Examination Outline Cross-reference:	Level	RO	SRO
203000G2.4.20	Tier #	2	
Knowledge of operational implications of EOP warnings, cautions and notes. RHR/LPCI: Injection Mode	Group #	1	
	K/A #	203000G2.4.20	
	Importance Rating	3.8	4.3

Proposed Question: **# 1**

A loss of all off-site power and main steam line break have occurred on Unit 3 resulting in the following plant conditions:

- RPV pressure — 50 psig and steady.
- RPV level — (-) 100 inches and steady, being maintained by RHR Loop I at 22,000 gpm.
- Drywell pressure — 19 psig and rising.
- Drywell temperature — 225 ^{O}F and rising.
- Suppression Pool Temp — 205 ^{O}F and rising.
- Both loops of Core Spray are unavailable.

Which ONE of the following describes the appropriate action to take and the basis for that action?

Initiating Drywell Sprays with RHR Loop II is ______(1)______ because ______(2)______.

	(1)	(2)
A.	inappropriate	RHR Loop II must be aligned for LPCI injection to ensure adequate core cooling.
B.	inappropriate	RHR Loop II must be aligned for LPCI injection UNTIL RPV level is restored to between (+) 2 and (+) 51 inches.
C.	appropriate	RHR Loop II is not required for LPCI injection to ensure adequate core cooling.
D.	appropriate	this will ensure adequate NPSH for RHR Loop I under the current plant conditions.

Proposed Answer: **C**

Explanation :

a. Part (1) is incorrect. Initiating DW Spray IS appropriate. It can be interpolated that a DW pressure of 19 psig corresponds to a Suppression Chamber pressure above 12 psig, (~5 psid) which requires DW Sprays. In addition, due to the relatively small ΔT between SP temp and DW temp, sprays will not reduce containment pressure low enough to exceed Curve #2 NPSH limits. This choice becomes plausible due to the wording of Caution #4, which warns of the potential for exceeding NPSH limits with a reduction in containment pressure. Only a detailed analysis of the containment conditions, based on fundamental heat transfer dynamics, can eliminate that potential. Part (2) is incorrect. As long as RPV level can be maintained above (-) 162 inches, RHR Loop II can be aligned for containment control.

b. Part (1) is incorrect. Part (2) is incorrect. As long as RPV level can be maintained above (-) 162 inches, RHR Loop II can be aligned for containment control. Although the required level band is +2 to +51 inches, as long as adequate core cooling is assured based on current conditions, using RHR for containment control is allowed.

c. Correct answer.

d. Part (1) is correct as stated in (a) above. Part (2) is incorrect. Initiating Drywell Sprays under this condition will actually reduce NPSH to the running RHR pumps, however as stated in (a) above, the small ΔT between SP temp and DW temperature would prevent containment pressure from dropping low enough to challenge NPSH limits.

Technical Reference(s):	3-EOI-1 and 3-EOI-2 flowcharts	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: 3-EOI-2 Flowchart

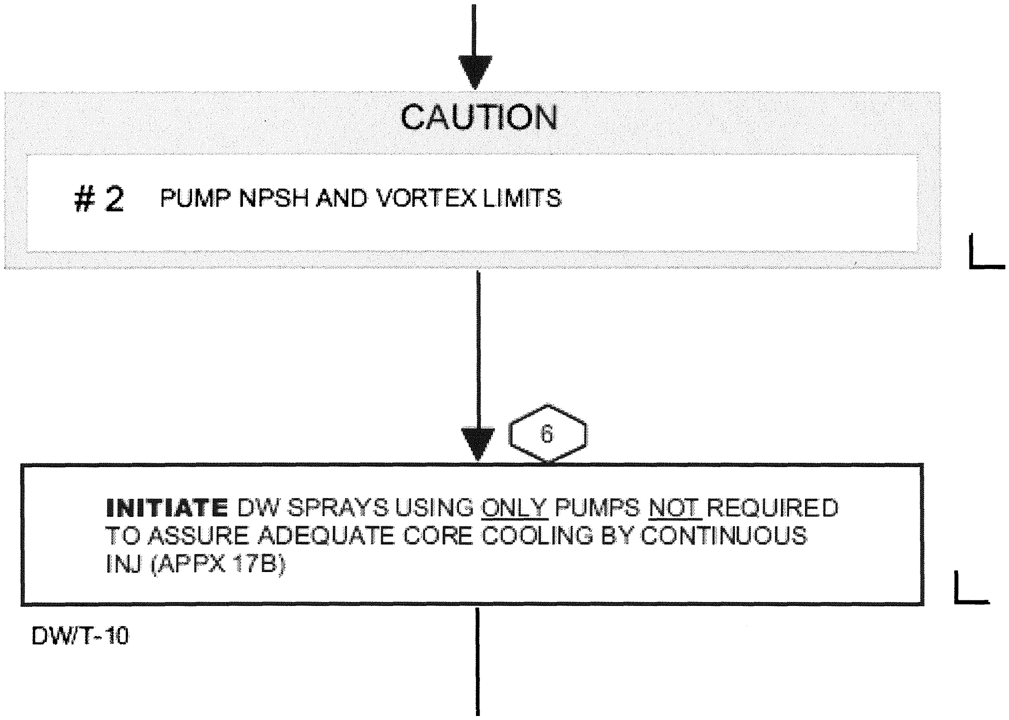
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	RMS 6/16/2008	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

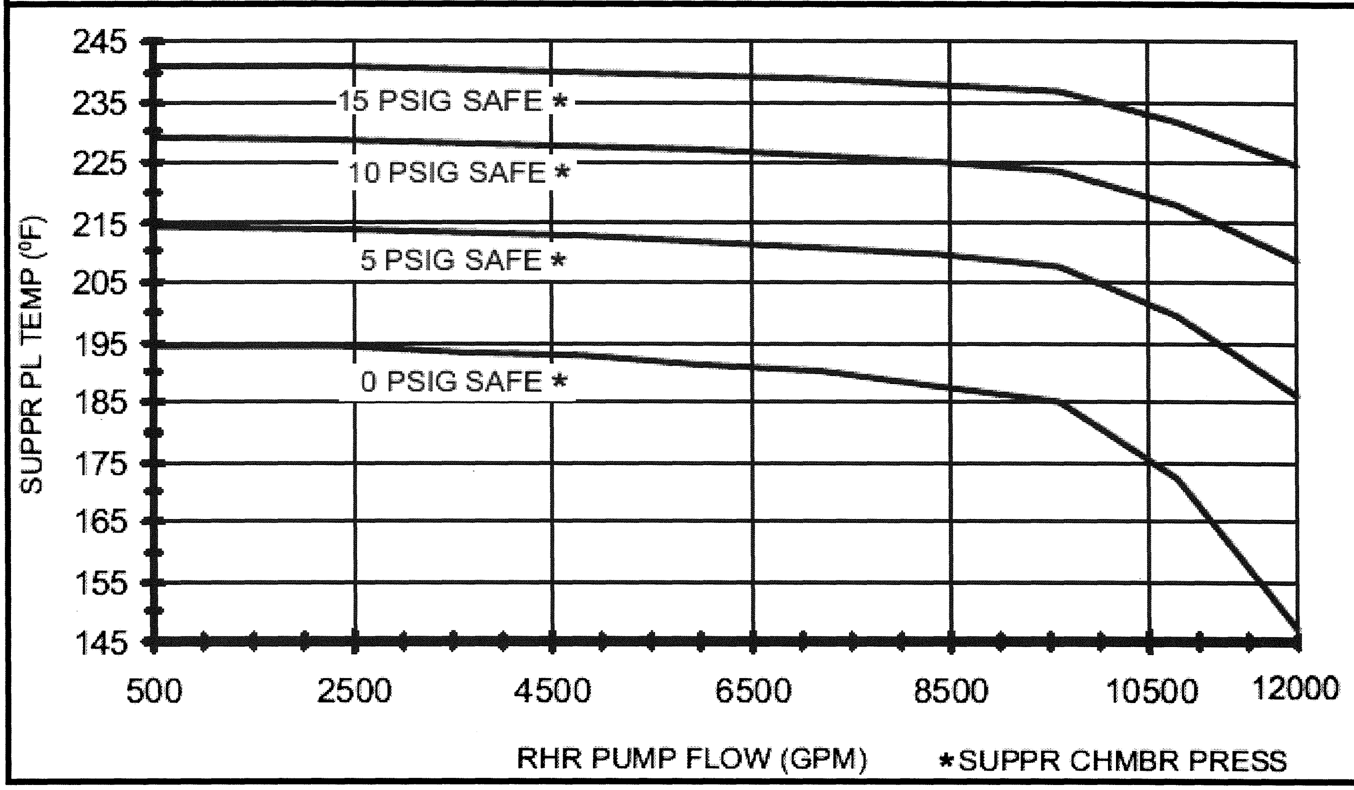
10 CFR Part 55 Content:	55.41	X
	55.43	

Comments: This has been evaluated as C/A due to the multiple plant conditions which must be evaluated against procedural requirements. The K/A is met by structuring the question to ensure RHR LPCI injection Mode is considered and maintained while performing other actions in accordance with EOIs.

Excerpt from 3-EOI-2 flowchart path DW/T:



CURVE 2
RHR NPSH LIMITS



CAUTIONS

CAUTION #2

OPERATION OF RHR OR CS WITH SUCTION FROM THE SUPPR PL MAY RESULT IN EQUIPMENT DAMAGE IF:

- PUMP FLOW IS ABOVE THE NPSH LIMIT (CURVE 1 OR 2)

  OR
- SUPPR PL LVL IS BELOW THE VORTEX LIMIT (10 FT).

CAUTION #4

REDUCING PC PRESS WILL REDUCE THE AVAILABLE NPSH FOR PUMPS TAKING SUCTION FROM THE SUPPR PL

Examination Outline Cross-reference:	Level	RO	SRO
203000G2.4.34	Tier #	2	
Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects: RHR/LPCI Injection Mode.	Group #	1	
	K/A #	203000G2.4.34	
	Importance Rating	4.2	4.1

Proposed Question: **RO # 2**

Given the following Unit 1 plant conditions:

- Unit 1 and 2 control rooms have been abandoned due to a toxic gas release.
- Control has been established at Backup Control Panel 1-25-32.
- RHR Loop II is in Suppression Pool Cooling with both RHR pumps running.
- Reactor Pressure is 110 psig and lowering due to the cooldown.
- Reactor water level is (-) 48 inches and lowering slowly.
- The Unit Supervisor has directed that RHR be lined up for LPCI injection in accordance with 1-AOI-100-2, "Control Room Abandonment."

Which ONE of the following describes the location where this lineup is performed and the method of monitoring injection flow?

Operating the LPCI Injection valves can be accomplished from _____(1)_____ and injection flow is monitored by _____(2)_____.

	(1)	(2)
A.	480V RMOV Board 1B	RHR Total Flow indication from Panel 1-25-32
B.	480V RMOV Board 1B	RHR pump amps from 4KV Shutdown Board C
C.	Backup Control Panel 1-25-32	RHR Total Flow indication from Panel 1-25-32
D.	Backup Control Panel 1-25-32	RHR pump amps from 4KV Shutdown Board C

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Part (1) is correct. Part (2) is incorrect. Pump amperage is used to verify RHRSW pump flow during operation from 1-25-32, not RHR pump flow.

c. Part (1) is incorrect. RHR injection valves cannot be controlled from 1-25-32. Only RCIC injection flow can be controlled from 1-25-32. Part (2) is correct.

d. Part (1) is incorrect as stated in (c) above. Part (2) is incorrect. Pump amperage is used to verify RHRSW pump flow during operation from 1-25-32, not RHR pump flow.

Technical Reference(s): 1-AOI-100-2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/07/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 1	Control Room Abandonment	1-AOI-100-2 Rev. 0016 Page 14 of 79

4.2 Unit 1 Subsequent Actions (continued)

- U-2 Unit Operator complete Attachment 7 ☐

[13] Upon completion of attachments, **RE-ESTABLISH** communication using the best available means and AWAIT further instructions. ☐

[14] **IF** CRD Pump 1B is to be aligned to Unit 1, **THEN**

PERFORM the following:

[14.1] **OPEN** 1-FCV-085-0008 using CRD PUMP 1B UNIT 1 DISCHARGE, 1-HS-085-0008C at 480V RMOV Bd 1C Compt 3B. ☐

[14.2] **PLACE** CRD PUMP 1B, 1-HS-085-0002C in CLOSE at 4KV SD Bd A, Compt. 13 to start CRD Pump 1B. ☐

[15] **INITIATE** RHR Suppression Pool Cooling as follows:

NOTE
Communication between 4160V Shutdown Board C and 480V RMOV Bd 1B is necessary for establishing RHRSW flow and to prevent exceeding 53 amps on RHRSW Pump B2.

[15.1] **PLACE** RHRSW PUMP B2 MOTOR, 0-HS-023-0019C in CLOSE at 4160V Shutdown Bd C, Compt. 16 to start RHRSW Pump B2. ☐

[15.2] **THROTTLE OPEN** RHR HX 1B RHRSW OUTLET VLV, using 1-HS-023-0046C at 480V RMOV Bd 1B, Compt. 14C2. ☐

[15.3] **WHEN** between 48 and 52 amps on RHR SERVICE WATER PUMP B2, **THEN**

STOP throttling, RHR HX 1B RHRSW OUTLET VLV, 1-HS-023-0046C. ☐

BFN Unit 1	Control Room Abandonment	1-AOI-100-2 Rev. 0016 Page 17 of 79

4.2 Unit 1 Subsequent Actions (continued)

[16] **MAINTAIN** Drywell temperature less than 160°F as follows:

[16.1] **MONITOR** DRYWELL AIR TEMPERATURE, 1-TIS-64-52AA at Panel 1-25-32, ☐

[16.2] **OPERATE** Drywell Blowers as required. ☐

Drywell Blower	Switch No.	Compt. No.	Switch Position	
	480V Shutdown Bd 1A			
1A-1	DW CLG UNIT 1A1 BLOWER, 1-HS-070-0037C	2C	CLOSE	☐
1A-2	DW CLG UNIT 1A2 BLOWER, 1-HS-070-0038C	2D	CLOSE	☐
	480V Shutdown Bd 1B			
1B-1	DW CLG UNIT 1B1 BLOWER, 1-HS-070-0042C	2C	CLOSE	☐
1B-2	DW CLG UNIT 1B2 BLOWER, 1-HS-070-0043C	2D	CLOSE	☐
	480V RMOV Bd 1A			
1A-3	DW CLG UNIT 1A3 BLOWER, 1-HS-070-0039C	17A	START	☐
1A-4	DW CLG UNIT 1A4 BLOWER, 1-HS-070-0040C	18A	START	☐
	1B-3 1B-4 COOLERS started by AUO in Attachment 4 Part B.			
	480V RMOV Bd 1C			
1A-5	DW CLG UNIT 1A5 BLOWER, 1-HS-070-0041C	1A	START	☐
1B-5	DW CLG UNIT 1B5 BLOWER, 1-HS-070-0046C	11A	START	☐

[17] **IF** Reactor makeup from RHR LPCI is desired **AND** RHR Pumps are operating in Suppression Pool Cooling, **THEN**

ESTABLISH RHR system flow to the Reactor as follows:(Otherwise N/A) ☐

[17.1] **MONITOR** RHR SYS II TOTAL FLOW, 1-FI-74-79 at Panel 1-25-32. ☐

BFN Unit 1	Control Room Abandonment	1-AOI-100-2 Rev. 0016 Page 18 of 79

4.2 **Unit 1 Subsequent Actions (continued)**

[17.2] **CLOSE** RHR SYS II SUPPR POOL CLG/TEST VLV using 1-HS-074-0073C at 480V RMOV Bd 1B, Compt. R11C. ☐

[17.3] **OPEN** RHR SYS II LPCI INBD INJECT VLV, using 1-HS-074-0067C at 480V RMOV Bd 1B, Compt. R10A. ☐

[17.4] **THROTTLE OPEN** RHR SYS II LPCI OUTBD INJECT VLV, using 1-HS-074-0066C at 480V RMOV Bd 1B, Compt. 3A, as necessary to maintain Reactor Water Level between +2 and +50 inches. ☐

NOTE
Step 4.2[18] will prepare the condensate system for injection in preparation for tripping RCIC when pressure is less than 50 psig.

[18] **IF** Reactor makeup from Condensate System is desired, **THEN**

PERFORM the following: (Otherwise N/A) ☐

[18.1] **DISPATCH** an operator to Unit 1 Turb Bldg El 617' at RFW START-UP LCV, 1-LCV-003-0053, and **ESTABLISH** communications. ☐

[18.2] **MONITOR** RX WATER LEVEL A , 1-LI-3-46A and RX WATER LEVEL B, 1-LI-3-46B, at Panel 1-25-32, ☐

[18.3] **DIRECT** operator to **THROTTLE** RFW START-UP LCV-3-53 BYPASS, 1-BYV-003-0053, as necessary to maintain Reactor Water Level between +2 and +50 inches. ☐

Examination Outline Cross-reference:	Level	RO	SRO
205000A3.02	Tier #	2	
Ability to monitor automatic operation of Shutdown Cooling including: Pump trips.	Group #	1	
	K/A #	205000A3.02	
	Importance Rating	3.2	3.2

Proposed Question: **RO # 3**

Given the following Unit 1 plant conditions:

- RHR Loop I is in Shutdown Cooling Mode with 1A and 1C RHR Pumps running.
- Unit 1 is in Mode 4 at 185 ^{O}F and lowering slowly.
- RHR SYSTEM I MIN FLOW INHIBIT switch, 1-HS-74-148 is in INHIBIT.
- RHR SYS I LPCI INBD INJECT VALVE, 1-FCV-74-53 is fully open.
- RHR SYS I LPCI OUTBD INJECT VALVE, 1-FCV-74-52 is throttled open.
- 480V RMOV Board B de-energizes due to an electrical fault.

Which ONE of the following describes the status of RHR Loop I following the loss of 480V RMOV Board B and the final position of the RHR Inboard Injection valve 1-FCV-74-53?

RHR Pumps 1A and 1C ________(1)__________. The RHR SYS I LPCI INBD INJECT VALVE, 1-FCV-74-53 __________(2)__________.

	(1)	(2)
A.	trip on loss of a suction path	closes due to Group II logic signal
B.	trip on loss of a suction path	fails open on loss of power
C.	remain in operation	closes due to Group II logic signal
D.	remain in operation	fails open on loss of power

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Part (1) is correct. The candidate must determine that 480V RMOV Bd B supplies RPS B which, when lost, causes a Group II isolation of the RHR Shutdown Cooling suction valves. Part (2) is incorrect. RHR Loop II inboard injection valve is powered from 480V RMOV Bd B, so the Loop I valve (480V RMOV Bd A) still has power to automatically close.

c. Part (1) is incorrect. When RPS B is lost due to the 480V RMOV Bd B loss, the Outboard Shutdown Cooling suction valve closes which causes a loss of suction path to BOTH RHR pumps. Part (2) is correct. The RHR Loop I inboard valve is powered from 480V RMOV Bd A so will have power to automatically close when the Group II logic signal is received.

d. Part (1) is incorrect. When RPS B is lost due to the 480V RMOV Bd B loss, the Outboard Shutdown Cooling suction valve closes which causes a loss of suction path to BOTH RHR pumps. Part (2) is incorrect. RHR Loop II inboard injection valve is powered from 480V RMOV Bd B, so the Loop I valve (480V RMOV Bd A) still has power to automatically close.

Technical Reference(s): 1-OI-74, 1-OI-99, Illustration 1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/08/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 1	Residual Heat Removal System	1-OI-74 Rev. 0055 Page 15 of 260

3.0 PRECAUTIONS AND LIMITATIONS (continued)

4. If Unit 1 reactor pressure exceeds 100 psig or a Group II isolation occurs on Unit 1 while Shutdown Cooling is in operation, the following will occur for the given condition:

   a. (100 psig) RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 1-FCV-74-47 and 1-FCV-74-48, close, thus tripping operating Unit 1 RHR Pumps.

   b. (Group II) RHR SYS I and II LPCI INBD INJECT VALVEs, 1-FCV-74-53 and 1-FCV-74-67, close and Unit 1 RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 1-FCV-74-47 and 1-FCV-74-48, close, thus tripping operating Unit 1 RHR Pumps.

5. To reopen RHR SYS I(II) LPCI INBD INJECT VALVE, 1-FCV-74-53(67), after a loss of Shutdown Cooling from one of the above conditions, RHR SYS I(II) SD CLG INBD INJECT(INJ) ISOL RESET, 1-XS-74-126(132) push-button must be depressed after either of the following occur:

   a. Isolation signal has been reset

   b. 1-FCV-74-47 or 1-FCV-74-48 are fully closed.

6. If after a GROUP II Isolation, RHR SYS I(II) LPCI INBD INJECT VALVE, 1-FCV-74-53(67) is given an OPEN signal prior to depressing the RHR SYS I(II) SD CLG INBD INJECT(INJ) ISOL RESET, 1-XS-74-126(132) then the valve will travel full open and full close unless given a close signal prior to traveling full open.

7. The RHR spray/cooling valves, 1-FCV-74-57(71), will receive an auto closure signal in the presence of a LPCI initiation signal and they are interlocked to prevent opening if the in-line torus spray valve, 1-FCV-74-58(72), is NOT fully closed. The in-line valve interlock can be by-passed if the following conditions exist.

   a. Reactor level is >2/3 core height

   b. LPCI initiation signal is present

   c. Select reset switch is in the SELECT position.

   The requirements for >2/3 core height and a LPCI initiation signal may be by-passed using the keylock bypass switch, 1-XS-74-122/130.

8. If primary containment cooling is desired with reactor level at <2/3 core height, the keylock bypass switch must be placed in BYPASS before the select reset switch is placed in SELECT to ensure relay logic is made up.

BFN Unit 1	Loss of Power to One RPS Bus	1-AOI-99-1 Rev. 0017 Page 5 of 9

3.0 AUTOMATIC ACTIONS

NOTES
An overview of the automatic actions is provided here. A detailed list of the actions is provided in 1-OI-99, Illustration 1 which lists the actions that occur when RPS buses are de-energized on a transfer of power supply.

A. An RPS trip logic A(B) half scram occurs.

B. One-half of the PCIS Group 1 trip logic is de-energized.

C. Inboard (outboard) isolation of PCIS Group 2, shutdown cooling mode of RHR.

D. Inboard and outboard isolation of PCIS Group 3, RWCU on loss of RPS A and outboard isolation on loss of RPS B.

E. Inboard and outboard isolation of PCIS Group 6, primary containment vent and purge, reactor building ventilation.

F. Group 8, TIP.

G. Control Bay Emergency Pressurization System A & B start.

H. Standby Gas Treatment System starts.

4.0 OPERATOR ACTIONS

4.1 Immediate Action

[1] **STOP** all testing that could cause RPS half scrams or PCIS Logic isolation signals.

BFN Unit 1	Reactor Protection System	1-OI-99 Rev. 0033 Page 50 of 68

Illustration 1
(Page 4 of 4)

RPS Bus A or B Power Transfer

C. Loss of power to RPS Bus B only will result in the following events in addition to those listed for RPS Bus A or B power loss

VALVE	FUNCTION/SYSTEM	ACTION
1-FCV-074-0047	RHR SD CLG SUCT OUTBD ISOL VLV	CLOSES
1-FCV-074-0067	RHR SYS II LPCI INBD INJECT VLV	CLOSES
1-FCV-075-0058	PSC PUMP SUCTION OUTBD ISOL	CLOSES
1-FCV-077-0015B	DRYWELL EQ DR SUMP OUTBD FCV	CLOSES
1-FCV-077-0002B	DRWELL FD SUMP OUTBD ISOLATION VLV	CLOSES
1-FCV-069-0002	RWCU OUTBD SUCT ISOL VLV	CLOSES
1-FCV-069-0012	RWCU SYS RETURN ISOL VLV	CLOSES
1-FCV-001-0015	MAIN STEAM LINE A OUTBD ISOL VLV AC control power	DE-ENERGIZES
1-FCV-001-0027	MAIN STEAM LINE B OUTBD ISOL VLV AC control power	DE-ENERGIZES
1-FCV-001-0038	MAIN STEAM LINE C OUTBD ISOL VLV AC control power	DE-ENERGIZES
1-FCV-001-0052	MAIN STEAM LINE D OUTBD ISOL VLV AC control power	DE-ENERGIZES
1-FCV-001-0056	MAIN STEAM LINE DRAIN OUTBD ISOL VLV	CLOSES
1-FCV-043-0014	REACTOR RECIRC OUTBD ISOLATION VLV	CLOSES

Examination Outline Cross-reference:

205000A3.03

Ability to monitor automatic operation of Shutdown Cooling including: Lights and alarms.

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	205000A3.03	
Importance Rating	3.5	3.3

Proposed Question: **RO # 4**

Given the following plant conditions:

- Unit 2 is aligned with RHR Loop I in Shutdown Cooling and RHR Loop II in standby readiness.

- A leak occurs in the RPV, which results in the following conditions:

  - RPV level at 0 inches and slowly lowering
  - Drywell Pressure at 3.0 psig and slowly rising
  - RHR Pumps 'A' and 'C' TRIPPED

Which ONE of the following describes the **minimum** actions required to align RHR Loop II for injection to the RPV?

A. After FCV-74-47 OR FCV-74-48 is closed, push the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132.

B. After FCV-74-47 AND FCV-74-48 are closed, start RHR Loop II pumps, reset PCIS, and open the inboard injection valve.

C. After FCV-74-47 OR FCV-74-48 is closed; reset PCIS, push the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132, and open the inboard injection valve.

D. After FCV-74-47 AND FCV-74-48 are closed, reset PCIS, push the RHR SYS II SD CLG INBD INJECT ISOL RESET 2-XS-74-132, and open BOTH injection valves.

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. The valve alignment is correct and resetting 2-XS-74-132 is correct. However, resetting PCIS, re-opening FCV 74-47 and re-starting RHR pumps are NOT required.

c. The valve alignment is incorrect. Both 74-47 and 74-48 must be closed. Resetting 2-XS-74-132 is correct. However, resetting PCIS is not required to re-start RHR pumps.

d. The valve alignment is correct and resetting 2-XS-74-132 is correct. However, resetting PCIS is not required to re-start RHR pumps and the injection valves will open automatically.

Technical Reference(s):	2-OI-74	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #	RO 295021G2.4.50	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	3/25/2008	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

BFN Unit 2	Residual Heat Removal System	2-OI-74 Rev. 0133 Page 22 of 367

3.5 **INTERLOCKS (continued)**

5. The RHR outboard LPCI injection valves, 2-FCV-74-52(66), have throttling capability. They receive an auto open signal in the presence of a LPCI initiation signal when Reactor pressure is ≤450 psig and are interlocked open under these conditions for 5 minutes, or until the appropriate LPCI SYS I (SYS II) OUTBD INJ VLV BYPASS SEL keylock Switch, 2-HS-74-155A(155B), is placed in the BYPASS position. Additionally these valves are interlocked to prevent opening when reactor pressure is >450 psig if its in-line companion valve 2-FCV-74-53(67) is not fully closed.

6. If Unit 2 reactor pressure exceeds 100 psig or a Group II isolation occurs on Unit 2 while Shutdown Cooling is in operation, the following will occur for the given condition:

(100 psig)	RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 and 2-FCV-74-48, close, thus tripping operating Unit 2 RHR Pumps.
(Group II)	RHR SYS I and II LPCI INBD INJECT VALVEs, 2-FCV-74-53 and 2-FCV-74-67, close and Unit 2 RHR SHUTDOWN COOLING SUCT OUTBD and INBD ISOL VLVs, 2-FCV-74-47 and 2-FCV-74-48, close, thus tripping operating Unit 2 RHR Pumps.

If RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67) is giving an OPEN signal prior to resetting the RHR SYS I(II) SD CLG INBD INJECT ISOL, after a GROUP II Isolation. The valve travels full open and full close unless given a close signal prior to traveling full open.

To reopen RHR SYS I(II) LPCI INBD INJECT VLV, 2-FCV-74-53(67), after a loss of Shutdown Cooling from one of the above conditions, RHR SYS I(II) SD CLG INBD INJECT ISOL RESET pushbutton is required to be depressed after either of following occur:

(1) Isolation signal has been reset OR

(2) 2-FCV-74-47 or 2-FCV-74-48 is fully closed.

Examination Outline Cross-reference:	Level	RO	SRO
206000A2.04	Tier #	2	
Ability to (a) predict the impacts of the following on the HPCI system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. Failures	Group #	1	
	K/A #	203000A2.04	
	Importance Rating	2.7	3.0

Proposed Question: **# 5**

Unit 2 is at 100% rated power when the following annunciators are received:

- "HPCI 120 VAC POWER FAILURE" (9-3F W7),
- "HPCI LOGIC POWER FAILURE" (9-3F W3).

Which ONE of the following describes the current HPCI status and the action required to return HPCI to a normal standby lineup?

The HPCI system ____(1)___ initiate and inject if required. ____________(2)____________ to restore HPCI to a normal standby lineup.

	(1)	(2)
A.	will	Restart Div II ECCS inverter per 2-AOI-57-11, "Loss of Power to an ECCS ATU Panel/ECCS Inverter."
B.	will	Transfer 250V RMOV Board A to ALTERNATE per 0-OI-57D, "DC Electrical System."
C.	will NOT	Restart Div II ECCS inverter per 2-AOI-57-11, "Loss of Power to an ECCS ATU Panel/ECCS Inverter."
D.	will NOT	Transfer 250V RMOV Board A to ALTERNATE per 0-OI-57D, "DC Electrical System."

Proposed Answer: **D**

Explanation :

a. HPCI will not initiate or inject with a loss of 120V power. In addition, the logic power failure annunciator would not occur is only the ECCS inverter was de-energized.

b. HPCI will not initiate or inject with a loss of 120V power. However, the transfer of the 250V RMOV Board to alternate is the correct action to take.

c. Past (1) is correct, however the logic power failure annunciator would not occur is only the ECCS inverter was de-energized.

d. correct answer

Technical Reference(s):	2-AOI-57-11	(Attach if not previously provided)
	0-OI-57D	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	RMS 6/20/2008	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

OPL171.042 Page 44

a. **The Division II ECCS ATU inverter (powered from 250V RMOV Board A) supplies power to the flow controller and the following instruments on Panel 9-3:	**SOER 83-3 (Recommendation 11) Obj. V.B.6
(1) PI-73-31A, Pump Discharge	Note that these loads are supplied from the HPCI Inverter on U-1
(2) PI-73-28A, Booster Pump Suction	
(3) PI-73-4A, Steam Supply	
(4) PI-73-21A, Turbine Exhaust	
(5) FIC-73-33, Flow Ind Controller	Obj. V.B.6 Obj. V.C.6
(6) If Division II ECCS inverter output is lost, HPCI 120VAC FAILURE (XA-55-3F-7) would alarm and flow controller fails downscale, Control valve closes if open. If accompanied by HPCI LOGIC POWER FAILURE (XA-55-3F-3), would indicate a loss of power from 250V RMOV Board A.	Note: on U-1 may indicate a loss of logic bus B
(7) If Division I ECCS inverter and converter output is lost, HPCI will not initiate from DIV I logic. LIS-3-58A and B will be lost.	
b. Relay Logic Bus A (Div I) is powered from 250V RMOV Board 2B. It supplies power to half of the low level circuit. It also supplies isolation logic Channel A (Div I). If lost, HPCI can still initiate and isolate on all signals.	XA-55-3F(3)

8.7 Transfer of Power Supplies to 250V Reactor MOV Boards

[1] **REVIEW** all Precautions and Limitations in Section 3.0. ☐

CAUTION

When any unit 250VDC RMOV Board A or B is transferred to alternate power supply, it is possible that a transfer of EHC from Reactor Pressure Control to Header Pressure Control to occur due to a los of power to 2 of the 4 reactor press instruments. [PER 109297]

NOTES

1) Tripping of the normal or alternate feeder breakers to 250V Reactor MOV Boards on overcurrent results in lockout of both breakers.
2) Both normal and alternate feeder breakers to a 250V Reactor MOV Board are electrically interlocked to prevent simultaneous closure (paralleling) of DC sources.
3) The normal and alternate feeder breakers are located on the 250V Reactor MOV Board which they supply.
4) Trip Test push-buttons are used only for testing racked out normal and alternate feeder breakers.
5) Transfer requires two operators due to the distance between the normal and alternate feeder breakers.
6) Prior to transferring any 250VDC RMOV Board to the alternate supply, Precaution and Limitation 3.0A must be complied with.
7) Transfer of 250V RMOV BD 3A will cause annunciation of the following alarms:
   - 3-XA-55-3C, Window 1, RCIC RELAY LOGIC POWER FAILURE
   - 3-XA-55-3C, Window 32, ADS BLOWNDOWN POWER FAILURE
   - 3-XA-55-3E, Window 23, 480V RX MOV BD D BACKUP SW IN EMER POSN
   - 3-XA-55-3F, Window 28, HPCI GLAND SEAL CONDENSER HOTWELL LEVEL LOW
   - 3-XA-55-5B, Window 34, PNL 9-47 FUSE FAILURE
   - 3-XA-55-4B, Window 22, 4160V RPT BD 3-II CONTROL ABNORMAL

[2] **CHECK** power availability of the EMERGENCY (NORMAL) supply breaker as follows:

A. Voltmeter indicates greater than or equal to 250 volts. ☐

B. Voltage Relay is reset. ☐

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0114 Page 194 of 241

8.7 **Transfer of Power Supplies to 250V Reactor MOV Boards (continued)**

[3] **PLACE** NORM/EMERG TRANSFER SWITCH in the ALT(NOR) position. ☐

[4] **HOLD** the EMERGENCY(Normal) supply BREAKER CONTROL SWITCH in CLOSE. ☐

[5] **TRIP** the Normal(Emergency) supply BREAKER CONTROL SWITCH. ☐

[6] **CHECK** Emergency(Normal) SUPPLY BREAKER is CLOSED. ☐

[7] **CHECK** NORMAL(EMERGENCY) SUPPLY BREAKER is OPEN/TRIP. ☐

[8] **RELEASE** the Breaker Control Switches. ☐

[9] **CHECK** control room panels for any abnormal indications. **REFER TO** Caution and Notes beginning this section. ☐

BFN Unit 2	Loss of Power to an ECCS ATU Panel/ECCS Inverter	2-AOI-57-11 Rev. 0008 Page 4 of 31

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions, operator actions, technical specification requirements, and reportability requirements resulting from a loss of power to ECCS ATU Panel 9-81 or 9-82 or loss of an ECCS inverter.

NOTES

1) Each Inverter provides electrical power to divisional logics plus one of the two redundant power supplies to its divisional ATU cabinet (**REFER TO** Illustration 5). The power supplies to ECCS ATU Panel 9-81 (Div. I), ECCS ATU Panel 9-82 (Div. II) and the ECCS inverters are as follows:

<u>Panel 9-81</u>

- Division I ECCS inverter — 250V RMOV Board 2B, compartment 8A.
- Division I 250/24vdc converter — 250V RMOV Board 2B, compartment 1B1.

<u>Panel 9-82</u>

- Division II ECCS inverter — 250V RMOV Board 2A, compartment 11A1
- Division II 250/24vdc converter — 250V RMOV Board 2A, compartment 9A1.

Power will be lost to an ECCS ATU Panel due to the loss of the respective 250V RMOV board listed above, opening/loss of both of the breakers listed above, loss of ECCS ATU Panel internal fuses, or simultaneous loss of both redundant 24vdc power supplies in each ECCS ATU panel.

2) The total loss of power to the ECCS ATU panels results in power loss to all instrumentation on:

- Division I — Panel 9-81 (Aux Instrument Room)
- Division II — Panel 9-82 (Aux Instrument Room)

RHR system I(II) containment spray valve operation will require the use of the manual override switch 2-XS-74-122(130) upon a loss of Panel 9-81(82) due to a loss of the two-thirds core height level channel.

Examination Outline Cross-reference:	Level	RO	SRO
209001K1.13	Tier #	2	
Knowledge of the physical connections and/or cause-effect relationships between Core Spray and the following: Leak detection.	Group #	1	
	K/A #	209001K1.13	
	Importance Rating	2.8	3.0

Proposed Question: **RO # 6**

Unit-1 is operating at 100% rated power when CORE SPRAY SYS I SPARGER BREAK (9-3C W14) alarms on Panel 9-3.

Which ONE of the following describes the principle of operation of the Core Spray Leak Detection instrument due to a Core Spray pipe break between the RPV wall and the core shroud?

The pressure sensed in the Core Spray pipe will be ____(1)____ causing a ____(2)____ ΔP to be sensed by the Leak Detection ΔP transmitter.

	(1)	(2)
A.	higher	lower
B.	higher	higher
C.	lower	lower
D.	lower	higher

Proposed Answer: **C**

Explanation:

a. Part (1) is incorrect. The reference leg of the ΔP transmitted normally senses pressure inside the core shroud below the steam separators and compares it to the pressure just above the core plate. If the Core Spray pipe breaks between the RPV wall and the core shroud, the ΔP transmitter reference leg will now sense pressure in the steam dome above the steam dryers, which is ~ 7 psig lower due to the pressure drop across the dryers and separators. Part (2) is correct. The reference leg pressure would lower, which would reduce the ΔP sensed by the ΔP transmitter.

b. Part (1) is incorrect as stated in (a) above. Part (2) is incorrect. However, this would be consistent with Part (1) if the break occurred in the variable leg rather than the reference leg of the ΔP transmitter.

c. Correct answer.

d. Part (1) is correct. Reference leg pressure lowers due to the break. Part (2) is incorrect. This would be correct if the break occurred in the variable leg rather than the reference leg of the ΔP transmitter.

Technical Reference(s): OPL171.045, Core Spray System (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/09/2008 RMS

Question History: Last NRC Exam

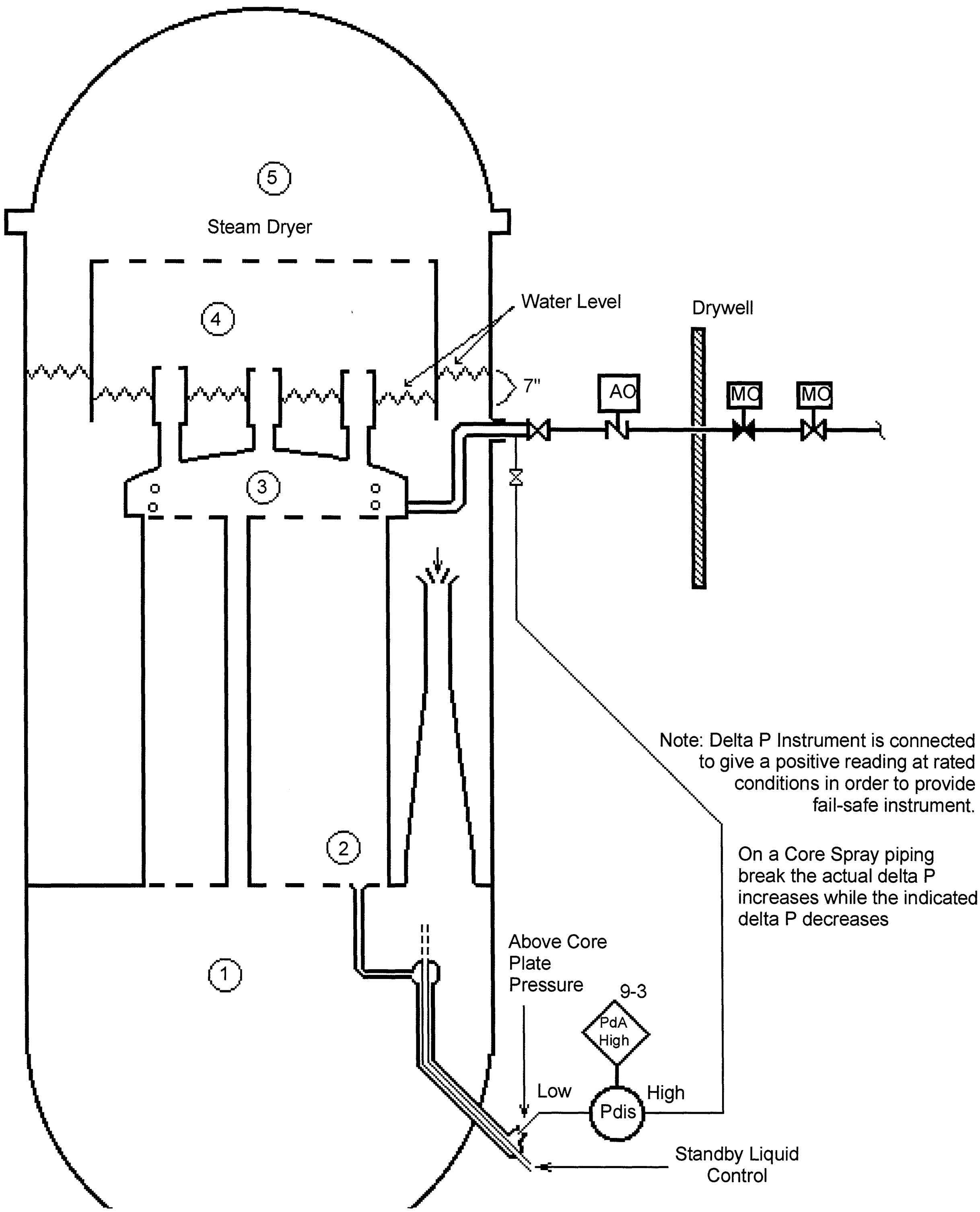
Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:


5
Steam Dryer
4
Water Level
Drywell
7"
AO
MO
MO
3
2
1
Note: Delta P Instrument is connected to give a positive reading at rated conditions in order to provide fail-safe instrument.
On a Core Spray piping break the actual delta P increases while the indicated delta P decreases
Above Core Plate Pressure
9-3
PdA High
Low
Pdis
High
Standby Liquid Control

Excerpt from OPL171.045 pages 29 & 30 of 49:

1. Leak Detection

Core Spray piping penetrates the drywell, reactor vessel and shroud. If a pipe break occurred between vessel wall and the shroud, Core Spray function would be lost. Pipe break detection system monitors the integrity of the Core Spray piping and alarms in Control Room.

a. Pressure 1 (P1) is greater than P5 due to the jet pump driving force.

b. P1 is greater than P2 due to the pressure drop across the core plate.

c. P2 is greater than P3 due to the pressure drop across the core. (This ΔP is small.)

d. P3 is greater than P4 by 7 psi due to the pressure drop across the steam separators.

e. P4 is greater than P5 by 7" of water due to the pressure drop across the steam dryer.

f. The low side of the detector senses above-core plate pressure (P2) plus the pressure due to the height of water in the vessel. Under normal conditions the high side of the detector senses core exit pressure (P3) plus pressure due to the height of water in the sensing leg. With the plant operating at rated conditions the detector reads +3.5 psid. P3 is slightly less than P2 due to the ΔP across the core. Therefore, the pressure differential detected is mainly due to the height of cold water (135°F) in the high leg of piping. If the Core Spray piping breaks between the reactor vessel and the shroud, piping is now sensing P5 instead of P3, and the high-side pressure at the detector would decrease by 7 psig. Sensed low-side pressure will remain the same. This would cause the ΔP to decrease, causing an alarm to sound at 2 psid decreasing (following a 15-sec time delay). During cold shutdown conditions this alarm will normally be in. This is due to low-side pressure being greater than high side pressure (negative ΔP).

BFN Unit 1	Panel 1-9-3 1-XA-55-3C	1-ARP-9-3C Rev. 0019 Page 17 of 38

CORE SPRAY SYS I SPARGER BREAK 1-PDA-75-28 (14)	Sensor/Trip Point:	
	1-PDIS-075-0028	2 psig lowering ΔP (15 second time delay)

(Page 1 of 1)

Sensor Location: 1-LPNL-925-0057
Rx Bldg, El 565', R-3 S-LINE

Probable Cause:
A. Indication of Core Spray piping break inside primary containment.
B. Low core flow.
C. Malfunction of sensor.

Automatic Aciton: None

Operator Action:

A. **DISPATCH** personnel to 1-LPNL-925-0057 to check CSS SYS I Hi ΔP, 1-PDIS-075-0028. **COMPARE** with CSS SYS II Hi ΔP, 1-PDIS-075-0056, on same panel. (The normal reading should be approximately 3.5 psid.) ☐
B. IF necessary, **THEN** **DISPATCH** IMs to VERIFY instrument operation. ☐
C. IF there are indications of a broken Core Spray header, **THEN** **CONSIDER** the associated Core Spray system INOPERABLE and **TAKE** appropriate action as required by Tech Spec 3.5.A **(TS Section 3.5.1)**. ☐
D. IF there are no indications of a Core Spray header break, **THEN** **REFER TO** Tech Spec table 3.2.B **(TS Section 3.3.5.1)**. ☐

References:

1-45E620-2-1	GE 730E930-2 & -8	47W600-59
1-47E610-75-1	(Technical Specifications 3.5.1 and 3.3.5.1)	

Examination Outline Cross-reference:	Level	RO	SRO
211000K6.03	Tier #	2	
Knowledge of the effect that a loss or malfunction of the following will have on the SLC System: AC Power.	Group #	1	
	K/A #	211000K6.03	
	Importance Rating	3.2	3.3

Proposed Question: **RO # 7**

Given the following Unit 3 plant conditions:

- 480 V Shutdown Board 3A tripped due to an electrical fault.
- An ATWS has occurred requiring the initiation of Standby Liquid Control (SLC).

Which ONE of the following describes the SLC pump which should be started and the status of the squib valves once the appropriate pump is started?

The OATC should start the __(1)__ SLC Pump. Once started, _____(2)_____ should fire.

	(1)	(2)
A.	3A	both squib valves.
B.	3A	only the "A" squib valve.
C.	3B	both squib valves.
D.	3B	only the "B" squib valve.

Proposed Answer: **C**

Explanation:

a. Part (1) is incorrect. SLC Pump 3A is powered from 480V Shutdown Board 3A, which is de-energized. Part (2) is correct. Both squib valves receive power from two sources. Each squib has two primers on separate power supplies.

b. Part (1) is incorrect as stated in (a) above. Part (2) is incorrect as described in (a) above.

c. Correct answer.

d. Part (1) is correct. 3B SLC pump has an operable power supply. Part (2) is incorrect as stated in (a) above.

Technical Reference(s):	3-OI-63, SLC System	(Attach if not previously provided)
	OPL171.039, SLC System	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #	211000K5.04	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

BFN Unit 3	Standby Liquid Control System	3-OI-63 Rev. 0020 Page 6 of 30

3.0 PRECAUTIONS AND LIMITATIONS

A. The Unit SRO/RO or Shift Manager are the only persons authorized to inject SLC solution.

B. SLC Pump Operation

1. 3A and 3B SLC PUMP HAND SWITCHES, 3-HS-063-0006AA and 3-HS-063-0006B, are for pump starting only. The squib valves will not fire when using these control switches.

2. Starting either SLC pump from the control room fires both squib valves.

3. The SLC pumps are interlocked so that only one pump can be run at a time. Operation of both SLC pumps simultaneously may result in overpressurization of the system.

4. [II/F] SLC pump abnormal noise (similar to uncoupled or no load condition), lack of normal test tank perturbations, or smell of burnt packing may indicate that the pump is air bound. These positive displacement pumps do not deliver flow if air bound. [Incident Investigation II-B-90-134]

C. SLC System Heating

1. The use of heat tracing is optional. Fuses 3-FU2-063-0005AB for the normal heat trace circuit (480V RMOV Bd 3A, Compt 12A) and 3-FU2-063-0005BB for the alternate heat trace circuit (480V RMOV Bd 3B, Compt 9A) will have to be installed if heat trace is to be used.

2. The SLC tank heaters are set to cut off at approximately 830 gallons in the tank.

D. Adequate mixing time (20 minutes) is required to be strictly enforced to ensure representative sampling. Excessive mixing times should be avoided (i.e. approximately 1 hour).

E. When SLC is being air mixed, the SLC system is to be considered INOPERABLE due to the possibility of air entrapment in the SLC pumps rendering them air bound. The SLC system will be OPERABLE when the air mix is no longer in operation.

Excerpt from OPL171.039 Pages 15 & 16:

1. SLC Pumps
   a) Two 100% capacity, triplex, positive displacement piston pumps are installed in parallel.
   b) 'A' pump is powered from 480V Shutdown Board A.
   c) 'B' pump is powered from 480V Shutdown Board B.
   d) Electrically interlocked so that only one pump will run at a time. This prevents system overpressurization.
   e) The pumps are manually started from the main control room using the key-lock switch on panel 9-5, or locally, using the Test Permissive Transfer Switch at Panel 25-19.
   f) A control room start signal will fire the explosive valves. A local start will <u>not</u> fire the explosive valves.

2. Explosive Valves
   a) Two 100% capacity explosive (Squib) valves, FCV 63-8A and B, are installed in parallel.
   b) Provide a zero leakage seal between the boron solution and the reactor.
   c) Each valve contains two firing primers, powered by the 250V DC control power from the 480V Shutdown Boards A and B, (unit specific).
   d) Either primer is capable of actuating the valve.
   e) The primer is fired by taking the main control room handswitch, HS-63-6A, to the START PUMP A or START PUMP B position. This forces the ram outward, which shears the end cap off the valve fitting, allowing flow to pass through the valve.
   f) After firing, the ram remains extended. This prevents the sheared cap from obstructing flow through the valve.
   g) The primer requires a minimum current of 2 amps to fire, and fires within 2 milliseconds after this circuit is applied. All the explosion by-products are retained in the trigger explosive chamber.
   h) Each valves firing circuit continuity is monitored by a blue indicating light on Panel 9-5 and a current meter located in the back of Panel 9-5.

Examination Outline Cross-reference:	Level	RO	SRO
212000A1.06	Tier #	2	
Ability to predict and/or monitor changes in parameters associated with operating the RPS controls including: Reactor Power.	Group #	1	
	K/A #	212000A1.06	
	Importance Rating	4.2	4.2

Proposed Question: **RO # 8**

Unit 1 is performing 1-SR-3.3.1.1.8(11), "Reactor Protection System Manual Scram Functional Test" with the following conditions:

- A manual scram was inserted on RPS Channel "A".
- All four (4) SCRAM SOLENOID GROUP A LOGIC RESET, red indicating lights extinguished.
- The OATC failed to properly reset the ½ scram on RPS "A" and SCRAM SOLENOID GROUP A LOGIC RESET red indicating lights 2 and 3 remained extinguished.
- A manual scram was then inserted on RPS Channel "B".

Which ONE of the following describes the final STEADY STATE condition of the plant to this event and the reason for that condition?

Inserting a manual ½ scram on RPS Channel "B" will cause ______(1)______ of the control rods to insert. This is caused by ________(2)________.

	(1)	(2)
A.	50%	Scram Discharge Volume high level.
B.	50%	Backup Scram Valve actuation.
C.	100%	Scram Discharge Volume high level.
B.	100%	Backup Scram Valve actuation.

Proposed Answer: **C**

Explanation:

a. Part (1) is correct. Initially, 50% of the control rods insert, but within a few seconds the Scram Discharge volume fills enough to initiate a full scram. Since the stem asks for STEADY STATE conditions, 100% of the control rods will insert. Part (2) is correct.

b. Part (1) is incorrect. Initially, 50% of the control rods begin to insert, but the remainder will begin to insert from a full scram signal before the first 50% reach full-in. Part (2) is incorrect. Backup Scram Valves will only actuate as a result of the SDV high level, which has already initiated a scram.

c. Correct answer.

d. Part (1) is correct as stated in (a) above. Part (2) is incorrect. Backup Scram Valves will only actuate as a result of the SDV high level, which has already initiated a scram.

Technical Reference(s): 1-SR-3.3.1.1.8(11) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/10/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments: **NOTE:** Each HCU accumulator hold approximately 5 gallons of water which enters the SDV on a scram. With Approximately 90 control rods inserting, 450 gallons of water are directed to the SDV, which will initiate a full scram at 46-50 gallons. It doesn't take long. Simulator times were on the order of 2 to 3 seconds.

Excerpt from OPL171.028 page 16 of 50:

a. SCRAM discharge volume high level,

(1) Initiates a SCRAM while adequate volume is available to receive SCRAM discharge water to assure that all operable drives will fully insert

(2) Level is sensed by two mechanical float switches and two electronic level switches (RTD's) in each instrument volume.

(3) East Instrument Volume

- LS-85-45E (A1-float)
- LS-85-45F (B1-float) 50 gal - float
- LS-85-45G (A2-thermal)
- LS-85-45H (B2-thermal) 46 gal - thermal

(4) West Instrument Volume

- LS-85-45A (A1-thermal)
- LS-85-45B (B1-thermal) 46 gal - thermal
- LS-85-45C (A2-float)
- LS-85-45D (B2-float) 50 gal - float

(5) When the instrument volume fills up to the setpoint, the sensors open contacts in both RPS Trip Systems. Therefore, one SDV that is full would initiate a full SCRAM

Examination Outline Cross-reference:

212000K5.01

Knowledge of the operational implications of the following concepts as they apply to RPS: Fuel Thermal Time Constant.

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	212000K5.01	
Importance Rating	2.7	2.9

Proposed Question: **RO # 9**

Which ONE of the following Reactor Protection System scram signals is delayed for six (6) seconds once the setpoint has been exceeded and the basis for that delay?

The scram signal for ______(1)________ is delayed for six seconds to take into consideration the ____________(2)______________.

	(1)	(2)
A.	APRM Flow-biased STP (.66W +66%)	fuel thermal time constant.
B	APRM Flow-biased STP (.66W +66%)	nominal MSIV closure time.
C.	APRM High Flux $\leq$ 120% RTP	fuel thermal time constant
D.	APRM High Flux $\leq$ 120% RTP	nominal MSIV closure time.

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Part (1) is correct. Part (2) is incorrect. The high flux trip of 120% is designed to provide protection against an MSIV closure and the resultant pressure transient.

c. Part (1) is incorrect. The 120% high flux trip is not delayed by RPS. The flow-biased high flux scram is designed for slow power increases such as feedwater heating problems. Part (2) is correct, but not for the setpoint given in Part (1).

d. Part (1) is incorrect. The 120% high flux trip is not delayed by RPS. The flow-biased high flux scram is designed for slow power increases such as feedwater heating problems. Part (2) is incorrect, but the MSIV closure time is relevant with regard to the 120% scram setpoint since the basis for that setpoint is to protect against a MSIV closure event.

Technical Reference(s):	TSR 3.3.1.1	(Attach if not previously provided)
	OPL171.148, PRNM	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/10/2008 RMS	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

(3) The reactor mode switch provides the "RUN" vs. "OUT of RUN" signal to determine if the SETDOWN function is applied. This setdown function refers to the reactor mode switch function.

(4) Each APRM provides a digital input that monitors the state of the Reactor Mode Switch. The input is active (signal present) when the switch is in the "RUN" position and inactive for all other switch positions.

(5) The APRM instrument defaults to the "RUN" mode of operation in the event of loss of Reactor Mode Switch signal processing power in order to prevent a full scram condition.

(6) The Bypass signal is used to indicate that only one APRM channel is bypassed. Otherwise, the signal is interpreted as NONE for the APRMs being bypassed. Once an APRM channel is bypassed, all trip function inputs to the voters are also bypassed. V.B.12

(7) The APRM Instrument receives a signal from the Two-Out-Of-Four Logic Module indicating that the APRM channel (as well as the LPRM and OPRM) is bypassed.

(8) The APRM calculates the average neutron flux. The average neutron flux is the average of non bypassed LPRM with gain correction applied so that the signal corresponds to reactor power. The averaged LPRM value is adjusted to read in units of "percent of rated core thermal power". V.B.13 V.C.4

(9) The STP signal is a result of applying a 6 second filter to the average flux signal which approximates the time response of reactor thermal power.

RPS Instrumentation
3.3.1.1

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 15% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.66 W + 66% RTP and ≤ 120% RTP(c)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP

(continued)

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

(b) Each APRM channel provides inputs to both trip systems.

(c) [0.66 W + 66% - 0.66 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than or equal to the Average Power Range Monitor Fixed Neutron Flux - High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor Fixed Neutron Flux - High Function will provide a scram signal before the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function setpoint is exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

Each APRM channel uses one total drive flow signal representative of total core flow. The total drive flow signal is generated by the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this function.

The clamped Allowable Value is based on analyses that take credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function for the mitigation of the loss of feedwater heating event. The THERMAL POWER time constant of < 7 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER. The term "W" in the equation for determining the Allowable Value is defined as total recirculation flow in percent of rated.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Average Power Range Monitor Fixed Neutron Flux - High

The Average Power Range Monitor Fixed Neutron Flux - High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 4, the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 5) takes credit for the Average Power Range Monitor Fixed Neutron Flux - High Function to terminate the CRDA.

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux - High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed in the CRDA analysis, which is applicable in MODE 2, the Average Power Range Monitor Neutron Flux - High, (Setdown) Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Range Monitor Fixed Neutron Flux - High Function is not required in MODE 2.

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
215003G2.4.8	Tier #	2	
Knowledge of how abnormal operating procedures are used in conjunction with EOPs: Intermediate Range Monitors	Group #	1	
	K/A #	215003G2.4.8	
	Importance Rating	3.8	4.5

Proposed Question: **RO # 10**

Given the following Unit 1 plant conditions:

- A reactor scram has occurred.
- All control rods did NOT fully insert following the scram.
- 1-EOI-1, "RPV Control" has been entered based on low RPV level.
- RPV level is (-) 15 inches and rising with feed water injection.
- 1-EOI-2, "Primary Containment Control" entry is NOT required.
- All Intermediate Range Monitors are fully inserted and reading on Range 5 and lowering.

Which ONE of the following describes the appropriate action to monitor and control reactor power?

______(1)______ path RC/Q of 1-EOI-1, "RPV Control" and control reactor power using ________(2)________.

	(1)	(2)
A.	Remain in	1-AOI-100-1, "Reactor Scram" and 1-OI-85, "CRD System."
B.	Remain in	1-EOI Appendix 1D, "Insert Control Rods using Reactor Manual Control System."
C.	Exit	1-AOI-100-1, "Reactor Scram" and 1-OI-85, "CRD System."
D.	Exit	1-EOI Appendix 1D, "Insert Control Rods using Reactor Manual Control System."

Proposed Answer: **C**

Explanation:

a. Part (1) is incorrect. With IRMs on Range 7 or below and no entry conditions for EOI-2, the reactor is subcritical with no boron injection. This requires exiting RC/Q and entry into AOI-100-1. Part (2) is correct. Actions in AOI-100-1 and OI-85 are authorized even if RC/Q is NOT exited as required so long as those actions do not interfere with EOI actions. In this case, they would not.

b. Part (1) is incorrect as stated in (a) above. Part (2) is incorrect. Actions in EOI Appendix 1D involve bypassing RWM and other more drastic actions that are not necessary if the reactor is subcritical without boron injection. Therefore, those actions should NOT be performed.

c. Correct answer.

d. Part (1) is correct. The given conditions indicate sub-criticality and no requirement for boron injection. Retainment Override step RC/Q-3 directs RC/Q exited and AOI-100-1 entered for power control. Part (2) is incorrect as stated in (b) above.

Technical Reference(s): 1-EOI-1, 1-AOI-100-1, 1-OI-85 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/11/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Excerpt from OPL171.202 page 49:

1. Step RC/Q-3

   a. The second retainment override statement directs the operator to transfer reactor power control actions if present plant conditions are such that the reactor is subcritical and no boron has been injected. As used in EOIs, the term "subcritical" means that reactor power is below the heating range and not increasing. If it is determined that the reactor is subcritical without having injected any boron into the RPV, an exit to AOI-100-1,

Excerpt from OPL171.201 page 25:

a. Determination of Shutdown Margin

   (1) EOI-1, C-1,C-2, C-4, C-5, requires that a determination of the ability of the reactor to remain subcritical under all conditions without boron, be made.

   (2) During ATWS conditions, when the reactor is subcritical, the conditions of EOI Note 1 should be evaluated. If necessary, Reactor Engineering should be requested to determine if the reactor will remain subcritical under all conditions without boron injection.

   (3) This request should be made as soon as possible after it is known that rod insertion is no longer possible in order to facilitate later actions in the EOIs, if needed.

Excerpt from OPL171.201 page 30:

b. Subcritical

   (1) When used in the EOIs, subcritical means reactor power below the heating range and not trending upward.

   (2) (Reactor power on range 7 and lowering of the IRMs with the IRMs inserted.)

BFN Unit 1	Reactor Scram	1-AOI-100-1 Rev. 0003 Page 10 of 61

4.2 **Subsequent Actions (continued)**

[8] [INPO/C] **CHECK** all control rods are fully inserted as indicated on the full core display or on the ICS NSSS FULL CORE DISPLAY and **REQUEST PRINT** ROD POSITION LOG on the ICS NSSS menu. [INPO SOER 80-006] ☐

NOTE
Step 4.2[8.1] may require support from off-site organizations and an extended period may elapse before results are obtained.

[8.1] **IF** all rods are **NOT** inserted to Position 02 or beyond, **THEN**

DIRECT Reactor Engineer to commence determination that the reactor will remain subcritical under all conditions without boron. ☐

[9] [INPO/C] **IF** any control rod fails to fully insert and it is required to Re-scram, **THEN**

PERFORM the following, as required. [INPO SOER 80-006]

[9.1] **RESET** the scram per Steps 4.2[23] thru 4.2[23.10]. ☐

[9.2] **VERIFY** WEST and EAST CRD DISCH VOL WTR LVL HIGH HALF SCRAM annunciators (1-XA-55-4A, window 1 and 1-XA-55-4A, window 29) are reset. ☐

[9.3] **INITIATE** a manual scram. **REPEAT** Step 4.2[9], as necessary, as long as rod motion is observed. ☐

[10] [INPO/C] **IF** any control rod fails to fully insert and it is required to Drive Control Rods, **THEN**

REFER TO 1-OI-85. [INPO SOER 80-006] ☐

Table of Contents (continued)

BFN UNIT 1	INSERT CONTROL RODS USING REACTOR MANUAL CONTROL SYSTEM	1-EOI APPENDIX-1D Rev. 0 Page 2 of 3

LOCATION: Unit 1 Control Room, Panel 1-9-5

ATTACHMENTS: 1. Core Position Map (√)

NOTE

This EOI Appendix may be executed concurrently with EOI Appendix 1A or 1B at SRO's discretion when time and manpower permit.

1. **VERIFY** at least one CRD pump in service. _____

NOTE

Closing 1-SHV-085-0586, CHARGING WTR ISOL, valve may reduce the effectiveness of EOI Appendix 1A or 1B.

2. IF .... Reactor Scram or ARI CANNOT be reset,

   THEN.......... **DISPATCH** personnel to close 1-SHV-085-0586, CHARGING WTR ISOL (RB NE, El 565 ft). _____

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 1-SHV-085-0586, CHARGING WTR ISOL (RB NE, El 565 ft). _____

END OF TEXT

Examination Outline Cross-reference:	Level	RO	SRO
215004A2.02	Tier #	2	
Ability to (a) predict the impacts of the following on the Source Range Monitor system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: SRM INOP condition.	Group #	1	
	K/A #	215004A2.02	
	Importance Rating	3.4	3.7

Proposed Question: **RO # 11**

Given the following Unit 1 plant conditions:

- A reactor startup is in progress following refueling, with ALL 8 Reactor Protection System (RPS) Shorting Links installed.
- The reactor is in Mode 2.
- IRM "G" is on Range 7, all other IRMs are on Range 8.

An electronic failure in the 'B' Source Range Monitor (SRM) drawer results in a SRM HIGH/INOP (9-5A W13) alarm.

Which ONE of the following describes the plant response and required action(s), if any, to continue the startup?

The SRM failure will initiate a ____(1)____. The startup may continue ____(2)____ bypassing SRM "B" in accordance with 1-OI-92, "Source Range Monitor System."

	(1)	(2)
A.	Control Rod Withdraw Block	after
B.	SRM HIGH/INOP alarm ONLY	without
C.	Control Rod Withdraw Block	without
D.	SRM HIGH/INOP alarm ONLY	after

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Part (1) is incorrect. If the reactor was in Mode 1, this would be correct with IRM "G" still on range 7. Part (2) is incorrect. Rod withdrawal is not possible with the SRM INOP until it is bypassed.

c. Part (1) is correct. If the reactor was in Mode 1, this would be incorrect . Part (2) is incorrect. Rod withdrawal is not possible with the SRM INOP until it is bypassed.

d. Part (1) is incorrect. If the reactor was in Mode 1, this would be correct with IRM "G" still on range 7. Part (2) is incorrect. Rod withdrawal is not possible with the SRM INOP until it is bypassed.

Technical Reference(s): 1-OI-92, Source Range Monitor System (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # 215004A3.03 attached

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Original question RO 215004A3.03:

Given the following plant conditions:

- A reactor startup is in progress following refueling, with ALL 8 Reactor Protection System (RPS) Shorting Links removed.
- The reactor is approaching criticality.
- An electronic failure in the 'B' Source Range Monitor (SRM) drawer results in an SRM HIGH/HIGH output signal.

Which ONE of the following describes the plant response?

A. A Rod Out Block ONLY.

B. A Rod Out Block and 1/2 Scram ONLY.

C. A "SRM HIGH/HIGH" alarm ONLY.

D. A Full Reactor Scram.

BFN Unit 1	Source Range Monitors	1-OI-92 Rev. 0006 Page 14 of 14

Illustration 1
(Page 1 of 1)

SRM Trip Outputs

TRIP SIGNAL	SETPOINT	ACTION
SRM High	6.8×10^4 counts per second	Rod block unless IRMs on Range 8 (or higher) or REACTOR MODE SWITCH in RUN
SRM Inop	A. Module unplugged B. Mode switch not in operate C. HV power supply low voltage D. Loss of +/-24 vdc	Rod block unless IRMs on Range 8 (or higher) or REACTOR MODE SWITCH in RUN
SRM Downscale	5 counts per second	Rod block unless IRMs on range 3 (or higher) or REACTOR MODE SWITCH in RUN
SRM Detector Wrong Position	145 counts per second	Rod block unless detector full-in, IRMs on range 3 (or higher), or REACTOR MODE SWITCH in RUN
SRM High-High	2×10^5 counts per second	Scram if shorting links removed

Examination Outline Cross-reference:	Level	RO	SRO
215005K2.02	Tier #	2	
Knowledge of electrical power supplies to the following: APRM/LPRM (APRM Channels)	Group #	1	
	K/A #	215005K2.02	
	Importance Rating	2.6	2.8

Proposed Question: **RO # 12**

Given the following Unit 1 plant conditions:

- Operating at 100% rated power.
- The normal feeder breaker to 4KV Shutdown Board C inadvertently trips.
- The alternate breaker from Shutdown Bus 1 fails to close.
- 4KV Shutdown Board C is now being powered from C Diesel Generator.

Which ONE of the following describes the effect on Average Power Range Monitors (APRM) and Rod Block Monitors (RBM) due to this electrical transient?

A. Power Range Neutron Monitoring (PRNM) is not affected by this transient.

B. All APRM channels generate a Critical Fault and RBM B generates a Non-critical fault.

C. All APRM channels generate a Non-critical Fault and RBM B generates a Critical fault.

D. All APRM channels and RBM channels generate a Non-critical fault.

Proposed Answer: **C**

Explanation:

a. Incorrect. APRM channels generate non-critical faults due to a loss of RPS B. RBM B generates a Critical Fault. The loss of RPS B can be determined because of the time required for C D/G to start and tie to the 4KV S/D board. The RPS MG set flywheel cannot hold speed and voltage long enough to prevent a UV trip.

b. Incorrect. APRM channels generate non-critical faults and RBM B generates a Critical Fault since its interface panel has lost power.

.

c. Correct answer.

d. Incorrect. RBM B generates a Critical Fault since its interface panel has lost power.

Technical Reference(s): 1-OI-92B, PRNM System (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/11/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 1	Average Power Range Monitoring	1-OI-92B Rev. 0008 Page 8 of 27

3.0 PRECAUTIONS AND LIMITATIONS (continued)

N. The OPRM channel/cell provides an INOPERATIVE ALARM (Inverse Video) when the quantity of operating OPRM cells is less than 23. When the number of LPRMs in a cell is reduced to less than 2, the OPRM Cell is considered Inoperable. The OPRM function is disabled when the reactor mode switch is in a position other than RUN or the Reactor is operating outside of the OPRM Auto Enable region.

O. The Operators Display Assembly, which normally monitors LPRM or APRM function, will automatically switch over to OPRM monitoring when the reactor is placed in the region of potential instability. The region of potential instability is bounded by at least 25% power and less than or equal to 60% total recirc drive flow (OPRM Auto Enable region) from any one of the channels.

P. The message "OPRM TRIP ENABLED" will be displayed for each APRM when entering the power/flow region where instability can occur. The message will be replaced with "ANTICIPATED INSTABILITY" whenever a Pre Trip (alarm) setpoint has been reached by any of the OPRM algorithms. If an oscillation trip exists, as defined by the OPRM trip setpoints, the message will be replaced with "INSTABILITY DETECTED" and when two of these types of trips occur, an RPS automatic scram is received.

Q. The operator has the ability to transfer the display back from OPRM to APRM by depressing the "ETC" softkey.

R. There are a total of four Operators Display Assemblies (ODAs), two for the APRMs/OPRMs and two for the RBM. Each APRM ODA provides indication for two APRMs/OPRMs. All four ODAs are powered by I & C BUS "A".

S. The following are power supplies for the APRM/OPRM:

Panel 1-9-14 is made up of 5 Chassis, 4 APRMs and 1 RBM.

There are five Quadruple Low Voltage Power Supplies, one per bay on Panel 1-9-14.

Each QLVPS receives power from both RPS busses. LVPS 1 and 2 are fed from RPS A, LVPS 3 and 4 are fed from RPS B.

For each QLVPS, LVPS 1 and 4 feed the APRM and RBM A Chassis and LVPS 2 and 3 feed LPRM and RBM B Chassis.

Each Voter is powered from the RPS bus it serves, such that those assigned to RPS sub-channels B1 and B2 are powered from RPS B and those assigned to RPS sub-channels A1 and A2 are powered from RPS A. These power supplies are seen at the bottom of the panels on the QLVPS and are indicated energized by the illuminated green lights.

BFN Unit 1	Average Power Range Monitoring	1-OI-92B Rev. 0008 Page 9 of 27

3.0 PRECAUTIONS AND LIMITATIONS (continued)

T. A loss of an RPS A or B will result in a Critical Fault on RBM A or B (respectively), and a Non-Critical Fault on APRM channels (All APRM and LPRM Chassis continue to operate).

U. LPRM Alarms high at 100% and downscale at 3%. The solid box above the bargraph indicates that the setpoint marker is presently exceeded while the hollow box indicates a past condition. A past condition can be reset by entering the TRIP STATUS display and pressing the RESET MEMORY softkey.

V. The total number of LPRMs that may be bypassed (failed) is 23. If the number of bypassed (failed) LPRM inputs exceeds the minimum number required in the APRM average, (<20 total or < 3 per level) an APRM INOP CONDITION is applied, resulting in a Rod Withdrawal Block and a trouble alarm on the APRM channel display in Inverse Video. This APRM INOP CONDITION is not an automatic trip but does render the associated APRM inoperable.

W. Bypassed LPRMs are not used by PRNM system for calibration, flux, or trip points. LPRMs are normally bypassed from Panel 1-9-14 using "BYP/HV ON" or "BYP/HV OFF".

BYP/HV ON — LPRM is manually bypassed with the voltage on the detector. Indication of the detector output is available, but the signal is not included for any input to the APRM/OPRM/RBM functions.

BYP/HV OFF — LPRM is manually bypassed with the voltage off. No detector output.

In addition, LPRMs may be bypassed as indicated below:

BYP/IV — LPRM is automatically bypassed as a remotely initiated I/V (current to Voltage check)process is in progress.

BYP/CAL — LPRM is automatically bypassed while the LPRM is being calibrated (CALIBRATE) or that the calibration is being checked (CAL CHECK).

BYP/SUB'D — LPRM is bypassed while undergoing TRIP CHECK

BYP/FAULT — LPRM is automatically bypassed as a LPRM self-test fault is detected.

Excerpt from OPL171.148 page 69:

(g) On the front panel of QLVPS, four green indicating lamps provide indication that there is power to the modules . An illuminated indicating lamp is only an indication of power to the LVPS module and is not an indication that the LVPS module is functioning properly.

Discuss failure of any power supply (Demonstrated on simulator)

(h) In case of one power supply failure, the APRM self testing process will identify it as a non-critical fault, and the normal operation of APRM will continue.

(2) Discuss the effects of losing "A" RPS power on PRNM.

(a) Voters for channels 1 and 3 lose power.

(b) Non critical fault on all APRM, LPRM, and RBM instruments.

(c) Critical fault on RBM channel A since it's interface panel has lost power. A Rod block is initiated from RBM channel "A".

(3) The loss of RPS "B" is similar.

(4) **If one voter is powered down with trip signal present and bypassed, then the trip signal will remain in and bypass function still works.**

If powered down with no trips and with no bypass, the X and Y relays are deenergized which will input to RPS (1/2 scram).

(5) The loss of a LVPS supply's output causes a self-test alarm.

Examination Outline Cross-reference:	Level	RO	SRO
217000K3.04	Tier #	2	
Knowledge of the effect that a loss or malfunction of the RCIC system will have on the following: Adequate Core Cooling.	Group #	1	
	K/A #	217000K3.04	
	Importance Rating	3.6	3.6

Proposed Question: **RO # 13**

Given the following Unit 2 plant conditions:

- A Group I isolation occurred from Steam Tunnel high temperature.
- Feedwater flow on "A" feedwater line indicated upscale.
- HPCI and RCIC automatically initiated and RPV level was slowly restored until both HPCI and RCIC tripped on high RPV level.
- RPV level again began to lower.
- With RPV level (-) 25 inches and lowering, the Board Unit Operator noticed that 2-71-8, RCIC Steam Supply Valve suddenly lost light indications for valve position.

Which ONE of the following describes the status of the RCIC system and, based on that status, the effect on RPV level recovery efforts?

The RCIC Steam Supply Valve, 2-71-8 will ____(1)____ without power when RCIC receives another initiation signal. Based on this response, RPV level will ______(2)______.

	(1)	(2)
A.	remain open	lower due to RCIC injection into the "A" feedwater line.
B.	remain open	rise due to RCIC injection into the "B" feedwater line.
C.	remain closed	lower due to HPCI injection into the "A" feedwater line.
D.	remain closed	rise due to HPCI injection into the "B" feedwater line.

Proposed Answer: **C**

Explanation:

a. Part (1) is incorrect. 2-71-8 will close on a high level trip. Part (2) is incorrect. RPV level will lower, but not based on RCIC injecting into the "A" FW line. HPCI injects into the "A" FW line.

b. Part (1) is incorrect. 2-71-8 will close on a high level trip. Part (2) is incorrect. RPV level will not rise even though RCIC does inject into the "B" FW line. RCIC will not run with 2-71-8 failed closed.

c. Correct answer.

d. Part (1) is correct. 2-71-8 will remain closed and RCIC will not initiate. Part (2) is incorrect. HPCI injects into the "A" FW line, which is isolated and broken. HPCI will run at rated flow but the water will be injected into the steam tunnel and flow into the Reactor Building, not the RPV.

Technical Reference(s): 2-OI-71, OPL171.040, OPL171.042 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/12/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 2	Reactor Core Isolation Cooling	2-OI-71 Rev. 0055 Page 9 of 70

3.0 PRECAUTIONS AND LIMITATIONS

A. Turbine controls provide for automatic shutdown of the RCIC turbine upon receiving any of the following signals (REFER TO Section 8.4 for auto actions):

1. High RPV water level (+51 in.); 579 in. above vessel zero. The RCIC TURBINE STEAM SUPPLY VLV, 2-FCV-71-8, and RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, will close at +51 in. and will **RE-OPEN** when RCIC re-initiates at -45 in. RPV water level.

2. Turbine overspeed (Mechanical, 122.3% of rated speed).

3. Pump low suction pressure (10 inches Hg vacuum).

4. Turbine high exhaust pressure (50 psig).

5. Any isolation signal.

6. Remote manual trip (RCIC TURBINE TRIP push-button, 2-HS-71-9A, depressed).

B. RCIC turbine steam supply will isolate from the following signals (REFER TO 2-AOI-64-2C for auto actions):

1. RCIC steamline space temperature at ≤180°F Torus Area or ≤180°F RCIC Pump Room.

2. RCIC turbine high steam flow (150% flow, 3-second time delay.)

3. RCIC turbine steam line low pressure (73 psig).

4. RCIC turbine exhaust diaphragms ruptured (10 psig).

5. Remote manual isolation (RCIC AUTO-INIT MANUAL ISOLATION push-button, 2-HS-71-54, depressed, only if RCIC initiation signal is present).

C. The RCIC turbine will auto initiate on RPV Low-Low Water Level, -45 in. (REFER TO Section 5.1 for auto actions.)

D. In the presence of a RCIC initiation signal, the RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, opens when system flow is below 60 gpm and closes when flow is above 120 gpm. The valve will NOT auto open on low flow if an initiation signal is NOT present.

E. RCIC PUMP MIN FLOW VALVE, 2-FCV-71-34, will open on receipt of an initiation signal even with RCIC turbine manually tripped resulting in slowly draining CST to Suppression Chamber.

Excerpt from OPL171.040, RCIC page 20:

m. Injection Valve (FCV-71-39) (Normally closed)

Powered from 250VDC RMOV Board C. Pump discharges through a thermal sleeve into B FW line downstream of the outboard isolation check valve.

Excerpt from OPL171.042, HPCI page 11:

a. Water path

(1) Normal condensate path from CST to the HPCI pump, to the A Feedwater line and into the reactor vessel

Examination Outline Cross-reference:	Level	RO	SRO
218000K6.03	Tier #	2	
Knowledge of the effect that a loss or malfunction of the following will have on the ADS System: Nuclear Boiler Instrument System (level indication).	Group #	1	
	K/A #	218000K6.03	
	Importance Rating	3.8	3.9

Proposed Question: **RO # 14**

Given the following Unit 1 plant conditions:

- A steam line break in the drywell has occurred.
- All control rods inserted on the scram.
- RPV level is (-) 70 inches and steady with automatic HPCI injection.
- All attempts to initiate Drywell Sprays have been unsuccessful.
- Drywell temperature is 390 ^{O}F and rising.
- RPV Saturation Temp (Curve 8) has been exceeded.
- RPV level instrumentation has become erratic.

Which ONE of the following describes the required Automatic Depressurization System (ADS) operation and the basis for that requirement?

ADS must be manually ________(1)________ in order to ______________(2)______________ .

	(1)	(2)
A.	inhibited	prevent adding more energy to the Primary Containment.
B.	initiated	establish the conditions necessary to flood the RPV.
C.	inhibited	prevent a further loss of RPV water inventory.
D.	initiated	return to the SAFE area of the RPV Saturation Temp (Curve 8).

Proposed Answer: **B**

Explanation:

a. Part (1) is incorrect. No conditions currently exist that would require ADS to be inhibited. All rods are in and RPV level was steady before indication was lost. Part (2) is incorrect. Adding more heat to the containment is certainly not desirable at this point, but ADS must be initiated to flood the RPV in accordance with 1-EOI-C4, "RPV Flooding."

b. Correct answer.

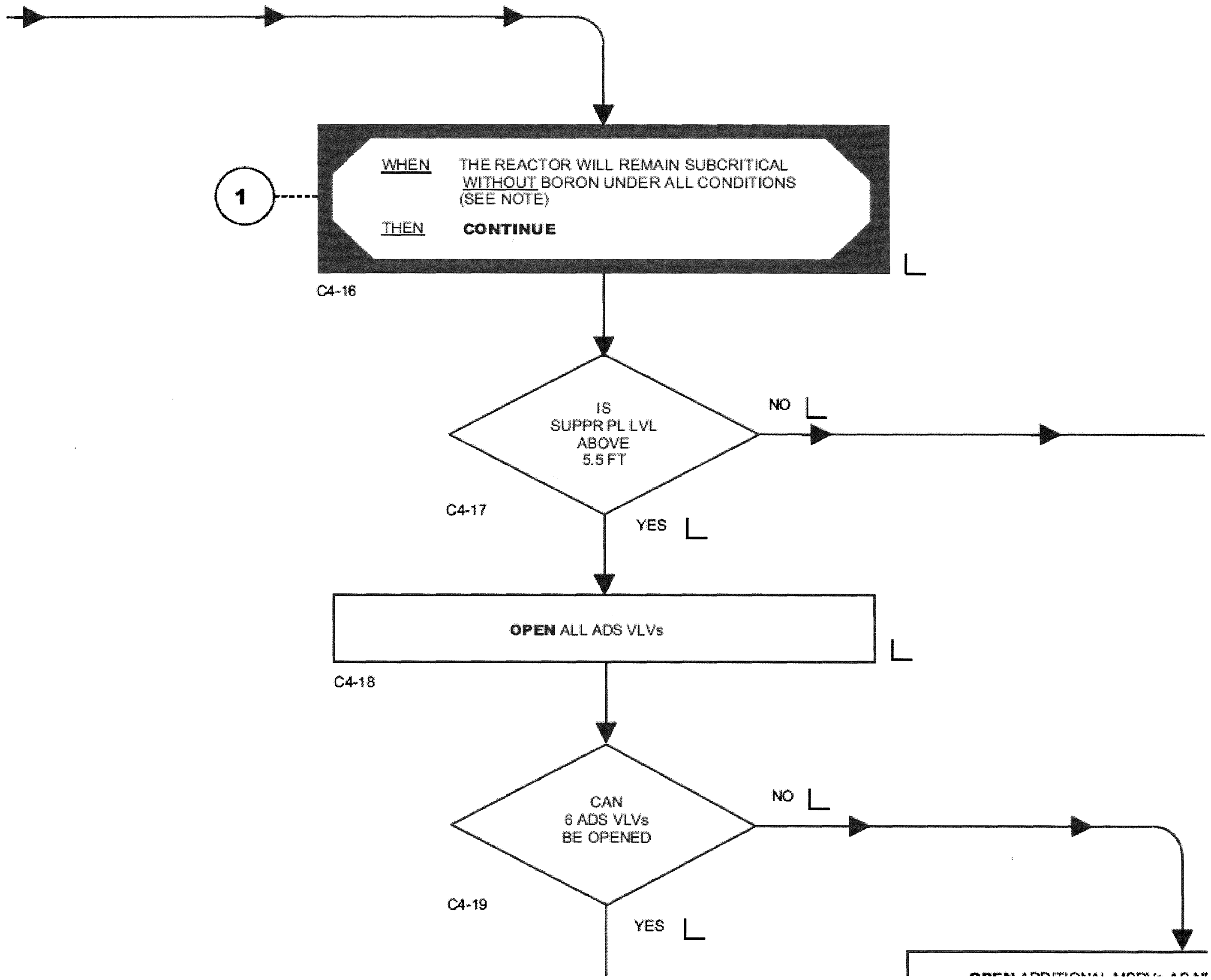
c. Part (1) is incorrect as stated in (a) above. Part (2) is incorrect. Losing additional inventory is a secondary concern without RPV level indication. Establishing the conditions to flood the RPV to a condition where adequate core cooling is assured is the priority.

d. Part (1) is correct. Part (2) is incorrect. Once Curve 8 has been exceeded and RPV level indication is lost, reducing pressure and returning to the SAFE area of the curve will not help restore RPV level instruments. The drywell must be cooled down and RPV level instrument reference legs must be refilled.

Technical Reference(s):	1-EOI-C4, "RPV Flooding."		(Attach if not previously provided)
Proposed references to be provided to applicants during examination:			None
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/13/2008 RMS	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

Excerpt from 1-EOI-C4:



Excerpt from 1-EOI-1 Path RC/L:

ABOVE
+2 IN.

YES

MAINTAIN RPV WATER LVL ABOVE -162 IN.

RC/L-6

WHILE EXECUTING THE FOLLOWING STEPS:	
IF	**THEN**
RPV WATER LVL DROPS BELOW -120 IN. OR THE ADS TIMER HAS INITIATED	**INHIBIT** ADS

RC/L-7

CAUTION

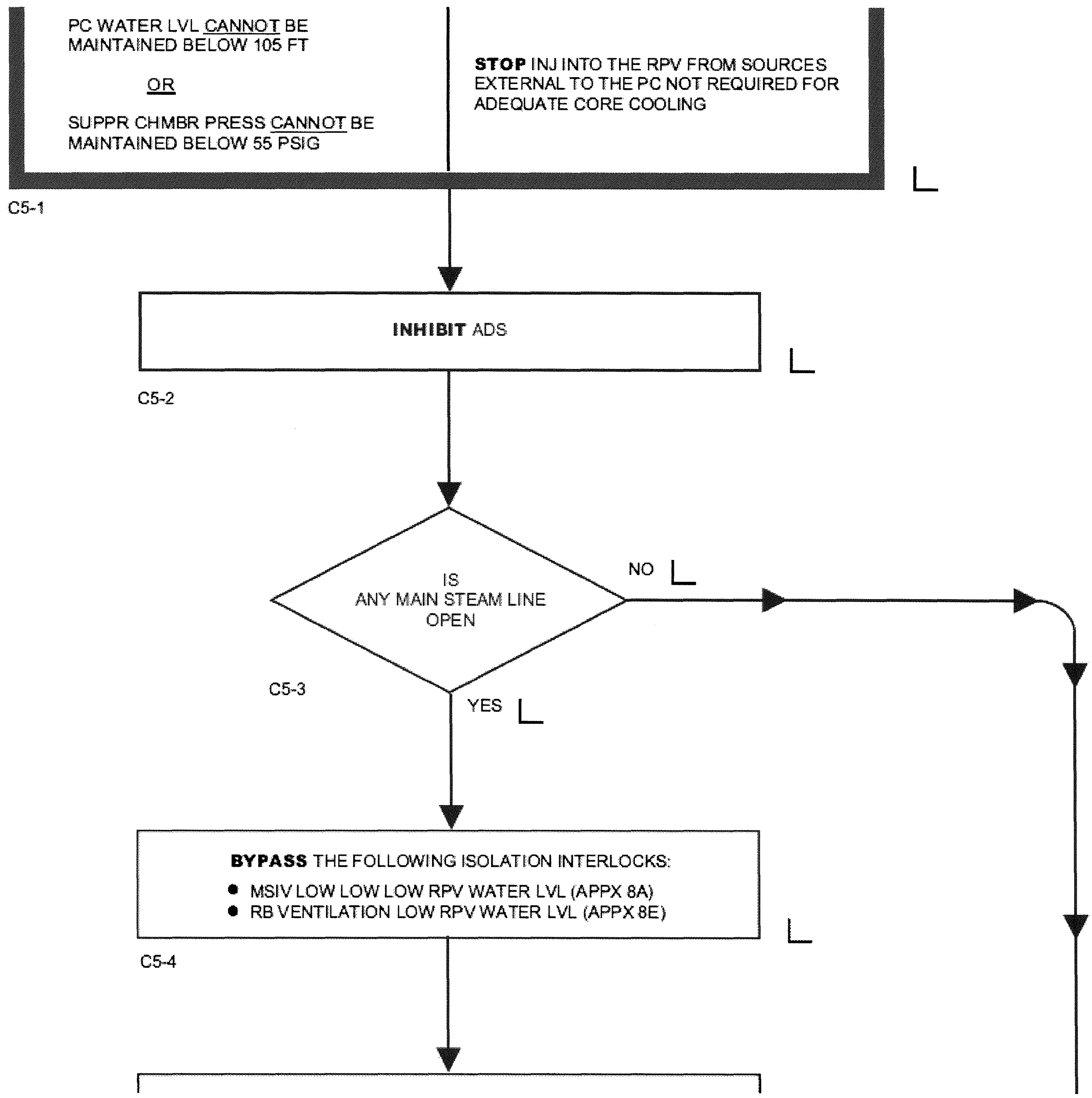
#3 ELEVATED SUPPR CHMBR PRESS MAY TRIP RCIC

#6 HPCI OR RCIC SUCTION TEMP ABOVE 140 °F

AUGMENT RPV WATER LVL CONTROL AS NECESSARY WITH ONE OR MORE OF THE FOLLOWING INJ SOURCES:

INJ SOURCE	APPX	INJ PRESS

Excerpt from 1-EOI-C5, "Level/Power Control."



Examination Outline Cross-reference:	Level	RO	SRO
223002K3.20	Tier #	2	
Knowledge of the effect that a loss or malfunction of the PCIS/NSSS system will have on the following: Standby Gas Treatment system.	Group #	1	
	K/A #	223002K3.20	
	Importance Rating	3.3	3.4

Proposed Question: **RO # 15**

Given the following Unit 1 plant conditions:

- Reactor power is at 100% with all systems in a normal lineup.
- Instrument Mechanics (IM) are performing calibrations on the drywell pressure sensors when an inadvertent Group 2 and Group 6 PCIS isolation signal is received on Unit 1.

Which ONE of the following describes the response of the Standby Gas Treatment (SGT) system and the effect these PCIS isolations will have on plant operation?

Standby Gas Treatment (SGT) trains ____(1)____ will be operating. As a result of this PCIS malfunction, the reactor will ____________(2)____________.

	(1)	(2)
A.	A, B and C	continue to operate until PCIS is restored to a normal condition.
B.	A, B and C	scram due to steam tunnel high temperature and Group I isolation.
C.	A and B only	continue to operate until PCIS is restored to a normal condition.
D.	A and B only	scram due to steam tunnel high temperature and Group I isolation.

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Part (1) is correct. Even though only two trains are required for 100% capacity and "C" SGT is powered from Unit 3, all three trains will initiate on a PCIS Group 2 or Group 6 isolation. Part (2) is incorrect. This was an issue due to the loss of Reactor building ventilation until a booster fan was recently installed in the steam tunnel to maintain temperature following a loss of normal ventilation.

c. Part (1) is incorrect. Even though only two trains are required for 100% capacity and "C" SGT is powered from Unit 3, all three trains will initiate on a PCIS Group 2 or Group 6 isolation. Part (2) is correct. Other than steam tunnel temperature, no other systems effected by the spurious isolations pose an immediate threat to continued operation.

d. Part (1) is incorrect. Even though only two trains are required for 100% capacity and "C" SGT is powered from Unit 3, all three trains will initiate on a PCIS Group 2 or Group 6 isolation. Part (2) is incorrect. This was an issue due to the loss of Reactor building ventilation until a booster fan was recently installed in the steam tunnel to maintain temperature following a loss of normal ventilation.

Technical Reference(s):	OPL171.017, PCIS	(Attach if not previously provided)
	OPL171.018, SGT OPL171.067, HVAC	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/12/2008 RMS	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Excerpt from OPL171.017 pages 16 & 18:

2. Group 2

This group includes Residual Heat Removal (RHR), drywell floor/equipment drain sump, and PSC Head Tank Pump suction valves.

The signals which will initiate a Group 2 isolation are:

RPV low level (+2"or Level 3)
Drywell High Pressure (+2.45 psig)

6. Group 6

This group provides for isolations of systems associated with Primary containment atmosphere control and sampling. Systems/lines isolated are as follows:

- Nitrogen/Air Purge
- Drywell/Suppression Chamber Exhaust
- Hydrogen/Oxygen Sample Lines
- Post Accident Sample System (PASS) Lines
- Drywell Air Compressors
- Drywell Leak Detection

The signals which will initiate a Group 6 isolation are:

- RPV Low Level (+2" or Level 3)
- Drywell High Pressure (2.45 psig)
- Reactor Bldg Vent Hi Radiation (72 mr/hr)
- Refuel Zone Hi Radiation (72 mr/hr)

Excerpt from OPL171.018 pages 21:

1. Initiation Signals

   a. System automatically starts with one or more of the following signals.

      (1) High drywell pressure (2.45 psi)

      (2) Low reactor water level: +2.0"

      (3) High radiation, Reactor Zone Ventilation System (72 mRem/hr) 2 monitors in the same channel or 2 monitors downscale (RE-142, -143) one in each channel.

      (4) High radiation, Refueling Zone (72 mRem/hr) 2 monitors in the same channel or 2 monitors downscale (RE-140, -141) one in each channel.

   These signals on any unit, will start all three SGT trains when the control switch is in AUTO

   .

   b. All 3 SGT trains auto-start on initiation and run until manually stopped.

   c. 2 of the 3 trains can provide design flow conditions.

   d. SGT will auto start with initiation signal as soon as power is available (slight delay until D/G powers SD Bd.)

   Train A gets its signals from DIV. I.

   Train B gets its signals from DIV. II.

   Train C gets its signals from both divisions.

Excerpt from OPL171.067 page 17:

1. Main Steam Vault Booster Fan

   a. Centrifugal booster fan installed in the Main Steam Vault exhaust duct to provide cooling during hot weather months or when normal ventilation is lost.

   b. This prevents unnecessary isolation of MSIVs due to ambient overheating. — Scram Frequency Reduction Effort

   c. Power Supplies:
   Unit-2 480V RMOV Bd 2C Compartment 5A
   Unit-3 480V RMOV Bd 3C Compartment 5A

   d. Fan operation is controlled from the breaker cubicle with a maintenance control switch located at the fan motor.

Examination Outline Cross-reference:	Level	RO	SRO
239002A4.02	Tier #	2	
Ability to manually operate and/or monitor in the control room: SRV Tail pipe Temperatures.	Group #	1	
	K/A #	239002A4.02	
	Importance Rating	3.6	3.7

Proposed Question: **RO # 16**

Unit-1 is operating at 100% rated power when the following annunciator alarmed:

MAIN STEAM RELIEF VALVES OPEN 1-FA-1-1 (9-3C W24)

Which ONE of the following describes the primary sensor that initiated the annunciator and one of the secondary indications which could be used to verify its accuracy?

The annunciator is initiated by the MSRV _____(1)_____ and can be verified using the MSRV _____(2)_____.

	(1)	(2)
A.	valve position indication	tail pipe temperature
B	valve position indication	tail pipe flow monitor
C.	tail pipe flow monitor	tail pipe temperature
D.	tail pipe temperature	tail pipe flow monitor

Proposed Answer: **C**

Explanation:

a. Part (1) is incorrect. Position indication is only indicating that the MSRV has received a demand to open, not its actual status. Part (2) is correct. Verifying the tail pipe temperature is responding to the demand is positive proof that the MSRV is passing steam from the RPV to the tail pipe.

b. Part (1) is incorrect as stated in (a) above. Part (2) is incorrect, but is also proof that the MSRV is passing steam from the RPV to the tail pipe. This would certainly be an indication to verify MSRV status, but it is also the input to the annunciator given in the stem of the question. Therefore, it is incorrect for the conditions given.

c. Correct answer.

d. Part (1) and (2) are incorrect. These are the two methods to positively verify MSRV operation, but the annunciator is based on tail pipe flow read by acoustic sensors in each tailpipe. Since a leaking (simmering) MSRV can also indicate tail pipe temperatures equal to an open MSRV, it is not used to annunciate an open MSRV.

Technical Reference(s):	1-ARP-9-3C Window 25		(Attach if not previously provided)
Proposed references to be provided to applicants during examination:			None
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/12/2008 RMS	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

BFN Unit 1	Panel 1-9-3 1-XA-55-3C	1-ARP-9-3C Rev. 0019 Page 28 of 38

MAIN STEAM RELIEF VALVES OPEN
1-FA-1-1
25

(Page 1 of 1)

Sensor/Trip Point:

1-FMT-1-4
MSRV Tailpipe Flow Monitor

Sensor Location: 1-FMT-1-4,Panel 1-9-3
Main Control Rm El 617'

Probable Cause: Main Steam Relief Valve is open or leaking.

Automatic Action: None

Operator Action:

A. CHECK MSRV Temp Recorder, 1-TR-1-1 on Panel 1-9-47 for raised temp and MSRV Tailpipe Flow Monitor, 1-FMT-1-4 on Panel 1-9-3 for flow indications. ☐
B. REFER TO 1-AOI-1-1. ☐
C. IF alarm is due to sensor malfunction, THEN REFER TO 0-OI-55 and OPDP-4. ☐

References: 1-47E610-1-1 1-45E620-2-1 GE 730E929-2

Examination Outline Cross-reference:	Level	RO	SRO
239002A4.04	Tier #	2	
Ability to manually operate and/or monitor in the control room: SRV Suppression Pool temperature.	Group #	1	
	K/A #	239002A4.04	
	Importance Rating	4.3	4.3

Proposed Question: **RO # 17**

Given the following Unit-3 plant conditions:

- A reactor scram has occurred due to a spurious Group I isolation.
- Over 50% of the control rods failed to insert on the scram.
- Reactor pressure is being maintained 800 to 1000 psig using MSRVs in accordance with 3-AOI-Appendix 11A, "Alternate RPV Pressure Control Systems, MSRVs."

Which ONE of the following describes the method of operating the MSRVs and the basis for the prescribed method?

MSRVs are opened in a ______(1)______ to ensure ________(2)__________.

	(1)	(2)
A.	numerical order by switch UNID	that both CAD tanks receive an equal loading.
B.	numerical order by switch UNID	even heat distribution in the Suppression Pool.
C.	specific order per the Appendix	that both CAD tanks receive an equal loading.
D.	specific order per the Appendix	even heat distribution in the Suppression Pool.

Proposed Answer: **D**

Explanation:

a. Part (1) is incorrect. Numerical order by switch position is a common procedure violation while performing Appendix 11A. Part (2) is incorrect for the given conditions. If Drywell Control Air is lost, MSRV operation is to ensure sufficient CAD tank volume is maintained. This is addressed in Appendix 11A.

b. Part (1) is incorrect as stated in (a) above. Part (2) is correct for the given conditions, but not for the answer given in Part (1) of this response.

c. Part (1) is correct. A specific order of opening is prescribed in the procedure. Part (2) is incorrect based on the given conditions. Drywell Control Air is available under these conditions unless a specific problem has occurred to cause it to be lost. No such problem was given, therefore Part (2) is incorrect.

d. Correct answer.

Technical Reference(s): 3-AOI-Appendix 11A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/12/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments: I evaluated this question as C/A because the candidate must determine, based on conditions in the stem, if Drywell Control Air is available.

3-EOI APPENDIX-11A
Rev. 2
Page 1 of 3

3-EOI APPENDIX-11A

ALTERNATE RPV PRESSURE CONTROL SYSTEMS MSRVs

LOCATION: Unit 3 Control Room

ATTACHMENTS: None

(√)

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.	____

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:

a.	6	3-PCV-1-42	MN STM LINE D RELIEF VALVE.	____
b.	7	3-PCV-1-30	MN STM LINE C RELIEF VALVE.	____
c.	4	3-PCV-1-31	MN STM LINE C RELIEF VALVE.	____
d.	13	3-PCV-1-34	MN STM LINE C RELIEF VALVE.	____
e.	10	3-PCV-1-41	MN STM LINE D RELIEF VALVE.	____
f.	2	3-PCV-1-180	MN STM LINE D RELIEF VALVE.	____
g.	12	3-PCV-1-18	MN STM LINE B RELIEF VALVE.	____

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:

a.	9	3-PCV-1-5	MN STM LINE A RELIEF VALVE.	____
b.	11	3-PCV-1-22	MN STM LINE B RELIEF VALVE.	____
c.	5	3-PCV-1-23	MN STM LINE B RELIEF VALVE.	____
d.	3	3-PCV-1-4	MN STM LINE A RELIEF VALVE.	____
e.	8	3-PCV-1-19	MN STM LINE B RELIEF VALVE.	____
f.	1	3-PCV-1-179	MN STM LINE A RELIEF VALVE.	____

6. IF ..... BOTH Drywell Control Air headers are depressurized,
THEN ... **PERFORM** the following as directed by EOI-1, RPV Control, RC/P Section:

- **PLACE** each MSRV control switch in CLOSE/AUTO, and **PLACE** 3-XS-1-202, MSRV AUTO ACTUATION LOGIC INHIBIT, to INHIBIT. ____

  AND

- **MINIMIZE** MSRV cycling by using sustained openings for RPV depressurization. ____

Examination Outline Cross-reference:	Level	RO	SRO
259002A4.01	Tier #	2	
Ability to manually operate and/or monitor in the control room: Reactor Water Level Control, All individual component controllers in the manual mode.	Group #	1	
	K/A #	259002A4.01	
	Importance Rating	3.8	3.6

Proposed Question: **RO # 18**

Given the following Unit 1 plant conditions:

- A reactor startup is in progress from cold conditions.
- The reactor is critical with a heat up in progress at 180 ^{O}F.
- The heat up rate is currently 80 ^{O}F/hr.
- 1A and 1B Condensate pumps and 1A Condensate Booster pump are operating.
- RPV level is (+) 30 inches and steady.

Which ONE of the following describes the appropriate method used to return RPV level to the normal control band under these conditions, and the reason for using that method?

A. CRD SYSTEM FLOW CONTROL in MANUAL can be used to raise injection flow to as high as 80 gpm.

B. RWCU BLDN FLOW CONT in MANUAL can be reduced to reject less water to the main condenser due to thermal expansion from the heat up.

C. RFW SU LVL CONT in AUTOMATIC can be used to control level and prevent distracting the OATC during the startup.

D. CNDS FLOW CONTROL SHORT CYCLE in MANUAL will raise Condensate Booster pump discharge pressure to raise injection flow.

Proposed Answer: **B**

Explanation:

a. Incorrect. Although the CRD flow controller can be adjusted to 80 gpm, this is NOT done in manual. In addition, doing so under this condition will eventually result in too much inventory and has a negative impact on CRD cooling water flow while its adjusted that high.

b. Correct answer. Also the preferred method.

c. Incorrect. Placing the S/U Level Control valve in automatic allows uncontrolled addition of water to the RPV which amounts to an uncontrolled reactivity addition under these conditions. Distracting the OATC takes a back seat to reactivity control.

d. This method will also raise RPV level but is not easily controlled. Therefore, this method is limited to controlling injection during a cooldown following a reactor shutdown.

Technical Reference(s): 1-OI-3, 1-OI-69, 1-OI-85, 1-GOI-100-1A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/13/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 1	Unit Startup	1-GOI-100-1A Rev. 0011 Page 93 of 173

5.0 INSTRUCTION STEPS (continued)

[45] **IF** Reactor pressure is less than 750 psig **AND** a RFP is NOT being used to maintain Reactor water level, **THEN**

MAINTAIN Reactor water level between 28 inches and 50 inches as indicated on RX VESSEL LEVEL/TOTAL FW FLOW recorder, 1-XR-3-53, and less than 48 inches on 1-LI-3-208A(B)(C)(D) using the following vessel makeup and level control systems: (N/A if RFP is being used to maintain Reactor water level)

- CRD System (40 to 65 gpm). (Control Rod Drive Hydraulic System Startup section of 1-OI-85).
- CRD System (up to 80 gpm). (CRD Pump Operation at Elevated Flow section of 1-OI-85).
- RWCU System. (1-OI-69).
- Condensate System. (1-OI-2).

(R) ____________ ____________ ____________
Initials Date Time

BFN Unit 1	Control Rod Drive System	1-OI-85 Rev. 0005 Page 60 of 179

6.10 CRD pump operation at elevated flow

[1] **VERIFY** CRD System in service in accordance with Section 5.1. ☐

[2] **REVIEW** all Precautions and Limitations in Section 3.1. ☐

CAUTIONS

1) Elevated flow rates are likely to reduce drive water and CRD cooling water D/P.

2) Elevated flows for extended periods (> 24 hours) are **NOT** recommended due to CRD Graphitar Seal erosion.

[3] **PERFORM** the following steps concurrently, as required, to establish a maximum of 80 gpm as indicated on CRD SYSTEM FLOW, 1-FIC-85-11:

- **THROTTLE** CRD PUMP DISCH THROTTLING VLV, 1-THV-085-0527, to maintain pressure less than or equal to 1500 psig as indicated on CRD ACCUM CHG WTR HDR PRESS, 1-PI-85-13A. ☐

- **ESTABLISH** the following by alternately adjusting the tape setpoint of the CRD SYSTEM FLOW CONTROL, 1-FIC-85-11, and the throttled position of the CRD DRIVE WATER PRESS CONTROL VLV, 1-HS-85-23A:

  - CRD CLG WTR HDR DP, 1-PDI-85-18A, of approximately 20 psid. ☐

  - CRD DRIVE WTR HDR DP, 1-PDI-85-17A, between 250 psid and 270 psid. ☐

[4] **CHECK** CRD STABILIZING FLOW, 1-FI-85-22 is approximately 6 gpm at 1-LPNL-925-0018B. ☐

[4.1] **IF** CRD Stabilizing Flow adjustment is necessary, **THEN**

CONTACT Technical Support and **REQUEST** performance of 0-TI-20 in order to adjust stabilizer needle valve settings. ☐

[5] **VERIFY** CRD DRIVE WTR HDR FLOW, 1-FI-85-15A is approximately 0 gpm. ☐

BFN Unit 1	Reactor Feedwater System	1-OI-3 Rev. 0009 Page 70 of 210

7.1 RFP/RFPT Shutdown (continued)

[10.4] With RFW SU LVL CONT, 1-LIC-3-53, in MANUAL. **ESTABLISH** approximately 70% demand signal on RFW SU LVL CONT, 1-LIC-3-53, using RAISE/LOWER push-buttons. ☐

[10.5] **VERIFY** the RFW START-UP LCV, 1-LCV-003-0053 is open via communications with the Operator at the valve. ☐

[10.6] **IF** required, **THEN**

HAVE the Operator open the RFW START-UP LCV 3-53 BYPASS, 1-BYV-003-0533. (**N/A** Otherwise) ☐

[10.7] **ADJUST** the CNDS FLOW CONTROL SHORT CYCLE, 1-FIC-2-29, as needed to raise or lower Condensate header pressure enabling vessel level control through the operating feedpump. ☐

[10.8] **WHEN** reactor pressure is approximately 270 psig, **THEN**

CLOSE 1-FCV-3-19(12)(5), RFP 1A(1B)(1C) DISCHARGE VALVE using handswitch 1-HS-3-19A(12A)(5A). ☐

[10.9] **ADJUST** the CNDS FLOW CONTROL SHORT CYCLE, 1-FIC-2-29, as needed to raise or lower Condensate header pressure enabling vessel level control. ☐

[10.10] **IF** the RFW START-UP LCV, 1-LCV-003-0053, fails to perform as expected **AND** makeup to the vessel is desired, **THEN**

RE-OPEN 1-FCV-3-19(12)(5), RFP 1A(1B)(1C) DISCHARGE VALVE using handswitch 1-HS-3-19A(12A)(5A) to re-establish feed to the vessel. (Otherwise **N/A**) ☐

[10.11] **WHEN** steady level has been obtained from the RFW START-UP LCV, 1-LCV-003-0053, **THEN**

DEPRESS RFPT 1A(1B)(1C) TRIP, 1-HS-3-125A(151A)(176A), to trip RFPT being removed from service. ☐

BFN Unit 1	Reactor Water Cleanup System	1-OI-69 Rev. 0037 Page 64 of 126

6.5 Blowdown Operation (continued)

NOTE

Blowdown to the Main Condenser is preferred to reduce Radwaste processing requirements.

CAUTIONS

1) During blowdown operation, failure to maintain the Non-Regenerative Heat Exchanger outlet temperature less than 130°F as indicated on 1-TR-69-6, will cause a reduction of resin efficiency and possible resin damage.
2) Opening RWCU MAIN CONDR BDV, 1-FCV-069-0016, and RWCU BLOWDOWN TO RADWASTE, 1-FCV-69-17, simultaneously during blowdown operations will result in a loss of condenser vacuum.
3) Failure to monitor Non-Regenerative Heat Exchanger inlet and outlet temperature to maintain less than 436°F across the heat exchanger during blowdown operations will cause damage to the Non-Regenerative Heat Exchanger. [BFPER 03-001802-000]
4) Failure to monitor Non-Regenerative Heat Exchanger outlet temperature closely during blowdown operations could result in an automatic RWCU isolation as sensed by RWCU NONREGEN HTX DISCH TEMP, 1-TIS-069-0011.
5) RWCU BLDN PRESS CNTL VLV, 1-PCV-069-0015, will automatically isolate on 5 psig low upstream pressure or 140 psig high downstream pressure.
6) Failure to closely monitor Reactor level during blowdown operations could result in uncontrollable level fluctuations.

[4] **IF** blowdown is to the Main Condenser, **THEN**

OPEN RWCU BLOWDOWN TO MAIN CNDR, using 1-HS-69-16A. ☐

Examination Outline Cross-reference:	Level	RO	SRO
261000K4.03	Tier #	2	
Knowledge of SGT design feature(s) and/or interlock(s) which provide for the following: Moisture Removal.	Group #	1	
	K/A #	261000K4.03	
	Importance Rating	2.5	2.7

Proposed Question: **RO # 19**

Regarding the Standby Gas Treatment (SGT) train, which ONE of the following describes the temperature set point that will trip the relative humidity heater and the basis for maintaining relative humidity within the SGT train?

The "SGT FILTER BK RH HTR CONT TEMPERATURE" annunciator will alarm and trip the Relative Humidity heater at a set point of ___(1)___ ^{O}F. Moisture is controlled within the SGT train to ___________(2)___________.

	(1)	(2)
A.	180	prevent lowering the adsorption properties of the charcoal.
B.	180	prevent damaging the charcoal and clogging the HEPA filter.
C.	80	prevent lowering the adsorption properties of the charcoal.
D.	80	prevent damaging the charcoal and clogging the HEPA filter.

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Part (1) is correct. The heater is tripped at 180 ^{O}F. Part (2) is incorrect. The basis of why the heater is tripped to prevent high temperatures is to prevent physical damage to the charcoal which would degrade it's mechanical filtration capability and also clog the HEPA filter, but it is NOT the basis for maintaining relative humidity.

c. Part (1) is incorrect. This value is the CHARCOAL BED VESSEL TEMP HIGH (9-52 W9) alarm for the charcoal bed in the Off-gas system, NOT the Standby Gas Treatment system. Part (2) is correct. This is the basis for removing moisture before the SGT flow enters the charcoal filter.

d. Part (1) is incorrect as stated in (c) above. Part (2) is incorrect as stated in (b) above.

Technical Reference(s): OPL171.030, OPL171.018 (Attach if not previously provided)

1-ARP-9-53

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/13/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BASES

BACKGROUND (continued)

The sizing of the SGT System equipment and components is based on the results of an infiltration analysis. The internal pressure of the SGT System boundary region is maintained at a negative pressure of 0.25 inches water gauge when the system is in operation. The Secondary Containment membrane limits infiltration to not more than the design flow requirements for the SGT System under neutral (< 5 mph) wind conditions. This allows the SGT System to evacuate the entire secondary containment volume to at least a negative 0.25 inches water gauge relative to outside the membrane.

The moisture separator is provided to remove entrained water in the air, while the electric heater reduces the relative humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides (however, no credit is taken in the radiological dose analyses for the charcoal), and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, the three charcoal filter train fans start and run until manually stopped. Two of the three subsystems can provide design flow conditions.

(continued)

Excerpt from OPL171.030 Appendix D:

Appendix D - CHARCOAL ADSORPTION PROCESS:

Activated, granulated charcoal provides a tremendous amount of surface area contact with many exposed chemically active sights.

Charcoal bed adsorption in our offgas system (including SBGT, containment purge units) is very effective in delay of noble gasses and even better delay on the halogen family (iodines).

Halogens such as iodine are delayed by chemical interaction with active sights in the activated charcoal. Halogens are very reactive with the hydrocarbon sights and actually combine chemically remaining as long as the halogen remains a halogen. Note that this is entirely dependent on the chemical properties and not on the nuclear properties. The charcoal beds will stop the halogens whether or not they are radioactive. The radioactive decay process of iodine normally beta decays to barium. Barium has totally different chemical properties and can actually combine with other iodine atoms. However barium will no longer remain chemically combined with the charcoal and thus would be released. The compounds which barium would form will be particulates which can be filtered out in the post filter. Not all halogens atoms pausing through the charcoal with actually combine chemically, but would still be delayed by the second interactive process described next.

Noble gasses and halogens also are delayed by electron cloud interaction with hydrocarbons. The hydrocarbon molecules normally present a slightly positive charge to the passing offgas molecules. The noble gasses and halogens are very electron rich thus having a slightly negative charge. This causes the entrained atoms and compounds to "stick" to the charcoal like Velcro. However, the gasses will still tumble along but be significantly delayed. The delay due to the chemical properties allows radioactive decay prior to release.

Note that halogens are very reactive prior to reaching the charcoal bed, so that a lot of the atoms will be already combined chemically with other atoms. This can prevent the chemical combination but will still be delayed by the electrostatic "Velcro" sticking.

Excerpt from OPL171.018 page 19:

i.	SGT FILTER BK A (B,C) RH HTR CONT TEMPERATURE TA 65-12A (-34A, -60)	>180˚F	Annunciation (U-1 and U-2 only) Turns off R-H heater control

Excerpt from OPL171.030 page 50:

y. Charcoal Bed vessel temperature high (80°F) Panel 9-53

BFN Unit 1	Panel 9-53 1-XA-55-53	1-ARP-9-53 Rev. 0014 Page 13 of 39

CHARCOAL BED VESSEL TEMP HIGH 1-TA-66-115

9

(Page 1 of 2)

Sensor/Trip Point:

1-TRS-66-115 80°F

Sensor Location: 1-TE-066-0115 A thru G
Off-Gas Bldg

Probable Cause:

A. OFFGAS REHEATER OUTLET TEMP CONTROL, 1-TIC-66-109, out of adjustment.

B. Adsorber vault air conditioning failure.
C. Carbon bed wetting due to high off-gas moisture.
D. Large quantities of adsorbed radioactive gas.

E. H_2 ignition in Off-Gas System.
F. Fire in charcoal vessel.
G. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **VERIFY** the following on Panel 1-9-53:
1. 1-TRS-66-115/(1-7), adsorber vessel and vault temperature, high. ☐
2. 1-TIC-66-109, OFFGAS REHEATER OUTLET TEMP CONTROL, 1-TIC-66-109, in AUTO and set at 77°F. ☐
3. 1-TRS-66-115/8, ABSORBER VAULT temperature,at about 70°F. ☐
4. REHEATER OUTLET (DEW POINT) temperature, 1-TRS-66-106/3, < 48°F. ☐
5. H_2 ANALYZER A (B) %H2, 1-XR-66-103/1(/2). **IF** H_2 ignition is suspected, **THEN** **REFER TO** 1-AOI-66-1. ☐

B. **VERIFY** Carbon Bed Rad Monitor, 1-RI-90-280, Panel 1-9-10 for normal reading. ☐

Continued on Next Page

Examination Outline Cross-reference:	Level	RO	SRO
262001K5.01	Tier #	2	
Knowledge of the operational implications of the following concepts as they apply to the AC Electrical Distribution: Principle involved with paralleling two A.C. sources.	Group #	1	
	K/A #	262001K5.01	
	Importance Rating	3.1	3.4

Proposed Question: **RO # 20**

Given the following plant conditions:

- Diesel Generator (DG) 3EA is running in parallel with the grid during the monthly load surveillance test.
- The DG Mode Selector Switch for 3EA DG is in the PARALLEL WITH SYSTEM position.
- 3EA DG load is currently 2400 KW and steady.

Which ONE of the following describes the expected response of 3EA DG if the DG Mode Selector Switch was moved to the UNITS IN PARALLEL position, and the basis for that response?

The 3EA DG would ________(1)________. This is a result of ____(2)____ speed droop control.

	(1)	(2)
A.	trip on Overload (51X)	zero
B.	trip on Overload (51X)	automatic
C.	continue to operate normally	automatic
D.	continue to operate normally	zero

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Part (1) is correct. Operating in "Parallel with System" allows automatic speed droop control which monitors DG speed, frequency AND load. When placed in "Units in Parallel" position, the speed droop control becomes "Zero" and the DG only monitors speed and frequency. Since the DG speed setpoint is maintained slightly above grid frequency during the surveillance, the governor will attempt to increase grid frequency by sending more and more fuel to the DG. The result is an overload condition. Part (2) is incorrect. The DG was in automatic speed droop control BEFORE the mode switch was moved out of "Parallel with System".

c. Part (1) is incorrect. The "Parallel with System" position is the only position that uses automatic speed droop control. Part (2) is incorrect. If the mode switch is placed in "Units in Parallel" or "Single Unit", speed droop control is set to zero and the DG will respond to grid frequency.

d. Part (1) is incorrect as stated in (c) above. Part (2) is correct for the switch position but would not result in continued operation of the DG.

Technical Reference(s): 0-OI-82, OPL171.038 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/15/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Excerpt from OPL171.038 page 20 and 21:

a. Speed Droop Control (Zero Droop Operation)

b.

(1) When the generator is the only power supply to a bus, it is desirable to have the speed governor maintain constant speed and frequency regardless of load on the bus. This is "Zero Droop" operation. The regulator system simply compares generator output frequency and setpoint frequency and actuates the fuel supply to maintain setpoint frequency. In effect for **Single Unit** and **Units in parallel.**

(2) **Zero Droop Operation** results in the speed/frequency remaining constant as KW load is increased.

(3) If the generator were to be tied to the grid, when in **Single Unit** or **Units in parallel**, as soon as the output breaker is shut the speed regulator senses output frequency, but now the generator output frequency is fixed by the other machines on the grid. If the diesel speed setpoint is higher than grid frequency, the zero droop governor will keep advancing the fuel supply to the diesel in order to try and raise grid/DG output frequency to the governor's setpoint. This will cause the diesel to overload. (495 amps.)

(4) **Droop operation** is in effect for **Parallel with System.** In droop mode the load carried by the diesel is sensed in addition to the output frequency. If the speed setpoint is higher than grid frequency, when the output breaker is shut the governor will see generator output frequency as being too low and start advancing fuel. This will cause the generator load to pick up. As load picks up, it sends a negative speed signal back to the regulator which cancels out the difference between grid frequency and setpoint frequency. When this happens the governor will stop advancing fuel and the engine will steady out at a certain amount of load. To pick up additional load the speed setpoint is adjusted upwards and the load builds up until it has canceled out the additional speed setpoint adjustment. If a droop mode generator was the sole supply to a board, its frequency versus kilowatt load would droop.

(5) Droop mode operation of the governor is controlled by the electronic governor and is in use **only** when the generator mode is "PARALLEL WITH SYSTEM."

(6) The governor control, when in parallel with the grid, serves to control the KILOWATT loading on the machine.

BFN Unit 0	Standby Diesel Generator System	0-OI-82 Rev. 0094 Page 72 of 178

8.1 Parallel with System Operation at Panel 9-23 (continued)

CAUTION

Only one Unit 1 and 2 Diesel Generator at a time is allowed to be operated in parallel with system.

[6] **PULL and PLACE** the associated Diesel Generator mode selector switch in PARALLELED WITH SYSTEM. ☐

Diesel	Handswitch Name	Handswitch No.	Panel
A	DG A MODE SELECT	0-HS-82-A/5A	0-9-23-7
B	DG B MODE SELECT	0-HS-82-B/5A	0-9-23-7
C	DG C MODE SELECT	0-HS-82-C/5A	0-9-23-8
D	DG D MODE SELECT	0-HS-82-D/5A	0-9-23-8

CAUTION

Failure of the PARALLELED WITH SYSTEM light to illuminate in the following step could indicate that the DG is still in SINGLE UNIT operation and result in overload when the DG output breaker is closed.

[7] **RELEASE** the Diesel Generator mode selector switch **and OBSERVE** PARALLELED WITH SYSTEM light illuminated. ☐

[8] **ADJUST** Diesel Generator frequency using the associated Diesel Generator governor control switch to obtain a synchroscope needle rotation of one revolution every 15 to 20 seconds in the FAST direction. ☐

Diesel	Handswitch Name	Handswitch No.	Panel
A	DG A GOVERNOR CONTROL	0-HS-82-A/3A	0-9-23-7
B	DG B GOVERNOR CONTROL	0-HS-82-B/3A	0-9-23-7
C	DG C GOVERNOR CONTROL	0-HS-82-C/3A	0-9-23-8
D	DG D GOVERNOR CONTROL	0-HS-82-D/3A	0-9-23-8

Examination Outline Cross-reference:	Level	RO	SRO
262002A1.02	Tier #	2	
Ability to predict and/or monitor changes in parameters associated with operating the UPC (AC/DC) controls including: Motor Generator outputs.	Group #	1	
	K/A #	262002A1.02	
	Importance Rating	2.5	2.9

Proposed Question: **RO # 21**

Given the following plant conditions:

- Unit 3 is in a normal lineup.
- The following alarm is received:
  - UNIT PFD SUPPLY ABNORMAL (9-8B W35)
- It is determined that the alarm is due to a Unit 3 Unit Preferred AC Generator Over-voltage condition

Which ONE of the following describes the result of this condition?

Unit 3 Breaker 1001 ____(1)____; Unit 2 Breaker 1003 ____(2)____; and the Motor-Motor-Generator (MMG) set ________(3)________.

	(1)	(2)	(3)
A.	trips OPEN;	is interlocked OPEN;	automatically shuts down.
B.	is interlocked OPEN;	trips OPEN;	automatically shuts down.
C.	trips OPEN;	is interlocked OPEN;	continues to run without excitation.
D.	is interlocked OPEN;	trips OPEN;	continues to run without excitation.

Proposed Answer: **C**

Explanation:

a. Part (1) and Part (2) are correct. Part (3) is incorrect. The MMG set does not automatically shut down.

b. Part (1) and Part (2) are incorrect. The breaker lineup is backward. Part (3) is incorrect. The MMG set does not automatically shut down.

c. Correct answer.

d. Part (1) and Part (2) are incorrect. The breaker lineup is backward. Part (3) is correct. The MMG set does not automatically shut down.

Technical Reference(s):	OPL171.102		(Attach if not previously provided)
	3-ARP-9-8B, (W 35)		

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #	262002A1.02	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	3/25/2008	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments: In order to answer this question correctly the candidate must determine the following:

1. The 1001 and 1003 breakers from an MMG set will trip on over voltage or under frequency at the output of the MMG.
2. Unit 2 MMG Breakers are interlocked to prevent alternate power to unit 1 and 3 at the same time.
3. When an over voltage condition exists at the Generator Output, the 1001 breaker from the MMG Set trips.
4. Excitation is lost and the MMG Set continues to run.
5. The "Hold to build up voltage" switch must be depressed to restore voltage.

Excerpt from OPL171.102 page 20 & 21:

b. MMG Sets (Unit 2&3)

(1) The MMG is normally driven By the AC motor, powered from 480V Shutdown Board A. Should this supply fail, the AC motor is automatically disconnected and the DC motor starts, powered from 250V Battery Board. The DC motor has an alternate power supply from another 250V Battery Board. Transfer to the alternate DC source is manual. Under frequency on the generator output will trip the DC motor. Transfer of the MMG set back to the AC motor is manual.

(2) The 1001 and 1003 breakers from an MMG set will trip on over voltage or under frequency at the output of the MMG. Also Unit 2 MMG Breakers are interlocked to prevent alternate power to unit 1 and 3 at the same time.

(3) When an under frequency or over voltage condition exists at the Generator Output the following occurs

(a) BB panel 10 breakers from the MMG Set trip.

U2	1001 (U2)	1003 (U1&3)
U3	1001 (U3)	1003 (U2)

(b) Excitation is lost and the MMG Set continues to run. (The Hold to build up voltage switch must be depressed to restore voltage.)

BFN Unit 3	Panel 9-8 3-XA-55-8B	3-ARP-9-8B Rev. 0014 Page 38 of 38

UNIT PFD SUPPLY ABNORMAL

35

(Page 1 of 1)

Sensor/Trip Point:

Relay SE -	loss of normal DC power source.
Relay TS -	DC transfer switch in emergency position.
Relay 2MS -	DC motor starts.
Relay 3k -	AC generator overvoltage, under frequency, or loss of output voltage.
Relay 13K -	difference between battery voltage and open circuit DC motor voltage greater than -7V DC or greater than +12V DC.

Sensor Location: 250V DC Battery Bd 3
El 593'

Probable Cause:

A. Loss of normal DC power source, UNIT PFD DC NORM/EMERG SELECTOR, 3-HS-252-02 in EMERG.
B. DC power transfer.
C. DC motor starts.
D. AC generator overvoltage.
E. AC generator under frequency.
F. AC generator loss of output voltage.
G. Relay failure.
H. Sensor malfunction.
I. Voltage differential between battery and motor is greater than -7V DC or greater than +12V DC.
J. Burned out light bulb on voltage indicating switch which activates Relay 13K.

Automatic Action: None

Operator Action:

A. IF 120V AC Unit Preferred is lost, THEN REFER TO 3-AOI-57-4. ☐
B. After determining situation, **REFER TO** appropriate portion of 0-OI-57C. ☐
C. IF voltage differential between battery and motor is greater than -7V DC or greater than +12V DC, **THEN** **PLACE** both local and remote OFF-AUTO-MANUAL select switch in MANUAL to prevent possible damage to DC drive motor. ☐
D. IF burned out light bulb is cause of alarm, **THEN** **INITIATE** WO to change out the light bulb due to it being hard wired. ☐

References: 0-45E641-2 3-45E620-11 3-3300D15A4585

Examination Outline Cross-reference:	Level	RO	SRO
263000A1.01	Tier #	2	
Ability to predict and/or monitor changes in parameters associated with operating the DC Electrical Distribution controls including: Battery charging/discharging rates.	Group #	1	
	K/A #	263000A1.01	
	Importance Rating	2.5	2.8

Proposed Question: **RO # 22**

Which ONE of the following describes the analyzed time in which the 250V Battery Board can carry a full electrical load without a battery charger connected, and the analyzed transient or accident that is the basis for that time?

The 250V DC battery board can carry a full load for ____(1)____. This capability is required to provide vital equipment power during a(an) ____(2)____.

	(1)	(2)
A.	30 minutes.	Appendix R fire.
B.	30 minutes.	Design Basis LOCA.
C.	60 minutes.	Appendix R fire.
D.	60 minutes.	Design Basis LOCA.

Proposed Answer: **B**

Explanation:

a. Part (1) is correct. Part (2) is incorrect. There are time sensitive actions during an Appendix R fire that must be taken to ensure the battery chargers are not lost, but the battery capacity is not the primary concern as it is during a DBA LOCA.

b. Correct answer.

c. Part (1) is incorrect. This is the time the battery is expected to carry electrical loads during normal operation. Part (2) is correct.

d. Part (1) is incorrect. This is the time the battery is expected to carry electrical loads during normal operation. Part (2) is incorrect. There are time sensitive actions during an Appendix R fire that must be taken to ensure the battery chargers are not lost, but the battery capacity is not the primary concern as it is during a DBA LOCA.

Technical Reference(s):	0-OI-57D, OPL171.037	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	06/15/2008 RMS	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0117 Page 16 of 247

3.0 **PRECAUTIONS AND LIMITATIONS (continued)**

E. All safety requirements concerning smoking, fires or sparks should be observed when in the Battery-Battery Board Rooms because of potential accumulation of hydrogen in flammable amounts.

F. 250V Unit Battery Charger 1,2A,2B and 3 Emergency ON select switch bypasses battery charger emergency load shed contacts. Placing the select switch in Emergency ON reestablishes charger operations with an accident signal present and Diesel Generator voltage available. Battery Charger 4 supply breaker, 480V Shutdown Board 3B, Compt 6D, receives a trip signal from the load shed logic and the breaker must be manually re-closed after a 40 second time delay to restore the charger to service. The annunciation circuit for the 250V Unit Battery Charger 3 does NOT work when the EMER/OFF/ON Select Switch is in the EMER Position.

G. [II/C] Neutron monitoring battery chargers are NOT stand alone power supplies and shall only be operated while connected to the neutron monitoring batteries. [BFPER 940662]

H. Within 30 minutes after the loss of the normal charger to a 250V Unit Battery another charger shall be placed in service to that battery and load reduced so that the battery is NOT discharging.

I. [NRC/C] Upon return to service of 24V DC Neutron Monitoring Battery A or B, Instrument Maintenance must perform functional tests on SRMs and IRMs that are powered from the affected battery board (In that the IRMs and SRMs are normally inoperable after entering RUN mode due to lack of testing, these tests are N/A for the IRMs and the SRMs if the Unit is in RUN Mode and the IRMs and SRMs are inoperable). Prior to calling the IRMs and SRMs operable, the tests have to be performed. [NRC IE Inspect Follow-up Item 86-40]

J. To return equipment to service following a failure or trip, the shutdown section of this instruction should be performed on the equipment failed. The initial conditions may NOT be applicable in this case.

K. [NRC/C] The transfer of 250VDC control power to a 4kV Shutdown Board with a diesel generator operating may cause an inadvertent start of a RHRSW pump. [LER 86021/25]

Excerpt from OPL171.037 page 13:

(2) Batteries

The 250 volt batteries are 120-cell lead-calcium type. The Unit Batteries (Mfg type LCUN-33) have a manufacturer 1 minute discharge rating of 2080 amps and an 8-hour discharge rating of 2320 amp-hours to a 210V DC minimum (required ECCS components must operate with as low as 200V). Two batteries can carry maximum expected load under DBA (Design Basis Accident) conditions without recharging for 30 minutes.

The Plant/Station Batteries (Mfg type LCR-33) have a manufacturer 1 minute discharge rating of 2240 amps and a 8 hours discharge rating of 2320 amp-hours.

BASES

BACKGROUND (continued)	Each battery charger for a DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 12 hours while supplying normal steady state loads (Ref. 4).
APPLICABLE SAFETY ANALYSES	The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter 6 (Ref. 5) and Chapter 14 (Ref. 5), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation. The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining DC sources OPERABLE during accident conditions in the event of: a. An assumed loss of all offsite AC power or all onsite AC power sources; and b. A postulated worst case single failure. The DC sources satisfy Criterion 3 of the NRC Policy Statement (Ref. 11).

(continued)

BFN Unit 0	Safe Shutdown Instructions	0-SSI-001 Rev. 0000 Page 88 of 97

Attachment 1
(Page 6 of 13)

TBD-31	Alignments of the 480V Control Bay Vent Board(s) and/or 480V HVAC Board B are necessary to ensure power is available to the guaranteed ventilation equipment.
TBD-32	Control Bay Chillers and Air Handling Units are necessary to maintain cooled areas at or below their design temperature limits.
TBD-33	An Electric Board Room AHU and Control Bay Chiller are placed in service to provide necessary cooling to Electric Board Rooms which have equipment required to support safe shutdown.
TBD-34	The CAD system is aligned to supply nitrogen to the main steam relief valve actuators as a backup supply to the accumulators for long term operation.
TBD-35	RHRSW pumps are started to provide RHRSW flow to RHR Heat Exchangers to be placed in service for decay heat removal.
TBD-36	FCV-23-34, 40, 46, or 52 will be opened to provide a cooling water flow path through the RHR Heat Exchanger for decay heat removal.
TBD-37	Battery Chargers are placed in service to provide long term DC power availability for designated MSRV solenoids and RPV instrumentation.
TBD-38	Lights are turned off in these rooms to reduce the heat load in the rooms.
TBD-39	I & C Bus transformers that are not designated safe shutdown equipment are removed from service to reduce the heat load in the associated rooms.
TBD-40	These doors are opened to provide natural circulation with adjacent areas. This will prevent or prolong the necessity of providing portable ventilation to these rooms.
TBD-41	Battery and Board Room Exhaust Fan 1A or 1B is started to provide ventilation to vital rooms on El. 593'. Board Room Supply Fan 1A or 1B is started to provide ventilation to Unit 1 El. 606' Mechanical Equipment Room. 3EA and/or 3EB Diesel Auxiliary Board Room Exhaust Fans are started to support operation of Diesel Auxiliary Boards on Unit 3.
TBD-42	This(These) Board(s) is(are) aligned to its(their) alternate power supply due to potential fire induced damage to the normal power supply.
TBD-43	480V RMOV Board B is aligned to remove power to FCV-23-46 and/or 52. This will allow unimpeded manual valve operation.

Examination Outline Cross-reference:	Level	RO	SRO
263000K5.01	Tier #	2	
Knowledge of the operational implications of the following concepts as they apply to the DC Electrical Distribution: Hydrogen Generation during battery charging.	Group #	1	
	K/A #	263000K5.01	
	Importance Rating	2.6	2.9

Proposed Question: **RO # 23**

Which ONE of the following is a concern to plant operation if the Plant/Station Battery Rooms HVAC units are not operating properly?

A. The design limit for hydrogen concentration in the rooms may be reached when the batteries are being charged.

B. Electrical Maintenance will not be able to obtain accurate Cell specific gravity readings if temperature is above 90 ^{O}F.

C. The lead-calcium batteries tend to release toxic gas into the atmosphere above 90 ^{O}F, and access to the room would be limited.

D. The Quarterly Battery SR frequency is lowered to weekly when temperatures are above the 70 ^{O}F to 90 ^{O}F temperature range.

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Incorrect. This would be correct for temperatures below 60 ^{O}F.

c. Incorrect. Lead-calcium batteries suffer degraded performance at high temperatures but do not release toxic gas as a result.

d. Incorrect. This would be correct if temperatures were below the temperature range, not above it.

Technical Reference(s): 0-OI-57D (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # 263000K5.01

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0117 Page 16 of 247

3.0 **PRECAUTIONS AND LIMITATIONS (continued)**

E. All safety requirements concerning smoking, fires or sparks should be observed when in the Battery-Battery Board Rooms because of potential accumulation of hydrogen in flammable amounts.

F. 250V Unit Battery Charger 1,2A,2B and 3 Emergency ON select switch bypasses battery charger emergency load shed contacts. Placing the select switch in Emergency ON reestablishes charger operations with an accident signal present and Diesel Generator voltage available. Battery Charger 4 supply breaker, 480V Shutdown Board 3B, Compt 6D, receives a trip signal from the load shed logic and the breaker must be manually re-closed after a 40 second time delay to restore the charger to service. The annunciation circuit for the 250V Unit Battery Charger 3 does NOT work when the EMER/OFF/ON Select Switch is in the EMER Position.

G. [II/C] Neutron monitoring battery chargers are NOT stand alone power supplies and shall only be operated while connected to the neutron monitoring batteries. [BFPER 940862]

H. Within 30 minutes after the loss of the normal charger to a 250V Unit Battery another charger shall be placed in service to that battery and load reduced so that the battery is NOT discharging.

I. [NRC/C] Upon return to service of 24V DC Neutron Monitoring Battery A or B, Instrument Maintenance must perform functional tests on SRMs and IRMs that are powered from the affected battery board (In that the IRMs and SRMs are normally inoperable after entering RUN mode due to lack of testing, these tests are N/A for the IRMs and the SRMs if the Unit is in RUN Mode and the IRMs and SRMs are inoperable). Prior to calling the IRMs and SRMs operable, the tests have to be performed. [NRC IE Inspect Follow-up item 86-40]

J. To return equipment to service following a failure or trip, the shutdown section of this instruction should be performed on the equipment failed. The initial conditions may NOT be applicable in this case.

K. [NRC/C] The transfer of 250VDC control power to a 4kV Shutdown Board with a diesel generator operating may cause an inadvertent start of a RHRSW pump. [LER 86021/25]

ACTIONS

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

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. Taking into consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable for operation prior to declaring the associated DC battery inoperable.

B.1

When any battery parameter is outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not ensured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing the Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells falling below 60°F for each Unit and Shutdown Board battery (except Shutdown Board battery 3EB) and 40°F for Shutdown Board battery 3EB and each DG battery, also are cause for immediately declaring the associated DC electrical power subsystem inoperable.

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
264000A3.05	Tier #	2	
Ability to monitor automatic operation of the EDGs including: Load shedding and sequencing.	Group #	1	
	K/A #	264000A3.05	
	Importance Rating	3.4	3.5

Proposed Question: **RO # 24**

Given the following Unit 3 plant conditions:

- Operating at 100% rated power with all equipment in a normal lineup.
- A total loss of all off-site power occurs in conjunction with a large break LOCA.
- Drywell pressure peaks at 22 psig and is subsequently lowered to 2.3 psig using Drywell Sprays.
- RPV pressure lowered to 400 psig and is stable.
- RPV level drops below (-) 122 inches.
- Assume no operator actions.

Which ONE of the following describes the expected response of the RHR pumps and SGT system as a result of these conditions?

When the DG output breakers close, RHR pumps will start ____(1)____. The B SGT fan ____(2)____.

	(1)	(2)
A.	in 7 seconds	auto starts ONLY if A SGT fan fails to start
B.	in 7 seconds	auto starts in 40 seconds
C.	immediately	auto starts ONLY if A SGT fan fails to start
D.	immediately	auto starts in 40 seconds

Proposed Answer: **D**

Explanation:

a. Part (1) is incorrect. This time is associated with starting Core Spray pumps. Part (2) is correct. Part (2) is incorrect. The B SGT fan will auto start in 40 seconds regardless of whether A SGT fan starts.

b. Part (1) is incorrect. This time is associated with starting Core Spray pumps. Part (2) is correct.

c. Part (1) is correct. Part (2) is incorrect. The B SGT fan will auto start in 40 seconds regardless of whether A SGT fan starts.

d. Correct answer.

Technical Reference(s): 0-AOI-57-1A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/15/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments: I rated this question as C/A due to conditions in the stem which indicate that conditions for a Common Accident Signal (CAS) may not be met since DW pressure dropped below the LOCA setpoint of 2.45 psig before RPV level dropped below -122 inches. The candidate must evaluate plant conditions to determine whether a Load Shed logic trip has been initiated.

BFN Unit 0	Loss of Offsite Power (161 and 500 KV)/Station Blackout	0-AOI-57-1A Rev. 0071 Page 7 of 71

3.0 **AUTOMATIC ACTIONS (continued)**

V. Unit 1/2 480V Load Shed occurs on a loss of offsite power in conjunction with a LOCA signal:

1. One RBCCW pump auto restarts (after 40 seconds on U1 and U2).

2. Drywell Blowers auto restart on non-accident unit (after 40 seconds). Drywell Blowers with their respective auto restart inhibit switches in the INHIBIT position will not auto restart.

3. Drywell coolers are manually restarted on the accident unit. A Drywell Blower with its auto restart inhibit switch in the INHIBIT position can be manually restarted after a ten minute time delay.

4. SGT TRAINS A & B trip, but will AUTO RESTART in 40 seconds when an initiation signal is present.

5. Loss of Control Bay Chilled Water Pumps A & B. (may be restarted after 10 minutes with use of bypass switch).

W. Unit 3 480V load shedding occurs as follows:

1. Division I 480V load shedding will occur when an accident signal is present and diesel generator voltage is available on the 4160V shutdown board supplying the 480V shutdown board 3A as follows:

   a. RBCCW pump 3A trips

   b. Drywell blowers 3A1 & 3A2 trip

   c. After a 40 second time delay, with the control switch in Normal After Start, RBCCW pump 3A restarts

   d. After a 40 second time delay, Drywell blowers 3A1 and 3A2 can be manually restarted

   e. Drywell blowers 3A3, 3A4 and 3A5 cannot be restarted until the load shed signal is corrected

Excerpt from OPL171.038 page 38:

a. If normal voltage is available, load will sequence on as follows: (**NVA**)

Time After Accident	S/D Board A	S/D Board C	S/D Board B	S/D Board D
0	RHR/CS A			
7		RHR/CS B		
14			RHR/CS C	
21				RHR/CS D
28	RHRSW*	RHRSW*	RHRSW*	RHRSW*

*RHRSW pumps assigned for EECW automatic start

b. **If normal voltage is NOT available: (DGVA)**

(1) After 5-second time delay, all 4kV Shutdown Board loads except 4160/480V transformer breakers are automatically tripped.

(2) Diesel generator output breaker closes when diesel is at speed.

(3) Loads sequence as indicated below

Time After Accident	S/D Board A	S/D Board B	S/D Board C	S/D Board D
0	RHR A	RHR C	RHR B	RHR D
7	CS A	CS C	CS B	CS D
14	RHRSW *	RHRSW *	RHRSW *	RHRSW *

*RHRSW pumps assigned for EECW automatic start

c. Certain 480V loads are shed whenever an accident signal is received in conjunction with the diesel generator tied to the board. (see OPL171.072)

Examination Outline Cross-reference:	Level	RO	SRO
300000K4.02	Tier #	2	
Knowledge of Instrument Air system design feature(s) and/or interlock(s) which provide for the following: Cross-over to other air systems.	Group #	1	
NOTE: Instrument Air at BFN is referred to as Control Air.	K/A #	300000K4.02	
	Importance Rating	3.0	3.0

Proposed Question: **RO # 25**

Given the following plant conditions:

- The Unit 3 Control Air system is aligned with the "G" Air compressor running and loaded.
- Subsequently, the Unit 3 Control Air system pressure falls to approximately 60 psig due to a rupture.

Which ONE of the following describes the final plant configuration pertaining to the Control Air system.

A. All Unit's Control Air system pressure will drop and a manual scram will be required for all units.

B. The effects on Unit 1 and 2 will not be as severe as on Unit 3 because of the automatic unit separation capability.

C. The effect on Unit 3 will not be experienced on Unit 1 and 2 because of the alignment of the closed manual header isolation valve between Unit 2 and 3.

D. All Unit's Control Air system pressures will drop, all Control Air Compressors will full load, and the Service Air Compressor will unload due to the Surge Condition experienced on Unit 3.

Proposed Answer: **B**

Explanation:

a. Incorrect. 2-PCV-032-3901 is installed on the main control air header between Unit 2 and Unit 3 and 1-PCV-032-3901 is installed on the main control air header between Unit 2 and Unit 1. Each valve automatically close when Control Air header pressure on either side of the valve drops below 65 psig. This action prevents a control air failure on any one Unit from resulting in a multi-unit scram. Illustration 3 provides more information on this feature.

b. Correct answer.

c. Incorrect. The valve between Unit 2 and 3 is not a manual isolation valve.

d. Incorrect. Partially true. All three unit's control air pressure will begin to drop, but will stabilize when the PCV between Unit 2 and Unit 3 closes.

Technical Reference(s): ______________________ (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # OPL171.054.10 minor format changes

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Original question OPL171.054.10:

The Unit 3 Control Air system is aligned with the "G" Air compressor running and loaded. Subsequently, the Unit 3 Control Air system pressure falls to approximately 60 psig due to a rupture.

Determine which one of the following describes the final plant configuration pertaining to the Control Air system.

A. Both Units Control Air system pressure will drop and a manual scram will be required for both units.

B. The effects on Unit 2 will not be as severe as on Unit 3 because of the automatic unit separation capability.

C. The effect on Unit 3 will not be experienced on Unit 2 because of the alignment of the closed manual header isolation valve between Unit 2 and 3.

D. Both Units Control Air system pressures will drop,all Control Air Compressors will full load, and the Service Air Compressor will unload due to the Surge Condition experienced on Unit 3.

BFN Unit 0	Control Air System	0-OI-32 Rev. 0114 Page 12 of 105

3.0 **PRECAUTIONS AND LIMITATIONS (continued)**

O. Control Air Compressor A, B, C, D Air Shutoff Valves To Unloaders, 0-SHV-032-4004A (B,C,D) and 0-SHV-032-4005A (B,C,D) can be used as an alternate method for manually loading Control Air Compressors A, B, C, D. Closing any one of these valves will cause the applicable compressor to Half Load. Closing both valves on the compressor will result in that compressor reaching Full Load. Section 8.4, Alternate Method for Manually Loading Control Air Compressors A, B, C, D, provides instruction for utilizing these valves to load A, B, C, D compressors.

P. Air flow should never be established through an air dryer unless power is supplied to the dryer.

Q. Section 6.5, Moisture Trap blowdown, is performed once per shift.

R. [QA/C] Header isolation valves 1-32-586 and 1-32-2378 should be closed during multi-unit operation so that a Control Air failure in Unit 1 will **NOT** result in the possibility of a scram of Unit 2. [CAQR BFP 910093].

S. 2-PCV-032-3901 is installed on the main control air header between Unit 2 and Unit 3 and 1-PCV-032-3901 is installed on the main control air header between Unit 2 and Unit 1. Each valve automatically close when Control Air header pressure on either side of the valve drops below 65 psig. This action prevents a control air failure on any one Unit from resulting in a multi-unit scram. Illustration 3 provides more information on this feature.

BFN Unit 0	Control Air System	0-OI-32 Rev. 0114 Page 72 of 105

Illustration 3
(Page 1 of 2)

Control Air System Unit Separation

Figure 1 (Normal Alignment)

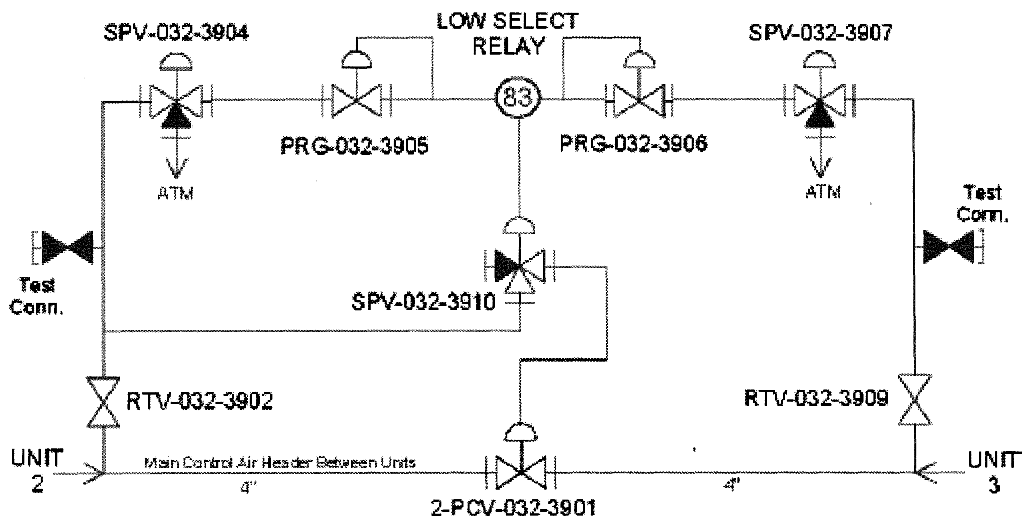
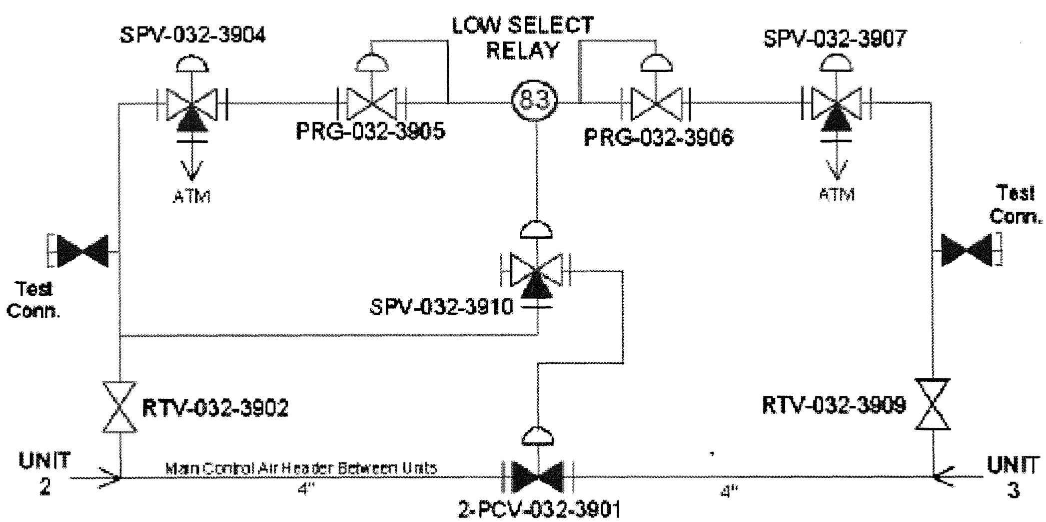


Figure 2 (Low Header Pressure Alignment)



Examination Outline Cross-reference:	Level	RO	SRO
400000K1.03	Tier #	2	
Knowledge of the physical connections and/or cause-effect relationships between Component Cooling Water system and the following: Radiation monitoring system.	Group #	1	
	K/A #	400000K1.03	
	Importance Rating	2.7	3.0

Proposed Question: **RO # 26**

Unit 2 is at rated power with the following indications:

- RECIRC PUMP MTR 'A' TEMP HIGH (9-4A W13).
- RECIRC PUMP MTR 'B' TEMP HIGH (9-4B W13).
- RBCCW EFFLUENT RADIATION HIGH (9-3A W17).
- RBCCW SURGE TANK LEVEL HIGH (9-4C W6).
- RX BLDG AREA RADIATION HIGH (9-3A W 22).
- Recirc Pump Motors "2A" and '2B' Winding and Bearing Temperature Recorder (2-TR-68-84) are reading 170 ^{O}F and 188 ^{O}F respectively and rising.
- RBCCW Pump Suction Header Temperature Indicator (2-TIS-70-3) is reading 94 ^{O}F and rising.
- RWCU NON-REGENERATIVE HX DISCH TEMP HIGH (9-4C W17).
- Area Radiation Monitor RE-90-13A and RE-90-14A are in alarm reading 55 mr/hr and rising.

Which ONE of the following describes the actions that should be taken in accordance with plant procedures?

REFERENCE PROVIDED

Enter 2-EOI-3, "Secondary Containment Control" and ______________________________

____.

A. Trip and isolate '2B' Recirc Pump.
Enter 2-AOI-68-1A "Recirc Pump Trip/Core Flow Decrease OPRMs Operable."

B. Trip and isolate '2B' Recirc Pump.
Commence a normal shutdown in accordance with 2-GOI-100-12A, "Unit Shutdown."

C. Trip RWCU pumps and isolate the RWCU system.
Commence a normal shutdown in accordance with 2-GOI-100-12A, "Unit Shutdown."

D. Trip RWCU pumps and isolate the RWCU system.
Close RBCCW Sectionalizing Valve 2-FCV-70-48 in accordance with 2-AOI-70-1, "Loss of RBCCW."

Proposed Answer: **D**

Explanation:

a. Incorrect. Although 2B Recirc pump motor temperature is high, insufficient RBCCW cooling to the motor could be the cause. This can be verified by the temperature of the 2A Recirc pump. Although not as high as the 2B pump, it is still well above normal for this condition, which leads to the conclusion that the cause of the high temperature is common to BOTH Recirc pumps and NOT an individual pump issue. The procedure, 2-AOI-68-1A would be correct if the Recirc pump was the cause o the problem.

b. Incorrect. The reason above explains why the Recirc pump is not the problem. Commencing a plant shutdown is directed by 2-EOI-3, but only if the cause of the leak into secondary containment is NOT isolated. In this case, there is no information given that the corrective actions taken will not be successful.

c. Incorrect. The action to trip and isolate RWCU is correct for the given indications. The action to commence a shutdown is incorrect for the reasons given in (b) above.

d. Correct answer.

Technical Reference(s):	2-EOI-3 flowchart, 2-ARP-9-4A & B (W13)	(Attach if not previously provided)
	2-ARP-9-3A (W17)	

Proposed references to be provided to applicants during examination: 2-EOI-3 flowchart

Question Source:	Bank #		
	Modified Bank #	RO 400000G2.4.31	Attached
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments: This question was used on the last NRC exam but has been significantly modified. It has been changed from Unit 3 to Unit 2 and the conditions in the stem have changed to provide a different source of the leak. Therefore, the answer is now different from the previous version of this question. I did not take credit for a previous NRC exam question in calculating the total number of questions from the last exam. If this is inappropriate, even given the substantial changes to the question, please let me know and I'll develop a different question.

Original Question RO 400000G2.4.31:

Unit 3 is at rated power with the following indications:

- RECIRC PUMP MTR 'B' TEMP HIGH (3-XA-9-4B, Window 13).
- RBCCW EFFLUENT RADIATION HIGH (3-XA-9-3A, Window 17).
- RBCCW SURGE TANK LEVEL HIGH (3-XA-9-4C, Window 6).
- RX BLDG AREA RADIATION HIGH (3-XA-9-3A, Window 22).
- Recirc Pump Motor '3B' Winding and Bearing Temperature Recorder (3-TR-68-84) is reading 170^{O}F and rising.
- RBCCW Pump Suction Header Temperature Indicator (3-TIS-70-3) is reading 104^{O}F and rising.
- RWCU NON-REGENERATIVE HX DISCH TEMP HIGH (3-XA-9-4C, Window 17).
- Area Radiation Monitor RE-90-13A and RE-90-14A are in alarm reading 55 mr/hr and rising.

Which ONE of the following describes the actions that should be taken in accordance with plant procedures?

REFERENCE PROVIDED

Enter 3-EOI-3, "Secondary Containment Control" and ________________________

____.

A. Trip and isolate '3B' Recirc Pump.
Enter 3-AOI-68-1A "Recirc Pump Trip/Core Flow Decrease OPRMs Operable."

B. Trip and isolate '3B' Recirc Pump.
Commence a normal shutdown in accordance with 3-GOI-100-12A, "Unit Shutdown."

C. Trip RWCU pumps and isolate the RWCU system.
Commence a normal shutdown in accordance with 3-GOI-100-12A, "Unit Shutdown."

D. Trip RWCU pumps and isolate the RWCU system.
Close RBCCW Sectionalizing Valve 3-FCV-70-48 to isolate non-essential loads and maximize cooling to '3B' Recirc Pump.

BFN Unit 2	Panel 9-4 2-XA-55-4A	2-ARP-9-4A Rev. 0034 Page 16 of 44

RECIRC PUMP MTR 2A TEMP HIGH 2-TA-68-58

13

(Page 1 of 2)

Sensor/Trip Point: Alarm is from 2-TR-68-58, Panel 2-9-21

2-TE-68-61A RECIRC PMP MTR 2A-THR BRG UPPER FACE (190°F)
2-TE-68-61C RECIRC PMP MTR 2A-THR BRG LOWER FACE (190°F)
2-TE-68-61E RECIRC PMP MTR 2A-UPPER GUIDE BRG (190°F)
2-TE-68-61N RECIRC PMP MTR 2A-LOWER GUIDE BRG (190°F)
2-TE-68-61G RECIRC PMP MTR 2A-MOTOR WINDING A (255°F)
2-TE-68-61J RECIRC PMP MTR 2A-MOTOR WINDING B (255°F)
2-TE-68-61L RECIRC PMP MTR 2A-MOTOR WINDING C (255°F)
2-TE-68-61T RECIRC PMP MTR 2A-SEAL NO. 2 CAVITY (NOTE 1)
2-TE-68-61U RECIRC PMP MTR 2A-SEAL NO. 1 CAVITY (NOTE 1)
2-TE-68-54 RECIRC PMP MTR 2A-CLG WTR FROM SEAL CLG (140°F)
2-TE-68-57 RECIRC PMP MTR 2A-CLG WTR FROM BRG (140°F)

NOTE

160°F with the original water seal (Model SU) installed or 180°F with the new water seal (Model N7500) installed.

Sensor Location: Temperature elements are located on Recirculation pump motor. Elevation 563.12, Unit 2 drywell.

Probable Cause:

A. Possible bearing failure.
B. Possible motor overload.
C. Insufficient cooling water.
D. Possible seal failure.
E. High drywell temperature.

Automatic Action: None

Operator Action:

A. **CHECK** following on Panel 2-9-4:
- RBCCW PUMP SUCTION HDR TEMP temperature indicating switch, 2-TIS-70-3 normal (summer 70-95°F, winter 60-80°F). ☐
- RBCCW PRI CTMT OUTLET handswitch, 2-HS-70-47A (2-FCV-70-47) OPEN. ☐

B. **CHECK** the temperature of the cooling water leaving the seal and bearing coolers <140°F on RECIRC PMP MTR 2A WINDING AND BRG TEMP temperature recorder, 2-TR-68-58 on Panel 2-9-21. ☐

C. **REDUCE** Recirc pump speed until the temperature drops below alarm setpoint. ☐

D. **CONTACT** System engineering to PERFORM a complete assessment and monitoring of all seal conditions particularly seal leakage, temperature, and pressure of all stages for Recirc Pump seal temperatures in excess of 200°F. ☐

Continued on Next Page

BFN Unit 2	Panel 9-4XA-55-4B	2-ARP-9-4B Rev. 0034 Page 17 of 46

RECIRC
PUMP MTR 2B
TEMP HIGH
2-TA-68-84
13

(Page 1 of 2)

Sensor/Trip Point:

Alarm is from 2-TR-68-84, Panel 2-9-21
2-TE-68-73A RECIRC PMP MTR 2B-THR BRG UPPER FACE (190°F)
2-TE-68-73C RECIRC PMP MTR 2B-THR BRG LOWER FACE (190°F)
2-TE-68-73E RECIRC PMP MTR 2B-UPPER GUIDE BRG (190°F)
2-TE-68-73N RECIRC PMP MTR 2B-LOWER GUIDE BRG (190°F)
2-TE-68-73G RECIRC PMP MTR 2B-MOTOR WINDING A (255°F)
2-TE-68-73J RECIRC PMP MTR 2B-MOTOR WINDING B (255°F)
2-TE-68-73L RECIRC PMP MTR 2B-MOTOR WINDING C (255°F)
2-TE-68-73T RECIRC PMP MTR 2B-SEAL NO. 2 CAVITY (Note 1)
2-TE-68-73U RECIRC PMP MTR 2B-SEAL NO. 1 CAVITY (Note 1)
2-TE-68-67 RECIRC PMP MTR 2B-CLG WTR FROM SEAL CLG (140°F)
2-TE-68-70 RECIRC PMP MTR 2B-CLG WTR FROM BRG (140°F)

NOTE

160°F with the original water seal (Model SU) installed or 180°F with the new water seal (Model N7500) installed.

Sensor Location: Temperature elements are located on Recirculation pump motor, Elevation 563.12, Unit 2 drywell.

Probable Cause:
A. Possible bearing failure.
B. Possible motor overload.
C. Insufficient cooling water.
D. Possible seal failure.
E. High drywell temperature.

Automatic Action: None

Operator Action:

A. CHECK following on Panel 2-9-4:
- RBCCW PUMP SUCTION HDR TEMP temperature indicating switch, 2-TIS-70-3 normal (summer 70-95°F, winter 60-80°F). ☐
- RBCCW PRI CTMT OUTLET handswitch, 2-HS-70-47A (2-FCV-70-47) OPEN. ☐

B. CHECK the temperature of the cooling water leaving the seal and bearing coolers < 140°F on RECIRC PMP MTR 2B WINDING AND BRG TEMP temperature recorder, 2-TR-68-84 on Panel 2-9-21. ☐

Continued on Next Page

BFN Unit 2	Panel 9-3 2-XA-55-3A	2-ARP-9-3A Rev. 0036 Page 25 of 50

RBCCW EFFLUENT RADIATION HIGH 2-RA-90-131A SOLID MAGENTA 17

(Page 1 of 2)

Sensor/Trip Point:

	HI	HI-HI
RM-90-131D	(NOTE 2)	(NOTE 2)

NOTE: HI alarm from recorder. HI HI alarm from drawer

Sensor Location: RE-90-131A RBCCW HX Rx Bldg, El 593, R-2 R-LINE

Probable Cause: HX tube leak into RBCCW system.

Automatic Action: None

Operator Action:

A. **DETERMINE** cause of alarm by observing following:
   1. RBCCW and RCW EFFLUENT RADIATION recorder, 2-RR-90-131/132 Red pen on Panel 2-9-2. ☐
   2. RBCCW EFFLUENT OFFLINE RAD MON radiation monitor, 2-RM-90-131D on Panel 2-9-10. ☐

B. **NOTIFY** Chemistry to sample RBCCW for total gamma activity. ☐

C. **DETERMINE** if source of leak is RWCU Non-regenerative, Fuel Pool Cooling, Reactor Water Sample or RWCU Recirc Pump A or B Seal Water heat exchanger(s). ☐

D. [NER/C] **CHECK** the following for indication of Reactor Recirculation Pump Seal Heat Exchanger leak:
   1. LOWERING Pressure in reactor Recirculation pump 2A(2B) No. 1 or 2 SEAL, 2-PI-68-64A or 2-PI-68-63A (2-PI-68-76A or 2-PI-68-75A) on Panel 2-9-4. ☐
   2. Rising Temperature on CLG WTR FROM SEAL CLG TE-68-54, on RECIRC PMP MTR 2A WINDING AND BRG TEMP temperature recorder, 2-TR-68-58, on Panel 2-9-21. ☐
   3. Rising Temperature on CLG WTR FROM SEAL CLG TE-68-67, on RECIRC PMP MTR 2B WINDING AND BRG TEMP temperature recorder, 2-TR-68-84, on Panel 2-9-21. ☐

Continued on Next Page

BFN Unit 2	Panel 9-3 2-XA-55-3A	2-ARP-9-3A Rev. 0036 Page 26 of 50

RBCCW EFFLUENT RADIATION HIGH 2-RA-90-131A, Window 17
(Page 2 of 2)

Operator Action: (Continued)

E. **IF** it is determined the source of leakage is from Reactor Recirc Pump A(B), **THEN**, at the discretion of the Shift Manager,

1. **ISOLATE** Reactor Recirculation Loop A(B). **REFER TO** 2-OI-68 . ☐

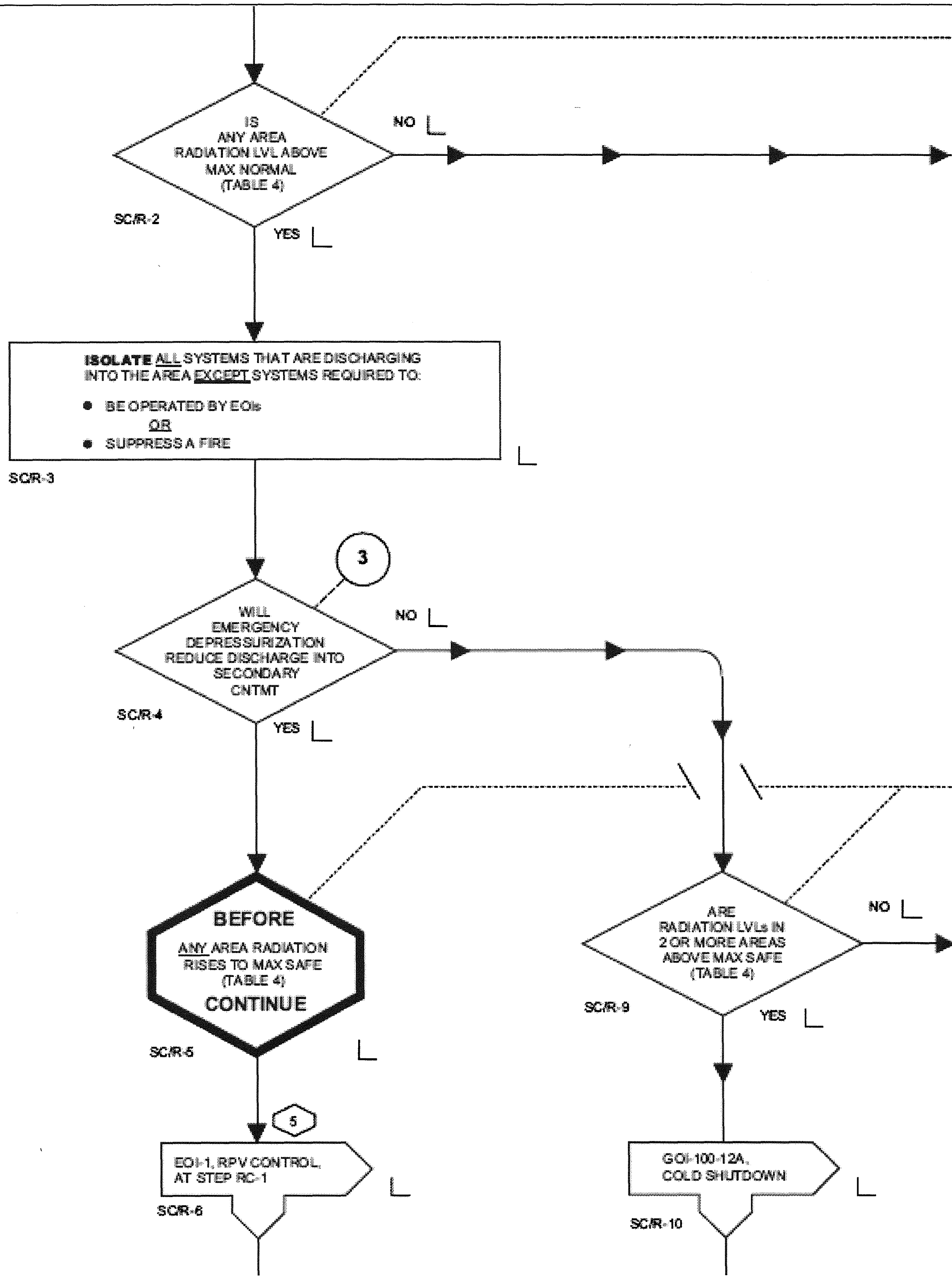
NOTE
Cooldown is required to prevent hangers or shock suppressers from exceeding their maximum travel range

2. **WHEN** primary system pressure is less than 125 psig, **THEN**, at the discretion of the Shift Manager,

   a. **ISOLATE** RBCCW System to preclude damage to RBCCW piping. [IEN 89-054, GE SIL-459] ☐

NOTE 2
Chemlab should be contacted for current setpoints per 0--TI-45.

References: 2-45E620-3 2-47E610-90-3 GE 2-729E814-3


IS ANY AREA RADIATION LVL ABOVE MAX NORMAL (TABLE 4)
NO
YES
SC/R-2
ISOLATE ALL SYSTEMS THAT ARE DISCHARGING INTO THE AREA EXCEPT SYSTEMS REQUIRED TO:
• BE OPERATED BY EOIs
OR
• SUPPRESS A FIRE
SC/R-3
3
WILL EMERGENCY DEPRESSURIZATION REDUCE DISCHARGE INTO SECONDARY CNTMT
NO
YES
SC/R-4
BEFORE ANY AREA RADIATION RISES TO MAX SAFE (TABLE 4) CONTINUE
SC/R-5
5
EOI-1, RPV CONTROL, AT STEP RC-1
SC/R-6
ARE RADIATION LVLs IN 2 OR MORE AREAS ABOVE MAX SAFE (TABLE 4)
NO
YES
SC/R-9
GOI-100-12A, COLD SHUTDOWN
SC/R-10

REFERENCE MATERIAL

Provided to

CANDIDATE

EOI-3
UNIT 2

EOI-3

RX ZONE VENTILATION EXH RADIATION ABOVE 72 MR/HR

REFUEL ZONE VENTILATION EXH RADIATION ABOVE 72 MR/HR

SECONDARY CNTMT FLOOR DRAIN SUMP WATER LVL ABOVE 64 IN.

TABLE 3
SECONDARY CNTMT AREA TEMP

AREA	PANEL 9-3 ALARM WINDOW (UNLESS NOTED)	PANEL 9-21 TEMP ELEMENT (UNLESS NOTED)	MAX NORMAL VALUE °F	MAX SAFE VALUE °F	POTENTIAL ISOLATION SOURCES
RHR SYS I PUMPS	XA-55-3E-4	74-95A	ALARMED	160	FCV-74-47, 48
RHR SYS II PUMPS	XA-55-3E-4	74-95B	ALARMED	210	FCV-74-47, 48
HPCI ROOM	XA-55-3F-10	73-55A	ALARMED	270	FCV-73-2, 3, 44, 81
CS SYS I PUMPS RCIC ROOM	XA-55-3D-10	71-41A	ALARMED	190	FCV-71-2, 3, 39
CS SYS II PUMPS	XA-55-3E-29	75-69B (PANEL 9-3)	ALARMED	150	NONE
TOP OF TORUS	XA-55-3D-10 XA-55-3F-10 XA-55-3E-4	71-41B, C, D 73-55B, C, D 74-95G, H	ALARMED ALARMED ALARMED	200 240 240	FCV-71-2, 3 FCV-73-2, 3, 81 FCV-74-47, 48
STEAM TUNNEL (RB)	XA-55-3D-24	1-60A (PANEL 9-3)	ALARMED	316	MSIVs FCV-71-2, 3, FCV-69-1, 2, 12
DW ACCESS	XA-55-3E-4	74-95E	ALARMED	170	FCV-74-47, 48
RB EL 565 W (RWCU PIPE TRENCH)	XA-55-5B-32 (PANEL 9-5) XA-55-5B-33 (PANEL 9-5)	69-835A, B, C, D (AUX INST ROOM)	ALARMED ALARMED	170	FCV-69-1, 2, 12
RWCU H. X. ROOM	XA-55-3D-17	69-29F, G, H	ALARMED	220	FCV-69-1, 2, 12
RWCU PUMP A	XA-55-3D-17	69-29D	ALARMED	215	FCV-69-1, 2, 12
RWCU PUMP B	XA-55-3D-17	69-29E	ALARMED	215	FCV-69-1, 2, 12
RB EL 593	XA-55-3E-4	74-95C, D	ALARMED	196	FCV-74-47, 48
RB EL 621	XA-55-3E-4	74-95F	ALARMED	166	FCV-43-13, 14

WHILE EXECUTING THIS PROCEDURE:

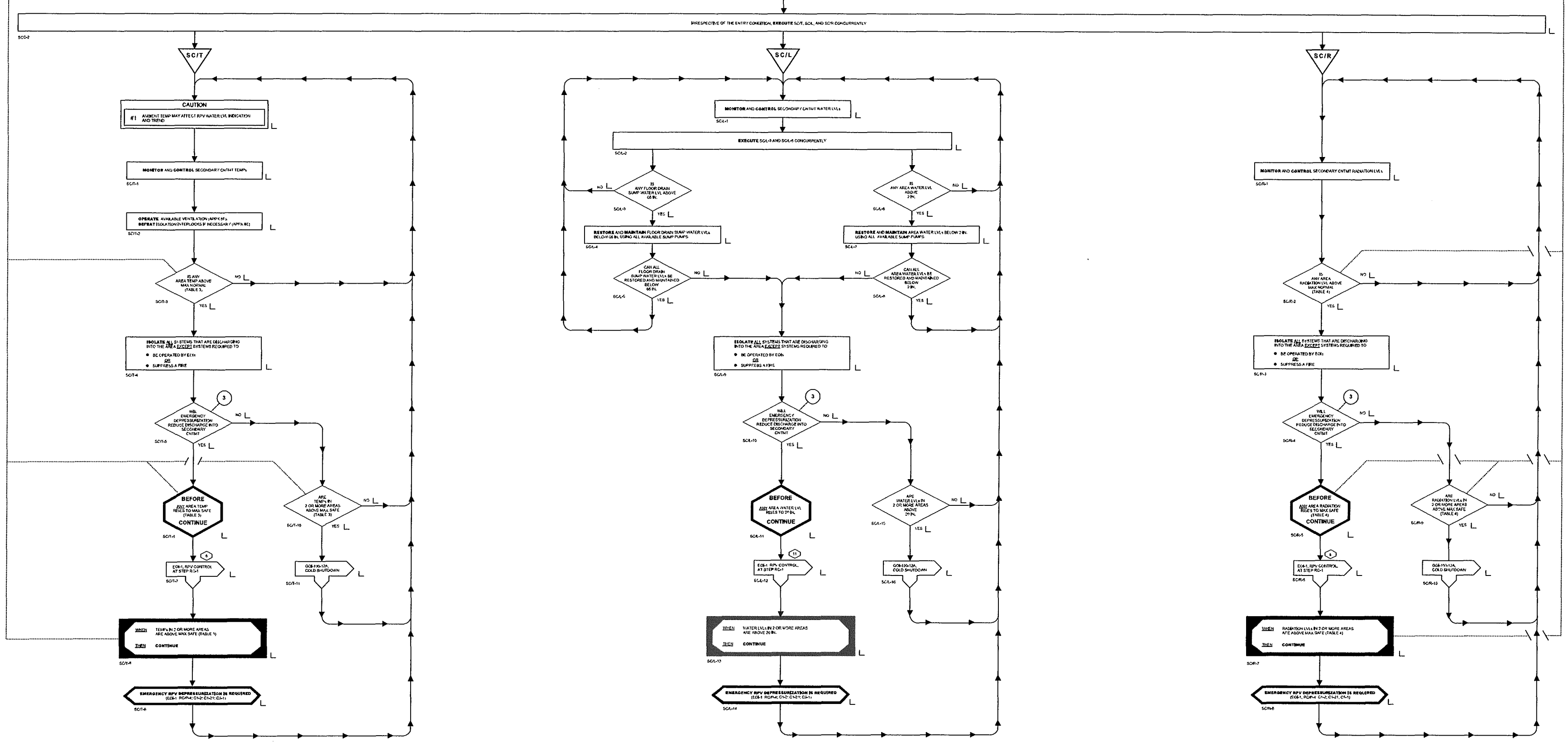
IF	THEN
SAMG ENTRY IS REQUIRED AND THE TSC SAM TEAM HAS ASSUMED COMMAND AND CONTROL	**EXIT** ALL EOI FLOWCHARTS AND **PERFORM** SAMG ACTIONS AS DIRECTED BY THE SAM TEAM
RX ZONE VENTILATION EXH RADIATION LVL IS ABOVE 72 MR/HR	**VERIFY** ISOLATION OF RX ZONE AND REFUEL ZONE VENTILATION AND **VERIFY** INIT OF SGTS
REFUEL ZONE VENTILATION EXH RADIATION LVL IS ABOVE 72 MR/HR	**VERIFY** ISOLATION OF REFUEL ZONE VENTILATION AND **VERIFY** INIT OF SGTS
RX ZONE VENTILATION IS ISOLATED AND RX ZONE VENTILATION EXH RADIATION LVL IS BELOW 72 MR/HR	**RESTART** RX ZONE VENTILATION (APPX 8F). **DEFEAT** ISOLATION INTERLOCKS IF NECESSARY (APPX 8E)
REFUEL ZONE VENTILATION IS ISOLATED AND REFUEL ZONE VENTILATION EXH RADIATION LVL IS BELOW 72 MR/HR	**RESTART** REFUEL ZONE VENTILATION (APPX 8F). **DEFEAT** ISOLATION INTERLOCKS IF NECESSARY (APPX 8E)

SCC-1

TABLE 4
SECONDARY CNTMT AREA RADIATION

AREA	APPLICABLE RADIATION INDICATORS	MAX NORMAL VALUE MR/HR	MAX SAFE VALUE MR/HR	POTENTIAL ISOLATION SOURCES
RHR SYS I PUMPS	90-25A	ALARMED	1000	FCV-74-47, 48
RHR SYS II PUMPS	90-28A	ALARMED	1000	FCV-74-47, 48
HPCI ROOM	90-24A	ALARMED	1000	FCV-73-2, 3, 44, 81
CS SYS I PUMPS RCIC ROOM	90-26A	ALARMED	1000	FCV-71-2, 3, 39
CS SYS II PUMPS	90-27A	ALARMED	1000	NONE
TOP OF TORUS GENERAL AREA	90-29A	ALARMED	1000	FCV-73-2, 3, 81 FCV-74-47, 48 FCV-71-2, 3
RB EL 565 W	90-20A	ALARMED	1000	FCV-69-1, 2, 12 SDV VENTS & DRAINS
RB EL 565 E	90-21A	ALARMED	1000	SDV VENTS & DRAINS
RB EL 565 NE	90-23A	ALARMED	1000	NONE
TIP ROOM	90-22A	ALARMED	100,000	TIP BALL VALVE
RB EL 593	90-13A, 14A	ALARMED	1000	FCV-74-47, 48
RB EL 621	90-9A	ALARMED	1000	FCV-43-13, 14
RECIRC MG SETS	90-4A	ALARMED	1000	NONE
REFUEL FLOOR	90-1A, 2A, 3A	ALARMED	1000	NONE

IRRESPECTIVE OF THE ENTRY CONDITION, **EXECUTE** SC/T, SC/L, AND SC/R CONCURRENTLY



CAUTIONS

CAUTION #1

- AN RPV WATER LVL INSTRUMENT MAY BE USED TO DETERMINE OR TREND LVL ONLY WHEN IT READS ABOVE THE MINIMUM INDICATED LVL ASSOCIATED WITH THE HIGHEST MAX DW OR SC RUN TEMP.
- IF DW TEMPS, OR SC AREA TEMPS (TABLE 6), AS APPLICABLE, ARE OUTSIDE THE SAFE REGION OF CURVE 8, THE ASSOCIATED INSTRUMENT MAY BE UNRELIABLE DUE TO BOILING IN THE RUN.

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR-64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A, B	EMERGENCY -155 TO +60	ON SCALE	N/A	BELOW 150
		-145	N/A	151 TO 200
		-140	N/A	201 TO 250
		-130	N/A	251 TO 300
		-120	N/A	301 TO 350
LI-3-53 LI-3-60 LI-3-206 LI-3-253 LI-3-208A, B, C, D	NORMAL 0 TO +60	ON SCALE	N/A	BELOW 150
		+5	N/A	151 TO 200
		+15	N/A	201 TO 250
		+20	N/A	251 TO 300
		+30	N/A	301 TO 350
LI-3-52 LI-3-52A	POST ACCIDENT -268 TO +32	ON SCALE	N/A	N/A
LI-3-55	SHUTDOWN FLOODUP 0 TO +400	+10	BELOW 100	N/A
		+15	100 TO 150	N/A
		+20	151 TO 200	N/A
		+30	201 TO 250	N/A
		+40	251 TO 300	N/A
		+50	301 TO 350	N/A
		+65	351 TO 400	N/A

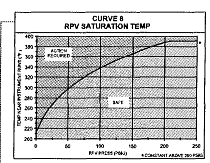


TABLE 6
SECONDARY CONTMT INSTRUMENT RUNS

INSTRUMENT	SC TEMP ELEMENTS AND LOCATIONS			
	EL 621 (74-95F)	EL 593 (74-95C AND D)	EL 565 (69-835A THRU D)	RWCU HXRM (69-29F, G, H)
LI-3-58A	°F	°F	N/A	°F
LI-3-58B	°F	°F	N/A	N/A
LI-3-53	°F	°F	N/A	°F
LI-3-60	°F	°F	N/A	N/A
LI-3-206	°F	°F	N/A	°F
LI-3-253	°F	°F	N/A	N/A
LI-3-52	°F	°F	°F	N/A
LI-3-52A	°F	°F	°F	N/A
LI-3-55	°F	°F	N/A	N/A
LI-3-208A, B	°F	°F	N/A	°F
LI-3-208C, D	°F	°F	N/A	N/A

NOTES

(3) TABLES 3 AND 4 CONTAIN INFORMATION THAT MAY BE USED TO DETERMINE IF A PRIMARY SYSTEM IS DISCHARGING INTO SECONDARY CNTMT (EMERGENCY DEPRESSURIZATION WILL REDUCE DISCHARGE)

EOI-3

2-EOI-3	PAGE 1 OF 1
SECONDARY CONTAINMENT CONTROL	
UNIT 2 BROWNS FERRY NUCLEAR PLANT	
REV: 11	

Examination Outline Cross-reference:	Level	RO	SRO
201001A2.04	Tier #	2	
Ability to (a) predict the impacts of the following on the CRD system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Scram conditions.	Group #	2	
	K/A #	201001A2.04	
	Importance Rating	3.8	3.9

Proposed Question: **RO # 27**

A scram has just occurred on Unit 2 with the following conditions:

- All blue Scram lights on the Full Core Display are energized.
- Three control rods indicate 48 in red and all remaining control rods indicate (- -) in green on the Full Core Display.
- All Accumulator Trouble lights and Rod Drift lights on the Full Core Display are energized.
- All IRMs are inserted and on Range 3 and lowering.

Which ONE of the following describes the current status of the CRD hydraulic system and the action(s) necessary to insert the remaining control rods?

CRD system flow is being directed to the charging water header and ______(1)______.
In order to insert the remaining control rods, the OATC must first ______(2)______.

REFERENCE PROVIDED

	(1)	(2)
A.	Scram Discharge Volume	close 2-85-586 in accordance with 2-EOI-Appendix 1D.
B.	Scram Discharge Volume	reset the RPS scram signal in accordance with 2-AOI-100-1.
C.	cooling water header	close 2-85-586 in accordance with 2-EOI-Appendix 1D.
D.	cooling water header	reset the RPS scram signal in accordance with 2-AOI-100-1.

Proposed Answer: **B**

Explanation:

a. Part (1) is correct. Part (2) is incorrect. Since the IRMs are below range 7, all actions to insert control rods are done using abnormal procedures, not emergency procedures.

b. Correct answer.

c. Part (1) is incorrect. Following a scram, the flow is prevented from going to the cooling water header because the CRD flow control valve closes. Part (2) is incorrect. Since the IRMs are below range 7, all actions to insert control rods are done using abnormal procedures, not emergency procedures.

d. Part (1) is incorrect as stated in (c) above. Part (2) is correct. Resetting the scram will allow the flow control valve to re-open and establish the required drive water pressure to allow control rod insertion. This is accomplished using 2-AOI-100-1.

Technical Reference(s): 2-AOI-100-1, (Attach if not previously provided)

2-EOI-1 Flowchart path RC/Q

Proposed references to be provided to applicants during examination: 2-EOI-1 flowchart

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/15/2008 RMS

Question History: Last NRC Exam

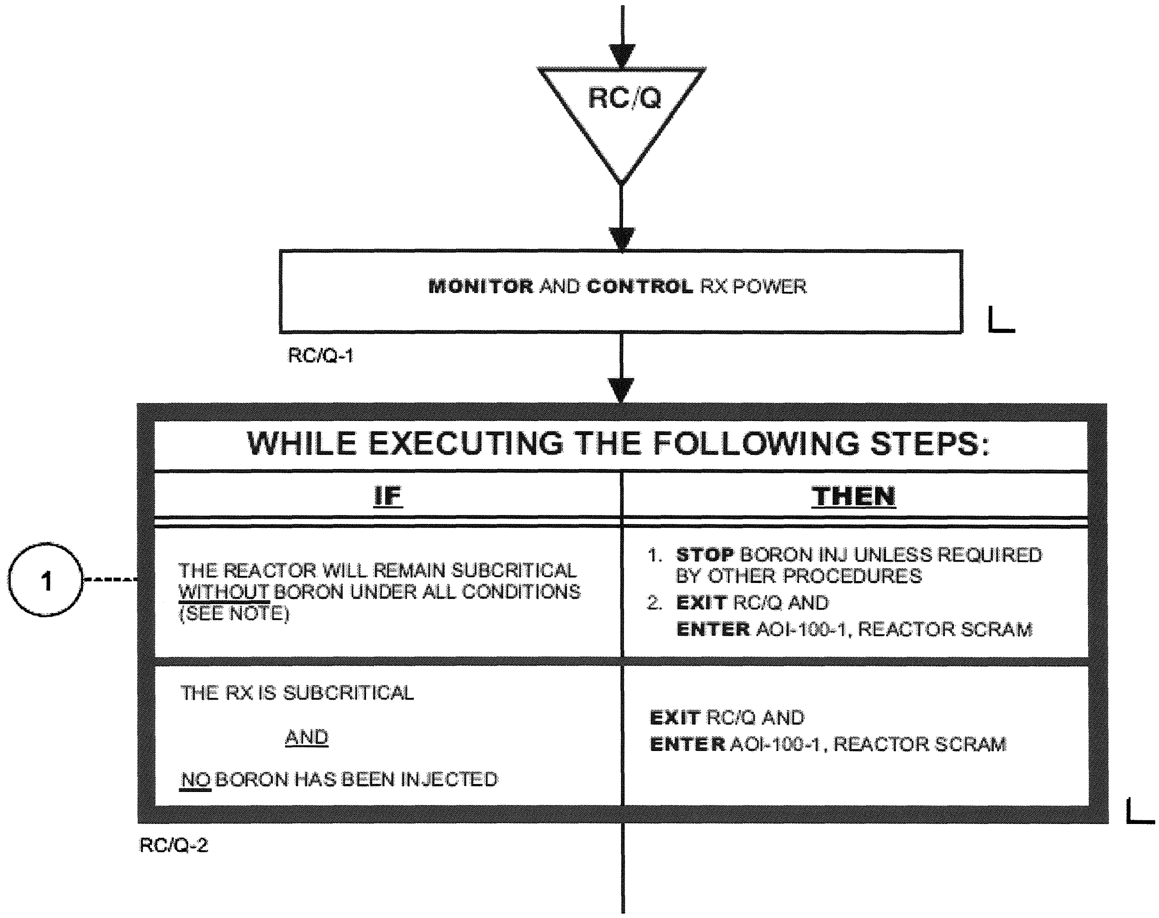
Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:



RC/Q
MONITOR AND CONTROL RX POWER
RC/Q-1
WHILE EXECUTING THE FOLLOWING STEPS:
IF
THEN
1
THE REACTOR WILL REMAIN SUBCRITICAL WITHOUT BORON UNDER ALL CONDITIONS (SEE NOTE)
1. STOP BORON INJ UNLESS REQUIRED BY OTHER PROCEDURES
2. EXIT RC/Q AND ENTER AOI-100-1, REACTOR SCRAM
THE RX IS SUBCRITICAL
AND
NO BORON HAS BEEN INJECTED
EXIT RC/Q AND ENTER AOI-100-1, REACTOR SCRAM
RC/Q-2

BFN Unit 2	Reactor Scram	2-AOI-100-1 Rev. 0087 Page 10 of 60

4.2 Subsequent Actions (continued)

NOTE
Step 4.2[8.1] may require support from off-site organizations and an extended period may elapse before results are obtained.

[8.1] **IF** all rods are **NOT** inserted to Position 02 or beyond, **THEN**

DIRECT Reactor Engineer to commence determination that reactor will remain subcritical under all conditions without boron. ☐

[9] [INPO/C] IF any control rod fails to fully insert and it is required to Re-scram, **THEN**

PERFORM the following, as required: [INPO SOER 80-006]

[9.1] **RESET** scram per Steps 4.2[22] thru 4.2[22.10]. ☐

[9.2] **VERIFY** WEST and EAST CRD DISCH VOL WTR LVL HIGH HALF SCRAM annunciators (2-XA-55-4A-1 and 4A-29) reset. ☐

[9.3] **INITIATE** manual scram. ☐

[9.4] **REPEAT** Step 4.2[9], as necessary, as long as rod motion is observed. ☐

[10] [INPO/C] IF any control rod fails to fully insert and it is required to Drive Control Rods, **THEN**

REFER TO 2-OI-85. [INPO SOER 80-006] ☐

[11] **IF** ANY PCIS isolation signal is received, **THEN**

VERIFY PCIS isolations using ANY of the following:

- Containment Isolation Status System on Panel 2-9-4 ☐
- PCIS Mimic and individual control switch indications ☐
- ICS ☐
- 2-OI-64 ☐

Excerpt from OPL171.005 page 48:

1. Scram

   a. Following a scram, but before the SDV is full, the control rod will be in an over travel-in position since there will still be a large differential pressure across the piston.

   b. Therefore, the green (full in) light on Panel 9-5 will be on but there will be no rod position readout displayed.

   c. After the SDV is full, there will be no differential pressure across the piston, and rod will settle into the 00 position.

Excerpt from OPL171.005 page 18:

(a) Runout protection

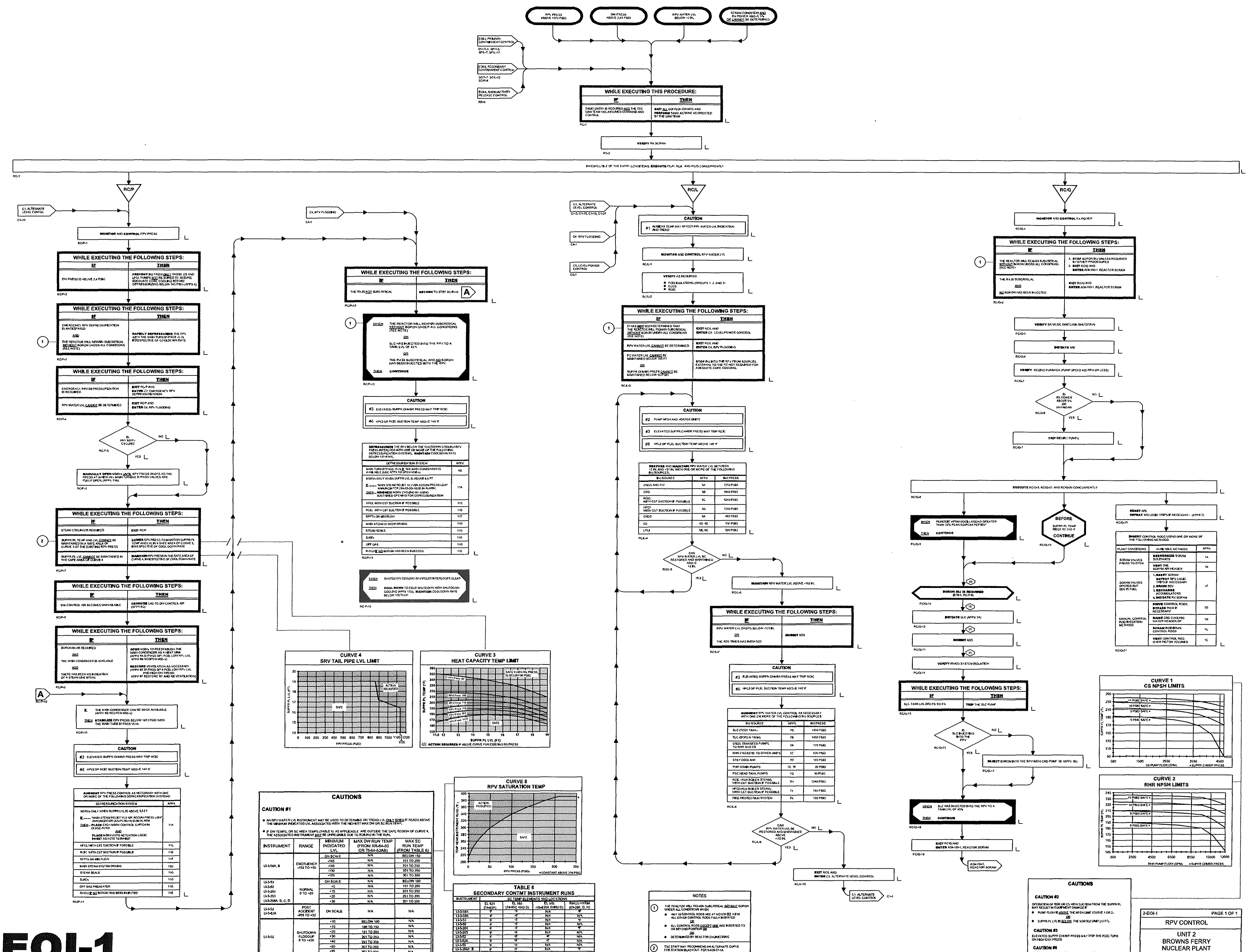
   i. During a scram, the HCU accumulators will be fully discharged.

   ii. The CRD pump will try to recharge all the accumulators at once. Flow through the charging header will cause the flow control valves to close.

REFERENCE MATERIAL

Provided to

CANDIDATE



CAUTIONS

CAUTION #2

OPERATION OF RHR OR CS WITH SUCTION FROM THE SUPPR PL MAY RESULT IN EQUIPMENT DAMAGE IF

- PUMP FLOW IS ABOVE THE NPSH LIMIT (CURVE 1 OR 2)

  OR

- SUPPR PL LVL IS BELOW THE VORTEX LIMIT (10 FT).

CAUTION #3

ELEVATED SUPPR CHMBR PRESS MAY TRIP THE RCIC TURB ON HIGH EXH PRESS

CAUTION #6

OPERATING HPCI OR RCIC TURBs WITH SUCTION TEMPs ABOVE 140 °F MAY RESULT IN EQUIPMENT DAMAGE

2-EOI-1	PAGE 1 OF 1
RPV CONTROL	
UNIT 2 BROWNS FERRY NUCLEAR PLANT	
REV: 12	

Examination Outline Cross-reference:	Level	RO	SRO
201003A4.02	Tier #	2	
Ability to manually operate and/or monitor in the control room: Control Rod and Drive Mechanism, CRD mechanism position.	Group #	2	
	K/A #	201003A4.02	
	Importance Rating	3.5	3.5

Proposed Question: **RO # 28**

Which ONE of the following combinations of the alarms and indications numbered below characterizes the possibility of an uncoupled control rod?

1. "Red" (- -) on the Full Core Display.
2. CONTROL ROD OVERTRAVEL (9-5A W14).
3. "Red" (48) on Four Rod Display.
4. CONTROL ROD DRIFT (9-5A W28)
5. Blank rod position indication on Four Rod Display.

A. 1, 2, 4

B. 2, 4, 5

C. 2, 3, 4

D. 1, 2, 5

Proposed Answer: **B**

Explanation:

a. Incorrect. The full core display will indicate blank just like the four rod display.

b. Correct answer.

c. Incorrect. The four rod display indicates blank.

d. Incorrect. The full core display will indicate blank just like the four rod display.

Technical Reference(s): 1-ARP-9-5A W14 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # OPL171.006.12

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 1	Panel 9-5 1-XA-55-5A	1-ARP-9-5A Rev. 0012 Page 18 of 43

CONTROL ROD OVERTRAVEL 14	Sensor/Trip Point:	
(Page 1 of 1)	Relay 3A-K3 Switch 50 in RPIS probe	Receive alarm when a control rod has been withdrawn past full out position.

Sensor Location: Panel 1-9-28
Elev. 593'
Control bldg.
Aux. Inst. Room

Probable Cause:
A. Rod is uncoupled.
B. Malfunction of sensor.
C. Fuse 1-FU1-85-5XA failure.

Automatic Action: The digital read out and background light will go out.

Operator Action:
A. **VALIDATE** alarm as follows:
1. Full core display will have no digital readout. ☐
2. Background light extinguished. ☐
3. Rod DRIFT light on. ☐

B. **IF** alarm is valid, **THEN** REFER TO 1-AOI-85-2. ☐
C. **NOTIFY** Reactor Engineer. ☐
D. REFER TO Tech Spec 3.1.3, 3.10.8.5, 3.3.2.1, Table 3/4.2.F and TRM Table 3.3.5-1. ☐

References: 1-45E620-6-1 1-730E321-10

BFN Unit 1	Uncoupled Control Rod	1-AOI-85-2 Rev. 0000 Page 4 of 8

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for an uncoupled control rod.

2.0 SYMPTOMS

NOTE

If a control rod is uncoupled and being withdrawn to any position other than position 48, the Rod Position Information System will display normal control rod movement. Power must be monitored to determine if the control rod is following its associated drive.

- Nuclear instrumentation does NOT respond to control rod movement.
- CONTROL ROD OVERTRAVEL annunciator (1-XA-55-5A, Window 14) in alarm.
- Digital display and red backlighting for the uncoupled control rod on the full core display is extinguished.

3.0 AUTOMATIC ACTIONS

None

BFN Unit 1	Panel 9-5 1-XA-55-5A	1-ARP-9-5A Rev. 0012 Page 35 of 43

CONTROL ROD DRIFT
28
(Page 1 of 1)

Sensor/Trip Point:

Relays 3A	-K37A,B,C,& D -K37A has 50 rods -K37B has 43 rods -K37C has 44 rods -K37D has 48 rods	Picked up by same relays for each individual rod drift.

Sensor Location: 1-PNLA-009-0028
Elev. 593'
Aux. Inst. Room
Control Bay 1C

Probable Cause:

A. Rod **NOT** selected, **NOT** being driven, and leaves even number reed switch (drift).
   1. Internal leakage of scram valves.
   2. Excessive cooling water pressure.

B. Rod settle timer timed out, and rod passes odd number reed switch.
   1. Drift-in following insert signal;
      a. Failure of insert valve to close.
   2. Drift-out following withdraw signal;
      a. Failure of withdraw valve to close.
      b. Failure of rod to latch after withdraw signal.

C. Defective RPIS alarm circuit.
D. Reactor SCRAM or Rod SCRAM.
E. Loss of 1 or more Rod Position Indications.

Automatic Action: None

Operator Action:

A. Determine which control rod is drifting from Full Core Display. ☐
B. IF control rod is drifting in, **THEN** REFER TO 1-AOI-85-5. ☐
C. IF control rod is drifting out, **THEN** REFER TO 1-AOI-85-6. ☐
D. REFER TO Tech Spec Section 3.1.3, 3.10.8. ☐

References: Technical Specifications 1-45E620-6-1 1-730E321-16

Examination Outline Cross-reference:	Level	RO	SRO
201006K5.10	Tier #	2	
Knowledge of the operational implications of the following concepts as they apply to the RWM: Withdraw error.	Group #	2	
	K/A #	201006K5.10	
	Importance Rating	3.2	3.3

Proposed Question: **RO # 29**

Unit 2 is starting up with the following plant conditions:

- Total steam flow is at 20% of rated flow.
- Rod Worth Minimizer (RWM) Group 22 is latched with limits from 00-04. (a double asterisk on the pull sheet applies for this group)
- The OATC selects the first rod in Group 22 and takes the ROD MOVEMENT CONTROL switch to the ROD OUT NOTCH position.
- The selected rod triple notches to position 06.

Which ONE of the following describes the RWM response to this condition and the reason for that response?

The RWM ROD BLOCK (9-5B W35) ____(1)____. The reason for that response is ____(2)____.

	(1)	(2)
A.	will alarm	a Withdraw Block is applied due to power below the Low Power Set Point (LPSP).
B.	will alarm	a Withdraw Block is applied since the single notch restraint limit for Group 22 control rods has been exceeded.
C.	will NOT alarm	Notch 06 is the Alternate Withdraw Limit for that control rod and will NOT result in a Withdraw Error.
D.	will NOT alarm	a Withdraw Block is NOT applied with power above the Low Power Set Point (LPSP).

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Part (1) is correct. On Unit-2, the LPSP is 24%. On Units 1 & 3 the LPSP is 16%. This is due to EPU differences. Part (2) is incorrect. Although the Group 22 rods are "single notch restricted" by the double asterisk on the pull sheet, this restriction is administrative and does NOT initiate any control rod blocks if exceeded. It is merely an added control to help enforce the Reduced Notch Worth Procedure (RNWP) restraints.

c. Part (1) is incorrect. This would be true for either Unit 1 or Unit 3, but not for Unit 2. Part (2) is incorrect. For group limits, RWM recognizes the Nominal Limits only. The Nominal Limit is the insert or withdraw limit for the group assigned by RWM. The Alternate Limit is no longer recognized by the RWM as an Acceptable Group Limit..

d. Part (1) and Part (2) are incorrect. This would be the correct answer for either Unit-1 or Unit 3.

Technical Reference(s):	2-OI-85, 1-OI-85, 3-OI-85		(Attach if not previously provided)
Proposed references to be provided to applicants during examination:			None
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/15/2008 RMS	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0104 Page 17 of 181

3.2 **Reactor Manual Control System (continued)**

J. Whenever there is fuel in the vessel, a peer check verification is required on all control rod selections, identification of final position, and verification of final position following movement, except as governed by the AOIs and/or EOIs. Peer check verification is required to be performed by an SRO, RO, STA, or Reactor Engineer.

K. While driving a Control Rod, if at any time a control rod moves unexpectedly more than two notches from its intended position, the control rod should be continuously inserted using the "EMERGENCY IN" switch. Notify the Control Room Unit Supervisor, Reactor Engineer, and obtain the Shift Manager's permission prior to resuming rod movement. If rod insertion to Position 00 is required and core thermal power is ≤ 10%, entry into LCO 3.1.6 may be required.

3.3 **Rod Worth Minimizer (RWM)**

A. The RWM System Rod Test/Touch screen function allows any one rod to be selected and moved to any position only if all other control rods are fully inserted. To get out of the rod test, the pushbutton needs to be depressed again (otherwise any single rod in any group can be selected and withdrawn).

B. [NER/C] When the RWM is bypassed, a second licensed operator, or other qualified member of the technical staff, is required to verify the Control Rod Sequence is followed. [INPO SOER-84-002]

C. 2-SR-3.3.2.1.7 is used to document independent verification of the RWM whenever the reactor is in startup or run, below 10% power.

D. [NER/C] Activities that can directly affect core reactivity are of a critical nature. Strict procedural compliance and conservative actions are required to be followed. [INPO SOER-84-002]

E. For RWM to enforce, Total Feedwater Flow or Total Steam Flow is required to be < 24%. To take RWM out of service automatically, Low Power Set Point (LPSP), Total Steam Flow AND Total Feedwater Flow is required to be > 24%.

The Low Power Alarm Point (LPAP) for the RWM is 27%, as sensed by Total Steam Flow. When the RWM is operating in the transition zone, between the LPSP (24%) and the LPAP (27%), no rod blocks are applied as a result of insert or withdraw errors, but the RWM will continue to provide alarm indications and error displays.

The monitoring functions of the RWM are automatically bypassed at power levels above the LPAP.

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0104 Page 18 of 181

3.3 Rod Worth Minimizer (RWM) (continued)

F. All the RWM blocks are applied in the event of a system hardware or software failure, when power is below the LPAP. At any Rx power, when a loss of ICS 2A occurs, a select block occurs due to the loss of power and cannot be bypassed using the RWM Bypass key.

G. An insert error occurs if:

1. A rod in the currently latched group is inserted past the insert limit for this group.

2. A rod in a group lower than the one that is presently latched is inserted past the withdraw limit for the lower group.

H. A withdraw error occurs if:

1. A rod in the currently latched group is withdrawn past the withdraw limit for the group.

2. A rod in a group lower than the one currently latched is withdrawn past the withdraw limit for its group.

3. A rod in a group higher than the one currently latched is withdrawn past the insert limit for its group.

I. A select error occurs if:

1. With the reactor operating below the LPAP, a rod other than one contained in the currently latched group is selected, unless conditions for latching up or down are met.

2. With a rod block applied, any rod other than an error rod is selected.

3. When operating in the Sequence Control Mode, a rod is skipped.

J. An insert block occurs if:

1. With two insert errors existing, a rod is moved to cause a third insert error.

2. A withdraw error has been made, a withdraw block applied, and a rod other than the withdraw error rod is selected.

K. A withdraw block occurs if:

1. A withdraw error is made.

2. With three insert errors existing and an insert block present, a rod other than one of the insert errors is selected.

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0104 Page 19 of 181

3.3 **Rod Worth Minimizer (RWM) (continued)**

L. A select block occurs if:

1. The RWM Bypass Switch is in normal and the RWM program is NOT running; i.e., following return to normal from bypass and the program has NOT been initialized.

2. The RWM Bypass Switch is in normal and the program stops due to software error.

M. For group limits only, RWM recognizes the Nominal Limits only. The Nominal Limit is the insert or withdraw limit for the group assigned by RWM. The Alternate Limit is no longer recognized by the RWM as an Acceptable Group Limit.

N. During RWM latching, the latched group will be the highest numbered group with 2 or less insert errors and having at least 1 rod withdrawn past its insert limits. With Sequence Control ON, latching occurs as follows. (Normally, startups are performed with Sequence Control ON).

1. RWM will latch down when all rods in the presently latched group have been inserted to the group insert limit and a rod in the next lower group is selected.

2. RWM will latch up when a rod within the next higher group is selected, provided that no more than two insert errors result.

   With Sequence Control OFF, latching occurs as follows:

3. For non-repeating groups, latching occurs as described above.

4. For repeating groups, latching occurs to the next setup or set down based on rod movement as opposed to rod selection.

O. Latching occurs at:

1. System initialization.

2. Following a "System Diagnostic" request.

3. When operator demands entry or termination of "Rod Test."

4. When power drops below LPAP.

5. When power drops below LPSP.

6. Every five seconds in the transition zone.

7. Following any full control rod scan when power is below LPAP.

BFN Unit 1	Control Rod Drive System	1-OI-85 Rev. 0005 Page 17 of 179

3.3 **Rod Worth Minimizer (RWM)**

A. The RWM system Rod Test/Touchscreen function allows any one rod to be selected and moved to any position only if all other control rods are fully inserted. To get out of the rod test, the pushbutton needs to be depressed again (otherwise any single rod in any group can be selected and withdrawn).

B. [NER/C] When the RWM is bypassed, a second licensed operator or other qualified member of the technical staff is required to verify the Control Rod Sequence is followed. [INPO SOER-84-002] (Not required with no fuel in RPV)

C. 1-SR-3.3.2.1.7 is used to document independent verification of the RWM whenever the reactor is in STARTUP or RUN, below 10% power.

D. [NER/C] Activities that can directly affect core reactivity are of a critical nature and strict procedural compliance, along with conservative actions, must be followed. [INPO SOER-84-002]

E. For RWM to enforce, Total Feedwater Flow or Total Steam Flow must be <16%. To take RWM out of service automatically, Low Power Set Point (LPSP), Total Steam Flow and Total Feedwater Flow must be >16%. The Low Power Alarm Point (LPAP) for the RWM is (21%) as sensed by Total Steam Flow. When the RWM is operating in the transition zone, between the LPSP (16%) and the LPAP (21%), no rod blocks will be applied as a result of insert or withdraw errors, but the RWM will continue to provide alarm indications and error displays.

F. The monitoring functions of the RWM are automatically bypassed at power levels above the LPAP.

G. All the RWM blocks will be applied in the event of a system hardware or software failure when power is below the LPAP. At any Reactor power, when a loss of ICS 1A occurs, a select block will occur due to the loss of power and cannot be bypassed using the RWM Bypass key.

H. An insert error occurs if:

1. A rod in the currently latched group is inserted past the insert limit for this group, OR

2. A rod in a group lower than the one that is presently latched is inserted past the withdraw limit for the lower group.

Examination Outline Cross-reference:	Level	RO	SRO
202002A4.05	Tier #	2	
Ability to manually operate and/or monitor in the control room: Recirculation Flow Control, Reactor level.	Group #	2	
	K/A #	202002A4.05	
	Importance Rating	3.4	3.4

Proposed Question: **RO # 30**

Unit 2 is at 100% rated power with the following plant conditions:

- 2C Reactor Feedwater Pump (RFP) tripped due to thrust bearing wear.
- RPV level lowered to (+) 24 inches before recovering to normal.

Which ONE of the following describes the Recirculation System response and the reason for that response?

The Reactor Recirculation Pumps (RRP) will _____(1)_____. The reason for that response is that _____(2)_____.

	(1)	(2)
A.	runback to 28% flow	total RFP flow is <19% AND one RRP discharge valve is <90% open.
B.	runback to 28% flow	total RFP flow is <19% OR one RRP discharge valve is <90% open.
C.	runback to 75% flow	individual RFP flow is <19% AND RPV level is <27 inches.
D.	runback to 75% flow	individual RFP flow is <19% OR RPV level is <27 inches.

Proposed Answer: **C**

Explanation:

a. Part (1) is incorrect. A single RFP trip from rated conditions will not result in a scram, therefore total RFP flow will not lower to <19%. Part (2) is incorrect. Even if total RFP flow dropped to <19%, either condition is all that is necessary to initiate the runback, therefore the "AND" makes this response incorrect.

b. Part (1) is incorrect as stated in (a) above. Part (2) would be the correct reason if Part (1) were correct for these conditions, but it is not.

c. Correct answer.

d. Part (1) is correct. Part (2) is incorrect. Both low RFP flow AND low RPV level are required to initiate the 75% runback.

Technical Reference(s): 2-OI-68 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/15/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 2	Reactor Recirculation System	2-OI-68 Rev. 0124 Page 14 of 166

3.0 **PRECAUTIONS AND LIMITATIONS (continued)**

M. Recirc pump controller limits are as follows:

1. When any individual RFP flow is < 19% and reactor water level is below 27 inches, or if Reactor Scram occurs, speed limit is set to 75% (~1130 RPM speed); and if speed is greater than 75% (~1130 RPM speed), recirc speed will run back to 75% (~1130 RPM speed).

2. When total feed water flow is < 19% (15 sec CA) or recirc pump discharge valve is less than 90% open, speed limit is set to 28% (~480 RPM speed); and if speed is greater than 28% (~480 RPM speed), recirc speed will run back to 28%(~480 RPM speed).

N. Observe the following recirc pump seal limitations:

1. The RBCCW System and CRD System provide cooling water to the recirc pump seals. As long as one of these systems is supporting recirc pump seals, the seals should remain functional and undamaged. If both systems fail to support recirc pump seals at normal operating conditions, recirc pump seals will begin to over heat in approximately seven minutes. (Refer to 2-OI-70 for RBCCW TCV adjustments).

2. When recirc pump seal temperatures are in excess of 200°F a complete assessment and monitoring of all seal conditions should be made; particularly seal leakage, temperature, and pressure of all stages.

3. [NER/C] For Model SU seals only, recirc pump seal life may be extended by minimizing operation of the recirc pumps at suction pressures below 300 psi, and minimizing recirc pump starts below 300 psi and ensuring recirc pumps are NOT operated with air in the seal cavities by thoroughly venting pump seals. BFN Unit 2 currently has N7500 Seals installed in both recirculation pumps. [GE SIL-203]

4. [II/C] Following recirculation pump seal maintenance, recirculation pump vents and drains are to remain open until seal purge is established and the seals are flushed and vented. This prevents small particles from entering the pump cavity and seals due to flooding without seal purge established. [BFPER 951603]

5. Seal purge alignment is to be secured (isolated) anytime drains backup and water is introduced onto the floor, and Radiation Protection notified.[PER 117112]

O. Actuation of 2-FSV-43-70 could cause erratic readings on Jet Pump No. 1 Flow Indication, 2-FI-68-15, if Reactor Recirculation System is in operation.

P. Keying a Radio while the recirc drive control cabinet door is open has caused recirc drive trips.

Examination Outline Cross-reference:	Level	RO	SRO
226001K2.02	Tier #	2	
Knowledge of electrical power supplies to the following: RHR/LPCI: CTMT Spray Mode, Pumps.	Group #	2	
	K/A #	226001K2.02	
	Importance Rating	2.9	2.9

Proposed Question: **RO # 31**

A LOCA has occurred on Unit 2 with the following plant conditions:

- RPV level is (-) 50 inches being restored with HPCI.
- RPV pressure is 700 psig and steady.
- Drywell pressure is 14 psig and lowering.
- RHR Loops I and II are operating in Drywell Spray mode in accordance with 2-EOI Appendix 17B, "RHR SYSTEM OPERATION DRYWELL SPRAYS."
- Unit 1 RPV level has just dropped below (-) 122 inches.

Which ONE of the following describes the Unit 2 RHR system response and the actions required to restore Drywell Sprays on Unit 2?

Unit 2 RHR pumps ____(1)____ will trip. RHR pumps ______(2)______ to re-establish Drywell Sprays on Unit 2.

	(1)	(2)
A.	2A and 2C	2B and 2D will continue to run and 2A and 2C can be restarted in 60 seconds
B.	2A and 2C	2B and 2D will continue to run and 2A and 2C can NOT be restarted until the Unit 1 CAS clears
C.	2A, 2B, 2C & 2D	2B and 2D can be restarted in 60 seconds. 2A and 2C can NOT be restarted until the Unit 1 CAS clears
D.	2A, 2B, 2C & 2D	2B and 2D can be restarted in 60 seconds. 2A and 2C can ALSO be restarted in 60 seconds

Proposed Answer: **D**

Explanation:

a. Part (1) is incorrect. Although these are the NON-PREFERRED pumps on Unit 2, all RHR pumps will trip since Unit 2 does not currently have an accident signal. Part (2) is incorrect. Unit 2 does not have an accident signal so 2B and 2C RHR pumps will trip.

b. Part (1) is incorrect as stated in (a) above. Part (2) is incorrect. This would be correct if Unit 2 also had an accident signal at the time of the Unit 1 accident signal.

c. Part (1) is correct. Part (2) is incorrect. All RHR pumps on the non-accident unit can be manually restarted after 60 seconds.

d. Correct answer.

Technical Reference(s): OPL171.044 pages 50 and 51 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/15/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Excerpt from OPL171.044 pages 50 and 51:

a. Accident Signal

(1) LOCA signals are divided into two separate signals, one referred to as a Pre Accident Signal (PAS) and the other referred to as a Common Accident Signal (CAS).

- PAS

  -122" Rx water level (Level 1)

  OR

  2.45 psig DW pressure

- CAS

  -122" Rx water level (Level 1)

  OR

  2.45 psig DW pressure **AND** <450 psig Rx pressure

(2) If a unit receives an accident signal, then all its respective RHR and Core Spray pumps will sequence on based upon power source to the SD Boards.

(3) All RHR and Core Spray pumps on the non-affected unit will trip (if running) and will be blocked from **manual** starting for 60 seconds.

(4) After 60 seconds all RHR pumps on the **non-affected** unit may be **manually** started.

(5) The **non-preferred** pumps on the **non-affected** unit are also prevented from **automatically** starting until the affected unit's accident signal is clear.

(6) The **preferred pumps** on the **non-affected** unit are locked out from automatically starting until the affected unit accident signal is clear **OR** the **non-affected** unit receives an accident signal.

Examination Outline Cross-reference:	Level	RO	SRO
233000A3.02	Tier #	2	
Ability to monitor automatic operation of the Fuel Pool Cooling/Cleanup system including: Pump trip(s).	Group #	2	
	K/A #	233000A3.02	
	Importance Rating	2.6	2.6

Proposed Question: **RO # 32**

Unit 2 is at 100% power with the following plant conditions:

- SIX MONTHS following a refueling outage.
- An electrical problem caused a trip of BOTH Fuel Pool Cooling Pumps approximately one hour ago.
- Pump restart is delayed.
- The Fuel Pool Temperature was 90 °F when the pumps tripped.

Which ONE of the following is the calculated time to reach the fuel pool temperature Operating Limit and Technical Requirements Manual Limit?

The Operating Limit will be reached in ___(1)___. The TRM limit will be reached in ___(2)___.

REFERENCE PROVIDED

	(1)	(2)
A.	16.7 hours.	32.3 hours.
B.	26.9 hours.	58.2 hours.
C.	26.9 hours.	64.9 hours.
D.	43.75.	52.9 hours.

Proposed Answer: **B**

Explanation:

a. Incorrect since this data is derived from information at 30 days after the start of the outage.

(125 - 90)/2.1 = 16.7 and 16.7 + 25/1.6 = 32.3

b. Correct answer. Uses data for 180 days after the start of the outage.

(125 - 90)/1.3 = 26.9 and 26.9 + 25/0.8 = 58.2

c. Incorrect since this uses data from 180 days after the start of the outage but the formula is performed such that the time to reach 125°F is added to 25°F and then divided by 0.8.

(125 - 90)/1.3 = 26.9 and (26.9 + 25)/0.8 = 64.9

d. Incorrect since this used data for 180 days but the X and Y values are switched.

(125-90)/0.8 = 43.75 and 43.75 + 25/1.3 = 52.9

Technical Reference(s):	2-AOI-78-1, Table 1	(Attach if not previously provided)
	TRM section 3.9.2	

Proposed references to be provided to applicants during examination: 2-AOI-78-1 Table 1 **w/o example**
TRM section 3.9.2

Question Source:	Bank #	233000A1.07	minor changes to format
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Student Handout Question #32

NOTE

This table is based on 2 year refuel cycle and a core "off-load" of ≈ 300 fuel bundles.

Table 1

Spent Fuel Pool Heat-up Rate at normal Fuel Pool level.

COLUMN A Decay Time Days	COLUMN B Rate to 125 degrees / hr X	COLUMN C Rate 125 to 150 degrees / hr Y	COLUMN D Max Temp
0	2.7	2.2	180 @ 90 hrs
30	2.1	1.6	168 @ 100 hrs
180	1.3	0.8	152 @ 144 hrs
365	1.0	0.8	152 @ 144 hrs
730(2 yr cycle)	1.0	0.8	152 @ 144 hrs

The information provided above is intended to cover all possible event scenarios. The heat up rates given are for a starting SFSP temperature of 90 degrees and are for the first hour without any cooling. They will provide a conservative estimate of the time to reach the given temperature.

4.2 Subsequent Actions (continued)

Use the following formula to determine time to reach 125°F.

Use Column A (# of days since the beginning of the last refueling outage) and B to determine current heatup rate.

$$\frac{125°F \text{ - Actual fuel pool temp (°F)}}{\textbf{X} \text{ (heatup rate determined from columns A and B (°F / hr))}} = \text{TIME (in hours) FOR FUEL POOL TO REACH 125°F.}$$

Use the following formula to determine time to reach 150°F.

Use Column A (# of days since the beginning of the last refueling outage) and C to determine current heatup rate

$$\underset{\text{(calculated above)}}{\textbf{Time to reach 125°F}} + \frac{25°F}{\textbf{Y} \text{ (heatup rate determined from columns A and C (°F / hr))}} = \text{TIME (in hours) FOR FUEL POOL TOREACH 150°F}$$

TR 3.9 REFUELING OPERATIONS

TR 3.9.2 Spent Fuel Pool Water Temperature

LCO 3.9.2 Fuel pool water temperature shall be ≤ 150° F.

APPLICABILITY: Whenever irradiated fuel is in the fuel pool

---NOTE---
TRM LCO 3.0.3 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Fuel pool water temperature > 150° F.	A.1 Initiate actions to lower the pool temperature.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.9.2.1	Whenever irradiated fuel is stored in the spent fuel pool, the temperature shall be measured and recorded daily.	24 hours

Examination Outline Cross-reference:	Level	RO	SRO
245000K4.05	Tier #	2	
Knowledge of Main Turbine Gen./Aux. system design feature(s) and/or interlock(s) which provide for the following: Turbine protection.	Group #	2	
	K/A #	245000K4.05	
	Importance Rating	2.9	3.0

Proposed Question: **RO # 33**

Which ONE of the following describes the reason for Extraction Non-Return Valve closure when a turbine trip signal is received?

A. Protect the heater tubes from excessive vibration when the steam flows back to the turbine.

B. Protect the turbine casing from over-pressurization when the steam flows back to the turbine.

C. Protect the moisture separators from over-pressurization on CIV closure.

D. Protect the turbine from overspeed when the steam flows back to the turbine.

Proposed Answer: **D**

Explanation:

a. Incorrect. Heater tubes normally operate in a steam environment.

b. Turbine casing pressure would lower to condenser vacuum when the turbine tripped.

c. Steam flow to the moisture separators stops when the turbine trips.

d. Correct answer.

Technical Reference(s): 1-OI-47 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # 295005AK3.05

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Excerpt from OPL171.010 page 28:

E. Extraction Non-return Valves

1. Purpose

To protect the turbine from an over-peed condition, which might occur when the turbine is tripped. A subsequent lowering of pressure in the turbine and heaters, due to vacuum in the condenser will cause hot water from the heater to flash to steam. The reverse steam flow back through the extraction steam piping to the Main Turbine could cause blade damage.

Examination Outline Cross-reference:	Level	RO	SRO
268000K3.04	Tier #	2	
Knowledge of the effect that a loss or malfunction of the Radwaste system will have on the following: Drain Sumps.	Group #	2	
	K/A #	268000K3.04	
	Importance Rating	2.7	2.8

Proposed Question: **RO # 34**

Given the following plant conditions:

- The Radwaste system Waste Collector Pump has failed due to bearing damage.
- Waste Collector Tank level is upscale at 38,000 gallons.

Which ONE of the following describes the action required to correct this problem and the effect on plant operation until it is completed?

To correct this problem perform the following:_____(1)_____. Until this action is complete, the ____(2)____ sump levels will continue to rise.

	(1)	(2)
A.	Lineup to pump the Waste Collector Tank to the Waste Surge Tank.	Drywell, Reactor Building and Turbine Building Floor Drain
B.	Cross-tie Waste Collector Pump suction to the Waste Surge Pump.	Drywell, Reactor Building and Turbine Building Floor Drain
C.	Lineup to pump the Waste Collector Tank to the Waste Surge Tank.	Drywell, Reactor Building and Turbine Building Equipment Drain
D.	Cross-tie Waste Collector Pump suction to the Waste Surge Pump.	Drywell, Reactor Building and Turbine Building Equipment Drain

Proposed Answer: **D**

Explanation:

a. Part (1) is incorrect. The Waste Collector Pump is required to complete that lineup. Part (2) is incorrect. Equipment Drains are directed to the Waste Collector Tank, not Floor Drains.

b. Part (1) is correct. A cross-tie line is provided with a normally closed manual valve. Part (2) is incorrect as stated in (a) above.

c. Part (1) is incorrect as stated in (a) above. Part (2) is correct.

d. Correct answer.

Technical Reference(s): OPL171.084 page 18 and TP-3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/15/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Excerpt from OPL171.084 page 18:

(2) Waste Collector pump and waste surge pump

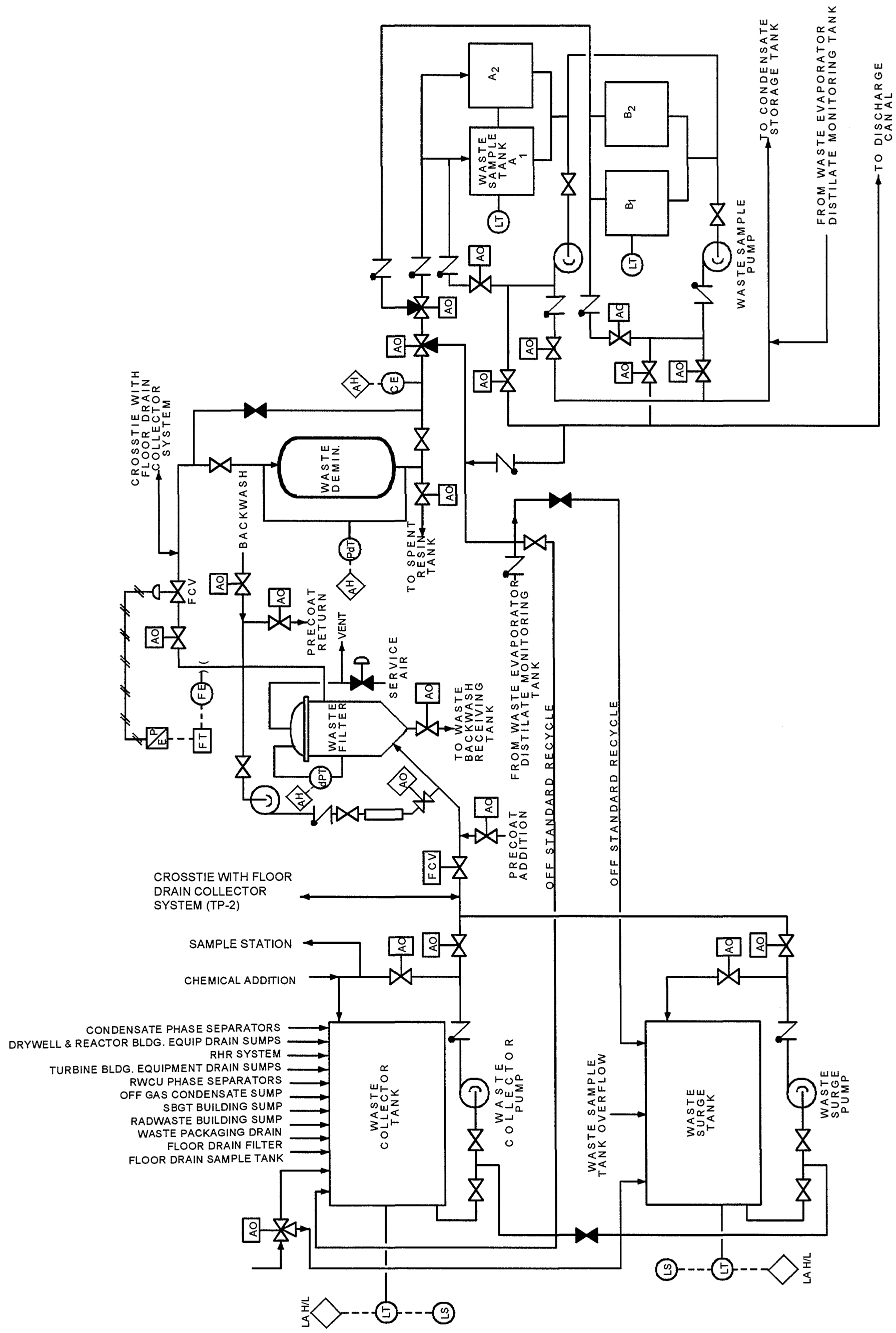
(a) 440 GPM centrifugal pumps

(b) Draw suction on waste collector and surge tanks

(c) Suction can be cross connected

(d) Both discharge to waste filter

Excerpt from OPL171.084 TP-3:



Examination Outline Cross-reference:	Level	RO	SRO
271000K6.02	Tier #	2	
Knowledge of the effect that a loss or malfunction of the following will have on the Off-Gas system: Process radiation monitoring system.	Group #	2	
	K/A #	271000K6.02	
	Importance Rating	3.0	3.2

Proposed Question: **RO # 35**

Unit 2 is in Mode 2 during a reactor startup at ~ 2% power.

The Off-gas system is aligned as follows:

- Adsorber control switch (HS-66-113) AUTO
- "A" SJAE is in service.
- Adsorber train "A" inlet (FCV 66-113A) CLOSED
- Adsorber bypass valve (FCV 66-113B) OPEN
- Off-gas system isolation valve (FCV 66-28) OPEN

Which ONE of the following describes the effect on the Off-gas system alignment should one of the OG Post-Treatment radiation monitors fail upscale?

The Adsorber INLET valve (66-113A) will ____(1)____. The Adsorber BYPASS valve (66-113B) will ____(2)____. The Off-gas isolation valve (66-28) will ____(3)____.

	(1)	(2)	(3)
A.	open	close	remain open
B.	open	close	close
C.	remain closed	remain open	remain open
D.	remain closed	remains open	close

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Part (1) and (2) are correct. Part (3) is incorrect. The Off-gas isolation valve (66-28) requires BOTH channels to initiate an isolation.

c. Part (1) and (2) are incorrect with the Adsorber control switch (HS-66-113) in AUTO. Part (3) is correct.

d. Part (1) and (2) are incorrect with the Adsorber control switch (HS-66-113) in AUTO. Part (3) is incorrect. The Off-gas isolation valve (66-28) requires BOTH channels to initiate an isolation.

Technical Reference(s):	2-AOI-66-2		(Attach if not previously provided)
Proposed references to be provided to applicants during examination:			None
Question Source:	Bank #	271000K4.07	minor format changes
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

BFN Unit 2	Offgas Post-Treatment Radiation HI-HI-HI	2-AOI-66-2 Rev. 0020 Page 5 of 9

3.0 **AUTOMATIC ACTIONS**

A. If the OFFGAS TREATMENT SELECT handswitch, 2-XS-66-113, Panel 9-53, is in AUTO when High radiation condition exists it will automatically align, or ensure alignment of, the charcoal adsorbers to the treatment mode, i.e., the charcoal inlet valve will receive an open signal and the charcoal bypass valve will receive a close signal.

B. OFFGAS SYSTEM ISOLATION VALVE, 2-FCV-66-28, automatically closes on any combination of Off Gas Post Treatment Hi Hi Hi, downscale, or inop simultaneously in both channels of the O.G. post treatment radiation monitoring system after 5 seconds. 2-FCV-066-0028 will not perform it's design function to automatically close, when it is mechanically restrained open due to plant conditions.

Original question 271000K4.07:

Unit 2 is in Mode 2 during a reactor startup at ~ 2% power
The Off-gas system is aligned as follows:

- Adsorber control switch (HS-66-113) in AUTO
- "A" SJAE in service
- Adsorber train "A" inlet (FCV 66-113A) closed
- Adsorber bypass valve (FCV 66-113B) open
- Off gas system isolation valve (FCV 66-28) open

How will the Off-gas system alignment be affected should one of the OG Post-Treat radiation monitors fail high?

A. Adsorber inlet valve (66-113A) remains closed, Adsorber bypass valve (66-113B) remains open, Off-gas isolation valve (66-28) closes

B. Adsorber inlet valve (66-113A) opens, Adsorber bypass valve (66-113B) closes, Off-gas isolation valve (66-28) closes

C. Adsorber inlet valve (66-113A) remains closed, Adsorber bypass valve (66-113B) remains open, Off-gas isolation valve (66-28) remains open

D. Adsorber inlet valve (66-113A) opens, Adsorber bypass valve (66-113B) closes, Off-gas isolation valve (66-28) remains open

Examination Outline Cross-reference:	Level	RO	SRO
288000G2.2.44	Tier #	2	
Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions: Plant Ventilation.	Group #	2	
	K/A #	288000G2.2.44	
	Importance Rating	4.2	4.4

Proposed Question: **RO # 36**

Unit 2 is operating at 100% rated power during the summer with the following conditions:

- Control Bay Chiller "A" breaker indicates closed and reading 32 amps on Panel 9-20.
- Control Bay Chiller "B" breaker indicates closed and reading 0 amps on Panel 9-20.
- Control Bay chilled water inlet temperature is indicating 42 ^{O}F on the ICS computer.

Which ONE of the following describes the effect of transferring the CONTROL PANEL MODE SELECT switch from "DIGITAL" to "ANALOG" on the CONTROL BAY CHILLER A LOCAL DISPLAY PANEL?

Control Bay Chiller "A" will ____________________.

A. continue to run, but chilled water inlet temperature will lower to 40 ^{O}F.

B. continue to run, but will no longer supply performance data to the ICS computer.

C. immediately trip, causing a loss of chilled water to the Control Bay Air Handling Units.

D. immediately trip, then will automatically restart in the Analog Mode.

Proposed Answer: **C**

Explanation:

a. Incorrect. The chiller will trip, however the chilled water temperature setpoint while operating in Analog mode is 2 ^{O}F lower at 40 ^{O}F.

b. Incorrect. The chiller will trip, however the Chiller will not supply digital data to ICS while running in Analog Mode

c. Correct answer.

d. Incorrect. The chiller will trip as stated in (c) above. The chiller must be manually shutdown prior to transferring to Analog control, then must be manually restarted.

Technical Reference(s): 0-OI-31 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/15/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 0	Control Bay and Off-Gas Treatment Building Air Conditioning System	0-OI-31 Rev. 0126 Page 16 of 283

3.2 **Unit 1/2 Control Bay Chillers**

A. Compressor sump heaters are required to be energized for a minimum of 24 hours prior to starting a chiller. This prevents compressor damage caused by liquid refrigerant in the compressor at startup. When the heaters are on, the bottom end (end closest to the chiller control panel) of the compressors will be warm. This requirement may be modified by the System Engineer taking into account the length of time the compressor was shutdown and the outside air temperatures.

B. Prior to any draining being performed on the Unit 1/2 Control Bay Chilled Water System, Chemistry and Environmental should be notified of the planned system draining.

C. When transferring between analog and digital control modes on a Chiller, the chiller must NOT be running. Therefore, whenever CONTROL PANEL MODE SELECT, 0-HS-031-2110AA(2210BA), switches are to be manipulated, the associated Chiller is required to be shutdown.

D. The Chiller Control Switch located on the local display panel for each Control Bay Chiller Control Panel has three positions; STOP/RESET, AUTO LOCAL, and AUTO REMOTE. The STOP/RESET position shuts down the chiller when in local control. The AUTO LOCAL position is used to start the chiller when in local control. The AUTO REMOTE position is NOT used. The control switch should NOT be selected to the AUTO REMOTE position.

E. The Control Bay Chilled Water Pumps A or B are tripped by load shed signal. Pumps may be restarted after 10 minutes with the use of "CHW PUMP LOAD SHED BYPASS SW for each pump. (A PUMP "0-HS-031-2101E" Location 0-LPNL-925-0165 PANEL D)(B PUMP "0-HS-031-2201E" Location 0-LPNL-925-0165 PANEL E) If 1 & 2 CONT. BAY CHW PUMP A(B) TRANSFER SWITCH, 0-XS-031-2101(2201) is in LOCAL POSITION load shed logic for these pumps is bypassed.

BFN Unit 0	Control Bay and Off-Gas Treatment Building Air Conditioning System	0-OI-31 Rev. 0126 Page 55 of 283

5.16 Startup of Unit 1/2 Control Bay Chiller A In Local Digital Control Mode (continued)

[6] **VERIFY** the following switch positions: (Unit 1/2 Control Bay Chiller A Control Panel):

- ANALOG CONTROL MODE SELECT, 0-HS-031-2110AB(HS2), in ANALOG STOP. ☐
- CONTROL PANEL MODE SELECT, 0-HS-031-2110AA(HS1), in DIGITAL CONTROL. ☐
- UNIT 1&2 CB CHILLER A REMOTE HS APPENDIX R DISC SW, 0-HS-031-2110 in LOCAL. ☐
- CONTROL BAY CHILLER A, 0-BKR-031-2110, in ON. ☐
- CONTROL BAY CHILLER A LOCAL DISPLAY PANEL 0-PMC-031-2100A Display Window is illuminated. ☐

NOTE

Step 5.16[7] starts the chiller. There is approximately a 2 minute time delay between when the switch is placed in AUTO/LOCAL and when the chiller actually starts.

[7] At CONTROL BAY CHILLER A LOCAL DISPLAY PANEL, 0-PMC-031-2100A, **PLACE** switch in AUTO LOCAL. ☐

[8] **CHECK** the following parameters for Chiller A:

- 1&2 CB CHW PUMP A INLET PRESSURE, 0-PI-031-0194, is approximately 15-30 psig. ☐
- 1&2 CB CHW PUMP A(B) DISCHARGE PRESSURE, 0-PI-031-0195, is approximately 85-105 psig. ☐
- On 0-PMC-031-2100A (Menu-PO point F), Evap Leaving Water Temp eventually lowers to 42 ± 2°F. ☐

BFN Unit 0	Control Bay and Off-Gas Treatment Building Air Conditioning System	0-OI-31 Rev. 0126 Page 63 of 283

5.18 Startup of Unit 1/2 Control Bay Chiller A In Analog Control Mode (continued)

NOTE
Step 5.18[8] starts the chiller at (Unit 1/2 Control Bay Chiller A Control Panel). There is approximately a 5 minute time delay between when the switch is placed in AUTO and when the chiller actually starts.

[8] **PLACE** ANALOG CONTROL MODE SELECT, 0-HS-031-2110AB(HS2), in ANALOG START. ☐

[9] **RECORD** switch manipulations in Steps 5.18[7] and 5.18[8] in Narrative Log. ☐

[10] **CHECK** the following parameters on Chiller A:

- 1&2 CB CHW PUMP A INLET PRESSURE, 0-PI-031-0194, is approximately 15-30 psig. ☐
- 1&2 CB CHW PUMP A(B) DISCHARGE PRESSURE, 0-PI-031-0195, is approximately 85-105 psig. ☐
- **VERIFY** Evaporator Leaving Water Temp eventually lowers to 40 ± 5°F on 1&2 CB CHILLER A CHW OUTLET TEMP IND, 0-TI-031-0023. ☐

APPENDIX D: Test Data Changed (TRANSACT Table)

CR NRC00028451
Report Name:
Quarterly Load - Office = "V"

Records for		**Changed to**:
Docket No	= 05000259	PA_NO = 122C91A
Week_Ending	= 10/4/08 & 10/11/08	
Initials	= KHD	
Docket No	= 05000260	PA_NO = 122C91A
Week_Ending	= 10/4/08 & 10/11/08	
Initials	= KHD	
Docket No	= 07007001	PA_NO = 333240A
Week_Ending	= 10/4/08 & 10/18/08	
Initials	= MY9	
Docket No	= 05000259	PIC_CD = 19
Week_Ending	= 10/4/08	
Initials	= IFH	

LFARB #1 - Office = "V"

Records for		**Changed to**:
Docket No	= 07000143	PA_NO = 333C91A
Week_Ending	= 10/4/08	
Initials	= GAQ	
Docket No	= 07000027	PA_NO = 333C91A
Week_Ending	= 10/4/08	
Initials	= SGQ	
Docket No	= 05000250	PA_NO = 122C91A
Week_Ending	= 10/4/08	
Initials	= JS8	
Docket No	= 05000251	PA_NO = 122C91A
Week_Ending	= 10/18/08	
Initials	= MKZ	
Docket No	= 05000259	PA_NO = 122C91A
Week_Ending	= 10/4/08 & 10/11/08	
Initials	= CRY & TLR	

LFARB #2 Report - Office = "V"

Records for		**Changed to:**
Docket_No	= 05000391	PA_NO = 122C92B
Week_ending	= 10/4/08 & 10/11/08	
Initials	= THN	
Docket No	= 05000325	PIC_CD = 19
Week_Ending	= 10/4/08 & 10/18/08	
Initials	= RNA	

LFARB #3 Report - Office = "M"

Records for		**Changed to:**
Docket No	= 00000700	PIC_CD = 19
TAC No	= MD2953	
Week_Ending	= 10/4/08 & 10/18/08	
Initials	= G9B & RML	

LFARB #4 Report - Office = "D"

Records for		**Changed to:**
Docket_No	= 05000219	PA_NO = 122122A
Week_Ending	= 10/11/08 & 10/18/08	
Initials	= DYA	
Docket_No	= 05000244	PA_NO = 122122A
Week_Ending	= 10/4/08, 10/11/08 & 10/18/08	
Initials	= DLP	
Docket_No	= 05000313	PA_NO = 171105AC
TAC No	= MD7067 & MD7178	
Week_Ending	= 10/4/08 & 10/11/08	
Initials	= ADW	
Docket No	= 05000368	PIC_CD = 19
TAC No	= MD5250 & MD7068	
Week_Ending	= 10/4/08 & 10/18/08	
Initials	= ADW	

Casework Fee Memo Report - Office = "D"

Records for		**Changed to:**
Docket No	= 07007011	PIC_CD = 19
Week_Ending	= 10/11/08	
Initials	= KWJ	

Examination Outline Cross-reference:

290001A1.01

Ability to predict and/or monitor changes in parameters associated with operating the Secondary Containment controls including: System Lineups.

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	290001A1.01	
Importance Rating	3.1	3.1

Proposed Question: **RO # 37**

Unit 3 Reactor Building Ventilation is running with the Standby Gas Treatment (SGT) Systems in a normal standby lineup.

An event occurs which results in the following Unit 3 plant conditions:

- Reactor Water Level — increasing from a low of +8 inches.
- Drywell Pressure — 1.5 psig and steady.
- Reactor Building Exhaust duct radiation — 62 mR/hr and increasing slowly.
- Refuel Floor Exhaust duct radiation — 65 mR/hr and steady.

Which ONE of the following describes the Secondary Containment ventilation lineup for these conditions?

The SGT systems are ____(1)____. Reactor Building ventilation is ____(2)____ and Refuel Floor Ventilation is ____(3)____.

	(1)	(2)	(3)
A.	running with suction from the Reactor Building.	secured with dampers isolated	running with dampers open.
B.	running with suction from the Refuel Floor.	running with dampers open	secured with dampers isolated.
C.	in a normal standby lineup.	running with dampers open	running with dampers open.
D.	running with suction from the HPCI system.	running with dampers open	running with dampers open.

Proposed Answer: **C**

Explanation:

a. Part (1) and (2) are incorrect. No isolation setpoint has been exceeded. If any of the given conditions were above the set point, Part (1) and (2) would be correct and Part (3) would be incorrect. With the given conditions, only Part (3) is correct. Refuel Floor ventilation is operating properly.

b. Part (1) and (3) are incorrect. No isolation setpoint has been exceeded. If any of the given conditions were above the set point, Part (1) and (3) would be correct and Part (2) would be incorrect. With the given conditions, only Part (2) is correct. Reactor Building ventilation is operating properly.

c. Correct answer.

d. Part (1) is incorrect. If Drywell pressure was above 2.45 psig, this would be a correct answer, but Part (1) and (2) would become incorrect. With the given conditions, Part (1) and (2) are incorrect.

Technical Reference(s):	OPL171.067		(Attach if not previously provided)
Proposed references to be provided to applicants during examination:			None
Question Source:	Bank #	290001K1.08	minor format changes
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

Original question 290001K1.08:

Unit 3 Reactor Building Ventilation is running with the Standby Gas Treatment (SGT) Systems in a normal standby lineup. An event occurs with the following conditions present on Unit 3:

- Reactor Water Level — increasing from a low of +8 inches
- Drywell Pressure — 1.5 psig and steady
- Exhaust duct radiation — 62 mR/hr and increasing slowly
- Refuel Floor radiation — 55 mR/hr and steady

Which ONE of the following describes the Secondary Containment ventilation lineup for these conditions?

The SGT systems are....

A. running with their exhaust to the Main Stack; Reactor Building supply and exhaust dampers closed.

B. running with their exhaust to the Reactor Building Vent Stack; Reactor Building supply and exhaust dampers closed.

C. in their standby lineup; Reactor Building ventilation remains in the normal lineup exhausting to the Reactor Building Vent Stack.

D. in their standby lineup; Reactor Building supply and exhaust dampers are isolated from the Reactor Building Vent Stack.

Excerpt from OPL171.067 page 16:

1. System Isolation (Group 6)

   a. The isolation signals are low reactor water level +2", high drywell pressure 2.45 psig, and high radiation in exhaust duct (72 MR/hr Refuel zone or Reactor zone).

   b. On auto isolation signal (except Refuel Zone high radiation) the unit reactor zone supply and exhaust fans trip, all refuel zone supply and exhaust fans trip, unit reactor zone and all refuel zone supply and exhaust isolation dampers close, and dampers to Standby Gas Treatment System open and SGT train blowers auto start.

   c. Note: Damper logic is as follows: PCIS Group 6 with A SGT running opens 64-41 and 45; PCIS Group 6 with B SGT running opens 64-40 and 44.

   d. On isolation due to high radiation in Refuel Zone all refuel zones isolate, refuel zone supply and exhaust fans trip, SGT starts and aligns dampers for refuel zone only.

Examination Outline Cross-reference:	Level	RO	SRO
290003K1.04	Tier #	2	
Knowledge of the physical connections and/or cause-effect relationships between Control Room HVAC system and the following: Nuclear Steam Supply Shut off System (NSSSS/PCIS).	Group #	2	
	K/A #	290003K1.04	
	Importance Rating	3.2	3.3

Proposed Question: **RO # 38**

Given the following Control Room Emergency Ventilation (CREV) system conditions:

- CREV Train "A" was started to prove operability following maintenance on the charcoal trays using the STOP-AUTO-START switch on Panel 9-22.
- The SYSTEM PRIORITY SELECTOR SWITCH is selected for "TRAIN-B".

Which ONE of the following describes the CREV system response should a valid CREV initiation signal be received?

On a valid initiation, CREV Train "B" would _____(1)_____ and CREV Train "A" would _____(2)_____.

	(1)	(2)
A.	initiate	shutdown.
B.	initiate	NOT shutdown.
C.	NOT initiate	shutdown.
D.	NOT initiate	NOT shutdown.

Proposed Answer: **B**

Explanation:

a. Part (1) is correct. CREV Train B will initiate without a time delay since the CREV UNIT PRIMARY SELECTOR SWITCH is selected for "TRAIN-B". Part (2) is incorrect. CREV will not automatically shutdown with a valid initiation signal present.

b. Correct answer.

c. Part (1) is incorrect. Normally, when an auto initiation signal is received, the TRAIN selected for "secondary" begins its start sequence but will not finish if the Primary CREV train is running. This is sensed by looking at the ΔP across the HEPA filter. Since Train B was selected as the Primary CREV unit, the start sequence does not look at the ΔP. Part (2) is incorrect. CREV will not automatically shutdown with a valid initiation signal present.

d. Part (1) is incorrect as stated in (c) above. Part (2) is correct. This would be incorrect if CREV Train A was started using the AUTO-INITIATE TEST switch, as would be the case during the periodic surveillance test.

Technical Reference(s): 0-OI-31 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # OPL171.067.49 minor format modifications

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Original question OPL171.067.49:

CREV Train A is running for testing to prove operability following maintenance on the charcoal trays.
The CREV UNIT PRIMARY SELECTOR SWITCH is selected for "B".

DETERMINE which of the following is the proper sequence of events and/or operator actions should a valid CREV initiation signal be received.

A. CREV Train B will initiate on a valid start signal, Train A will shutdown, no operator action required.

B. CREV Train B will initiate on a valid start signal and Train A must be manually shutdown.

C. Neither Train would initiate on a valid start signal since Train A is being tested and would shutdown automatically. Train B must be manually started.

D. CREV Train B will NOT initiate on a valid start signal. Train B must be manually started and Train A must be manually shutdown.

3.6 CREV and CREV instrumentation operability issues (continued)

B. The main control room boundary may be opened intermittently under administration controls. For openings other than normal entry and exit, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the main control room and whose task is to close the opening when main control room isolation is indicated. With both CREVs inoperable in Modes 1, 2 or 3, for other than a control room boundary issue, enter LCO 3.0.3 Immediately. With two CREV subsystems inoperable during OPDRVs, initiate action to suspend OPDRVs. Reference TS 3.7.3.

C. When there is an automatic actuation of CREVS, the following automatic isolation dampers and hatch is required to be closed for CREVS to be considered operable.

1. 0-FCO-31-150B, 0-FCO-31-150D, 0-FCO-31-150E, 0-FCO-31-150F, 0-FCO-31-150G.
2. Removable equipment hatch in U-3 Mechanical Equipment Room, floor Elevation 617'.

D. The CREV system utilizes 15.45 kW Duct heaters to control moisture buildup in the charcoal adsorber. A malfunction of the 15.45 kW duct heater makes the applicable CREV unit inoperable. [Reference Functional Evaluation in PER 74959 and 75680]

E. One of the UNIT 1 & 2 Control Bay Supply Fans and one of the UNIT 3 Control Bay Supply Fans and their associated power and control circuits is required to be operable for CREVS instrumentation (control bay high radiation)to be considered operable. Reference Tech Spec 3.3.7.1.

F. CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214 may be placed in either the "A" or "B" position, depending on the operability status of the CREV trains. When a CREV train is inoperable, it will NOT be selected as lead. When both CREV trains are operable, the preferred position for CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214, is in TRAIN "A" which makes the "A" CREV the lead train. In the event that "A" CREV is INOP, CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214 is required to be placed in the TRAIN "B" position so the "B" CREV will initiate, without a time delay, as the lead train.

G. When one of the CREV trains is inoperable for testing, the CREV UNIT PRIMARY SELECTOR SWITCH, 0-XSW-031-7214 is required to be aligned to the train which is NOT under testing conditions to ensure the non-test train will initiate under an actual initiation signal.

Examination Outline Cross-reference:	Level	RO	SRO
295001AK1.03	Tier #	1	
Knowledge of the operational implications of the following concepts as they apply to the Partial or Complete Loss of Forced Core Flow Circulation: Thermal Limits.	Group #	1	
	K/A #	295001AK1.03	
	Importance Rating	3.6	4.1

Proposed Question: **RO # 39**

Which ONE of the following describes the response of thermal limits due to the trip of a single recirculation pump from 100% rated power?

A single recirculation pump trip from 100% rated power will cause the value of Critical Power to ____(1)____ and the Critical Power Ratio will ____(2)____.

	(1)	(2)
A.	lower	lower
B.	lower	rise
C.	rise	lower
D.	rise	rise

Proposed Answer: **B**

Explanation:

a. Part (1) is correct. As the actual power goes down, the power required to cause the onset of transition boiling also goes down. Part (2) is incorrect. Although actual power goes down and critical power goes down, the power required to cause the onset of transition boiling does NOT go down as far as actual power due to the higher void fraction. Therefore, the Critical Power Ratio rises.

b. Correct answer

c. Part (1) is incorrect. As actual power goes down, void fraction increases. This causes critical power to lower, although not as significantly as actual power. Part (2) is incorrect as stated in (a) above.

d. Part (1) is incorrect as stated in (c) above. Part (2) is correct. Since actual power drops farther than critical power, the Critical Power Ratio gets larger.

Technical Reference(s):	GFES Thermal Limits	(Attach if not previously provided)
	OPL171.007, Recirculation System LP	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	9/2/2008 RMS	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Excerpt from OPL171.007 Page 28 of 86:

b. The protective action of a scram is the normal means of terminating this transient. At end of cycle conditions, control rods may all be fully withdrawn, while thermal neutron flux has shifted upwards in the core. This will delay the effect of negative reactivity from a control rod scram. In order to provide a means for adding additional negative reactivity, an EOC-RPT system has been designed to trip recirc pumps to allow additional void formation.

When voids collapse, both actual power and critical power go up, however, actual power goes up more than critical power. This results in being closer to CPR limits.

Excerpt from GFES Lesson Plan on Thermal Limits (General Physics Corp © 2000):

Table 9-1 Factors Affecting Critical Power

FACTOR	CRITICAL POWER	BUNDLE POWER	CPR
INLET SUBCOOLING:			
INCREASES	↑	↑	↓
DECREASES	↓	↓	↑
MASS FLOW RATE:			
INCREASES	↑	↑	↓
DECREASES	↓	↓	↑
PRESSURE:			
INCREASES	↓	↑	↓
DECREASES	↑	↓	↑
LOCAL PEAKING FACTOR			
INCREASES	↓	↔	↓
DECREASES	↑	↔	↑
AXIAL POWER DISTRIBUTION			
INCREASES	↓	↔	↓
DECREASES	↑	↔	↑

STEADY STATE AND TRANSIENT

The primary design objective is to maintain nucleate boiling and avoid OTB. The CPR thermal limit is set to maintain adequate margin between nucleate boiling and OTB. The steady state and transient MCPR thermal limits are derived from this single design basis requirement.
Transients caused by single operator error or equipment malfunction shall be limited so that, considering uncertainties in monitoring the core operating state, more than 99.9% of the fuel rods are expected to avoid OTB.
The transients most likely to limit operation because of MCPR considerations are:

- Turbine trips or generator load rejections without bypass valve capability
- Loss of feedwater heating or inadvertent high pressure coolant injection
- Feedwater controller failure (maximum demand)

MAXIMUM FRACTION OF LIMITING CRITICAL POWER RATIO (MFLCPR)

The process computer calculates CPR data evaluating core conditions to ensure limits are not exceeded. One of the most useful forms of this data output is a ratio called the "fraction of limiting critical power ratio" (FLCPR). This ratio compares the flow-adjusted operating (steady-state) maximum CPR for the fuel bundle to the actual bundle CPR. From this, the maximum fraction of limiting critical power ratio (MFLCPR - pronounced "miffle-sipper"), which is the maximum fraction of limiting critical power ratio (MFLCPR) and is the ratio of the flow-adjusted CPR operating limit for that fuel type to the bundle CPR, is developed. For most nuclear plants the MFLCPR ratio takes the following form:

$$\text{MFLCPR} = \frac{\text{CPR}_{\text{Limit}} \times K_f}{\text{CPR}}$$

Equation 9-16

Examination Outline Cross-reference:	Level	RO	SRO
295003AK1.02	Tier #	1	
Knowledge of the operational implications of the following concepts as they apply to the Partial or Complete Loss of AC: Load Shedding.	Group #	1	
	K/A #	295003AK1.02	
	Importance Rating	3.1	3.4

Proposed Question: **RO # 40**

Given the following Unit-2 conditions:

- A Loss of Off-site power and LOCA has occurred.
- 480V Load Shedding Logic has actuated.
- The Unit Operator immediately clears the "A" RBCCW pump white disagreement light while surveying panel 9-4.
- No other actions were performed.

Which ONE of the following describes the effect on the RBCCW system?

A. "A" pump auto starts after 40 sec; "B" pump can be manually started immediately.

B. "B" pump auto starts after 43 sec; "A" pump can be manually started after 40 sec.

C. "A" pump auto starts after 40 sec; "B" pump auto starts after 43 secs.

D. "B" pump auto starts after 43 sec; "A" pump can be manually started immediately.

Proposed Answer: **B**

Explanation:

a. Part (1) is incorrect. This is true only if the breaker control switch was left in the "normal-after-start" position. Part (2) is incorrect. All RBCCW pumps are prevented from starting for 40 seconds following a load shed regardless of switch positions.

b. Correct answer.

c. Part (1) is incorrect as stated in (a) above. Part (2) is correct.

d. Part (1) is correct. Part (2) is incorrect. All RBCCW pumps are prevented from starting for 40 seconds following a load shed regardless of switch positions.

Technical Reference(s): 2-OI-70, Rev 59 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # OPL171.072.03 attached

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Original question: OPL171.072.03

480V Load Shedding Logic has actuated on Unit 2 when the operator clears the "A" RBCCW pump white disagreement light while surveying pnl. 9-4 with no other actions.

Which one of the following statements describes the effect on the RBCCW system?

A. "A" pump auto starts after 40 sec; "B" pump auto starts 3 sec. later.

B. "B" pump auto starts after 43 sec; "A" pump can be manually started after 40 sec.

C. "B" pump auto starts after 40 sec; "A" pump can be manually started immediately.

D. "A" pump auto starts after 40 secs; "B" may then be manually started.

BFN Unit 2	Reactor Building Closed Cooling Water System	2-OI-70 Rev. 0059 Page 10 of 62

3.0 PRECAUTIONS AND LIMITATIONS (continued)

P. With an accident signal present (low Reactor water level or high Drywell pressure) on Unit 1 or Unit 2 and any Diesel Generator output breaker closing on U-1 and 2 Shutdown boards, the following occurs:

1. Unit 1 and Unit 2 RBCCW pumps trip.

2. Unit 1 and Unit 2 Drywell blowers trip.

3. 1-FCV-70-48 and 2-FCV-70-48 close when power is restored (close signal present for 40 seconds). This auto closure is bypassed if 2(1)-XS-70-48 at 480 RMOV board 2(1)B is in the EMERGENCY position.

4. After a 40 second time delay, the following occurs:

   a. With the control switch in Normal After Start, RBCCW Pump A restarts for Unit 1 and Unit 2.

   b. If RBCCW pump A fails to start, RBCCW Pump B will automatically start after a 3 second time delay for Unit 1 and Unit 2 (with the control switch in normal after start).

   c. The Drywell Blowers on the unit without the accident will automatically restart (Unit 2 blowers will have staggered auto start times). Unit 2 Drywell blowers with their respective Auto Start Inhibit switch in the INHIBIT position will not auto start, but can, however, be manually started after a ten minute time delay.

   d. The Drywell Blowers A1, B1, A2, and B2 on the unit with the accident may be manually restarted after 40 seconds.

   e. The Drywell Blowers A3, B3, A4, B4, A5 and B5 on the unit with the accident will remain tripped.

Q. Rotork valve operator indicators have three indications. "Full Open," "Full Closed," and "Mid Position". The "Mid Position" merely indicates that the valve is neither "Full Open" nor "Full Closed," It does not represent a percentage Open or Closed.

R. Temperature Control Valves are required to be isolated very SLOWLY (controlled manner)to ensure no erratic system perturbations result in ESF initiations. [PER 01-005343-000]

Examination Outline Cross-reference:	Level	RO	SRO
295004AA2.03	Tier #	1	
Ability to determine and interpret the following as they apply to a Partial or Total Loss of DC Power: Battery Voltage.	Group #	1	
	K/A #	295004AA2.03	
	Importance Rating	2.8	2.9

Proposed Question: **RO # 41**

The following plant conditions exist:

- Complete loss of offsite power on Unit 2.
- All 4KV Shutdown boards are being supplied by their Diesel Generators.
- DW pressure: 2 psig slowly rising.
- RPV level: (-) 140 inches and stable.
- RPV pressure: 800 psig and stable.
- HPCI and RCIC are injecting to the vessel.
- 250V Reactor MOV BD 2A UV (9-8C W4) is in alarm.

What ONE of the following describes the actions required to restore 2A 250v Reactor MOV Board voltage to normal?

On 250V Battery Charger ___(1)___, perform the following: ______(2)______.

	(1)	(2)
A.	2A	Place the Emergency ON select switch in "Emergency ON."
B.	2A	Manually re-close the normal feeder breaker following a 40 second time delay.
C.	1	Place the Emergency ON select switch in "Emergency ON."
D.	1	Manually re-close the normal feeder breaker following a 40 second time delay.

Proposed Answer: **A**

Explanation:

a. correct answer.

b. Part (1) is correct. Part (2) is incorrect. This action would be correct for Battery Charger 4, but is incorrect for Battery Charger 2A.

c. Part (1) is incorrect. This would be the correct battery charger for 250V Reactor MOV Board 1A. (wrong unit/train issue) Part (2) is correct and would allow Battery Charger 1 to restart as well.

d. Part (1) is incorrect as stated in (c) above. Part (2) is incorrect. This action would be correct for Battery Charger 4, but is incorrect for Battery Charger 2A.

Technical Reference(s):	2-ARP-9-8C, 0-OI-57D	(Attach if not previously provided)
	OPL171.037, DC Distribution	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #	RO 295004AK3.01	attached
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Original question RO 295004AK3.01:

The following plant conditions exist:

- Complete loss of offsite power on Unit 2
- All 4kv S/D boards are being supplied by their Diesel Generators
- DW pressure: 2 psig slowly rising
- RPV level: -140 inches and stable
- RPV pressure: 800 psig and stable
- HPCI and RCIC are injecting to the vessel

What actions are required to restore 2A 250v Battery Charger?

A. The battery charger can only be energized when the accident signal clears.

B The battery charger can be re-energized by placing the emergency bypass switch to bypass.

C. The battery charger is energized, it automatically restarts forty seconds following an accident signal and requires no manual actions to restore it.

D. The battery charger is energized, it requires no manual actions to restore it.

BFN Unit 0	DC Electrical System	0-OI-57D Rev. 0117 Page 16 of 247

3.0 PRECAUTIONS AND LIMITATIONS (continued)

E. All safety requirements concerning smoking, fires or sparks should be observed when in the Battery-Battery Board Rooms because of potential accumulation of hydrogen in flammable amounts.

F. 250V Unit Battery Charger 1,2A,2B and 3 Emergency ON select switch bypasses battery charger emergency load shed contacts. Placing the select switch in Emergency ON reestablishes charger operations with an accident signal present and Diesel Generator voltage available. Battery Charger 4 supply breaker, 480V Shutdown Board 3B, Compt 6D, receives a trip signal from the load shed logic and the breaker must be manually re-closed after a 40 second time delay to restore the charger to service. The annunciation circuit for the 250V Unit Battery Charger 3 does NOT work when the EMER/OFF/ON Select Switch is in the EMER Position.

G. [II/C] Neutron monitoring battery chargers are NOT stand alone power supplies and shall only be operated while connected to the neutron monitoring batteries. [BFPER 940862]

H. Within 30 minutes after the loss of the normal charger to a 250V Unit Battery another charger shall be placed in service to that battery and load reduced so that the battery is NOT discharging.

I. [NRC/C] Upon return to service of 24V DC Neutron Monitoring Battery A or B, Instrument Maintenance must perform functional tests on SRMs and IRMs that are powered from the affected battery board (In that the IRMs and SRMs are normally inoperable after entering RUN mode due to lack of testing, these tests are N/A for the IRMs and the SRMs if the Unit is in RUN Mode and the IRMs and SRMs are inoperable). Prior to calling the IRMs and SRMs operable, the tests have to be performed. [NRC IE Inspect Follow-up Item 86-40]

J. To return equipment to service following a failure or trip, the shutdown section of this instruction should be performed on the equipment failed. The initial conditions may NOT be applicable in this case.

K. [NRC/C] The transfer of 250VDC control power to a 4kV Shutdown Board with a diesel generator operating may cause an inadvertent start of a RHRSW pump. [LER 88021/25]

Excerpt from OPL171.037, DC Distribution, page 14 of 70:

They also supply alternate control power for Units 1 and 2 4kV Shutdown Boards; however, on Unit 3, the A, C, and D 4kV Shutdown Boards receive both normal and alternate control power from the 250V DC Unit Systems. (3EB receives alternate control power only.) The 250V DC RMOV Boards are supplied from the Unit Battery Board as follows: BB-1 supplies 250V RMOV Boards 1A, 2C, 3B. BB-2 supplies 250V RMOV Bds 2A, 1C, 3C. BB-3 supplies 250V RMOV Boards 3A, 1B, 2B.

Excerpt from OPL171.037, DC Distribution, page 31 of 70:

250V Battery Charger	Normal Source	Alternate Source (Charger Service bus)
1	480V SD Bd 1A, Comp 6D	480V Common Bd 1, Comp 3A
2A	480V SD Bd 2A Comp 6D	480V Common Bd 1, Comp 3A
2B	480V SD Bd 2B, Comp 6D	480V Common Bd 1, Comp 3A
3	480V SD Bd 3A, Comp 6D	480V Common Bd 1, Comp 3A
4	480V SD Bd 3B, Comp 6D	480V Common Bd 1, Comp 3A
5	480V Com Bd 1 Comp 5C	(no alternate)
6	480V Com Bd 3 Comp 3D	(no alternate)

The 2B spare charger DC output can be directed to any of four feeders. Three DC outputs can be connected to battery board 1, 2, or 3. The fourth DC output is connected to output transfer switch (BBR 4) to batteries 4, 5, or 6. Mechanical interlock permits closing only one output feeder at a time. (A slide bar is utilized in battery board room 2 and a Kirk key interlock is used in battery board room 4.)

250V DC battery chargers 1, 2A and 2B will load shed upon receipt of a Unit 1 or Unit 2 accident signal and any Unit 1/2 shutdown board being supplied by its respective diesel generator or cross tied to a Unit 3 shutdown board and a unit three Diesel Generator. 250 VDC Battery Charger 3 will load shed on a unit 3 load shed signal. The load shedding feature can be bypassed by placing the "Emergency" switch on the charger to the "EMERG" position.

Station Battery charger 4 does not have load shed logic; however, battery charger 4 will deenergize when 3B 480 S/D Board deenergizes and will return when the 480V S/D Board voltage returns.

They also supply alternate control power for Units 1 and 2 4kV Shutdown Boards; however, on Unit 3, the A, C, and D 4kV Shutdown Boards receive both normal and alternate control power from the 250V DC Unit Systems. (3EB receives alternate control power only.) The 250V DC RMOV Boards are supplied from the Unit Battery Board as follows:
BB-1 supplies 250V RMOV Boards 1A, 2C, 3B.
BB-2 supplies 250V RMOV Bds 2A, 1C, 3C.

Examination Outline Cross-reference:	Level	RO	SRO
295005AA2.04	Tier #	1	
Ability to determine and interpret the following as they apply to a Main Turbine Generator Trip: Reactor Pressure	Group #	1	
	K/A #	295005AA2.04	
	Importance Rating	3.7	3.8

Proposed Question: **RO # 42**

Given the following plant conditions:

- Reactor power is 38% power.
- Main turbine load is 23%.
- Turbine bypass valves are partially open.

Which ONE of the following describes the response of the reactor if the Main Turbine Generator inadvertently trips?

The reactor will ________________________.

A. scram on High Reactor Pressure.

B. immediately scram on Turbine Stop Valve closure.

C. continue to operate at 38% power with Bypass Valves open.

D. continue to operate above 38% power due to a loss of Feedwater heating.

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Incorrect due to low turbine first stage pressure. If turbine load was slightly higher, the reactor would scram on TSV closure.

c. Incorrect due to power slightly greater than Bypass Valve capacity. If power were lower, than the reactor would continue to operate.

d. Incorrect due to power slightly above Bypass Valve capacity. If power were slightly lower and the reactor did not scram, the reduction in Feedwater temperature would cause a small power increase.

Technical Reference(s): 1-OI-99, Reactor Protection System (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # 295005AA2.05 Attached

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Bank Question 295005AA2.05

Given the following plant conditions:

- Reactor power is 38% power.
- Main turbine load is 23%.
- Turbine bypass valves are partially open.

Which one of the following describes the response of the reactor if the Generator Breaker inadvertently OPENS?

A. Reactor immediately scrams on turbine stop valve 10% closure.

B. Reactor scrams on high reactor pressure.

C. Reactor continues to operate at 38% power.

D. Reactor continues to operate and power decreases to 30%.

BFN Unit 1	Reactor Protection System	1-OI-99 Rev. 0033 Page 52 of 68

Illustration 2
(Page 2 of 2)

Unit 1 Reactor Scram Initiation Signal

	Scram	Setpoint	Bypass
J.	OPRM TRIP	Any one of the three algorithms, period, growth, or amplitude for an operable OPRM cell has exceeded its trip value conditions:	Reactor is NOT operating in the AUTO ENABLE Region of the Power/Flow Map.
K.	Low RPV Water Level (Level 3)	+2.0"	N/A
L.	Hi RPV Pressure	1088 psig	N/A
M.	Hi DW Pressure	2.45 psig	N/A
N.	MSIV closure	90% open (3 Main Steam Lines)	NOT in RUN
O.	Scram Discharge Instrument Volume Hi Hi	• Thermal level switches 49 gallons (LS-85-45A,B,G,H) • Float level switches 45 gallons (LS-85-45C,D,E,F)	Mode Switch in SHUTDOWN or REFUEL with keylock switch in BYPASS
P.	TSV Closure	90% open (3 TSVs)	< 30% Rx Power (≤ 148.5 psig 1st stage pressure)
Q.	TCV Fast Closure (load reject)	40% mismatch (amps to cross-under pressure); 850 psig EHC RETS at TCV (1 or 3) & (2 or 4)	< 30% Rx Power (≤ 148.5 psig 1st stage pressure)
R.	Loss of RPS Power	N/A	N/A
S.	Scram Channel Test Switches	Key-locked in AUTO Panels 1-9-15 & 1-9-17	N/A

Examination Outline Cross-reference:	Level	RO	SRO
295006AK1.03	Tier #	1	
Knowledge of the operational implications of the following concepts as they apply to a Scram: Reactivity Control	Group #	1	
	K/A #	295006AK1.03	
	Importance Rating	3.7	4.0

Proposed Question: **RO # 43**

Unit 2 has received a scram signal but some of the control rods failed to fully insert. The Unit Supervisor has directed you to insert control rods as directed by 2-OI-85, "Control Rod Drive System."

Which ONE of the following control rod insertion processes can ONLY be accomplished in the main control room as directed by 2-OI-85, "Control Rod Drive System?"

A. Removal and replacement of RPS scram solenoid fuses.

B. Venting and re-pressurizing the Scram Pilot Air Header.

C. Insertion of control rods by venting the over piston area.

D. Control rod insertion using raised cooling water differential pressure.

Proposed Answer: **D**

Explanation:

a. Incorrect answer. This action must be performed from the Aux Instrument Room.

b. Incorrect answer. This action must be performed in the Reactor Building.

c. Incorrect answer. This action must also be performed in the Reactor Building.

d. Correct answer.

Technical Reference(s):	2-OI-85 Section 8.19		(Attach if not previously provided)
Proposed references to be provided to applicants during examination:			None
Question Source:	Bank #	295006AK1.03	Attached
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0104 Page 134 of 181

8.19 Control Rods Which Fail to FULLY INSERT After Scram

NOTE

The operator should determine the most effective method to insert rods from the following sections:

- Removal and Replacement of RPS Scram Solenoid Fuses (Section 8.19[1]).
- Venting and Repressurizing the Scram Pilot Air Header (Section 8.19[2]).
- Individually Scram Control Rods (Section 8.19[3]).
- Insert Control Rods using Reactor Manual Control System (Section 8.19[4]).
- Manual Insertion of Control Rods by Venting the Over Piston Area (Section 8.19[5]).
- Control Rod Insertion using Raised Cooling Water Differential Pressure (Section 8.19[6]).

[1] **IF** Removal and Replacement of RPS Scram Solenoid Fuses is desired, **THEN**

PERFORM the following:

[1.1] **OBTAIN** fuse pullers and **PROCEED TO** Unit 2 Auxiliary Instrument Room. ☐

[1.2] **LOCATE** terminal strip CC inside Panel 9-15, Bay 2, RPS CHANNEL A Panel (Rear). ☐

[1.3] **REMOVE** the following RPS Bus "A" fuses (located at bottom of terminal strip CC, Panel 9-15) **AND DOCUMENT** removal on Illustration 6. ☐

FUSE LOCATION	FUSE	FUSE ID
CC-4FU	5A-F18A	2-FU1-085-0037AA
CC-5FU	5A-F18E	2-FU1-085-0039A/2
CC-6FU	5A-F18C	2-FU1-085-0039A/3
CC-7FU	5A-F18G	2-FU1-085-0039A/4

[1.4] **LOCATE** terminal strip CC inside Panel 9-17, Bay 2, RPS CHANNEL B Panel (Rear). ☐

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8.19 **Control Rods Which Fail to FULLY INSERT After Scram (continued)**

[1.5] **REMOVE** the following RPS Bus "B" fuses (located at bottom of terminal strip CC, Panel 9-17) **AND DOCUMENT** removal on Illustration 6. ☐

FUSE LOCATION	FUSE	FUSE ID
CC-4FU	5A-F18B	2-FU1-085-0037BA
CC-5FU	5A-F18F	2-FU1-085-0039B/2
CC-6FU	5A-F18D	2-FU1-085-0039B/3
CC-7FU	5A-F18N	2-FU1-085-0039B/4

[1.6] **WHEN** ALL fuses are removed, **THEN**

NOTIFY Unit Operator. ☐

[1.7] **WHEN** Shift Manager/Unit Supervisor directs replacement of fuses, **THEN**

REPLACE fuses listed in Steps 8.19[1.3] and 8.19[1.5] above **AND DOCUMENT** replacement on Illustration 6. ☐

[1.8] **WHEN** ALL fuses are replaced, **THEN**

NOTIFY Unit Operator. ☐

[2] **IF** Venting and Repressurizing the Scram Pilot Air Header is desired, **THEN**

PERFORM the following:

[2.1] **CLOSE** 2-85-331, CONT AIR SPLY HDR SOV (located on RX Bldg North wall near Scram Air Header Pressure Regulators). ☐

[2.2] **OPEN** instrument drain valves for the following pressure switch and gauge (located on 2-PNL-925-0018B, East end):

- 2-PS-85-38, CRD SCRAM VALVE PILOT AIR HEADER PRESS. ☐
- 2-PI-85-38, CRD SCRAM VALVE PILOT AIR HEADER PRESS. ☐

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0104 Page 138 of 181

8.19 Control Rods Which Fail to FULLY INSERT After Scram (continued)

[4.6] **REFER TO** Illustration 4 and **DEPRESS** the appropriate CRD Rod Select pushbutton on 2-XS-85-40. ☐

[4.7] **CHECK** backlit CRD ROD SELECT pushbutton is brightly illuminated and white indicating light on Full Core Display illuminated. ☐

[4.8] **CONTINUOUSLY INSERT** control rod to Position 00, by holding CRD CONTROL SWITCH, 2-HS-85-48, in ROD IN **OR** CRD NOTCH OVERRIDE SWITCH, 2-HS-85-47, in EMERG ROD IN. ☐

[4.9] **IF** control rod is difficult to insert, **THEN**

REFER TO Section 8.16. ☐

[4.10] **REPEAT** Steps 8.19[4.6] through 8.19[4.8] for each Control Rod to be inserted. ☐

[4.11] **PLACE** Rod Worth Minimizer Normal Bypass Switch in NORMAL in accordance with Section 8.18. ☐

[4.12] [INPO/C] **PLACE** the Reactor Mode Switch in SHUTDOWN. [INPO SOER 80-006 recommendation 9] ☐

[4.13] **VERIFY OPEN** CHARGING WATER SHUTOFF, 2-SHV-085-0586 (RB, EL 565 NE Corner). ☐

[5] **IF** Manual Insertion of Control Rods by Venting the Over Piston Area is desired, **THEN**

PERFORM the following:

[5.1] **OBTAIN** the following equipment:

- Catwalk key from Unit 2 Control Room Key Cabinet. ☐
- Square valve stem operators (one L-shaped, T-shaped, 10" crescent and speed wrench). ☐
- 50 feet high temperature hose with quick disconnect fitting on one end. ☐

[5.2] **ESTABLISH** communications with Control Room. ☐

BFN Unit 2	Control Rod Drive System	2-OI-85 Rev. 0104 Page 139 of 181

8.19 Control Rods Which Fail to FULLY INSERT After Scram (continued)

[5.3] **REFER TO** Illustration 4 and **OBTAIN** recommended control rod insert sequence from Unit Operator. ☐

[5.4] **REFER TO** Illustration 5 and **PERFORM** the following to vent each CRD in insert sequence recommended by Unit Operator. ☐

NOTE
ONLY those CRD modules on the extreme North and South ends of each row contain EOI Identification Tags.

[5.4.1] **UNLOCK AND CLOSE** WITHDRAW RISER ISOL 2-ISV-085-615. ☐

[5.4.2] **DIRECT** end of vent hose without quick disconnect coupling to Radwaste floor drain **AND SECURE** hose to floor drain cover. ☐

[5.4.3] **PROCEED TO** catwalk with tools and equipment. ☐

[5.4.4] **CONNECT** quick disconnect end of vent hose to coupling WITHDRAW RISER VENT TEST CONN, 2-85-623. ☐

[5.4.5] **IF** valve stem cap is installed for WITHDRAWAL RISER VENT, 2-VTV-085-614, **THEN**

REMOVE valve stem cap from WITHDRAWAL RISER VENT, 2-VTV-085-614. ☐

CAUTION
Opening of WITHDRAW RISER VENT valve more than two turns may result in burn and contamination hazard to personnel at the valve or floor drain area.

[5.4.6] **SLOWLY OPEN** WITHDRAW RISER VENT, 2-VTV-085-614, using T- or L-shaped wrench. ☐

Examination Outline Cross-reference:	Level	RO	SRO
295016G2.4.41	Tier #	1	
Knowledge of emergency action level thresholds and classifications. Control Room Abandonment	Group #	1	
	K/A #	295016G2.4.41	
	Importance Rating	2.9	4.6

Proposed Question: **RO # 44**

Which ONE of the following describes the emergency action level required by EPIP-1, "Emergency Classification Procedure" when the control room must be abandoned and AOI-100-2, "Control Room Abandonment" is entered?

If the control room is evacuated and backup control from Panel 25-32 is NOT established within ______(1)______, EPIP-1 classifies the event as a/an _________(2)_________.

	(1)	(2)
A.	20 minutes	Alert
B.	20 minutes	Site Area Emergency
C.	15 minutes	Alert
D.	15 minutes	Site Area Emergency

Proposed Answer: **B**

Explanation:

a. Part (1) is correct. Part (2) is incorrect. An Alert is declared immediately upon entering AOI-100-2.

b. Correct Answer.

c. Part (1) is incorrect. 15 minutes is plausible because it is the time limit associated with upgrading an emergency from a NOUE to an Alert in the event of a fire. (EAL 6.4-U1) Part (2) is incorrect as stated in (a) above.

d. Part (1) is incorrect as stated in (c) above. Part (2) is correct.

Technical Reference(s):	1-AOI-100-2, EPIP-1		(Attach if not previously provided)
Proposed references to be provided to applicants during examination:			None
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	8/27/2008 RMS	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

1.2 Responsibilities (continued)

NOTE
Evacuation or anticipated evacuation is classified by EPIP-1 as an Alert. If the control room is evacuated and backup control from Panel 1-25-32 is **NOT** established within 20 minutes, EPIP-1 classifies the event as a Site Area Emergency. Details are contained in EPIP-1 and Technical Bases.

D. If Unit 3 Control Room is **NOT** affected, Unit 3 Unit Supervisor (SRO) assumes responsibility for EPIP implementation.

E. If ALL Control Rooms are affected, Shift Manager/Unit Supervisor (SRO) assumes responsibility for EPIP implementation.

F. Responsibility for completing panel checklists is assigned to individuals stationed in the area of equipment to be checked. Attachment 1 provides backup control station assignments.

2.0 SYMPTOMS

A. Dense smoke in Unit 1/2 Control Room.

B. Toxic gas released through ventilation system.

C. A fire in the Unit 1/2 Control Room **NOT** meeting Appendix R entry conditions.

3.0 AUTOMATIC ACTIONS

None

BROWNS FERRY	EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX	EPIP-1

FIRE / EXPLOSION

Description	Description	
6.4-U1 TABLE	6.4-U2	UNUSUAL EVENT
Confirmed fire in ANY plant area listed in Table 6.4-U1 AND NOT extinguished within 15 minutes. OPERATING CONDITION: ALL	Unanticipated explosion within the protected area resulting in visible damage to ANY permanent structure or equipment. OPERATING CONDITION: ALL	
6.4-A TABLE		ALERT
Fire or explosion in ANY plant area listed in Table 6.4-A affecting safety system performance OR Fire or explosion causing visible damage to permanent structure of safety systems in ANY plant area listed in Table 6.4-A. OPERATING CONDITION: ALL		
		SITE EMERGENCY
		GENERAL EMERGENCY

Examination Outline Cross-reference:	Level	RO	SRO
295018AK2.01	Tier #	1	
KNOWLEDGE OF THE INTERRELATIONS BETWEEN Partial or Total Loss of CCW and the following: System Loads.	Group #	1	
	K/A #	295018AK2.01	
	Importance Rating	3.3	3.4

Proposed Question: **RO # 45**

Which ONE of the following describes the response to a partial loss of RBCCW?

Should the system discharge header pressure drop to less than __(1)___, isolation valve FCV-70-48 would close, causing RBCCW System loads ______(2)______ to lose RBCCW cooling.

1. RBED sump HX
2. FPC HX
3. Recirc pump seal coolers
4. RR system sample coolers
5. RWCU pump seal coolers
6. DWED sump HX
7. RWCU non-regenerative HX
8. Recirc pump motor coolers
9. DW coolers

	(1)	(2)
A.	47 psig;	3,6,8,9.
B.	57 psig;	3,6,8,9.
C.	47 psig;	1,2,4,5,7.
D.	57 psig;	1,2,4,5,7.

Proposed Answer: **D**

Explanation:

a. Part (1) is incorrect. The pressure is 10 psig below the setpoint. Part (2) is incorrect. This is a list of loads that do NOT lose cooling.

b. Part (1) is correct. Part (2) is incorrect as stated in (a) above.

c. Part (1) is incorrect as stated in (c) above. Part (2) is correct. This is the list of loads that will lose cooling.

d. Correct answer.

Technical Reference(s):	2-OI-70, RBCCW System	(Attach if not previously provided)
	OPL171.047, RBCCW LP	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #	OPL171.047.17	Attached
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Original question OPL171.047.17:

Concerning the RBCCW system, should the system discharge header pressure drop to less than _______, the non-essential loop isolation valve FCV-70- _____ would close, causing which ones of the following to lose RBCCW.

1. RBED sump HX
2. FPC HX
3. DW control A/C
4. RR system sample clrs
5. RWCU pump seal clrs
6. DWED sump HX
7. RWCU non-reg HX
8. RR pump mtr & seal clrs
9. DW atmos clrs

A. 48 psig; 48; 3,6,8,9.

B. 57 psig; 47; 1,2,3,5,7.

C. 47 psig; 47; 1,2,3,4,5,7.

D. 57 psig; 48; 1,2,4,5,7.

BFN Unit 2	Reactor Building Closed Cooling Water System	2-OI-70 Rev. 0059 Page 7 of 62

3.0 PRECAUTIONS AND LIMITATIONS

A. The Spare RBCCW Pump and RBCCW Heat Exchanger are common to all three (3) units. When it becomes necessary for the Spare RBCCW Pump and RBCCW Heat Exchanger to be placed in service, operators on all three units are required to communicate and be fully aware of RBCCW conditions.

B. When removing an RBCCW heat exchanger from service, RBCCW flow is required to be stopped prior to stopping RCW or EECW cooling water flow.

C. Any water removed from RBCCW is considered potentially contaminated.

D. On low RBCCW Pump discharge header pressure (57 psig), nonessential equipment isolation valve 2-FCV-70-48 automatically closes and is required to be reopened manually using 2-HS-70-48A. This interlock is bypassed when 2-XS-70-48 (480V RMOV board 2B, compartment 5A) is placed in EMERGENCY.

E. When placing an RBCCW heat exchanger in service, prior to opening an RBCCW heat exchanger inlet valve, placing the RBCCW SECTIONALIZING VLV TRANSFER switch at the 480V reactor MOV board 2B, compartment 5A to EMERG will temporarily bypass the header low pressure auto closure for 2-FCV-70-48.

F. [NRC/C] When the RCW, EECW, or RBCCW supplied to any RBCCW heat exchanger is put into service or taken out of service, the Chemistry Laboratory Shift Supervisor is required to be notified so any required sampling can be initiated or stopped as determined by the status of RCW, EECW, and RBCCW. [NRC LER 259/88010]

G. [CAQR/C] When the RBCCW system is drained for more than 30 days, the Technical Support Supervisor is required to be notified in order that a specific lay-up configuration is established by the System Engineer to prevent system degradation. All planned draining of RBCCW is required to be coordinated with the Chemistry Unit Supervisor and the Radwaste Coordinator. In addition, the Chemical Lab should be notified of any leaks discovered which allows RBCCW to discharge to floor drains. [CAQR BFP900249]

Excerpt from OPL171.047 page 10 of 41.

1. RBCCW Heat Loads

a. Essential loop loads

- Drywell Blowers(10)
- Reactor recirculation pump motor coolers (2)
- Reactor recirculation pump seal coolers (2)
- Drywell equipment drain sump heat exchanger (1)

b. Non-essential loop loads

- Reactor Building equipment drain sump heat exchanger (1)
- Reactor water cleanup pump seal water coolers and bearing oil coolers (2)
- RWCU Non-regenerative heat exchangers (2)
- Fuel pool cooling heat exchangers (2)
- Reactor recirculation pump discharge sample cooler (1)

Examination Outline Cross-reference:	Level	RO	SRO
295019AA1.04	Tier #	1	
Ability to operate and/or monitor the following as they apply to a Partial or Total Loss of Instrument Air: Service Air isolation valves.	Group #	1	
NOTE: Instrument Air at BFN is referred to as Control Air.	K/A #	295019AA1.04	
	Importance Rating	3.3	3.2

Proposed Question: **RO # 46**

Units 2 and 3 are operating at 100% power when a leak develops in the Control Air header, causing pressure to lower slowly. All available compressors are in service.

Which ONE of the following statements describes the response of the Service Air System?

Service Air to Control Air Crosstie Valve (0-FCV-33-1) will OPEN at __(1)__ and will ____(2)____ when Control Air pressure drops below 30 psig.

	(1)	(2)
A.	70 psig	fail open
B.	70 psig	fail closed
C.	85 psig	fail open
D.	85 psig	fail closed

Proposed Answer: **D**

Explanation:

a. Part (1) is incorrect. 70 psig is the Control Air header low pressure alarm set point. Part (2) is incorrect. 0-FCV-33-1 fails closed on low pneumatic pressure.

b. Part (1) is incorrect as stated in (a) above. Part (2) is correct.

c. Part (1) is correct. Part (2) is incorrect as stated in (a) above.

d. Correct answer.

Technical Reference(s):	0-OI-33, Service Air System	(Attach if not previously provided)
	OPL171.054, Control Air LP	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #	OPL171.054.17	attached
	Modified Bank #		
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Original question OPL171.054.17:

Units 2 and 3 are operating at 100% power when a leak develops in the control air header, causing pressure to depressurize slowly. All available compressors are in service.

Which ONE of the following statements describes the operation of the Service Air to Control Air Crosstie Valve (33-1)?

A. The valve will open at 90 psig and closes when air pressure drops to 15 psig.

B. The valve will open at 85 psig and closes when air pressure drops to 30 psig.

C. The valve will open at 80 psig and remains open until air pressure is restored to above 90 psig.

D. The valve will open at 65 psig and closes when air pressure rises above 85 psig.

BFN Unit 0	Service Air System	0-OI-33 Rev. 0064 Page 10 of 94

3.0 PRECAUTIONS AND LIMITATIONS (continued)

D. Service Air Isolation Valve, 0-FCV-33-1 will open on low Control Air System pressure. This provides a service air backup to the Control Air System.

E. During a loss of Service Air the Amertap system may release the condenser tube cleaning balls to the river.

Excerpt from OPL171.054 page 27 of 72:

(a) Service air supply valve from control air header (0-FCV-33-1). Can be operated from panel 1-9-20 and/or 3-9-20. The switch positions are **CLOSE-AUTO-OPEN,** with position indication lamps just above each control switch. The valve automatically opens if control air pressure falls to 85 psig and closes at $\approx$ 30 psig (due to insufficient air pressure to keep the valve open).

Examination Outline Cross-reference:	Level	RO	SRO
295021G2.2.36	Tier #	1	
Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions of operations: Loss of Shutdown Cooling	Group #	1	
	K/A #	295021G2.2.36	
	Importance Rating	3.1	4.2

Proposed Question: **RO # 47**

Unit 3 is in Cold Shutdown with the following plant conditions:

- Both reactor recirc pumps are removed from service for maintenance.
- RHR Loop II is in shutdown cooling with 3B RHR pump running.
- At 08:00, RHR Loop II was taken out of shutdown cooling to adjust the packing on the RHR pumps.
- At 09:30, RHR Loop II was returned to the shutdown cooling mode of operation.
- During this time RHR Loop I remained in standby.
- At 12:00, the Unit Supervisor is informed that an RHR Loop I surveillance needs to be performed that will require declaring RHR Loop I inoperable.

Which ONE of the following describes the EARLIEST time and the MAXIMUM duration that RHR Loop I may be made inoperable for surveillance testing?

RHR Loop I may be made inoperable _______(1)_______ and can remain inoperable _______(2)_______.

REFERENCE PROVIDED

	(1)	(2)
A.	immediately	as long as RHR Loop II is operable.
B.	immediately	for no longer than 2 hours.
C.	after 16:00	as long as RHR Loop II is operable.
D.	after 16:00	for no longer than 2 hours.

Proposed Answer: **A**

Explanation:

a. Correct answer

b. Part (1) is correct. Part (2) is incorrect. Since a separate entry condition is allowed for each RHR Shutdown Cooling subsystem, as long as one RHR Shutdown Cooling subsystem remains in operation, the LCO is met. Specifically, there are TWO RHR Shutdown Cooling subsystems per RHR loop.

c. Part (1) is incorrect. The 8 hour time frame is for BOTH RHR Shutdown Cooling subsystems inoperable. Part (2) is correct since the LCO is met with RHR Loop II in shutdown cooling.

d. Part (1) is incorrect as stated in (c) above. Part (2) is incorrect as stated in (b) above.

Technical Reference(s): U3 TSR 3.4.8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: U3 TSR 3.4.8

Question Source: Bank # 295021G2.2.22

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

LCO 3.4.8 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

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

1. Both required RHR shutdown cooling subsystems and recirculation pumps may not be in operation for up to 2 hours per 8 hour period.
2. One required RHR shutdown cooling subsystem may be inoperable for up to 2 hours for performance of Surveillances.

APPLICABILITY: MODE 4.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable required RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1	Verify reactor coolant circulating by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND		
	B.2	Monitor reactor coolant temperature and pressure.	Once per hour

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

BASES

BACKGROUND

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to maintain the temperature of the reactor coolant ≤ 212°F. This decay heat removal is in preparation for performing refueling or maintenance operations, or for keeping the reactor in the Cold Shutdown condition.

The RHR System has two loops with each loop consisting of two motor driven pumps, two heat exchangers, and associated piping and valves. There are two shutdown cooling subsystems per RHR System loop. Both loops have a common suction from the same recirculation loop. The four redundant, manually controlled shutdown cooling subsystems of the RHR System provide decay heat removal. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via the associated recirculation loop. The RHR heat exchangers transfer heat to the RHR Service Water System. Any one of the four RHR shutdown cooling subsystems can provide the required decay heat removal function.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	Decay heat removal by operation of the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result. The RHR Shutdown Cooling System meets Criterion 4 of the NRC Policy Statement (Ref. 1).
LCO	Two RHR shutdown cooling subsystems are required to be OPERABLE, and when no recirculation pump is in operation, one RHR shutdown cooling subsystem must be in operation. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, one RHRSW pump capable of providing cooling to the heat exchanger, and the associated piping and valves. The subsystems have a common suction source and are allowed to have common discharge piping. Since piping is a passive component that is assumed not to fail, it is allowed to be common to the subsystems. In MODE 4, the RHR cross tie valve (FCV-74-46) may be opened to allow pumps in one loop to discharge through the opposite recirculation loop to make a complete subsystem. Additionally, each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

(continued)

BASES

LCO (continued)	Note 1 permits both required RHR shutdown cooling subsystems and recirculation pumps to not be in operation for a period of 2 hours in an 8 hour period. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.
APPLICABILITY	In MODE 4, the RHR Shutdown Cooling System must be OPERABLE and shall be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 212°F. Otherwise, a recirculation pump is required to be in operation. In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR low pressure permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR low pressure permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS-Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

(continued)

REFERENCE MATERIAL

Provided to

CANDIDATE

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

LCO 3.4.8 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

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

1. Both required RHR shutdown cooling subsystems and recirculation pumps may not be in operation for up to 2 hours per 8 hour period.
2. One required RHR shutdown cooling subsystem may be inoperable for up to 2 hours for performance of Surveillances.

APPLICABILITY: MODE 4.

ACTIONS

--NOTE--

Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or two required RHR shutdown cooling subsystems inoperable.	A.1	Verify an alternate method of decay heat removal is available for each inoperable required RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1	Verify reactor coolant circulating by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND		
	B.2	Monitor reactor coolant temperature and pressure.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify one required RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

Examination Outline Cross-reference:	Level	RO	SRO
288000G2.2.44	Tier #	2	
Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions: Plant Ventilation.	Group #	2	
	K/A #	288000G2.2.44	
	Importance Rating	4.2	4.4

Proposed Question: **RO # 36**

Unit 2 is operating at 100% rated power during the summer with the following conditions:

- Control Bay Chiller "A" breaker indicates closed and reading 32 amps on Panel 9-20.
- Control Bay Chiller "B" breaker indicates closed and reading 0 amps on Panel 9-20.
- Control Bay chilled water inlet temperature is indicating 42 ^{O}F on the ICS computer.

Which ONE of the following describes the effect of transferring the CONTROL PANEL MODE SELECT switch from "DIGITAL" to "ANALOG" on the CONTROL BAY CHILLER A LOCAL DISPLAY PANEL?

Control Bay Chiller "A" will ____________________.

A. continue to run, but chilled water inlet temperature will lower to 40 ^{O}F.

B. continue to run, but will no longer supply performance data to the ICS computer.

C. immediately trip, causing a loss of chilled water to the Control Bay Air Handling Units.

D. immediately trip, then will automatically restart in the Analog Mode.

Proposed Answer: **C**

Explanation:

a. Incorrect. The chiller will trip, however the chilled water temperature setpoint while operating in Analog mode is 2 ^{O}F lower at 40 ^{O}F.

b. Incorrect. The chiller will trip, however the Chiller will not supply digital data to ICS while running in Analog Mode

c. Correct answer.

d. Incorrect. The chiller will trip as stated in (c) above. The chiller must be manually shutdown prior to transferring to Analog control, then must be manually restarted.

Technical Reference(s): 0-OI-31 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/15/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

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3.2 Unit 1/2 Control Bay Chillers

A. Compressor sump heaters are required to be energized for a minimum of 24 hours prior to starting a chiller. This prevents compressor damage caused by liquid refrigerant in the compressor at startup. When the heaters are on, the bottom end (end closest to the chiller control panel) of the compressors will be warm. This requirement may be modified by the System Engineer taking into account the length of time the compressor was shutdown and the outside air temperatures.

B. Prior to any draining being performed on the Unit 1/2 Control Bay Chilled Water System, Chemistry and Environmental should be notified of the planned system draining.

C. When transferring between analog and digital control modes on a Chiller, the chiller must NOT be running. Therefore, whenever CONTROL PANEL MODE SELECT, 0-HS-031-2110AA(2210BA), switches are to be manipulated, the associated Chiller is required to be shutdown.

D. The Chiller Control Switch located on the local display panel for each Control Bay Chiller Control Panel has three positions; STOP/RESET, AUTO LOCAL, and AUTO REMOTE. The STOP/RESET position shuts down the chiller when in local control. The AUTO LOCAL position is used to start the chiller when in local control. The AUTO REMOTE position is NOT used. The control switch should NOT be selected to the AUTO REMOTE position.

E. The Control Bay Chilled Water Pumps A or B are tripped by load shed signal. Pumps may be restarted after 10 minutes with the use of "CHW PUMP LOAD SHED BYPASS SW for each pump. (A PUMP "0-HS-031-2101E" Location 0-LPNL-925-0165 PANEL D)(B PUMP "0-HS-031-2201E" Location 0-LPNL-925-0165 PANEL E) If 1 & 2 CONT. BAY CHW PUMP A(B) TRANSFER SWITCH, 0-XS-031-2101(2201) is in LOCAL POSITION load shed logic for these pumps is bypassed.

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5.16 Startup of Unit 1/2 Control Bay Chiller A In Local Digital Control Mode (continued)

[6] **VERIFY** the following switch positions: (Unit 1/2 Control Bay Chiller A Control Panel):

- ANALOG CONTROL MODE SELECT, 0-HS-031-2110AB(HS2), in ANALOG STOP. ☐
- CONTROL PANEL MODE SELECT, 0-HS-031-2110AA(HS1), in DIGITAL CONTROL. ☐
- UNIT 1&2 CB CHILLER A REMOTE HS APPENDIX R DISC SW, 0-HS-031-2110 in LOCAL. ☐
- CONTROL BAY CHILLER A, 0-BKR-031-2110, in ON. ☐
- CONTROL BAY CHILLER A LOCAL DISPLAY PANEL 0-PMC-031-2100A Display Window is illuminated. ☐

NOTE
Step 5.16[7] starts the chiller. There is approximately a 2 minute time delay between when the switch is placed in AUTO/LOCAL and when the chiller actually starts.

[7] At CONTROL BAY CHILLER A LOCAL DISPLAY PANEL, 0-PMC-031-2100A, **PLACE** switch in AUTO LOCAL. ☐

[8] **CHECK** the following parameters for Chiller A:

- 1&2 CB CHW PUMP A INLET PRESSURE, 0-PI-031-0194, is approximately 15-30 psig. ☐
- 1&2 CB CHW PUMP A(B) DISCHARGE PRESSURE, 0-PI-031-0195, is approximately 85-105 psig. ☐
- On 0-PMC-031-2100A (Menu-PO point F), Evap Leaving Water Temp eventually lowers to 42 ± 2°F. ☐

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5.18 Startup of Unit 1/2 Control Bay Chiller A In Analog Control Mode (continued)

NOTE
Step 5.18[8] starts the chiller at (Unit 1/2 Control Bay Chiller A Control Panel). There is approximately a 5 minute time delay between when the switch is placed in AUTO and when the chiller actually starts.

[8] **PLACE** ANALOG CONTROL MODE SELECT, 0-HS-031-2110AB(HS2), in ANALOG START. ☐

[9] **RECORD** switch manipulations in Steps 5.18[7] and 5.18[8] in Narrative Log. ☐

[10] **CHECK** the following parameters on Chiller A:

- 1&2 CB CHW PUMP A INLET PRESSURE, 0-PI-031-0194, is approximately 15-30 psig. ☐
- 1&2 CB CHW PUMP A(B) DISCHARGE PRESSURE, 0-PI-031-0195, is approximately 85-105 psig. ☐
- **VERIFY** Evaporator Leaving Water Temp eventually lowers to 40 ± 5°F on 1&2 CB CHILLER A CHW OUTLET TEMP IND, 0-TI-031-0023. ☐

Examination Outline Cross-reference:	Level	RO	SRO
295023AK3.03	Tier #	1	
Knowledge of the reasons for the following responses as they apply to Refueling Accidents: Ventilation isolation.	Group #	1	
	K/A #	295023AK3.03	
	Importance Rating	3.3	3.6

Proposed Question: **RO # 48**

Unit 1 is performing a fuel pool cleanout when a failure of the Main Grapple hooks result in an irradiated fuel bundle being dropped into the reactor vessel.

Unit 1 Ventilation Radiation Monitors read as follows:

- Channel A	Reactor Zone Detector A	1-RM-90-142A	100 MR/HR
	Reactor Zone Detector B	1-RM-90-142B	100 MR/HR
	Refuel Zone Detector A	1-RM-90-140A	40 MR/HR
	Refuel Zone Detector B	1-RM-90-140B	60 MR/HR
- Channel B	Reactor Zone Detector A	1-RM-90-143A	45 MR/HR
	Reactor Zone Detector B	1-RM-90-143B	62 MR/HR
	Refuel Zone Detector A	1-RM-90-141A	100 MR/HR
	Refuel Zone Detector B	1-RM-90-141B	100 MR/HR

Based on the above conditions, which ONE of the following describes the response of the plant ventilation system?

Refuel Zone Supply and Exhaust fans on ______(1)______ and Reactor Zone Supply and Exhaust fans on ______(2)______.

	(1)	(2)
A.	Unit 1 only will trip	all three units will trip.
B.	Unit 1 only will trip	Unit 1 only will trip.
C.	all three units will trip.	all three units will trip.
D.	all three units will trip.	Unit 1 only will trip.

Proposed Answer: **D**

Explanation:

a. Part (1) is incorrect. All Refuel Zones are tied together and will trip in high radiation. Part (2) is incorrect. Reactor Zone ventilation is separate for each unit. Only the effected unit's RB ventilation will trip.

b. Part (1) is incorrect as stated in (a) above. Part (1) is correct.

c. Part (1) is correct. Part (2) is incorrect as stated in (a) above.

d. Correct answer.

Technical Reference(s):	OPL171.067, HVAC	(Attach if not previously provided)
	U1 TSB Section 3.3.6.2	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #	295023AK2.05	Attached
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments: This question was considered MODIFIED due to the original question being incorrect. It implied that ONLY Unit1 Refuel Zone trips when all three units would trip. The correct answer was derived by eliminating the three distracters due to incorrect statements. That made the required answer the "most correct" choice rather than "the" correct choice.

Original question: 295023AK2.05

Unit 1 is performing a fuel pool cleanout when a failure of the Reactor Building overhead crane results in an irradiated LPRM string raised above fuel pool water level.

Unit 1 Ventilation Rad Monitors read as follows:

- Channel A	Reactor Zone Detector A	1-RM-90-142A	100 MR/HR
	Reactor Zone Detector B	1-RM-90-142B	100 MR/HR
	Refuel Zone Detector A	1-RM-90-140A	40 MR/HR
	Refuel Zone Detector B	1-RM-90-140B	60 MR/HR
- Channel B	Reactor Zone Detector A	1-RM-90-143A	45 MR/HR
	Reactor Zone Detector B	1-RM-90-143B	62 MR/HR
	Refuel Zone Detector A	1-RM-90-141A	100 MR/HR
	Refuel Zone Detector B	1-RM-90-141B	100 MR/HR

Based on the above conditions, which ONE of the following describes the response of plant ventilation systems?

A. Unit 1's Refueling AND Unit 1's Reactor Zone Supply and Exhaust Fans trip.

B. All Refueling AND Unit 1's Reactor Zone Supply and Exhaust fans trip and both CREV units start.

C. All Reactor Zone Supply and Exhaust fans trip and the CREV System starts and all SGTS fans start.

D. All SGTS fans start, the preferred CREV unit and the Unit 1/3 Board Room Emergency Supply Fans start.

Excerpt from OPL171.067 page 18 of 71:

1. System Isolation (Group 6)

   a. The isolation signals are low reactor water level +2", high drywell pressure 2.45 psig, and high radiation in exhaust duct (72 MR/hr Refuel zone or Reactor zone).

   b. On auto isolation signal (except Refuel Zone high radiation) the unit reactor zone supply and exhaust fans trip, all refuel zone supply and exhaust fans trip, unit reactor zone and all refuel zone supply and exhaust isolation dampers close, and dampers to Standby Gas Treatment System open and SGT train blowers auto start.

   c. Note: Damper logic is as follows: PCIS Group 6 with A SGT running opens 64-41 and 45; PCIS Group 6 with B SGT running opens 64-40 and 44.

   d. On isolation due to high radiation in Refuel Zone all refuel zones isolate, refuel zone supply and exhaust fans trip, SGT starts and aligns dampers for refuel zone only.

Excerpt from OPL171.017 page 22 of 56:

n. Reactor Building Ventilation Exhaust Radiation High

- Reactor Building ventilation exhaust radiation is monitored by two sets of two (four detectors total) gamma-sensitive radiation monitors located on the reactor building ventilation exhaust duct.
- One channel at the high trip setpoint or two channels in a combination of downscale and/or inop will cause isolation.

o. Refuel Zone Ventilation Exhaust Radiation High

- Refuel zone ventilation exhaust radiation is monitored by two sets of two (four detectors total) gamma-sensitive radiation monitors located on the refuel zone ventilation exhaust duct.
- One channel at the high trip setpoint or two channels in a combination of downscale and/or inop will cause isolation.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Drywell Pressure - High (PIS-64-56A-D) (continued)

The Allowable Value was chosen to be the same as the ECCS Drywell Pressure - High Function Allowable Value (LCO 3.3.5.1) since this is indicative of a loss of coolant accident (LOCA).

The Drywell Pressure - High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This Function is not required in MODES 4 and 5 because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES.

3, 4. Reactor Zone Exhaust and Refueling Floor Radiation - High (RM-90-140, 141, 142, 143)

High secondary containment exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB. When Exhaust Radiation - High is detected, secondary containment isolation and actuation of the SGT System are initiated to limit the release of fission products as assumed in the FSAR safety analyses (Ref. 4).

The Exhaust Radiation - High signals are initiated from radiation detectors located on the reactor zone ventilation exhaust and the common refueling zone. There are two radiation monitors and two divisional trip systems for each unit (Units 1, 2, and 3). Each monitor has one channel of Reactor Zone Exhaust Radiation - High and one channel of Refueling Floor Radiation - High. Each monitor's channels provide signals to its associated divisional trip system. Each channel has two radiation elements which monitor the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	3, 4. Reactor Zone Exhaust and Refueling Floor Radiation - High (RM-90-140, 141, 142, 143) (continued)

ventilation exhaust both of which must be OPERABLE or tripped for the channel to be OPERABLE. Both radiation elements must provide a High signal to trip the associated channel (two-out-of-two). However, the output relays from the divisional trip systems are arranged in logic systems such that if either channel for a zone trips, a secondary containment isolation signal is initiated (one-out-of-two). Six channels of Reactor Zone Exhaust Radiation - High Function and six channels of Refueling Floor Radiation - High Function are available (two channels of each Function from each unit) and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Values are chosen to provide timely detection of nuclear system process barrier leaks inside containment but are far enough above background levels to avoid spurious isolation.

The Reactor Zone Exhaust and Refueling Floor Radiation - High Functions are required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, these Functions are not required. In addition, the Functions are also required to be OPERABLE during OPDRVs because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery) must be provided to ensure that offsite dose limits are not exceeded.

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
295024EA1.07	Tier #	1	
Ability to operate and/or monitor the following as they apply to High Drywell Pressure: PCIS/NSSSS.	Group #	1	
	K/A #	295024EA1.07	
	Importance Rating	3.8	3.9

Proposed Question: **RO # 49**

Unit 2 is at 100% rated power with the following conditions:

- DRYWELL NORM OPERATING PRESS HIGH (9-3B W 19) in alarm.
- DRYWELL PRESS APPROACHING SCRAM (9-3B W 30) in alarm.
- DRYWELL PRESSURE ABNORMAL (9-5B W 31) in alarm.
- DRYWELL FD SUMP LEVEL ABN (9-4C W 2) in alarm.
- Drywell venting is in progress using 2-AOI-64-1, "Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell."
- Drywell pressure is 2.2 psig and steady.

Assuming no further operator action, which ONE of the following describes the plant response if 480V Shutdown Board 2A de-energized due to an electrical fault?

The Drywell vent lineup would be ____(1)____. Drywell pressure would ________(2)______.

	(1)	(2)
A.	unaffected	lower due to non-essential RBCCW loads isolating.
B.	unaffected	rise due to RPS A de-energizing.
C.	isolated	lower due to non-essential RBCCW loads isolating.
D.	isolated	rise due to RPS A de-energizing.

Proposed Answer: **D**

Explanation:

a. Part (1) is incorrect. RPS A would de-energize, causing the vent lineup to isolate, since RPS supplies PCIS isolation valve logic which would fail closed on loss of power. Part (2) is incorrect. Under normal circumstances, DW pressure would lower due to non-essential loads isolating on low RBCCW discharge pressure. This acts to increase cooling flow to the drywell blowers. However, with the vent lineup isolated and a leak in the drywell, pressure would begin to rise again.

b. Part (1) is incorrect as stated in (a) above. Part (2) is correct. With the vent lineup isolated and a leak in the drywell, pressure would begin to rise again.

c. Part (1) is correct. RPS A would de-energize, causing the vent lineup to isolate. Part (2) is incorrect as stated in (a) above.

d. Correct answer.

Technical Reference(s):	2-AOI-64-1, ARPs 9-5B, 9-3B	(Attach if not previously provided)
	2-AOI-99-1	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/03/2008 RMS	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

BFN Unit 2	Drywell Pressure and/or Temperature High, or Excessive Leakage into Drywell	2-AOI-64-1 Rev. 0023 Page 5 of 12

2.3 Symptoms for High Drywell Temperature

- DRYWELL NORM OPERATING PRESS HIGH (2-XA-55-3B, Window 19)
- Drywell temperature rising, as indicated on DRYWELL TEMPERATURE/PRESSURE, 2-XR-064-050 (Panel 2-9-3)
- Drywell pressure rising, as indicated on DRYWELL TEMPERATURE/PRESSURE, 2-XR-064-050 (Panel 2-9-3)

2.4 Symptoms for Drywell Excessive Leakage

- DRYWELL NORM OPERATING PRESS HIGH (2-XA-55-3B, Window 19)
- DRYWELL FD SUMP LEVEL ABN (2-XA-55-4C, Window 2)
- DRYWELL EQPT DR SUMP LEVEL ABN (2-XA-55-4C, Window 9)
- RBCCW SURGE TANK LEVEL LOW (2-XA-55-4C, Window 13)
- DRYWELL EQPT DR SUMP TEMP HIGH (2-XA-55-4C, Window 16)
- REACTOR WATER LEVEL ABNORMAL (2-XA-55-5A, Window 8)
- RECIRC PUMP A NO. 2 SEAL LEAKAGE HIGH 2-FA-68-55 (2-XA-55-4A, Window 18)
- RECIRC PUMP A NO. 1 SEAL LEAKAGE ABN 2-FA-68-62 (2-XA-55-4A, Window 25)
- RECIRC PUMP B NO. 2 SEAL LEAKAGE HIGH 2-FA-68-68 (2-XA-55-4B, Window 18)
- RECIRC PUMP B NO. 1 SEAL LEAKAGE ABN 2-FA-68-74 (2-RA-55-4B, Window 25)

BFN Unit 2	2-XA-55-3B	2-ARP-9-3B Rev. 0020 Page 22 of 38

DRYWELL NORM OPERATING PRESS HIGH 2-PA-64-135 (19)	Sensor/Trip Point: 2-PS-64-135 or 2-PS-64-136	1.6 psig

(Page 1 of 1)

Sensor Location: Panel 9-19
Elevation 593'
Auxiliary Instrument Room

Probable Cause:
A. Excessive N_2 makeup.
B. Low pressure front moving through area.
C. Drywell DP compressor malfunction.
D. Drywell blowers failure.
E. Steam or water leak inside drywell.
F. Possible Drywell Control Air System In leakage.
G. Sensor malfunction.
H. Reactor Start-up.

Automatic Action: None

Operator Action:
A. **CHECK** drywell pressure and temperature for rise **AND CHECK** weather report for atmospheric pressure. ☐
B. **CHECK** to see if Drywell DP Compressor is running. **IF** Drywell DP Compressor is running, **THEN** **STOP** compressor. ☐
C. **CHECK** N_2 makeup valves to Suppression Chamber and Drywell closed. ☐
D. **CHECK** Drywell Control Air System Flow Elements 2-FIQ-032-00092 (Rx Bldg 565' R10-S) and 2-FIQ-032-0075 (Rx Bldg 565' R20-T0) < 1.7 SCFM. ☐
E. **IF** pressure rise is due to normal startup, **THEN** **REFER TO** 2-OI-64 for normal venting instructions. ☐
F. **IF** Drywell pressure is high, **THEN** **REFER TO** 2-AOI-64-1. ☐

References: 2-45E620-8 2-45E777-21

BFN Unit 2	2-XA-55-3B	2-ARP-9-3B Rev. 0020 Page 33 of 38

DRYWELL PRESS
APPROACHING
SCRAM
2-PA-64-58

RED BAR 30

(Page 1 of 1)

Sensor/Trip Point:

PIS-64-58E
PIS-64-58F 1.96 psig
PIS-64-58G
PIS-64-58H

Sensor Location:	PIS-64-58E, 58G Aux Inst. Rm. Panel 9-81	PIS-64-58F, 58H Aux Inst. Rm. Panel 9-82

Probable Cause:

A. Drywell pressure rising.
B. Drywell cooler(s) failure.
C. Steam or water leak inside Drywell.
D. Loss of RBCCW to Drywell coolers.

Automatic Action: None

Operator Action:

A. **CHECK** containment pressure and temperature using multiple indications. ☐
B. **REFER TO** 2-AOI-64-1. ☐

References: 45N620-3   2-47E610-64-1   47W600-57

BFN Unit 2	Panel 9-5 2-XA-55-5B	2-ARP-9-5B Rev. 0025 Page 36 of 43

DRYWELL PRESSURE ABNORMAL 2-PA-64-56

31

(Page 1 of 1)

Sensor/Trip Point:

2-PS-64-56E	1.96 psig rising
2-PS-64-56F	0.1 psig lowering

Sensor Location: Panel 25-5B
Elevation 593
Col R-3 S-LINE

Probable Cause:

A. Drywell DP air compressor failure.
B. Loss of RBCCW.
C. Breach of Primary Containment.
   1. Drywell vent valves open or leaking.
   2. Drywell vacuum breaker open or leaking.

D. LOCA.
E. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **VERIFY** alarm using multiple indications. ☐
B. **IF** RBCCW has been lost, **THEN** **REFER TO** 2-AOI-70-1. ☐
C. **REFER TO** 2-AOI-64-1. ☐

References:

2-45E620-6	2-47E610-64-1	2-730E915-17
2-AOI-70-1	2-AOI-64-1	

BFN Unit 2	Loss of Power to One RPS Bus	2-AOI-99-1 Rev. 0025 Page 4 of 8

3.0 AUTOMATIC ACTIONS

NOTE
An overview of automatic actions for RPS Bus A(B) is provided here. A detailed list of actions is provided in 2-OI-99, Illustration 1, which lists actions that occur when RPS buses are de-energized on a transfer of power supply.

A. RPS trip logic A(B) half-scram occurs.

B. PCIS Group 1 half-trip logic de-energizes.

C. PCIS Group 2 isolation, RHR Shutdown Cooling Mode:

1. Bus A inboard.
2. Bus B outboard.

D. PCIS Group 3 isolation, RWCU:

1. Bus A inboard and outboard.
2. Bus B outboard.

E. PCIS Group 6 isolation, Primary Containment Vent and Purge and Reactor Building Ventilation:

1. Bus A or B inboard and outboard.

F. Group 8 isolation, TIP.

G. Control Room Emergency Ventilation System start.

H. Standby Gas Treatment System starts.

Examination Outline Cross-reference:	Level	RO	SRO
295025G2.1.31	Tier #	1	
Ability to locate control room switches, controls and indications and to determine that they are correctly reflecting the desired plant lineup: High Reactor Pressure.	Group #	1	
	K/A #	295025G2.1.31	
	Importance Rating	4.6	4.3

Proposed Question: **RO # 50**

Given Unit 1 at 100% rated power:

Which ONE of the following describes the unit response if the Max Combined Flow Limit setting was inadvertently reduced from 125% to 75% over two minutes?

Main Generator load would _____(1)_____, Main Turbine Bypass Valves would _____(2)_____ and reactor pressure would _____(3)_____.

	(1)	(2)	(3)
A.	remain the same	open	remain the same
B.	lower	remain closed	rise
C.	remain the same	remain closed	remain the same
D.	lower	open	rise

Proposed Answer: **B**

Explanation:

a. Part (1) is incorrect. Generator load will remain the same until the MCFL setting drops below 100%. At that point the TCVs will begin to close. Part (2) is incorrect. If the Generator Load Set setting was being lowered, this would be correct. Part (3) is incorrect, but is consistent with the expected response if the Generator Load Set setting was being lowered.

b. Correct answer.

c. Part (1) is incorrect as stated in (a) above. Part (2) is correct. Part (3) is incorrect but is consistent with the expected response as the MCFL is lowered from 125% to 100%. Once the MCFL is lowered past 100%, the conditions in (b) would occur.

d. Part (1) is correct as the MCFL lowers below 100%. Part (2) is incorrect as stated in (a) above. Part (3) is correct for the given conditions but is not consistent with Bypass Valves opening.

Technical Reference(s): 1-OI-47, OPL171.228 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/04/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 1	Turbine-Generator System	1-OI-47 Rev. 0013 Page 12 of 220

3.0 PRECAUTIONS AND LIMITATIONS

A. A Turbine-Generator trip is **NOT** to be reset before the cause of the trip is clearly established and corrective actions considered.

B. If the hydrogen seal oil is lost, the Turbine is to be tripped and the hydrogen dumped and the Generator purged immediately in accordance with 1-OI-35.

C. The Turbine is to be tripped if stator coolant conductivity exceeds 9.9 µmho.

D. Do **NOT** select a set speed which is lower than current Turbine speed. If deceleration is desired, trip the Turbine and place on turning gear.

E. The EHC Control System can be used in either Reactor Pressure control or Header Pressure control. While in Header Pressure control, a single header pressure input failing high could cause the bypass valves to open. While in Reactor Pressure control, a single Reactor Pressure input failing high will **NOT** affect the bypass valves. For this reason, Reactor Pressure control is the preferred mode of operation for the EHC Control System.

F. The COLR Thermal Limit analysis allows for a Turbine Bypass Valve and/or the Recirc Pump Trip to be out of service. Therefore, the EOC-RPT logic can remain disabled should a Turbine Bypass Valve become inoperative. The Unit 1 TRM COLR should be referred to for the appropriate Thermal limits and off-rated corrections when either Turbine Bypass out-of-service conditions exist or when the Recirculation Pump Trip is out-of-service.

G. The following pertain to the Max Combined Flow Limit:

1. Max combined flow limit setting of 150% (upper limit) precludes exceeding thermal limits during a single Turbine control valve closure.

2. The max combined flow upper and lower setting limits are 50% and 150%. Normally it is set at 125%.

3. The max combined flow limit setting is adjustable only on the EHC WORK STATION computer (Panels 1-9-7 and 1-9-31).

4. Max combined flow limit setpoint can be found on the following computer screens:

   a. On ICS, EHC TURBINE CONTROL (EHCTC) screen.

   b. On EHC WORK STATION, TURBINE CONTROL screen.

Excerpt from OPL171.228 page 13 of 81:

a. The **Maximum Combined Flow Limiter** limits the valve opening for the control valves and bypass valves and is **only adjustable through the EHC Workstation**.

Obj. V.B.3
50-150% setting
OI-47 sets this at 125% flow demand

b. In the event the maximum combined flow limit becomes active, the MAX CF indicator will illuminate at the MCR panel and the EHC Workstation.

c. All of the above parameters are low signal selected on a low signal select block. The output of the low signal select block will be the lowest input value provided the value is not higher than the high limit or lower than the low limit.

The low limit and high limit values are set for 0% and 100%.

d. The lowest input to the low signal select bus and the parameter that is in control is illuminated on the MCR panel and the EHC workstation.

Examination Outline Cross-reference:	Level	RO	SRO
295026EK3.01	Tier #	1	
Knowledge of the reasons for the following responses as they apply to Suppression Pool High Water Temp: Emergency/Normal depressurization.	Group #	1	
	K/A #	295026EK3.01	
	Importance Rating	3.8	4.1

Proposed Question: **RO # 51**

Which ONE of the following describes the Suppression Pool temperature limit and bases for injecting Standby Liquid Control into the reactor?

Standby Liquid Control must be injected prior to exceeding __(1)__ ^{O}F. This is based on ________(2)________ prior to Emergency Depressurization being required due to high Suppression Pool temperature.

	(1)	(2)
A.	95	ensuring the reactor is subcritical
B.	95	maintaining Suppression Pool pH
C.	110	ensuring the reactor is subcritical
D.	110	maintaining Suppression Pool pH

Proposed Answer: **C**

Explanation:

a. Part (1) is incorrect. This is the entry condition for Primary Containment Control based on SP temperature. Part (2) is correct.

b. Part (1) is incorrect as stated in (a) above. Part (2) is incorrect for the given conditions. SLC injection is required for pH control of the Suppression Pool but is based on Drywell high radiation, not Suppression Pool high temperature.

c. Correct answer.

d. Part (1) is correct. Although "prior to 95 ^{O} F" is also "prior to 110 ^{O} F", there is no procedure guidance to inject SLC before 95 ^{O} F based on SP temperature. Part (2) is incorrect as stated in (b) above.

Technical Reference(s):	EOIPM SECTION 0-V-C page 116	(Attach if not previously provided)
	1-EOI-1 flowchart, OPL171.203	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/04/2008 RMS	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

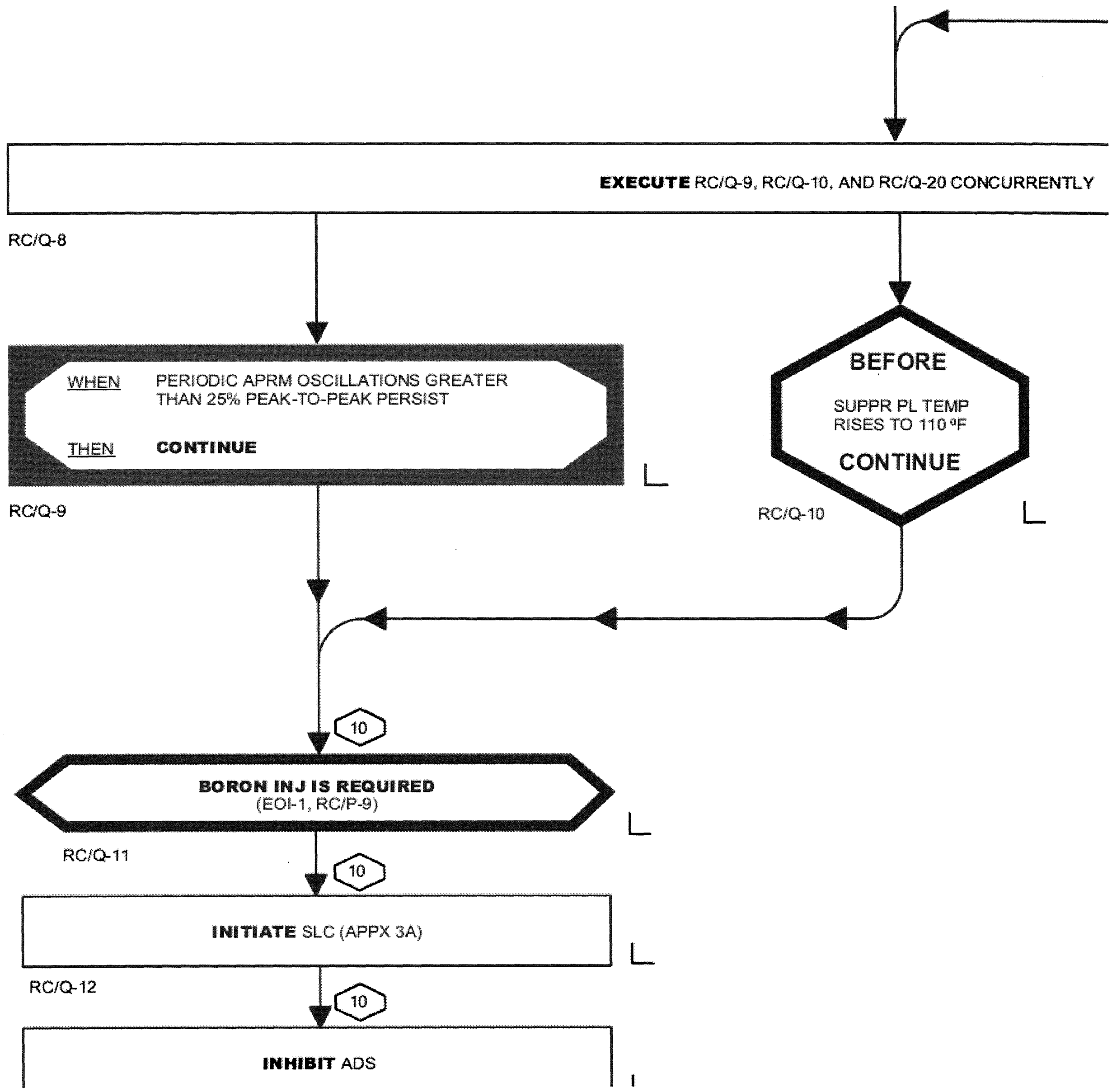
Excerpt from OPL171.203 page 42 of 66.

1. Step SP/T-4

   a. This before decision step has the operator evaluate current and future efforts to reduce suppression pool temperature, in relation to the current value and trend of suppression pool temperature, to determine if a reactor shutdown is necessary.

   b. The before decision step requires that this determination and subsequent actions be performed before suppression pool temperature reaches 110°F, the temperature at which boron injection would be required if the reactor was not subcritical.

   c. Calculations have determined that if suppression pool temperature reaches 110°F before boron injection is initiated, there is no assurance that the reactor will be subcritical when emergency RPV depressurization is required due to exceeding the Heat Capacity Temperature Limit.


EXECUTE RC/Q-9, RC/Q-10, AND RC/Q-20 CONCURRENTLY
RC/Q-8
WHEN PERIODIC APRM OSCILLATIONS GREATER THAN 25% PEAK-TO-PEAK PERSIST
THEN CONTINUE
RC/Q-9
BEFORE
SUPPR PL TEMP RISES TO 110 ºF
CONTINUE
RC/Q-10
10
BORON INJ IS REQUIRED
(EOI-1, RC/P-9)
RC/Q-11
10
INITIATE SLC (APPX 3A)
RC/Q-12
10
INHIBIT ADS

BFN Unit 1	Panel 9-7 1-XA-55-7C	1-ARP-9-7C Rev. 0017 Page 20 of 41

DRYWELL RADIATION HIGH 1-RA-90-272 SOLID MAGENTA 15	

(Page 1 of 1)

Sensor/Trip Point:

1-RE-90-272A	10 R/HR
1-RE-90-273A	10 R/HR
1-RE-90-272B	No set point (later)
1-RE-90-273B	No set point (later)

Sensor Location:
1-RM-90-272A, Panel 1-9-54
1-RM-90-273A, Panel 1-9-55
1-RM-90-272B, Panel 1-9-54
1-RM-90-273B, Panel 1-9-55

Probable Cause:
A. Noise spikes.
B. High radiation (post accident monitor).

Automatic Action: None

Operator Action:
A. **VERIFY** alarm on 1-RR-90-272 on Panel 1-9-54 and 1-RR-90-273 on Panel 1-9-55. ☐
B. **CHECK** 1-RR-90-256 for rising indication. ☐
C. **ATTEMPT** to isolate equipment to stop source. ☐
D. **IF** the alarm is determined to be valid, **THEN**, **PERFORM** the following within 2 hours of alarm:
- **OPEN** UPSTREAM MSL DRAIN TO CONDENSER 1-FCV-001-0058. ☐
- **OPEN** DOWNSTREAM MSL DRAIN TO CONDENSER 1-FCV-001-0059. ☐
- **ENSURE** 1-PCV-001-0147 is Closed by taking STEAM SEAL REGULATOR, 1-HS-1-147 to **CLOSE**. (Panel 1-9-7) ☐
- **CLOSE** MAIN STEAM TO OG PREHEATER 1A ISOL, 1-SHV-001-0741.(SJAE RM B) ☐
- **CLOSE** MAIN STEAM TO OG PREHEATER 1B ISOL, 1-SHV-001-0743.(SJAE RM B) ☐

Continued on Next Page

BFN Unit 1	Panel 9-7 1-XA-55-7C	1-ARP-9-7C Rev. 0017 Page 21 of 41

E. IF ALL the following conditions exist:
- Alarm is determined to be valid. ☐
- The reactor will remain sudcritical without boron injection under all conditions ☐
- Leakage of primary coolant into primary containment is indicated ☐

THEN within 2 hours of alarm, **INJECT** SLC for alternate source term control by placing SLC PUMP 1A/1B, 1-HS-63-6A in the START A OR START B position. ☐

F. **REFER TO** EPIPs. ☐

G. IF started at Operator Action Step 5, **THEN WHEN** SLC tank reaches "0", **STOP** the running SLC Pump. ☐

References: 1-45E620-9-1, 2 0-47E610-90-2
Technical Specifications 3.3.3.1

Examination Outline Cross-reference:	Level	RO	SRO
295028EK2.03	Tier #	1	
Knowledge of the interrelations between High Drywell Temperature and Reactor Water Level indication.	Group #	1	
	K/A #	295028EK2.03	
	Importance Rating	3.6	3.8

Proposed Question: **RO # 52**

A LOCA has occurred on Unit 2 resulting in the following conditions:

- 2-EOI-1, "RPV Control" and 2-EOI-2, "Primary Containment Control" are being executed
- Drywell Sprays could not be initiated due to logic failures.
- Drywell pressure 15 psig and slowly rising.
- Drywell temperature 305 ^{O}F and steady.
- Suppression Pool level 15.5 feet.
- Suppression pool temperature 140 ^{O}F and steady.
- ADS was manually initiated due to high Drywell temperature.
- The six ADS valves have now closed due to low reactor pressure
- Normal range level indicates (+) 34 inches.
- Emergency range level indicates (+) 58 inches.
- Shutdown Floodup level indicates (+) 30 inches.

Which ONE of the following describes the current status of RPV level instrumentation?

Reactor water level ___(1)___ be determined. The Shutdown Floodup instrument _____(2)_____ be used for trend indication.

REFERENCE PROVIDED

	(1)	(2)
A.	CAN	CAN
B.	CAN	CANNOT
C.	CANNOT	CAN
D.	CANNOT	CANNOT

Proposed Answer: **B**

Explanation:

a. Part (1) is correct. Although the "Action Required" region of Curve 8 has been entered, no indication of erratic level instruments have been given in the stem. Caution 1 part B states instruments MAY be unreliable. Part (2) is incorrect. The Shutdown Floodup level indicates below the Minimum Indicated Level on Caution 1 which implies actual level may be below the variable leg instrument tap.

b. Correct answer.

c. Part (1) is incorrect. Both Normal and Emergency range level indicators are reliable under the given conditions. Part (2) is incorrect as stated in (a) above.

d. Part (1) is incorrect as stated in (c) above. Part (2) is correct. The Shutdown Floodup level indicates below the Minimum Indicated Level on Caution 1 which implies actual level may be below the variable leg instrument tap.

Technical Reference(s):	2-EOI-2 Curve 8 and Caution 1	(Attach if not previously provided)
	OPL171.201	

Proposed references to be provided to applicants during examination: EOI Curve 8 and Caution 1

Question Source:	Bank #		
	Modified Bank #	OPL171.203.82	Attached
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Original question OPL171.203.82:

A LOCA coupled with an inability to spray the drywell has resulted in the operators Emergency Depressurizing the reactor.

The following conditions are present:

DW pressure 15 psig and slowly rising
DW temperature 305 degrees and steady
Suppression Pool level 15.5 feet
Suppression pool temperature 140 degrees and steady
The six ADS valves have just closed due to low reactor pressure
The normal and emergency systems range level indicators are reading off-scale high
The 3-55 level indicator is reading +50"
EOI-1 & 2 are being executed

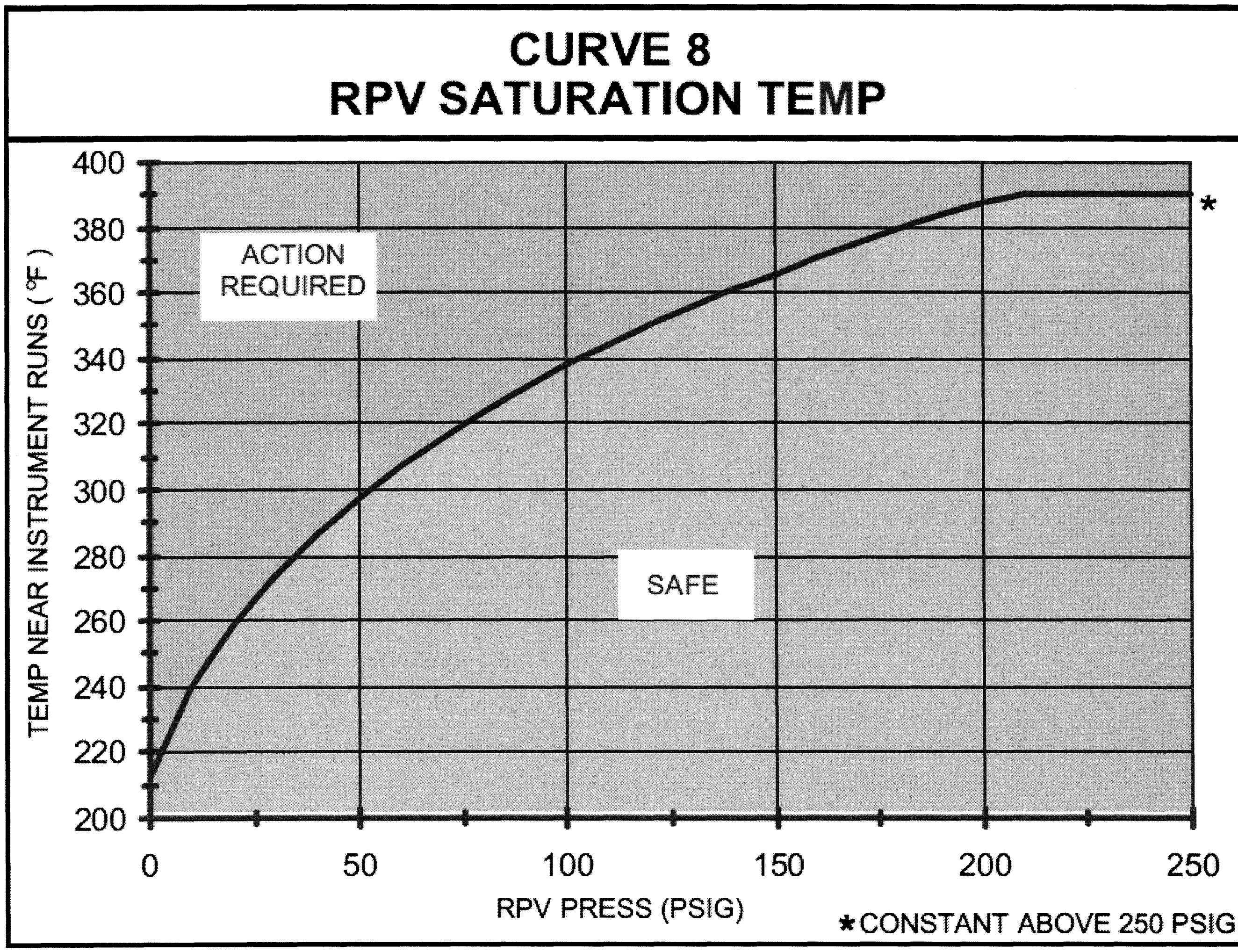
What additional actions (if any) should be taken?

A. Reactor level CAN be determined, continue to execute EOI-1 and EOI-2.

B. Reactor level CAN be determined, enter C-1.

C. Reactor level CANNOT be determined, enter C-4.

D. Reactor level CANNOT be determined, enter C-1.


CURVE 8
RPV SATURATION TEMP
400
380
360
340
320
300
280
260
240
220
200
TEMP NEAR INSTRUMENT RUNS (°F)
ACTION REQUIRED
SAFE
0
50
100
150
200
250
RPV PRESS (PSIG)
*
* CONSTANT ABOVE 250 PSIG

CAUTIONS

CAUTION #1

- AN RPV WATER LVL INSTRUMENT MAY BE USED TO DETERMINE OR TREND LVL ONLY WHEN IT READS ABOVE THE MINIMUM INDICATED LVL ASSOCIATED WITH THE HIGHEST MAX DW OR SC RUN TEMP.
- IF DW TEMPS, OR SC AREA TEMPS (TABLE 6), AS APPLICABLE, ARE OUTSIDE THE SAFE REGION OF CURVE 8, THE ASSOCIATED INSTRUMENT MAY BE UNRELIABLE DUE TO BOILING IN THE RUN.

INSTRUMENT	RANGE	MINIMUM INDICATED LVL	MAX DW RUN TEMP (FROM XR-64-50 OR TI-64-52AB)	MAX SC RUN TEMP (FROM TABLE 6)
LI-3-58A, B	EMERGENCY -155 TO +60	ON SCALE	N/A	BELOW 150
		-145	N/A	151 TO 200
		-140	N/A	201 TO 250
		-130	N/A	251 TO 300
		-120	N/A	301 TO 350
LI-3-53 LI-3-60 LI-3-206 LI-3-253 LI-3-208A, B, C, D	NORMAL 0 TO +60	ON SCALE	N/A	BELOW 150
		+5	N/A	151 TO 200
		+15	N/A	201 TO 250
		+20	N/A	251 TO 300
		+30	N/A	301 TO 350
LI-3-52 LI-3-62A	POST ACCIDENT -268 TO +32	ON SCALE	N/A	N/A
LI-3-55	SHUTDOWN FLOODUP 0 TO +400	+10	BELOW 100	N/A
		+15	100 TO 150	N/A
		+20	151 TO 200	N/A
		+30	201 TO 250	N/A
		+40	251 TO 300	N/A
		+50	301 TO 350	N/A
		+65	351 TO 400	N/A

Excerpt from OPL171.201 Page 32 of 117:

1. **Caution #1**

a. RPV water level instrument systems sense liquid level in the vessel downcomer region by measuring differential pressure (dP) between a variable leg water column and a reference leg water column. The reference leg **remains** full of water from steam condensing in the chamber located at the top of the reference leg water column. Excess condensate drains back into the RPV. To ensure reference leg water remains gas free a trickle flow of CRDH water is continuously injected into the 4 primary reference legs.

b. When water level in the reactor vessel lowers, variable leg height of water decreases, sensed dP increases, and indicated RPV water level lowers. The converse occurs when water level in the reactor vessel increases; variable leg height of water increases, sensed dP decreases, and indicated RPV water level increases.

c. Changes in height or density of water in the instrument reference leg can cause changes in indicated RPV water level. For example: if actual RPV water level is constant at some on-scale value and the instrument reference leg head of water (height and/or density) decreases, sensed dP decreases and indicated RPV water level increases. Under extreme conditions, a high and increasing drywell or containment temperature can decrease the density of water in the reference leg such that the instrument falsely indicates an on-scale and steadily increasing water level even though the actual RPV water level is decreasing and well below the elevation of the instrument variable leg tap.

d. It is important to note that the information presented in Caution #1 is not just a simple accommodation for inaccuracies in RPV water level indication which occur when plant conditions are different from those for which the instruments are calibrated. Rather, the caution defines conditions under which the displayed value and the indicated trend of RPV water level cannot be relied upon.

e. Part B of Caution #1 identifies the limiting conditions beyond which water in instrument legs may boil. Water in the RPV water level instrument legs is maintained in a liquid state by cooling action of the surrounding atmosphere and pressure in the reactor vessel. Water in the instrument legs will boil, however, if its temperature exceeds saturation temperature for the existing RPV pressure.

f. Boiling is a concern in both horizontal and vertical reference and variable instrument leg runs. Boil-off from reference leg water inventory reduces the reference head of water, decreases dP sensed by the instrument, and results in an erroneously high indicated RPV water level. Boiling in the instrument's variable leg exerts increased pressure on the variable leg side of the dP cell. This effect results in a lower sensed dP and an erroneously high indicated RPV water level.

g. Part B of Caution #1 references the RPV Saturation Temperature Curve (Curve 8) The RPV Saturation Temperature Curve is generic, based simply on the properties of water. The axis for RPV pressure is plotted from atmospheric pressure to the pressure setpoint of the lowest lifting MSRV. Note that the temperature axis of the RPV Saturation Temperature Curve is not simply drywell temperature. Depending upon the relative location of instrument reference legs and variable legs, indications from monitors near instrument runs must be considered.

h. Because BFN does not have the capability of directly reading temperature indications near instrument runs located in secondary containment, the RPV Saturation Temperature Curve (Curve 8) is supplemented with Table 6, Secondary Containment Instrument Runs. Table 6 identifies the temperature elements and general locations for the instrument runs to each RPV water level instrument.

i. Caution 1 part B says instruments “may be unreliable” if Curve 8 is exceeded. This means instruments may continue to be used until and unless erratic indication is observed since momentary excursions (expected in some post LOCA situations) into curve 8 unsafe region will not result in boiling. If, however, indications of boiling are observed then that instrument is unusable until the instrument lines can be cooled and refilled.

j. Part A of Caution #1 allows the operator to determine if each indicated RPV water level range is reliable by being above the Minimum Indicated Level for each of a series of instrument run temperature ranges. Engineering calculations have determined that when indicated RPV water level is above the Minimum Indicated Level, the operator is assured that actual RPV water level is above the instrument variable leg tap, and trends are valid.

k. The Minimum Indicated Level is defined to be the highest RPV water level instrument indication which results from off-calibration instrument run temperature conditions when RPV water level is actually at the elevation of the instrument variable leg tap. Separate levels are provided for each RPV water level instrument.

l. The table in Part A is structured to give a Minimum Indicated Level corresponding to several temperature ranges for each of the RPV water level instrument ranges. This yields more usable instrument range than would be available if single values were used.

m. There are two items to note concerning this table: 1) Except for the Shutdown Floodup instrument, all drywell temperatures are not applicable, because there is very little vertical pipe run in the drywell. This means very little error can be caused by elevated drywell temperatures (until boiling occurs), and 2) The "MAX SC RUN TEMP" is the highest temperature reading which can be obtained from Table 6, Secondary Containment Instrument Runs.

Examination Outline Cross-reference:

295030EA2.03

Ability to determine and interpret the following as they apply to Low Suppression Pool Water Level: Reactor Pressure.

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	295030EA2.03	
Importance Rating	3.7	3.9

Proposed Question: **RO # 53**

Which ONE of the following describes the condition where Emergency RPV Depressurization is required based on Suppression Pool level and the reason for that requirement?

Suppression Pool water level below __(1)__ requires Emergency RPV Depressurization based on level below the ____________(2)________________.

	(1)	(2)
A.	12.75 feet	downcomer pipe exits.
B.	12.75 feet	HPCI Exhaust Line exit.
C.	11.50 feet	downcomer pipe exits.
D.	11.50 feet	HPCI Exhaust Line exit.

Proposed Answer: **C**

Explanation:

a. Part (1) is incorrect. This level (12.75 feet) corresponds to uncovering the HPCI exhaust line. Below this level, HPCI operation must be prevented because the exhaust line high pressure isolation setpoint is above the Primary Containment design pressure. Therefore, Part (2) is incorrect for this condition, but is correct for 11.50 feet.

b. Part (1) is incorrect as stated in (a) above. Part (2) is incorrect. However, this is the correct basis for 12.75 feet, but the stem asks for a condition requiring Emergency RPV depressurization. This condition does not warrant that action.

c. Correct answer.

d. Part (1) is correct. Below this level, the downcomer pipe exits are uncovered, which bypasses the suppression function of the Suppression Chamber in the event of a LOCA. Therefore, Emergency RPV Depressurization is required to place the RPV in the lowest energy state while sufficient Suppression Pool level is available to allow MSRV operation.

   Part (2) is incorrect as stated in (a) above.

Technical Reference(s): EOIPM Section 0-V-D page2 102 & 104 (Attach if not previously provided)

OPL171.203 pages 59-60

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/05/2008 RMS

Question History: Last NRC Exam

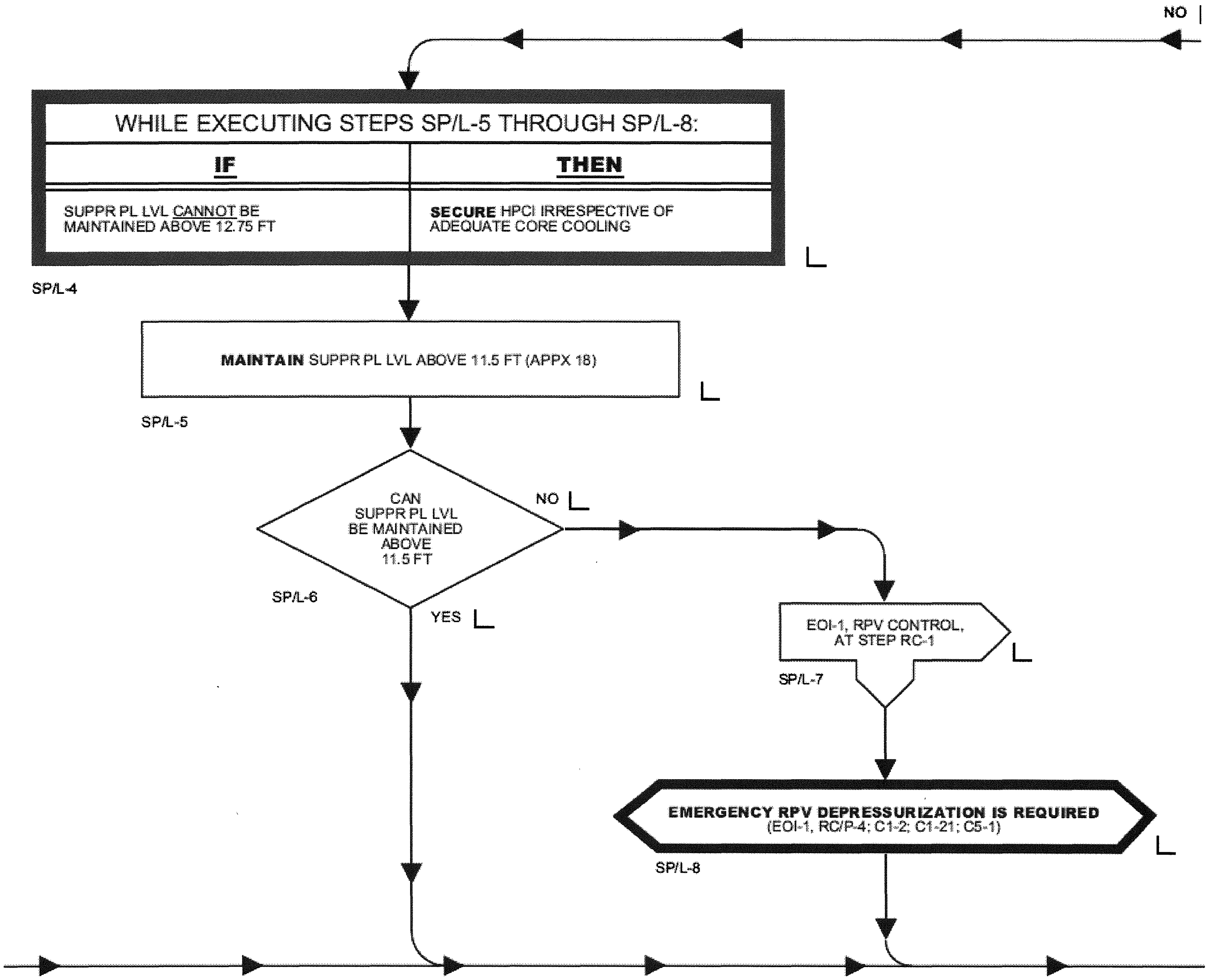
Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:


NO
WHILE EXECUTING STEPS SP/L-5 THROUGH SP/L-8:
IF
THEN
SUPPR PL LVL CANNOT BE MAINTAINED ABOVE 12.75 FT
SECURE HPCI IRRESPECTIVE OF ADEQUATE CORE COOLING
SP/L-4
MAINTAIN SUPPR PL LVL ABOVE 11.5 FT (APPX 18)
SP/L-5
CAN SUPPR PL LVL BE MAINTAINED ABOVE 11.5 FT
NO
YES
SP/L-6
EOI-1, RPV CONTROL, AT STEP RC-1
SP/L-7
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED (EOI-1, RC/P-4; C1-2; C1-21; C5-1)
SP/L-8

Excerpt from OPL171.203 page 49 of 66:

1. Step SP/L-4

   a. This retainment override step applies while performing Steps SP/L-5 through SP/L-8.

   b. The operator is directed to secure HPCI operation if suppression pool level cannot be maintained above the HPCI exhaust discharge device.

   c. Suppression pool level of 12.75 ft. corresponds to HPCI exhaust discharge device.

   d. Operation of the HPCI System with its exhaust discharge device not submerged will directly pressurize the suppression chamber. If this condition exists, HPCI operation is secured, irrespective of maintaining adequate core cooling, to prevent the failure of primary containment from overpressurization.

Excerpt from OPL171.203 page 50 of 66:

2. Step SP/L-5

   a. This action step directs the operator to maintain suppression pool level, using methods in EOI Appendix 18, above 11.5 ft.

   b. Calculations have determined that failure of containment, failure of equipment necessary for safe shutdown of the plant, and loss of pressure suppression function of containment, are prevented when suppression pool level is maintained above 11.5 ft.

   c. 11.5 ft. corresponds to the bottom of the downcomer openings

   d. If suppression pool level cannot be maintained above 11.5 ft., the operator continues at Step SP/L-7. The RPV is not permitted to remain at pressure if suppression of steam discharged from the RPV to the suppression pool cannot be assured.

Examination Outline Cross-reference:	Level	RO	SRO
295031EK3.01	Tier #	1	
Knowledge of the reasons for the following responses as they apply to Reactor Low Water Level: Automatic Depressurization System actuation.	Group #	1	
	K/A #	295031EK3.01	
	Importance Rating	3.9	4.2

Proposed Question: **RO # 54**

Given the following Unit 1 plant conditions:

- HPCI 120VAC POWER FAILURE (9-3F W7) is in alarm.
- A LOCA has occurred initiating a scram on Low Reactor Water Level.
- Reactor water level (-) 122 inches and lowering
- Drywell pressure 1.8 psig and steady
- A Pre-Accident Signal (PAS) has just been received and all ECCS equipment respond as designed.
- Assume NO operator actions.

Which ONE of the following describes the time that must elapse before ADS automatically initiates and the basis for this response?

ADS will initiate in ___(1)___. This actuation is in response to a ________(2)________.

	(1)	(2)
A.	265 seconds	LOCA inside the Drywell
B.	360 seconds	LOCA inside the Drywell
C.	265 seconds	LOCA outside the Drywell
D.	360 seconds	LOCA outside the Drywell

Proposed Answer: **D**

Explanation:

a. Part (1) is incorrect. This time delay is associated with -122 inches received without a high DW pressure (>2.45 psig), which is given in the stem. However, once this timer times out, if ECCS pumps are running, a 95 second timer initiates and must time out before ADS initiates. This makes the total time 360 seconds. Part (2) is incorrect. This is the basis for ADS initiation with BOTH high DW pressure AND low RPV level.

b. Part (1) is correct as stated in (a) above. Part (2) is incorrect as stated in (a) above.

c. Part (1) is correct as stated in (a) above. Part (2) is correct. ADS initiation in the absence of high DW pressure is due to decay heat boil-off following a LOCA outside the Drywell with MSIV isolation.

d. Correct answer.

Technical Reference(s): OPL171.043 pages 13 & 14 of 30 (Attach if not previously provided)

1-ARP-9-3F Window 7

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/06/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments: The addition of "Assume NO operator action" was added due to procedural guidance which would inhibit ADS initiation under this condition. In this condition, 1-EOI-1 flowchart path RC/L would allow ADS to be inhibited below -100 inches. In addition, 1-EOI-C1 would be entered below approximately -120 inches and direct that ADS be inhibited. In fact, there are no foreseeable circumstances where ADS would be allowed to auto initiate by procedure.

The HPCI 120VAC Power Failure annunciator is to provide realistic conditions where ADS would auto initiate. If HPCI were operable, ADS would not be required under these conditions.

Excerpt from OPL171.043 pages 13 & 14 of 30:

1. Automatic Depressurization Initiation Logic

a. The following conditions must be met before automatic depressurization will occur:

1) Two coincident signals of high drywell pressure (+2.45 psig) and low low low reactor vessel water level (-122")

OR
-122" for 265 sec.

2) A confirmatory low reactor vessel water level signal (+2") (Tech Spec Value 0")

3) Any one of the four RHR pumps or either A or B and either C or D Core Spray pumps running

NOTE:
This signal comes from pressure switches on the discharge of the pumps which give permissives in the logic above a set pressure of 100 psig for RHR pumps and 185 psig for the Core Spray pumps.

RHR	CS
PS-74-8A and 8B (Pump A)	PS-75-7 (Pump A)
PS-74-31A and 31B (Pump B)	PS-75-35 (Pump B)
PS-74-19A and 19B (Pump C)	PS-75-16 (Pump C)
PS-74-42A and 42B (Pump D)	PS-75-44 (Pump D)

4) A 95-second timer must be timed out

b. The high drywell pressure signal seals in immediately upon receipt of the signal

1) Must be manually reset after the signal has cleared

2) Indicative of a breach in the process system barrier inside the drywell

c. The reactor vessel low water level signals (-122" and +2") indicate that fuel is in danger of becoming overheated

1) The -122" water level signal would not normally occur unless the HPCI System had failed

2) These signals do not seal

3) The -122" water level initiation setpoint is selected to open the SRVs and depressurize the reactor vessel in time to allow fuel cooling by the Core Spray and LPCI Systems following a LOCA, in the event that the other makeup systems (Feedwater, CRD Hydraulic, RCIC, and HPCI) fail to maintain vessel level

4) The -122" setpoint will also initiate 265 second timers that seal in and will run even if water level is restored to >-122". The timers can be reset (if Rx. Level >-122") using pushbuttons in the auxiliary instrument room.

5) Once these timers have timed out, the drywell pressure contacts are bypassed, but other relays (that are not sealed in) must still sense reactor level <-122"

6) If so, and the other conditions are met (<+2" and low pressure pumps running), the 95 second timers will start.

7) This feature is based on a LOCA outside of the drywell which has been isolated. Level is below -122" and inventory is boiling off due to decay heat.

8) General Electric calculations have determined that the core will remain covered for 15 minutes after the -122" level is reached. Our system will initiated within the 15 minutes calculated by GE

BFN Unit 1	Panel 9-3 XA-55-3F	1-ARP-9-3F Rev. 0016 Page 10 of 39

HPCI 120VAC POWER FAILURE

7

(Page 1 of 1)

Sensor/Trip Point:

Relay 23A-K50 Loss of 120VAC from DIV II ECCS ATU inverter and loss of power to HPCI Flow Ind. Controller (1-FIC-73-33).

Sensor Location: 1-PNLA-009-0019
Auxiliary Instrument Room, El 593'

Probable Cause:
A. Failed fuse 1-FU2-073-0033C, Panel 1-9-82
B. DIV II ECCS ATU inverter failure
C. Loss of 250V DC power supply to DIV II ECCS ATU inverter. (250V Reactor MOV Bd 1A Compt. 11A1)

Automatic Action:
A. HPCI controller loses power. (HPCI is inoperable)
B. If HPCI is in service, HPCI TURBINE STOP VALVE, 1-FCV-73-18, closes and HPCI controller loses power. (HPCI becomes inoperable)
C. If fuse 1-FU2-073-0033C is cleared, 1-PI-064-0067B will lose power and become inoperable

Operator Action:
A. **DISPATCH** personnel to check the following:
- Inverter fuse 1-FU2-073-0033C, Panel 1-9-82 ☐
- DIV II ECCS ATU inverter ☐
- RMOV Bd 1A, Compt 11A1 ☐

B. **REFER TO** Tech Spec 3.3.5.1 and Table 3.3.5.1-1. ☐
C. **REFER TO** Tech Spec 3.3.1.1, Table 3.3.3.1-1, and TRM 3.3.5 for inoperable indicator 1-PI-064-0067B. ☐

References: 1-45E620-1-2 1-730E928-2 and -4 TRM 3.3.5
Technical Specifications 3.5.1, 3.5.2, 3.3.3.1

Examination Outline Cross-reference:	Level	RO	SRO
295037EA1.04	Tier #	1	
Ability to operate and/or monitor the following as they apply to SCRAM condition present and Power Above APRM Downscale or Unknown: Standby Liquid Control.	Group #	1	
	K/A #	295037EA1.04	
	Importance Rating	4.5	4.5

Proposed Question: **RO # 55**

An ATWS has occurred on Unit 1. The following conditions exist after the OATC has initiated Standby Liquid Control (SLC) injection using the 3A SLC pump:

• Pump Running Red Light	On
• Squib Valve Continuity Lights	Off
• SLC SQUIB VALVE CONTINUITY LOST (9-5 W20)	in alarm
• SLC Pressure	1200 psig
• Reactor Pressure	1000 psig
• RWCU is in service.	

Given these control board indications, Which ONE of the following is the appropriate action(s)?

Standby Liquid Control ____(1)____ injecting to the RPV. Perform the following:_____(2)_____.

	(1)	(2)
A.	is NOT	Initiate SLC pump 3B.
B.	is NOT	Fire the squid valves locally.
C.	IS	Manually isolate RWCU.
D.	IS	Fire the squid valves locally.

Proposed Answer: **C**

Explanation:

a. Part (1) is incorrect. Loss of continuity would indicate a SLC system failure BEFORE SLC is manually initiated. This is expected indication AFTER SLC is initiated. Part (2) is incorrect. The given conditions of SLC pressure above RPV pressure indicate that SLC Pump 3A is injecting.

b. Part (1) is incorrect as stated in (a) above. Part (2) is incorrect since the given indications are correct for SLC manual initiation. This would be a correct answer if discharge pressure was NOT above RPV pressure.

c. Correct answer. RWCU should have automatically isolated on SLC initiation. 1-EOI-Apendix 3A requires RWCU manually isolated if auto isolation does not occur.

d. Part (1) is correct. Part (2) is incorrect since the given indications are correct for SLC manual initiation. This would be a correct answer if discharge pressure was NOT above RPV pressure.

Technical Reference(s): 1-EOI-Appendix 3B, 1-ARP-95B W20 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # OPL171.201.1 (Attached)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Original question OPL171.201.1:

An ATWS has occurred and the following conditions exist after the Unit Operator has initiated SLC injection using the "A" pump:

* Red Light On
* Squib Continuity Lights Off
* Flow Light On
* Alarm "SLC Injection Flow to Reactor"
* Alarm "SLC Squib Valve Continuity Lost"
* SLC Pressure 1200 psig
* Reactor Pressure 1000 psig
* Tank Level 50%, lowering

Given these control board indications, DETERMINE which of the following is the appropriate action.

A. Start SLC Pump B and continue running SLC Pump A.

B. Initiate Alternate SLC Injection.

C. Stop SLC Pump A and start SLC Pump B.

D. Continue running SLC Pump A.

BFN Unit 1	Panel 9-5 1-XA-55-5B	1-ARP-9-5B Rev. 0016 Page 23 of 42

SLC
SQUIB VALVE
CONTINUITY LOST
1-EA-63-8
20

(Page 1 of 1)

Sensor/Trip Point:

1-XM-63-8A and 8B — fall 3.0 ma
rise 7.0 ma

Sensor Location: Behind Panel 1-9-5
Main Control Room

Probable Cause:
A. Cleared fuse.
B. Loss of power supply.
C. Blown Photohelic Bulb.
D. Sensor malfunction.
E. SLC pump start from the control room.

Automatic Action: None

Operator Action:
A. IF SLC has been initiated, **THEN**
REFER TO 1-EOI-1 or 1-AOI-79-2. ☐
B. IF SLC has **NOT** been initiated, **THEN**
PERFORM the following:
1. **CHECK** amber indicating lights on Panel 1-9-5 to determine which valve ignition circuit failed. ☐
2. **CHECK** sensor and amp meter in back of Panel 1-9-5. ☐
3. **DISPATCH** personnel to the SLC tank, RB El. 639', to investigate squib valve wiring connection. ☐

C. REFER TO Tech Spec 3.1.7. ☐

References: 1-47E610-63-1 1-45E620-6-2 1-729E854-1

BFN UNIT 1	SLC INJECTION	1-EOI APPENDIX-3A Rev. 0 Page 1 of 2

LOCATION:	Unit 1 Control Room
ATTACHMENTS:	None (✓)

1. **UNLOCK** and **PLACE** 1-HS-63-6A, SLC PUMP 1A/1B, 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 1-EA-63-8 Annunciator in alarm on Panel 1-9-5 (1-XA-55-5B, Window 20). _____
   - 1-PI-63-7A, SLC PUMP DISCH PRESS, indicates above RPV pressure. _____
   - System flow, as indicated by 1-IL-63-11, SLC FLOW, red light illuminated on Panel 1-9-5, _____
   - SLC INJECTION FLOW TO REACTOR 1-FA-63-11, Annunciator in alarm on Panel 1-9-5 (1-XA-55-5B, Window 14). _____

3. IF .... Proper system operation <u>CANNOT</u> be verified,

   THEN .... **RETURN** to Step 1 and **START** other SLC pump. _____

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

5. **VERIFY** ADS inhibited. _____

6. **MONITOR** reactor power for downward trend. _____

Examination Outline Cross-reference:	Level	RO	SRO
295038EK2.03	Tier #	1	
Knowledge of the interrelations between High Off-site Release Rate and the following: Plant Ventilation Systems.	Group #	1	
	K/A #	295038EK2.03	
	Importance Rating	3.6	3.8

Proposed Question: **RO # 56**

Given the following plant conditions:

- a Site Area Emergency has been declared due to gaseous effluent releases above 100 mrem TEDE.
- High radiation has been detected in the air inlet to the Unit 3 Control Room.
- Radiation Monitor RE-90-259B is reading 275 cpm above background.

Which ONE of the following describes the Control Room Emergency Ventilation (CREV) System response?

A. NEITHER CREV unit will automatically start at the current radiation level.

B. BOTH CREV units will automatically start with suction from the normal outside air path to Elevation 3C.

C. The Selected CREV unit will automatically start and will continue to run until Control Bay Ventilation is restarted; then, it will automatically stop.

D. The Selected CREV unit will automatically start. The Standby CREV unit will begin to auto-start; but, will ONLY run if the selected CREV unit fails to develop sufficient flow.

Proposed Answer: **D**

Explanation:

a. This is incorrect. This is plausible because the Tech Spec initiation setpoint is 270 cpm, which is less than the given radiation level. However, the actual CREV initiation setpoint is 221 cpm.

b. This is incorrect. This is plausible since both CREV units receive a start signal on a valid initiation. However, the CREV unit NOT selected will experience a 30 second time delay on initiation and will only complete its start sequence if the selected CREV unit fails to start.

c. This is incorrect. This is plausible because the start sequence is correct. However, once initiated, CREV must be manually secured. There is no automatic shutdown capability, only trips.

d. Correct answer.

Technical Reference(s): ______________________ (Attach if not previously provided)

____

Proposed references to be provided to applicants during examination: None

Question Source: Bank # RO 290003A3.01

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 3/25/2008

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 0	Control Bay and Off-Gas Treatment Building Air Conditioning System	0-OI-31 Rev. 0126 Page 21 of 283

3.6 **CREV and CREV instrumentation operability issues (continued)**

B. The main control room boundary may be opened intermittently under administration controls. For openings other than normal entry and exit, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the main control room and whose task is to close the opening when main control room isolation is indicated. With both CREVs inoperable in Modes 1, 2 or 3, for other than a control room boundary issue, enter LCO 3.0.3 Immediately. With two CREV subsystems inoperable during OPDRVs, initiate action to suspend OPDRVs. Reference TS 3.7.3.

C. When there is an automatic actuation of CREVS, the following automatic isolation dampers and hatch is required to be closed for CREVS to be considered operable.

1. 0-FCO-31-150B, 0-FCO-31-150D, 0-FCO-31-150E, 0-FCO-31-150F, 0-FCO-31-150G.

2. Removable equipment hatch in U-3 Mechanical Equipment Room, floor Elevation 617'.

D. The CREV system utilizes 15.45 kW Duct heaters to control moisture buildup in the charcoal adsorber. A malfunction of the 15.45 kW duct heater makes the applicable CREV unit inoperable. [Reference Functional Evaluation in PER 74959 and 75680]

E. One of the UNIT 1 & 2 Control Bay Supply Fans and one of the UNIT 3 Control Bay Supply Fans and their associated power and control circuits is required to be operable for CREVS instrumentation (control bay high radiation)to be considered operable. Reference Tech Spec 3.3.7.1.

F. CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214 may be placed in either the "A" or "B" position, depending on the operability status of the CREV trains. When a CREV train is inoperable, it will NOT be selected as lead. When both CREV trains are operable, the preferred position for CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214, is in TRAIN "A" which makes the "A" CREV the lead train. In the event that "A" CREV is INOP, CREV UNIT PRIMARY SELECTOR, 0-XSW-031-7214 is required to be placed in the TRAIN "B" position so the "B" CREV will initiate, without a time delay, as the lead train.

G. When one of the CREV trains is inoperable for testing, the CREV UNIT PRIMARY SELECTOR SWITCH, 0-XSW-031-7214 is required to be aligned to the train which is NOT under testing conditions to ensure the non-test train will initiate under an actual initiation signal.

Excerpt from OPL171.013, HVAC page 34:

a. Flow switches are provided, one for each division/unit, to start the standby unit if the selected unit does not start or trips off. The selected unit not starting is sensed by low differential pressure across the common HEPA filter in the Unit 2 vent tower. Low differential pressure exists when a fan is not operating; this signal will normally be present. The circuit for each unit is such that its initiation sequence is begun upon either of the following:

(1) Unit is selected as primary unit and CR1 relay for that division is energized.

(2) Other unit is selected as primary unit, low differential pressure exists across the common HEPA filter, and CR1 relay for that division has been energized for approx. 30 seconds.

b. With this circuit design, when an accident signal is initially received, the selected unit will enter its initiation sequence immediately and the other unit will enter its initiation sequence approx. 30 seconds later. Once the selected unit fan has been started (taking approx. 75 seconds -- 70 for the damper and 5 for the fan), the low differential pressure signal will no longer be present in the standby unit circuitry and its damper will return to the fail-close position.

c. If the selected unit fails to start properly, it will itself be turned off by the trips noted above, and the standby unit will continue in its initiation sequence. The time delay for startup of the standby unit will be selected to ensure that regardless of the primary unit failure, both fans will not be running at the same time.

d. If the selected unit starts properly, but then trips at a later time, the standby unit will only be missing the low differential pressure signal to receive its start signal. The standby unit will start when the selected system has completed its shutdown process and the fan has been de-energized.

Examination Outline Cross-reference:	Level	RO	SRO
600000AA1.01	Tier #	1	
Ability to operate and/or monitor the following as they apply to Plant Fire On Site: Respirator air pack.	Group #	1	
	K/A #	600000AA1.01	
	Importance Rating	3.0	2.9

Proposed Question: **RO # 57**

Concerning a fully pressurized (60 minute) Self-Contained Breathing Apparatus (SCBA), which ONE of the following describes the value of the protection factor and the significance of receiving an alarm while wearing a SCBA?

The protection factor provided by a SCBA is ______(1)______. An alarm sounding while wearing the above SCBA indicates that ________(2)________ remain(s) before all air is expired.

	(1)	(2)
A.	10,000	approximately 5 minutes
B.	10,000	approximately 10 minutes
C.	1,000	approximately 5 minutes
D.	1,000	approximately 10 minutes

Proposed Answer: **B**

Explanation:

a. Part (1) is correct. Part (2) is incorrect. Approximately 5 minutes of time remaining is applicable to a 30 minute tank. This could be correct if the tank was NOT fully pressurized before use.

b. Correct answer.

c. Part (1) is incorrect. This is lower by a factor of 10. Approximately 5 minutes of time remaining is applicable to a 30 minute tank. This could be correct if the tank was NOT fully pressurized before use.

d. Part (1) is incorrect as stated in (c) above. Part (2) is correct. A 60 minute tank will alarm at approximately 10 to 15 minutes.

Technical Reference(s):	SCBA 063.002	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/08/2008 RMS	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Excerpt from SCBA 063.002 page 7 of 21:

C. Capabilities and Limitations of the SCBA: Objective 5

1. Capabilities:

a. Offers highest assigned protection factor against gases, vapors and particulates (PF = 10,000).

b. An alarm will sound when air supply drops to ~1/4 of remaining pressure (10-15 minutes on 60 minute tanks and ~5 minutes on 30 minute tanks).

Work Expectation - Conservative Decision Making -Your response is to exit the area immediately even though you may think you have time to finish the job..

Examination Outline Cross-reference:	Level	RO	SRO
700000G2.4.34	Tier #	1	
Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects: Generator Voltage and Electric Grid Disturbances.	Group #	1	
	K/A #	700000G2.4.34	
	Importance Rating	4.2	4.1

Proposed Question: **RO # 58**

Given the following plant conditions:

- A significant voltage transient on the grid initiated a fault on Unit Station Service Transformer (USST) 2A.
- A fire on USST 2A resulted in actuation of fire suppression systems.
- Subsequently, all off-site power was lost due to continued voltage transients on the grid.

Which ONE of the following describes the required operator actions to restore Electric Fire Pump B to service and the location where these actions are performed?

Proceed to _____(1)_____ and perform the following: _____(2)_____.

	(1)	(2)
A.	4KV Shutdown Board B	Place the NORMAL/EMERGENCY switch for Fire Pump B to EMERGENCY and then back to NORMAL.
B.	4KV Shutdown Board B	Re-close the breaker for Fire Pump B.
C.	4KV Shutdown Board C	Place the NORMAL/EMERGENCY switch for Fire Pump B to EMERGENCY and then back to NORMAL.
D.	4KV Shutdown Board C	Re-close the breaker for Fire Pump B.

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Part (1) is correct. This appears easier than actual to determine, however the pumps assigned to 4KV S/D boards B & C are typically backwards. For instance, RHR pump B and Core Spray pump B are powered from 4KV S/D Board C and vice versa. Application of the same logic to the Fire Pumps is a common error. Part (2) is incorrect. Re-closing the breaker will not work. The breaker will re-open unless the closing coils are reset. This is accomplished by placing the NORM/EMER switch to EMER and back to NORM.

c. Part (1) is incorrect. Refer to the explanation in (b) above. Part (2) is correct. The closing coils must be reset by interrupting DC control power to the breaker using the NORM/EMER switch. Once accomplished, the Fire Pump will automatically re-start as long as the initiation signal is still present.

d. Part (1) is incorrect. Refer to the explanation in (b) above. Part (2) is incorrect. Re-closing the breaker will not work. The breaker will re-open unless the closing coils are reset. This is accomplished by placing the NORM/EMER switch to EMER and back to NORM.

Technical Reference(s):	0-AOI-57A	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/08/2008	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

BFN Unit 0	Loss of Offsite Power (161 and 500 KV)/Station Blackout	0-AOI-57-1A Rev. 0071 Page 18 of 71

4.2 Subsequent Actions (continued)

> **NOTE**
>
> [NER/C] If electrical-driven fire pumps were running due to an automatic initiation, the 52Y relay will lock-out the pump breaker; thus preventing the pump(s) from being restarted. In order to restart the fire pumps, Step 4.2[21] should be performed to momentarily interrupt DC control power to the closing coils of the fire pump breakers. [NRC IE 88-075]

[21] [NER/C] **IF** electrical-driven fire pumps were running due to an automatic initiation signal prior to loss of off-site power **AND** the automatic initiation signal is still present, **THEN**

PERFORM the following to restart fire pumps necessary for fire suppression: (Otherwise N/A)

[21.1] **PLACE** the NORMAL/EMERGENCY switch for Fire Pump A (B) (C) on 4kV Shutdown Bd. A (B) (C) compt. 11(11)(10) to EMERGENCY and back to NORMAL. ☐

[21.2] **VERIFY** that the fire pump(s) start. [NER IE 88-075] ☐

[22] **MONITOR** Batt Bd amps and VOLTS on Panels 9-8. **PLACE** the battery charger back in service within 30 minutes after loss of the charger to the battery. ☐

> **NOTE**
>
> Step 4.2[22.1] will ensure adequate voltage is available to operate switchyard breakers.

[22.1] **TRANSFER** breaker control power from NORMAL (Battery Board 4) to ALTERNATE (Battery Board 2) as follows (at Panel 9-24):

- **OPEN** BKR 642 NOR FDR FROM BATT BD 4 BKR 219. ☐
- **CLOSE** BKR 641 ALT FDR FROM BATT BD 2 BKR 501. ☐

BFN Unit 0	Loss of Offsite Power (161 and 500 KV)/Station Blackout	0-AOI-57-1A Rev. 0071 Page 71 of 71

Illustration 2
(Page 1 of 1)

Loss of Diesel Generator Cooling

1.0 LOSS OF DIESEL GENERATOR ENGINE COOLING

SBO Unit	DGs Lost on SBO	No available #3 EECW Pumps with the following DGs Lost
Unit 1	A and C	D, 3A and 3B
Unit 2	B and D	C, 3A and 3B
Unit 3	3A and 3C	C, D and 3B

2.0 LOSS OF DIESEL GENERATOR ROOM COOLING

SBO Unit	DGs Lost on SBO	No available Diesel Generator Room Cooling with the following DGs Lost
Unit 1	A and C	*D
Unit 2	B and D	*A
Unit 3	3A and 3C	*No Room Cooling

*Diesel Generator Room Cooling requires power to 1 DSL Aux Board on U1/U2 and U3

3.0 EECW PUMP DIESEL GENERATOR POWER SUPPLIES

DG	A	B	C	D	3A	3B	3C	3D
North EECW	A1	C1			A3	C3		
South EECW			B3	D3			B1	D1

4.0 EECW VALVE POWER SUPPLIES

0-FCV-067-0048	D1 X-TIE	DSL Aux Board B	4KV SD BD D (Nor)	4KV SD BD B (Alt)
0-FCV-067-0049	C1 X-TIE	DSL Aux Board A	4KV SD BD A (Nor)	4KV SD BD B (Alt)

Examination Outline Cross-reference:	Level	RO	SRO
295008AK3.04	Tier #	1	
Knowledge of the reasons for the following responses as they apply to High Reactor Water Level: **Reactor Feed Pump Trip**.	Group #	2	
	K/A #	295008AK3.04	
	Importance Rating	3.3	3.5

Proposed Question: **# RO 59**

Which ONE of the following describes the logic arrangement for the Reactor Feed Pump High Water Level Trip and the Technical Specification basis for that trip?

The Reactor Feed Pump (RFP) High Water Level Trip logic is _____(1)_____ and is designed to prevent damage to the ________(2)________.

	(1)	(2)
A.	one-out-of-two-twice	RFP turbines
B.	one-out-of-two-twice	Main Turbine
C.	two-out-of-two-once	RFP turbines
D.	two-out-of-two-once	Main Turbine

Proposed Answer: **D**

Explanation:

a. One-out-of-two-twice is the RPS logic arrangement which will initiate a scram after the Main Turbine trips due to high reactor Water Level. In addition, although tripping the RFP turbines will protect them from damage, the **basis** for the trip is to protect the Main Turbine from damage.

b. Logic is incorrect, Basis is correct. Protecting the Main Turbine is the basis for the High Reactor Water Level Trip.

c. Logic is correct. Basis is incorrect.

d. Correct answer

Technical Reference(s): U1 TSB 3.3.2.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New RMS 6/16/2008

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments: The logic arrangement is properly covered in the lesson plan but the basis for the trip is not. However, an objective does exist in the lesson plan which requires knowledge of the bases for RFP limitations contained in the Operating Procedure OI-3. High RPV Level trip is addressed within OI-3. I rated this question as MEM because no significant analysis should be required to answer the question correctly.

B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

BASES

BACKGROUND

The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow.

With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level reference point, causing the trip of the three feedwater pump turbines and the main turbine.

Reactor Vessel Water Level - High signals are provided by level sensors that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Two channels of Reactor Vessel Water Level - High instrumentation per trip system are provided as input to a two-out-of-two initiation logic that trips the three feedwater pump turbines and the main turbine. There are two trip systems, either of which will initiate a trip. The channels include electronic equipment, LS-3-208A, LS-3-208B, LS-3-208C, and LS-3-208D (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a main feedwater and turbine trip signal to the trip logic.

A trip of the feedwater pump turbines limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from damage due to water entering the turbine.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The reactor vessel high water level trip indirectly initiates a reactor scram from the main turbine trip (above 30% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR. Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).
LCO	The LCO requires two channels of the Reactor Vessel Water Level - High instrumentation per trip system to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Reactor Vessel Water Level - High signal. Both channels in either trip system are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.3. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The actual setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
295009AK2.02	Tier #	1	
Knowledge of the interrelations between Low Reactor Water Level and the following: Reactor water level control.	Group #	2	
	K/A #	295009AK2.02	
	Importance Rating	3.9	3.9

Proposed Question: **RO # 60**

Given the following Unit 1 plant conditions:

- A scram has occurred and all control rods did not fully insert.
- Reactor power 28%
- Reactor water level (-) 85 inches and stable
- Reactor Water Level Band (-) 55 to (-) 100 inches using HPCI
- Reactor Pressure 985 psig and rising slowly
- Reactor Pressure band 800 to 1000 psig using MSRVs

Which ONE of the following describes the Reactor Water Level response to opening a MSRV and the reason for that response?

Opening a MSRV will cause indicated level to be ____(1)____ due to ________(2)________.

	(1)	(2)
A.	lower	lower ΔP across the dryers and separators.
B.	lower	higher ΔP across the dryers and separators.
C.	higher	lower ΔP across the dryers and separators.
D.	higher	higher ΔP across the dryers and separators.

Proposed Answer: **D**

Explanation:

a. Part (1) is incorrect. Opening an MSRV causes a reduction in pressure outside the shroud where level is measured. This causes reference leg pressure to lower, measured Δp to lower, and indicated level to rise. Part (2) is incorrect. ΔP across the dryers and separators gets higher.

b. Part (1) is incorrect as stated in (a) above. Part (2) is correct. Pressure outside the core shroud will lower when the MSRV is opened, but that will cause indicated level to rise as stated in (a) above. ΔP across the dryers and separators gets higher.

c. Part (1) is correct. Part (2) is incorrect. ΔP across the dryers and separators gets higher.

d. Correct answer.

Technical Reference(s): OPL171.003 page 30 of 54 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 09/06/2005 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Excerpt from OPL171.003 page 29 and 30:

h. Steam flow effect on reactor water level

(1) Steam flowing through the dryers is forced to change direction several times, resulting in a pressure drop across the dryers.

(2) At 100 percent steam flow, the pressure drop is 7 inches of water. On a reactor scram, this ΔP "goes away", due to the void collapse, causing lower back pressure on the recirc pumps and jet pumps. Water from the annulus is relocated to the core area, causing sensed (and indicated) water level (in the annulus) to drop.

(3) Therefore, at 100 percent steam flow, P_1 is 7 inches of water less than P_2.

(4) The level outside the dryer skirt (down comer region) is 7 inches higher than inside the skirt.

(5) Since the vessel level instruments compare the reference column height to the down comer (variable column) height, setpoints are adjusted to compensate for this error.

NOTE: The principles discussed here apply to this question, but from a different perspective. The effect discussed in the lesson plan applies to an increase in RPV pressure where the question relates to a decrease in pressure as the MSRV is opened. The theory of operation is identical.

Examination Outline Cross-reference:	Level	RO	SRO
295012AK1.02	Tier #	1	
Knowledge of the operational implications of the following concepts as they apply to High Drywell Temperature: Reactor Power Level Control.	Group #	2	
	K/A #	295012AK1.02	
	Importance Rating	3.1	3.2

Proposed Question: **# RO 61**

Given a condition where a rising Drywell Average Air Temperature CANNOT be restored or maintained, which ONE of the following temperatures will require initiating a reactor scram and the bases for that required action?

Before Drywell Average Air Temperature exceeds _(1)_ ^{O}F, a manual reactor scram is required in accordance with EOI-2, "Primary Containment Control?"

A. 150

B. 160

C. 200

D. 280

Proposed Answer: **B**

Explanation:

a. Incorrect. 150 ^{O}F is the Tech Spec limitation requiring a normal shutdown, but not a manual scram.

b. Incorrect. 160 ^{O}F is the limit requiring entry into EOI-2.

c. Correct answer. 200 ^{O}F requires entry into EOI-1 which initiates a manual scram.

d. Incorrect. 280 ^{O}F is the Drywell Average Air Temperature limitation requiring Emergency Depressurization

Technical Reference(s):	ARP-9-3B Window 3 & 16	(Attach if not previously provided)
	EOI-2 Flowchart	
	Tech Spec bases 3.6.1.4	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	RMS 6/16/2008	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments: I rated this question as C/A because the candidate must analyze the temperature and apply that to differing procedure guidance to ascertain the correct action.

BFN Unit 1	Panel 1-9-3 1-XA-55-3B	1-ARP-9-3B Rev. 0020 Page 6 of 38

DRYWELL ATMOSPHERIC TEMP HIGH 1-TA-80-1 (3)	Sensor/Trip Point:	Alarms off 1-TR-80-1
	1-TE-080-0001 through -0012	148°F
	1-TE-080-0013, -0014	200°F
	1-TE-080-0015	185°F
	1-TE-080-0016 through -0025	148°F
	1-TE-080-0026	370°F
	1-TE-080-0027	370°F
	1-TE-080-0028	370°F
	1-TE-080-0029	265°F
	1-TE-080-0030 through -0032	148°F

(Page 1 of 1)

Sensor Location: Multiple locations in Drywell

Probable Cause:
A. Drywell Cooler(s) failure
B. Loss of cooling water (RBCCW) to Drywell Coolers
C. Sensor malfunction

Automatic Action: None

Operator Action:
A. **CHECK** Drywell temperature and pressure. ☐
B. **VERIFY OPEN** RBCCW PRI CTMT OUTLET VALVE, 1-HS-70-47A, Panel 1-9-4. ☐
C. **START** all available Drywell Coolers. ☐
D. REFER TO 1-AOI-64-1. ☐

References:

1-45E620-3-2	1-47E610-80-1	0-47W600-90
GE Drawing 1-730E933-1	1-47E605-173A	

BFN Unit 1	Panel 1-9-3 1-XA-55-3B	1-ARP-9-3B Rev. 0020 Page 19 of 38

DRYWELL TEMP HIGH TA-64-52 16	Sensor/Trip Point:
	1-TE-064-0052C ≥ 154°F (Alarm comes off recorder)

(Page 1 of 1)

Sensor Location: 1-TE-064-0052A
Rx Bldg (Drywell)
EL 584'
AZ 225°

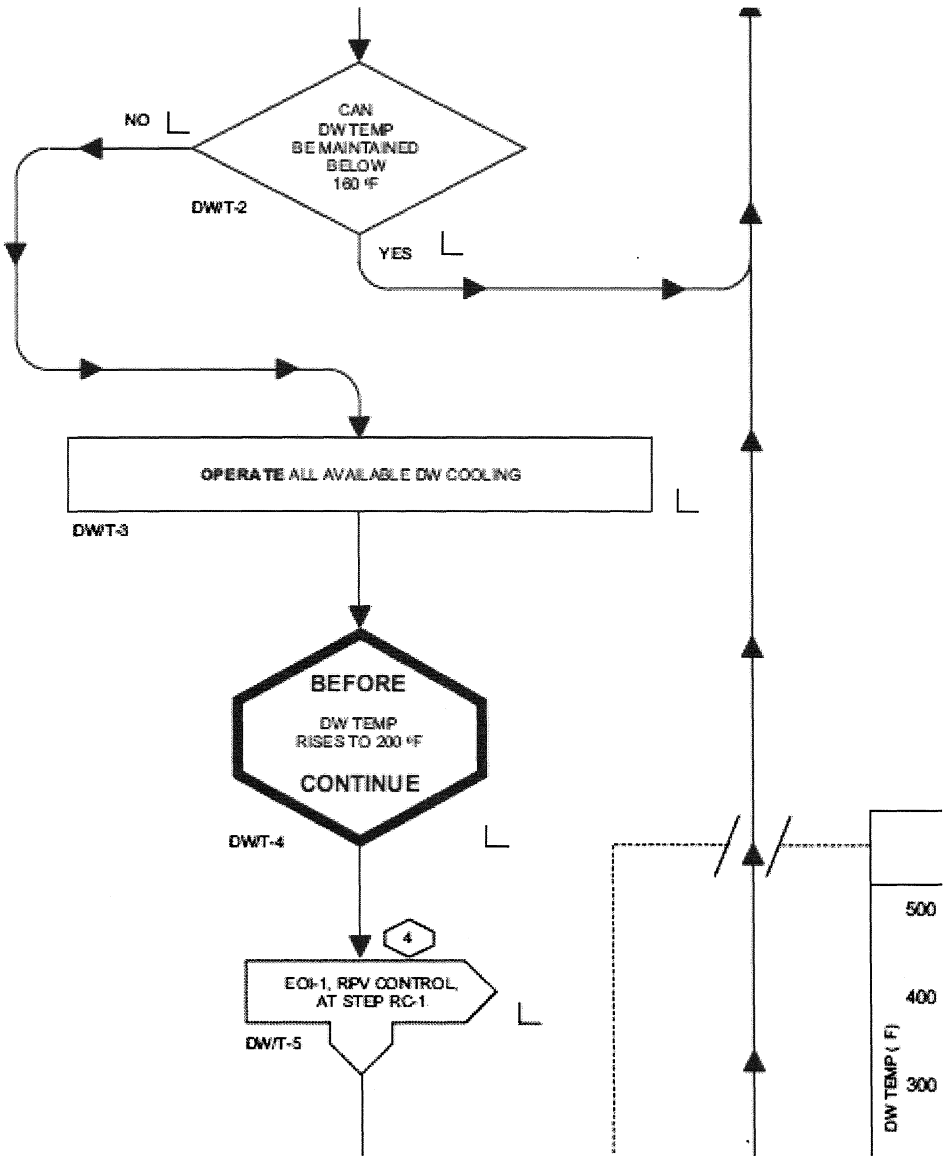
Probable Cause:
A. Drywell Cooler(s) failure
B. Loss of cooling water (RBCCW) to Drywell Coolers
C. Sensor malfunction

Automatic Action: None

Operator Action:
A. **CHECK** Drywell temperature and pressure using multiple indications. ☐
B. **VERIFY** Drywell Coolers running and **START** spare Drywell Cooler(s). ☐
C. **VERIFY OPEN** RBCCW PRI CTMT OUTLET VALVE, 1-HS-70-47A, 1-PLNA-009-0004. ☐
D. **START** additional RCW pumps. **REFER TO** 1-OI-24. ☐
E. **IF** high Drywell temperature continues, **THEN** **REFER TO** 1-AOI-64-1. ☐
F. **IF** Drywell temperature is due to a loss of RBCCW, **THEN**

REFER TO 1-AOI-70-1. ☐
G. **IF** temperature is above 160°F, **THEN** **ENTER** 1-EOI-2 Flowchart. ☐

References: 1-45E620-3-2 1-47E610-64-1 0-47W600-90


CAN DW TEMP BE MAINTAINED BELOW 160 °F
NO
YES
DWT-2
OPERATE ALL AVAILABLE DW COOLING
DWT-3
BEFORE
DW TEMP RISES TO 200 °F
CONTINUE
DWT-4
4
EOI-1, RPV CONTROL, AT STEP RC-1
DW/T-5
500
400
300
DW TEMP (F)

3.6 CONTAINMENT SYSTEMS

3.6.1.4 Drywell Air Temperature

LCO 3.6.1.4 Drywell average air temperature shall be ≤ 150°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell average air temperature not within limit.	A.1 Restore drywell average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

BASES

BACKGROUND	The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.
APPLICABLE SAFETY ANALYSES	Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Ref. 1). Analyses assume an initial average drywell air temperature of 150°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak drywell temperature does not exceed the maximum allowable temperature of 336°F (Ref. 2). Exceeding this temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident. Drywell air temperature satisfies Criterion 2 of the NRC Policy Statement (Ref. 3).

(continued)

Examination Outline Cross-reference:	Level	RO	SRO
295020AA1.01	Tier #	1	
Ability to operate and/or monitor the following as they apply to Inadvertent Containment Isolation: PCIS/NSSSS.	Group #	2	
	K/A #	295020AA1.01	
	Importance Rating	3.6	3.6

Proposed Question: **RO # 62**

Given the following Unit 2 plant conditions:

- An inadvertent Group 6 isolation occurs due to a bag of contaminated trash being brought too close to the Unit 2 Reactor Zone Ventilation Radiation Monitors.
- When the bag is removed, the NUMAC radiation monitor readings return to normal.

Which ONE of the following describes the MINIMUM operator actions required to reset the Group 6 isolation?

The NUMAC radiation monitors _______(1)______ reset. The PCIS isolation indication on Control Room Panel 9-4 _______(2)_______ reset.

	(1)	(2)
A.	must be manually	must be manually
B.	must be manually	will automatically
C.	will automatically	must be manually
D.	will automatically	will automatically

Proposed Answer: **C**

Explanation:

a. Part (1) is incorrect. NUMAC monitors are widely used in each unit. Some applications require manually resetting trips after the condition clears, while others reset automatically. In the case of Reactor Building High Radiation NUMAC monitors, the trip automatically resets. Part (2) is correct. The PCIS logic contacts re-close automatically, but the trip relay must be reset by manual action from Panel 9-4.

b. Part (1) is incorrect as stated in (a) above. Part (2) is incorrect. Only the relay contacts associated with High Radiation re-close automatically. The trip relay must be manually reset.

c. Correct answer.

d. Part (1) is correct as stated in (a) above. Part (2) is incorrect as stated in (a) and (b) above.

Technical Reference(s):	OPL171.148 pages 29 and 52	(Attach if not previously provided)
	OPL171.033 page 26	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #	OPL171.033.10	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Original question OPL171.033.10:

An inadvertent Group 6 isolation occurs due to a contaminated bag of trash being brought too close to the Unit 2 Reactor Zone Ventilation Radiation Monitors.
When the bag is moved, the monitors return to normal readings.

What is the MINIMUM operator actions to reset the Group 6 isolation?

A. Reset rad monitors at both NUMAC drawers.

B. Reset PCIS at Panel 9-4.

C. Reset rad monitors on the NUMAC drawer in alarm and reset PCIS at Panel 9-4.

D. Reset rad monitors at both NUMAC drawers and reset PCIS at Panel 9-4.

12. **LPRM alarms (Panel 9-5)** — V.B.2

a. LPRM Upscale and LPRM Downscale — Total scale = 0 to 125%

(1) The upscale and downscale set point markers are displayed inside the bargraphs and a status indication is displayed above the bargraphs. The solid box above the bargraph indicates that the set point marker is presently exceeded while a hollow box indicates a past condition.

V.B.4
↗ Vs. →
Above each LPRM bargraph

Downscale is less than or equal to 3% of scale AND upscale is greater than or equal to 100 % of scale.

(2) A past condition may be reset by entering the TRIP STATUS display and pressing the RESET MEMORY softkey.

(3) Trip memory indicates trip conditions that have occurred in the past but are no longer in the "tripped" condition. The trip memory is cleared on either of two occasions: — Obj V.D.5

(a) Manual reset
OR
(b) The instrument condition changes from INOP to OPER.

(4) The alarm status of each LPRM Detector is transmitted to the RBM instruments.

(5) The alarm status of each LPRM Detector is indicated on the APRM and LPRM instrument front panel display and on its associated operator's display (ODA).

o. **APRM Instrument (Panel 9-14)**

(1) The controls and indications are similar to those on the other NUMAC instruments that already exist in the control room.

Located on Pnl. 9-14
TP-8
Attention to Detail

(2) The status of the upper display will either be in normal video, blank, or inverse video.

(3) Inverse video indicates abnormal conditions requiring attention i.e. RPS trip, rod block active, bypass condition, faults, and when in the INOP mode.

(4) Area of display will either be blank or normal video when condition is cleared.

(5) When normal video is displayed, the operator must reset the memory for that NUMAC instrument to clear the normal video display.

So, for instance, if a rod block is displayed for APRM channel 1 in inverse video, then cleared, both the normal video on the ODA AND on the APRM instrument must be reset to clear the normal video display.

(6) Three bargraphs are presented on the APRM chassis as a default display for the OPER and INOP-CAL mode: APRM Flux, STP (Simulated Thermal Power) and Flow with the numerical values (to the right of each graph) in "percent of rated".

Each bargraph contains the alarm and trip set point markers (up or down arrows) for the various limits that are being monitored by the instrument.

▲ or ▼

Excerpt from OPL171.033 page 26:

a. Trips

(1) Reactor zone and refueling zone monitors work independently of each other for trip actuation

(2) High radiation trip setpoint is 72 mr/hr for the refueling and reactor zones

Rad-monitor auto resets when alarm is clear

2-EOI APPENDIX-8F

RESTORING REFUEL ZONE AND REACTOR ZONE VENTILATION FANS FOLLOWING GROUP 6 ISOLATION

LOCATION: Unit 2 Control Room

ATTACHMENTS: None (√)

1. **VERIFY** PCIS Reset. ____

2. **PLACE** Refuel Zone Ventilation in service as follows (Panel 2-9-25):

   a. **VERIFY** 2-HS-64-3A, REFUEL ZONE FANS AND DAMPERS, control switch is in OFF. ____

   NOTE: When Refuel Zone supply and exhaust fans start, Refuel Zone supply and exhaust dampers open automatically.

   b. **PLACE** 2-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 2-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:

      - 2-FCO-64-5, REFUEL ZONE SPLY OUTBD ISOL DMPR ____
      - 2-FCO-64-6, REFUEL ZONE SPLY INBD ISOL DMPR ____
      - 2-FCO-64-9, REFUEL ZONE EXH OUTBD ISOL DMPR ____
      - 2-FCO-64-10, REFUEL ZONE EXH INBD ISOL DMPR. ____

Examination Outline Cross-reference:	Level	RO	SRO
295034EA1.03	Tier #	1	
Ability to operate and/or monitor the following as they apply to Secondary Containment Ventilation High Radiation: Secondary containment ventilation.	Group #	2	
	K/A #	295034EA1.03	
	Importance Rating	4.0	3.9

Proposed Question: **RO # 63**

Given the following Unit 2 conditions:

- RX ZONE EXH RADIATION MONITOR DNSC (9-3A W35) is in alarm
- Reactor Zone Radiation detector 2-RE-90-142A has failed to a DOWNSCALE condition.

Which ONE of the following subsequent instrumentation failures will cause a Reactor Zone Isolation?

Radiation monitor ______(1)______ will initiate a Reactor Zone isolation if it fails ______(2)______.

	(1)	(2)
A.	2-RE-90-142B	upscale
B.	2-RE-90-143B	downscale
C.	2-RE-90-143C	upscale
D.	2-RE-90-142C	downscale

Proposed Answer: **B**

Explanation:

a. Part (1) is correct if the monitor fails downscale, but Part (2) makes this choice incorrect.

b. Correct answer. The logic is 2 of 2 taken once. This requires either two upscale trips in either channel or one downscale/INOP trip in both channels. The downscale/INOP condition initiates a trip due to preventing an upscale trip from occurring.

c. Part (1) is incorrect. This monitor is in the same division as 142A which requires both monitors upscale to initiate a trip.

d. Part (1) is incorrect. The downscale/INOP trip logic is 1 of 2 taken twice. Both channels downscale/INOP in the same division will NOT initiate a trip.

Technical Reference(s): OPL171.033, Process Radiation Monitoring (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # OPL171.033.21 Attached

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Original question OPL171.033.21:

One Reactor Zone Radiation detector 2-RE-90-142A (normally powered from RPS "A") has failed to a DOWNSCALE condition.

Which subsequent instrumentation failure will cause a Reactor Zone Isolation and a Primary Containment Group 6 Isolation?

A. 2-RE-90-142B (Reactor Zone powered from RPS A) upscale

B. 2-RE-90-143B (Reactor Zone powered from RPS B) downscale

C. 2-RE-90-143B (Reactor Zone powered from RPS B) upscale

D. 2-RE-90-142B (Reactor Zone powered from RPS A) downscale

Excerpt from OPL171.033 pages 26 & 27 of 78:

a. Trips

(1) Reactor zone and refueling zone monitors work independently of each other for trip actuation

(2) High radiation trip setpoint is 72 mr/hr for the refueling and reactor zones

(3) Trip logic for the refueling and the reactor zones is identical, and the following combinations will generate a trip

(a) Two high level trips in the <u>same</u> channel, (division)

-OR-

(b) One downscale trip in <u>each</u> channel (division)

-OR-

(c) One monitor INOP in <u>each</u> channel (division)

-OR-

(4) Loss of RPS power to <u>either</u> channel

(5) Automatic actions

(a) Refuel Zone Trip

(i) Isolate Refuel Zone on all 3 units

(ii) Start SGT and opens SGT suction to refuel zone

(iii) Group 6 PCIS

(iv) Start CREVs

(v) Isolate fresh air paths to Control Bay Elev 3C

(b) Reactor Zone – Same as Refuel Zone, plus:

(i) Isolate affected unit reactor zone

(ii) Open SGT suction to affected unit reactor zone

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RX ZONE EXH RADIATION MONITOR DNSC 3-RA-90-142B	
	35

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Sensor/Trip Point:

3-RE-90-142A	.20 MR/HR
3-RE-90-142B	.20 MR/HR
3-RE-90-143B	.20 MR/HR
3-RE-90-143B	.20 MR/HR

Sensor Location: Rx Bldg, El 664' (Refuel Floor), R-18 P-LINE
Panel 9-42

Probable Cause:
A. Sensor malfunction.
B. Loss of power to detector.

Automatic Action:
A. One downscale trip: None
B. [NRC/C] Two out of two taken once upscale trips or one out of two taken twice downscale trips or taking both OPER/INOP switches out of OPERATE will cause the following to occur:
1. Reactor zone and refuel zone isolate.
2. SGTS initiates.
3. Control Room Emergency Pressurization units start.
4. H_2O_2 analyzers isolate and pumps trip.
5. Drywell CAM, 3-RM-90-256, isolates and pump trips.
6. Drywell and Supp Chbr purge and exhaust valves close. [NRC NCO 870331002, LER 87-026]

NOTE
Trips on the Reactor Zone Radiation monitors will automatically reset when the alarming condition resets.

Operator Action:
A. **VERIFY** alarm condition on the following:
1. REACTOR ZONE EXHAUST RADIATION recorder, 3-RR-90-140 on Panel 3-9-2. ☐
2. RX & REFUEL ZONE EXH CH A RAD MON RTMR, 3-RM-90-140/142 on Panel 3-9-10. ☐
3. RX & REFUEL ZONE EXH CH B RAD MON RTMR, 3-RM-90-141/143 on Panel 3-9-10. ☐

Continued on Next Page

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RX ZONE EXH RADIATION MONITOR DNSC 3-RA-90-142B, Window 35
(Page 2 of 2)

Operator
Action: (Continued)

B. CHECK power supply to the radiation monitors. ☐
C. CHECK other radiation monitors for radiation levels below release limits. ☐
D. REFER TO Technical Specification Section 3.3.6.2. and 3.3.7.1. ☐
E. IF Group 6 isolation occurs, THEN REFER TO 3-AOI-64-2d. ☐

References: 3-47E620-3 3-47E610-90-1 GE 3-730E927-21
Technical Specifications 3.3.6.2 and 3.3.7.1

Examination Outline Cross-reference:	Level	RO	SRO
295035EA2.02	Tier #	1	
Ability to determine and interpret the following as they apply to Secondary Containment High Differential Pressure: Off-site release rate.	Group #	2	
	K/A #	295035EA2.02	
	Importance Rating	2.8	4.1

Proposed Question: **# RO 64**

Given the following plant conditions:

- Unit 2 is at 100% power.
- During the backwash of a Reactor Water Cleanup (RWCU) Demineralizer, the Backwash Receiving Tank ruptured.
- The RWCU system has been isolated.
- Secondary Containment conditions are as follows:
  - ALL Reactor and Refuel Zone radiation monitors trip on high radiation.
  - NO Standby Gas Treatment (SGT) train can be started.

Refuel zone pressure:	(-) 0.12 inches of water
Reactor zone pressure:	(+) 0.02 inches of water

- AREA RADIATION LEVELS

RB EL 565 W, 565 E, 565 NE:	250 mr/hr
RB EL 593	upscale
RB EL 621	upscale

Which ONE of the following describes the required action and the type of radioactive release in progress?

REFERENCE PROVIDED

A. Initiate a shutdown per 2-GOI-100-12A, "Unit Shutdown." An Elevated radiation release is in progress.

B. Initiate a shutdown per 2-GOI-100-12A, "Unit Shutdown." A Ground-level radiation release is in progress.

C. Scram the reactor and Emergency Depressurize the RPV. An Elevated radiation release is in progress.

D. Scram the reactor and Emergency Depressurize the RPV. A Ground-level radiation release is in progress.

Proposed Answer: **B**

Explanation:

a. The release from the Reactor Building is NOT elevated. This is plausible because the required actions are correct except the differential pressure results in a ground-level release.

b. correct answer.

c. Conditions DO NOT warrant a scram at this point. In addition, the release from the Reactor Building is NOT elevated. This is plausible if the candidate fails to recognize that a primary system is NOT discharging to the Reactor Building.

d. Conditions DO NOT warrant a scram at this point. This is plausible if the candidate fails to recognize that a primary system is NOT discharging to the Reactor Building.

Technical Reference(s): 2-EOI-3 Flowchart (Attach if not previously provided)

Proposed references to be provided to applicants during examination: 2-EOI-3 Flowchart

Question Source: Bank # 295035EA2.02

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam X

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments: In order to answer this question correctly the candidate must determine the following:

1. Which area(s) are above or approaching Max Safe.
2. Based on Item #1 above, determine the appropriate action.
3. Whether plant conditions indicate an elevated or ground-level release.

NOTE: EOI-3 steps SC/R-8 and SC/R-9 apply, requiring shutdown per 2-GOI-100-12A because two (2) or more areas are above max safe rad levels; but, a primary system is NOT discharging to the Reactor Building. Insufficient Reactor Building-to-atmosphere dp (greater than -0.25 inches of water) indicates loss of secondary containment integrity. The positive Reactor Zone pressure is causing an unmonitored and uncontrolled ground-level release of radioactive contaminants.

BROWNS FERRY	EMERGENCY CLASSIFICATION PROCEDURE	EPIP-1

TERM/PHRASE	MEANING/DEFINITION
Projectile	An object ejected, thrown, or launched towards a plant structure. The source of a projectile may be offsite or onsite. Damage is sufficient to cause concern regarding the integrity of the affected structure or the operability or reliability of safety equipment contained therein.
Protected Area	All areas within the security protected area fence.
PSIG	Pounds Per Square Inch Gauge
R	Rad
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
REP	Radiological Emergency Plan
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
Sabotage	Deliberate damage, misalignment, misoperation of plant equipment with the intent to render equipment inoperable.
SAMG	Severe Accident Management Guideline
SEC	Second
Secondary Containment	The spaces immediately adjacent to or surrounding, the primary containment from which the Reactor Building Ventilation System and the Standby Gas Treatment System provides a filtered elevated release.
SED	Site Emergency Director
SGTS	Standby Gas Treatment System
Significant Transient	An unplanned event involving one or more of the following: (1) Automatic turbine runback greater than 25% thermal reactor power or (2) Electrical load reduction greater than 25% full electrical load, or (3) <u>Thermal</u> power oscillations greater than 10%, or (4) Reactor scram, or (5) Valid ECCS initiation.

OPL171.067, HVAC Lesson Plan

V. Training Objectives:

A. Terminal Objective

Upon completion of this lesson, the operator will demonstrate satisfactory knowledge of Plant Ventilation, Heating and Air Conditioning (HVAC) Systems by scoring at least 80% (70% NLO) on a written exam.

B. Enabling Objectives (HLT/LOR)

1. Identify the purposes of the HVAC systems and how these purposes are accomplished.

2. Identify the relationships between HVAC systems and the following, evaluate how loss or malfunction of the following effects HVAC, and evaluate how loss or malfunction of HVAC effects the following:

a) AC Electrical Distribution
b) Secondary containment
c) Standby Gas Treatment (SGT)
d) EECW
e) Process Radiation Monitoring
f) Control Air
g) Process instrumentation (drywell pressure, reactor level)
h) Static pressure control
i) Fire Protection
j) Control Room Habitability
k) Area Temperatures (Reactor Bldg and Control Bay)

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a. Reactor zone and refueling zone static pressure is maintained at .25 inch water negative by use of larger exhaust fans and static pressure regulation. Dampers mounted in each fan inlet are designed to gradually close or open in response to static pressure regulators to maintain building pressure and ensure an elevated, monitored release.

Obj. V.B.2
Obj. V.C.2

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2. Normal ventilation flow path, ΔP control, and isolation capability ensure secondary containment integrity during normal operation. Loss of the normal HVAC system under isolation (PCIS) conditions does not affect secondary containment integrity if SGT auto starts and has a suction path to maintain pressure at -0.25 inches H_2O inside the reactor building. This ensures leakage is into the building, and releases are elevated (stack) and monitored. However if secondary containment integrity is lost, leakage may be at ground levels, and site boundary doses may be higher than calculated in the accident analysis.

Obj. V.B.2/ V.B.5
Obj. V.C.8/V.B.4
Obj.V.C.19
T.S. 3.6.4.1, 3.6.4.2

<u>EFFECTIVE COMMUNICATION</u>

Examination Outline Cross-reference:

295036G2.4.34

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects: Secondary Containment High Sump/Area Water Level.

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295036G2.4.34	
Importance Rating	4.2	4.1

Proposed Question: **RO # 65**

Which ONE of the following EOI-3, "Secondary Containment Control" entry conditions CANNOT be determined using control room indications and the appropriate operator action if it has been exceeded?

The entry condition is ______(1)______. Operate ALL available floor drain and equipment drain sump pumps from the ______(2)______.

	(1)	(2)
A.	SECONDARY CNTMT FLOOR DRAIN SUMP WATER LVL ABOVE 66 IN.	main control room panel 9-4.
B.	SECONDARY CNTMT FLOOR DRAIN SUMP WATER LVL ABOVE 66 IN.	radwaste control room.
C.	ANY SECONDARY CNTMT AREA WATER LVL ABOVE 2 IN.	main control room panel 9-4.
D.	ANY SECONDARY CNTMT AREA WATER LVL ABOVE 2 IN.	radwaste control room.

Proposed Answer: **B**

Explanation:

a. Part (1) is correct. This indication is only available from the radwaste control room or locally at the sump. Part (2) is incorrect. The floor drain and equipment drain sump pumps operated from panel 9-4 are for the Drywell, not the Reactor Building.

b. Correct answer.

c. Part (1) is incorrect. This indication is available from high level annunciators for each identified area in the reactor building. If the annunciator is in alarm, the level in that area is above 2 inches. Part (2) is incorrect. Rx BLDG floor and equipment drain sump pumps are operated from the radwaste control room.

d. Part (1) is incorrect as stated in (c) above. part (2) is correct. Rx BLDG floor and equipment drain sump pumps are operated from the radwaste control room.

Technical Reference(s):	2-EOI-3 flowchart	(Attach if not previously provided)
	1-ARP-9-4C (attached annunciator)**	

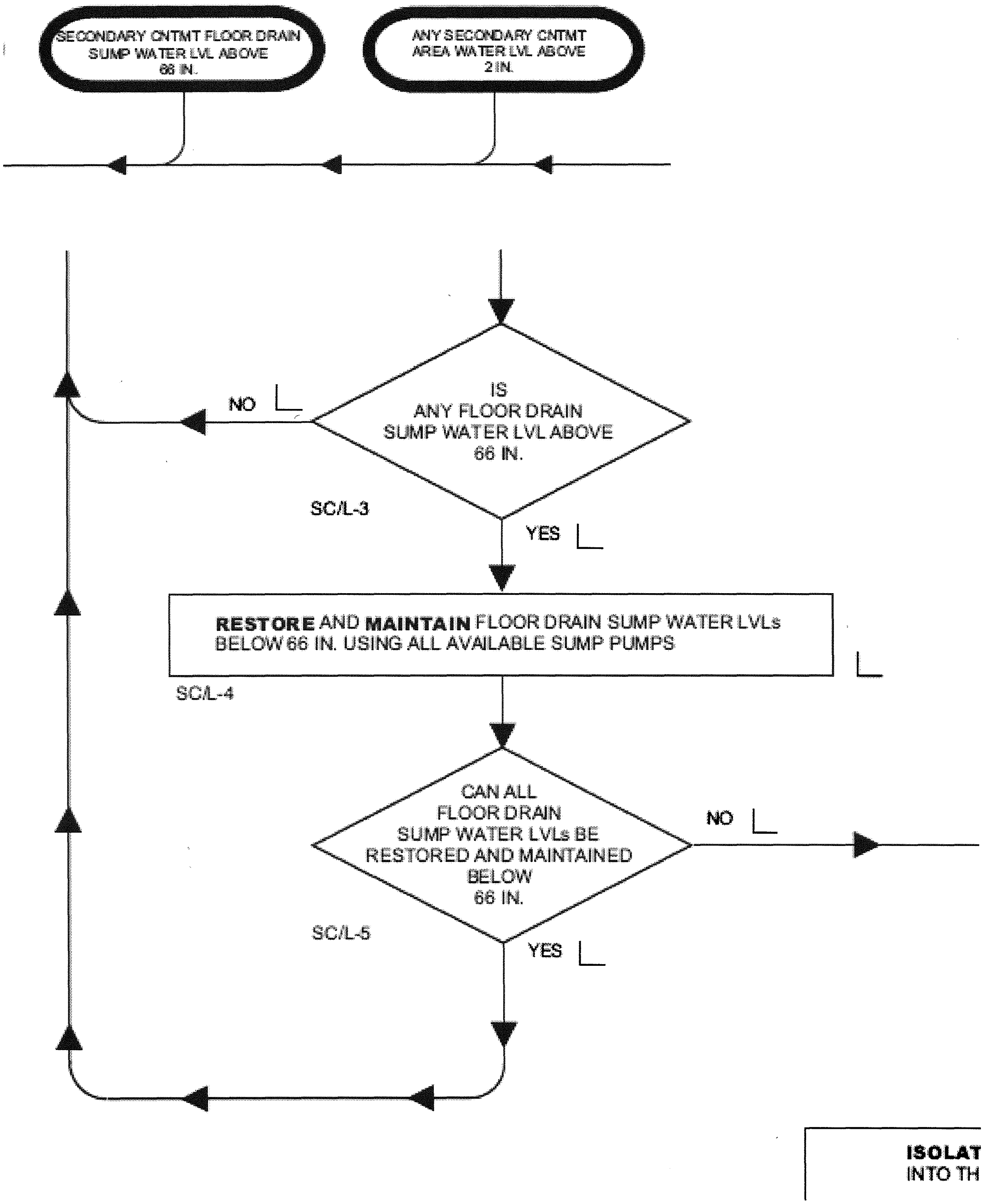
Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/07/2008 RMS	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments: ** Other annunciators indicating >2 inches are also available in the control room. This is only one example.


SECONDARY CNTMT FLOOR DRAIN SUMP WATER LVL ABOVE 66 IN.
ANY SECONDARY CNTMT AREA WATER LVL ABOVE 2 IN.
IS ANY FLOOR DRAIN SUMP WATER LVL ABOVE 66 IN.
NO
YES
SC/L-3
RESTORE AND MAINTAIN FLOOR DRAIN SUMP WATER LVLs BELOW 66 IN. USING ALL AVAILABLE SUMP PUMPS
SC/L-4
CAN ALL FLOOR DRAIN SUMP WATER LVLs BE RESTORED AND MAINTAINED BELOW 66 IN.
NO
YES
SC/L-5
ISOLAT
INTO TH

BFN Unit 1	Panel 9-4 1-XA-55-4C	1-ARP-9-4C Rev. 0016 Page 7 of 43

SUPPR CHMBR RM FLOOD LEVEL HIGH 1-LA-77-25F 3	Sensor/Trip Point: 1-LS-77-25F $\geq$ 2 Inches of Water on the Floor

(Page 1 of 1)

Sensor Location: Sensor is located near the floor of the Suppression Chamber room, Column R-4 N-Line.

Probable Cause: Greater than two inches of water on the floor.

Automatic Action: None

Operator Action:

A. **DISPATCH** personnel to visually check the suppression chamber room. ☐

B. **IF** alarm is valid, **THEN** **PERFORM** the following:
- **CHECK** the floor drain sump pumps running. ☐
- **CHECK** the floor drains for proper drainage. ☐
- **IF** possible, **THEN** **DETERMINE** the source of the leak and the leak rate. ☐
- **ENTER** 1-EOI-3 Flowchart. ☐

NOTE
The floor drain and equipment drain sump pumps may need to be locked out to prevent Radwaste flooding.

- **NOTIFY** Radwaste operator to monitor drain collector tank and waste collector tank levels. ☐
- **NOTIFY** Radiation Protection. ☐

References: 0-47E610-77-1 0-47E600-8
FSAR Sections 13.6.2 and F.7.15

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.2 Conduct of Operations	Tier #	3	
Knowledge of operator responsibilities during all modes of plant operation.	Group #		
	K/A #	G2.1.2	
	Importance Rating	3.0	4.0

Proposed Question: **RO # 66**

Given the following conditions involving Foreign Material Exclusion (FME):

- An outage worker was placing a plastic FME cover on a vacuum breaker inside the Torus.
- He inadvertently dropped the cover into the Suppression Pool.
- The cover immediately sank to the bottom of the Torus.
- The cover was still visible from the catwalk.

Which ONE of the following describes the required actions for this situation?

Work in the Torus ____(1)____. The FME cover MUST be retrieved ____(2)____.

	(1)	(2)
A.	MUST be stopped.	before job closeout.
B.	MUST be stopped.	immediately.
C.	may continue.	before job closeout.
D.	may continue.	immediately.

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Part (1) is correct. The job supervisor may eventually allow work to continue since the cover is visible, but all work is stopped until a retrieval plan has been finalized. Part (2) is incorrect. Torus work requires an underwater inspection by divers for closeout requirements. The cover can be retrieved then if it doesn't interfere with other work being done.

c. Part (1) is incorrect. The job supervisor may eventually allow work to continue since the cover is visible, but all work is stopped until a retrieval plan has been finalized. Part (2) is correct. Torus work requires an underwater inspection by divers for closeout requirements. The cover can be retrieved then if it doesn't interfere with other work being done.

d. Part (1) is incorrect. The job supervisor may eventually allow work to continue since the cover is visible, but all work is stopped until a retrieval plan has been finalized. Part (2) is incorrect. Torus work requires an underwater inspection by divers for closeout requirements. The cover can be retrieved then if it doesn't interfere with other work being done.

Technical Reference(s): SPP-6.5, Foreign Material Control (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # OPL171.217.10 minor format changes

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments: The requirement to stop work if FME control is lost has only recently been changed. Previously, the Job Supervisor made the determination of whether to stop work or not based on the specific situation. New restrictions, based on continuing problems with FME control, have led to more restrictive requirements. Notice that the original question had distracters related to initiating a Problem Evaluation Report (PER). Only under certain conditions was a PER required. Now a PER is written for any FME issue. For that reason, I removed that section of the question because this fact has been widely taught to the entire plant population and would no longer be discriminatory.

Original Question OPL717.217.10:

An outage worker was placing a plastic FME cover on a vacuum breaker inside the Torus when he inadvertently dropped the cover into the Suppression Pool. The cover immediately sank to the bottom of the Torus, but was still visible from the catwalk. Select the actions required for this situation.

A. Work in the Torus must be stopped to immediately retrieve the FME cover. Initiating a PER is required.

B. Work in the Torus must be stopped to immediately retrieve the FME cover. Initiating a PER is NOT required.

C. Work in the Torus may continue. Initiating a PER is required. Retrieve the FME cover before job closeout.

D. Work in the Torus may continue. Initiating a PER is NOT required. Retrieve the FME cover before job closeout.

TVAN Standard Programs and Processes	Foreign Material Control	SPP-6.5 Rev. 0012 Page 14 of 51

3.3 Work Performance (continued)

CAUTION

Failure to remove maintenance residue from radioactive components will result in production of radiation source term hazards to personnel.

13. While performing grinding or cutting activities, precautions shall be made to reduce the amount of debris produced. It may be necessary to erect shielding to prevent debris from maintenance activities from becoming foreign material in the system in question or another close by. Setting up a vacuum during activities can be a viable method for reducing debris.

14. Immediately following system breaches, the system or component shall be cleaned of foreign debris. Vacuuming is the preferred method of cleaning. Cleaning shall be done before removal of pipe dams, plugs, or barriers, and should be done again after the devices have been removed.

15. Stop all work if FME control is lost and investigate in accordance with Section 3.5.

16. For small items (screws, nuts, washers, etc.) that cannot be made fail-safe, consider the use of re-sealable fail-safe containers, double bagging or other methods to lessen the probability of a loss of FME control event.

17. Temporary Tie Wraps should be bright in color, non Metallic (not containing metallic locking tabs) and if at all possible, float in order to facilitate recovery.

18. Use of wire brushes on systems that come in contact with the Reactor Coolant System is prohibited.

NOTE

For Work involving electrical and I & C components or systems further guidance is provided in Appendix A.

J. Ensure any parts/particles (particularly valve stellite hard-facing) are cleaned from the system before closure.

3.4 **Suspension of the Job**

A. Often plant conditions change and work within a FMEA is stopped for a period of time. Anytime continuous or immediate access through opening is not required or if work is stopped during a shift, between shifts, or for a period of time the following actions are required.

TVAN Standard Programs and Processes	Foreign Material Control	SPP-6.5 Rev. 0012 Page 16 of 51

3.4 Suspension of the Job (continued)

The following materials are unacceptable for use as temporary plugs, covers or seals: paper products or rags inserted into pipes or openings.

3.5 Recovering from Loss of FME Controls

A. For all loss of FME control events, a PER shall be initiated, and the following notifications shall be made:

1. Notify the FME monitor.

2. Notify the Work Supervisor.

3. For events associated with systems that connect with the Reactor Coolant, or Spent Fuel Cooling Systems, notify Shift Operations or the Refuel Floor Senior Reactor Operator (RFF SRO).

4. Notify the Department Manager over the group performing the task.

5. During outages, notify the Outage Control Center.

B. If the foreign material can be easily retrieved (i.e., without further disassembly of the system or component), Stop, Review your actions, then the retrieval may be performed with slow deliberate moves to prevent lodging the foreign material deeper into the system or creating additional debris.

C. If the foreign material can not be easily retrieved, Stop, Notify the Work Supervisor and FME monitor to coordinate development of a retrieval plan.

D. If the missing item can not be accounted for, the job supervisor shall determine if work should be stopped and the item retrieved or if the search and retrieval will be performed just before closure.

E. Initiate a request for Site Engineering to evaluate the effects of the foreign material on the system(s) or component(s) if not retrieved.

F. A Technical Evaluation in accordance with the requirements of SPP-9.3 shall be completed for items that fall into the reactor vessel, reactor internals, spent fuel pool/transfer canal or into systems with a direct path to reactor vessel and which cannot be retrieved.

3.6 Completing the Job Closeout

A. Before a system opening is closed, the responsible supervisor/designee shall ensure all PERs identifying Foreign Material have been evaluated and the system determined to be acceptable for closure.

Examination Outline Cross-reference:	Level	RO	SRO
G2.1.41 Conduct of Operations	Tier #	3	
Knowledge of the refueling processes.	Group #		
	K/A #	G2.1.41	
	Importance Rating	2.8	3.7

Proposed Question: **RO # 67**

Which ONE of the following describes the proper orientation of a fuel bundle within a control rod cell while performing a Core Load Verification in accordance with 0-GOI-100-3C, "Fuel Movement Operations During Refueling", Attachment 6?

A. Channel spacer buttons are adjacent to the control blade and adjacent to each other.

B. The bundle serial number is readable as viewed from the center of the reactor core.

C. Channel fasteners for each bundle in the cell are aligned to the outside corners.

D. Orientation boss on the lifting bails point towards the center of the reactor core.

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Incorrect. The bundle serial number is readable as viewed from the center of the cell, not the core.

c. Incorrect. Channel fasteners for each bundle in the cell are grouped in the center of the cell.

d. Incorrect. Orientation boss on the lifting bails point towards the center of the cell, not the core.

Technical Reference(s): 0-GOI-100-3C (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New 06/17/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 0	Fuel Movement Operations During Refueling	0-GOI-100-3C Rev. 0062 Page 102 of 127

Attachment 6
(Page 2 of 3)

Core Verification

Verification of Bundle Orientation

Verification of bundle orientation requires two reviewers. The two reviewers must be familiar with the fuel design features listed below and with Figure 1.

A separate video scan is performed to verify that all bundles are oriented properly. The camera is positioned above the core to allow one complete control cell to be viewed at a time. The reviewers will be provided a core map which has the video path marked with the beginning and ending points.

Experience has demonstrated that certain design features are clearly visible, so that any misoriented fuel assembly will be readily distinguished during core verification. Five separate visual indications of proper fuel assembly orientation for interior cells exist:

A. Channel fasteners for each bundle in the cell are grouped in the center of the cell.

B. Orientation boss on the lifting bails point towards the center of the cell.

C. Channel spacer buttons are adjacent to the control blade and adjacent to each other.

D. The bundle serial number is readable as viewed from the center of the cell.

E. There is cell-to-cell symmetry.

Additional care must be exercised when viewing the partial cells around the core periphery. The fuel assemblies in these cells should have the same orientation as if the core contained a complete control cell including this fuel. These bundles may also be checked against approved core maps.

For peripheral bundle locations, pay particular attention to the channel spacing between the peripheral bundle and the face adjacent bundles. The spacing should be present and symmetric, and if not, further investigation and review is needed.

It was observed that this spacing did not exist when a bundle was misseated. The spacing loss was a result of the bundle leaning and touching its adjacent bundle due to not being inserted into the peripheral support piece.

Examination Outline Cross-reference:	Level	RO	SRO
G2.2.17 Equipment Control	Tier #	3	
Knowledge of the process for managing maintenance activities during power operations.	Group #		
	K/A #	G2.2.17	
	Importance Rating	2.6	3.8

Proposed Question: **RO # 68**

Which ONE of the following describes the management level required to provide FINAL approval of maintenance with a RED RISK classification in accordance with BP-336, "RISK DETERMINATION AND RISK MANAGEMENT?"

A. Risk SRO

B. Shift Manager

C. Operations Manager

D. Plant Manager

Proposed Answer: **D**

Explanation:

a. Incorrect. Risk SRO approves Yellow Risk.

b. Incorrect. Provides approval for all normal GREEN activities.

c. Incorrect. Operations Manager is a member of the Critical Evolutions Review Committee, but does not provide final approval.

d. Correct answer.

Technical Reference(s): BP-336 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:
Bank #
Modified Bank # (Note changes or attach parent)
New 06/17/2008 RMS

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content:
55.41 X
55.43

Comments:

4.1.2 Classification of Repetitive Activities or Tests

4.1.2.1 Repetitive activities or tests which have been previously risk assessed and have not been revised since the last performance will not require a new assessment to be performed.

4.1.2.2 Prior to being included in the schedule, the work control Operations Representative will review the previous assessment to ensure that it is still correct.

4.1.3 Schedule Risk Assessment

4.1.3.1 The Operations Shift Manager/Unit Supervisor shall have ultimate responsibility for ensuring that the impact on overall plant risk is evaluated before systems or components are removed from service for maintenance or surveillance activities.

4.2 Green Risk

4.2.1 Green Risk activities will be performed in accordance with the NPG Human Performance Pocket Guide.

4.3 Yellow Risk

4.3.1 Yellow Risk activities will be approved by the Risk SRO, or designee, and will be documented by signing the Work Implementation Schedule.

4.3.2 If the risk classification is due to:

- PSA/PRA - activities should not be performed on equipment remaining in service which would cause the plant to enter the next level of risk. Consideration shall be given to protecting this equipment as deemed necessary by the Risk SRO, refer to Attachment 5.
- SENTINEL - activities will be managed in accordance with SPP-7.1.
- ORAM - activities will be managed in accordance with SPP-7.2.

4.4 Orange Risk

4.4.1 The Critical Evolutions Review Committee will approve all Orange Risk activities.

4.4.2 If the risk classification is due to:

- PSA/PRA - activities shall not be performed on equipment remaining in service which would cause the plant to enter the next level of risk. Equipment shall be protected as deemed by the Risk SRO, refer to Attachment 5.
- SENTINEL - activities shall not be performed on equipment remaining in service which would cause the plant to enter the next level of risk. Equipment shall be protected as deemed by the Risk SRO, refer to Attachment 5.
- ORAM - activities will be managed in accordance with SPP-7.2.
- ACTIVITY RISK ASSESSMENT (ATTACHMENT 2) - activities which screen to orange risk will be managed in accordance with the barriers and actions prescribed per the applicable Attachment 2. A Responsible Task Lead will be assigned in accordance with SPP-7.1. Management oversight should be considered by the Critical Evolutions Review Committee.
- GRID RELIABILITY - no activities will be scheduled that would increase the probability of loss of offsite power or station blackout. No activities will be scheduled on equipment that is used to mitigate the consequences of a loss of offsite power or station blackout. Any equipment that already has been removed from service that meets the above criteria will be returned to functional status as soon as possible. Refer to Attachment 4 for systems/equipment that meets these criteria.
- PLANNED POWER REDUCTIONS – activities will be coordinated with load dispatch and reactivity management oversight will be provided by Reactor Engineering and Operations. Outage Management will have overall responsibility for schedule development and preparation. Management oversight will be provided as designated by the Critical Evolutions Review Committee.

- REDUCED MARGIN - activities will be assigned to a Responsible Task Lead per SPP-7.1. Other equipment shall be protected as deemed by the Risk SRO, refer to Attachment 5. Activities will be worked around the clock. Management oversight should be considered by the Critical Evolutions Review Committee.
- TIME REMAINING ON S/D LCO (≤ 7 DAYS BUT >72 HOURS) - activities will be worked around the clock and a Responsible Task Lead will be assigned in accordance with SPP-7.1. Management oversight shall be assigned as deemed by the Critical Evolutions Review Committee or Plant Manager/designee.
- COMPLEXITY (½ TRIP, ACTUATION OR ISOLATION & MULTIPLE LOCATIONS & > ANNUAL FREQUENCY) - activities will be managed in accordance with the barriers and actions prescribed per the applicable Attachment 2. A Responsible Task Lead will be assigned in accordance with SPP-7.1 as deemed by the Critical Evolutions Review Committee.
- COMPLEX INTEGRATED ACTIVITIES (WHICH CREATE RISK TO SHUTDOWN SAFETY OR GENERATION) - activities will be assigned to a Responsible Task Lead per SPP-7.1. Management oversight shall be assigned as deemed by the Critical Evolutions Review Committee or Plant Manager/designee.
- OUTAGE ACTIVITIES - activities will be assigned to a Responsible Task Lead per SPP-7.1. Management oversight shall be assigned as deemed by the Critical Evolutions Review Committee or Plant Manager/designee. These activities may be managed per SPP-7.2 and ORAM as applicable.

4.4.3 Level of Oversight and Supervision will be determined by the Critical Evolutions Review Committee.

4.5 Red Risk

4.5.1 The Plant Manager, or designee, will approve all Red Risk activities. The Critical Evolutions Review Committee will approve all Red Risk activities.

Examination Outline Cross-reference:	Level	RO	SRO
G2.2.35 Equipment Control	Tier #	3	
Ability to determine Technical Specification Mode of Operation.	Group #		
	K/A #	G2.2.35	
	Importance Rating	3.6	4.5

Proposed Question: **RO # 69**

Given the following plant conditions:

- Unit 2 is coming out of an outage making preparations for re-start.
- Reactor Coolant temperature is 150 ^{O}F.
- Mode Switch is in the REFUEL position.

As the Unit Operator, you receive a phone call from the Refuel Floor SRO to inform the Control Room that the Reactor Pressure Vessel (RPV) Head is fully tensioned.

Which ONE of the following describes the correct operating MODE in accordance with Technical Specifications?

A. Mode 2

B. Mode 3

C. Mode 4

D. Mode 5

Proposed Answer: **C**

Explanation:

a. Correct answer.

b. Incorrect. Reactor coolant temperature is less than 212 ^{O}F.

c. Incorrect. This would be correct for the given conditions IF the Mode switch was in the SHUTDOWN position.

d. Incorrect. With the RPV head fully tensioned, the plant is no longer in Mode 5.

Technical Reference(s):	U3 TSR Table 1.1-1	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/17/2008 RMS	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel[(a)] or Startup/Hot Standby	NA
3	Hot Shutdown[(a)]	Shutdown	> 212
4	Cold Shutdown[(a)]	Shutdown	≤ 212
5	Refueling[(b)]	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Examination Outline Cross-reference:	Level	RO	SRO
G2.2.7 Equipment Control	Tier #	3	
Knowledge of the process for conducting special or infrequent tests.	Group #		
	K/A #	G2.2.7	
	Importance Rating	2.9	3.6

Proposed Question: **RO # 70**

Which ONE of the following satisfies the criteria that identifies a "Complex Infrequently Performed Test or Evolution?" (CIPTE)

A. Primary System/Reactor Coolant System Barrier Hydrostatic Pressure Test.

B. Rod Worth Minimizer functional test prior to startup in accordance with 1-GOI-100-1A.

C. HPCI Main and Booster Pump Set Developed Head and Flow Rate Test at Rated Reactor Pressure per 3-SR-3.5.1.7.

D. Functional test conducted per the Work Order to verify operability of the RHR Inboard Injection Valve following maintenance.

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Incorrect. This is covered by an approved procedure, is not complex and does not pose an operational risk.

c. Incorrect. This procedure is complex and poses a potential operational risk, but it is covered by an approved procedure and performed routinely.

d. Incorrect. This procedure is not covered by an approved abnormal or normal procedure and does pose an operational risk, but is a simple evolution exempted from the criteria of a CIPTE.

Technical Reference(s):	SPP-2.2	(Attach if not previously provided)
	SPP-8.1	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #	OPL171.078.26	attached
	New		
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Original question OPL171.078.26:

Infrequently performed tests or evolutions that have the potential to significantly degrade the plant's margin of safety that warrant additional management oversight and control are called "Complex Infrequently Performed Tests or Evolutions." (i.e., CIPTE)

Which ONE of the following is a criteria that identifies a CIPTE?

A. Tests/evolutions not specifically covered by existing normal or abnormal operating procedures.

B. Data taking, for example gauge reading, annunciator observations, data compilation, and inspection or inventory type tests.

C. Critical procedures such as Emergency Operating Instructions (EOIs), Abnormal Operating Instructions (AOIs) and Annunciator Response Procedures (ARPs).

D. Functional tests conducted by work control documents such as stroking valves, bumping motors, calibrations, visual inspections, leak/pressure tests, and electrical continuity checks.

5.0 DEFINITIONS

Affected Organization Review - A review by organizations for impacts and implementation requirements in their area of responsibility as a result of a procedure or process change.

Approval Authority - The approval authority for a procedure is the manager/supervisor of the organizational unit responsible for the procedure. This authority shall not be routinely delegated to a lower level within the organization. Higher level managers may approve procedures within the area of responsibility.

Audit Trail - A QA record generated by electronic workflow routing which provides the procedure electronic review and approval information.

Complex Infrequently Performed Tests or Evolutions - Infrequently performed tests or evolutions that have the potential to significantly degrade the plant's margin of safety that warrant additional management oversight and control. The below criteria shall be used to identify these types of tests/evolutions:

A. Tests/evolutions not specifically covered by existing normal or abnormal operating procedures.

B. Tests/evolutions that are seldom performed even though covered by existing normal or abnormal procedures (for example, plant startup after a prolonged outage or after any outage that involves significant changes to systems, equipment, or procedures related to the core, reactivity control, or reactor protection).

C. Special, infrequently performed surveillance testing that involves complicated sequencing or placing the plant in an unusual configuration.

D. Tests/evolutions that require the use of special test procedures in conjunction with existing procedures.

Critical Step - A critical step is a work-related step or action that, if performed incorrectly, will significantly harm plant equipment or significantly impact plant operation. A step, action, or phase of a task is considered critical, if it satisfies all of the following conditions:

A. The consequences of incorrect performance are intolerable to reactor safety, generation, or to plant equipment (see DEFINITION of **Intolerable Consequences to the Plant**).

B. The consequences are realized immediately.

NOTE
"Immediately" should be construed as "during implementation of the procedure." For example, if the incorrect performance of Step A could cause an immediate intolerable consequence when Step B is performed, then Step A should be considered a critical step.

1.0 PURPOSE

This procedure describes the requirements, responsibilities, and administrative controls necessary for conducting test activities at TVA Nuclear's (TVAN's) facilities.

This procedure provides guidelines to help ensure that:

A. Test activities are performed in a manner to reduce the possibility of an undesired manipulation or actuation of plant equipment.

B. Test activities are performed by qualified personnel.

C. Personnel who perform test activities understand their duties and responsibilities.

2.0 SCOPE

A. This procedure applies to personnel involved in the administration, supervision, and performance of operations phase test activities on installed equipment at TVAN's facilities.

Test activities on equipment not under jurisdictional control of the Plant Manager on unlicensed units are excluded from this procedure.

B. This procedure contains the minimum requirements for conducting the following tests except as noted in item 2.0.C below.

1. Surveillance tests
2. Functional tests
3. Postmaintenance/Postmodification tests
4. Complex Infrequently Performed Tests or Evolutions (CIPTEs) including Special Tests
5. Technical related Instructions that perform a test (i.e., Technical Instructions, etc.)
6. Contractor/Vendor test

C. The following activities must also be performed in accordance with this procedure, but do not require Form SPP-8.1-1, "Test Director Assignment Sheet," Form SPP-8.1-2, "Chronological Test Log (CTL)," or prebrief unless required by Operations.

1. Simple functional tests conducted by work control documents, such as stroking valves, bumping motors, simple calibrations, and visual inspections or site implementing procedures, such as leak/pressure tests, electrical continuity.

2. Tests that do not require changes in plant status and/or configuration such as data taking (e.g., gauge reading, annunciator observations, data compilation), inspection or inventory type tests, or inspections.

3. Simple tests performed on a routine basis such as RadioChem Lab sampling instructions.

Examination Outline Cross-reference:	Level	RO	SRO
G2.3.15 Radiation Control	Tier #	3	
Knowledge of radiation monitoring systems.	Group #		
	K/A #	G2.3.15	
	Importance Rating	2.9	3.1

Proposed Question: **RO # 71**

Which ONE of the following Process Radiation Monitor systems will NOT be adversely affected by a loss of Reactor Protection System (RPS) bus A?

A. Main Steam Line radiation monitors.

B. Refuel Zone Ventilation radiation monitors.

C. Reactor Zone Ventilation radiation monitors.

D. Control Room Emergency Ventilation radiation monitors.

Proposed Answer: **D**

Explanation:

a. Incorrect. Two channels are powered from each RPS bus.

b. Incorrect. Two channels are powered from each RPS bus.

c. Incorrect. Two channels are powered from each RPS bus.

d. Correct answer.

Technical Reference(s):	OPL171.033	(Attach if not previously provided)
Proposed references to be provided to applicants during examination:	None	
Question Source:	Bank # OPL171.033.18	
	Modified Bank #	(Note changes or attach parent)
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Excerpt from OPL171.033 pages 25 and 35:

1. Reactor Building/Refuel Zone Ventilation Radiation Monitoring System (RM-90-140, 141, 142, and 143)

   a. Purpose

      (1) Indicates whenever abnormal amounts of radioactive material exists in the exhaust plenum of the reactor building/refuel zone and isolate the inlet and exhaust air flows

   b. Four gamma sensitive GM instrumentation channels monitor the radiation from the reactor zone exhaust and four identical channels monitor the radiation from the refueling zone

      (1) These are physically located on the side of the ventilation ducts on the refuel floor

      (2) Monitors RM-90-140(A & B) and 90-142(A & B) are fed from RPS ‘A’

      (3) Monitors RM-90-141(A & B) and 90-143(A & B) are fed from RPS ‘B’

2. Control Room Ventilation Radiation Monitoring System (90-259 A and B)

      (1) The external power supplies for the control room radiation monitor assemblies are:

         (a) 0-RM-90-259A - Panel 1-9-9 Cabinet 2 (Unit 1 I&C A) Breaker 222

         (b) 0-RM-90-259B - Panel 3-9-9 Cabinet 3 (Unit 3 I&C B) Breaker 325

3. Main Steam Line Radiation Monitoring System

      (1) RPS "A" supplies "A" (RM 90-136-A1) and "C" (RM-90-137-A2) radiation monitors. RPS "B" supplies "B" (RM-90-138-B1) and "D" (RM-90-139-B2) radiation monitors

Examination Outline Cross-reference:	Level	RO	SRO
G2.3.4 Radiation Control	Tier #	3	
Knowledge of radiation exposure limits under normal and emergency conditions.	Group #		
	K/A #	G2.3.4	
	Importance Rating	3.2	3.7

Proposed Question: **RO # 72**

Given the following plant conditions:

- A Fuel Pool cleanout is in progress of Unit 2.
- A failure of the Refuel Bridge monorail hoist allowed a bucket of irradiated stellite ball bearings to be raised above the fuel pool water level.
- The Refuel Floor radiation levels initiated a Group 6 isolation on all three units.
- The AUO operating the Refuel Bridge received a dose of 12 rem while manually lowering the bucket below the water level.

Which ONE of the following describes whether or not this constitutes an emergency exposure in accordance with EPIP-15, "Emergency Exposures" and the basis for this conclusion?

In accordance with EPIP-15, "Emergency Exposures," this ____(1)____ constitute an Emergency Exposure. The basis for this conclusion is ____________(2)____________.

	(1)	(2)
A.	DOES	only 25 rem is allowed for equipment problems.
B.	does NOT	only planned exposures are covered under EPIP-15 limits.
C.	DOES	spontaneous actions taken to mitigate a problem are covered by the EPIP-15 exposure limits.
D.	does NOT	this exposure is classified as a Planned Special Exposure.

Proposed Answer: **B**

Explanation:

a. Incorrect. The 25 rem limit is not applicable for this situation.

b. Correct answer.

c. Incorrect. Spontaneous actions taken to mitigate a problem are NOT covered by the EPIP-15 exposure limits.

d. Incorrect. This exposure was not planned, it was spontaneous.

Technical Reference(s):	EPIP-15, "Emergency Exposure".		(Attach if not previously provided)
Proposed references to be provided to applicants during examination:			None
Question Source:	Bank #	OPL171.000.04	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	X	
	55.43		

Comments:

BROWNS FERRY	EMERGENCY EXPOSURES	EPIP-15

1.0 INTRODUCTION

1.1 Purpose

This procedure provides guidance for authorizations of personnel dose limits under emergency conditions as described in EPA-400, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents".

These limits apply only to emergency exposure authorizations and not to spontaneous reactions by individuals attempting to mitigate an emergency situation. This procedure provides guidance for the approval and administration of radiation exposures received during emergencies in excess of 10 CFR 20.1201 entitled "Occupational Dose Limits for Adults". This procedure does not provide direction for the approval and administration for exposures received during other activities involving radiation dose to individuals.

NOTE: For the purpose of this implementing procedure, radiation exposure as expressed in unit of R/hr and sub-units, thereof, are equivalent to dose (rad) and dose equivalent (rem) based on ANSI N 13.11 development and terminology. Any acute dose greater than 10 rem is generally denoted in units of rad since that level is considered as the accident range of personnel exposure. Any dose less than that level is considered as the protective range of personnel exposure. For purposes of this procedure the assumptions of 1 rad = 1 rem is assumed for all levels of exposure.

2.0 REFERENCES

2.1 Industry Documents

A. EPA-400-R92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents"
B. 10 CFR 20.1201, Code of Federal Regulations

2.2 Plant Instructions

A. TVA Radiological Emergency Plan
B. EPIP-10, "Medical Emergency Procedure"
C. EPIP-14, "Radiological Control Procedures"
D. SPP 5.10, "Radiological Respiratory Protection Program"
E. SPP 5.1, "Radiological Controls"

BROWNS FERRY	EMERGENCY EXPOSURES	EPIP-15

3.3.7 Personnel shall not enter any area where dose rates are unknown or not measurable with instruments and dosimetry immediately available.

NOTE: The value below corresponds to the ratio of external (measured) dose rate to estimate TEDE dose, in accordance with default values in TVA's Dose Assessment model. When accident specific nuclide assessment are available, more definitive dose assessments should be performed to adjust the correction factors.

3.3.8 Until isotopic assessments of airborne radioactivity are available, an administrative correction factor of 2 should be used to estimate TEDE exposures in airborne activity areas:

Estimated TEDE = Dosimeter Reading **X** 2

3.4 **Dose Limits for Workers During Emergencies**

3.4.1 Doses to all workers during emergencies should, to the extent practicable be limited to 10 CFR 20.1201 limits. There are, however, some emergency situations for which higher emergency exposures may be justified. Whenever these situations are justified and ALARA considerations have been evaluated the following limits can be administered.

3.4.2 Radiation Protection (RP) considers the to-date annual accrued dose to individuals when establishing the maximum dose limits for workers during emergencies. The to-date annual accrued dose would be subtracted from the applicable emergency dose limit to determine the authorized allowable dose for the emergency.

3.4.3 Dose Limits for the Protection of Valuable Property

Dose Limit (Rem)	Receptor
10	Whole Body (TEDE)
30	Lens of the Eye
100	All Other Organs

3.4.4 Dose Limits for Lifesaving Activities and the Protection of Large Populations

Dose Limit (Rem)	Receptor
25	Whole Body (TEDE)
75	Lens of the Eye
250	All Other Organs

Examination Outline Cross-reference:	Level	RO	SRO
G2.4.20 Emergency Procedures/Plans	Tier #	3	
Knowledge of operational implications of EOP warnings, cautions and notes.	Group #		
	K/A #	G2.4.20	
	Importance Rating	3.8	4.3

Proposed Question: **RO # 73**

Given the following Unit 1 plant conditions:

- A reactor scram has occurred.
- HPCI is needed to maintain reactor water level.
- Suppression pool temperature is 145 ^{O}F.

Which ONE of the following describes the reason HPCI is operated with a suction from the CST if possible?

A. The HPCI turbine exhaust pressure is likely to exceed the Primary Containment Pressure limit.

B. The HPCI pump shaft seals are not designed to operate at temperatures in excess of 140 ^{O}F and may fail.

C. The suppression pool provides insufficient NPSH to the HPCI pump and cavitation may occur at rated flow.

D. The HPCI lube oil will exceed allowable temperatures and the HPCI function could be lost due to damaged bearings.

Proposed Answer: **D**

Explanation:

a. Incorrect. This condition is possible, but is an issue with SP level below 12.75 feet, not high temperature.

b. Incorrect. HPCI pump shaft seals are capable of operating at higher temperatures even though they are cooled by water from the suppression pool.

c. Incorrect. NPSH will be lower at high SP temperatures, but will remain within the allowable levels for HPCI operation.

d. Correct answer. High lube oil temperatures degrade the viscosity of the lube oil and could result in bearing damage.

Technical Reference(s):	1-EOI-1 Flowchart	(Attach if not previously provided)
	EOIPM	
Proposed references to be provided to applicants during examination:		None
Question Source:	Bank # OPL171.201.11	minor format changes
	Modified Bank #	(Note changes or attach parent)
	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 X	
	55.43	

Comments:

Original Question OPL171.201.11:

A reactor scram has occurred and HPCI is needed to maintain reactor water level. Suppression pool temperature is $145^{O}F$.

SELECT the statement that correctly describes the reason HPCI is operated with a suction from the CST if possible.

A. The suppression pool provides insufficient NPSH to the HPCI pump and cavitation may occur at rated flow.

B. The HPCI pump shaft seals are not designed to operate at temperatures in excess of $140^{O}F$ and may fail.

C. The HPCI lube oil will exceed allowable temperatures and the HPCI function could be lost due to damaged bearings.

D. The HPCI turbine exhaust pressure is likely to exceed the turbine trip setpoint.

Excerpt from OPL171.201 page 38:

1. **Caution #6**

   "Operating HPCI or RCIC Turbines with suction temperatures above 140 °F may result in equipment damage"

   a. The HPCI and RCIC Lube Oil Coolers are cooled by routing part of the pump discharge fluid to the cooler. At elevated temperatures in the suppression pool, the turbine lube oil may get too hot to provide adequate lubrication.

   b. Only during EOI operations will the system be needed at such an extreme suppression pool temperature. Therefore, the EOIs are an appropriate location for this caution.

Examination Outline Cross-reference:	Level	RO	SRO
G2.4.23 Emergency Procedures/Plans	Tier #	3	
Knowledge of the basis for prioritizing emergency procedure implementation during emergency operations.	Group #		
	K/A #	G2.4.23	
	Importance Rating	3.4	4.4

Proposed Question: **RO # 74**

Given the following plant conditions:

- An accident has occurred on Unit 1 which has resulted in entry into Severe Accident Management Guidelines (SAMG).
- SAMGs are being implemented from the Technical Support Center (TSC).

Which ONE of the following procedure classifications is inappropriate to be used in conjunction with SAMG implementation?

A. Abnormal Operating Instructions (AOI).

B. Emergency Plan Implementing Procedures (EPIP).

C. Emergency Operating Instruction (EOI) Flowcharts.

D. Emergency Operating Instruction (EOI) Appendices.

Proposed Answer: **C**

Explanation:

a. Incorrect. Although some AOI guidance may conflict with SAMG guidance, this is NOT prohibited. However, SAMG implementation takes precedence over AOI guidance IF a conflict exists.

b. Incorrect. EPIP implementation is authorized and will certainly be required under the given conditions.

c. Correct answer.

d. Incorrect. SAMG procedures have several Appendices specific to SAMG implementation, but some EOI Appendices are still appropriate and are used.

Technical Reference(s):	OPL171.212 page 6, 0-SSI-001	(Attach if not previously provided)
	OPL171.201 page 22	

Proposed references to be provided to applicants during examination: None

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	09/18/2008 RMS	
Question History:	Last NRC Exam		

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Excerpt from OPL171.212 page 6:

A. EOI Transition into SAMG - Loss of Coolable Geometry

1. The SAMGs are entered, then the core geometry is assumed to be changed and NOT coolable. The EOI strategies are employed for accidents inside BFN design basis. When accidents progress to a point where BFN design basis is exceeded, SAMG entry will be required.

3. At each of these specific points, we cannot assume a coolable geometry exists and SAMG entry is required.

4. Once the SAMGs are entered, the EOI flowcharts no longer apply because the configuration of the core may no longer be amenable to adequate cooling. All EOI flowcharts will be exited and will not be referred to again. Any subsequent EOI entry condition which is received will NOT result in EOI entry.

6. Other procedures (AOIs, ARP's, EPIPs, etc.) have event specific entry conditions and may be used to supplement SAMGs. The control room staff may continue to use several such procedures in response to lower-level plant alarms, lineups, etc.

   - Actions that contradict any direction provided by the SAM Team shall NOT be performed.

Examination Outline Cross-reference:	Level	RO	SRO
G2.4.31 Emergency Procedures/Plans	Tier #	3	
Knowledge of annunciators alarms, indications or response procedures.	Group #		
	K/A #	G2.4.31	
	Importance Rating	4.2	4.1

Proposed Question: **RO # 75**

Unit 3 is performing a Reactor Startup with the following conditions:

- RPV pressure is at 750 psig.
- Control Rod withdrawal is in progress.
  Reactor power is at Range 6 on the IRMs.
- The Woodward Governor for 3A RFP fails upscale and the Reactor scrams on APRM High-High.
- The Operating Crew stabilizes the Unit.
- After the scram is reset the OATC notes the following annunciators:

  - DRYWELL PRESSURE HIGH HALF SCRAM (9-4A W8).
  - DRYWELL TEMP HIGH (9-3B W16).
  - DRYWELL/SUPPR CHAMBER RADIATION HIGH (9-7C W15)
  - OG PRETREATMENT RADIATION HIGH (9-3A W5)

Which ONE of the following actions should the Unit Supervisor direct to be completed within 2 hours?

A. Inject Standby Liquid Control.

B. Place SJAEs on Auxiliary Boiler Steam supply.

C. Open 3-FCV-1-56 Main Steam Line Drain.

D. Place Steam Seals on Auxiliary Boiler Steam supply.

Proposed Answer: **A**

Explanation:

a. Correct answer.

b. Incorrect. Transferring SJAEs is not required.

c. Incorrect. 3-FCV-1-58 and 59 should be opened, not 3-FCV-1-56.

d. Incorrect. There is no requirement to place steam seals on Aux Boiler steam. The requirement is to adjust the regulator to zero.

Technical Reference(s): ______________________ (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Question Source: Bank # OPL171.033 62

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

BFN Unit 3	Panel 9-7 3-XA-55-7C	3-ARP-9-7C Rev. 0026 Page 20 of 42

DRYWELL/SUPPR CHAMBER RADIATION HIGH 3-RA-90-272

15

(Page 1 of 2)

Sensor/Trip Point:

3-RM-90-272A	100 R/HR
3-RM-90-273A	160 R/HR
3-RM-90-272B	9.99 E6 R/HR (Disabled Alarm Input)
3-RM-90-273B	9.99 E6 R/HR (Disabled Alarm Input)

Sensor Location: 3-RR-90-272, Panel 3-9-54, Main Control Room.
3-RR-90-273, Panel 3-9-55, Main Control Room.

Probable Cause:
A. High radiation levels.
B. Sensor malfunction.
C. Noise spikes.

Automatic Action: None

Operator Action:
A. **VERIFY** alarm on 3-RR-90-272 (Panel 3-9-54), and 3-RR-90-273 (Panel 3-9-55). ☐
B. **CHECK** 3-RR-90-256 for rise. ☐
C. **ATTEMPT** to isolate equipment to stop source. ☐
D. **IF** the alarm is determined to be valid, **THEN PERFORM** the following within 2 hours of the alarm:
- **OPEN** UPSTREAM MSL DRAIN TO CONDENSER 3-FCV-001-0058. ☐
- **OPEN** DOWNSTREAM MSL DRAIN TO CONDENSER 3-FCV-001-0059. ☐
- **ENSURE** 3-PCV-001-0147 is Closed by taking STEAM SEAL REGULATOR, 3-HS-1-147 to CLOSE. (Panel 9-7) ☐

Continued on Next Page

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DRYWELL/SUPPR CHAMBER RADIATION HIGH 3-RA-90-272, Window 15
(Page 2 of 2)

Operator
Action: (Continued)

E. **IF** ALL the following conditions exist (E.1,E.2, and E.3):
   1. Alarm is determined to be valid, **AND**
   2. The reactor will remain subcritical without boron injection under all conditions, **AND**
   3. Leakage of primary coolant into primary containment is indicated, **THEN**

   Within 2 hours of alarm, **INJECT** SLC for alternate source term control by placing SLC PUMP 3A/3B, 3-HS-63-6A in the START A or **START** B position. ☐

F. **REFER TO** EPIPs. ☐

G. **IF** started at Operator Action Step E, **THEN**
   WHEN SLC tank reaches 0", **STOP** the running SLC Pump ☐

References: 3-45E620-9 0-47E610-90-2 NESSD 3R-090-273A-00
Technical Specifications Section 3.3.3.1 NESSD 3R-090-272A-00

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0037 Page 10 of 51

OG PRETREATMENT RADIATION HIGH 3-RA-90-157A 5	Sensor/Trip Point:	HI
	3-RM-90-157	15.95 R/HR

(Page 1 of 2)

Sensor Location: RE-90-157, Turb Bldg OG pretreatment sample chamber, El 565', T-14 B-LINE

Probable Cause:
A. High radiation in the off-gas pretreatment system.
B. Resin trap failure (RWCU or Cond Demin).
C. Possible fuel element failure.
D. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **VERIFY** high radiation on following:
   1. OFFGAS PRETREATMENT RADIATION recorder, 3-RR-90-157 on Panel 3-9-2. ☐
   2. OFFGAS RADIATION recorder, 3-RR-90-160 on Panel 3-9-2. ☐
   3. OG PRETREATMENT RAD MON RTMR, 3-RM-90-157 on Panel 3-9-10. ☐
   4. OFFGAS RAD MON RTMR, 3-RM-90-160 on Panel 3-9-10. ☐

B. **CHECK** off-gas flow normal. ☐

C. **CHECK** following radiation recorders and associated radiation monitors:
   1. MAIN STEAM LINE RADIATION, 3-RR-90-135 on Panel 3-9-2. ☐
   2. OFFGAS POST-TREATMENT RADIATION, 3-RR-90-265 on Panel 3-9-2. ☐
   3. STACK GAS/CONT RM RADIATION FROM STACK GAS, 0-RR-90-147 on Panel 1-9-2. ☐

D. **NOTIFY** RADCON. ☐

E. **REQUEST** Chemistry perform radiochemical analysis to determine source. ☐

Continued on Next Page

BFN Unit 3	Panel 9-3 3-XA-55-3A	3-ARP-9-3A Rev. 0037 Page 11 of 51

OG PRETREATMENT RADIATION HIGH 3-RA-90-157A, Window 5
(Page 2 of 2)

Operator
Action: (Continued)

F. IF Offgas System Isolation Valve, 3-FCV-66-28 is manually restrained in the OPEN position and it has been determined that this is a valid alarm, THEN

UNRESTRAIN Offgas System Isolation Valve, 3-FCV-66-28.

G. REFER TO 0-SI-4.8.B.1.a.1 and 1(2)(3)-SR-3.4.6.1(A) for ODCM compliance and to determine if power level reduction is required. ☐

H. IF directed by Unit Supervisor, THEN
REDUCE reactor power to maintain off-gas radiation within ODCM limits. ☐

I. IF ODCM limits are exceeded, THEN
REFER TO EPIP-1. ☐

References: 3-45E620-3 3-47E610-90-1 GE 3-729E814-4 3-SIMI-90B

BFN Unit 3	Panel 9-3 3-XA-55-3B	3-ARP-9-3B Rev. 0018 Page 19 of 38

DRYWELL TEMP HIGH 3-TA-64-52 (16)	Sensor/Trip Point:	
	TE-64-52C	≥ 154°F (Alarm comes off recorder 3-XR-64-50)

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Sensor Location:
TE-64-52A
Rx Bldg (Drywell)
El 584'
225° (AZ)

Probable Cause:
A. Drywell cooler(s) failure.
B. Loss of RBCCW to Drywell Cooler(s).
C. Possible leak in Drywell.

Automatic Action: None

Operator Action:
A. **CHECK** Drywell temperatures and pressures using multiple indications. ☐
B. **VERIFY** Drywell coolers running and **START** spare Drywell Cooler(s). ☐
C. **CHECK OPEN** RBCCW PRI CNTMT OUTLET VALVE, 3-HS-70-47A (Panel 3-9-4). ☐
D. **START** additional RCW pumps. ☐
E. **IF** high Drywell temperature continues, **THEN** **REFER TO** 3-AOI-64-1. ☐
F. **IF** high Drywell temperature is due to a loss of RBCCW, **THEN** **REFER TO** 3-AOI-70-1. ☐
G. **IF** temperature is above 160°F, **THEN** **ENTER** 3-EOI-2 Flowchart. ☐

References: 3-45E620-3 3-47E610-64-1 47W600-90

BFN **Unit 3**	**Panel 9-4** **3-X-55-4A**	**3-ARP-9-4A** **Rev. 0037** **Page 11 of 45**

DRYWELL PRESSURE HIGH HALF SCRAM (8)	Sensor/Trip Point:	
	3-PIS-064-0056A	2.45 psi
	3-PIS-064-0056B	positive
	3-PIS-064-0056C	pressure in
	3-PIS-064-0056D	the drywell.

(Page 1 of 1)

Sensor Location: 3-PNLA-009-0083, 0084, 0085, 0086 U3 Aux Inst Room

Probable Cause:
A. ≥2.45 psig in the drywell.
B. Sensor malfunction.
C. SI or SR in progress.

Automatic Action:
A. Half scram if one sensor actuates.
B. Reactor scram and group 2, 6 and 8 PCIS if one sensor per channel actuates.

Operator Action:
A. **VERIFY** alarm by multiple indications. ☐
B. IF drywell pressure is ≥2.45 psig AND reactor has **NOT** scrammed, **THEN**
MANUALLY SCRAM the reactor. **ENTER** 3-EOI-1 FLOWCHART & 3-EOI-2 FLOWCHART. ☐
C. **DISPATCH** personnel to the pressure switches to check for abnormal condition. ☐
D. IF alarm is **NOT** valid or initiating condition is corrected, **THEN** with SRO permission, **RESET** Half Scram. **REFER TO** 3-OI-99. ☐

References: 3-45N620-5 GE 730E915 FSAR Section 7.2.3.5 and 13.6.2