method of monitoring, **AND** contingency actions	High	Low

Comments / Notes: Rod Out Notch Override is authorized, for Rod Out Notch Override follow the guidance in 2-OI-85 section 6.6.4.

Step Complete **AND** Reviewed by:__________________/________
Unit Supervisor / Date

Attachment 8

Reactivity Maneuver Instructions

STEP 2 of 2 **Reactivity Maneuver Plan # U2 NRC Exam 1**

Description of Step: Raise reactor power to 100% using core flow. NO Ramp Rate Limits apply.

Conditions : To be recorded at the Completion of Step Recorded: ____________/______ (by RO) (Date)

QRE presence required in the Control Room? Yes _____ No _X_ (check)

	Predicted (may be ranges)	Actual		Predicted (may be ranges)	Actual
MW Electric	1150		MFLCPR	.85 - .95	
MW Thermal	3400-3450		MAPRAT	.60 - .70	
Core Flow	85 – 95 mlbm/hr		MFDLRX	.70 - .75	
Loadline	102-104				
Core Power	100%		Other		

Critical Parameters: To be recorded DURING Step. **IF** parameters are outside of the predictions, **THEN** discuss with the RE **AND** record conclusions in the Comments / Notes section.

Description including frequency, method of monitoring, **AND** contingency actions	High	Low
N/A		

Comments / Notes:

1. Raise Reactor Power to 100% RTP
2. Document core flow changes on Attachment 10

Step Complete **AND** Reviewed by:_________________/________
Unit Supervisor / Date

Attachment 10
(Page 1 of 1)

Recirc Flow Maneuver Instructions

Reactivity Control Plan # U2 NRC Exam 1

RCP Step #	Flow Step #	Time	**Target Power (%RTP or MWe)**	Delta ±(MWe)	**Target Flow (MLb/Hr)**	Completed (RO)
2	**1**		100%			

Comments / Notes:

Reviewed by: ____________________/________
Unit Supervisor / Date

BFN	Reactivity Control Plan	

ATTACHMENT 4

ROD PATTERN STEP THROUGH MAPS
Reactivity Maneuver Plan # *U2 NRC Exam 1*

Prior to RCP 1

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	
59																59
55																55
51																51
47								00								47
43																43
39						00		00		00						39
35																35
31				00		00		00		00		00				31
27																27
23						00		00		08						23
19																19
15								00								15
11																11
07																07
03																03
	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	

After RCP 1

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	
59																59
55																55
51																51
47								16								47
43																43
39						16		12		16						39
35																35
31				16		12				12		16				31
27																27
23						16		12		16						23
19																19
15								16								15
11																11
07																07
03																03
	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	

ATTACHMENT 2
(Page 1 of 1)

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	22-31	00	12		
N/A	30-39	00	12		
N/A	38-31	00	12		
N/A	30-23	00	12		
N/A	30-31	00	48		
N/A	30-15	00	16		
N/A	46-31	00	16		
N/A	30-47	00	16		
N/A	14-31	00	16		
N/A	22-23	00	16		
N/A	38-23	00	16		
N/A	38-39	00	16		
N/A	22-39	00	16		

REMARKS[4]: Control Rod Pattern Adjustment

NOTES:

(1) RWM Group may be marked "N/A" if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position "48", this signoff verifies coupling integrity was checked in accordance with 2-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by:____________ / ______ Issued by ____________ / ______
Unit Supervisor Date Reactor Engineer Date

Facility: **Browns Ferry NPP** Scenario No.: **NRC - 2** Op-Test No.: **1108**

Examiners:________________ Operators: **SRO:**________________

________________ **ATC:**________________

________________ **BOP:**________________

Initial Conditions: 86% power, CCW pump 3A is ready to return to service.

Turnover: Return to service Condenser Circulating Water pump 3A per 3-OI-27 section 8.2. Raise power to 100%

Event No.	Malf. No.	Event Type*	Event Description
1		C-BOP C-SRO	Returning to service Condenser Circulating Water Pump 3A, IAW 3-OI-27 section 8.2, Pump Disch Valve Failure to Open
2		R-ATC R-SRO	Commence power increase with flow
3	RC02	C-BOP TS-SRO	Inadvertent RCIC start w/ trip pushbutton failure
4	RD01a	C-ATC C-SRO	CRD Pump 3A trip
5	RD07 46-19	C-ATC TS-SRO	Control Rod 46-19 drifts in to position 40
6	RC10	C-BOP TS-SRO	Steam leak in the RCIC room RCIC Steam line isolation valves 3-FCV-71-2 and 3 will not auto isolate.
7	MS06A MS06B TH35A	M-ALL	MSL A Break in Reactor BLDG with MSL A valves failing to close
8	RP07	I	RPS Fails to de-energize, ARI inserts all Rods
9	HP015	C	HPCI flow controller failure in Auto to 10%

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Events

1. BOP returns the Condenser Circulating Water Pump 3A to service IAW 3-OI-27 Condenser Circulating Water System, section 8.2. BOP Operator determines that 3A CCW Pump discharge valve fails to open. Orders a reset of breaker or secures 3A CCW Pump.

2. ATC commences power increase 100% using recirculation flow.

3. Inadvertent start of RCIC. BOP will attempt to trip RCIC, RCIC trip pushbutton fails BOP will close FCV-71-9 Valve and SRO will determine RCIC System inoperable, Technical Specification 3.5.3 Condition A

4. CRDH pump 3A trips ATC will perform 3-AOI-85-3 actions to start the Standby CRD Pump and restore CRD parameters.

5. When CRD Pump 3B is started Control rod 46-19 will drift in to position 40. ATC will respond IAW 3-AOI-85-5 Control Rod Drift In. ATC will insert Control Rod 46-19 until it becomes stuck at position 20. SRO will determine Control Rod 46-19 is Inoperable Technical Specification 3.1.3 Condition A.

6. A RCIC Steam Leak will result in high Room temperature with a failure of RCIC to Isolate. The BOP will isolate RCIC. The SRO will determine RCIC Isolation Valves inoperable Technical Specification 3.6.1.3 Condition A.

7. MSL break in Reactor Building with MSL A valves failing to close, with small fuel failure on scram. SRO will enter EOI-3 and transition to EOI-1 and Scram the Reactor Crew will monitor secondary containment radiation levels. Eventually the SRO will determine that ED on Radiation Levels is required.

8. On the Scram RPS will fail to de-energize, ATC will initiate ARI to insert control rods

9. RFPTs will trip on the scram, HPCI is available for level control but the HPCI flow controller will fail in Auto at 10%. Crew will take manual control to restore and maintain reactor level.

Terminate the scenario when the following conditions are satisfied or upon request of Lead Examiner:

Control Rods are inserted

Emergency Depressurization complete

Reactor Level is restored and maintained

Critical Tasks - Four

CT#1-With reactor at power and with a primary system discharging into the secondary containment, manually scram the reactor before MSL Tunnel temperature reaches 189°F OR any area radiation reaches 1000 mr/hr.

1. Safety Significance:
   Scram reduces the decay heat energy that the RPV may be discharging into the secondary containment

2. Cues:
   Procedural compliance
   Secondary containment area temperature, level, and radiation indication
   Field reports

3. Measured by:
   Observation - With a primary system discharging into secondary containment, a reactor scram is initiated before a maximum safe condition is reached.
   OR
   Observation - With a primary system discharging into secondary containment, US transitions to EOI-1 and RO initiates scram upon report that a maximum safe condition has been reached.

4. Feedback:
   Control rod positions
   Reactor power decrease

CT#2-With a primary system discharging into the secondary containment, when two or more areas are greater than their maximum safe operating values for the same parameter, RO initiates Emergency Depressurization as directed by US. Within 5 minutes of the parameters exceeding max safe.

1. Safety Significance:
   Places the primary system in the lowest possible energy state, rejects heat to the suppression pool in preference to outside the containment, and reduces driving head and flow of system discharging into the secondary containment.

2. Cues:
   Procedural compliance
   Secondary containment area temperatures, level, and radiation indication
   Field reports

3. Measured by:
   Observation - US transitions to C-2 and RO opens at least 6 SRV's when two or more areas are greater than their maximum safe operating values for the same parameter.

4. Feedback:
   RPV pressure trend
   SRV status indications

CT#3-With a primary system discharging into the secondary containment, take action to manually isolate the break.

1. Safety Significance:
   Isolating high energy sources can preclude failure of secondary containment and subsequent radiation release to the public.

2. Cues:
   Procedural compliance
   Area temperature indication

3. Measured by:
   With the reactor at pressure and a primary system discharging into the secondary containment, operator takes action to manually isolate the break.

4. Feedback:
   Valve position indication
   In field reports

CT#4-With a reactor scram required and the reactor not shutdown, take action to reduce power by initiating ARI to cause control rod insertion.

1. Safety Significance:
   Shutting down reactor can preclude failure of containment or equipment necessary for the safe shutdown of the plant.
   Correct reactivity control

2. Cues:
   Reactor power indication
   Procedural compliance

3. Measured by:
   Observation - ARI pushbuttons armed and depressed to cause control rod insertion.

4. Feedback:
   Reactor power trend
   Rod status indication

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER: 2

9	Total Malfunctions Inserted: List (4-8)
4	Malfunctions that occur after EOI entry: List (1-4)
4	Abnormal Events: List (1-3)
2	Major Transients: List (1-2)
2	EOI's used: List (1-3)
1	EOI Contingencies used: List (0-3)
75	Validation Time (minutes)
4	Crew Critical Tasks: (2-5)
YES	Technical Specifications Exercised (Yes/No)

Scenario Tasks

TASK NUMBER	K/A	RO	SRO
Condenser Circ Water Pump Start			
RO U-027-NO-5	400000A4.01	3.1	3.0
Raise Power with Recirc Flow			
RO U-068-NO-17 SRO S-000-NO-138	2.1.23	4.3	4.4
RCIC Inadvertent Start			
RO U-071-NO-5	217000A2.01	3.8	3.7
RCIC Steam Leak			
RO U-071-AL-20 SRO S-000-EM-12	217000A2.15	3.8	3.8
CRD Pump Trip			
RO U-085-AL-07 SRO S-085-AB-03	201001A2.01	3.2	3.3
Control Rod Drift			
RO U-085-AL-12 SRO S-085-AB-5	201003A2.03	3.4	3.7
Secondary Containment High Radiation			
RO U-090-AL-4 SRO S-000-EM-15 SRO S-000-EM-10	295033EA2.01	3.8	3.9

Simulator Instructor - IC-199

#CCW 3A Disch Valve FTO
ior zaoei2710a (e1 0) 330
ior zlohs2713a[1] on

#RCIC inadvertent start
imf rc02 (e5 0)
ior zdihs719a[1] null

#RCIC steam leak
imf rc10
ior zdihs712a[2] auto
imf rc09 (e6 0) 50 120 10
ior zdihs719a[1] null

#CR 46-19 drift in
imf rd01a (e10 0)
imf rd07r4619 (e12 0)
imf rd06r4619 (e13 0)

#MSL A break inside containment
imf th35a (e15 0) 3 600 0
imf ms06a
imf ms06b
imf hp03 (e15 0) 10
ior xa557c[8] alarm_off
imf rp07

#Fuel failure
imf th23 (e20 120) 4 600 1
ior zdihs03125[1] (e20 10) trip
ior zdihs03151[1] (e20 10) trip
ior zdihs03176[1] (e20 10) trip

Facility: **Browns Ferry NPP** Scenario No.: **NRC – 2** Op-Test No.: **1108**

Examiners:____________________ Operators: **SRO:**____________________

____________________ **ATC:**____________________

____________________ **BOP:**____________________

Initial Conditions: 86% power, CCW pump 3A is ready to return to service.

Turnover: Return to service Condenser Circulating Water pump 3A per 3-OI-27 section 8.2. Raise power to 100%

Event No.	Malf. No.	Event Type*	Event Description
1		C-BOP C-SRO	Returning to service Condenser Circulating Water Pump 3A, IAW 3-OI-27 section 8.2, Pump Disch Valve Failure to Open
2		R-ATC R-SRO	Commence power increase with flow
3	RC02	C-BOP TS-SRO	Inadvertent RCIC start w/ trip pushbutton failure
4	RD01a	C-ATC C-SRO	CRD Pump 3A trip
5	RD07 46-19	C-ATC TS-SRO	Control Rod 46-19 drifts in to position 40
6	RC10	C-BOP TS-SRO	Steam leak in the RCIC room RCIC Steam line isolation valves 3-FCV-71-2 and 3 will not auto isolate.
7	MS06A MS06B TH35A	M-ALL	MSL A Break in Reactor BLDG with MSL A valves failing to close
8	RP07	I	RPS Fails to de-energize, ARI inserts all Rods
9	HP015	C	HPCI flow controller failure in Auto to 10%

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Critical Tasks - Four

CT#1-With reactor at power and with a primary system discharging into the secondary containment, manually scram the reactor before MSL Tunnel temperature reaches 189°F OR any area radiation reaches 1000 mr/hr.

1. Safety Significance:
   Scram reduces the decay heat energy that the RPV may be discharging into the secondary containment

2. Cues:
   Procedural compliance
   Secondary containment area temperature, level, and radiation indication
   Field reports

3. Measured by:
   Observation - With a primary system discharging into secondary containment, a reactor scram is initiated before a maximum safe condition is reached.
   OR
   Observation - With a primary system discharging into secondary containment, US transitions to EOI-1 and RO initiates scram upon report that a maximum safe condition has been reached.

4. Feedback:
   Control rod positions
   Reactor power decrease

CT#2-With a primary system discharging into the secondary containment, when two or more areas are greater than their maximum safe operating values for the same parameter, RO initiates Emergency Depressurization as directed by US. Within five minutes of the parameters exceeding max safe.

1. Safety Significance:
   Places the primary system in the lowest possible energy state, rejects heat to the suppression pool in preference to outside the containment, and reduces driving head and flow of system discharging into the secondary containment.

2. Cues:
   Procedural compliance
   Secondary containment area temperatures, level, and radiation indication
   Field reports

3. Measured by:
   Observation - US transitions to C-2 and RO opens at least 6 SRV's when two or more areas are greater than their maximum safe operating values for the same parameter.

4. Feedback:
   RPV pressure trend
   SRV status indications

CT#3-With a primary system discharging into the secondary containment, take action to manually isolate the break.

1. Safety Significance:
   Isolating high energy sources can preclude failure of secondary containment and subsequent radiation release to the public.

2. Cues:
   Procedural compliance
   Area temperature indication

3. Measured by:
   With the reactor at pressure and a primary system discharging into the secondary containment, operator takes action to manually isolate the break.

4. Feedback:
   Valve position indication
   In field reports

CT#4-With a reactor scram required and the reactor not shutdown, take action to reduce power by initiating ARI to cause control rod insertion.

1. Safety Significance:
   Shutting down reactor can preclude failure of containment or equipment necessary for the safe shutdown of the plant.
   Correct reactivity control

2. Cues:
   Reactor power indication
   Procedural compliance

3. Measured by:
   Observation - ARI pushbuttons armed and depressed to cause control rod insertion.

4. Feedback:
   Reactor power trend
   Rod status indication

Events

1. BOP returns the Condenser Circulating Water Pump 3A to service IAW 3-OI-27 Condenser Circulating Water System, section 8.2. BOP Operator determines that 3A CCW Pump discharge valve fails to open. Orders a reset of breaker or secures 3A CCW Pump.

2. ATC commences power increase 100% using recirculation flow.

3. Inadvertent start of RCIC. BOP will attempt to trip RCIC, RCIC trip pushbutton fails BOP will close FCV-71-9 Valve and SRO will determine RCIC System inoperable, Technical Specification 3.5.3 Condition A

4. CRDH pump 3A trips ATC will perform 3-AOI-85-3 actions to start the Standby CRD Pump and restore CRD parameters.

5. When CRD Pump 3B is started Control rod 46-19 will drift in to position 40. ATC will respond IAW 3-AOI-85-5 Control Rod Drift In. ATC will insert Control Rod 46-19 until it becomes stuck at position 20. SRO will determine Control Rod 46-19 is Inoperable Technical Specification 3.1.3 Condition A.

6. A RCIC Steam Leak will result in high Room temperature with a failure of RCIC to Isolate. The BOP will isolate RCIC. The SRO will determine RCIC Isolation Valves inoperable Technical Specification 3.6.1.3 Condition A.

7. MSL break in Reactor Building with MSL A valves failing to close, with small fuel failure on scram. SRO will enter EOI-3 and transition to EOI-1 and Scram the Reactor Crew will monitor secondary containment radiation levels. Eventually the SRO will determine that ED on Radiation Levels is required.

8. On the Scram RPS will fail to de-energize, ATC will initiate ARI to insert control rods

9. RFPTs will trip on the scram, HPCI is available for level control but the HPCI flow controller will fail in Auto at 10%. Crew will take manual control to restore and maintain reactor level.

Terminate the scenario when the following conditions are satisfied or upon request of Lead Examiner:

Control Rods are inserted

Emergency Depressurization complete

Reactor Level is restored and maintained

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER: 2

9	Total Malfunctions Inserted:	List (4-8)
4	Malfunctions that occur after EOI entry:	List (1-4)
4	Abnormal Events:	List (1-3)
1	Major Transients:	List (1-2)
2	EOI's used:	List (1-3)
1	EOI Contingencies used:	List (0-3)
75	Validation Time (minutes)	
4	Crew Critical Tasks:	(2-5)
YES	Technical Specifications Exercised (Yes/No)	

Scenario Tasks

TASK NUMBER	K/A	RO	SRO
Condenser Circ Water Pump Start			
RO U-027-NO-5	400000A4.01	3.1	3.0
Raise Power with Recirc Flow			
RO U-068-NO-17 SRO S-000-NO-138	2.1.23	4.3	4.4
RCIC Inadvertent Start			
RO U-071-NO-5	217000A2.01	3.8	3.7
RCIC Steam Leak			
RO U-071-AL-20 SRO S-000-EM-12	217000A2.15	3.8	3.8
CRD Pump Trip			
RO U-085-AL-07 SRO S-085-AB-03	201001A2.01	3.2	3.3
Control Rod Drift			
RO U-085-AL-12 SRO S-085-AB-5	201003A2.03	3.4	3.7
Secondary Containment High Radiation			
RO U-090-AL-4 SRO S-000-EM-15 SRO S-000-EM-10	295033EA2.01	3.8	3.9

Procedures Used/Referenced:

Procedure Number	Procedure Title	Procedure Revision
3-OI-27	Condenser Circulating Water System	Rev 58
3-GOI-100-12	Power Maneuvering	Rev 35
3-OI-68	Reactor Recirculation System	Rev 80
3-ARP-9-3B, W27	RCIC Gland Seal Vacuum Tank Pressure High	Rev 20
TS 3.5.3	RCIC System	Amd 244
3-AOI-85-3	CRD System Failure	Rev 10
3-AOI-85-5	Rod Drift In	Rev 10
TS 3.1.3	Control Rod Operability	Amd 212
3-ARP-9-3A, W22	Reactor Building Area Radiation High	Rev 43
3-ARP-9-3D, W10	RCIC Steam Line Leak Detection Temperature High	Rev 28
3-EOI-3	Secondary Containment Control	Rev 10
TS 3.6.1.3	Primary Containment Isolation Valves	Amd 212
3-ARP-9-3D, W24	Main Steam Line Leak Detection Temperature High	Rev 28
3-AOI-100-1	Reactor Scram	Rev 53
3-EOI-1	RPV Control	Rev 8
3-EOI-Appendix-8F	Restoring Refuel Zone and Reactor Zone Ventilation Fans Following Group 6 Isolation	Rev 2
3-EOI-Appendix-8E	Bypassing Group 6 Low RPV level and High Drywell Pressure Isolation Interlocks	Rev 1
3-EOI-Appendix-11A	Alternate Pressure Control Systems MSRVs	Rev 2
3-EOI-Appendix-5D	Injection System Lineup HPCI	Rev 5
3-EOI-3-C-2	Emergency RPV Depressurization	Rev 8
3-EOI-Appendix-6A	Injection Subsystems Lineup Condensate	Rev 2
3-EOI-2	Primary Containment Control	Rev 8
3-EOI-Appendix-17A	RHR System Operation Suppression Pool Cooling	Rev 5
EPIP-1	Emergency Classification	Rev 46

Simulator Instructor - IC-199

#CCW 3A Disch Valve FTO
ior zaoei2710a (e1 0) 330
ior zlohs2713a[1] on

#RCIC inadvertent start
imf rc02 (e5 0)
ior zdihs719a[1] null

#RCIC steam leak
imf rc10
ior zdihs712a[2] auto
imf rc09 (e6 0) 50 120 10
ior zdihs719a[1] null

#CR 46-19 drift in
imf rd01a (e10 0)
imf rd07r4619 (e12 0)
imf rd06r4619 (e13 0)

#MSL A break inside containment
imf th35a (e15 0) 3 600 0
imf ms06a
imf ms06b
imf hp03 (e15 0) 10
ior xa557c[8] alarm_off
imf rp07

#Fuel failure
imf th23 (e20 120) 4 600 1
ior zdihs03125[1] (e20 10) trip
ior zdihs03151[1] (e20 10) trip
ior zdihs03176[1] (e20 10) trip

Scenario 2

		DESCRIPTION/ACTION
Simulator Setup	manual	Reset to IC 199
Simulator Setup	**Load Batch**	bat nrc1108-2
Simulator Setup	manual	
Simulator Setup		Verify file loaded
Simulator Setup		

RCP required (86% - 100% with control rods and flow) and RCP for Urgent Load Reduction
Provide marked up copy of 3-GOI-100-12

Simulator Event Guide:

Event 1 Component: Returning to service CCW Pump 3A, Disch Valve FTO

<table>
<tr><td></td><td>SRO</td><td>Directs CCW Pump 3A returned to service IAW 3-OI-27, section 8.2</td></tr>
<tr><td></td><td>Driver</td><td>Insert trigger one when operator starts 3A CCW Pump to ensure pump discharge valve does not fully open and amps indicate correctly</td></tr>
<tr><td></td><td>BOP</td><td>8.2 Returning a CCW Pump to Service

[4] VERIFY CLOSED the CCW PUMP 3A(3B)(3C) DISCH ISOL VALVE, 3-FCV-27-13(21)(29), on Panel 3-9-20.

CAUTIONS
1) Capacitor bank fuses are subject to clearing when the unit boards are being supplied from the 161kV source and large pumps are started. Unit Supervisors should evaluate placing the Capacitor Banks in Manual prior to starting RHR, CS or CCW pumps.
2) When returning a pump to service with at least one pump already in operation, the pump being placed in service may experience perturbations in flow and motor amps. It may be necessary to throttle Condenser Water Box Discharge Valves as stated in Section 6.1 to stabilize pump.

[6] START CCW PUMP 3A(3B)(3C) using 3-HS-27-10A(18A)(26A) on Panel 3-9-20 and VERIFY the respective CCW PUMP 3A(3B)(3C) DISCH ISOL VALVE, 3-FCV-27-13(21)(29), automatically travels to the full open position.</td></tr>
<tr><td></td><td>BOP</td><td>Verifies CCW Pump 3A Discharge valve closed and starts CCW Pump 3A, recognizes CCW Pump 3A discharge valve does not fully open</td></tr>
<tr><td></td><td></td><td></td></tr>
</table>

Simulator Event Guide:

Event 1 Component: Returning to service CCW Pump 3A, Disch Valve FTO

	NRC	When BOP recognizes failure of CCW Pump 3A Disch Valve to fully open candidate may either 1) call AUO to reset breaker OR 2) Secure the Pump
	Driver	If crew secures 3A CCW pump contact Unit Supervisor as the Shift Manager and inform that power needs to be increased to 100% or 26" Hg Vacuum whichever comes first by ODS request
	Driver	If crew contacts AUO to investigate breaker or reset, wait 3 minutes and report that 3-FCV-27-13 breaker on 480V Water Supply Board Compt 25B indicates thermal overload If ordered to reset breaker, report breaker has been reset ask operator to take discharge valve to Open, wait a few seconds and DOR zlohs2713a[1] on And DOR zaoei2710a to return 3A CCW pump amps to normal
	BOP	If crew ordered AUO to reset breaker, BOP shall verify that 3A CCW pump discharge valve travels full open or open the valve
	BOP	IF breaker is NOT reset, BOP operator secures 3A CCW Pump

Simulator Event Guide:

Event 2 Reactivity: Power increase with Recirc Flow

	SRO	Notifies ODS of power increase.
		Directs Power increase using Recirc Flow, per 3-GOI-100-12. [21] **WHEN** desired to restore Reactor power to 100%, **THEN PERFORM** the following as directed by Unit Supervisor and recommended by the Reactor Engineer: • **RAISE** power using control rods or core flow changes. **REFER TO** 3-SR-3.3.5(A) and 3-OI-68.
	ATC	Raise Power w/Recirc, IAW 3-OI-68, Section 6.2
		D. Individual pump speeds should be mismatched by ~60 RPM during dual pump operation between 1200 and 1300 RPM to minimize harmonic vibration (this requirement may be waived for short time periods for testing or maintenance). [1] **IF** desired to control Recirc Pumps 3A and/or 3B speed with Recirc Individual Control, **THEN PERFORM** the following; • Raise Recirc Pump 3A using, RAISE SLOW (MEDIUM), 3-HS-96-15A(15B). **AND/OR** • Raise Recirc Pump 3B using, RAISE SLOW (MEDIUM), 3-HS-96-16A(16B).
		[2] **WHEN** desired to control Recirc Pumps 3A and/or 3B speed with the RECIRC MASTER CONTROL, **THEN ADJUST** Recirc Pump speed 3A & 3B using the following push buttons as required: RAISE SLOW, 3-HS-96-31 RAISE MEDIUM, 3-HS-96-32
	NOTE	One push of Raise Medium on the recirc speed controller is equivalent to 5 RPM speed change at 1 RPM per second
	NRC	Crew may only be increasing power to 26"Hg Vacuum depending on outcome of previous event. When satisfied with Reactivity Manipulation, Inadvertent start of RCIC
	Driver	When directed by NRC, Trigger 5 Inadvertent start of RCIC

Simulator Event Guide:

Event 3 Component: Inadvertent RCIC start w/ trip pushbutton failure

	BOP	Responds to alarm 9-3B, Window 27, RCIC Gland Seal Vacuum Tank Pressure High
		A. **VERIFY** RCIC VACUUM PUMP, 3-HS-71-31A, running. B. **VERIFY** RCIC VACUUM TANK CONDENSATE PUMP, 3-HS-71-29A, running. C. **VERIFY** the following valves open: • RCIC LUBE OIL COOLING WTR VLV, 3-FCV-71-25A • RCIC VACUUM PUMP DISCHARGE VLV, 3-HCV-71-32
	BOP	While responding to alarm determines RCIC is running and reports to SRO
		Verifies by multiple indications that initiation signal is not valid and reports it to SRO
	SRO	Directs BOP to trip RCIC
	BOP	Attempts to trip RCIC, recognizes RCIC failed to trip with the Trip Pushbutton. Operator performs actions that should have automatically occurred when tripped IAW 3-OI-71, Section 8.4
		8.4 RCIC Turbine Trip **NOTES** 1) The following signals cause a RCIC turbine trip. The RPV High Water Level Trip Signal closes the RCIC TURBINE STEAM SUPPLY VLV, 3-FCV-71-8, and RCIC PUMP MIN FLOW VALVE, 3-FCV-71-34. All other trip signals close the RCIC TURBINE TRIP/THROTTLE VLV, 3-FCV-71-9 and RCIC PUMP MIN FLOW -VALVE, 3-FCV-71-34. • High RPV Water Level (+51 inches, Auto Reset at -45") • Low Pump Suction Pressure (10 inches HG vacuum) • High Turbine Exhaust Pressure (50 psig) • Turbine Overspeed (Mechanical, 122.3% of rated signal) • Automatic Isolation • Manual Pushbutton 2) All operations are performed at Panel 3-9-3 unless otherwise noted. [1] **IF** RCIC Turbine did **NOT** trip from high RPV water level, **THEN VERIFY** the following automatic actions: A. RCIC TURB TRIP/THROTTLE VLV, 3-FCV-71-9, closes B. RCIC PUMP MIN FLOW VALVE, 3-FCV-71-34, closes C. RCIC TURB SPEED, 3-SI-71-42A, indicates zero rpm
		Operator shuts the 71-9 and 71-34. Operator recognizes turbine is now shutting down, however, the RCIC Min Flow Valve will not remain shut because an inadvertent initiation signal is sealed in, BOP reports this to SRO
	SRO	Directs BOP to close RCIC Min Flow Valve and have operator in field open breaker

Simulator Event Guide:

Event 3 Component: Inadvertent RCIC start w/ trip pushbutton failure

	ATC	Reports power/level/pressure stable after RCIC secured
	BOP	Dispatches personnel to RCIC Min flow valve breaker at 250V RMOV BD 3B, Compt 5D to open breaker when valve is closed
	BOP	Dispatches Instrument Mechanics to investigate inadvertent initiation signal
	Driver	Acknowledge dispatch to breaker, wait 3 minutes and report on station at 250V RMOV BD 3B, Compt 5D, when directed insert override to open breaker for 71-34 valve: **ior ypovfcv7134 fail_power_now** Acknowledge dispatch as Instrument Mechanic
	BOP	Reports to SRO that 71-34 valve is closed and breaker is open
	SRO	Evaluates Technical Specification 3.5.3 Condition A: RCIC system inoperable Required Action A.1: Verify by administrative means HPCI system is operable Required Action A.2: Restore RCIC system to operable status Completion Time A.1: Immediately Completion Time A.2: 14 days
	NRC	When ready, CRD Pump 3A trip
	Driver	When directed by NRC, call the BOP operator to have candidate check which RCW pumps are running and when BOP is away from panel 3-9-5 insert trigger 10 for CRD Pump 3A trip

Simulator Event Guide:

Event 4 Component: CRD Pump 3A trip

	ATC	Reports Trip of CRD Pump 3A.
	SRO	Announces entry into 3-AOI-85-3, "CRD System Failure".
		4.1 Immediate Actions [1] **IF** operating CRD PUMP has failed **AND** the standby CRD Pump is available, **THEN PERFORM** the following at Panel 3-9-5: [1.1] **PLACE** CRD SYSTEM FLOW CONTROL, 3-FIC-85-11, in MAN at minimum setting. [1.2] **START** associated standby CRD Pump using one of the following: • CRD PUMP 3B, using 3-HS-85-2A [1.3] **ADJUST** CRD SYSTEM FLOW CONTROL, 3-FIC-85-11, to establish the following conditions: • CRD CLG WTR HDR DP, 3-PDI-85-18A, approximately 20 psid • CRD SYSTEM FLOW CONTROL, 3-FIC-85-11, between 40 and 65 gpm. [1.4] **BALANCE** CRD SYSTEM FLOW CONTROL, 3-FIC-85-11, and **PLACE** in AUTO or BALANCE.
	Driver	If Dispatched to CRD Pump 3A, pump is extremely hot to touch. CRD Pump 3B - oil levels in band, pump ready for start, conditions normal after the start. CRD 3A - report breaker tripped on over current, Electrical Maint called.
	NRC	When ATC begins to restore CRD parameters, Control Rod 46-19 drifts in to position 40
	Driver	When directed by NRC, insert trigger 12, Control Rod 46-19 drifts in. **When rod gets to position 42 on Full Core Display delete the rod drift.**

Simulator Event Guide:

Event 5 Component: Control Rod 46-19 drifts in to position 40

	ATC	Report Control Rod Drift Alarm 5A-28, reports Control Rod 46-19 drifting in.
	SRO	Enter 3-AOI-85-5 Rod Drift In.
	ATC	**4.1 Immediate Actions** [1] **IF** multiple rods are drifting into core, **THEN MANUALLY SCRAM** Reactor. Refer to 3-AOI-100-1.
	SRO	**4.2 Subsequent Actions** [1] **IF** a Control Rod is moving from its intended position without operator actions, **THEN INSERT** the Control Rod to position 00 using CONTINUOUS IN. [2] **NOTIFY** the Reactor Engineer to Evaluate Core Thermal Limits and Preconditioning Limits for the current Control Rod pattern. [3] **IF** another Control Rod Drift occurs before Reactor Engineering completes the evaluation, **THEN MANUALLY SCRAM** Reactor and enter 3-AOI-100-1. [4] **CHECK** Thermal Limits on ICS (RUN OFFICIAL 3D). [5] **ADJUST** control rod pattern as directed by Reactor Engineer and **CHECK** Thermal Limits on ICS (RUN OFFICIAL 3D).
	ATC	Reports rod 46-19 stopped drifting at position 40
	Driver	When ATC begins to insert control rod 46-19 stick the rod at position 20 by inserting **trigger 13**
	ATC	Inserts Control Rod 46-19 to position 00.
	ATC	Reports Control Rod 46-19 is stuck at position 20
	SRO	May or May NOT direct actions for Control Rod difficult to insert
	ATC	**8.16 Control Rod Difficult to Insert** [1] **VERIFY** the control rod will not notch in, in accordance with Section 6.7 or Section 8.19. [2] **REVIEW** all Precautions and Limitations in Section 3.0. [3] [NRC/C] **IF** RWM is enforcing, **THEN VERIFY** RWM operable and LATCHED in to the correct ROD GROUP. [NRC IR 84-02] [4] **CHECK** CRD SYSTEM FLOW is between 40 gpm and 65 gpm, indicated by 3-FIC-85-11.

Simulator Event Guide:

Event 5 Component: Control Rod 46-19 drifts in to position 40

	ATC	**8.16 Control Rod Difficult to Insert (contd)** [5] **CHECK** CRD DRIVE WTR HDR DP, 3-PDI-85-17A is between 250 psid and 270 psid. [6] **IF** CRD SYSTEM FLOW or CRD DRIVE WTR HDR DP had to be adjusted, **THEN PROCEED** to Section 6.7. [7] **IF** control rod motion is observed, but the CRD fails to notch-in with normal operating drive water pressure, **THEN**: [7.1] **NOTIFY** Reactor Engineer to determine what parameters should be recorded for further evaluation. [7.2] **RAISE** CRD DRIVE WTR HDR DP, 3-PDI-85-17A, not to exceed 300 psid, using CRD DRIVE WATER PRESS CONTROL VLV, 3-HS-85-23A. [7.3] **INSERT** control rod as directed in Section 6.7. [7.4] **LOWER** CRD DRIVE WTR HDR DP, 3-PDI-85-17A, to between 250 psid and 270 psid using CRD DRIVE WATER PRESS CONTROL VLV, 3-HS-85-23A. [12] **IF** the control rod still fails to notch in, **THEN**: [12.1] **NOTIFY** the Unit Supervisor and Reactor Engineer to Refer to section Stuck Control Rod-Test to distinguish a Hydraulic Problem from Mechanical Binding, 0-TI-20, and **RETURN** to Section 8.16. [12.2] **REQUEST** the Unit Supervisor and Reactor Engineer to evaluate the control rod operability. Refer to Tech Spec 3.1.

Simulator Event Guide:

Event 5 Component: Control Rod 46-19 drifts in to position 40

	Driver	As Reactor engineer acknowledge rod drift, if asked for rod pattern adjustment, inform crew that you are working on it As AUO after dispatched report scram valves are normal.	
	SRO	Evaluate Tech Spec 3.1.3	
		Condition A	One withdrawn control rod stuck
		Required Action A.1 Completion Time	Verify stuck control rod separation criteria are met Immediately
		AND	
		Required Action A.2 Completion Time	Disarm the associated CRD 2 Hours
		AND	
		Required Action A.3 Completion Time	Perform SR3.1.3.3 for each withdrawn OPERABLE control rod. 24 Hours from discovery of Condition A
		AND	
		Required Action A.4 Completion Time	Perform SR3.1.1.1 72 Hours
NRC	**NOTE**	TRM 3.3.5 may be appropriate IF control rod 46-19 is stuck at a point where no position indication is available.	
	NRC	When ready, Steam leak in the RCIC room RCIC Steam line isolation valves 3-FCV-71-2 and 3 will not auto isolate	
	Driver	When directed by NRC, insert trigger 6 for RCIC room steam leak	

Simulator Event Guide:

Event 6 Component: Steam leak in the RCIC room
RCIC Steam line isolation valves 3-FCV-71-2 and 3 will not auto isolate

	BOP	Respond to Annunciator RX BLDG AREA RADIATION HIGH A. **DETERMINE** area with high radiation level on Panel 3-9-11. (Alarm on Panel 3-9-11 will automatically reset if radiation level lowers below setpoint.) C. **NOTIFY** RADCON. D. **IF** the TSC is **NOT** manned and a "VALID" radiological condition exists, **THEN USE** public address system to evacuate area where high airborne conditions exist.
	BOP	Determine RCIC Area Radiation Monitor is in Alarm and report, Evacuate affected area and notify radiation protection.
	BOP	Respond to annunciator RCIC STEAM LINE LEAK DETECTION TEMP HIGH If temperature continues to rise it will cause isolation of the following valves at steam line space temperature of 165°F Torus Area or 165°F RCIC Pump Room. • RCIC STEAM LINE INBD ISOLATION VLV, 3-FCV-71-2 • RCIC STEAM LINE OUTBD ISOLATION VLV, 3-FCV-71-3 A. **CHECK** RCIC temperature switches on LEAK DETECTION SYSTEM TEMPERATURE indicator, 3-TI-69-29 on Panel 3-9-21. B. **IF** RCIC is **NOT** in service **AND** 3-FI-71-1A(B), RCIC STEAM FLOW indicates flow, **THEN ISOLATE** RCIC and **VERIFY** temperatures lowering. C. **IF** high temperature is confirmed, **THEN ENTER** 3-EOI-3 Flowchart. D. **CHECK** CS/RCIC ROOM El 519 RX BLDG radiation indicator, 3-RI-90-26A on Panel 3-9-11 and **NOTIFY** RADCON if rising radiation levels are observed. E. **DISPATCH** personnel to investigate.

Simulator Event Guide:

Event 6 Component: Steam leak in the RCIC room
RCIC Steam line isolation valves 3-FCV-71-2 and 3 will not auto isolate

	BOP	Reports rising temperature in RCIC, reports that 71-2 and 71-3 failed to auto isolate. Based on amber lights on Panel 3-9-3.
	BOP	Reports 3-FCV-71-2 failed to close manually, 3-FCV-71-3 is closed
	SRO	Enter EOI-3 on Secondary Containment Area Radiation
	Driver	If dispatched to RCIC area report after 5 minutes that cannot access area at this time.
	SRO	If Reactor Zone or Refuel Zone Exhaust Radiation Level is above 72 mr/hr. Then verify isolation of Reactor Zone or Refuel Zone and verify SGTS initiates
		If above 72 mr/hr direct Operator to verify isolation of ventilation system and SGTS initiated
	ATC/BOP	Verifies Reactor Zone and Refuel Zone Ventilation Systems isolated and SGTS initiated
	SRO	If Reactor Zone or Refuel Zone Exhaust Ventilation isolated and ventilation radiation levels are below 72 mr/hr Then Restart Reactor Zone and Refuel Zone Ventilation per Appendix 8F
		If ventilation isolated and below 72 mr/hr directs Operator to perform Appendix 8F
CT#3	SRO	**Enters EOI-3 on High Secondary Containment Temperature** **Secondary Containment Temperature** Monitor and Control Secondary Containment Temperature Operate available ventilation per Appendix 8F Is Any Area Temp Above Max Normal - **YES** Isolate all systems that are discharging into the area except systems required to: • Be operated by EOIs <u>OR</u> • Suppress a Fire
CT#3	BOP	Isolates RCIC Steam Lines and reports Temperatures and Radiation Levels lowering
	SRO	Evaluates Technical Specification 3.6.1.3 Condition B Condition B — One or more penetration flow paths with two PCIVs inoperable except due to MSIV leakage not within limits. Required Action B.1 — Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. Completion Time — 1 Hour
NRC	**NOTE**	SRO may initially choose TS 3.3.6.1, PCIS Instrumentation.

Simulator Event Guide:

Event 6 Component: Steam leak in the RCIC room
RCIC Steam line isolation valves 3-FCV-71-2 and 3 will not auto isolate

	SRO	**Enters EOI-3 on High Secondary Containment Temperature (continued)** **Secondary Containment Radiation** Monitor and Control Secondary Containment Radiation Levels Is Any Area Radiation Level Max Normal - **NO** Isolate all systems that are discharging into the area except systems required to: • Be operated by EOIs OR • Suppress a Fire
		Ensures no systems are still discharging to Secondary Containment, remains in EOI-3 until entry conditions are cleared.
	SRO	**Enters EOI-3 on High Secondary Containment Temperature (continued)** **Secondary Containment Level** Monitor and Control Secondary Containment Water Levels Is Any Floor Drain Sump Above 66 inches - **NO** AND Is Any Area Water Level Above 2 inches - **NO**
	NRC	When ready, MSL A Break in Reactor BLDG with MSL A valves failing to close
	Driver	When directed by NRC, insert trigger 15 for MSL A Break in Rx Building

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	BOP	Responds to Rx Building Area Radiation High Alarm, 3A-22 and Rx BLDG, TURB BLDG, RF ZONE EXH Radiation High 3A-4 alarms.
		A. **DETERMINE** area with high radiation level on Panel 2-9-11. (Alarm on Panel 2-9-11 will automatically reset if radiation level lowers below setpoint.) D. **NOTIFY** RAD PRO. E. **IF** the TSC is **NOT** manned and a "VALID" radiological condition exists, **THEN USE** public address system to evacuate area where high airborne conditions exist. G. **MONITOR** other parameters providing input to this annunciator frequently as these parameters will be masked from alarming while this alarm is sealed in. J. For all radiation indicators except FUEL STORAGE POOL radiation indicator, 2-RI-90-30, **ENTER** 2-EOI-3 Flowchart.
	BOP	Determines Suppression Pool Area ARM, 90-29A, is in alarm and several other ARMs on Panel 9-11 are showing elevated radiation Uses Public Address System to evacuate the affected area(s) Reports to SRO the current Radiological conditions and trends and reports EOI-3 entry conditions
	SRO	Enters **EOI-3** on Secondary Containment Radiation
	BOP	Responds to Main Steam Line Leak Detection Temperature High alarm, 3D-24
		A. **CHECK** the following temperature indications: • MN STEAM TUNNEL TEMP temperature indicator, 3-TIS-1-60A on Panel 3-9-3.
	BOP	Determines Main Steam Tunnel Temperature on 3-TIS-1-60A is rising and reports to SRO
CT #1	SRO	Determines leak is in the Main Steam Tunnel from a MSL and determines a trigger value for Rx Scram and MSIV isolation before Main Steam Tunnel temperature reaches 189F OR any area radiation reading reaches 1000 mr/hr.
	SRO	May or May NOT direct Core flow runback prior to Reactor Scram
	ATC	If directed insert Core Flow Runback, by depressing the core flow runback pushbutton.
	Driver	When ATC arms and depresses ARI, insert trigger 20 for RFPT trips and Fuel Failure

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	Driver	After ATC has armed and depressed ARI, insert trigger 20 for RFPT trips and Fuel Failure
CT #1	SRO	Directs ATC to insert manual Rx Scram prior to MSIV isolation at a Steam Tunnel Temperature of 189F OR any area radiation reading reaches 1000 mr/hr. Directs BOP to shut MSIVs after Scram and prior to MSIV isolation at a Steam Tunnel Temperature of 189F
CT #1	ATC	Inserts Manual Rx Scram and performs immediate actions of 3-AOI-100-1, Reactor Scram
		[1] **DEPRESS** REACTOR SCRAM A and B, 3-HS-99-5A/S3A and 3-HS-99-5A/S3B, on Panel 3-9-5. [2] **IF** scram is due to a loss of RPS, **THEN PAUSE** in START & HOT STBY mode for approximately 5 seconds before going to REFUEL. (Otherwise N/A) [3] Refuel Mode One Rod Permissive Light check [3.1] **PLACE** REACTOR MODE SWITCH, 3-HS-99-5A-S1, in REFUEL. [3.2] **CHECK** REFUEL MODE ONE ROD PERMISSIVE light, 3-XI-85-46, illuminates. [3.3] **IF** REFUEL MODE ONE ROD PERMISSIVE light, 3-XI-85-46, is **NOT** illuminated, **THEN CHECK** all control rod positions at Full-In Overtravel, or Full-In. (Otherwise N/A) [4] **PLACE** REACTOR MODE SWITCH, 3-HS-99-5A-S1, in the SHUTDOWN position. [5] **IF** all control rods **CAN NOT** be verified fully inserted, **THEN INITIATE** ARI by Arming and Depressing: (Otherwise N/A) • ARI Manual Initiate, 3-HS-68-119A **OR** • ARI Manual Initiate, 3-HS-68-119B [6] **REPORT** the following status to the US: • Reactor Scram • Mode Switch is in Shutdown • "All rods in" or "rods out " • Reactor Water Level and trend (recovering or lowering). • Reactor pressure and trend • MSIV position (Open or Closed) • Power level [7] US **REPEAT** back status to UO, eye contact is not necessary.
CT #4 **[#4**	ATC	Depresses Reactor Scram A and B pushbuttons, places the Mode Switch in Shutdown, and reports "No Rod Motion." Initiates ARI by Arming and Depressing one of the ARI Manual Initiate collars and pushbuttons then reports "I have rod motion." Verifies all rods insert and makes Scram Report to the SRO

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	SRO	Provides repeat back of Scram Report with All Rods Inserted and enters **EOI-1** on Low Reactor Water Level after Scram
	BOP	After Reactor Scram and Turbine Trip, Shuts all MSIVs to isolate the leak Reports to the SRO that the A MSL MSIVs failed to isolate manually or automatically
	SRO	**Enters EOI-3 on High Secondary Containment Temperature or Radiation**
	SRO	**IF** Reactor Zone **or** Refuel Zone Exhaust Ventilation isolated and ventilation radiation levels are below 72 mr/hr **THEN** Restart Reactor Zone and Refuel Zone Ventilation, per Appendix 8F. Defeat isolation interlocks if necessary, Appendix 8E.
		If ventilation isolated and below 72 mr/hr, directs Operator to perform Appendix 8F.
	ATC/BOP	3-EOI Appendix 8F 1. **VERIFY** PCIS Reset. 2. **PLACE** Refuel Zone Ventilation in service as follows (Panel 3-9-25): a. **VERIFY** 3-HS-64-3A, REFUEL ZONE FANS AND DAMPERS, control switch is in OFF. b. **PLACE** 3-HS-64-3A, REFUEL ZONE FANS AND DAMPERS, control switch to SLOW A (SLOW B). c. **CHECK** two SPLY/EXH A(B) green lights above 3-HS-64-3A, REFUEL ZONE FANS AND DAMPERS, control switch extinguish and two SPLY/EXH A(B) red lights illuminate. d. **VERIFY OPEN** the following dampers: • 3-FCO-64-5, REFUEL ZONE SPLY OUTBD ISOL DMPR • 3-FCO-64-6, REFUEL ZONE SPLY INBD ISOL DMPR • 3-FCO-64-9, REFUEL ZONE EXH OUTBD ISOL DMPR • 3-FCO-64-10, REFUEL ZONE EXH INBD ISOL DMPR
	BOP	Dispatches personnel to investigate A MSIVs and manually close Outboard MSIVs
	Driver	If requested, wait 3 minutes and report Appendix 8E complete, **enter** bat app08e If dispatched for A MSIVs, acknowledge dispatch

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	SRO	Monitor and Control Secondary Containment Temperature.
		Operate available ventilation, per Appendix 8F.
		Is Any Area Temp Above Max Normal? - **YES**
		Isolate all systems that are discharging into the area except systems required to: • Be operated by EOIs **OR** • Suppress a Fire
CT #2		Will Emergency Depressurization Reduce Discharge Into Secondary Containment? - **YES**
		Proceeds to the STOP sign Before any area temp rises to Max Safe (table 5) Continue:
CT #1		**Enters** EOI-1 RPV Control **and** directs Reactor Scram **before** any temperature exceeds MAX Safe. (Reactor Scram already conducted to prevent automatic Scram from occurring when MSIVs isolated on High Temperature)
CT #2		Stops at Stop sign When temperatures in two or more areas are above Max Safe, Then Emergency Depressurization is required.
	Crew	Monitors for Max Safe Temperatures
	SRO	EOI-3 Secondary Containment (Level)
		Monitor and Control Secondary Containment Water Levels.
		Is Any Floor Drain Sump Above 66 inches?– **NO** Is Any Area Water Level Above 2 inches? - **NO**

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	SRO	EOI-3 Secondary Containment (Radiation)
		Monitor and Control Secondary Containment Radiation Levels.
		Is Any Area Radiation Level Above Max Normal? - YES
		Isolate all systems that are discharging into the area except systems required to: • Be operated by EOIs **OR** • Suppress a Fire (MSIVs have already been shut to prevent automatic isolation, however MSL A MSIVs did not shut)
		Will Emergency Depressurization Reduce Discharge Into Secondary Containment? - **YES**
		Before any area radiation rises to Max Safe (table 4) Continue and enter EOI-1 (EOI-1 has already been entered after Reactor Scram)
CT #2		Stops at Stop sign When radiation levels in two or more areas are above Max Safe, Then Emergency Depressurization is required.
	Crew	Monitors for Max Safe Radiation and reports (Suppression Pool Area, 90-29A, and CRD West, 90-20A, will be the first two Max Safe Radiation Areas in that order)
	ATC	Reports that RFPTs tripped after Reactor Scram and Reactor Water Level and pressure are lowering

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	SRO	Enters **EOI-1** on Low Reactor Water Level after Scram
	SRO	Reactor Pressure
		Monitor and Control Reactor Pressure
		IF Drywell Pressure Above 2.4 psig ?- **NO**
		IF Emergency Depressurization is Anticipated and the Reactor will remain subcritical without boron under all conditions, **THEN** Rapidly depressurize the RPV with the Main Turbine Bypass Valves irrespective of cooldown rate. May Answer **YES**; during Scenario and direct Bypass Valves opened to Depressurize through the open MSIVs on the A MSL.
		IF Emergency Depressurization is required, **THEN** exit RC/P and enter C2 Emergency Depressurization. Answers **YES**; when two area radiation levels have reached MAX Safe.
		IF RPV water level cannot be determined? - **NO**
		Is any MSRV Cycling? - **NO**
		IF Steam cooling is required? - **NO**
		IF Suppression Pool level and temperature cannot be maintained in the safe area of Curve 3? - **NO**
		IF Suppression Pool level cannot be maintained in the safe area of Curve 4? - **NO**
		IF Drywell Control air becomes unavailable? - **NO**
		IF Boron injection is required? - **NO**
	SRO	Directs a Pressure Band, however, Reactor Pressure will be slowly lowering due to leak on the A MSL. If ED is not anticipated directs Reactor Pressure controlled using SRVs, if necessary, using 3-EOI-Appendix-11A
	ATC/BOP	Controls Reactor Pressure as directed and if ED anticipated opens Bypass Valves to Rapidly Depressurize the RPV irrespective of cooldown rate.
	Driver	If ED anticipated, Fuel Failure may have to be increased to force crew to ED on two Max Safe Rad Levels

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	BOP	**Maintains prescribed pressure band per 3-EOI-Appendix-11A, if necessary** 1. IF Drywell Control Air is NOT available, THEN **EXECUTE** EOI Appendix 8G, CROSSTIE CAD TO DRYWELL CONTROL AIR, **CONCURRENTLY** with this procedure. 2. IF Suppression Pool level is at or below 5.5 ft, THEN **CLOSE** MSRVs and **CONTROL** RPV pressure using other options. 3. **OPEN** MSRVs using the following sequence to control RPV pressure as directed by SRO: a. 1 3-PCV-1-179 MN STM LINE A RELIEF VALVE. b. 2 3-PCV-1-180 MN STM LINE D RELIEF VALVE. c. 3 3-PCV-1-4 MN STM LINE A RELIEF VALVE. d. 4 3-PCV-1-31 MN STM LINE C RELIEF VALVE. e. 5 3-PCV-1-23 MN STM LINE B RELIEF VALVE. f. 6 3-PCV-1-42 MN STM LINE D RELIEF VALVE. g. 7 3-PCV-1-30 MN STM LINE C RELIEF VALVE. h. 8 3-PCV-1-19 MN STM LINE B RELIEF VALVE. i. 9 3-PCV-1-5 MN STM LINE A RELIEF VALVE. j. 10 3-PCV-1-41 MN STM LINE D RELIEF VALVE. k. 11 3-PCV-1-22 MN STM LINE B RELIEF VALVE. l. 12 3-PCV-1-18 MN STM LINE B RELIEF VALVE. m. 13 3-PCV-1-34 MN STM LINE C RELIEF VALVE.
	SRO	Reactor Level
		Monitor and Control Reactor Level
		Verify as required PCIS isolations
	SRO	IF It has not been determined that the reactor will remain subcritical? - **NO**
		IF RPV water level cannot be determined? - **NO**
		IF PC water level cannot maintained below 105 feet? - **NO**
		Restores and Maintains RPV Water Level between +2 and +51 inches, with one of the following injection sources: Directs a Level Band of (+) 2 to (+) 51 inches with HPCI, 3-EOI-Appendix-5D.

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	BOP	**Maintains the prescribed level band, per 3-EOI-Appendix-5D.** 1. IF Suppression Pool level drops below 12.75 ft during HPCI operation, THEN **TRIP** HPCI and **CONTROL** injection using other options. 2. IF Suppression Pool level CANNOT be maintained below 5.25 in., THEN **EXECUTE** EOI Appendix 16E concurrently with this procedure to bypass HPCI High Suppression Pool Water Level Suction Transfer Interlock. 3. IF BOTH of the following exist: • High temperature exists in the HPCI area, **AND** • SRO directs bypass of HPCI High Temperature Isolation interlocks, THEN **PERFORM** the following: a. **EXECUTE** EOI Appendix 16L concurrently with this procedure. b. **RESET** auto isolation logic using 3-XS-73-58A(B) HPCI AUTO-ISOL LOGIC A(B) RESET pushbuttons. **CAUTION** • Operating HPCI Turbine below 2400 rpm may result in unstable system operation and equipment damage. • Operating HPCI Turbine with suction temperatures above 140°F may result in equipment damage. 4. **VERIFY** 3-IL-73-18B, HPCI TURBINE TRIP RX LVL HIGH amber light extinguished. 5. **VERIFY** at least one SGTS train in operation. 6. **VERIFY** 3-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller in AUTO and set for 5300 gpm. NOTE HPCI Auxiliary Oil Pump will NOT start UNTIL 3-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, starts to open.

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	BOP	**Maintains the prescribed level band, per 3-EOI-Appendix-5D (cont'd).** 7. **PLACE** 3-HS-73-47A, HPCI AUXILIARY OIL PUMP, handswitch in START. 8. **PLACE** 3-HS-73-10A, HPCI STEAM PACKING EXHAUSTER, handswitch in START. 9. **OPEN** the following valves: • 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE • 3-FCV-73-44, HPCI PUMP INJECTION VALVE. 10. **OPEN** 3-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, to start HPCI Turbine. 11. **CHECK** proper HPCI operation by observing the following: a. HPCI Turbine speed accelerates above 2400 rpm. b. 3-FCV-73-45, HPCI TESTABLE CHECK VLV, opens by observing 3-ZI-73-45A, DISC POSITION, red light illuminated. c. HPCI flow to RPV stabilizes and is controlled automatically at 5300 gpm. d. 3-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes as flow exceeds 1200 gpm. 12. **VERIFY** HPCI Auxiliary Oil Pump stops and the shaft-driven oil pump operates properly. 13. WHEN HPCI Auxiliary Oil Pump stops, THEN **PLACE** 3-HS-73-47A, HPCI AUXILIARY OIL PUMP, handswitch in AUTO. 14. **ADJUST** 3-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller as necessary to control injection.
	BOP	Reports to SRO that HPCI Flow Control Valve has failed in automatic control
		Takes manual control of HPCI Flow Control Valve and controls injection to maintain prescribed level band

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	SRO	Reactor Power
		Monitor and control Reactor Power
		If the Reactor is Subcritical and no Boron has been injected then exit RC/Q and enter 3-AOI-100-1, Reactor Scram - **YES**
	ATC	When time permits performs subsequent actions of 3-AOI-100-1
CT #2	SRO	Enters 3-C-2, "Emergency Depressurization" when two Max Safe Rad levels are reached
		Will the Reactor Remain Subcritical Without Boron Under All Conditions ?- **YES**
		Is Drywell Pressure Above 2.4 psig? - **NO**
		Is Suppression Pool Level Above 5.5 feet? - **YES**
		Directs All ADS Valves Open.
CT #2	ATC/BOP	Opens 6 ADS Valves within five minutes of exceeding the MAX safe values.
	SRO	Can 6 ADS Valves Be Opened? - **YES**
	SRO	Directs Level Control transitioned to Condensate per 3-EOI-Appendix-6A
	ATC	**Maintains prescribed level band per 3-EOI-Appendix-6A** 1. **VERIFY CLOSED** the following Feedwater heater return valves: • 3-FCV-3-71, HP HTR 3A1 LONG CYCLE TO CNDR • 3-FCV-3-72, HP HTR 3B1 LONG CYCLE TO CNDR • 3-FCV-3-73, HP HTR 3C1 LONG CYCLE TO CNDR 2. **VERIFY CLOSED** the following RFP discharge valves: • 3-FCV-3-19, RFP 3A DISCHARGE VALVE • 3-FCV-3-12, RFP 3B DISCHARGE VALVE • 3-FCV-3-5, RFP 3C DISCHARGE VALVE 3. **VERIFY OPEN** the following drain cooler inlet valves: • 3-FCV-2-72, DRAIN COOLER 3A5 CNDS INLET ISOL VLV • 3-FCV-2-84, DRAIN COOLER 3B5 CNDS INLET ISOL VLV • 3-FCV-2-96, DRAIN COOLER 3C5 CNDS INLET ISOL VLV 4. **VERIFY OPEN** the following heater outlet valves: • 3-FCV-2-124, LP HEATER 3A3 CNDS OUTL ISOL VLV • 3-FCV-2-125, LP HEATER 3B3 CNDS OUTL ISOL VLV • 3-FCV-2-126, LP HEATER 3C3 CNDS OUTL ISOL VLV

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	ATC	**Maintains prescribed level band per 3-EOI-Appendix-6A (cont'd)** 5. VERIFY OPEN the following heater isolation valves: • 3-FCV-3-38, HP HTR 3A2 FW INLET ISOL VLV • 3-FCV-3-31, HP HTR 3B2 FW INLET ISOL VLV • 3-FCV-3-24, HP HTR 3C2 FW INLET ISOL VLV • 3-FCV-3-75, HP HTR 3A1 FW OUTLET ISOL VLV • 3-FCV-3-76, HP HTR 3B1 FW OUTLET ISOL VLV • 3-FCV-3-77, HP HTR 3C1 FW OUTLET ISOL VLV 6. VERIFY OPEN the following RFP suction valves: • 3-FCV-2-83, RFP 3A SUCTION VALVE • 3-FCV-2-95, RFP 3B SUCTION VALVE • 3-FCV-2-108, RFP 3C SUCTION VALVE 7. VERIFY at least one condensate pump running. 8. VERIFY at least one condensate booster pump running. 9. ADJUST 3-LIC-3-53, RFW START-UP LEVEL CONTROL, to control injection (Panel 3-9-5). 10. VERIFY RFW flow to RPV.
	ATC	Verifies RFP discharge valves are closed prior to Reactor Pressure dropping below condensate system discharge pressure to prevent overfeeding the Reactor

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	SRO	Enters EOI-2 on High Suppression Pool Temperature
		EOI-2 (Drywell Temperature)
	SRO	Monitor and Control DW Temp Below 160°F, using available DW Cooling.
		Can Drywell Temp Be Maintained Below 160°F? - **YES**
	SRO	Verify H2O2 Analyzers placed in service, Appendix 19.
	BOP	Places H2O2 analyzers in service, IAW Appendix 19.
	SRO	EOI-2 Primary Containment (Pressure)
		Monitor and Control PC Pressure Below 2.4 psig, Using the Vent System As Necessary. (Appendix 12)
		Can Primary Containment pressure be maintained below 2.4 psig? - **YES**
	SRO	EOI-2 Suppression Pool (Temperature)
		Monitor and Control Suppression Pool Temperature Below 95°F, Using Available Suppression Pool Cooling As Necessary. (Appendix 17A)
		Can Suppression Pool Temperature Be Maintained Below 95°F? - NO
		Operate all available suppression pool cooling, using only RHR Pumps not required to assure adequate core cooling by continuous injection. (Appendix 17A)
	BOP/ATC	Places RHR in Suppression Pool Cooling, (IAW Appendix 17A)

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	SRO	EOI-2 Suppression Pool Level
		Monitor and Control Suppression Pool Level between -1 inch and -6inch, (Appendix 18).
		Can Suppression Pool Level be maintained above -6 inches - **Yes**
		Can Suppression Pool Level be maintained below -1 inches - **Yes**
	BOP	**Places RHR in Suppression Pool Cooling IAW 3-EOI-Appendix-17A** 1. IF Adequate core cooling is assured, **OR** Directed to cool the Suppression Pool irrespective of adequate core cooling, THEN **BYPASS** LPCI injection valve auto open signal as necessary by **PLACING** 3-HS-74-155A(B), LPCI SYS I(II) OUTBD INJ VLV BYPASS SEL in **BYPASS**. 2. **PLACE** RHR SYSTEM I(II) in Suppression Pool Cooling as follows: a. **VERIFY** at least one RHRSW pump supplying each EECW header. b. **VERIFY** RHRSW pump supplying desired RHR Heat Exchanger(s). c. **THROTTLE** the following in-service RHRSW outlet valves to obtain between 1350 and 4500 gpm RHRSW flow: • 3-FCV-23-34, RHR HX 3A RHRSW OUTLET VLV • 3-FCV-23-46, RHR HX 3B RHRSW OUTLET VLV • 3-FCV-23-40, RHR HX 3C RHRSW OUTLET VLV • 3-FCV-23-52, RHR HX 3D RHRSW OUTLET VLV. d. IF Directed by SRO, THEN **PLACE** 3-XS-74-122(130), RHR SYS I(II) LPCI 2/3 CORE HEIGHT OVRD in MANUAL OVERRIDE. e. IF LPCI INITIATION Signal exists, THEN MOMENTARILY PLACE 3 XS-74-121(129), RHR SYS I(II) CTMT SPRAY/CLG VLV SELECT in SELECT. f. IF 3-FCV-74-53(67), RHR SYS I(II) LPCI INBD INJECT VALVE, is OPEN, THEN **VERIFY CLOSED** 3-FCV-74-52(66), RHR SYS I(II) LPCI OUTBD INJECT VALVE. g. **OPEN** 3-FCV-74-57(71), RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV. h. **VERIFY** desired RHR pump(s) for Suppression Pool Cooling are operating.

Simulator Event Guide:

Event 7 Major: MSL A Break in Reactor BLDG with MSL A valves failing to close

	BOP	**Places RHR in Suppression Pool Cooling IAW 3-EOI-Appendix-17A (cont'd)** **CAUTION** RHR System flows below 7000 gpm or above 10000 gpm for one-pump operation may result in excessive vibration and equipment damage. i. **THROTTLE** 3-FCV-74-59(73), RHR SYS I(II) SUPPR POOL CLG/TEST VLV, to maintain EITHER of the following as indicated on 3-FI-74-50(64), RHR SYS I(II) FLOW: • Between 7000 and 10000 gpm for one-pump operation. **OR** • At or below 13000 gpm for two-pump operation. j. **VERIFY CLOSED** 3-FCV-74-7(30), RHR SYSTEM I(II) MIN FLOW VALVE. k. **MONITOR** RHR Pump NPSH using Attachment 1. l. **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers. m. IF Additional Suppression Pool Cooling flow is necessary, THEN **PLACE** additional RHR and RHRSW pumps in service using Steps 2.b through 2.l.
	SRO	Emergency Plan Classification 3.2-S

SHIFT TURNOVER SHEET

Equipment Out of Service/LCO's:

None

Operations/Maintenance for the Shift:

Return Condenser Circulating Water Pump 3A to service, IAW 3-OI-27, section 8.2[4]
Section 8.2 of 3-OI-27has been completed thru [3.3]. Cooling Towers are not in service.

Commence a power increase to 100%

Unit 1 and 2 are at 100% Power

Unusual Conditions/Problem Areas:

None

ATTACHMENT 2

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	22-23	10	00		
N/A	22-39	10	00		
N/A	38-23	10	00		
N/A	38-39	10	00		
N/A	30-31	14	00		
N/A	14-15	16	00		
N/A	14-47	16	00		
N/A	46-47	16	00		
N/A	46-15	16	00		
N/A	14-31	12	00		
N/A	30-47	12	00		
N/A	46-31	12	00		
N/A	30-15	12	00		
N/A	22-31	48	00		
N/A	30-39	48	00		
N/A	38-31	48	00		
N/A	30-23	48	00		

REMARKS[4]: Emergency Shove Sheet – Loadline reduction or Unit Shutdown Insert Rods Continuously to 00. Insertion may stop after completion of any group.

NOTES:

(1) RWM Group may be marked "N/A" if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position "48", this signoff verifies coupling integrity was checked in accordance with 3-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by: ____________________ / __________ Issued by ____________________ / __________
Unit Supervisor Date Reactor Engineer Date

ATTACHMENT 2
(Page 2 of 2)

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	22-47	48	00		
N/A	38-47	48	00		
N/A	38-15	48	00		
N/A	22-15	48	00		
N/A	14-39	48	00		
N/A	46-39	48	00		
N/A	46-23	48	00		
N/A	14-23	48	00		
N/A	06-31	48	00		
N/A	30-55	48	00		
N/A	54-31	48	00		
N/A	30-07	48	00		
N/A	06-39	48	00		
N/A	54-39	48	00		
N/A	54-23	48	00		
N/A	06-23	48	00		

REMARKS[4]: Emergency Shove Sheet – Loadline reduction or Unit Shutdown Insert Rods Continuously to 00. Insertion may stop after completion of any group.

NOTES:

(1) RWM Group may be marked "N/A" if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position "48", this signoff verifies coupling integrity was checked in accordance with 3-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by: ____________________ / __________ Issued by ____________________ / __________
Unit Supervisor Date Reactor Engineer Date

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

Reactivity Maneuver Plan U3 NRC Exam 2

Raise Reactor Power to 100%

Attachment 7
(Page 1 of 2)

Reactivity Control Plan Form

BFN Unit: 3 Valid Date(s): 8/8/11 – 8/19/11 Reactivity Control Plan #: **U3 NRC Exam 2**

Are Multiple Activations Allowed: No (If yes, US may make additional copies)

Prepared by: ____________ / ______
Reactor Engineer Date

Reviewed by: ____________ / ______
Qualified Reactor Engineer Date

Approved by: ____________ / ______
RE Supervisor Date

Concurrence: ____________ / ______
WCC/Risk/US SRO Date

Approved by: ____________ / ______
Ops Manager or Supt. Date

Authorized by: ____________ / ______
Shift Manager Date

RCP Activated: ____________ / ______
Unit Supervisor Date

RCP Terminated: ____________ / ______
Unit Supervisor Date

Title of Evolution: Raise Power to 100%

Purpose/Overview of Evolution: Raise Power to 100%

Maneuver Steps

1. Raise reactor power to 100% using core flow. (NO Ramp Rate Limits Apply)

Attachment 7
(Page 2 of 2)

Reactivity Control Plan Form

Operating Experience and General Issues: U3 NRC Exam 2

This plan is NOT valid if the unit is operating with a suspected or known fuel leaker and is not to be used. Contact Reactor Engineering if there are indications of a fuel leak.

Known Issues:

Cautions/Error Likely Situations/Special Monitoring Requirements/Contingencies:

NONE

Attachment 8

Reactivity Maneuver Instructions

STEP 1 of 1 **Reactivity Maneuver Plan # U3 NRC Exam 2**

Description of Step: Raise reactor power to 100% using core flow. No Ramp Rate Limits apply.

Conditions : To be recorded at the Completion of Step Recorded: ____________/______ (by RO) (Date)

QRE presence required in the Control Room? Yes _____ No _X_ (check)

	Predicted (may be ranges)	Actual		Predicted (may be ranges)	Actual
MW Electric	900-1150		MFLCPR	.80 - .85	
MW Thermal	2850-3450		MAPRAT	.55 - .65	
Core Flow	65-96 mlbm/hr		MFDLRX	.65 - .75	
Loadline	105-108				
Core Power	80% - 100%		Other		

Critical Parameters: To be recorded DURING Step. **IF** parameters are outside of the predictions, **THEN** discuss with the RE **AND** record conclusions in the Comments / Notes section.

Description including frequency, method of monitoring, **AND** contingency actions	High	Low
MWth Limit	3458	

Comments / Notes:

1. Raise Reactor Power to 100% RTP
2. Document core flow changes on Attachment 10

Step Complete **AND** Reviewed by:_________________/_______

Unit Supervisor / Date

Attachment 10
(Page 1 of 1)

Recirc Flow Maneuver Instructions

Reactivity Control Plan # U3 NRC Exam 2

RCP Step #	Flow Step #	Time	**Target Power (%RTP or MWe)**	Delta $\pm$(MWe)	**Target Flow (MLb/Hr)**	Completed (RO)
1			**100%**			

Comments / Notes:

Reviewed by: ____________________/________

Unit Supervisor / Date

Facility: **Browns Ferry NPP** Scenario No.: **NRC - 4** Op-Test No.: **1108**

Examiners: ____________________ Operators: **SRO:** ____________________

____________________ **ATC:** ____________________

____________________ **BOP:** ____________________

Initial Conditions: 100% power. HPCI is out of service.

Turnover: Transfer 4kV Unit board 3A from USST to Start Bus 1A 0-OI-57A section 8.15.1 starting at step [4]. Lower reactor power to 90% using recirc for surveillance testing.

Event No.	Malf. No.	Event Type*	Event Description
1		N-BOP N-SRO	Transfer 4KV UB-3A from USST 3B to Start Bus 1A IAW 0-OI-57A section 8.15.1
2		R-ATC R-SRO	Power decrease with flow
3	Batch File	TS-SRO	Core Spray Loop 1 Inoperable failed FCV-75-25
4	EG03	C-BOP C-SRO	Turbine Generator Voltage Regulator Failure
5	TH10/11b	C-ATC R-ATC TS-SRO	LOCA - Recirculation Pump B Inboard and Outboard seal failure
6	EG02	C-BOP C-SRO	Stator Water Cooling Pump Trip
7	TC10b	C-ATC C-SRO	EHC Pressure Transducer Failure
8		M-ALL	ATWS, without MSIVs
9	RC08	C	RCIC steam supply valve fails to auto open
10	IOR	C	CRD Controller Fails Low (FIC-85-11)

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Events

1. BOP Transfers 4KV Unit Board - 3A from USST 3B to Start Bus 1A IAW 0-OI-57A section 8.15.1 starting at step [4].

2. ATC lowers power with flow.

3. Core Spray Loop #1 – FCV-75-25 Loss of Power in Close position. SRO will determine Technical Specification 3.5.1 Condition A and D is applicable 72 hours to restore HPCI or Core Spray Loop 1 to Operable.

4. Turbine Generator Voltage Regulator will fail high in automatic and not transfer to manual. BOP will respond according to ARPs and transfer the voltage regulator to manual and restore Generator MVAR loading to normal.

5. #1 and #2 recirc pump seal failure – ATC will note alarm and report #2 seal carrying full pressure. A short time later seal #2 will fail ATC will note that a small LOCA exists. ATC will trip and isolate B RR Pump IAW with 3-AOI-68-1A. ATC will insert control rods to exit Region 2 of the power to flow map. SRO will determine Technical Specification 3.4.1 Condition A, is applicable with 24 hours to establish single loop conditions. Can follow up with RCS Operational Leakage Technical Specification prior to RR Loop isolation, Technical Specification 3.4.4 Condition A.

6. Stator Water Cooling Pump trip, BOP operator starts standby pump and restores stator water cooling prior to a turbine trip.

7. EHC Pressure Transducer Failure – non-operating pressure regulator takes control. This results in slowly decreasing reactor pressure. ATC inserts a scram and the BOP operator closes the MSIVs prior to reactor pressure lowering to less than 900 psig IAW 3-AOI-47-2.

8. ATWS exists on the scram the crew will enter EOI-1, EOI-2 and EOI-C-5. Crew will insert control rods, control reactor pressure on SRVs, initiate SLC.

9. RCIC steam supply valve will not auto-open on initiation signal, level will degrade until RCIC is manually started. Once started RCIC will maintain level above TAF.

10. CRD Controller will fail low ATC takes manual control of controller and restores CRD parameters

Terminate the scenario when the following conditions are satisfied or upon request of Lead Examiner:

Control Rods are being inserted

Reactor Level is being maintained

Reactor Pressure Controlled on SRVs

CRITICAL TASKS - Three

CT#1-With a reactor scram required and the reactor not shutdown, initiate action to reduce power by injecting boron (If still critical with challenge to BIIT) and inserting control rods.

1. Safety Significance:
   Shutting down reactor can preclude failure of containment or equipment necessary for the safe shutdown of the plant.

2. Cues:
   Procedural compliance
   Suppression Pool temperature

3. Measured by:
   Observation - If operating IAW EOI-1 and C-5, US determines that SLC is required (indicated by verbal direction or EOI placekeep5,ing action) before exceeding 110 degrees in the Suppression Pool.
   AND
   RO places SLC A / B Pump control switch in ON, when directed by US.
   AND
   Control Rod insertion commenced in accordance EOI Appendices.

4. Feedback:
   Reactor Power trend
   Control Rod indications
   SLC tank level

CT#2 – RPV Level maintained above -162 inches, RCIC has been manually initiated.

1. Safety Significance:
   Maintaining adequate core cooling

2. Cues:
   RPV level indication

3. Measured by:
   RCIC injecting at 600 gpm

4. Feedback:
   RPV level trend
   RCIC injection valve open

CT#3 - With reactor scram required and the reactor not shutdown, to prevent an uncontrolled RPV depressurization and subsequent power excursion, inhibit ADS.

1. Safety Significance:
   Precludes core damage due to an uncontrolled reactivity addition

2. Cues:
   Procedural compliance

3. Measured by:
   ADS logic inhibited prior to an automatic initiation unless all required injection systems are Terminated and Prevented.

4. Feedback:
   RPV pressure trend
   RPV level trend
   ADS annunciator status

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER: 4

7	Total Malfunctions Inserted: List (4-8)
2	Malfunctions that occur after EOI entry: List (1-4)
4	Abnormal Events: List (1-3)
1	Major Transients: List (1-2)
2	EOI's used: List (1-3)
1	EOI Contingencies used: List (0-3)
75	Validation Time (minutes)
3	Crew Critical Tasks: (2-5)
YES	Technical Specifications Exercised (Yes/No)

Scenario Tasks

TASK NUMBER	K/A	RO	SRO
Transfer 4KV Unit Board			
RO U-57A-NO-1 SRO S-57A-NO-4	262001A4.05	3.3	3.3
Lower Power with Recirc Flow			
RO U-068-NO-17 SRO S-000-NO-138	2.1.23	4.3	4.4
Stator Water Cooling Pump Trip			
RO U-35A-AL-2 SRO S-070-AB-1	245000A4.03	2.7	2.8
Turbine Generator Voltage Regulator Failure			
RO U-47-AL-20 SRO S-57A-AB-4	262001A2.09	3.1	3.4
RR Pump Seal Failure			
RO U-068-AL-9 SRO S-068-AB-1	203000A4.02	4.1	4.1
EHC Pressure Transducer Failure			
RO U-047-AB-2 SRO S-047-AB-2	241000A2.03	4.1	4.2
ATWS			
RO U-000-EM-35 SRO S-000-EM-1 SRO S-000-EM-2 SRO S-000-EM-3	295015AA2.01	4.1	4.3

Simulator Instructor - **IC-204**

#HPCI tagout
bat nrchpcito

#Tech Spec call SRO Core Spray System #1
ior ypovfcv7525 (e1 0) fail_now
ior xa553c[27] (e1 0) crywolf

#B stator water pump trip
irf eg02 (e5 0) off
ior ypobkrscwpa (e5 0) fail_ccoil
ior zdihs3535a[2] (e5 0) stop
ior zlohs3535a[1] (e5 0) off

#Turbine Generator Voltage Regulator failure
imf eg03 (e10 0)

#B Recirc pump seal failures
imf th12b (e15 0)
imf th10b (e15 0) 100
imf th11b (e15 180) 100 60 0

#B EHC Pressure transducer failure
bat atws70
ior zdihs0116[1] (e20 0) select
ior zdihs47204[1] (e20 0) null
ior zlohs0116[1] off
ior zlohs47204[1] on
imf tc10b (e20 0) 86 1200 79

#RCIC steam supply valve fails to auto open
imf rc08
trg 25 = bat sdv
trg 26 = bat atws-1
trg 27 = bat app01f
trg 28 = bat app02
trg 29 = bat app08ae

#After Scram manually insert under DI Overide
#3-FIC-85-11 0-100(L)

Facility: **Browns Ferry NPP** Scenario No.: **NRC – 4** Op-Test No.: **1108**

Examiners:____________________ Operators: **SRO:**____________________

____________________ **ATC:**____________________

____________________ **BOP:**____________________

Initial Conditions: 100% power. HPCI is out of service.

Turnover: Transfer 4kV Unit board 3A from USST to Start Bus 1A 0-OI-57A section 8.15.1 starting at step [4]. Lower reactor power to 90% using recirc for surveillance testing.

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2		R-ATC R-SRO	Power decrease with flow
3	Batch File	TS-SRO	Core Spray Loop 1 Inoperable failed FCV-75-25
4	EG03	C-BOP C-SRO	Turbine Generator Voltage Regulator Failure
5	TH10/11b	C-ATC R-ATC TS-SRO	LOCA - Recirculation Pump B Inboard and Outboard seal failure
6	EG02	C-BOP C-SRO	Stator Water Cooling Pump Trip
7	TC10b	C-ATC C-SRO	EHC Pressure Transducer Failure
8		M-ALL	ATWS, without MSIVs
9	RC08	C	RCIC steam supply valve fails to auto open
10	IOR	C	CRD Controller Fails Low (FIC-85-11)

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

CRITICAL TASKS - Three

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1. Safety Significance:
   Shutting down reactor can preclude failure of containment or equipment necessary for the safe shutdown of the plant.

2. Cues:
   Procedural compliance
   Suppression Pool temperature

3. Measured by:
   Observation - If operating IAW EOI-1 and C-5, US determines that SLC is required (indicated by verbal direction or EOI placekeeping action) before exceeding 110 degrees in the Suppression Pool.
   AND
   RO places SLC A / B Pump control switch in ON, when directed by US.
   AND
   Control Rod insertion commenced in accordance EOI Appendices.

4. Feedback:
   Reactor Power trend
   Control Rod indications
   SLC tank level

CT#2 – RPV Level maintained above -162 inches, RCIC has been manually initiated.

1. Safety Significance:
   Maintaining adequate core cooling

2. Cues:
   RPV level indication

3. Measured by:
   RCIC injecting at 600 gpm

4. Feedback:
   RPV level trend
   RCIC injection valve open

CT#3 - With reactor scram required and the reactor not shutdown, to prevent an uncontrolled RPV depressurization and subsequent power excursion, inhibit ADS.

1. Safety Significance:
   Precludes core damage due to an uncontrolled reactivity addition

2. Cues:
   Procedural compliance

3. Measured by:
   ADS logic inhibited prior to an automatic initiation unless all required injection systems are Terminated and Prevented.

4. Feedback:
   RPV pressure trend
   RPV level trend
   ADS annunciator status

Events

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2. ATC lowers power with flow.

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5. #1 and #2 recirc pump seal failure – ATC will note alarm and report #2 seal carrying full pressure. A short time later seal #2 will fail ATC will note that a small LOCA exists. ATC will trip and isolate B RR Pump IAW with 3-AOI-68-1A. ATC will insert control rods to exit Region 2 of the power to flow map. SRO will determine Technical Specification 3.4.1 Condition A, is applicable with 24 hours to establish single loop conditions. Can follow up with RCS Operational Leakage Technical Specification prior to RR Loop isolation, Technical Specification 3.4.4 Condition A.

6. Stator Water Cooling Pump trip, BOP operator starts standby pump and restores stator water cooling prior to a turbine trip.

7. EHC Pressure Transducer Failure – non-operating pressure regulator takes control. This results in slowly decreasing reactor pressure. ATC inserts a scram and the BOP operator closes the MSIVs prior to reactor pressure lowering to less than 900 psig IAW 3-AOI-47-2.

8. ATWS exists on the scram the crew will enter EOI-1, EOI-2 and EOI-C-5. Crew will insert control rods, control reactor pressure on SRVs, initiate SLC.

9. RCIC steam supply valve will not auto-open on initiation signal, level will degrade until RCIC is manually started. Once started RCIC will maintain level above TAF.

10. CRD Controller will fail low ATC takes manual control of controller and restores CRD parameters

Terminate the scenario when the following conditions are satisfied or upon request of Lead Examiner:

Control Rods are being inserted

Reactor Level is being maintained

Reactor Pressure Controlled on SRVs

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER: 4

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4	Abnormal Events:	List (1-3)
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1	EOI Contingencies used:	List (0-3)
75	Validation Time (minutes)	
3	Crew Critical Tasks:	(2-5)
YES	Technical Specifications Exercised (Yes/No)	

Scenario Tasks

TASK NUMBER	K/A	RO	SRO
Transfer 4KV Unit Board			
RO U-57A-NO-1 SRO S-57A-NO-4	262001A4.05	3.3	3.3
Lower Power with Recirc Flow			
RO U-068-NO-17 SRO S-000-NO-138	2.1.23	4.3	4.4
Stator Water Cooling Pump Trip			
RO U-35A-AL-2 SRO S-070-AB-1	245000A4.03	2.7	2.8
Turbine Generator Voltage Regulator Failure			
RO U-47-AL-20 SRO S-57A-AB-4	262001A2.09	3.1	3.4
RR Pump Seal Failure			
RO U-068-AL-9 SRO S-068-AB-1	203000A4.02	4.1	4.1
EHC Pressure Transducer Failure			
RO U-047-AB-2 SRO S-047-AB-2	241000A2.03	4.1	4.2
ATWS			
RO U-000-EM-35 SRO S-000-EM-1 SRO S-000-EM-2 SRO S-000-EM-3	295015AA2.01	4.1	4.3

Procedures Used/Referenced:

Procedure Number	Procedure Title	Procedure Revision
0-OI-57A	Switchyard and 4160V AC Electrical System	Rev.141
3-GOI-100-12	Power Maneuvering	Rev. 35
3-OI-68	Reactor Recirculation System	Rev. 80
3-ARP-9-3C	Alarm Response Procedure	Rev. 26
3-TSR	BFN-UNIT 3 Tech Spec 3.5-1	Amend No. 244 December 1, 2003
3-ARP-9-7A	Alarm Response Procedure	Rev. 22
3-ARP-9-8A	Alarm Response Procedure	Rev. 34
3-ARP-9-4B	Alarm Response Procedure	Rev. 42
3-AOI-64-1	Drywell Pressure and/or Temperature High, or Excessive Leakage Into Drywell	Rev. 3
3-AOI-68-1A	Recirc Pump Trip/Core Flow Decrease OPRMs Operable	Rev. 6
3-AOI-47-2	Turbine EHC Control System Malfunctions	Rev. 6
3-EOI-1	RPV CONTROL FLOWCHART	Rev. 8
3-EOI APPENDIX-1D	INSERT CONTROL RODS USING REACTOR MANUAL CONTROL SYSTEM	Rev. 2
3-EOI-2	PRIMARY CONTAINMENT CONTROL FLOWCHART	Rev.7
3-EOI APPENDIX-5C	INJECTION SYSTEM LINEUP RCIC	Rev. 3
3-EOI APPENDIX-1F	MANUAL SCRAM	Rev. 2
3-EOI APPENDIX-2	DEFEATING ARI LOGIC TRIPS	Rev. 4
3-EOI-C-5	LEVEL-POWER CONTROL FLOWCHART	Rev. 9

Simulator Instructor - **IC-204**

#HPCI tagout
bat nrchpcito

#Tech Spec call SRO Core Spray System #1
ior ypovfcv7525 (e1 0) fail_now
ior xa553c[27] (e1 0) crywolf

#B stator water pump trip
irf eg02 (e5 0) off
ior ypobkrscwpa (e5 0) fail_ccoil
ior zdihs3535a[2] (e5 0) stop
ior zlohs3535a[1] (e5 0) off

#Turbine Generator Voltage Regulator failure
imf eg03 (e10 0)

#B Recirc pump seal failures
imf th12b (e15 0)
imf th10b (e15 0) 100
imf th11b (e15 180) 100 60 0

#B EHC Pressure transducer failure
bat atws70
ior zdihs0116[1] (e20 0) select
ior zdihs47204[1] (e20 0) null
ior zlohs0116[1] off
ior zlohs47204[1] on
imf tc10b (e20 0) 86 1200 79

#RCIC steam supply valve fails to auto open
imf rc08
trg 25 = bat sdv
trg 26 = bat atws-1
trg 27 = bat app01f
trg 28 = bat app02
trg 29 = bat app08ae

#After Scram manually insert under DI Overide
#3-FIC-85-11 0-100(L)

Scenario 4

		DESCRIPTION/ACTION
Simulator Setup	manual	Reset to IC 204
Simulator Setup	**Load Batch**	bat nrc1108-4
Simulator Setup	manual	Verify file loaded
Simulator Setup	Manual	Hang clearance on HPCI

Simulator Event Guide:

Event 1 Normal: Transfer 4KV UB-3A from USST 3B to Start Bus 1A

	SRO	Directs Transfer 4KV UB-3A from USST 3B to Start Bus 1A per 0-OI-57A section 8.15.1
	BOP	**8.15.1 Transfer 4kV Unit Board 3A from USST to Start Bus** [1] **REVIEW** all Precautions and Limitations in Section 3.0.
		CAUTIONS 1) This board transfer can cause a power interruption causing a loss of Computer Rooms and Communication Battery Board ACU, Computer UPS ACU, and Communication rooms ACU. 2) Capacitor bank fuses are subject to clearing when Unit Boards are supplied from the 161 source and large pumps are started. Unit Supervisors should evaluate placing the Capacitor Banks in Manual prior to starting Condensate, CBP, RHR, CS or CCW pumps. 3) If 4kV Unit Board 3A is fed from the Alternate Power Supply (Start Bus), then Auto transfer must be blocked for: • 4kV UNIT BD 1A, 1B, 1C, 2A, 2B, and 2C. (Ref. 3-45E721 OPL) • 4kV COM BD A and B. (3-45E721 OPL) 4) If either 4kV UNIT BD 1A, 1B, 2A or 2B is aligned to a Start Bus, prior to aligning UNIT BD 3A to the Start Bus, check Technical Specifications 3.8.1.a and 3.8.2.a to determine operability of qualified AC circuits between the offsite transmission network and the onsite Class 1E Electrical Power Distribution System.
		NOTES 1) All procedural steps are performed from Control Room Panel 3-9-8, unless specified. 2) This procedure section contains actions ensure electrical load restrictions are not exceeded when 4kV UNIT BD 3A is placed on Alternate Supply (Start Bus).
		[2] Ensure the 4kV Start Busses are aligned Normal. [2.1] On Panel 9-23-2, **VERIFY** 4kV Start Bus 1A ALT FDR BKR 1518 OPEN. [2.2] On Panel 9-23-2, **VERIFY** 4kV Start Bus 1B ALT FDR BKR 1414 OPEN.
		[3] **RE-ALIGN** 4kV Auto Transfers to met Load Restrictions [3.1] On Panel 1-9-8, **PLACE** 1-XS-57-4, 4kV UNIT BD 1A MAN/AUTO SELECT switch to MAN. [3.2] On Panel 1-9-8, **PLACE** 1-XS-57-7, 4kV UNIT BD 1B MAN/AUTO SELECT switch to MAN.

Simulator Event Guide:

Event 1 Normal: Transfer 4KV UB-3A from USST 3B to Start Bus 1A

		[3.3] On Panel 1-9-8, **PLACE** 1-XS-57-10, 4kV UNIT BD 1C MAN/AUTO SELECT switch to MAN. [3.4] On Panel 3-9-8, **PLACE** 3-XS-57-4, 4kV UNIT BD 2A MAN/AUTO SELECT switch to MAN. [3.5] On Panel 3-9-8, **PLACE** 3-XS-57-7, 4kV UNIT BD 2B MAN/AUTO SELECT switch to MAN. [3.6] On Panel 3-9-8, **PLACE** 3-XS-57-10, 4kV UNIT BD 2C MAN/AUTO SELECT switch to MAN. [3.7] On Panel 0-9-23-3, **PLACE** 0-43-203-A, 4kV COM BD A MAN/AUTO SELECT switch to MAN. [3.8] On Panel 0-9-23-4, **PLACE** 0-43-203-B, 4kV COM BD B MAN/AUTO SELECT switch to MAN.
DRIVER	DRIVER	When requested to **RE-ALIGN** 4kV UNIT BD Auto Transfer Scheme. Report switches for 4 KV **Unit** Boards 1A, 1B, 1C, 2A, 2B, 2C, AND **Common** Boards A and B have been place in **MANUAL** .
		[4] **TRANSFER** 4kv UNIT BD 3A to the ALT FD [4.1] **PLACE** 3-XS-57-4, 4kV UNIT BD 3A MAN/AUTO SELECT switch to MAN. [4.2] **PLACE** 3-XS-202-1, 4kV BD/BUS/XFMR VOLTAGE SELECT switch to START BUS 1A. [4.3] **CHECK** START BUS 1A Voltage on 3-EI-57-28 is between 3950 and 4400 Volts. [4.4] **PLACE** and **HOLD** 3-HS-57-5, 4kV UNIT BD 3A ALT FDR BKR 1432 switch to CLOSE. [4.5] **PLACE** 3-HS-57-3, 4kV UNIT BD 3A NORM FDR BKR 1312 switch to TRIP. [4.6] **CHECK CLOSED** the 4kV UNIT BD 3A, ALT FDR BREAKER 1432.

Simulator Event Guide:

Event 1 Normal: Transfer 4KV UB-3A from USST 3B to Start Bus 1A (continued)

		[4.7] **CHECK OPEN** the 4kV UNIT BD 3A, NORM FDR BREAKER 1312. [4.8] **RELEASE** BKRs 1432 and 1312 control switches. [4.9] **PLACE** 3-XS-202-1, 4kV BD/BUS/XFMR VOLTAGE SELECT SWITCH TO UNIT BD 3A. [4.10] **CHECK** 4kV UNIT BD 3A voltage is between 3950 and 4400 Volts. [4.11] **VERIFY LOCALLY** 4kV BKR 1432 closing spring target indicates charged and the amber breaker spring charged light is on. [4.12] As directed by the Unit Supervisor, **PLACE** a Caution Order on the Condensate, CBP, CS, RHR or CCW Pump stating, "Evaluate the need to place CAP Banks in Manual prior to starting Pump." [4.13] **RETURN** the Computer Rooms, Communication Battery Board, Computer UPS, and Communication rooms ACUs to service per 0-OI-31.
DRIVER	DRIVER	When requested, acknowledge that a Caution Order will need to be placed on the Condensate, CBP, CS, RHR or CCW Pump stating, "Evaluate the need to place CAP Banks in Manual prior to starting Pump."
DRIVER	DRIVER	When requested, acknowledge that the Computer Rooms, Communication Battery Board, Computer UPS, and Communication rooms ACUs are to be returned to service per 0-OI-31. There are no simulator actions required to complete this step.

Simulator Event Guide:

Event 2 Reactivity: Lower Reactor Power with Recirc Flow

	SRO	Notify ODS of power decrease
		Directs Power Reduction using Recirc Flow per 3-GOI-100-12: [9] **REDUCE** reactor power by a combination of control rod insertions and core flow changes, as recommended by Reactor Engineer. **REFER TO** 3-SR-3.1.3.5(A) and 3-OI-68. (N/A if entering 3-GOI-100-12 to recover from Recirc Pump Trip)
	ATC	**3-OI-68 Precaution and Limitations** **3.5.3 Dual Pump Operation** D. Individual pump speeds should be mismatched by ~60 RPM during dual pump operation between 1200 and 1300 RPM to minimize harmonic vibration (this requirement may be waived for short time periods for testing or maintenance).
	ATC	Lowers Power w/Recirc using 3-OI-68, section 6.2
		[1] **IF** desired to control Recirc Pumps 3A and/or 3B speed with Recirc Individual Control, **THEN** **PERFORM** the following; (Otherwise N/A) • Raise Recirc Pump 3A using, RAISE SLOW (MEDIUM), 3-HS-96-15A(15B). (Otherwise N/A) • Lower Recirc Pump 3A using SLOW (MEDIUM) (FAST), 3-HS-96-17A(17B)(17C). (Otherwise N/A) AND/OR • Raise Recirc Pump 3B using, RAISE SLOW (MEDIUM), 3-HS-96-16A(16B). (Otherwise N/A) • Lower Recirc Pump 3B using SLOW (MEDIUM) (FAST), 3-HS-96-18A(18B)(18C). (Otherwise N/A)
		[2] **WHEN** desired to control Recirc Pumps 3A and/or 3B speed with the RECIRC MASTER CONTROL, **THEN** **ADJUST** Recirc Pump speed 3A & 3B using the following push buttons as required: **RAISE** SLOW, 3-HS-96-31 **RAISE** MEDIUM, 3-HS-96-32 **LOWER** SLOW, 3-HS-96-33 **LOWER** MEDIUM, 3-HS-96-34 **LOWER** FAST, 3-HS-96-35

Simulator Event Guide:

Event 3: Core Spray Loop 1 Inoperable failed FCV-75-25

NRC	NRC	When satisfied with Reactivity manipulation, move on to Core Spray Loop 1 Inoperable failed FCV-75-25
DRIVER	DRIVER	Insert TRIGGER 1 to cause a loss of power to 3-FCV-~~25-75~~ 75-25, CORE SPRAY SYS I INBD INJECT VALVE.
DRIVER	DRIVER	As Reactor Building AUO, call the control room and report that you discovered breaker 480V RMOV Bd 3A, Comp 14B, for 3-FCV-~~25-75~~ 75-25, CORE SPRAY SYS I INBD INJECT VALVE, **"tripped AND will NOT reset"**.
	BOP	Relays field report to US. And recognizes 3-FCV-~~25-75~~ 75-25, CORE SPRAY SYS I INBD INJECT VALVE, does not have indication.
	SRO	References Tech Spec 3.5.1 and enters Conditions A and D.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

NOTE
LCO 3.0.4.b is not applicable to HPCI.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. OR One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days
D. HPCI System inoperable. AND Condition A entered.	D.1 Restore HPCI System to OPERABLE status. OR D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours

BM

Simulator Event Guide:

Event 4 Component: Turbine Generator Voltage Regulator Failure

DRIVER	DRIVER	Insert TRIGGER 10 to cause the Turbine Generator Voltage Regulator to fail high in automatic.
	NRC	**This failure will take approximately 9 minutes before the first annunciator is received.**
	BOP	Reports the following alarms: GENERATOR EXCTR PWR RECTIFIER TEMP HIGH GEN VOLTS PER CYCLE HIGH GEN HYDROGEN SYSTEM ABNORMAL
	BOP	**GEN VOLTS PER CYCLE HIGH, 3-9-8A window 9** A. **VERIFY** VOLTAGE REG TRANSFER switch in MANUAL. B. At Panel 3-9-8, **ADJUST** EXCITER FIELD VOLTAGE 70P MANUAL (3-HS-57-25) to maintain the following: 1. GENERATOR VOLTS, 3-EI-57-39, between 20,900V and 23,100V. 2. GENERATOR MVARS, 3-EI-57-51, within the generator capability curve. **REFER TO** 3-OI-47, Illustration 6. C. **IF** Turbine/Generator trips and power is less than ~30%, **THEN** **VERIFY** Bypass Valves Controlling Reactor Pressure. **REFER TO** 3-AOI-47-1. D. **IF** Reactor scrams, **THEN** **REFER TO** 3-AOI-100-1.
	BOP	Takes Voltage Regulator to Manual and adjusts MVARs to comply with 0-GOI-300-4, Switchyard Manual P&L I: A 300 MVAR maximum outgoing limit applies to all units for both the 500kV and 161kV offsite power source qualification. If the outgoing MVAR limit is exceeded for a unit and is not corrected within 15 minutes, the TOp must immediately inform BFN that both offsite power sources are disqualified for the unit that is exceeding the limit. Offsite power qualification is not impaired for the unit(s) whose outgoing MVARs are under the limit.
	Crew	Make notifications. Must notify Load Dispatch when voltage regulator not in Auto
Driver	Driver	Acknowledge notifications
Driver	Driver	At NRC direction initiate trigger 15 for Reactor Recirc Pump B Seal Failure

Simulator Event Guide:

Event 5 Component: LOCA - Recirculation Pump B Inboard and Outboard seal failure

DRIVER	DRIVER	At NRC direction, insert **TRIGGER 15** to cause the B Recirc pump seals to fail.
	ATC	Reports failure of the #1 Reactor Recirc Pump B Seal
		RECIRC PUMP B NO. 1 SEAL LEAKAGE ABN, 3-9-4B Window 25: A. **DETERMINE** initiating cause by comparing No. 1 and 2 seal cavity pressure indicators on Panel 3-9-4 or ICS. • Plugging of No. 1 RO - No. 2 seal cavity pressure indicator drops toward zero. • Plugging of No. 2 RO - No. 2 seal pressure approaches no. 1 seal pressure. • Failure of No. 1 seal - No. 2 seal pressure is greater than 50% of the pressure of No. 1. • Failure of No. 2 seal - no. 2 seal pressure is less than 50% of the No. 1 seal.
		NOTE 1) Possible indications of dual seal failure include: • Window 18 on this panel alarming in conjunction with this window. • Rising drywell pressure and/or temperature. • Increased leakage into the drywell sump. • Increased vibration of the recirc pump.
	ATC	Identifies that the #2 seal is also failed/failing.
		D. **IF** dual seal failure is indicated, **THEN** 1. SHUTDOWN Recirc Pump 3B by DEPRESSING RECIRC DRIVE 3B SHUTDOWN, 3-HS-96-20. 2. **VERIFY TRIPPED**, RECIRC DRIVE 3B NORMAL FEEDER, 3-HS-57-14.

Simulator Event Guide:

Event 5 Component: LOCA - Recirculation Pump B Inboard and Outboard seal failure

		3. **VERIFY TRIPPED**, RECIRC DRIVE 3B ALTERNATE FEEDER, 3-HS-57-12. 4. **CLOSE** Recirculation Pump 3B suction valve. 5. **CLOSE** Recirculation Pump 3B discharge valve. 6. **REFER TO** 3-AOI-68-1A or 3-AOI-68-1B AND 3-OI-68. 7. **DISPATCH** personnel to SECURE Recirculation Pump 3B seal Water
	SRO	Enters: 3-AOI-68-1A, Recirc Pump Trip/Core Flow Decrease OPRMs Operable, 3-AOI-64-1, Drywell Pressure and/or Temperature High, or Excessive Leakage Into Drywell.
		3-AOI-68-1A, Recirc Pump Trip/Core Flow Decrease OPRMs Operable [2] **IF** a single Recirc Pump tripped, **THEN** **CLOSE** tripped Recirc Pump discharge valve. [3] **IF** Region I or II of the Power to Flow Map is entered, **THEN** (Otherwise N/A) **IMMEDIATELY** take actions to INSERT control rods to less than 95.2% loadline. **REFER TO** 0-TI-464, Reactivity Control Plan Development and Implementation. [4] **RAISE** core flow to greater than 45%. **REFER TO** 3-OI-68. [5] **INSERT** control rods to exit regions if not already exited. Refer to 0-TI-464, Reactivity Control Plan Development and Implementation. **NOTE** The remaining subsequent action steps apply to a single Reactor Recirc Pump trip. [6] **MAINTAIN** operating Recirc pump flow less than 46,600 gpm. **REFER** to 3-OI-68. [7] **WHEN** plant conditions allow, **THEN**, (Otherwise N/A) **MAINTAIN** operating jet pump loop flow greater than 41 x 106 lbm/hr (3-FI-68-46 or 3-FI-68-48).

Simulator Event Guide:

Event 5 Component: LOCA - Recirculation Pump B Inboard and Outboard seal failure

	ATC	Inserts Rods per Emergency shove sheet to get below 95% loadline. 1. Rod 30-31 48 to 00 2. Rod 14-15 12 to 00 3. Rod 14-47 12 to 00 4. Rod 46-47 12 to 00 5. Rod 46-15 12 to 00 6. Rod 14-31 12 to 00 7. Rod 30-47 12 to 00 8. Rod 46-31 12 to 00 9. Rod 30-15 12 to 00 10. Rod 22-31 48 to 00 11. Rod 30-39 48 to 00. 12. Rod 38-31 48 to 00. 13. Rod 30-23 48 to 00
	ATC	When less than 95% load line, raises core flow to greater than 45%.
	SRO	AOI-64-1 Directs BOP to Vent the Drywell
	BOP	**3-AOI-64-1Drywell Pressure and/or Temperature High, or Excessive Leakage Into Drywell** [3] **VENT** Drywell as follows: [3.1] **CLOSE** SUPPR CHBR INBD ISOLATION VLV 3-FCV-64-34 (Panel 3-9-3). [3.2] **VERIFY OPEN**, DRYWELL INBD ISOLATION VLV, 3-FCV-64-31 (Panel 3-9-3). [3.3] **VERIFY** 3-FIC-84-20 is in AUTO and SET at 100 scfm (Panel 3-9-55). [3.4] **VERIFY** Running, required Standby Gas Treatment Fan(s) SGTS Train(s) A, B, C (Panel 3-9-25). [3.5] **IF** required, **THEN** **REQUEST** Unit 1 Operator to START Standby Gas Treatment Fan(s) SGTS Train(s) A, B. (Otherwise **N/A)**
DRIVER	DRIVER	When requested to start a standby gas fan remote function pc01a or b or c

Simulator Event Guide:

Event 5 Component: LOCA - Recirculation Pump B Inboard and Outboard seal failure

	SRO	Evaluates Tech Spec 3.4.1 and enters Condition A
		3.4.1 Recirculation Loops Operating LCO 3.4.1Two recirculation loops with matched flows shall be in operation OR One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable: a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR; b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR; c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation; APPLICABILITY: MODES 1 and 2.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	24 hours
B. Required Action and associated Completion Time of Condition A not met. OR No recirculation loops in operation.	B.1 Be in MODE 3.	12 hours

Simulator Event Guide:

Event 6 Component: Stator Water Cooling Pump Trip

DRIVER	DRIVER	At NRC direction Insert TRIGGER 5 to cause a trip of the B stator water cooling pump.
	BOP	Responds to annunciator 3-9-7A window 22, GEN STATOR COOLANT SYS ABNORMAL: A. IF while performing the action of this ARP 3-XA-55-9-8A Window 1 alarms THEN, 1. VERIFY all available Stator Cooling Water Pumps running. 2. Attempt to RESET alarm 3. IF alarm fails to reset, AND reactor power is above turbine bypass valve capability THEN SCRAM the Reactor B. VERIFY a stator cooling water pump is running and CHECK stator temperature recorder, 3-TR-57-59, Panel 3-9-8.
	BOP	Responds to annunciator ARP 3-XA-55-9-8A Window 1, TURBINE TRIP TIMER INITIATED: Automatic Action: Main Turbine Trip 60 seconds after the alarm is received (Total of 70 seconds) Operator Action: A. **CHECK** Stator Cooling Water Flow and Temperature and Generator Stator temperatures using ICS. B. **VERIFY** all available Stator Cooling Water Pumps running. C. **IF** all of the following conditions exist • Alarm fails to reset, • Low Stator Cooling Water flow OR High Generator or Stator • Cooling temperatures are observed on ICS, • Reactor Power is above turbine bypass valve capability, **THEN ,** **SCRAM** the reactor.
	BOP	Operator starts the standby Stator Water Cooling Pump and restores Stator Water Cooling .
DRIVER	DRIVER	When/If dispatched to investigate the B Stator Water Cooling Pump, wait 5 minutes and report unable to determine cause of trip

Simulator Event Guide:

Event 7 Component: EHC Pressure Transducer Failure

DRIVER	DRIVER	At NRC direction, insert **TRIGGER 20** to cause the B EHC Pressure transducer to fail. **Verify tc10b initial setpoint prior to inserting trigger**. One Main Steam line will fail to isolate.
	ATC/BOP	Responds to annunciator, 3-9-7B Window 4, HEADER PRESS SETPOINT OUT OF RANGE: A. **CHECK** header pressure setpoint >960 on EHC SETPOINT, 3-PI-47-204. B. **STOP** any power ascensions until alarm can be reset (power reductions may be performed). C. **IF** it is desired to be in Reactor Pressure Control, **THEN TRANSFER** control to Reactor Pressure Control. REFER TO Transferring EHC Pressure Control from Header Pressure To Reactor Pressure section in 3-OI-47. (N/A if Step D will be performed). D. **IF** Reactor Pressure Control is **NOT** available, **THEN LOWER** setpoint to below 960 psig by using LOWER pushbutton, 3-HS-47-162A (N/A if Step C was performed). E. **VERIFY** alarm will reset. F. **RECORD** events in narrative log.
	ATC	Recognizes lowering Reactor Pressure and generator megawatts.
	SRO	Directs entry into 3-AOI-47-2.
		3-AOI-47-2 Turbine EHC Control System Malfunctions [1] **IF** Reactor Pressure lowers to or below 900 psig, **THEN MANUALLY SCRAM** the Reactor and **CLOSE** the MSIVs.
	SRO	Directs manual scram, closing of the MSIV's, and entry into 3-AOI-100-1.
	ATC	Manually scrams the reactor.
DRIVER	DRIVER	After Scram manually insert under **DI Override 3-FIC-85-11 0-100(L)** and insert **TRIGGER 25** to enter bat SDV
	BOP	Recognizes one main steam line D failed to isolate. Closes the MSIV's.
	SRO	Enter 3-EOI-1, "RPV Control".
	SRO	EOI-1 (Reactor Pressure)
		Monitor and Control Reactor Pressure
		IF Drywell Pressure Above 2.4 psig? - NO
		IF Emergency Depressurization is Anticipated and the Reactor will remain subcritical without boron under all conditions **THEN** Rapidly depressurize the RPV with the Main Turbine Bypass Valves irrespective of cooldown rate ?- NO
		IF Emergency Depressurization is required THEN exit RC/P and enter C2 Emergency Depressurization? - NO
		IF RPV water level cannot be determined? - NO
		Is any MSRV Cycling? - YES
		IF Steam cooling is required? - NO
		IF Suppression Pool level and temperature cannot be maintained in the safe area of Curve 3? - NO

Simulator Event Guide:

Event 8 Major: ATWS, without MSIVs

	SRO	3-EOI-1 (Reactor Pressure)
		IF Suppression Pool level cannot be maintained in the safe area of Curve 4? - NO
		IF Drywell Control air becomes unavailable? – NO. **THEN** crosstie CAD to Drywell Control Air, Appendix 8G.
		IF Boron injection is required? - NO
	SRO	Direct a Pressure Band of 800 to 1000 psig, Appendix 11A.
	ATC/BOP	Maintain directed pressure band, IAW Appendix 11A.
	SRO	EOI-1 RPV Pressure – Augment RPV Pressure control as necessary with one or more of the following depressurization systems: HPCI Appendix 11C, RCIC Appendix 11B, RFPTs on minimum flow Appendix 11F, Main Steam System Drains Appendix 11D, Steam Seals Appendix 11G, SJAEs Appendix 11G, Off Gas Preheater Appendix 11G, RWCU Appendix 11E.
	ATC/BOP	Pressure Control IAW Appendix11A, RPV Pressure Control SRVs
		1. IF Drywell Control Air is **NOT** available, **THEN**: **EXECUTE** EOI Appendix 8G, CROSSTIE CAD TO DRYWELL CONTROL AIR, **CONCURRENTLY** with this procedure.
		2. IF Suppression Pool level is at or below 5.5 ft, **THEN:** **CLOSE** MSRVs and **CONTROL** RPV pressure using other options.
		3. **OPEN** MSRVs; using the following sequence to control RPV pressure, as directed by SRO:
		a. 3-PCV-1-179 MN STM LINE A RELIEF VALVE
		b. 3-PCV-1-180 MN STM LINE D RELIEF VALVE.
		c. 3-PCV-1-4 MN STM LINE A RELIEF VALVE
		d. 3-PCV-1-31 MN STM LINE C RELIEF VALVE
		e. 3-PCV-1-23 MN STM LINE B RELIEF VALVE
		f. 3-PCV-1-42 MN STM LINE D RELIEF VALVE
		g. 3-PCV-1-30 MN STM LINE C RELIEF VALVE

Simulator Event Guide:

Event 8 Major: ATWS, without MSIVs

	ATC/BOP	Pressure Control IAW Appendix11A, RPV Pressure Control SRVs (continued)
		h. 3-PCV-1-19 MN STM LINE B RELIEF VALVE.
		i. 3-PCV-1-5 MN STM LINE A RELIEF VALVE.
		j. 3-PCV-1-41 MN STM LINE D RELIEF VALVE
		k. 3-PCV-1-22 MN STM LINE B RELIEF VALVE
		l. 3-PCV-1-18 MN STM LINE B RELIEF VALVE
		m. 3-PCV-1-34 MN STM LINE C RELIEF VALVE
	SRO	EOI-1 (Reactor Level)
		Monitor and Control Reactor Level.
		Verify as required PCIS isolations group (1,2 and 3), ECCS and RCIC, Directs group 2 and 3 verified.
	ATC/BOP	Verifies Group 2 and 3 isolation.
	SRO	IF it has not been determined that the reactor will remain subcritical, **THEN** Exit RC/L; **ENTER** C5 Level / Power Control.
		If Emergency Depressurization is required? - NO
		RPV Water level cannot be determined? – NO
		The reactor will remain subcritical without Boron under all conditions? - NO
		PC water level cannot be maintained below 105 feet **OR** Suppression Chamber pressure cannot be maintained below 55 psig? - NO
CT#3	SRO	Directs ADS Inhibited.
CT#3	ATC/BOP	Inhibits ADS.
	SRO	Is any Main Steam Line Open?- NO

Simulator Event Guide:

Event 8 Major: ATWS, without MSIVs

	SRO	C5 Level / Power Control
		IF Suppression Pool Temperature is above 110°F **AND** Reactor Power is above 5% **AND** a MSRV is open or cycling **OR** drywell pressure is above 2.4 psig **AND** RPV water level is above -162 inches? – NO
		Is Reactor Power above 5% ?- YES
		Stop and Prevent all injection into the RPV except from RCIC, CRD, and SLC (Appendix 4). **WHEN** RPV Level drops below -50 inches; **THEN** Continue:
	SRO	Direct Terminate and Prevent IAW Appendix 4.
		IF Suppression Pool Temperature is above 110°F **AND** Reactor Power is above 5% **AND** a MSRV is open or cycling **OR** drywell pressure is above 2.4 psig **AND** RPV water level is above -162 inches – IF YES?
		Stop and Prevent all injection into the RPV except from RCIC, CRD, and SLC; irrespective of any consequent reactor power or reactor water level oscillations.
		WHEN RPV Level drops below -50 inches and any of the following exist: • Power drops below 5% **OR** • All MSRVs remain closed and DW pressure remains below 2.4 psig **OR** • Water level reaches -162 inches **THEN** Continue:
	ATC/BOP	Terminate and Prevent IAW Appendix 4
	BOP/ATC	**Appendix 4** 1. **PREVENT** injection from HPCI by performing the following: a. IF HPCI Turbine is **NOT** at zero speed, **THEN PRESS** and **HOLD** 3-HS-73-18A, HPCI TURBINE TRIP push-button. b. **WHEN** HPCI Turbine is at zero speed, **THEN PLACE** 3-HS-73-47A, HPCI AUXILIARY OIL PUMP control switch in PULL TO LOCK and **RELEASE** 3-HS-73-18A, HPCI TURBINE TRIP push-button. 3. **PREVENT** injection from CORE SPRAY following an initiation signal by **PLACING** ALL Core Spray pump control switches in STOP.

Simulator Event Guide:

Event 8 Major: ATWS, without MSIVs

		4. **PREVENT** injection from LPCI SYSTEM I by performing the following: **NOTE** Injection may be prevented by performing EITHER step 4.a or step 4.b. a. Following automatic pump start, PLACE RHR SYSTEM I pump control switches in STOP. **OR** b. BEFORE RPV pressure drops below 450 psig, 1) PLACE 3-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in BYPASS. **AND** 2) VERIFY CLOSED 3-FCV-74-52, RHR SYS I LPCI OUTBD INJECT VALVE.
		5. **PREVENT** injection from LPCI SYSTEM II by performing the following:
		NOTE Injection may be prevented by performing EITHER step 5.a or step 5.b. a. Following automatic pump start, PLACE RHR SYSTEM II pump control switches in STOP. **OR** b. BEFORE RPV pressure drops below 450 psig, 1) PLACE 3-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYPASS SEL in BYPASS. **AND** 2) VERIFY CLOSED 3-FCV-74-66, RHR SYS II LPCI OUTBD INJECT VALVE
		6. **PREVENT** injection from CONDENSATE and FEEDWATER by performing the following: a. **IF** Immediate injection termination from a reactor feedwater pump is required, **THEN PERFORM** step 6.d for the desired pump.

Simulator Event Guide:

Event 8 Major: ATWS, without MSIVs

		Appendix 4 (continued) c. **CLOSE** the following valves BEFORE RPV pressure drops below 500 psig: • 3-FCV-3-19, RFP 2A DISCHARGE VALVE • 3-FCV-3-12, RFP 2B DISCHARGE VALVE • 3-FCV-3-5, RFP 2C DISCHARGE VALVE • 3-LCV-3-53, RFW START-UP LEVEL CONTROL
		d. **TRIP** RFPTs as necessary to prevent injection by **DEPRESSING** the following push-buttons: • 3-HS-3-125A, RFPT 3A TRIP • 3-HS-3-151A, RFPT 3B TRIP • 3-HS-3-176A, RFPT 3C TRIP.
CT#2	SRO	**WHEN** RPV Level drops below -50 inches **THEN** Continue: **OR** **WHEN** RPV Level has dropped below -50 inches **AND** Power is below 5% **OR** Reactor Level reaches -162 inches, **THEN** Continue: Directs a Level Band with RCIC.

Simulator Event Guide:

Event 8 Major: ATWS, without MSIVs (continued)

	SRO	EOI-1 (Power Control)
		Monitor and Control Reactor Power.
		Will the reactor will remain sub subcritical without boron under all conditions? - NO
		If the reactor subcritical and No boron has been injected?- NO
		Verify Reactor Mode Switch in Shutdown.
		Initiate ARI.
	ATC	Initiates ARI.
	SRO	Verify Recirc Runback (pump speed 480 rpm).
	ATC	Verifies Recirc Runback.
	SRO	Is Power above 5%? - YES
		Directs tripping Recirc Pumps.
	ATC	Trips Recirc Pumps.
CT#1	SRO	Before Suppression Pool temperature rises to 110°F, continue:
		Insert Control Rods Using **one or more** of the following methods: • Appendix 1F • Appendix 1D
	DRIVER	**WHEN** directed to perform Appendix 1F and Appendix 2, wait 4 minutes and insert **TRIGGER 27** and **TRIGGER 28 THEN** report appendix 2 complete and field action for appendix 1F complete. **WHEN** the Scram has been reset **THEN** insert **TRIGGER 26** to enter bat ATWS-1
CT#1	ATC	Inserts Control Rods, IAW Appendix 1D and 1F.

Simulator Event Guide:

Event 8 Major: ATWS, without MSIVs

	ATC	Insert Control Rods, IAW Appendix 1F.
		2. **WHEN** RPS Logic has been defeated, **THEN RESET** Reactor Scram. 3. **VERIFY OPEN** Scram Discharge Volume vent and drain valves. 4. **DRAIN** SDV UNTIL the following annunciators clear: • WEST CRD DISCH VOL WTR LVL HIGH HALF SCRAM (Panel 3-9-4, 3-XA-55-4A, Window 1) • EAST CRD DISCH VOL WTR LVL HIGH HALF SCRAM (Panel 3-9-4, 3-XA-55-4A, Window 29). 5. **DISPATCH** personnel to **VERIFY OPEN** 3-SHV-085-0586, CHARGING WATER SHUTOFF. 6. **WHEN** CRD Accumulators are recharged, **THEN INITIATE** manual Reactor Scram and ARI. 7. **CONTINUE** to perform Steps 1 through 6, **UNTIL ANY** of the following exists: • ALL control rods are fully inserted, **OR** • NO inward movement of control rods is observed, **OR** • SRO directs otherwise.

Simulator Event Guide:

Event 8 Major: ATWS, without MSIVs

<table>
<tr><td>CT#1</td><td>BOP/ATC</td><td>Initiate SLC IAW Appendix 3A</td></tr>
<tr><td></td><td></td><td>1. UNLOCK and PLACE 3-HS-63-6A, SLC PUMP 2A/2B, control switch in START-A or START-B position.

2. CHECK SLC System for injection by observing the following:
• Selected pump starts, as indicated by red light illuminated above pump control switch.
• Squib valves fire, as indicated by SQUIB VALVE A and B CONTINUITY blue lights extinguished.
• SLC SQUIB VALVE CONTINUITY LOST Annunciator in alarm on Panel 3-9-5 (3-XA-55-5B, Window 20).
• 3-PI-63-7A, SLC PUMP DISCH PRESS, indicates above RPV pressure.
• System flow, as indicated by 3-IL-63-11, SLC FLOW, red light illuminated on Panel 3-9-5.
• SLC INJECTION FLOW TO REACTOR Annunciator in alarm on Panel 3-9-5 (3-XA-55-5B, Window 14).</td></tr>
<tr><td></td><td></td><td>3. IF Proper system operation CANNOT be verified, THEN RETURN to Step 1 and START other SLC pump.

4. VERIFY RWCU isolation by observing the following:
• RWCU Pumps 2A and 2B tripped.
• 3-FCV-69-1, RWCU INBD SUCT ISOLATION VALVE closed.
• 3-FCV-69-2, RWCU OUTBD SUCT ISOLATION VALVE closed.
• 3-FCV-69-12, RWCU RETURN ISOLATION VALVE closed.

5. VERIFY ADS inhibited.

6. MONITOR reactor power for downward trend.

7. MONITOR 3-LI-63-1A, SLC STORAGE TANK LEVEL, and CHECK that level is dropping approximately 1% per minute.</td></tr>
</table>

Simulator Event Guide:

Event 8 Major: ATWS, without MSIVs

	SRO	**ENTER** 3-EOI-2, "Primary Containment Control"
		EOI-2 (Drywell Temperature)
	SRO	Monitor and Control DW Temp Below 160°F using available DW Cooling.
		Can Drywell Temp Be Maintained Below 160°F? - YES
	SRO	EOI-2 (Primary Containment Hydrogen)
		If PCIS Group 6 isolation exists? – YES **THEN DIRECTS**: 1. Place analyzer isolation bypass keylock switches to bypass. 2. Select Drywell or suppression chamber and momentarily pull out select switch handle to start sample pumps.
	BOP	1. Place analyzer isolation bypass keylock switches to bypass. 2. Select Drywell or suppression chamber and momentarily pull out select switch handle to start sample pumps.
	SRO	EOI-2 (Suppression Pool Temperature)
		Monitor and Control Suppression Pool Temperature Below 95°F, Using Available Suppression Pool Cooling As Necessary (Appendix 17A)
		Can Suppression Pool Temperature Be Maintained Below 95°F? – NO
		Operate all available Suppression pool cooling, using only RHR Pumps not required to assure adequate core cooling by continuous injection, Appendix 17A.
	ATC/BOP	Place an RHR System in Pool Cooling, when directed IAW Appendix 17A.
	SRO	Before Suppression Pool Temperature rises to 110°F Continue in EOI-1 RPV Control
		Can Suppression Pool temperature and level be maintained within a safe area of curve 3? - YES
	SRO	EOI-2 (Suppression Pool Level)
		Monitor and Control Suppression Pool Level between -1 inch and -6 inches, (Appendix 18).
		Can Suppression Pool Level be maintained above -6 inches? – YES
		Can Suppression Pool Level be maintained below -1 inch? – YES

Simulator Event Guide:

Event 8 Major: ATWS, without MSIVs

	SRO	EOI-2 (Primary Containment Pressure)
		Monitor and Control PC Pressure Below 2.4 psig, Using the Vent System As Necessary, (Appendix 12)
	SRO	Can Primary Containment pressure be maintained below 2.4 psig? – YES
	ATC	Place Suppression Pool Cooling in service, IAW Appendix 17A.
		1. **IF** Adequate core cooling is assured, **OR** Directed to cool the Suppression Pool irrespective of adequate core cooling, **THEN BYPASS** LPCI injection valve open interlock AS NECESSARY: • **PLACE** 3-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in BYPASS. • **PLACE** 3-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYPASS SEL in BYPASS.
		2. **PLACE** RHR SYSTEM I(II) in Suppression Pool Cooling as follows: a. **VERIFY** at least one RHRSW pump supplying each EECW header. b. **VERIFY** RHRSW pump supplying desired RHR Heat Exchanger(s). c. **THROTTLE** the following in-service RHRSW outlet valves to obtain between 1350 and 4500 gpm RHRSW flow: • 3-FCV-23-34, RHR HX 2A RHRSW OUTLET VLV • 3-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV • 3-FCV-23-40, RHR HX 2C RHRSW OUTLET VLV • 3-FCV-23-52, RHR HX 2D RHRSW OUTLET VLV. d. **IF** Directed by SRO, **THEN PLACE** 3-XS-74-122(130), RHR SYS I(II) LPCI 2/3 CORE HEIGHT OVRD in MANUAL OVERRIDE. e. **IF** LPCI INITIATION Signal exists, **THEN** MOMENTARILY PLACE 3-XS-74-121(129), RHR SYS I(II) CTMT SPRAY/CLG VLV SELECT in SELECT.

Simulator Event Guide:

Event 8 Major: ATWS, without MSIVs

		f. **IF** 3-FCV-74-53(67), RHR SYS I(II) LPCI INBD INJECT VALVE, is OPEN, THEN **VERIFY CLOSED** 3-FCV-74-52(66), RHR SYS I(II) LPCI OUTBD INJECT VALVE. g. **OPEN** 3-FCV-74-57(71), RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV. h. **VERIFY** desired RHR pump(s) for Suppression Pool Cooling are operating.
		i. **THROTTLE** 3-FCV-74-59(73), RHR SYS I(II) SUPPR POOL CLG/TEST VLV, to maintain **EITHER** of the following as indicated on 3-FI-74-50(64), RHR SYS I(II) FLOW: • Between 7000 and 10000 gpm for one-pump operation. **OR** • At or below 13000 gpm for two-pump operation. j. **VERIFY CLOSED** 3-FCV-74-7(30), RHR SYSTEM I(II) MIN FLOW VALVE. k. **MONITOR** RHR Pump NPSH using Attachment 1.

Simulator Event Guide:

Event 9 Component: RCIC steam supply valve fails to auto open

CT#2	ATC/BOP	Recognize that 3-FCV-71-8, RCIC TURBINE STEAM SUPPLY VLV fails to open on a RCIC automatic initiation signal. Manually starts RCIC.
	ATC/BOP	Maintain Directed Level Band with RCIC, Appendix 5C..
		3. **VERIFY RESET** and **OPEN** 3-FCV-71-9, RCIC TURB TRIP/THROT VALVE RESET.
		4. **VERIFY** 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, controller in AUTO with setpoint at 600 gpm.
		5. **OPEN** the following valves: • 3-FCV-71-39, RCIC PUMP INJECTION VALVE • 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE • 3-FCV-71-25, RCIC LUBE OIL COOLING WTR VLV.
		6. **PLACE** 3-HS-71-31A, RCIC VACUUM PUMP, handswitch in START.
		7. **OPEN** 3-FCV-71-8, RCIC TURBINE STEAM SUPPLY VLV, to start RCIC Turbine.
		8. **CHECK** proper RCIC operation by observing the following: a. RCIC Turbine speed accelerates above 2100 rpm. b. RCIC flow to RPV stabilizes and is controlled automatically at 600 gpm. c. 3-FCV-71-40, RCIC Testable Check Vlv, opens by observing 3-ZI-71-40A, DISC POSITION, red light illuminated. d. 3-FCV-71-34, RCIC PUMP MIN FLOW VALVE, closes as flow rises above 120 gpm.
		9. IF BOTH of the following exist? - NO
		10. **ADJUST** 3-FIC-71-36A, RCIC SYSTEM FLOW/CONTROL, controller as necessary to control injection.

Simulator Event Guide:

Event 10 Component: CRD Controller Fails Low (FIC-85-11)

	ATC	Recognizes CRD flow controller 3-FIC-85-11 has failed to control in automatic.
		Takes manual control of 3-FIC-85-11 and restores CRD flow.
CT#1	ATC	Insert Control Rods IAW Appendix 1D
		1. **VERIFY** at least one CRD pump in service. 2. **IF** Reactor Scram or ARI CANNOT be reset, **THEN DISPATCH** personnel to **CLOSE** 3-SHV-085-0586, CHARGING WATER SHUTOFF (RB NE, El 565). 3. **VERIFY** REACTOR MODE SWITCH in SHUTDOWN. 4. **BYPASS** Rod Worth Minimizer. 5. **REFER** to Attachment 2 and **INSERT** control rods in the area of highest power as follows: a. **SELECT** control rod. b. **PLACE** CRD NOTCH OVERRIDE switch in EMERG ROD IN position UNTIL control rod is NOT moving inward. c. **REPEAT** Steps 5.a and 5.b for each control rod to be inserted. 6. **WHEN** NO further control rod movement is possible or desired, **THEN DISPATCH** personnel to **VERIFY OPEN** 3-SHV-085-0586, CHARGING WATER SHUTOFF (RB NE, El 565 ft).
	DRIVER	**WHEN** dispatched to close Charging Water Shutoff, wait 2 minutes and report 3-SHV-085-0586 closed. (mrf rd06 close) **WHEN** asked to open Charging Water Shutoff, wait 2 minutes and report 3-SHV-085-0586 open. (mrf rd06 open).
		REP classification is 1.2-S

Terminate the scenario when the following conditions are satisfied or upon request of Lead Examiner:

Control Rods are being inserted

Reactor Level is being maintained

Reactor Pressure Controlled on SRVs

SHIFT TURNOVER SHEET

Equipment Out of Service/LCO's:

None

Operations/Maintenance for the Shift:

100% power. HPCI is out of service.

Complete transfer of 4kV Unit board 3A from USST to Start Bus 1A 0-OI-57A section 8.15.1 starting at step [4]. Steps [1], [2], and [3] of 0-OI-57A section 8.15.1 are complete.

When 4kV Unit board 3A transfer is complete, lower reactor power to 90% using recirc for surveillance testing.

Unit 1 and 2 at 100% Power

Unusual Conditions/Problem Areas:

None

ATTACHMENT 2

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	30-31	48	00		
N/A	14-15	12	00		
N/A	14-47	12	00		
N/A	46-47	12	00		
N/A	46-15	12	00		
N/A	14-31	12	00		
N/A	30-47	12	00		
N/A	46-31	12	00		
N/A	30-15	12	00		
N/A	22-31	48	00		
N/A	30-39	48	00		
N/A	38-31	48	00		
N/A	30-23	48	00		

REMARKS[4]: Emergency Shove Sheet – Loadline reduction or Unit Shutdown Insert Rods Continuously to 00. Insertion may stop after completion of any group. ____________________

NOTES:

(1) RWM Group may be marked "N/A" if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position "48", this signoff verifies coupling integrity was checked in accordance with 3-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by: ____________________ / __________ Issued by ____________________ / __________

Unit Supervisor Date Reactor Engineer Date

ATTACHMENT 2
(Page 2 of 2)

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	22-47	48	00		
N/A	38-47	48	00		
N/A	38-15	48	00		
N/A	22-15	48	00		
N/A	14-39	48	00		
N/A	46-39	48	00		
N/A	46-23	48	00		
N/A	14-23	48	00		
N/A	06-31	48	00		
N/A	30-55	48	00		
N/A	54-31	48	00		
N/A	30-07	48	00		
N/A	06-39	48	00		
N/A	54-39	48	00		
N/A	54-23	48	00		
N/A	06-23	48	00		

REMARKS[4]: Emergency Shove Sheet – Loadline reduction or Unit Shutdown Insert Rods Continuously to 00. Insertion may stop after completion of any group.

NOTES:

(1) RWM Group may be marked "N/A" if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position "48", this signoff verifies coupling integrity was checked in accordance with 3-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by: ______________ / ________ Issued by ______________ / ________
Unit Supervisor Date Reactor Engineer Date

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

Reactivity Maneuver Plan U3 NRC Exam 4

Lower Reactor Power to 90%

Attachment 7
(Page 1 of 2)

Reactivity Control Plan Form

BFN Unit: 3 Valid Date(s): 8/8/11 – 8/19/11 Reactivity Control Plan #: **U3 NRC Exam 4**

Are Multiple Activations Allowed: No (If yes, US may make additional copies)

Prepared by: ____________________ / ________
Reactor Engineer Date

Reviewed by: ____________________ / ________
Qualified Reactor Engineer Date

Approved by: ____________________ / ________
RE Supervisor Date

Concurrence: ____________________ / ________
WCC/Risk/US SRO Date

Approved by: ____________________ / ________
Ops Manager or Supt. Date

Authorized by: ____________________ / ________
Shift Manager Date

RCP Activated: ____________________ / ________
Unit Supervisor Date

RCP Terminated: ____________________ / ________
Unit Supervisor Date

Title of Evolution: Lower Power to 90%

Purpose/Overview of Evolution: Lower Power to 90%

Maneuver Steps

1. Lower reactor power to 90% using core flow. NO Ramp Rate Limits Apply

Reactivity Control Plan Form

Operating Experience and General Issues: U3 NRC Exam 4

This plan is NOT valid if the unit is operating with a suspected or known fuel leaker and is not to be used. Contact Reactor Engineering if there are indications of a fuel leak.

Known Issues:

Cautions/Error Likely Situations/Special Monitoring Requirements/Contingencies:

NONE

Attachment 8

Reactivity Maneuver Instructions

STEP 1 of 1 **Reactivity Maneuver Plan # U3 NRC Exam 4**

Description of Step: Lower reactor power to 90% using core flow. No Ramp Rate Limits apply

Conditions : To be recorded at the Completion of Step Recorded: ____________/_____ (by RO) (Date)

QRE presence required in the Control Room? Yes _____ No _X_ (check)

	Predicted (may be ranges)	Actual		Predicted (may be ranges)	Actual
MW Electric	950-1050		MFLCPR	.80 - .85	
MW Thermal	2950-3150		MAPRAT	.55 - .65	
Core Flow	75-80 mlbm/hr		MFDLRX	.65 - .75	
Loadline	105-108				
Core Power	88% - 90%		Other		

Critical Parameters: To be recorded DURING Step. **IF** parameters are outside of the predictions, **THEN** discuss with the RE **AND** record conclusions in the Comments / Notes section.

Description including frequency, method of monitoring, **AND** contingency actions	High	Low

Comments / Notes:

1. Lower Reactor Power to 90% RTP
2. Document core flow changes on Attachment 10

Step Complete **AND** Reviewed by:__________________/________
Unit Supervisor / Date

Attachment 10
(Page 1 of 1)

Recirc Flow Maneuver Instructions

Reactivity Control Plan # U3 NRC Exam 4

RCP Step #	Flow Step #	Time	**Target Power (%RTP or MWe)**	Delta ±(MWe)	**Target Flow (MLb/Hr)**	Completed (RO)
1			**90%**			

Comments / Notes:

Reviewed by: ____________________/_________
Unit Supervisor / Date

Facility: **Browns Ferry NPP** Scenario No.: **NRC - 6** Op-Test No.: **1108**

Examiners:______________ Operators: **SRO:**______________

______________ **ATC:**______________

______________ **BOP:**______________

Initial Conditions: 80% power. RCIC is out of service and Breaker 1624 Alternate Feed to SD BD C.

Turnover: Place RFPT A in service from 600 RPM in accordance with 2-OI-3section 5.7 and then raise power to 100%

Event No.	Malf. No.	Event Type*	Event Description
1		N-BOP N-SRO	Place RFPT A in service from 600 RPM in accordance with 2-OI-3section 5.7
2		R-ATC R-SRO	Raise Power with Control Rods
3	RD06r3016	C-ATC C-SRO	CR 30-15 Difficult to withdraw at position 00
4	OG04a	C-BOP C-SRO	Loss of SJAE A
5	DG01c ED09c	C-BOP TS-SRO	C Shutdown Board Supply Breaker trips DG C fails to auto start
6	Batch file	C-ATC TS-SRO	RBCCW pump B trips, RBCCW sectionalizing valve fails to auto close
7	OG05a OG01	M-ALL	Explosion in Off-gas system, Loss of condenser vacuum
8	TH21	C	LOCA, Loss of SD BD C
9	IOR	I TS-SRO	RHR Sys 1 Containment Spray Valve select switch failure

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Events

1. BOP places RFPT A in service from 600 RPM in accordance with 2-OI-3section 5.7

2. ATC increases power with Control Rods

3. Control Rod 30-15 difficult to withdraw. ATC refers to 2-OI-85 CRD System section and determines double clutching is to be used initially. Double clutching will work to withdraw rod 30-15.

4. Loss of SJAE A, BOP operator swaps to B SJAE IAW 2-AOI-47-3 Loss of Condenser Vacuum.

5. Maintenance work in the area of Shutdown Board C will cause the Normal Supply Breaker to trip. Diesel Generator C will fail to automatically start and tie to the shutdown board. The BOP will respond and start DG C and tie to the shutdown board. The SRO will evaluate Technical Specifications and determine TS 3.8.1 Condition B is entered. Since the Alternate Feeder Breaker is also out of service for SD BD C, Condition G is also entered and Shutdown Board C is declared Inoperable. The SRO will then evaluate Technical Specification 3.8.7 and Condition A is entered.

6. RBCCW Pump will trip and the sectionalizing valve will fail to close automatically. ATC will take actions IAW 2-AOI-70-1 and trip RWCU Pumps and close the sectionalizing valve for RBCCW. SRO to evaluate TRM 3.4.1 and inform Chemistry that Reactor Coolant Sampling will for conductivity will have to be performed every 4 hours.

7. Explosion in Off Gas due to high hydrogen – Loss of condenser Vacuum. Crew scrams the reactor and Enter EOI-1. Bypass valves are unavailable for pressure control and HPCI is the only high pressure system available for level control.

8. LOCA will develop and crew enters EOI-2 to control degrading Containment parameters. Loss of SD BD C occurs.

9. RHR System 1 Containment Spray/Cooling Valve Select will fail. RHR Loop 2 is available for Drywell Spray. Directs spraying the drywell before exceeding the PSP curve or reaching 280°F and drywell sprays will be secured when drywell pressure lowers to 1.0 psig. SRO to evaluate Technical Specification for RHR System 1 Select Logic Failure, Technical Specification 3.6.2.5 Condition B.

Terminate the scenario when the following conditions are satisfied or upon request of Lead Examiner:

Control Rods are inserted

Drywell has been sprayed

Reactor Level is restored and maintained

Critical Tasks - Two

CT#1-When Suppression Chamber Pressure exceeds 12 psig, initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit(DSIL) curve and prior to exceeding the PSP limit.

1. Safety Significance:
   Precludes failure of containment

2. Cues:
   Procedural compliance
   High Drywell Pressure and Suppression Chamber Pressure

3. Measured by:
   Observation - US directs Drywell Sprays IAW with EOI Appendix 17B
   AND
   Observation - RO initiates Drywell Sprays

4. Feedback:
   Drywell and Suppression Pressure lowering
   RHR flow to containment

OR

CT#1- Before Drywell temperature rises to 280°F, initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit(DSIL) curve.

1. Safety Significance:
   Precludes failure of containment

2. Cues:
   Procedural compliance
   High Drywell Pressure and Suppression Chamber Pressure

3. Measured by:
   Observation - US directs Drywell Sprays IAW with EOI Appendix 17B
   AND
   Observation - RO initiates Drywell Sprays

4. Feedback:
   Drywell and Suppression Pressure lowering
   RHR flow to containment

CT#2- Terminate Drywell/Suppression Chamber Sprays before Drywell/Suppression Chamber pressure drops below 0 psig.

1. Safety Significance:
   Precludes failure of containment

2. Cues:
   Procedural compliance
   Drywell Pressure at or below 1.0 psig

3. Measured by:
   Observation - US directs Drywell Sprays secured IAW with EOI Appendix 17B
   AND
   Observation - RO secures Drywell Sprays

4. Feedback:
   RHR flow to containment lowering
   RHR Sprays Valves closed

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER: 6

8 Total Malfunctions Inserted: List (4-8)

2 Malfunctions that occur after EOI entry: List (1-4)

4 Abnormal Events: List (1-3)

1 Major Transients: List (1-2)

2 EOI's used: List (1-3)

0 EOI Contingencies used: List (0-3)

75 Run Time (minutes)

2 Crew Critical Tasks: (2-5)

YES Technical Specifications Exercised (Yes/No)

Scenario Tasks

TASK NUMBER	K/A	RO	SRO
Place RFPT A in Service			
RO U-003-NO-4	259002A4.03	3.8	3.6
Raise Power with Control Rods			
RO U-085-NO-7 SRO S-000-AD-31	2.2.2	4.6	4.1
Control Rod Difficult to Withdraw			
RO U-085-NO-19	201003A2.01	3.4	3.6
Loss of SJAE A			
RO U-066-NO-7 SRO S-047-AB-3	295002AA2.01	2.9	3.1
DG C Auto Start Failure			
RO U-082-AL-7 SRO S-000-AD-27	264000A4.04	3.7	3.7
Loss of RBCCW			
RO U-070-AL-3 SRO S-070-AB-1	206000A2.17	3.9	4.3
LOCA			
RO U-000-EM-1 RO U-000-EM-5 SRO S-000-EM-1 SRO S-000-EM-2 SRO S-000-EM-5	295024EA1.11	4.2	4.2

SIMULATOR Instructor – IC92

Batch File 1108-6
Imf dg01c
Trg e4 NRC/dgstart
Trg e4 = dmf ed09c
Ior zdi0hs2110c02a[1] trip
Ior zlo0hs2110c02a[1] off
Ior zlo0hs2110c02a[2] on
Ior zlohs7048a[2] on
Ior zlohs7048a[1] off
Ior xa554c19 alarm_off
Trg e1 7048-1
Trg e1 = bat NRC/110806-1
Ior zlohs661a[1] on
Ior zlohs661a[2] off
Ior zdixs74121[1] reset
Trg e2 modesw
Imf th21 (e2 180) 0.5 600 0.1
Imf dg03c (e2 0)

Pref File 110806
F3 bat NRC/1108rcicto
F4
F5 bat NRC/110806
F6 imf rd06r3015
F7 dmf rd06r3015
F8 imf og04a
F9 imf ed09c
F10 imf sw02b
F11 mrf sw02 align
F12 imf og01
S1 imf og05a 80 1200 100
S2 ior zdihs661a open

Facility: **Browns Ferry NPP** Scenario No.: **NRC - 6** Op-Test No.: **1108**

Examiners:__________________ Operators: **SRO:**__________________

__________________ **ATC:**__________________

__________________ **BOP:**__________________

Initial Conditions: 80% power. RCIC is out of service and Breaker 1624 Alternate Feed to SD BD C.

Turnover: Place RFPT A in service from 600 RPM in accordance with 2-OI-3section 5.7 and then raise power to 100%

Event No.	Malf. No.	Event Type*	Event Description
1		N-BOP N-SRO	Place RFPT A in service from 600 RPM in accordance with 2-OI-3section 5.7
2		R-ATC R-SRO	Raise Power with Control Rods
3	RD06r3016	C-ATC C-SRO	CR 30-15 Difficult to withdraw at position 00
4	OG04a	C-BOP C-SRO	Loss of SJAE A
5	DG01c ED09c	C-BOP TS-SRO	C Shutdown Board Supply Breaker trips DG C fails to auto start
6	Batch file	C-ATC TS-SRO	RBCCW pump B trips, RBCCW sectionalizing valve fails to auto close
7	OG05a OG01	M-ALL	Explosion in Off-gas system, Loss of condenser vacuum
8	TH21	C	LOCA, Loss of SD BD C
9	IOR	I TS-SRO	RHR Sys 1 Containment Spray Valve select switch failure

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Critical Tasks - Two

CT#1 When Suppression Chamber Pressure exceeds 12 psig, initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit(DSIL) curve and prior to exceeding the PSP limit.

1. Safety Significance:
   Precludes failure of containment

2. Cues:
   Procedural compliance.
   High Drywell Pressure and Suppression Chamber Pressure

3. Measured by:
   Observation - US directs Drywell Sprays IAW with EOI Appendix 17B
   AND
   Observation - RO initiates Drywell Sprays

4. Feedback:
   Drywell and Suppression Pressure lowering
   RHR flow to containment

OR

CT#1 - Before Drywell temperature rises to 280°F, initiate Drywell Sprays while in the safe region of the Drywell Spray Initiation Limit(DSIL) curve.

1. Safety Significance:
   Precludes failure of containment

2. Cues:
   Procedural compliance.
   High Drywell Pressure and Suppression Chamber Pressure

3. Measured by:
   Observation - US directs Drywell Sprays IAW with EOI Appendix 17B
   AND
   Observation - RO initiates Drywell Sprays

4. Feedback:
   Drywell and Suppression Pressure lowering
   RHR flow to containment

CT#2 - Terminate Drywell/Suppression Chamber Sprays before Drywell/Suppression Chamber pressure drops below 0 psig.

1. Safety Significance:
   Precludes failure of containment

2. Cues:
   Procedural compliance.
   Drywell Pressure at or below 1.0 psig

3. Measured by:
   Observation - US directs Drywell Sprays secured IAW with EOI Appendix 17B
   AND
   Observation - RO secures Drywell Sprays

4. Feedback:
   RHR flow to containment lowering
   RHR Sprays Valves closed

REP Classification is an Alert. EAL 2.1-A

Events

1. BOP places RFPT A in service from 600 RPM in accordance with 2-OI-3section 5.7

2. ATC increases power with Control Rods

3. Control Rod 30-15 difficult to withdraw. ATC refers to 2-OI-85 CRD System section and determines double clutching is to be used initially. Double clutching will work to withdraw rod 30-15.

4. Loss of SJAE A, BOP operator swaps to B SJAE IAW 2-AOI-47-3 Loss of Condenser Vacuum.

5. Maintenance work in the area of Shutdown Board C will cause the Normal Supply Breaker to trip. Diesel Generator C will fail to automatically start and tie to the shutdown board. The BOP will respond and start DG C and tie to the shutdown board. The SRO will evaluate Technical Specifications and determine TS 3.8.1 Condition B is entered. Since the Alternate Feeder Breaker is also out of service for SD BD C, Condition G is also entered and Shutdown Board C is declared Inoperable. The SRO will then evaluate Technical Specification 3.8.7 and Condition A is entered.

6. RBCCW Pump will trip and the sectionalizing valve will fail to close automatically. ATC will take actions IAW 2-AOI-70-1 and trip RWCU Pumps and close the sectionalizing valve for RBCCW. SRO to evaluate TRM 3.4.1 and inform Chemistry that Reactor Coolant Sampling will for conductivity will have to be performed every 4 hours.

7. Explosion in Off Gas due to high hydrogen – Loss of condenser Vacuum. Crew scrams the reactor and Enter EOI-1. Bypass valves are unavailable for pressure control and HPCI is the only high pressure system available for level control.

8. LOCA will develop and crew enters EOI-2 to control degrading Containment parameters. Loss of SD BD C occurs.

9. RHR System 1 Containment Spray/Cooling Valve Select will fail. RHR Loop 2 is available for Drywell Spray. Directs spraying the drywell before exceeding the PSP curve or reaching 280°F and drywell sprays will be secured when drywell pressure lowers to 1.0 psig. SRO to evaluate Technical Specification for RHR System 1 Select Logic Failure, Technical Specification 3.6.2.5 Condition B.

Terminate the scenario when the following conditions are satisfied or upon request of Lead Examiner:

Control Rods are inserted

Drywell has been sprayed

Reactor Level is restored and maintained

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER: 6

8	Total Malfunctions Inserted: List (4-8)
2	Malfunctions that occur after EOI entry: List (1-4)
4	Abnormal Events: List (1-3)
1	Major Transients: List (1-2)
2	EOI's used: List (1-3)
0	EOI Contingencies used: List (0-3)
75	Run Time (minutes)
2	Crew Critical Tasks: (2-5)
YES	Technical Specifications Exercised (Yes/No)

Scenario Tasks

TASK NUMBER	K/A	RO	SRO
Place RFPT A in Service			
RO U-003-NO-4	259002A4.03	3.8	3.6
Raise Power with Control Rods			
RO U-085-NO-7 SRO S-000-AD-31	2.2.2	4.6	4.1
Control Rod Difficult to Withdraw			
RO U-085-NO-19	201003A2.01	3.4	3.6
Loss of SJAE A			
RO U-066-NO-7 SRO S-047-AB-3	295002AA2.01	2.9	3.1
DG C Auto Start Failure			
RO U-082-AL-7 SRO S-000-AD-27	264000A4.04	3.7	3.7
Loss of RBCCW			
RO U-070-AL-3 SRO S-070-AB-1	206000A2.17	3.9	4.3
LOCA			
RO U-000-EM-1 RO U-000-EM-5 SRO S-000-EM-1 SRO S-000-EM-2 SRO S-000-EM-5	295024EA1.11	4.2	4.2

Procedures Used/Referenced:

Procedure Number	Procedure Title	Procedure Revision
2-OI-3	Reactor Feedwater System	Revision 136
2-GOI-100-12	Power Maneuvering	Revision 40
2-OI-85	Control Rod Drive System	Revision 128
2-OI-3	Reactor Feedwater System	Revision 136
2-ARP-9-5A	Alarm Response Procedure Panel 2-9-5A	Revision 48
2-AOI-47-3	Loss of Condenser Vacuum	Revision 19
2-ARP-9-53	Alarm Response Procedure Panel 2-9-53	Revision 36
ODCM	Offsite Dose Calculation Manual	Revision 20
TS 3.8.1	AC Sources - Operating	Amendment 269
TS 3.8.7	AC Distribution	Amendment 269
2-AOI-66-1	Off-Gas H2 High	Revision 19
2-AOI-100-1	Reactor Scram	Revision 95
2-EOI-1	RPV Control Flowchart	Revision 12
2-EOI-2	Primary Containment Control Flowchart	Revision 12
2-EOI-2-C-1	Alternate Level Control Flowchart	Revision 9
2-EOI-2-C-2	Emergency RPV Depressurization	Revision 6
2-EOI Appendix-6D	Injection Subsystems Lineup Core Spray System I	Revision 7
2-EOI-APPENDIX-17A	RHR System Operation Suppression Pool Cooling	Revision 12
2-EOI Appendix-5C	Injection System Lineup RCIC	Revision 5
2-EOI Appendix-7B	Alternate RPV Injection System Lineup SLC System	Revision 6

Procedures Used/Referenced Continued:

Procedure Number	Procedure Title	Procedure Revision
2-EOI Appendix-11A	Alternate RPV Pressure Control Systems MSRVs	Revision 4
2-EOI Appendix-12	Primary Containment Venting	Revision 4
2-EOI Appendix-5B	Injection System Lineup CRD	Revision 3
2-EOI Appendix-6B	Injection Subsystems Lineup RHR System I LPCI Mode	Revision 8
2-EOI Appendix-17C	RHR System Operation Suppression Chamber Sprays	Revision 11
EPIP-1	Emergency Classification Procedure	Revision 46
EPIP-5	General Emergency	Revision 41

Console Operator Instructions

A. Scenario File Summary

Batch File 1108-6
Imf dg01c
Imf dg03c
Trg e4 NRC/dgstart
Trg e4 = dmf ed09c
Ior zdi0hs2110c02a[1] trip
Ior zlo0hs2110c02a[1] off
Ior zlo0hs2110c02a[2] on
Ior zlohs7048a[2] on
Ior zlohs7048a[1] off
Ior xa554c19 alarm_off
Trg e1 7048-1
Trg e1 = bat NRC/110806-1
Ior zlohs661a[1] on
Ior zlohs661a[2] off
Ior zdixs74121[1] reset
Trg e2 modesw
Imf th21 (e2 180) 0.5 600 0.1
Imf dg03c (e2 0)

Pref File 110806

F3 bat NRC/1108rcicto
F4
F5 bat NRC/110806
F6 imf rd06r3015
F7 dmf rd06r3015
F8 imf og04a
F9 imf ed09c
F10 imf sw02b
F11 mrf sw02 align
F12 imf og01
S1 imf og05a 80 1200 100
S2 ior zdihs661a open

Scenario 6

		DESCRIPTION/ACTION
Simulator Setup	manual	Reset to IC 92
Simulator Setup	Load Batch	RestorePref NRC/110806
Simulator Setup	manual	F3 and F5
Simulator Setup		Verify file loaded

Simulator Event Guide:

Event 1 Normal: Place RFPT A in service from 600 RPM in accordance with 2-OI-3 section 5.7

	SRO	Directs Placing RFPT A in service from 600 rpm.
	BOP	Places RFPT A in service from 600 rpm.
		2-OI-3 section 5.7 Placing the Second and Third RFP/RFPT In Service **CAUTIONS** 1) FAILURE to monitor SJAE/OG CNDR CNDS FLOW, 2-FI-2-42, on Panel 2-9-6 for proper flow (between 2 x 10^6 and 3 x 10^6 lbm/hr) may result in SJAE isolation. 2) Changes in Condensate System flow may require adjustment to SPE CNDS BYPASS, 2-FCV-002-0190. **NOTE** Placing RFP 2A(2B)(2C) MIN FLOW VALVE, 2-HS-3-20(13)(6) in OPEN position will lock it open, preventing minimum flow valve oscillations at low flow. [1] **NOTIFY** Radiation Protection that an RPHP is in effect for the impending action to place RFPT 2A(2B)(2C) in service. **RECORD** time Radiation Protection notified in NOMS Narrative Log. [1.1] **VERIFY** appropriate data and signatures recorded on Appendix A per Appendix A instructions [3] **VERIFY** RFP 2A MIN FLOW VALVE, 2-HS-3-20, in OPEN position. • **CHECK** OPEN MIN FLOW VALVE, 2-FCV-3-20. [4] **SLOWLY RAISE** speed of RFPT, using RFPT 2A SPEED CONT RAISE/LOWER, 2-HS-46-8A, to establish flow to vessel and maintain level. [5] **IF** discharge valve was not opened in Step 5.6[2.2.8] **AND** RFPT discharge pressure is within 250 psig of Reactor pressure, **THEN** (Otherwise N/A) **OPEN** RFP 2A DISCHARGE VALVE, 2-FCV-3-19.
		[6] **SLOWLY RAISE** RFPT speed, using RFPT 2A SPEED CONT RAISE/LOWER switch, 2-HS-46-8A, to slowly raise RFP discharge pressure and flow on the following indications (Panel 2-9-6): • RFP Discharge Pressure - RFP 2A, 2-PI-3-16A. • RFP Discharge Flow - RFP 2A, 2-FI-3-20.
		[7] **WHEN** sufficient flow is established to maintain RFP 2A MIN FLOW VALVE, 2-FCV-3-20, in CLOSED position (≈ 2 x 106 lbm/hr), **THEN PLACE** RFP 2A MIN FLOW VALVE, 2-HS-3-20, in AUTO.

Simulator Event Guide:

Event 1 Normal: Place RFPT A in service from 600 RPM in accordance with 2-OI-3 section 5.7

	BOP	[8] **OBSERVE** lowering of speed and discharge flows on other operating RFPs. **NOTE** Steps 5.7[9] and 5.7[10] transfers control of RFPT from MANUAL GOVERNOR to individual RFPT Speed Control PDS. [9] **PULL** RFPT 2A SPEED CONT RAISE/LOWER switch, 2-HS-46-8A, to FEEDWATER CONTROL position. • **CHECK** amber light at switch extinguished.
		[10] **PERFORM** the following on RFPT 2A SPEED CONTROL (PDS), 2-SIC-46-8 (Panel 2-9-5): [10.1] **SELECT** Column 3. [10.2] **VERIFY** PDS in MANUAL. **NOTE** Performance of Steps 5.7[11] through 5.7[13] will transfer control of RFPT to REACTOR WATER LEVEL CONTROL PDS, 2-LIC-46-5. [11] **VERIFY** REACTOR WATER LEVEL CONTROL (PDS), 2-LIC-46-5 functioning properly and ready to control second or third RFP.□ [12] **SLOWLY RAISE** RFP speed. • **CHECK** discharge flow and discharge pressure rise. [13] **WHEN** RFP speed is approximately equal to operating RFP(s) speed, **THEN** on RFPT 2A SPEED CONTROL (PDS), 2-SIC-46-8: [13.1] **PLACE** PDS in AUTO. [13.2] **VERIFY** Column 3 selected.
		[14] **WHEN** RFP is in automatic mode on REACTOR WATER LEVEL CONTROL, (PDS) 2-LIC-46-5, **THEN CLOSE** the following valves: • RFPT 2A LP STOP VLV ABOVE SEAT DR, 2-FCV-6-120 • RFPT 2A LP STOP VLV BELOW SEAT DR, 2-FCV-6-121 • RFPT 2A HP STOP VLV ABOVE SEAT DR, 2-FCV-6-122 • RFPT 2A HP STOP VLV BELOW SEAT DR, 2-FCV-6-123 • RFPT 2A FIRST STAGE DRAIN VLV, 2-FCV-6-124 • RFPT A HP STEAM SHUTOFF ABOVE SEAT DRAIN, 2-FCV-6-153 (local control) • RFPT A(B)(C) LP STEAM SHUTOFF ABOVE SEAT DRAIN, 2-FCV-6-154 (local control)
	Driver	When called report 2-FCV-6-163 and 2-FCV-6-154 closed

Simulator Event Guide:

Event 1 Normal: Place RFPT A in service from 600 RPM in accordance with 2-OI-3 section 5.7

	BOP	[15] **VERIFY CLOSED** the following valves on first RFP started in Section 5.5: • RFPT (2B)(2C) LP STOP VLV ABOVE SEAT DR, 2-FCV-6-(125)(130) • RFPT (2B)(2C) LP STOP VLV BELOW SEAT DR, 2-FCV-6-(126)(131) • RFPT (B)(C) LP STEAM SHUTOFF ABOVE SEAT DR, 2-FCV-6-(156)(158) (local control) [16] **VERIFY** both RFPT Main Oil Pumps running.
	Driver	When called report 2-FVC-6-156/158 are closed
		[17] **IF** desired to stop Turning Gear for in service RFPT, **THEN PLACE** appropriate handswitch in **STOP** and **RETURN** to AUTO: • RFPT 2A TURNING GEAR MOTOR, 2-HS-3-101A [18] **GO TO** Section 6.0. [18.1] **CONTROL** and **MONITOR** RFW System operation.

Simulator Event Guide:

Event 2 Reactivity: Raise Power with Control Rods

	SRO	Direct Power Increase IAW RCP
	SRO	Notify ODS of power increase
	ATC	Raise Power with Control Rods per 2-OI-85, section 6.6. Control Rods 22-31, 30-39, 38-31, and 30-23 from 00 to 08, 30-31 from 00 to 48, 30-15 00 to 24
	ATC	Withdraw control rods IAW 2-OI-85 **NOTES** 1) Continuous control rod withdrawal may be used when a control rod is to be withdrawn greater than three notches. 2) When in areas of high notch worth, single notch withdrawal should be used instead of continuous rod withdrawal. Information concerning high notch worth is identified by Reactor Engineering in Control Rod Coupling Integrity Check, 2-SR-3.1.3.5A. 3) When continuously withdrawing a control rod to a position other than position 48, the CRD Notch Override Switch is held in the Override position and then the CRD Control Switch is held in the Rod Out Notch position. • Both switches should be released when the control rod reaches two notches prior to its intended position. (Example: If a control rod is to be withdrawn from position 00 to position 12, the CRD Notch Override Switch and the CRD Control Switch would be used to move the control rod until reaching position 08, then both switches would be released.) • If the rod settles in a notch prior to the intended position, the CRD Control Switch should be used to withdraw the rod to the intended position. (using the above example; If the control rod settles at a notch prior to the intended position of 12, the CRD Control Switch would be used to withdraw the control rod to position 12.)
		6.6.1 Initial Conditions Prior to Withdrawing Control Rods [2] **VERIFY** the following prior to control rod movement: • CRD POWER, 2-HS-85-46 in ON. • Rod Worth Minimizer is operable and LATCHED into the correct ROD GROUP when Rod Worth Minimizer is enforcing.

Event 2 Reactivity: Raise Power with Control Rods

<table>
<tr><td></td><td>ATC</td><td>6.6.2 Actions Required During and Following Control Rod Withdrawal

[4] OBSERVE the following during control rod repositioning:
• Control rod reed switch position indicators (four rod display) agree with the indication on the Full Core Display.
• Nuclear Instrumentation responds as control rods move through the core. (This ensures control rod is following drive during Control Rod movement.)

[5] ATTEMPT to minimize automatic RBM Rod Block as follows:
• STOP Control Rod withdrawal (if possible) prior to reaching any RBM Rod Block using the RBM displays on</td></tr>
<tr><td></td><td>ATC</td><td>6.6.3 Control Rod Notch Withdrawal
[1] SELECT the desired control rod by depressing the appropriate CRD ROD SELECT pushbutton, 2-XS-85-40.

[2] OBSERVE the following for selected control rod:
• CRD ROD SELECT pushbutton is brightly ILLUMINATED.
• White light on the Full Core Display ILLUMINATED
• Rod Out Permit light ILLUMINATED.

[3] VERIFY ROD WORTH MINIMIZER operable and LATCHED in to correct ROD GROUP when Rod Worth Minimizer is enforcing.

[4] PLACE CRD CONTROL SWITCH, 2-HS-85-48, in ROD OUT NOTCH and RELEASE.

[5] OBSERVE control rod settles into desired position AND ROD SETTLE light extinguishes.</td></tr>
<tr><td></td><td></td><td>[6] IF control rod is notch withdrawn to rod notch Position 48, THEN PERFORM control rod coupling integrity check</td></tr>
<tr><td></td><td></td><td></td></tr>
<tr><td></td><td></td><td></td></tr>
</table>

Simulator Event Guide:

Event 2 Reactivity: Raise Power with Control Rods

		[5] **ATTEMPT** to minimize Automatic RBM Rod block as follows: • **STOP** Control Rod Withdrawal (if possible) prior to reaching any RBM Rod Block using the RBM Displays on Panel 9-5 and perform step 6.6.2[6].
		[6] **IF** Control Rod movement was stopped to keep from exceeding a RBM Setpoint or was caused by a RBM Rod Block, **THEN PERFORM** the following at the Unit Supervisors discretion to "REINITIALIZE" the RBM: [6.1] **PLACE** the CRD Power, 2-HS-85-46 to the OFF position to deselect the control Rod. [6.2] **PLACE** the CRD Power, 2-HS-85-46 to the ON position. [6.3] **IF** desired, **THEN CONTINUE** to withdraw Control Rods and PERFORM applicable section for Control Rod withdraw
	ATC	Responds to annunciator 9-5A Window 7, CONTROL ROD WITHDRAWAL BLOCK. Operator Action: A. DETERMINE initiating condition from corresponding rod withdrawal block alarm(s) and REFER TO operator action for alarm(s). ☐
	ATC	Responds to annunciator 9-5A Window 24, RBM HIGH/INOP. Operator Action: A. IF moving control rods for start-up or power maneuvering, THEN PERFORM the following: (otherwise N/A) 1. VERIFY correct control rod selected. ☐ 2. VERIFY Rod Out Permit light is not illuminated to ensure selected rod withdrawal is inhibited. ☐ 3. CHECK annunciator LPRM HIGH (1-xa-55-5a, Window 12) and matrix light, Panel 1-9-5 to determine if the alarm is due to high flux. ☐ 4. DESELECT then RESELECT the desired Control Rod to reset the alarm and reinitialize the RBM back to normalized 100%. ☐
Event 3	DRIVER	Insert Malfunction to F6 (imf RD06r3015) to stick rod 30-15

Simulator Event Guide:

Event 3 Component: Control Rod Difficult to Withdraw

NOTE		Control Rod 30-15 will fail to withdraw from position 00
	ATC	Report Control Rod 30-15 fail to withdraw from position 00
	SRO	Direct 2-OI-85 Section 8.15
	ATC	**8.15 Control Rod Difficult to Withdraw** [1] **VERIFY** the control rod will not notch out. Refer to Section 6.6. [2] **REVIEW** all Precautions and Limitations in Section 3.0 **CAUTION** [NER/C] Never pull control rods except in a deliberate, carefully controlled manner, while closely monitoring the Reactor's response. [INPO SOER-96-001] [3] [NRC/C] **IF** RWM is enforcing, **THEN** **VERIFY** RWM is operable and LATCHED in to the correct ROD GROUP. [NRC-IR 84-02] **NOTES** 1) Steps 8.15[4] through 8.15[6] should be used when the control rod is at Position 00 while Step 8.15[7] should be used when the control rod is at OR between Positions 02 and 46. 2) Double clutching of a control rod at Position 00 will place the rod at the "overtravel in" stop, independent of the RMCS timer, allowing maximum available time to establish over-piston pressure required to maintain the collet open and prevent the collet fingers from engaging the 00 notch. 3) Step 8.15[4] may be repeated as necessary until it is determined that this method will <u>not</u> free the control rod.

Simulator Event Guide:

Event 3 Component: Control Rod Difficult to Withdraw (continued)

NRC	DRIVER NOTE	Delete malfunction F7 (dmf rd06r3015) when double clutch is used. **The procedure first directs double clutching be used**
	ATC	[4] **IF** the control rod problem is not believed to be air in the hydraulic system, **THEN** **PERFORM** the following to double clutch the control rod at Position 00: [4.1] **PLACE** AND **HOLD** CRD NOTCH OVERRIDE, 2-HS-85-47, in EMERG ROD IN, for several seconds. [4.2] **CHECK** the control rod full in indication (double green dashes) on the Full Core Display for the associated control rod. [4.3] **SIMULTANEOUSLY PLACE** CRD NOTCH OVERRIDE, 2-HS-85-47, in NOTCH OVERRIDE **AND** CRD CONTROL SWITCH, 2-HS-85-48, in ROD OUT NOTCH.
	ATC	[4.4] **WHEN** EITHER of the following occur: • Control rod begins to move, **OR** • It is determined the rod will not move, **THEN** **RELEASE** 2-HS-85-47 AND 2-HS-85-48. [4.5] **IF** the control rod successfully notches out, **THEN** **PROCEED TO** Section 6.6 and **WITHDRAW** the control rod to the appropriate position. [4.6] **IF** Desired, **THEN** **REPEAT** Steps 8.15[4.1] through 8.15[4.5] several times prior to raising drive water pressure in Step 8.15[5].

Event 4 Component: Loss of SJAE A

<table>
<tr><td>Event 3</td><td>DRIVER</td><td>Insert Malfunction to F8 (imf OG04a) to cause a loss of SJAE A</td></tr>
<tr><td></td><td>SRO</td><td>Enters AOI-47-3 Loss of Condenser Vacuum.</td></tr>
<tr><td></td><td>BOP</td><td>Offgas Panel 9-53 Alarms:

Window 4, OG HOLDUP LINE INLET FLOW LOW:
Operator action:

VERIFY OPEN, FCV-66-28, off-gas system isolation valve.

VERIFY that SJAE auto isolation has NOT occurred.

Window 10, H2 WATER CHEMISTRY ABNORMAL:
Operator action:
None at this time

Window 20, H2 WATER CHEMISTRY SHUTDOWN:
Operator action:
None at this time</td></tr>
<tr><td></td><td>BOP</td><td>Swaps to B SJAE IAW 2-AOI-47-3 Loss of Condenser Vacuum.</td></tr>
<tr><td></td><td>BOP</td><td>4.2 Subsequent Actions (continued)

[11] IF a failure of the in-service SJAE is indicated, THEN

PLACE the standby SJAE in service as follows:

NOTES
1) This section may be used to return either SJAE to service following a shutdown or an isolation.
2) Potential causes of PCV valve closure are:
• Condensate pressure from SJAE A(B) less than 60 psig, 2-PI-2-34(40), Panel 25-105.
• SJAE 2A(2B) CONDENSATE INLET VALVE closed at 2-HS-2-31A(36), Panel 2-9-6.
• SJAE 2A(2B) CONDENSATE OUTLET VALVE closed at 2-HS-2-35A(41A), Panel 2-9-6.
• STEAM TO SJAE A(B) STAGE I & II, 2-PI-1-150(152), Panel 25-105 is less than 155 psig. (disabled for the SJAE selected by 2-HS-001-0375)
• Loss of I&C bus A(B), power is required to be restored to return the SJAE to service.
3) 2-HS-001-0375, SJAE TRAIN PERMISSIVE, should be placed in the position for the SJAE being placed in service. This switch will normally be in the position of the standby SJAE.</td></tr>
</table>

Simulator Event Guide:

Event 4 Component: Loss of SJAE A (continued)

	BOP	[11.1] **PLACE** SJAE TRAIN PERMISSIVE 2-HS-001-0375 in the position for the SJAE being placed in service. This switch will normally be in the position of the Standby SJAE. (Panel 925-105 on junction box 8595) (N/A if Placing the standby SJAE in service) [11.2] **VERIFY** off gas isolation is reset, using OG OUTLET/DRAIN ISOLATION VLVS, 2-HS-90-155, Panel 2-9-8. [11.3] **VERIFY** the following valves are OPEN: • SJAE 2A(2B) INLET VALVE, 2-HS-66-11(15), Panel 2-9-8 • STEAM TO SJAE 2A(2B), 2-HS-1-155A(156A), Panel 2-9-7
	BOP	[11.4] **VERIFY** SJAE 2A(2B) OG OUTLET VALVE, 2-HS-66-14(18), AUTO/OPEN (Panel 2-9-8) [11.5] **PLACE** SJAE 2A(2B) PRESS CONTROLLER 2-HS-1-150(152) in CLOSE and then in OPEN at Panel 2-9-7. [11.6] **VERIFY** the following valves OPEN (red lights illuminated) at Panel 2-9-7. • STEAM TO SJAE 2A(2B) STAGES 1,2, AND 3, 2-PCV-1-151/166 (153/167). • SJAE 2A(2B) INTMD CONDENSER DRAIN 2-FCV-1-150(152). [11.7] **MONITOR** hotwell pressure as indicated on HOTWELL PRESS AND TEMP recorder, 2-XR-2-2 (Panel 2-9-6). [11.8] For the SJAE not being placed in service, • **VERIFY CLOSED** SJAE 2B(2A) OG OUTLET VALVE, 2-HS-66-18(14) (Panel 2-9-8). • **VERIFY CLOSED** SJAE 2B(2A) PRESSURE CONTROLLER, 2-HS-1-152(150) (Panel 2-9-7) [11.9] **VERIFY** SJAE TRAIN PERMISSIVE, 2-HS-001-0375, in the position for the SJAE selected for Standby operation SJAE A(SJAE B). (Panel 925-105 on junction box 8595)

Simulator Event Guide:

Event 4 Component: Loss of SJAE A (continued)

	BOP	[11.10] **IF** the HWC System had previously been in service, (otherwise N/A) **AND** **WHEN** stable SJAE operation is confirmed, **THEN REFER TO** 2-OI-4, HWC System, for shut down and restart guidance.
	BOP	Notifies Chemistry of HWC Shutdown

Simulator Event Guide:

Event 5 Component: DG C Auto Start Failure

	DRIVER	Insert malfunction F9 (imf ED09c) to cause a loss of Shutdown Board C, and immediately delete the malfunction.
	BOP	Recognizes Loss of Shutdown Board C failure of to DG C start, and Manually Starts DG C and close DG Supply Breaker
	BOP	Reports Loss of Shutdown Board C, failure of DG C to start, and manual start of DG C to SRO.
	DRIVER	When requested to investigate the shutdown board, report that the NORMAL Feeder Breaker to Shutdown Board C is tripped and the smell of smoke at the compartment.
	SRO	Evaluates Tech Specs 3.8.7 (condition A) and 3.8.1 (condition B)
NRC	NOTE	SRO may identify 3.8.1 G (Off site Circuit Inop) and this action is met with the 3.8.7 action.
		3.8 ELECTRICAL POWER SYSTEMS 3.8.7 Distribution Systems - Operating LCO 3.8.7 The following AC and DC electrical power distribution subsystems shall be OPERABLE: a. Unit 1 and 2 4.16 kV Shutdown Boards; b. Unit 2 480 V Shutdown Boards; c. Unit 2 480 V RMOV Boards 2A, 2B, 2D, and 2E; d. Unit 1 and 2 DG Auxiliary Boards; e. Unit DC Boards and 250 V DC RMOV Boards 2A, 2B, and 2C; f. Unit 1 and 2 Shutdown Board DC Distribution Panels; and g. Unit 1 and 3 AC and DC Boards needed to support equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," and LCO 3.7.3, "Control Room Emergency Ventilation (CREV) System." APPLICABILITY: MODES 1, 2, and 3.

Simulator Event Guide:

Event 5 Component: DG C Auto Start Failure

<table>
<tr><td></td><td>SRO</td><td>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Unit 1 and 2 4.16 kV Shutdown Board inoperable.	------------------NOTE--------------- Enter applicable Conditions and Required Actions of Condition B, C, D, and G when Condition A results in no power source to a required 480 volt board. ---	
	A.1 Restore the Unit 1 and 2 4.16 kV Shutdown Board to OPERABLE status.	5 days AND 12 days from discovery of failure to meet LCO
	AND	
	A.2 Declare associated diesel generator inoperable.	Immediately

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Unit 1 and 2 diesel generators (DGs) with two divisions of 480 V load shed logic and common accident signal logic OPERABLE; and

c. Unit 3 DG(s) capable of supplying the Unit 3 4.16 kV shutdown board(s) required by LCO 3.8.7, "Distribution Systems - Operating."

APPLICABILITY: MODES 1, 2, and 3.

</td></tr>
</table>

Simulator Event Guide:

Event 5 Component: DG C Auto Start Failure

	SRO	TSR 3.8.1

Condition	Required Action		Completion Time
B. One required Unit 1 and 2 DG inoperable.	B.1	Verify power availability from the offsite transmission network.	1 hour AND Once per 8 hours thereafter
	AND		
B. (continued)	B.2	Declare required feature(s), supported by the inoperable Unit 1 and 2 DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Determine OPERABLE Unit 1 and 2 DG(s) are not inoperable due to common cause failure.	24 hours
	OR		
	B.3.2	Perform SR 3.8.1.1 for OPERABLE Unit 1 and 2 DG(s).	24 hours
	AND		
	B.4	Restore Unit 1 and 2 DG to OPERABLE status.	7 days AND 14 days from discovery of failure to meet LCO

Simulator Event Guide:

Event 6 Component: Loss of RBCCW

	DRIVER	Insert malfunction F10 (sw02b) to cause a loss of RBCCW
	BOP/ATC	Responds to alarm 4C-12, RBCCW PUMP DISCH. HDR PRESS LOW Report Trip of RBCCW Pump 2B.
	BOP/ATC	Automatic Action: Closes 2-FCV-70-48, non-essential loop, closed cooling water sectionalizing MOV. A. **VERIFY** 2-FCV-70-48 CLOSING/CLOSED. B. **VERIFY** RBCCW pumps A and B in service. C. **VERIFY** RBCCW surge tank low level alarm is reset. D. **DISPATCH** personnel to check the following: • RBCCW surge tank level locally. • RBCCW pumps for proper operation. E. **REFER TO** 2-AOI-70-1, for RBCCW System failure and 2-OI-70, for starting spare pump.
	SRO	Enters 2-AOI-70-1.
	ATC	Closes 2-FCV-70-48 and report the sectionalizing valve failed to close automatically
	BOP	Dispatch Personnel to investigate RBCCW Pump 2B trip
	ATC	**2-AOI-70-1**
		4.1 Immediate Actions [1] **IF** RBCCW Pump(s) has tripped, **THEN** Perform the following • **SECURE** RWCU Pumps. • **VERIFY** RBCCW SECTIONALIZING VLV, 2-FCV-70-48 **CLOSED**.
	ATC	Secures RWCU Pumps and Closes 2-FCV-70-48.
	SRO	**4.2 Subsequent Actions** [1] **IF** Reactor is at power **AND** Drywell Cooling cannot be immediately restored, **AND** core flow is above 60%,**THEN:** (Otherwise N/A): [2] **IF** any EOI entry condition is met, **THEN ENTER** appropriate EOI(s) (Otherwise N/A).

Simulator Event Guide:

Event 6 Component: Loss of RBCCW (continued)

		Steps 1 and 2 are NA
		[3] **IF** RBCCW Pump(s) has tripped and it is desired to restart the tripped RBCCW pump, **THEN PERFORM** the following (Otherwise N/A): [3.1] **INSPECT** the tripped RBCCW pump and its associated breaker for any damage or abnormal conditions. [3.2] **IF** no damage or abnormal conditions are found, **THEN ATTEMPT** to restart tripped RBCCW pump(s).
	DRIVER	When dispatched, report RBCCW Pump 2B breaker is tripped. There is also a smell of burnt wiring and charring on the breaker.
	SRO	[4] **IF** unable to restart a tripped pump, **THEN PLACE** Spare RBCCW Pump in service. REFER TO 2-OI-70. Direct Unit 1 to place Spare RBCCW Pump in service
	DRIVER	When called to place spare RBCCW Pump in service, wait 3 minutes (IRF SW02 align). **THEN** inform Unit 2 Operator that spare RBCCW Pump is in service.
	SRO	[5] **IF** RBCCW flow was restored to two pump operation by placing the Spare RBCCW pump in service in the preceding step, **THEN PERFORM** the following: [5.1] **REOPEN** RBCCW SECTIONALIZING VLV, 2-HS-70-48A. [5.2] **RESTORE** the RWCU system to operation. (REFER TO 2-OI-69) Directs ATC or BOP to Open Sectionalizing Valve and Restore RWCU.
	ATC	Responds to alarm RBCCW 2-FCV-70-48 Closed
		B. **OPEN** 2-FCV-70-48, RBCCW Sectionalizing Valve, when conditions permit. C. **IF** unable to reopen 2-FCV-70-48, **THEN** if desired, **REMOVE** RWCU from service. **REFER TO** 2-OI-69. 1. **NOTIFY** CHEMISTRY if RWCU is removed from service (Reference TRM 3.4.1).
	Crew	Notifies Chemistry of TRM, TSR 3.4.1.1 for Reactor Coolant Conductivity monitoring
	ATC	Opens Sectionalizing Valve, 2-FCV-70-48.

Simulator Event Guide:

Event 6 Component: Loss of RBCCW (continued)

TR 3.4 REACTOR COOLANT SYSTEM

TR 3.4.1 Coolant Chemistry

LCO 3.4.1 Reactor coolant chemistry shall be maintained within the limits of Table 3.4.1-1.

APPLICABILITY: According to Table 3.4.1-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Conductivity greater than the limit of Table 3.4.1-1 Column B but ≤ 10 µmho/cm at 25°C.	A.1 Verify by administrative means that conductivity has not been > 1.0 µmho/cm at 25°C for > 2 weeks in the past year.	Immediately
B. Chloride concentration greater than the limit of Table 3.4.1-1 Column B or E but ≤ 0.5 ppm.	B.1 Verify by administrative means that chloride concentration has not been > 0.2 ppm for > 2 weeks in the past year.	Immediately
C. pH not within limits of Table 3.4.1-1 Column A, B, and E.	C.1 Restore pH to within limits.	24 hours

(continued)

Simulator Event Guide:

Event 6 Component: Loss of RBCCW (continued)

		TSR 3.4.1.1	Monitor reactor coolant conductivity. Continuously OR 4 hours when the continuous conductivity monitor is inoperable and the reactor is not in MODE 4 or 5 OR 8 hours when the continuous conductivity monitor is inoperable and the reactor is in MODE 4 or 5

Simulator Event Guide:

Event 6 Component: Loss of RBCCW (continued)

	ATC/BOP	2-OI-69 RWCU **3.10 Nuclear Heat Balance (NHB)** A. When a RWCU demin is removed from service it is required to ensure proper heat balance auto subsitution for the RWCU Demin removed from service per Section 8.16.
	ATC/BOP	**7.0 SYSTEM SHUTDOWN** **7.1 ICS Temperature Point Subsitution for Heat Balance** [1] **IF** removing Reactor Water Cleanup System from service when operating at power, **THEN PERFORM** RWCU ICS Temperature Point Substitution for Heat Balance adjustments: **NOTE** The following values are to be substituted for RWCU Inlet and Outlet temperatures so RWCU parameters provide conservative input to the Integrated Computer System (ICS) thermal power calculation. • 525 degrees F for 69-6A, RWCU LOOP INLET TEMP. • 420 degrees F for 69-6D, RWCU LOOP OUTLET TEMP.
		A. **TYPE** SV in the yellow block at the top of the ICS display and depress Enter key.
		B. At the prompt "ENTER POINT ID," **TYPE** 69-6A and **DEPRESS** Return key.
		C. At the prompt "ENTER SUBSTITUTE VALUE," **TYPE** 525 and **DEPRESS** Return key.
		D. At the prompt "ENTER POINT ID," **TYPE** 69-6D and **DEPRESS** Return key.
		E. At the prompt "ENTER SUBSTITUTE VALUE," **TYPE** 420 and **DEPRESS** Return key.
		F. At the prompt "ENTER SUBSTITUTE VALUE," **DEPRESS** the CANCEL key.

Simulator Event Guide:

Event 7 Major: Explosion in Off Gas due to high hydrogen – Loss of condenser Vacuum.

	DRIVER	Insert F12 (imf og01) and Shift F1, to cause High Offgas Hydrogen
	BOP	Responds to alarm the following alarms: HIGH OFFGAS % H2 TRAIN A (2-XA-55-53, Window 3) HIGH OFFGAS % H2 TRAIN B (2-XA-55-53, Window 13) OFFGAS MONITOR PANEL TROUBLE,(2-XA-55-589, Window 07)
	BOP	Reports a rise in hydrogen concentration on OFF GAS HYDROGEN ANALYZER (CH 1-Analyzer 2A, CH 2-Analyzer 2B) recorder, 2-H2R-66-96, Panel 9-53.
	SRO	Enters 2-AOI-66-1, Off-Gas H_2 High.
	DRIVER	Insert Shift F2 when many alarms are received on OFF GAS panel (ior zdihs661a open), opens condenser vacuum breaker
	BOP	Responds to alarm 9-53-Window 14 OG HOLDUP LINE INLET FLOW HIGH.
	ATC	Report degrading condenser Vacuum.
	ATC	Inserts Reactor Scram when directed; and places mode switch in shutdown.
	ATC	Recognizes reactor scram. Verifies rods inserted. Reports Scram announcement.
	SRO	Enters EOI-1 and EOI-2.
	SRO	EOI-1 (Reactor Pressure)
		Monitor and Control Reactor Pressure
		IF Drywell Pressure Above 2.4 psig? – YES, but action Not Required.
		IF Emergency Depressurization is Anticipated and the Reactor will remain subcritical without boron under all conditions, **THEN** Rapidly depressurize the RPV with the Main Turbine Bypass Valves irrespective of cooldown rate? - NO
		IF Emergency Depressurization is required THEN exit RC/P and enter C2 Emergency Depressurization? - NO
		IF RPV water level cannot be determined? - NO
	SRO	Is any MSRV Cycling? – YES. Directs Manually open MSRVs until RPV Pressure drops to the pressure at which all turbine bypass valves are open. (Appendix 11A)
		IF Steam cooling is required? - NO
		IF Suppression Pool level and temperature cannot be maintained in the safe area of Curve 3?- NO

Simulator Event Guide:

Event 7 Major: Explosion in Off Gas due to high hydrogen – Loss of condenser Vacuum. (continued)

		IF Suppression Pool level cannot be maintained in the safe area of Curve 4? - NO
		IF Drywell Control air becomes unavailable? - NO
		IF Boron injection is required? - NO
	SRO	Directs a Pressure Band with SRVs, IAW Appendix 11A. Should begin to lower Reactor Pressure, not to exceed 100°F/hr cooldown.
	ATC	Control Reactor Pressure in assigned band, IAW Appendix 11A.
	ATC/BOP	Pressure Control IAW Appendix11A, "RPV Pressure Control SRVs".
	NA	1. **IF** Drywell Control Air is **NOT** available, **THEN EXECUTE** EOI Appendix 8G, CROSSTIE CAD TO DRYWELL CONTROL AIR, **CONCURRENTLY** with this procedure.
	NA	2. **IF** Suppression Pool level is at or below 5.5 ft, **THEN CLOSE** MSRVs and **CONTROL** RPV pressure using other options.
		3. **OPEN** MSRVs, using the following sequence, to control RPV pressure as Directed by SRO:
		a. 2-PCV-1-179 MN STM LINE A RELIEF VALVE
		b. 2-PCV-1-180 MN STM LINE D RELIEF VALVE
		c. 2-PCV-1-4 MN STM LINE A RELIEF VALVE
		d. 2-PCV-1-31 MN STM LINE C RELIEF VALVE
		e. 2-PCV-1-23 MN STM LINE B RELIEF VALVE
		f. 2-PCV-1-42 MN STM LINE D RELIEF VALVE
		g. 2-PCV-1-30 MN STM LINE C RELIEF VALVE
		h. 2-PCV-1-19 MN STM LINE B RELIEF VALVE.
		i. 2-PCV-1-5 MN STM LINE A RELIEF VALVE.
		j. 2-PCV-1-41 MN STM LINE D RELIEF VALVE
		k. 2-PCV-1-22 MN STM LINE B RELIEF VALVE
		l. 2-PCV-1-18 MN STM LINE B RELIEF VALVE
		m. 2-PCV-1-34 MN STM LINE C RELIEF VALVE

Simulator Event Guide:

Event 7 Major: Explosion in Off Gas due to high hydrogen – Loss of condenser Vacuum. (continued)

	ATC/BOP	Pressure Control IAW Appendix11A RPV Pressure Control SRVs
	NA	4. **IF** Drywell Control Air header supplied from CAD System A, shows indications of being depressurized, as determined by Appendix 8G, **THEN OPEN** MSRVs supplied by CAD System B; using the following sequence to control RPV pressure; as directed by SRO:
	NA	5. **IF** Drywell Control Air header supplied from CAD System B, shows indications of being depressurized, as determined by Appendix 8G, **THEN OPEN** MSRVs supplied by CAD System A; using the following sequence to control RPV pressure; as directed by SRO:
	SRO	EOI-1 RPV Pressure – Augment RPV Pressure control, as necessary; with one or more of the following depressurization systems: • HPCI Appendix 11C • RCIC Appendix 11B • RFPTs on minimum flow Appendix 11F • Main Steam System Drains Appendix 11D • Steam Seals Appendix 11G • SJAEs Appendix 11G • Off Gas Preheater Appendix 11G • RWCU Appendix 11E.
	ATC/BOP	Augments RPV Pressure Control, if directed by SRO.
	SRO	EOI-1 (Reactor Level)
		Monitor and Control Reactor Water Level. Directs Verification of PCIS isolations.
	ATC/BOP	Verifies PCIS isolations.
	SRO	Restores and Maintains RPV Water Level between (+) 2 to (+) 51 inches with one or more of the following injection sources. (HPCI, Appendix 5D)
	ATC	Maintains the prescribed level band, IAW Appendix 5D.
		1. **IF** Suppression Pool level drops below 12.75 ft during HPCI operation, **THEN TRIP** HPCI and **CONTROL** injection using other options.
		2. **IF** Suppression Pool level CANNOT be maintained below 4.25 in., **THEN EXECUTE** EOI Appendix 16E concurrently with this procedure to bypass HPCI High Suppression Pool Water Level Suction Transfer Interlock.

Simulator Event Guide:

Event 7 Major: Explosion in Off Gas due to high hydrogen – Loss of condenser Vacuum. (continued)

		3. IF .... BOTH of the following exist: • High temperature exists in the HPCI area, **AND** • SRO directs bypass of HPCI High Temperature Isolation interlocks, THEN .... **PERFORM** the following: a. **EXECUTE** EOI Appendix 16L concurrently with this procedure. b. **RESET** auto isolation logic using 2-XS-73-58A(B) HPCI AUTO-ISOL LOGIC A(B) RESET pushbuttons.
		CAUTION • Operating HPCI Turbine below 2400 rpm may result in unstable system operation and equipment damage. • Operating HPCI Turbine with suction temperatures above 140°F may result in equipment damage.
		4. **VERIFY** 2-IL-73-18B, HPCI TURBINE TRIP RX LVL HIGH amber light extinguished. 5. **VERIFY** at least one SGTS train in operation.
		6. **VERIFY** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller in AUTO and set for 5300 gpm.
		NOTE HPCI Auxiliary Oil Pump will NOT start UNTIL 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, starts to open.
		7. **PLACE** 2-HS-73-47A, HPCI AUXILIARY OIL PUMP, handswitch in START. 8. **PLACE** 2-HS-73-10A, HPCI STEAM PACKING EXHAUSTER, handswitch in START. 9. **OPEN** the following valves: • 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE • 2-FCV-73-44, HPCI PUMP INJECTION VALVE. 10. **OPEN** 2-FCV-73-16, HPCI TURBINE STEAM SUPPLY VLV, to start HPCI Turbine.

Simulator Event Guide:

Event 7 Major: Explosion in Off Gas due to high hydrogen – Loss of condenser Vacuum. (continued)

		11. **CHECK** proper HPCI operation by observing the following: a. HPCI Turbine speed accelerates above 2400 rpm. b. 2-FCV-73-45, HPCI TESTABLE CHECK VLV, opens by observing 2-ZI-73-45A, DISC POSITION, red light illuminated. c. HPCI flow to RPV stabilizes and is controlled automatically at 5300 gpm. d. 2-FCV-73-30, HPCI PUMP MIN FLOW VALVE, closes as flow exceeds 1200 gpm. 12. **VERIFY** HPCI Auxiliary Oil Pump stops and the shaft-driven oil pump operates properly. 13. WHEN .... HPCI Auxiliary Oil Pump stops, THEN ....**PLACE** 2-HS-73-47A, HPCI AUXILIARY OIL PUMP, handswitch in AUTO.
		14. **ADJUST** 2-FIC-73-33, HPCI SYSTEM FLOW/CONTROL, controller as necessary to control injection.

Simulator Event Guide:

Event 8 Component: LOCA, Loss of SD BD C

	SRO	Enters EOI-2, all legs.
		EOI-2 (Drywell Temperature)
	SRO	Monitor and Control DW Temp Below 160°F, using available DW Cooling.
		Can Drywell Temp Be Maintained Below 160°F? - NO
	SRO	Directs H2O2 Analyzers placed in service, IAW Appendix 19.
	BOP	Places H2O2 analyzers in service, IAW Appendix 19.
	SRO	EOI-2 (Primary Containment Pressure) Monitor and Control PC Pressure Below 2.4 psig, Using the Vent System As Necessary. (Appendix 12)
	SRO	Directs venting of Primary Containment, per Appendix 12.
	BOP	Vents Primary Containment, IAW Appendix 12.
		1. **VERIFY** at least one SGTS train in service. 2. **VERIFY CLOSED** the following valves (Panel 2-9-3 or Panel 2-9-54): • 2-FCV-64-31, DRYWELL INBOARD ISOLATION VLV • 2-FCV-64-29, DRYWELL VENT INBD ISOL VALVE • 2-FCV-64-34, SUPPR CHBR INBOARD ISOLATION VLV • 2-FCV-64-32, SUPPR CHBR VENT INBD ISOL VALVE
		Steps 3, 4, 5 and 6 are If / Then steps that do not apply.
		7. **CONTINUE** in this procedure at: Step 8 to vent the Suppression Chamber through 2-FCV-84-19, **OR** Step 9 to vent the Suppression Chamber through 2-FCV-84-20.

Simulator Event Guide:

Event 9 Instrument: RHR Sys 1 Containment Spray Valve select switch failure

		8. **VENT** the Suppression Chamber using 2-FIC-84-19, PATH B VENT FLOW CONT, as follows: a. **PLACE** keylock switch 2-HS-84-35, DW/SUPPR CHBR VENT ISOL BYP SELECT, to SUPPR-CHBR position (Panel 2-9-54). b. **VERIFY OPEN** 2-FCV-64-32, SUPPR CHBR VENT INBD ISOL VALVE (Panel 2-9-54). c. **PLACE** 2-FIC-84-19, PATH B VENT FLOW CONT, in AUTO with setpoint at 100 scfm (Panel 2-9-55). d. **PLACE** keylock switch 2-HS-84-19, 2-FCV-84-19 CONTROL, in OPEN (Panel 2-9-55). e. **VERIFY** 2-FIC-84-19, PATH B VENT FLOW CONT, is indicating approximately 100 scfm. f. **CONTINUE** in this procedure at step 12.
	SRO	Can PC Pressure Be Maintained Below 2.4 psig? - NO
	SRO	Directs Suppression Chamber Sprays per Appendix 17C
	NOTE	Sprays are unavailable on Loop I of RHR due to failed Select Logic.
	ATC/BOP	Sprays the Suppression Chamber per Appendix 17C
		1. BEFORE Suppression Chamber pressure drops below 0 psig, **CONTINUE** in this procedure at Step 6.
		2. IF ..... Adequate core cooling is assured **OR** Directed to spray the Suppression Chamber irrespective of adequate core cooling, THEN ... **BYPASS** LPCI injection valve open interlock as necessary: • **PLACE** 2-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in **BYPASS**. • **PLACE** 2-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYPASS SEL in **BYPASS.**
		3. IF ..... Directed by SRO to spray the Suppression Chamber using Standby Coolant Supply, THEN ... **CONTINUE** in this procedure At Step 7 using RHR Loop I OR At Step 8 using RHR Loop II.

Simulator Event Guide:

Event 9 Instrument: RHR Sys 1 Containment Spray Valve select switch failure (continued)

		4. IF ..... Directed by SRO to spray the Suppression Chamber using Fire Protection, THEN ... **CONTINUE** in this procedure at Step 9.
		5. **INITIATE** Suppression Chamber Sprays as follows: a. **VERIFY** at least one RHRSW pump supplying each EECW header. b. IF.....EITHER of the following exists: • LPCI Initiation signal is NOT present, **OR** • Directed by SRO, THEN...**PLACE** keylock switch 2-XS-74-122(130), RHR SYS I(II) LPCI 2/3 CORE HEIGHT OVRD, in MANUAL OVERRIDE.
		c. **MOMENTARILY PLACE** 2-XS-74-121(129), RHR SYS I(II) CTMT SPRAY/CLG VLV SELECT, switch in SELECT.
		d. IF.....2-FCV-74-53(67), RHR SYS I(II) INBD INJECT VALVE, is OPEN, THEN...**VERIFY CLOSED** 2-FCV-74-52(66), RHR SYS I(II) OUTBD INJECT VALVE.
		e. **VERIFY OPERATING** the desired RHR System I(II) pump(s) for Suppression Chamber Spray.
		f. **VERIFY OPEN** 2-FCV-74-57(71), RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV.
		g. **OPEN** 2-FCV-74-58(72), RHR SYS I(II) SUPPR CHBR SPRAY VALVE.
		h. IF.....RHR System I(II) is operating ONLY in Suppression Chamber Spray mode, THEN...**CONTINUE** in this procedure at Step 5.k.
		i. **VERIFY CLOSED** 2-FCV-74-7(30), RHR SYSTEM I(II) MIN FLOW VALVE.
		j. **RAISE** System flow by placing the second RHR System I(II) pump in service as necessary.
		k. **MONITOR** RHR Pump NPSH using Attachment 2.
		l. **VERIFY** RHRSW pump supplying desired RHR Heat Exchanger(s).
		m. **THROTTLE** the following in-service RHRSW outlet valves to obtain between 1350 and 4500 gpm flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VLV • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VLV • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VLV

Simulator Event Guide:

Event 9 Instrument: RHR Sys 1 Containment Spray Valve select switch failure (continued)

		n. **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.
		6. WHEN ... EITHER of the following exists: • Before Suppression Pool pressure drops below 0 psig, **OR** • Directed by SRO to stop Suppression Chamber Sprays, THEN ... **STOP** Suppression Chamber Sprays as follows: a. **CLOSE** 2-FCV-74-58(72), RHR SYS I(II) SUPPR CHBR SPRAY VALVE. b. **VERIFY CLOSED** the following valves: • 2-FCV-74-100, RHR SYS I U-1 DISCH XTIE • 2-FCV-74-101, RHR SYS II U-3 DISCH XTIE. c. IF.....RHR operation is desired in ANY other mode, THEN...**EXIT** this EOI Appendix. d. **STOP** RHR Pumps 2A and 2C (2B and 2D). e. **CLOSE** 2-FCV-74-57(71), RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV.

Simulator Event Guide:

Event 9 Instrument: RHR Sys 1 Containment Spray Valve select switch failure (continued)

	SRO	EOI-2 (Suppression Pool Level)
		Monitor and Control Suppression Pool Level between (-) 1 inch and (-) 6 inches. (Appendix 18)
		Can Suppression Pool Level Be Maintained above (-) 6 inches? - YES
		Can Suppression Pool Level Be Maintained below (-) 1 inch? - YES
	BOP	Places H2O2 analyzers in service, IAW Appendix 19.
		1. IF............A Group 6 PCIS signal exists, THEN.......**PLACE** 2-HS-76-69, H2/O2 ANALYZER ISOLATION BYPASS switch in BYPASS (Panel 2-9-54). 2. **DEPRESS** 2-HS-76-91, H2/O2 ANALYZER ISOLATION RESET. 3. IF.......... H2/O2 Analyzer is to sample the Suppression Chamber, THEN....... **ALIGN** Analyzer as follows (Panel 2-9-54): a. **PLACE** 2-HS-76-110, H2/O2 ANALYZER DW/SUPPR CHBR SELECT in SUPPR CHBR position. b. **VERIFY** SUPPR CHBR SMPL VLVS 2-FSV-76-55/56 OPEN using 2-IL-76-49-1. c. **VERIFY** OPEN SMPL RTN VLVS 2-FSV-76-57/58 using 2-IL-76-49-3. 4. IF......... H2/O2 Analyzer is to sample the Drywell, THEN.......**ALIGN** Analyzer as follows (Panel 2-9-54): a. **PLACE** 2-HS-76-110, H2/O2 ANALYZER DW/SUPPR CHBR SELECT in DRYWELL position. b. **VERIFY OPEN** DRYWELL SMPL VLVS 2-FSV-76-49/50 using 2-IL-76-49-2. c. **VERIFY OPEN** SMPL RTN VLVS 2-FSV-76-57/58 using 2-IL-76-49-3.

Simulator Event Guide:

Event 9 Instrument: RHR Sys 1 Containment Spray Valve select switch failure (continued)

	BOP	Places H2O2 analyzers in service, IAW Appendix 19.
		5. **IF** H2/O2 Analyzer is in STANDBY at 2-MON-76-110 (Panel 2-9-55), **THEN PLACE** H2/O2 Analyzer in service at as follows: a. **TOUCH** 2-MON-76-110 display screen. b. **DEPRESS** Go To Panel PROCESS VALUES soft key. c. **DEPRESS** Go To Panel MAINT MENU soft key. d. **DEPRESS** LOG ON soft key. e. **ENTER** password 1915 on soft keypad. f. **DEPRESS** ENT soft key on keypad. g. **DEPRESS** STANDBY MODE ON soft key to enable sample pump operation. h. **VERIFY** soft key reads STANDBY MODE OFF. i. **DEPRESS** Go To Panel PROCESS VALUES soft key. j. **DEPRESS** Go To Panel MAIN soft key. k. **VERIFY** STANDBY MODE is NOT displayed.
		6. **VERIFY** H2/O2 ANALYZER SAMPLE PUMP running using 2-XI-76-110 (Panel 2-9-55). 7. **VERIFY** red LOW FLOW indicating light extinguished at 2-MON-76-110, H2/O2 ANALYZER (Panel 2-9-55).
		8. **WHEN** H2/O2 Analyzer has been aligned and sampling for 10 minutes or greater, **THEN OBTAIN** H2 and O2 readings from 2-XR-76-110 H2/O2 CONCENTRATION recorder (Panel 2-9-54).
	SRO	EOI-2 (Suppression Pool Temperature)
		Monitor and Control Suppression Pool Temperature Below 95°F, Using Available Suppression Pool Cooling As Necessary. (Appendix 17A)
		Can Suppression Pool Temperature Be Maintained Below 95°F? - NO
	ATC	Places Suppression Pool Cooling in service, IAW Appendix 17A using Loop I of Residual Heat Removal.

Simulator Event Guide:

Event 9 Instrument: RHR Sys 1 Containment Spray Valve select switch failure (continued)

	ATC/BOP	Places Suppression Pool Cooling in service, IAW Appendix 17A.
		1. **IF** Adequate core cooling is assured, **OR** Directed to cool the Suppression Pool irrespective of adequate core cooling, **THEN BYPASS** LPCI injection valve auto open signal as necessary; by **PLACING** 2-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYPASS SEL in **BYPASS**.
		2. **PLACE** RHR SYSTEM II in Suppression Pool Cooling as follows: a. **VERIFY** at least one RHRSW pump supplying each EECW header. b. **VERIFY** RHRSW pump supplying desired RHR Heat Exchanger(s). c. **THROTTLE** the following in-service RHRSW outlet valves to obtain between 1350 and 4500 gpm RHRSW flow: • 2-FCV-23-46, RHR HX 3B RHRSW OUTLET VLV • 2-FCV-23-52, RHR HX 3D RHRSW OUTLET VLV d. **IF** Directed by SRO, **THEN PLACE** 2-XS-74-130, RHR SYS II LPCI 2/3 CORE HEIGHT OVRD in MANUAL OVERRIDE. e. **IF** LPCI INITIATION Signal exists, **THEN** MOMENTARILY PLACE 2-XS-74-129, RHR SYS II CTMT SPRAY/CLG VLV SELECT in SELECT.
		f. **IF** 2-FCV-74-67, RHR SYS II LPCI INBD INJECT VALVE, is OPEN, **THEN VERIFY CLOSED** 2-FCV-74-66, RHR SYS II LPCI OUTBD INJECT VALVE. g. **OPEN** 2-FCV-74-71, RHR SYS II SUPPR CHBR/POOL ISOL VLV. h. **VERIFY** desired RHR pump(s) for Suppression Pool Cooling are operating.
		i. **THROTTLE** 2-FCV-74-73, RHR SYS II SUPPR POOL CLG/TEST VLV, to maintain **EITHER** of the following as indicated on 2-FI-74-64, RHR SYS II FLOW: • Between 7000 and 10000 gpm for one-pump operation. **OR** • At or below 13000 gpm for two-pump operation. j. **VERIFY CLOSED** 2-FCV-74-30, RHR SYSTEM II MIN FLOW VALVE. k. **MONITOR** RHR Pump NPSH using Attachment 1.

Simulator Event Guide:

Event 9 Instrument: RHR Sys 1 Containment Spray Valve select switch failure (continued)

CT #1	SRO	When Suppression Chamber Pressure exceeds 12 psig, determines that Drywell Sprays are required. Directs Spraying the Drywell before exceeding the PSP curve or reaching 280° with **Loop II** of RHR to be placed in Drywell Sprays per EOI Appendix 17B.
CT #1	ATC/BOP	**Drywell Sprays per appendix 17B** 1. IF................Adequate core cooling is assured **OR** Directed to spray the Drywell irrespective of adequate core cooling, THEN........... **BYPASS** LPCI injection valve open interlock as necessary: • **PLACE** 1-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in **BYPASS**. • **PLACE** 1-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYPASS SEL in **BYPASS.**
		2. **VERIFY** Recirc Pumps and Drywell Blowers shutdown.
		3. IF .... Directed by SRO to spray the Drywell using RHR System I(II), THEN..........**CONTINUE** in this procedure at Step 6 using RHR Loop I(II).
		NOTE Step 6 is performed ONLY if directed by Step 3 to spray the Drywell using RHR Loops I(II).
		6. **INITIATE** Drywell Sprays using RHR Loop I(II) as follows: a. BEFORE drywell pressure drops below 0 psig, **CONTINUE** in this procedure at Step 9. b. **VERIFY** at least one RHRSW pump supplying each EECW header.
		c. IF ....EITHER of the following exists: • LPCI Initiation signal is NOT present, **OR** • Directed by SRO, THEN..........**PLACE** keylock switch 1-XS-74-122(130), RHR SYS I(II) LPCI 2/3 CORE HEIGHT OVRD, in MANUAL OVERRIDE.

Simulator Event Guide:

Event 9 Instrument: RHR Sys 1 Containment Spray Valve select switch failure (continued)

		d. **MOMENTARILY PLACE** 1-XS-74-121(129), RHR SYS I(II) CTMT SPRAY/CLG VLV SELECT, switch in SELECT.
		e. IF ....1-FCV-74-53(67), RHR SYS I(II) LPCI INBD INJECT VALVE, is OPEN, THEN..........**VERIFY CLOSED** 1-FCV-74-52(66), RHR SYS I(II) LPCI OUTBD INJECT VALVE.
		f. **VERIFY OPERATING** the desired System I(II) RHR pump(s) for Drywell Spray.
		g. **OPEN** the following valves: • 1-FCV-74-60(74), RHR SYS I(II) DW SPRAY OUTBD VLV • 1-FCV-74-61(75), RHR SYS I(II) DW SPRAY INBD VLV.
		h. **VERIFY CLOSED** 1-FCV-074-0007(0030), RHR SYSTEM I(II) MIN FLOW VALVE.
		i. IF ....Additional Drywell Spray flow is necessary, THEN..........**PLACE** the second System I(II) RHR Pump in service.
		j. **MONITOR** RHR Pump NPSH using Attachment 2.
		k. **VERIFY** RHRSW pump supplying desired RHR Heat Exchanger(s).
		l. **THROTTLE** the following in-service RHRSW outlet valves to obtain between 1,350 and 4,500 gpm RHRSW flow: • 1-FCV-23-34, RHR HX 1A RHRSW OUTLET VLV • 1-FCV-23-46, RHR HX 1B RHRSW OUTLET VLV • 1-FCV-23-40, RHR HX 1C RHRSW OUTLET VLV • 1-FCV-23-52, RHR HX 1D RHRSW OUTLET VLV.

Simulator Event Guide:

Event 9 Instrument: RHR Sys 1 Containment Spray Valve select switch failure (continued)

CT #2		9. WHEN .... EITHER of the following exists: • Before drywell pressure drops below 0 psig, **OR** • Directed by SRO to stop Drywell Sprays, THEN...........**STOP** Drywell Sprays as follows: a. **VERIFY CLOSED** the following valves: • 1-FCV-74-60(74), RHR SYS I(II) DW SPRAY OUTBD VLV • 1-FCV-74-61(75), RHR SYS I(II) DW SPRAY INBD VLV • 1-FCV-74-101, UNITS 1-2 DISCHARGE CROSSTIE b. IF ....RHR pumps are running THEN...........**VERIFY OPEN** 1-FCV-74-7(30), RHR SYS I(II) MIN FLOW VALVE.
	SRO	REP Classification is an Alert. EAL 2.1-A

SHIFT TURNOVER SHEET

The unit is at approximately 80% power.

Equipment Out of Service/LCO's:

RCIC is out of service.
Breaker 1624 Alternate Feed to SD BD C is out of service

Operations/Maintenance for the Shift:

RFPT A is operating at 600 RPM.

Place RFPT A in service from 600 RPM in accordance with 2-OI-3section 5.7.

Once RFPT A is in service perform Rod Pattern adjustment and then raise power to 100% with flow in accordance with the RCP.

Units 1 and 3 are at 100% Power

Unusual Conditions/Problem Areas:

ATTACHMENT 2
(Page 1 of 2)

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	30-15	24	00		
N/A	46-31	24	00		
N/A	30-47	24	00		
N/A	14-31	24	00		
N/A	30-31	48	00		
N/A	22-31	08	00		
N/A	30-39	08	00		
N/A	38-31	08	00		
N/A	30-23	08	00		
N/A	22-23	16	00		
N/A	38-23	16	00		
N/A	38-39	16	00		
N/A	22-39	16	00		
N/A	14-23	48	00		
N/A	14-39	48	00		
N/A	46-39	48	00		
N/A	46-23	48	00		

REMARKS[4]: Emergency Shove Sheet – Loadline reduction or Unit Shutdown Insert Rods Continuously to 00. Insertion may stop after completion of any group. ____________

NOTES:

(1) RWM Group may be marked "N/A" if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position "48", this signoff verifies coupling integrity was checked in accordance with 2-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by: ____________________ / __________ Issued by ____________________ / __________
Unit Supervisor Date Reactor Engineer Date

ATTACHMENT 2
(Page 2 of 2)

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	22-47	48	00		
N/A	38-47	48	00		
N/A	38-15	48	00		
N/A	22-15	48	00		
N/A	14-47	48	00		
N/A	46-47	48	00		
N/A	46-15	48	00		
N/A	14-15	48	00		
N/A	06-31	48	00		
N/A	30-55	48	00		
N/A	54-31	48	00		
N/A	30-07	48	00		
N/A	06-39	48	00		
N/A	54-39	48	00		
N/A	54-23	48	00		
N/A	06-23	48	00		

REMARKS[4]: Emergency Shove Sheet – Loadline reduction or Unit Shutdown Insert Rods Continuously to 00. Insertion may stop after completion of any group.

NOTES:

(1) RWM Group may be marked "N/A" if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position "48", this signoff verifies coupling integrity was checked in accordance with 2-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by: ____________________ / __________ Issued by ____________________ / __________
Unit Supervisor Date Reactor Engineer Date

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

Reactivity Maneuver Plan U2 NRC Exam 6

Unit 2 Rod Pattern Adjustment

Attachment 7

Reactivity Control Plan Form

BFN Unit: 2 Valid Date(s): 8/7/11 – 8/19/11 Reactivity Control Plan #: **U2 NRC Exam 6**

Are Multiple Activations Allowed: No (If yes, US may make additional copies)

Prepared by: ____________ / ______ Reactor Engineer Date

Reviewed by: ____________ / ______ Qualified Reactor Engineer Date

Approved by: ____________ / ______ RE Supervisor Date

Concurrence: ____________ / ______ WCC/Risk/US SRO Date

Approved by: ____________ / ______ Ops Manager or Supt. Date

Authorized by: ____________ / ______ Shift Manager Date

RCP Activated: ____________ / ______ Unit Supervisor Date

RCP Terminated: ____________ / ______ Unit Supervisor Date

Title of Evolution: Unit 2 Rod Pattern Adjustment

Purpose/Overview of Evolution: Adjust Rod Pattern for 100% power operation after RFPT A returned to service.

Maneuver Steps

1. ***Withdraw Control rods IAW Attachment 2 provided by Reactor Engineer.***
2. ***Increase flow to 100% power. No Ramp Rate Limits apply***

Attachment 7
(Page 2 of 2)

Reactivity Control Plan Form

Operating Experience and General Issues: U2 NRC Exam 6

<u>Previously known control rod issues:</u>

4	172292	05/28/2009	Control Rod 46-27 double notched during the performance of the Unit 2 sequence exchange, 00 to 04.
4	150002	08/10/2008	During power ascension activities, control rod 46-27 double notched from position 00 to 04.
4	149981	08/09/2008	Control Rod 38-35 double notched during control rod withdrawal from 00 to 04.
4	148263	07/12/2008	While pulling control rods during U2 startup, CR 38-03 double-notched twice. 10 to 14 and 14 to 18

Cautions/Error Likely Situations/Special Monitoring Requirements/Contingencies:

- Rod Out Notch Override is authorized, for Rod Out Notch Override follow the guidance in 2-OI-85 section 6.6.4.
- This plan is NOT valid if the unit is operating with a suspected or known fuel leaker and is not to be used. Contact Reactor Engineering if there are indications of a fuel leak.

BFN	**Reactivity Control Plan**	

Attachment 8

Reactivity Maneuver Instructions

STEP 1 of 2 **Reactivity Maneuver Plan # U2 NRC Exam 6**

Description of Step: ***Withdraw Control rods IAW Attachment 2 provided by Reactor Engineer.***

Conditions : To be recorded at the Completion of Step Recorded: ____________/_____ (by RO) (Date)

QRE presence required in the Control Room? Yes _____ No _X_ (check)

	Predicted (may be ranges)	Actual		Predicted (may be ranges)	Actual
MW Electric	875-1100		MFLCPR	.80 - .85	
MW Thermal	2650-3200		MAPRAT	.45 - .55	
Core Flow	80-84mlbm/hr		MFDLRX	.65 - .70	
Loadline	103-106				
Core Power	88-92%		Other		

Critical Parameters: To be recorded DURING Step. **IF** parameters are outside of the predictions, **THEN** discuss with the RE **AND** record conclusions in the Comments / Notes section.

Description including frequency, method of monitoring, **AND** contingency actions	High	Low

Comments / Notes: Rod Out Notch Override is authorized, for Rod Out Notch Override follow the guidance in 2-OI-85 section 6.6.4.

Step Complete **AND** Reviewed by:__________________/________
Unit Supervisor / Date

Attachment 8

Reactivity Maneuver Instructions

STEP 2 of 2 **Reactivity Maneuver Plan # U2 NRC Exam 6**

Description of Step: Raise reactor power to 100% using core flow. No Ramp Rate Limits apply

Conditions : To be recorded at the Completion of Step Recorded: ____________/______
(by RO) (Date)

QRE presence required in the Control Room? Yes _____ No _X_ (check)

	Predicted (may be ranges)	Actual		Predicted (may be ranges)	Actual
MW Electric	1150		MFLCPR	.85 - .95	
MW Thermal	3400-3450		MAPRAT	.60 - .70	
Core Flow	85 – 95 mlbm/hr		MFDLRX	.70 - .75	
Loadline	103-106				
Core Power	100%		Other		

Critical Parameters: To be recorded DURING Step. **IF** parameters are outside of the predictions, **THEN** discuss with the RE **AND** record conclusions in the Comments / Notes section.

Description including frequency, method of monitoring, **AND** contingency actions	High	Low
N/A		

Comments / Notes:

1. Raise Reactor Power to 100% RTP
2. Document core flow changes on Attachment 10

Step Complete **AND** Reviewed by:__________________/________
Unit Supervisor / Date

Attachment 10
(Page 1 of 1)

Recirc Flow Maneuver Instructions

Reactivity Control Plan # U2 NRC Exam 6

RCP Step #	Flow Step #	Time	**Target Power (%RTP or MWe)**	Delta $\pm$(MWe)	**Target Flow (MLb/Hr)**	Completed (RO)
2	**1**		100%			

Comments / Notes:

Reviewed by: ____________________/________

Unit Supervisor / Date

ATTACHMENT 4

ROD PATTERN STEP THROUGH MAPS

Reactivity Maneuver Plan # *U2 NRC Exam 6*

Prior to RCP 6

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	
59																59
55																55
51																51
47								00								47
43																43
39						16		00		16						39
35																35
31				00		00		00		00		00				31
27																27
23						16		00		16						23
19																19
15								00								15
11																11
07																07
03																03
	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	

After RCP 6

	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	
59																59
55																55
51																51
47								24								47
43																43
39						16		08		16						39
35																35
31				24		08				08		24				31
27																27
23						16		08		16						23
19																19
15								24								15
11																11
07																07
03																03
	02	06	10	14	18	22	26	30	34	38	42	46	50	54	58	

ATTACHMENT 2
(Page 1 of 1)

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	22-31	00	08		
N/A	30-39	00	08		
N/A	38-31	00	08		
N/A	30-23	00	08		
N/A	30-31	00	48		
N/A	30-15	00	24		
N/A	46-31	00	24		
N/A	30-47	00	24		
N/A	14-31	00	24		

REMARKS[4]: Control Rod Pattern Adjustment

NOTES:

(1) RWM Group may be marked "N/A" if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position "48", this signoff verifies coupling integrity was checked in accordance with 2-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by:______________/________ Issued by ______________/________
Unit Supervisor Date Reactor Engineer Date

Facility: **Browns Ferry NPP** Scenario No.: **NRC - 7** Op-Test No.: **1108**

Examiners:______________________ Operators: **SRO:**______________________

______________________ **ATC:**______________________

______________________ **BOP:**______________________

Initial Conditions: 95% power. Loop 2 Core Spray is tagged out.

Turnover: Start SBGT Fan C and align to Reactor Bldg IAW 0-OI-65 section 5.2 and then raise reactor power to 100% with Recirculation.

Event No.	Malf. No.	Event Type*	Event Description
1		N-BOP TS-SRO	Start SBGT Fan C and align to Reactor Bldg IAW 0-OI-65 section 5.2, Relative Humidity heater fails for TS action
2		R-ATC R-SRO	Raise Power with Flow
3	AD01a	R-ATC TS-SRO C-BOP	ADS SRV 1-5 fails open
4	TH18d	C-ATC C-SRO	VFD Cooling Water Pump 2B trips with failure of the standby pump to auto start
5	FW05b	R-ATC C-BOP C-SRO	B2 Feedwater Heater Leak
6	FW30a	C-ATC C-SRO	Feedwater Pump 2A Governor Drifts Up
7	Batch File	M-ALL	Earthquake, Loss of All High Pressure injection
8		C	Loss of LPCI MG sets, loss of ALL Level Control Systems – Steam Cooling
9		ALL	Emergency Depressurization

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

EVENTS

1. BOP starts SBGT Fan C and aligns to Reactor Bldg IAW 0-OI-65 section 5.2. The relative humidity heater will fail to start and the SRO will evaluate Technical Specification 3.6.4.3 and determine Condition A is entered.

2. ATC raises Power with flow

3. ADS SRV 1-5 will fail open. ATC will lower power to less than 90%. When power is below 90% the BOP operator will perform 2-AOI-1-1 actions to close SRV. SRO will refer to Tech Specs and determine TS 3.5.1 condition F

4. The VFD Cooling Water Pump for the B Reactor Recirc VFD will trip and the standby pump will fail to start. The ATC will start the standby VFD Cooling Water Pump to restore cooling water preventing a VFD and Reactor Recirc Pump trip.

5. A tube leak on High Pressure Feedwater Heater B2 results in isolation of Extraction Steam to the heater. The crew will respond in accordance with 2-AOI-6-1A or 1C. The ATC will lower reactor power by 5%. The BOP Operators refers to 2-AOI-6-1A or 1C and determine that all automatic actions failed to occur and will isolate Heater B2.

6. RFPT A flow controller will slowly fail high, level will remain unchanged, RFPT A speed will continue to increase until the ATC or Crew notices. The controller will fail to respond until the ATC takes manual control with handswitch. The Operator will be able to restore RFPT A speed in manual. SRO should direct entry into 2-AOI-3-1.

7. Earthquake and Feedwater line Break – Loss of High Pressure Injection. On the scram, a feedwater line will break requiring the crew to isolate feedwater and HPCI. The crew will respond IAW EOI-1, EOI-2 and EOI-3.

8. Loss of LPCI MG Sets – Loss of RHR and Core Spray Pumps. Electrical faults will result in all injection to the core being lost. The SRO will transition to C-1, at -180 inches the SRO will transition to Steam Cooling. Once steam cooling is entered repairs will be completed to one electrical bus and an ECCS low pressure system will be restored for vessel injection. The SRO will transition to C-2, direct Emergency Depressurization and level restored to +2 to +51 inches.
   Loss of All injection sources – When crew enters steam cooling, one LPCI MG set will be restore to service, Crew will ED and restore reactor level.

Terminate the scenario when the following conditions are satisfied or upon request of Lead Examiner.

All Control Rods are inserted.

Emergency Depressurization is complete

Reactor Level is restored and maintained

Critical Tasks - Four

CT#1 - With NO injection system(s) operating and the reactor shutdown and at pressure, after RPV water level drops to -195 inches, direct Emergency Depressurization prior to -215 inches.

1. Safety Significance:
   Maintain adequate core cooling, prevent degradation of fission product barrier.

2. Cues:
   Procedural compliance .
   Water level trend.

3. Measured by:
   Observation - At least 6 SRV's must be opened when RPV level lowers to -200 inches.

4. Feedback:
   RPV pressure trend.
   SRV status indications.

CT#2 - With RPV pressure below the Shutoff Head of the available Low Pressure system(s), operate available Low Pressure system(s) to restore RPV water level above TAF.

1. Safety Significance:
   Maintaining adequate core cooling.

2. Cues:
   Procedural compliance.
   Pressure below low pressure ECCS system(s) shutoff head.

3. Measured by:
   Operator manually starts <u>or</u> initiates at least one low pressure ECCS system and injects into the RPV to restore water level above TAF.

4. Feedback:
   Reactor water level trend.
   Reactor pressure trend.

CT#3-To prevent an uncontrolled RPV depressurization when Reactor level cannot be restored and maintained above -162 inches, inhibit ADS .

1. Safety Significance:
   Maintain adequate core cooling, prevent degradation of fission product barrier.

2. Cues:
   Procedural compliance.

3. Measured by:
   ADS logic inhibited prior to an automatic initiation.

4. Feedback:
   RPV pressure trend.
   RPV level trend.
   ADS "ADS LOGIC BUS A/B INHIBITED" annunciator status.

CT#4 - With a SRV(s) open due to failure or incorrect automatic actuation, initiate action to close the SRV(s).

1. Safety Significance:
   Preclude exceeding Tech. Spec limit.
   Degradation of fission product barrier.

2. Cues:
   Procedural compliance.
   "SRV OPEN" annunciator status.

3. Measured by:
   Observation - SRV closed when the MSRV Inhibit Switch placed in OFF.

4. Feedback:
   Suppression Pool temperature trend.
   SRV status indications.

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER: 7

10 Total Malfunctions Inserted: List (4-8)

6 Malfunctions that occur after EOI entry: List (1-4)

4 Abnormal Events: List (1-3)

2 Major Transients: List (1-2)

3 EOI's used: List (1-3)

2 EOI Contingencies used: List (0-3)

75 Validation Time (minutes)

4 Crew Critical Tasks: (2-5)

YES Technical Specifications Exercised (Yes/No)

Scenario Tasks

TASK NUMBER	K/A	RO	SRO
Manual Initiation of SBGT Fan C			
RO U-065-NO-02 SRO S-000-AD-27	261000A4.07	3.1	3.2
Raise Power with Recirc Flow			
RO U-068-NO-17 SRO S-000-NO-138	2.1.23	4.3	4.4
ADS SRV Fails Open			
RO U-001-AB-1 SRO S-0001-AB-1	239002A2.03	4.1	4.2
VFD Cooling Water Pump Failure			
RO U-068-AL-33 SRO S-068-AB-01	202001A2.22	3.1	3.2
Loss of Feedwater Heating			
RO U-006-AB-01 SRO S-006-AB-01	2.1.43	4.1	4.3
Reactor Feed Pump Turbine Governor Failure			
RO U-003-AL-9 SRO S-003-AB-1	259002A4.01	3.8	3.6
Steam Cooling			
RO U-000-EM-15 SRO S-000-EM-15 SRO T-000-EM-16	295031EA2.04	4.6	4.8

Simulator Instructor – IC93

Batch 1108-7
Trg 11 NRC/msrvinhibit
Trg 11 = dmf ad01a
Ior zlohs682b2a[1] on
Ior zlohs682b2a[2] off
Mrf th18d trip
Trg 15 NRC/bvfd
Trg 15 = bat NRC/110807-1
Trg 1 modesw
Trg 1 = bat NRC/110807-4
Ior zdihs858a[1] close
Trg 17 NRC/rcic
Imf rc09 (e17 1:00) 100 1:00
Trg 10 NRC/rfptamaual
Trg 10 = dmf fw30a
Ior ypovfcv0521 fail_power_now
Ior zlohs0521a[2] on
Trg 5NRC/fwheating
Trg 5 = bat NRC/110807-2
Ior ypomtrsbgtrrh fail_control_power

Preference File 110807
F3 bat NRC/110807
F4 bat csloop2to
F5 imf ado1a 70
F6 ior zdihs682b1a[1] off
F7 imf fw05b 100 300 75
F8
F9 th22 100 5:00 50
F10 dmf ed12a
F11 mrf ed09 norm
F12 mrf rp02 reset
S1 mrf sl01 align
S2 imf ad01a 100

zdihs0521a[1].eq.1

Batch 110807-4
Imf sl01a and sl01b
Imf ed11a and ed11b (e1 4:00)
Imf ed12a and ed13a (e1 1:00)
Imf ed11c and ed11d (e1 5:00)
Imf fw19 (e1 0) 100 3:00

Imf th21 (e1 8:00) .25 1200

Imf rd01a (e1 3:00)

Batch 110807-2
Ior ypovfcv0521 fail_power_now
Ior zlohs0521a[2] on

Batch 110807-2
mrf th18d close

dor zlohs682b2a[1]

dor zlohs682b2a[2]

Manually Enter FW30A

Facility: **Browns Ferry NPP** Scenario No.: **NRC - 7** Op-Test No.: **1108**

Examiners:____________________ Operators: **SRO:**____________________

ATC:____________________

BOP:____________________

Initial Conditions: 95% power. Loop 2 Core Spray is tagged out.

Turnover: Start SBGT Fan C and align to Reactor Bldg IAW 0-OI-65 section 5.2 and then raise reactor power to 100% with Recirculation.

Event No.	Malf. No.	Event Type*	Event Description
1		N-BOP TS-SRO	Start SBGT Fan C and align to Reactor Bldg IAW 0-OI-65 section 5.2, Relative Humidity heater fails for TS action
2		R-ATC R-SRO	Raise Power with Flow
3	AD01a	R-ATC TS-SRO C-BOP	ADS SRV 1-5 fails open
4	TH18d	C-ATC C-SRO	VFD Cooling Water Pump 2B trips with failure of the standby pump to auto start
5	FW05b	R-ATC C-BOP C-SRO	B2 Feedwater Heater Leak
6	FW30a	C-ATC C-SRO	Feedwater Pump 2A Governor Drifts Up
7	Batch File	M-ALL	Earthquake, Loss of All High Pressure injection
8		C	Loss of LPCI MG sets, loss of ALL Level Control Systems – Steam Cooling
9		ALL	Emergency Depressurization

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Critical Tasks - Four

CT#1 - With NO injection system(s) operating and the reactor shutdown and at pressure, after RPV water level drops to -195 inches, direct Emergency Depressurization prior to -215 inches.

1. Safety Significance:
   Maintain adequate core cooling, prevent degradation of fission product barrier.

2. Cues:
   Procedural compliance .
   Water level trend.

3. Measured by:
   Observation - At least 6 SRV's must be opened when RPV level lowers to -200 inches.

4. Feedback:
   RPV pressure trend.
   SRV status indications.

CT#2 - With RPV pressure below the Shutoff Head of the available Low Pressure system(s), operate available Low Pressure system(s) to restore RPV water level above TAF.

1. Safety Significance:
   Maintaining adequate core cooling.

2. Cues:
   Procedural compliance.
   Pressure below low pressure ECCS system(s) shutoff head.

3. Measured by:
   Operator manually starts or initiates at least one low pressure ECCS system and injects into the RPV to restore water level above TAF.

4. Feedback:
   Reactor water level trend.
   Reactor pressure trend.

CT#3-To prevent an uncontrolled RPV depressurization when Reactor level cannot be restored and maintained above -162 inches, inhibit ADS .

1. Safety Significance:
   Maintain adequate core cooling, prevent degradation of fission product barrier.

2. Cues:
   Procedural compliance.

3. Measured by:
   ADS logic inhibited prior to an automatic initiation.

4. Feedback:
   RPV pressure trend.
   RPV level trend.
   ADS "ADS LOGIC BUS A/B INHIBITED" annunciator status.

CT#4 - With a SRV(s) open due to failure or incorrect automatic actuation, initiate action to close the SRV(s).

1. Safety Significance:
   Preclude exceeding Tech. Spec limit.
   Degradation of fission product barrier.

2. Cues:
   Procedural compliance.
   "SRV OPEN" annunciator status.

3. Measured by:
   Observation - SRV closed when the MSRV Inhibit Switch placed in OFF.

4. Feedback:
   Suppression Pool temperature trend.
   SRV status indications.

EVENTS

1. BOP starts SBGT Fan C and aligns to Reactor Bldg IAW 0-OI-65 section 5.2. The relative humidity heater will fail to start and the SRO will evaluate Technical Specification 3.6.4.3 and determine Condition A is entered.

2. ATC raises Power with flow

3. ADS SRV 1-5 will fail open. ATC will lower power to less than 90%. When power is below 90% the BOP operator will perform 2-AOI-1-1 actions to close SRV. SRO will refer to Tech Specs and determine TS 3.5.1 condition F

4. The VFD Cooling Water Pump for the B Reactor Recirc VFD will trip and the standby pump will fail to start. The ATC will start the standby VFD Cooling Water Pump to restore cooling water preventing a VFD and Reactor Recirc Pump trip.

5. A tube leak on High Pressure Feedwater Heater B2 results in isolation of Extraction Steam to the heater. The crew will respond in accordance with 2-AOI-6-1A or 1C. The ATC will lower reactor power by 5%. The BOP Operators refers to 2-AOI-6-1A or 1C and determine that all automatic actions failed to occur and will isolate Heater B2.

6. RFPT A flow controller will slowly fail high, level will remain unchanged, RFPT A speed will continue to increase until the ATC or Crew notices. The controller will fail to respond until the ATC takes manual control with handswitch. The Operator will be able to restore RFPT A speed in manual. SRO should direct entry into 2-AOI-3-1.

7. Earthquake and Feedwater line Break – Loss of High Pressure Injection. On the scram, a feedwater line will break requiring the crew to isolate feedwater and HPCI. The crew will respond IAW EOI-1, EOI-2 and EOI-3.

8. Loss of LPCI MG Sets – Loss of RHR and Core Spray Pumps. Electrical faults will result in all injection to the core being lost. The SRO will transition to C-1, at -180 inches the SRO will transition to Steam Cooling. Once steam cooling is entered repairs will be completed to one electrical bus and an ECCS low pressure system will be restored for vessel injection. The SRO will transition to C-2, direct Emergency Depressurization and level restored to +2 to +51 inches.
   Loss of All injection sources – When crew enters steam cooling, one LPCI MG set will be restore to service, Crew will ED and restore reactor level.

Terminate the scenario when the following conditions are satisfied or upon request of Lead Examiner.

All Control Rods are inserted.

Emergency Depressurization is complete

Reactor Level is restored and maintained

SCENARIO REVIEW CHECKLIST

SCENARIO NUMBER: 7

10	Total Malfunctions Inserted:	List (4-8)
6	Malfunctions that occur after EOI entry:	List (1-4)
4	Abnormal Events:	List (1-3)
2	Major Transients:	List (1-2)
3	EOI's used:	List (1-3)
2	EOI Contingencies used:	List (0-3)
75	Validation Time (minutes)	
4	Crew Critical Tasks:	(2-5)
YES	Technical Specifications Exercised (Yes/No)	

Scenario Tasks

TASK NUMBER	K/A	RO	SRO
Manual Initiation of SBGT Fan C			
RO U-065-NO-02 SRO S-000-AD-27	261000A4.07	3.1	3.2
Raise Power with Recirc Flow			
RO U-068-NO-17 SRO S-000-NO-138	2.1.23	4.3	4.4
ADS SRV Fails Open			
RO U-001-AB-1 SRO S-0001-AB-1	239002A2.03	4.1	4.2
VFD Cooling Water Pump Failure			
RO U-068-AL-33 SRO S-068-AB-01	202001A2.22	3.1	3.2
Loss of Feedwater Heating			
RO U-006-AB-01 SRO S-006-AB-01	2.1.43	4.1	4.3
Reactor Feed Pump Turbine Governor Failure			
RO U-003-AL-9 SRO S-003-AB-1	259002A4.01	3.8	3.6
Steam Cooling			
RO U-000-EM-15 SRO S-000-EM-15 SRO T-000-EM-16	295031EA2.04	4.6	4.8

Procedures Used/Referenced:

Procedure Number	Procedure Title	Procedure Revision
0-OI-24	Standby Gas Treatment System	Revision 53
TS 3.6.4.3	Containment Systems	Amendment 290
2-GOI-100-12	Power Maneuvering	Revision 40
2-OI-68	Reactor Recirculation System	Revision 138
2-ARP-9-3C	Alarm Response Procedure Panel 2-9-3C	Revision 20
2-AOI-1-1	Relief Valve Stuck Open	Revision 26
2-OI-74	RHR System	Revision 157
2-ARP-9-4B	Alarm Response Procedure Panel 2-9-4B	Revision 39
2-ARP-9-4C	Alarm Response Procedure Panel 2-9-4C	Revision 30
2-ARP-9-7A	Alarm Response Procedure Panel 2-9-7A	Revision 27
2-ARP-9-6A	Alarm Response Procedure Panel 3-9-6A	Revision 28
2-AOI-6-1A	High Pressure Feedwater Heater String/Extraction Steam Isolation	Revision 17
2-AOI-6-1C	High and Low Pressure Feedwater Heater String/Extraction Steam Isolation	Revision 14
2-OI-6	Feedwater Heating and Misc Drains System	Revision 84
2-ARP-9-5A	Alarm Response Procedure Panel 3-9-5A	Revision 48
2-ARP-9-6C	Alarm Response Procedure Panel 3-9-6C	Revision 19
TS 3.5.1	ECCS - Operating	Amendment 269
2-AOI-3-1	Loss of Reactor Feedwater or Reactor Water Level High/Low	Revision 20
0-AOI-100-5	Earthquake	Revision 33
2-AOI-100-1	Reactor Scram	Revision 95
2-EOI-1	RPV Control Flowchart	Revision 12
2-EOI-2	Primary Containment Control Flowchart	Revision 12
2-EOI-2-C-1	Alternate Level Control Flowchart	Revision 9
2-EOI-2-C-2	Emergency RPV Depressurization	Revision 6
2-EOI Appendix-6D	Injection Subsystems Lineup Core Spray System I	Revision 7
2-EOI-APPENDIX-17A	RHR System Operation Suppression Pool Cooling	Revision 12
2-EOI Appendix-5C	Injection System Lineup RCIC	Revision 5

Procedure Number	Procedure Title	Procedure Revision
2-EOI Appendix-5B	Injection System Lineup CRD	Revision 3
2-EOI Appendix-6B	Injection Subsystems Lineup RHR System I LPCI Mode	Revision 0
2-EOI Appendix-7B	Alternate RPV Injection System Lineup SLC System	Revision 6
2-EOI Appendix-11A	Alternate RPV Pressure Control Systems MSRVs	Revision 4
2-EOI Appendix-12	Primary Containment Venting	Revision 4
2-EOI Appendix-17C	RHR System Operation Suppression Chamber Sprays	Revision 0
EPIP-1	Emergency Classification Procedure	Revision 46
EPIP-5	General Emergency	Revision 41

Console Operator Instructions

Batch 1108-7
Trg 11 NRC/msrvinhibit
Trg 11 = dmf ad01a
Ior zlohs682b2a[1] on
Ior zlohs682b2a[2] off
Mrf th18d trip
Trg 15 NRC/bvfd
Trg 15 = bat NRC/110807-1
Trg 1 modesw
Trg 1 = bat NRC/110807-4
Ior zdihs858a[1] close
Trg 17 NRC/rcic
Imf rc09 (e17 1:00) 100 1:00
Trg 10 NRC/rfptamaual
Trg 10 = dmf fw30a
Ior ypovfcv0521 fail_power_now
Ior zlohs0521a[2] on
Trg 5NRC/fwheating
Trg 5 = bat NRC/110807-2
Ior ypomtrsbgtrrh fail_control_power

Preference File 110807
F3 bat NRC/110807
F4 bat csloop2to
F5 imf ado1a 70
F6 ior zdihs682b1a[1] off
F7 imf fw05b 100 300 75
F8
F9 th22 100 5:00 50
F10 dmf ed12a
F11 mrf ed09 norm
F12 mrf rp02 reset
S1 mrf sl01 align
S2 imf ad01a 100

zdihs0521a[1].eq.1

Batch 110807-4
Imf sl01a and sl01b
Imf ed11a and ed11b (e1 4:00)
Imf ed12a and ed13a (e1 1:00)
Imf ed11c and ed11d (e1 5:00)
Imf fw19 (e1 0) 100 3:00
Imf th21 (e1 8:00) .25 1200
Imf rd01a (e1 3:00)

Batch 110807-2
Ior ypovfcv0521 fail_power_now
Ior zlohs0521a[2] on

Batch 110807-2
mrf th18d close
dor zlohs682b2a[1]
dor zlohs682b2a[2]

Manually Enter FW30A

Scenario 7

		DESCRIPTION/ACTION
Simulator Setup	manual	Reset to IC 93
Simulator Setup	**Load Batch**	RestorePref NRC/110807
Simulator Setup	manual	Tag out Core Spray Loop 2
Simulator Setup	manual	F3 and F4
Simulator Setup		Verify file loaded

RCP required (95% - 100% with flow) and RCP for Urgent Load Reduction
Provide marked up copy of 2-GOI-100-12

Simulator Event Guide:

Event 1 Normal: Start SBGT Fan C and align to Reactor Bldg IAW 0-OI-65 section 5.2

	SRO	Directs Start SBGT Fan C and align to Reactor Bldg IAW 0-OI-65 section 5.2
	BOP	Start SBGT Fan C and align to Reactor Bldg IAW 0-OI-65 section 5.2
		5.2 Standby Gas Treatment System Manual Initiation [1] **VERIFY** the following requirements are satisfied: • SGT Train A(B)(C) in standby readiness. • Main Stack Radiation Monitoring in Service. [2] **REVIEW** the Precautions and Limitations in Section 3.0. [3] **VERIFY** suction path is aligned to SGT System as follows: [3.2] **IF** alignment to Reactor Zone Ventilation suction path is desired, **THEN VERIFY OPEN** the following dampers for the desired unit(s) to be aligned. • REACTOR ZONE EXH TO SGTS dampers, 2-HS-64-40 and 2-HS-64-41 on Panel 2-9-25 [4] **START** SGT FAN C as follows: [4.2] **IF** starting SGT FAN C from Panel 2-9-25, **THEN PLACE** SGTS FAN C, 0-HS-65-69A/2 in START. [5] **CHECK** SGT TRAIN C INLET DAMPER as follows: [5.3] **IF** SGT FAN C was started, **THEN CHECK OPEN** SGTS TRAIN C INLET DAMPER, 0-HS-65-51A indicates OPEN on Panel 2-9-25. [6] **CHECK** SGT TRAIN C RH CONTROL HTR as follows: [6.2] **IF** SGT FAN C was started, **THEN CHECK** ENERGIZED SGTS TRAIN C RH CONTROL HTR, 0-HS-65-60 on Panel 2-9-25. [7] **RECORD** start time and filter bank differential pressure for SGT Train as follows: [7.2] **IF** SGT FAN C was started, **THEN RECORD** start time and FILTER BANK DIFFERENTIAL PRESSURE, 0-PDI-65-53 on Panel 2-9-25, in the Narrative Log. [8] **DISPATCH** Operator to the Standby Gas Treatment building as soon as time allows to check for abnormal conditions (i.e. belt tightness, rubbing or vibration noises). [9] **MONITOR** Standby Gas Treatment Train operation. **REFER TO** Section 6.0.
	BOP	Reports failure of RH Heater

Simulator Event Guide:

Event 1 Normal: Start SBGT Fan C and align to Reactor Bldg IAW 0-OI-65 section 5.2

		BOP should identify failure of the RH during procedure execution, step 6.2 on previous page. If BOP turns the RH control switch out of the AUTO position, 2-9-3B, window 5 (SGT TRAIN C SWITCHES MISALIGNED), will alarm, however, the RH will not work with switch in either position (AUTO or ON)
NRC /DRIVER	NRC /DRIVER	**IF** the BOP fails to inform the SRO that the relative humidity heater failed to energize, **THEN** the Chief examiner will notify the booth driver to call the SRO (as UO) and inform him of the problem.
		2-ARP-9-3B, Window 5 – SGT TRAIN C SWITCHES MISALIGNED
		A. **CHECK** each hand switch in normal operating position in accordance with 0-OI-65, Attachment 2. B. If possible, **CLEAR** initiating signal. Otherwise **REFER TO** Tech Spec 3.6.4.3. C. **NOTIFY** UNIT SUPERVISOR/SRO and Unit 1 and Unit 3.
	SRO	SRO Evaluate Technical Specification 3.6.4.3
		LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE.
		Condition A One SGT subsystem inoperable Required Action A.1 Restore SGT subsystem to OPERABLE status Completion Time 7 Days NOTE: This LCO applies to ALL 3 UNITs

Simulator Event Guide:

Event 2 Reactivity: Raise Reactor Power with flow

	SRO	Direct Power Increase IAW RCP
	SRO	Notify ODS of power increase
		Direct Power increase using Recirc Flow per 2-GOI-100-12 [21] **WHEN** desired to restore Reactor power to 100%, **THEN PERFORM** the following as directed by Unit Supervisor and recommended by the Reactor Engineer: • **RAISE** power using control rods or core flow changes. **REFER TO** 2-SR-3.3.5(A) and 2-OI-68.
	ATC	Raise Power w/Recirc IAW 2-OI-68, section 6.2
		[1] **IF** desired to control Recirc Pumps 2A and/or 2B speed with Recirc Individual Control, **THEN PERFORM** the following; • Raise Recirc Pump 2A using, RAISE SLOW (MEDIUM), 2-HS-96-15A(15B). AND/OR • Raise Recirc Pump 2B using, RAISE SLOW (MEDIUM), 2-HS-96-16A(16B).
		[2] WHEN desired to control Recirc Pumps 2A and/or 2B speed with the RECIRC MASTER CONTROL, THEN ADJUST Recirc Pump speed 2A & 2B using the following push buttons as required: RAISE SLOW, 2-HS-96-31 RAISE MEDIUM, 2-HS-96-32
NRC	NRC	When satisfied with Reactivity Manipulation ADS SRV Fails Open requiring power to be lowered to less than 90%
Driver	Driver	At lead floor instructor direction F5, for failure of ADS SRV 1-5

Simulator Event Guide:

Event 3 Component: ADS SRVs Fail Open

	BOP	Report alarm MAIN STEAM RELIEF VALVE OPEN (2-9-3C Window 25)
		A. **CHECK** MSRV DISCHARGE TAILPIPE TEMPERATURE, 2-TR-1-1, on Panel 2-9-47 and SRV Tailpipe Flow Monitor on Panel 2-9-3 for raised temperature and flow indications. B. **REFER TO** 2-AOI-1-1.
	SRO	Enters 2-AOI-1-1
	BOP	**4.1 Immediate Action** [1] **IDENTIFY** stuck open relief valve by **OBSERVING** the following: • SRV TAILPIPE FLOW MONITOR, 2-FMT-1-4, on Panel 2-9-3, OR • MSRV DISCHARGE TAILPIPE TEMPERATURE recorder, 2-TR-1-1 on Panel 2-9-47.
	BOP	Identifies ADS SRV 1-5 open
	ATC	[2] **IF** relief valve transient occurred while operating above 90% power, **THEN PERFORM** the following (Otherwise N/A): [2.1] **INITIATE** a load reduction to ≤90% power with recirc flow.
	ATC	Lowers reactor power to ≤90% with recirc flow.
	BOP	[3] **WHILE OBSERVING** the indications for the affected Relief valve on the Acoustic Monitor; **CYCLE** the affected relief valve control switch several times as required: • CLOSE to OPEN to CLOSE positions
		4.2 Subsequent Action **4.2.2 Attempt to close valve from Panel 9-3:** [1] **PLACE** the SRV TAILPIPE FLOW MONITOR POWER SWITCH in the OFF position. [2] **PLACE** the SRV TAILPIPE FLOW MONITOR POWER SWITCH in the ON position. [3] **IF** all SRVs are CLOSED, **THEN CONTINUE** at Step 4.2.4. (Otherwise N/A) [4] **PLACE** MSRV AUTO ACTUATION LOGIC INHIBIT, 2-XS-1-202 in INHIBIT:
CT#4		Observe and report when 2-XS-1-202 is placed in Inhibit, ADS SRV 1-5 closes.

Simulator Event Guide:

Event 3 Component: ADS SRVs Fail Open

	BOP	[5] **IF** relief valve closes, **THEN OPEN** breaker or **PULL** fuses as necessary using Attachment 1 (Unit 2 SRV Solenoid Power Breaker/Fuse Table). [6] **PLACE** MSRV AUTO ACTUATION LOGIC INHIBIT 2-XS-1-202, in AUTO. Operator Does NOT perform step 6 until Breaker opened or fuses pulled
Driver	Driver	If MSRV AUTO Actuation Logic Inhibit Switch is returned to Auto prior to pulling fuses insert imf ad01a (shift F2)
	SRO	Evaluate Tech Spec 3.5.1
		Condition E One ADS valve inoperable Required Action E.1 Restore ADS Valve to OPERABLE status Completion Time 14 Days **AND** Condition F One ADS valve inoperable AND Condition A entered Required Action F.1 Restore ADS Valve to OPERABLE status Completion Time 72 Hours OR Required Action F.2 Restore low pressure ECCS spray subsystem to OPERABLE status Completion Time 72 Hours
	BOP	Directs AUO to Remove Power from SRV 1-5 **REMOVE** the power from 2-PCV-1-5 by performing one of the following: A. **OPEN** the following breakers (Preferred method) • 2C 250V RMOV, compartment 8A • Battery Board 1, breaker 727 OR B. In Panel 2-25-32 **PULL** the following fuses as necessary • Fuse 2E-F6B (Block AA, F14) • Fuse 2E-F4B (Block AA, F6)

Simulator Event Guide:

Event 3 Component: ADS SRVs Fail Open

Driver	Driver	When directed to remove power from SRV 1-5, insert mrf ad01a OUT in two minutes
	SRO	May direct Suppression Pool Cooling placed in service IAW 2-OI-74
	BOP	If Directed places Suppression Pool Cooling in Service Loop 1
		[6] **VERIFY** at least one RHRSW Pump is operating on each EECW Header.
		[7] **PLACE** RHR Pump and Heat Exchanger A(C) in service as follows: [7.1] **START** an RHRSW Pump to supply RHR Heat Exchanger A(C).
		[7.2] **ESTABLISH** RHRSW flow by performing one the following: [7.2.2] **THROTTLE OPEN** RHR HX 2A(2C) RHRSW OUTLET VLV, 2-FCV-23-34(40), as required for cooling (if another is maintaining minimum flow) and/or to maintain between 4000 and 4500 gpm RHRSW flow as indicated on 2-FI-23-36(42), RHR HTX 2A(2C) RHRSW FLOW. [7.3] **VERIFY CLOSED** RHR SYS I LPCI INBD INJECT VALVE, 2-FCV-74-53. [7.4] **VERIFY CLOSED** RHR SYS I SUPPR POOL CLG/TEST VLV, 2-FCV-74-59.
		[7.5] **VERIFY CLOSED** RHR SYS I SUPPR CHBR SPRAY VALVE, 2-FCV-74-58. [7.6] **VERIFY CLOSED** RHR SYS I DW SPRAY OUTBD VLV, 2-FCV-74-60. [7.7] **VERIFY OPEN** RHR SYS I SUPPR CHBR/POOL ISOL VLV, 2-FCV-74-57.
		[7.9] **START** RHR PUMP A(C) using 2-HS-74-5A(16A). [7.10] **THROTTLE** RHR SYS I SUPPR POOL CLG/TEST VLV, 2-FCV-74-59, to maintain RHR flow within limits, as indicated on RHR SYS I CTMT SPRAY FLOW, 2-FI-74-56. <table><tr><td>RHR Pumps in Operation</td><td>1</td><td>2</td></tr><tr><td>Loop Flow</td><td>7,000 to 10,000 gpm & Blue light illuminated</td><td><13,000 gpm & Blue light illuminated</td></tr></table>
		[7.11] **IF** desired to raise Suppression Pool Cooling flow and only one Loop I pump is in service, **THEN PLACE** the second Loop I RHR Pump and Heat Exchanger in service by **REPERFORMING** Step 8.5[7] for the second pump.

Simulator Event Guide:

Event 3 Component: ADS SRVs Fail Open

	BOP	If Directed places Suppression Pool Cooling in Service Loop 2
		[10] **PLACE** RHR Pump and Heat Exchanger B(D) in service as follows: [10.1] **START** an RHRSW Pump to supply RHR Heat Exchanger B(D).
		[10.2] **ESTABLISH** RHRSW flow by performing one the following: [10.2.2] **THROTTLE OPEN** RHR HX 2B(2D) RHRSW OUTLET VLV, 2-FCV-23-46(52), as required for cooling (if another is maintaining minimum flow) and/or to maintain between 4000 and 4500 gpm RHRSW flow as indicated on 2-FI-23-48(54), RHR HTX 2B(2D) RHRSW FLOW. [10.3] **VERIFY CLOSED** RHR SYS II LPCI INBD INJECT VALVE, 2-FCV-74-67. [10.4] **VERIFY CLOSED** RHR SYS II SUPPR POOL CLG/TEST VLV, 2-FCV-74-73.
		[10.5] **VERIFY CLOSED** RHR SYS II SUPPR CHBR SPRAY VALVE, 2-FCV-74-72. [10.6] **VERIFY CLOSED** RHR SYS II DW SPRAY OUTBD VLV, 2-FCV-74-74. [10.7] **VERIFY OPEN** RHR SYS II SUPPR CHBR/POOL ISOL VLV, 2-FCV-74-71.
		[10.9] **START** RHR PUMP B(D) using 2-HS-74-28A(39A). [10.10] **THROTTLE** RHR SYS II SUPPR POOL CLG/TEST VLV, 2-FCV-74-73, to maintain RHR flow within limits, as indicated on RHR SYS II CTMT SPRAY FLOW, 2-FI-74-70. RHR Pumps in Operation: 1 = 7,000 to 10,000 gpm & Blue light illuminated; 2 = <13,000 gpm & Blue light illuminated
		[10.11] **IF** desired to raise Suppression Pool Cooling flow and only one Loop II pump is in service, **THEN PLACE** the second Loop II RHR Pump and Heat Exchanger in service by **REPERFORMING** Step 8.5[10] for the second pump.
Driver	Driver	At lead floor instructor direction F6, for trip of 2-B-1 VFD Cooling Pump

RHR Pumps in Operation	1	2
Loop Flow	7,000 to 10,000 gpm & Blue light illuminated	<13,000 gpm & Blue light illuminated

Simulator Event Guide:

Event 4 Component: VFD Cooling Water Pump 2-B-1 Failure

	ATC	Reports the following annunciators 4B-12, 28 and 32 RECIRC DRIVE 2B COOLANT FLOW LOW, RECIRC DRIVE 2B PROCESS ALARM, and RECIRC DRIVE 2B DRIVE ALARM
	ATC	Reports the 2-B-1 VFD Cooling Water Pump for the B Recirc Pump, has tripped.
	ATC	Reports Standby Recirc Drive Cooling Water Pump2-B-2, failed to auto start.
	ATC	RECIRC DRIVE 2B COOLANT FLOW LOW **STARTS** RECIRC DRIVE cooling water pump 2-B-2 **and** **DISPATCHES** personnel to the RECIRC DRIVE, to check the operation of the Recirc Drive cooling water system.
	SRO	Concurs with start of Standby VFD Pump.
	BOP	RECIRC DRIVE 2B DRIVE ALARM A. **REFER TO** ICS Group Display "GD @VFDBDA" and determine cause of alarm. B. **IF** a problem with the cooling water system is indicated, **THEN VERIFY** proper operation of cooling water system. C. **IF** the problem is conductivity in the cooling water system, **THEN VERIFY** demineralizer is in service. D. **IF** a problem with power supplies is indicated, **THEN VERIFY** all the low voltage supply breakers are CLOSED/ON. E. For all other alarms, or any problems encountered **CONTACT** system engineering.
	Crew	Verifies Standby pump started by pulling up ICS displays.
	BOP	Dispatches personnel to VFD.
	DRIVER	Wait 4 minutes after dispatched, **THEN** report tripped VFD Pump 2-B-1 is "hot to the touch", internal bkr closed, 480 volt bkr tripped (480 V SD BD 2A-5D).
	DRIVER	Upon Lead examiner direction F7 for Loss of Feedwater Heating

Simulator Event Guide:

Event 5 Component: B2 Feedwater Heater Leak

	DRIVER	When directed by NRC insert F7 for Loss of Feedwater Heating and 2-FCV-5-21, HP HEATER 2B2 EXTR ISOL VLV Fail to isolate.
	ATC/BOP	Announces "BYPASS VALVE TO CONDENSER NOT CLOSED" and refers to 2-ARP-9-6A, window 18. A. **CHECK** heater high or low level or moisture separator high or low level alarm window illuminated on Panel 2-9-6 or 2-9-7 to identify which bypass valve is opening. B. **CHECK** ICS to determine which bypass valve is open. C. **DISPATCH** personnel to check which valve's light is extinguished on junction box.
	DRIVER	Acknowledge dispatch, wait 1-2 minutes and report 2-LCV-6-22B light is out on junction box 34-21.
	ATC/BOP	Announces "HEATER B2 LEVEL HIGH" and refers to 2-ARP-9-6A window 9. A. **CHECK** the following indications: • Condensate flow recorder 2-29, Panel 2-9-6. Rising flow is a possible indication of a tube leak. • Heater B2 shell pressure, 2-PI-5-22 and drain cooler B5 flow, 2-FI-6-34, Panel 2-9-6. High or rising shell pressure or drain cooler flow is possible indication of a tube leak. B. **CHECK** drain valve 2-FCV-6-95 open. C. **CHECK** level on ICS screen, FEEDWATER HEATER LEVEL (FWHL). • **IF** the 2B2 heater indicates HIGH (Yellow), **THEN VERIFY** proper operation of the Drain and Dump Valves. • **DISPATCH** personnel to local Panel 2-LPNL-925-562C to VERIFY and MANUALLY control the level. D. **IF** a valid HIGH HIGH level is received, **THEN GO TO** 2-AOI-6-1A or 2-AOI-6-1C.
	ATC/BOP	Checks condensate flow recorder, Heater B2 shell pressure and Drain Cooler B5 flow for indications of a tube leak Checks drain valve 2-FCV-6-95 open Checks 2B2 Heater level on ICS and dispatches personnel to verify and manually control level
	DRIVER	Acknowledge order to verify and manually control level on B2 Heater. Wait 6 minutes and report unable to take manual control of B2 Heater.

Simulator Event Guide:

Event 5 Component: B2 Feedwater Heater Leak

	ATC/BOP	Announces B1 and B2 High Pressure Heater Extraction Isolation
	SRO	Directs crew to enter 2-AOI-6-1A or 2-AOI-6-1C
	ATC/BOP	**2-AOI-6-1A High Pressure Feedwater Heater String/Extraction Steam Isolation** **4.1 Immediate Actions** [1] **REDUCE** Core Thermal Power to ≥ 5% below initial power level to maintain thermal margin. **4.2 Subsequent Actions** [1] **REFER** TO 2-OI-6 for turbine/heater load restrictions. [2] **REQUEST** Reactor Engineer EVALUATE and ADJUST thermal limits, as required. [3] **ADJUST** reactor power and flow as directed by Reactor Engineer/Unit Supervisor to stay within required thermal and feedwater temperature limits. **REFER TO** 2-GOI-100-12 or 2-GOI-100-12A for the power reduction. [4] **ISOLATE** heater drain flow from the feedwater heater string that isolated by closing the appropriate FEEDWATER HEATER B-2 DRAIN TO HTR B-3, 2-FCV-6-95. [5] **IF** a tube leak is indicated, **THEN** **PERFORM** manual actions of Attachment 1 for affected heaters. [6] **VERIFY** automatic actions occur. **REFER TO** Attachment 1. [7] **MONITOR** TURB THRUST BEARING TEMPERATURE, 2-TR-47-23, for rises in metal temperature and possible active/passive plate reversal. [8] **DETERMINE** cause which required heater isolation and **PERFORM** necessary corrective action.

Simulator Event Guide:

Event 5 Component: B2 Feedwater Heater Leak

	ATC/BOP	**2-AOI-6-1A High Pressure Feedwater Heater String/Extraction Steam Isolation (continued)** **4.2 Subsequent Actions (continued)** [9] **WHEN** the condition which required heater isolation is no longer required, **THEN** **RESTORE** affected heater. **REFER TO** 2-OI-6.
	ATC BOP	Lower Reactor Power greater than 5% below initial power level using Recirc Pump flow adjustments Refers to 2-OI-6 for turbine/heater load restrictions Contacts Reactor Engineer to evaluate and adjust Thermal Limits, if needed Isolates heater drain flow B2 Heater Drain to B3 Heater by shutting 2-FCV-6-95
	SRO	Directs isolating FW to B HP heater string based on indications of tube leak by performing manual actions of Attachment 1 and verifying automatic actions occur Directs power reduction to 920 MWe (79%) power (Power Reduction with RCP flow or Control Rods) per 2-OI-6, Illustration 1 **2-OI-6 Illustration 1** **HEATERS OUT (Tube and Shell Side)** ** One HP string 920 MWe (79%) One LP string 920 MWe (79%) One HP and LP string 920 MWe (79%) Enters 2-GOI-100-12, Power Maneuvering Notifies Rx Eng. And ODS of Feedwater Heater isolation and power reduction

Simulator Event Guide:

Event 5 Component: B2 Feedwater Heater Leak

	BOP	**2-AOI-6-1A Attachment 1** Closes the following Feedwater Valves Manually 2-FCV-3-31, HP HTR 2B2 FW INLET ISOL VALVE 2-FCV-3-76, HP HTR 2B1 FW OUTLET ISOL VALVE Verifies the following valves close automatically 2-FCV-5-9, HP HEATER 2B1 EXTR ISOL VLV 2-FCV-5-21, HP HEATER 2B2 EXTR ISOL VLV 2-FCV-6-74, MOISTURE SEP LC RES B1 ISOL VLV 2-FCV-6-172, MOISTURE SEP LC RES B2 ISOL VLV **Takes action to manually shut 2-FCV-5-21 upon determining the valve did not automatically close, and reports to SRO** Recognizes HTR level lowers as a result of isolating the Condensate side of 2B HP HTR string (i.e. tube leak) and reports to crew
	DRIVER	After HS for 2-FCV-5-21 taken to closed, verify **Trigger 5** goes active. As Reactor Engineer , when contacted direct crew to follow the guidance of urgent load reduction and 2-OI-6
	ATC	Lower Reactor Power to <920 MWe/<79% power by lowering recirc flow.
	SRO	Direct ATC to insert the first group of control rods on the Emergency Shove Sheet per Reactor Engineer recommendation.
	ATC	Inserts the first group of rods on the Emergency Shove Sheet using a peer check as directed by Rx Engineer & Unit Supervisor

Simulator Event Guide:

Event 6: Feedwater Pump 2A Governor Drifts Up

	DRIVER	When NRC directs, insert imf fw30a check current setting of fw30a and then ramp to 100 over 20 minutes for Feedwater Pump Governor Failure. When operator takes the RFPT Governor to manual the malfunction is automatically deleted, therefore, **IF** the operator pulls the Governor control knob back out, the malfunction must be manually reinserted and deleted when the operator returns the Governor control knob back down to force the operator to control level manually. For Example (imf fw30a 100 1200 67.05)
NOTE	NRC	Annunciator 2-9-6C Window 32, RFP DISCH FLOW LOW, will alarm at approximately 83% of malfunction severity if the crew does not notice the failure before the alarm.
	ATC	Report Rising Reactor Water Level and RFPT is not responding.
	SRO	Direct manual control of operating RFPT and Enter 2-AOI-3-1.
NOTE	NRC	The crew may decide to trip the 2A RFPT per 2-AOI-3-1 step 4.2 [6].
		4.2 Subsequent Actions [1] **VERIFY** applicable automatic actions. **6.0 HIGH REACTOR WATER LEVEL** [2] **IF** Feedwater Control System has failed, **THEN PERFORM** the following: [2.1] **PLACE** individual RFPT Speed Control Raise/Lower switches in MANUAL GOVERNOR (depressed position with amber light illuminated). [2.2] **ADJUST** RFP Discharge flows with RFPT Speed Control Raise/Lower switches as necessary to maintain level. [6] **IF** level continues to rise, **THEN TRIP** a RFP, as necessary. [8] **IF** RFPs are in manual control, **THEN LOWER** speed of operating RFPs. [9] **EXPECT** a possible Reactor power rise due to a rise in moderation. [10] **IF** unit remains on-line, **THEN PERFORM** the following: • **RETURN** Reactor water level to normal operating level of 33" (normal range). • **REQUEST** Nuclear Engineer check core limits.
	ATC	Take MANUAL GOVERNOR control of RFPT and maintain Reactor Water Level Manually in the Normal Level Band. Operator may attempt to control RFPT with PDS. PDS will not respond.
	DRIVER	If a scram is inserted or at NRC direction initiate F9 for LOCA and make Earthquake calls

Simulator Event Guide:

Event 7 Major: Earthquake

Driver	**Driver**	Report confirmed earthquake Unit 1 is handling 0-AOI-100-5, Earthquake
	ATC/BOP	Reports rising Drywell pressure
	SRO	Establishes Drywell Pressure to insert a Reactor Scram
	ATC	Insert Manual SCRAM when directed
	SRO	Enters 2-AOI-100-1, EOI-1 and EOI-2 on High Drywell Pressure
	ATC	2-AOI-100-1 [1] **DEPRESS** REACTOR SCRAM A and B, 2-HS-99-5A/S3A and 2-HS-99-5A/S3B, on Panel 2-9-5 [2] **IF** scram is due to a loss of RPS, **THEN** (Otherwise N/A) [3] REFUEL MODE ONE ROD PERMISSIVE light check: [3.1] **PLACE** REACTOR MODE SWITCH, 2-HS-99-5A-S1, in REFUEL. [3.2] **CHECK** REFUEL MODE ONE ROD PERMISSIVE light, 2-XI-85-46, illuminates. [3.3] **IF** REFUEL MODE ONE ROD PERMISSIVE light, 2-XI-85-46, is not illuminated, **THEN CHECK** all control rod positions at Full-In Overtravel, or Full-In. (Otherwise N/A) [4] **PLACE** REACTOR MODE SWITCH, 2-HS-99-5A-S1, in SHUTDOWN position.
Driver	**Driver**	Ensure **trigger 1** goes active on MODESWITCH

Simulator Event Guide:

Event 7 Major: Earthquake – Feedwater Line Break

Driver	Driver	Report confirmed earthquake Unit 1 is handling 0-AOI-100-5, Earthquake
	ATC	Determines Feedwater Leak on the A Feedwater Line due to Feedwater Line A Flow high and Feedwater line B flow lowering to 0 and Reactor Feed Pump Flows Increasing with a Lowering Reactor Water Level.
	SRO	Directs Reactor Feed Pumps to be tripped, Reactor Feed Pump Discharge Valves shut, and Condensate Booster Pumps then Condensate Pumps secured. (Isolate and stop leak) Also directs HPCI locked out due to Feedwater Line Break on the A line.
	ATC	Trips Reactor Feed Pumps and shuts Reactor Feed Pump Discharge Valves. Secures Condensate Booster Pumps then Condensate Pumps.
	BOP	Trips HPCI if running and places HPCI Aux Oil Pump in PTL when HPCI speed lowers to 0 rpm.
	SRO	**Enters EOI-1 on Low Reactor Water Level and High Drywell Pressure** **RC/Q** Monitor and Control Reactor Power. Directs Exit of EOI-1 RC/Q Leg, after ATC reports All Rods In on Scram Report. **RC/P** Monitor and Control RPV Pressure. Answers **NO** to: Is any MSRV cycling? Directs BOP to maintain RPV Pressure 500 -1000 psig using Appendix 11A.. **RC/L** Monitor and Control RPV Water Level. Verify as Required: • PCIS Isolations (Groups 1, 2 and 3) • ECCS • RCIC Directs level band of +2 to +51 inches, with Appendix 5C, 5B and/or 7B.

Simulator Event Guide:

Event 7 Major: Earthquake – Feedwater Line Break

	ATC/BOP	Pressure Control IAW Appendix11A, RPV Pressure Control SRVs
		1. IF Drywell Control Air is **NOT** available, **THEN**: **EXECUTE** EOI Appendix 8G, CROSSTIE CAD TO DRYWELL CONTROL AIR, **CONCURRENTLY** with this procedure.
		2. IF Suppression Pool level is at or below 5.5 ft, **THEN:** **CLOSE** MSRVs and **CONTROL** RPV pressure using other options.
		3. **OPEN** MSRVs; using the following sequence to control RPV pressure, as directed by SRO:
		a. 2-PCV-1-179 MN STM LINE A RELIEF VALVE
		b. 2-PCV-1-180 MN STM LINE D RELIEF VALVE.
		c. 2-PCV-1-4 MN STM LINE A RELIEF VALVE
		d. 2-PCV-1-31 MN STM LINE C RELIEF VALVE
		e. 2-PCV-1-23 MN STM LINE B RELIEF VALVE
		f. 2-PCV-1-42 MN STM LINE D RELIEF VALVE
		g. 2-PCV-1-30 MN STM LINE C RELIEF VALVE
		h. 2-PCV-1-19 MN STM LINE B RELIEF VALVE.
		i. 2-PCV-1-5 MN STM LINE A RELIEF VALVE.
		j. 2-PCV-1-41 MN STM LINE D RELIEF VALVE
		k. 2-PCV-1-22 MN STM LINE B RELIEF VALVE
		l. 2-PCV-1-18 MN STM LINE B RELIEF VALVE
		m. 2-PCV-1-34 MN STM LINE C RELIEF VALVE

Simulator Event Guide:

Event 8 Major: Loss of LPCI MG sets, loss of ALL Level Control Systems – Steam Cooling

NOTE	NOTE	When RCIC is started, a break will occur on the RCIC Steam Line prior to FCV 71-8.
	ATC/BOP	Reports alarm RCIC STEAM LINE LEAK DETECTION TEMP HIGH and rising temperatures in RCIC
	SRO	Directs RCIC Isolation verified
	ATC/BOP	Verifies RCIC automatically isolates.
		Attempt to align SLC per Appendix 7B. Recognize and report trip of both SLC Pumps.
		Report trip of CRD Pump 2A and inability to align CRD Pump 1B due to 2-85-8A will not open.
	CREW	Recognizes loss of all High Pressure Injection sources
	ATC/BOP	Report loss of 480 V RMOV Bd 2A / RMOV Bd 2E / RMOV Bd 2D
	CREW	Recognizes loss of all Injection sources
CT#3	SRO	**EOI-1 (cont)** Answers **NO** to: Can water level be Restored and Maintained above (+) 2 inches? Maintain RPV Water Level above (-) 162 inches. Directs ADS inhibited when RPV Water Level drops below -120 inches. Augments RPV Water Level Control with SLC, per Appendix 7B. Answers **NO** to: Can RPV Water Level be maintained above (-) 162 inches? Exits RC/L and enters C-1, "Alternate Level Control".
CT#3	ATC/BOP	Inhibits ADS

Simulator Event Guide:

Event 8 Major: Loss of LPCI MG sets, loss of ALL Level Control Systems – Steam Cooling

	SRO	**Enters C-1, Alternate Level Control** Verifies ADS Inhibited Directs lineup of Injection Systems Irrespective of Pump NPSH and Vortex limits (LPCI and CS) per Appendix 6B and 6D Answers **NO** to can 2 or more CNDS, LPCI or CS Injection Subsystems be aligned with pumps running When RPV Water Level drops to -162 inches, Then continues Answers **NO** to is any CNDS, LPCI or CS Injection Subsystem aligned with at least one pump running Before RPV Water Level drops to -180 inches continue Answers **NO** to are pumps running that can restore and maintain RPV Water Level above -180 inches after Emergency Depressurization When RPV Water Level drops to -180 inches continue Answers **NO** to is any CNDS Injection Source aligned with at least one pump running Steam Cooling is Required
Driver	Driver	Once steam cooling is entered insert F10 (dmf ed12a). Then close normal feeder breaker to RMOV Bd 2A insert F11 (mrf ed09 norm). Notify crew that RMOV Bd 2A is restored. Then insert F12 (mrf rp02 reset) to reset RPS B.
NOTE	NOTE	Restoration of RMOV Bd 2A makes Core Spray Loop I available.
CT#1	SRO	**C-1, Alternate Level Control** If any Injection Source aligned with at least one pump running and Reactor Level is < -180 inches continue Emergency Depressurization is required Enters C-2

Simulator Event Guide:

Event 8 Major: Loss of LPCI MG sets, loss of ALL Level Control Systems – Steam Cooling

CT#1	SRO	**C-1, Alternate Level Control** (Cont.) If RPV Water Level drops to -195 inches continue Emergency Depressurization is required Enters C-2 Directs maximizing RPV Injection from all available sources irrespective of pump NPSH and Vortex Limits Directs Emergency Depressurization before RPV Level reaches -215"
CT#1		**Enters C-2, Emergency RPV Depressurization** Answers **Yes** to will the Reactor remain subcritical without Boron under all conditions Answers **Yes** to is Drywell Pressure above 2.4 psig Does not prevent Injection from any Core Spray or LPCI pumps because they are all needed to assure adequate core cooling Answers **Yes** to is Suppression Pool Level above 5.5 feet Directs opening of all ADS Valves Answers **NO** to can 6 ADS Valves be opened Open additional MSRVs as necessary to establish 6 MSRVs Open Answers **YES** to are at least 4 MSRVs Open
CT#1	BOP/ATC	Open 5 ADS Valves and one additional SRV due to Inoperable ADS SRV
CT#2	BOP/ATC	With RPV pressure below the Shutoff Head of the available Low Pressure system(s), operate available Low Pressure system(s) to restore RPV water level above TAF.

Simulator Event Guide:

Event 8 Major: Loss of LPCI MG sets, loss of ALL Level Control Systems – Steam Cooling

	BOP/ATC	**Appendix 6D, Loop I Core Spray** 1. **VERIFY OPEN** the following valves: • 2-FCV-75-2, CORE SPRAY PUMP 2A SUPPR POOL SUCT VLV • 2-FCV-75-11, CORE SPRAY PUMP 2C SUPPR POOL SUCT VLV • 2-FCV-75-23, CORE SPRAY SYS I OUTBD INJECT VALVE. 2. **VERIFY CLOSED** 2-FCV-75-22, CORE SPRAY SYS I TEST VALVE. 3. **VERIFY** CS Pump 2A and/or 2C running. 4. WHEN ... RPV pressure is below 450 psig, THEN ... **THROTTLE** 2-FCV-75-25, CORE SPRAY SYS I INBD INJECT VALVE, as necessary to control injection at or below 4000 gpm per pump.
	SRO	**C-1, Alternate Level Control** (Cont.) Answers **Yes** to can RPV Water Level be restored and maintained above -180 inches Exits C-1 and enters EOI-1, RPV Control at step RC/L-1
	SRO	**Enters EOI-2 on High Drywell Pressure** **DW/T** Monitor and control Drywell temperature below 160F using available Drywell cooling Answers **No** to can Drywell Temperature be maintained below 160F Operate all available drywell cooling Before Drywell Temperature rises to 200F enter EOI-1 and Scram Reactor (this will already be complete at this time) Before Drywell Temperature rises to 280F continue Answers **Yes** to is Suppression Pool Level below 18 Feet Answers **Yes** to are Drywell Temperatures and Pressures within the safe area of curve 5 Directs Shutdown of Recirc Pumps and Drywell Blowers (should leave Drywell Blowers running due to being unable to spray because adequate core cooling is not assured)

Simulator Event Guide:

Event 8 Major: Loss of LPCI MG sets, loss of ALL Level Control Systems – Steam Cooling

	SRO	**Enters EOI-2 on High Drywell Pressure (cont)** **PC/P** Monitor and control Primary Containment pressure below 2.4 psig using the Vent System (Appendix 12) as necessary Direct Appendix 12
	ATC/BOP	Vent Containment IAW Appendix 12
		1. VERIFY at least one SGTS train in service. 2. VERIFY CLOSED the following valves (Panel 2-9-3 or Panel 2-9-54): 2-FCV-64-31, DRYWELL INBOARD ISOLATION VLV, 2-FCV-64-29, DRYWELL VENT INBD ISOL VALVE, 2-FCV-64-34, SUPPR CHBR INBOARD ISOLATION VLV, 2-FCV-64-32, SUPPR CHBR VENT INBD ISOL VALVE.
		Steps 3, 4, 5 and 6 are If / Then steps that do not apply
		7. **CONTINUE** in this procedure at: Step 8 to vent the Suppression Chamber through 2-FCV-84-19, **OR** Step 9 to vent the Suppression Chamber through 2-FCV-84-20.
		8. VENT the Suppression Chamber using 2-FIC-84-19, PATH B VENT FLOW CONT, as follows: a. PLACE keylock switch 2-HS-84-35, DW/SUPPR CHBR VENT ISOL BYP SELECT, to SUPPR-CHBR position (Panel 2-9-54). b. VERIFY OPEN 2-FCV-64-32, SUPPR CHBR VENT INBD ISOL VALVE (Panel 2-9-54). c. PLACE 2-FIC-84-19, PATH B VENT FLOW CONT, in AUTO with setpoint at 100 scfm (Panel 2-9-55). d. PLACE keylock switch 2-HS-84-19, 2-FCV-84-19 CONTROL, in OPEN (Panel 2-9-55). e. VERIFY 2-FIC-84-19, PATH B VENT FLOW CONT, is indicating approximately 100 scfm. f. CONTINUE in this procedure at step 12.

Simulator Event Guide:

Event 8 Major: Loss of LPCI MG sets, loss of ALL Level Control Systems – Steam Cooling

	BOP	Vents Primary Containment IAW Appendix 12
		9. VENT the Suppression Chamber using 2-FIC-84-20, PATH A VENT FLOW CONT, as follows: a. VERIFY OPEN 2-FCV-64-141, DRYWELL DP COMP BYPASS VALVE (Panel 2-9-3). b. PLACE keylock switch 2-HS-84-36, SUPPR CHBR/DW VENT ISOL BYP SELECT, to SUPPR-CHBR position (Panel 2-9-54). c. VERIFY OPEN 2-FCV-64-34, SUPPR CHBR INBOARD ISOLATION VLV (Panel 2-9-54). d. VERIFY 2-FIC-84-20, PATH A VENT FLOW CONT, in AUTO with setpoint at 100 scfm (Panel 2-9-55). e. PLACE keylock switch 2-HS-84-20, 2-FCV-84-20 ISOLATION BYPASS, in BYPASS (Panel 2-9-55). f. VERIFY 2-FIC-84-20, PATH A VENT FLOW CONT, is indicating approximately 100 scfm. g. CONTINUE in this procedure at step 12.
		12. ADJUST 2-FIC-84-19, PATH B VENT FLOW CONT, or 2-FIC-84-20, PATH A VENT FLOW CONT, as applicable, to maintain ALL of the following: Stable flow as indicated on controller, AND 2-PA-84-21, VENT PRESS TO SGT HIGH, alarm light extinguished, AND Release rates as determined below: iii. IF Venting for ANY other reason than items i or ii above, THEN MAINTAIN release rates below Stack release rate of 1.4 x 107 µCi/s AND 0-SI-4.8.B.1.a.1 release fraction of 1.
	DRIVER	Acknowledge Notification

Simulator Event Guide:

Event 8 Major: Loss of LPCI MG sets, loss of ALL Level Control Systems – Steam Cooling

	SRO	**Enters EOI-2 on High Drywell Pressure (cont)** **PC/P** Monitor and control Primary Containment pressure below 2.4 psig using the Vent System (Appendix 12) as necessary Direct Appendix 12 Answers **No** to can Primary Containment Pressure be maintained below 2.4 psig Before Suppression Chamber Pressure rises to 12 psig Initiate Suppression Chamber Sprays using only those pumps not required for Adequate Core Cooling Directs Drywell Spray
	ATC/BOP	Initiate Suppression Chamber Sprays per Appendix 17C
		1. BEFORE Suppression Chamber pressure drops below 0 psig, **CONTINUE** in this procedure at Step 6. 2. IF ..... Adequate core cooling is assured **OR** Directed to spray the Suppression Chamber irrespective of adequate core cooling, THEN ... **BYPASS** LPCI injection valve open interlock as necessary: • **PLACE** 2-HS-74-155A, LPCI SYS I OUTBD INJ VLV BYPASS SEL in **BYPASS.** • **PLACE** 2-HS-74-155B, LPCI SYS II OUTBD INJ VLV BYPASS SEL in **BYPASS.**
	NRC	EOI Program Manual 0-VIII-A 3.9 I. Use of Containment Cooling Modes While Executing C-1 Alternate Level Control. During execution of C-1 Alternate Level Control, if less than two Condensate, LPCI, CS Injection Subsystems can be aligned with pumps running per step C1-5, then available RHR injection subsystems *must* be aligned until the two subsystem requirement is met. Containment Cooling must be secured from those RHR subsystems that are aligned for injection. Step C1-5 does *not* count the number of *pumps* running. It counts the number of independent injection subsystem *paths* aligned that have at least one pump running, from the following five subsystems: Condensate, LPCI System I, LPCI System II, CS System I, CS System II. If at least two Condensate, LPCI, CS Injection Subsystems are aligned with pumps running in each per step C1-5, any RHR loop that is excess to the two required injection subsystems may be aligned for Containment Cooling mode.

Simulator Event Guide:

Event 8 Major: Loss of LPCI MG sets, loss of ALL Level Control Systems – Steam Cooling

	ATC/BOP	5. **INITIATE** Suppression Chamber Sprays as follows: a. **VERIFY** at least one RHRSW pump supplying each EECW header. b. IF.....EITHER of the following exists: • LPCI Initiation signal is NOT present, **OR** • Directed by SRO, THEN...**PLACE** keylock switch 2-XS-74-122(130), RHR SYS I(II) LPCI 2/3 CORE HEIGHT OVRD, in MANUAL OVERRIDE. c. **MOMENTARILY PLACE** 2-XS-74-121(129), RHR SYS I(II) CTMT SPRAY/CLG VLV SELECT, switch in SELECT. d. IF.....2-FCV-74-53(67), RHR SYS I(II) INBD INJECT VALVE, is OPEN, THEN...**VERIFY CLOSED** 2-FCV-74-52(66), RHR SYS I(II) OUTBD INJECT VALVE. e. **VERIFY OPERATING** the desired RHR System I(II) pump(s) for Suppression Chamber Spray. f. **VERIFY OPEN** 2-FCV-74-57(71), RHR SYS I(II) SUPPR CHBR/POOL ISOL VLV. g. **OPEN** 2-FCV-74-58(72), RHR SYS I(II) SUPPR CHBR SPRAY VALVE. h. IF.....RHR System I(II) is operating ONLY in Suppression Chamber Spray mode, THEN...**CONTINUE** in this procedure at Step 5.k. i. **VERIFY CLOSED** 2-FCV-74-7(30), RHR SYSTEM I(II) MIN FLOW VALVE. j. **RAISE** System flow by placing the second RHR System I(II) pump in service as necessary. k. **MONITOR** RHR Pump NPSH using Attachment 2.
	ATC/BOP	l. **VERIFY** RHRSW pump supplying desired RHR Heat Exchanger(s). m. **THROTTLE** the following in-service RHRSW outlet valves to obtain between 1350 and 4500 gpm flow: • 2-FCV-23-34, RHR HX 2A RHRSW OUTLET VLV • 2-FCV-23-46, RHR HX 2B RHRSW OUTLET VLV • 2-FCV-23-40, RHR HX 2C RHRSW OUTLET VLV • 2-FCV-23-52, RHR HX 2D RHRSW OUTLET VLV. n. **NOTIFY** Chemistry that RHRSW is aligned to in-service RHR Heat Exchangers.
	SRO	The Emergency classification is 1.1-G1

SHIFT TURNOVER SHEET

Equipment Out of Service/LCO's:

Core Spray Loop 2 is out of service and tagged out, Technical Specifications have been addressed

Operations/Maintenance for the Shift:

Start SBGT Fan C and align to Reactor Bldg IAW 0-OI-65 section 5.2

Once completed raise reactor power to 100% with Recirculation.

Units 1 and 3 are at 100% power.

Unusual Conditions/Problem Areas:

None

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT

Reactivity Maneuver Plan U2 NRC Exam 7

Raise Reactor Power to 100%

Attachment 7
(Page 1 of 2)

Reactivity Control Plan Form

BFN Unit: 2 Valid Date(s): 8/7/11 – 8/19/11 Reactivity Control Plan #: **U2 NRC Exam 7**

Are Multiple Activations Allowed: No (If yes, US may make additional copies)

Prepared by: ________________ / ______
Reactor Engineer Date

Reviewed by: ________________ / ______
Qualified Reactor Engineer Date

Approved by: ________________ / ______
RE Supervisor Date

Concurrence: ________________ / ______
WCC/Risk/US SRO Date

Approved by: ________________ / ______
Ops Manager or Supt. Date

Authorized by: ________________ / ______
Shift Manager Date

RCP Activated: ________________ / ______
Unit Supervisor Date

RCP Terminated: ________________ / ______
Unit Supervisor Date

Title of Evolution: Power increase with flow to 100%

Purpose/Overview of Evolution: Raise Power to 100%

Maneuver Steps

1. ***Increase flow to 100% power. No Ramp Rate Limits apply***

Attachment 7
(Page 2 of 2)

Reactivity Control Plan Form

Operating Experience and General Issues: U2 NRC Exam 7

- This plan is NOT valid if the unit is operating with a suspected or known fuel leaker and is not to be used. Contact Reactor Engineering if there are indications of a fuel leak.

Cautions/Error Likely Situations/Special Monitoring Requirements/Contingencies:

NONE

Attachment 8

Reactivity Maneuver Instructions

STEP 1 of 1 **Reactivity Maneuver Plan # U2 NRC Exam 7**

Description of Step: Power increase with flow to 100%. No Ramp Rate Limits apply

Conditions : To be recorded at the Completion of Step Recorded: ____________/_____
(by RO) (Date)

QRE presence required in the Control Room? Yes _____ No X (check)

	Predicted (may be ranges)	Actual		Predicted (may be ranges)	Actual
MW Electric	1050-1150		MFLCPR	.85 - .95	
MW Thermal	3300-3450		MAPRAT	.60 - .70	
Core Flow	85-94mlbm/hr		MFDLRX	.70 - .75	
Loadline	104-108				
Core Power	95-100%		Other		

Critical Parameters: To be recorded DURING Step. **IF** parameters are outside of the predictions, **THEN** discuss with the RE **AND** record conclusions in the Comments / Notes section.

Description including frequency, method of monitoring, **AND** contingency actions	High	Low
Maximum MWth	3458	

Comments / Notes:

1. Raise Reactor Power to 100% RTP
2. Document core flow changes on Attachment 10

Step Complete **AND** Reviewed by:_________________/________
Unit Supervisor / Date

Attachment 10
(Page 1 of 1)

Recirc Flow Maneuver Instructions

Reactivity Control Plan # U2 NRC Exam 7

RCP Step #	Flow Step #	Time	**Target Power (%RTP or MWe)**	Delta ±(MWe)	**Target Flow (MLb/Hr)**	Completed (RO)
1	**1**		100%			

Comments / Notes:

Reviewed by: ____________________/________
Unit Supervisor / Date

ATTACHMENT 2
(Page 1 of 2)

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	30-31	22	00		
N/A	22-23	08	00		
N/A	38-23	08	00		
N/A	38-39	08	00		
N/A	22-39	08	00		
N/A	30-15	48	00		
N/A	46-31	48	00		
N/A	30-47	48	00		
N/A	14-31	48	00		
N/A	14-23	48	00		
N/A	14-39	48	00		
N/A	46-39	48	00		
N/A	46-23	48	00		

REMARKS[4]: Emergency Shove Sheet – Loadline reduction or Unit Shutdown Insert Rods Continuously to 00. Insertion may stop after completion of any group.

NOTES:

(1) RWM Group may be marked "N/A" if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position "48", this signoff verifies coupling integrity was checked in accordance with 2-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by: ____________________ / __________ Issued by ____________________ / __________

Unit Supervisor   Date   Reactor Engineer   Date

ATTACHMENT 2

Date: Today

CONTROL ROD MOVEMENT DATA SHEET

RWM[1] GP	ROD NUMBER	FROM	TO	Rod Movement Completed INITIALS	
				UO(AC)[2]	2nd(AC) / Peer Check[3]
N/A	22-47	48	00		
N/A	38-47	48	00		
N/A	38-15	48	00		
N/A	22-15	48	00		
N/A	14-47	48	00		
N/A	46-47	48	00		
N/A	46-15	48	00		
N/A	14-15	48	00		
N/A	06-31	48	00		
N/A	30-55	48	00		
N/A	54-31	48	00		
N/A	30-07	48	00		
N/A	06-39	48	00		
N/A	54-39	48	00		
N/A	54-23	48	00		
N/A	06-23	48	00		

REMARKS[4]: Emergency Shove Sheet – Loadline reduction or Unit Shutdown Insert Rods Continuously to 00. Insertion may stop after completion of any group.

NOTES:

(1) RWM Group may be marked "N/A" if not applicable (i.e., when above the LPSP).

(2) For all rod moves to position "48", this signoff verifies coupling integrity was checked in accordance with 2-OI-85.

(3) Second-party verification by a second UO, RE, or STA is required ONLY when the RWM is inoperable or bypassed with core thermal power < 10%. A Peer Checker (not required in emergencies) may initial when second party is not required. "N/A" if not applicable.

(4) Record the rod number and any problems encountered, as applicable.

(5) Peer check by RE or SRO. The SRO should be checking the FROM and TO control rod positions as a minimum. The RE or SRO should be checking the positions identified for agreement with the predictor cases. Anytime the SRO feels the Peer check is beyond his knowledge level, then call in a second RE to perform the required Peer check.

Reviewed by: ____________________ / __________ Issued by ____________________ / __________
Unit Supervisor Date Reactor Engineer Date