Examination Outline Cross-reference:			
295003 Partial or Complete Loss of AC / 6	Level	[redacted]	SRO
G2.4.50 (10CFR 55.43.5 - SRO Only)	Tier #	[redacted]	1
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	Group #	[redacted]	1
	K/A #		295003G2.4.50
	Importance Rating	[redacted]	4.0

Proposed Question: **# 76**

Unit 1 is operating at 100% Reactor Power, when the following alarms are received:

- CONTROL ROD WITHDRAWAL BLOCK,1-9-5A, (Window 7)
- RPIS INOPERATIVE, 1-9-5A, (Window 14)
- CONTROL ROD OVERTRAVEL, 1-9-5A, (Window 35)
- **ALL** Control Rod position indication has been lost

(ASSUME NO OPERATOR ACTIONS)

Which ONE of the following completes the statement?

Based on the above conditions, entry into __(1)__ is required **AND** the loss of **ALL** control rod position indication will require the operating crew to __(2)__ .

[REFERENCE PROVIDED]

A. **(1)** 0-AOI-57-3, "Loss of Plant Preferred,"
   (2) insert **ALL** control rods in 3 hours **AND** disarm the associated CRDs in 4 hours.

B. **(1)** 0-AOI-57-3, "Loss of Plant Preferred,"
   (2) be in Mode 3 in 12 hours

C. **(1)** 1-AOI-57-4,"Loss of Unit Preferred,"
   (2) insert **ALL** control rods in 3 hours **AND** disarm the associated CRDs in 4 hours.

D. **(1)** 1-AOI-57-4,"Loss of Unit Preferred,"
   (2) be in Mode 3 in 12 hours

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 = incorrect, these alarms indicate a loss of Unit Preferred NOT Plant Preferred. Part 2 = incorrect, although entry into TS 3.1.3 Condition C is required, the inoperable control rods can not be inserted. Per1-AOI-57-4 Step 4.2[2], if control rod movement is required while RPIS and the process computer are inoperable, THEN INSERT a MANUAL SCRAM REFER TO 1-AOI-100-1.

B INCORRECT: Part 1 = incorrect, these alarms indicate a loss of Unit Preferred NOT Plant Preferred. Part 2 = correct, as required by TS 3.1.3 Condition E.

C INCORRECT: Part 1 = correct, applicable procedure to reference. Part 2 = incorrect, as explained above.

D **CORRECT:** Part 1 = correct, applicable procedure to reference. Part 2 = correct, with a loss of all position indication, all control rods are inoperable. Per TS 3.1.3 Condition E, if 9 or more control rods are inoperable, required action is to be in Mode 2 in 12 hours.

Technical Reference(s): 1-9-5A Rev 12, 0-AOI-57-3 Rev 40 (Attach if not previously provided)

1-AOI-57-4 Rev 26 / TS 3.1.3

Proposed references to be provided to applicants during examination: TS 3.1.3

Learning Objective: ____________ (As available)

____

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	<u>AND</u>		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	------------NOTE------------ RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. --------------------------------	
		Fully insert inoperable control rod.	3 hours
	<u>AND</u>		
	C.2	Disarm the associated CRD.	4 hours

(continued)

Control Rod OPERABILITY
3.1.3

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. -------------NOTE------------ Not applicable when THERMAL POWER > 10% RTP. --------------------------------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1	Restore compliance with BPWS.	4 hours
	<u>OR</u>		
	D.2	Restore control rod to OPERABLE status.	4 hours
E. Required Action and associated Completion Time of Condition A, C, or D not met. <u>OR</u> Nine or more control rods inoperable.	E.1	Be in MODE 3.	12 hours

Control Rod OPERABILITY
3.1.3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	24 hours

BFN Unit 1	Loss of Unit Preferred	1-AOI-57-4 Rev. 0026 Page 8 of 29

4.2 **Subsequent Actions (continued)**

[3] Perform the following for the CRD system:

[3.1] Monitor CRD temperatures while CRD SYS FLOW CONTROL VLV 1A /B, 1-FCV-85-11A/B are closed. ☐

[3.2] **IF** CRD seal temperatures rise to the alarm setpoint **OR** the Unit Preferred system cannot be restored within one hour, **THEN**

DISPATCH personnel to **MANUALLY OPEN** CRD SYS FLOW CONTROL VLV 1A /B, 1-FCV-85-11A/B.
REFER TO 1-OI-85
(Otherwise N/A) ☐

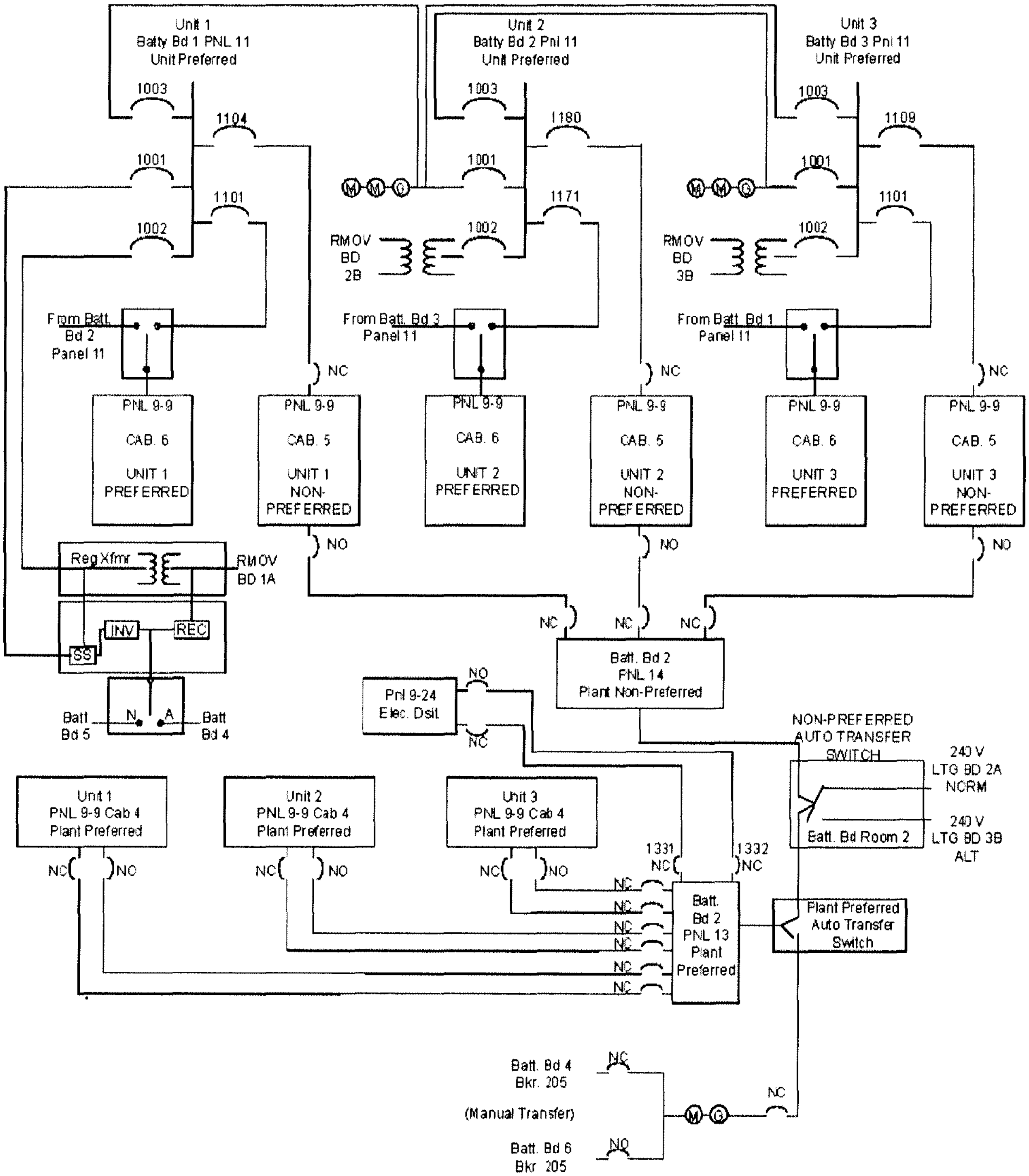
[3.3] **IF** Cabinet 5 fails to transfer or is otherwise de-energized, and Cabinet 6 is energized **THEN**

PLACE controller 1-FIC-85-11 in manual and adjust to normal Drive Water Pressure is obtained. ☐

BFN Unit 1	Loss of Unit Preferred	1-AOI-57-4 Rev. 0026 Page 14 of 29

Illustration 1
(Page 1 of 1)

Vital 120V AC Distribution


Unit 1
Batty Bd 1 PNL 11
Unit Preferred
Unit 2
Batty Bd 2 Pnl 11
Unit Preferred
Unit 3
Batty Bd 3 Pnl 11
Unit Preferred
1003
1104
1001
1101
1002
1180
1171
1109
RMOV BD 2B
RMOV BD 3B
From Batt. Bd 2 Panel 11
From Batt. Bd 3 Panel 11
From Batt. Bd 1 Panel 11
NC
NO
PNL 9-9 CAB. 6 UNIT 1 PREFERRED
PNL 9-9 CAB. 5 UNIT 1 NON-PREFERRED
PNL 9-9 CAB. 6 UNIT 2 PREFERRED
PNL 9-9 CAB. 5 UNIT 2 NON-PREFERRED
PNL 9-9 CAB. 6 UNIT 3 PREFERRED
PNL 9-9 CAB. 5 UNIT 3 NON-PREFERRED
Reg Xfmr
RMOV BD 1A
INV
REC
SS
Batt Bd 5
Batt Bd 4
N
A
Batt. Bd 2 PNL 14 Plant Non-Preferred
Pnl 9-24 Elec. Dsil
NON-PREFERRED AUTO TRANSFER SWITCH
240 V LTG BD 2A NORM
240 V LTG BD 3B ALT
Batt. Bd Room 2
Unit 1 PNL 9-9 Cab 4 Plant Preferred
Unit 2 PNL 9-9 Cab 4 Plant Preferred
Unit 3 PNL 9-9 Cab 4 Plant Preferred
1331
1332
Batt. Bd 2 PNL 13 Plant Preferred
Plant Preferred Auto Transfer Switch
Batt. Bd 4 Bkr. 205
(Manual Transfer)
Batt. Bd 6 Bkr. 205

BFN Unit 1	Loss of Unit Preferred	1-AOI-57-4 Rev. 0026 Page 7 of 29

[2] **IF** control rod movement is required while RPIS and the process computer are inoperable, **THEN**

INSERT a MANUAL SCRAM **REFER TO** 1-AOI-100-1. (Otherwise N/A) ☐

BFN Unit 1	Loss of Unit Preferred	1-AOI-57-4 Rev. 0026 Page 5 of 29

2.0 SYMPTOMS (continued)

G. The following rod control annunciators are in alarm simultaneously due to a loss of power to the associated circuits.

1. CONTROL ROD WITHDRAWAL BLOCK (1-XA-55-5A, Window 7) on Panel 1-9-5 is in alarm.
2. CONTROL ROD OVERTRAVEL (1-XA-55-5A, Window 14) on Panel 1-9-5 is in alarm.
3. RPIS INOPERATIVE (1-XA-55-5A, Window 35) on Panel 1-9-5 is in alarm.

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

RPIS INOPERABLE 35	Sensor/Trip Point:	
	Relay 3A-K5	Receive alarm if there is an electronic malfunction such as: A. Card pulled. B. Internal logic stall.

(Page 1 of 1)

Sensor Location: 1-PNLA-009-0028, Elev. 593', Aux. Inst. Room

Probable Cause:

A. 120V unit preferred breaker 612 on Panel 1-9-9 tripped.
B. 1-PX-58-5X(5Y)(6X)(6Y) fuse cleared or internal breaker open (1-PNLA-009-0027, Aux. Inst. Room).
C. Malfunction of a card in 1-PNLA-009-0027.
D. Spurious trip of sensor.

Examination Outline Cross-reference:			
295016 Control Room Abandonment / 7	Level		SRO
G2.4.3 (10CFR 55.43.5 - SRO Only)	Tier #		1
Ability to identify post-accident instrumentation.	Group #		1
	K/A #	295016G2.4.3	
	Importance Rating		3.9

Proposed Question: **# 77**

Unit 2 is operating at 100% Reactor Power, when conditions cause the Control Room to be abandoned.

The following conditions exist:

- Time 0730, control has been established per 2-AOI-100-2, "Control Room Abandonment"
- Reactor Water Level indicates (+) 27 inches and stable
- Reactor Pressure indicates 850 psig and lowering slowly
- Time 0800, Unit Operator reports that Reactor Water Level Indicator, 2-LI-3-46A, is reading off-scale low and appears broken
- Time 1200, control has been shifted back to the Control Room

Which ONE of the following completes the statement?

Entry into Tech Spec(s) __________.

A. 3.3.3.1, "PAM Instrumentation," is required upon discovery.

B. 3.3.3.2, "Backup Panel Instrumentation," is required upon discovery.

C. 3.3.3.1, "PAM Instrumentation," **AND** 3.3.3.2, "Backup Panel Instrumentation," are required upon discovery.

D. 3.3.3.2, "Backup Panel Instrumentation," is required upon discovery, **AND** 3.3.3.1, "PAM Instrumentation," when control is shifted back to the Control Room.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: 2-LI-3-46A is **not** a PAM instrument per T.S. 3.3.3.1.

B **CORRECT:** 2-LI-3-46A is applicable to T.S. 3.3.3.2 only.

C INCORRECT: 2-LI-3-46A is **not** a PAM instrument per T.S. 3.3.3.1.

D INCORRECT: 2-LI-3-46A is **not** a PAM instrument per T.S. 3.3.3.1. Even if it were, delay until control is shifted back to the Main Control Room is **not** prudent.

Technical Reference(s): T.S.3.3.3.1, T.S.3.3.3.2 (Attach if not previously provided)

(Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

Backup Control System
B 3.3.3.2

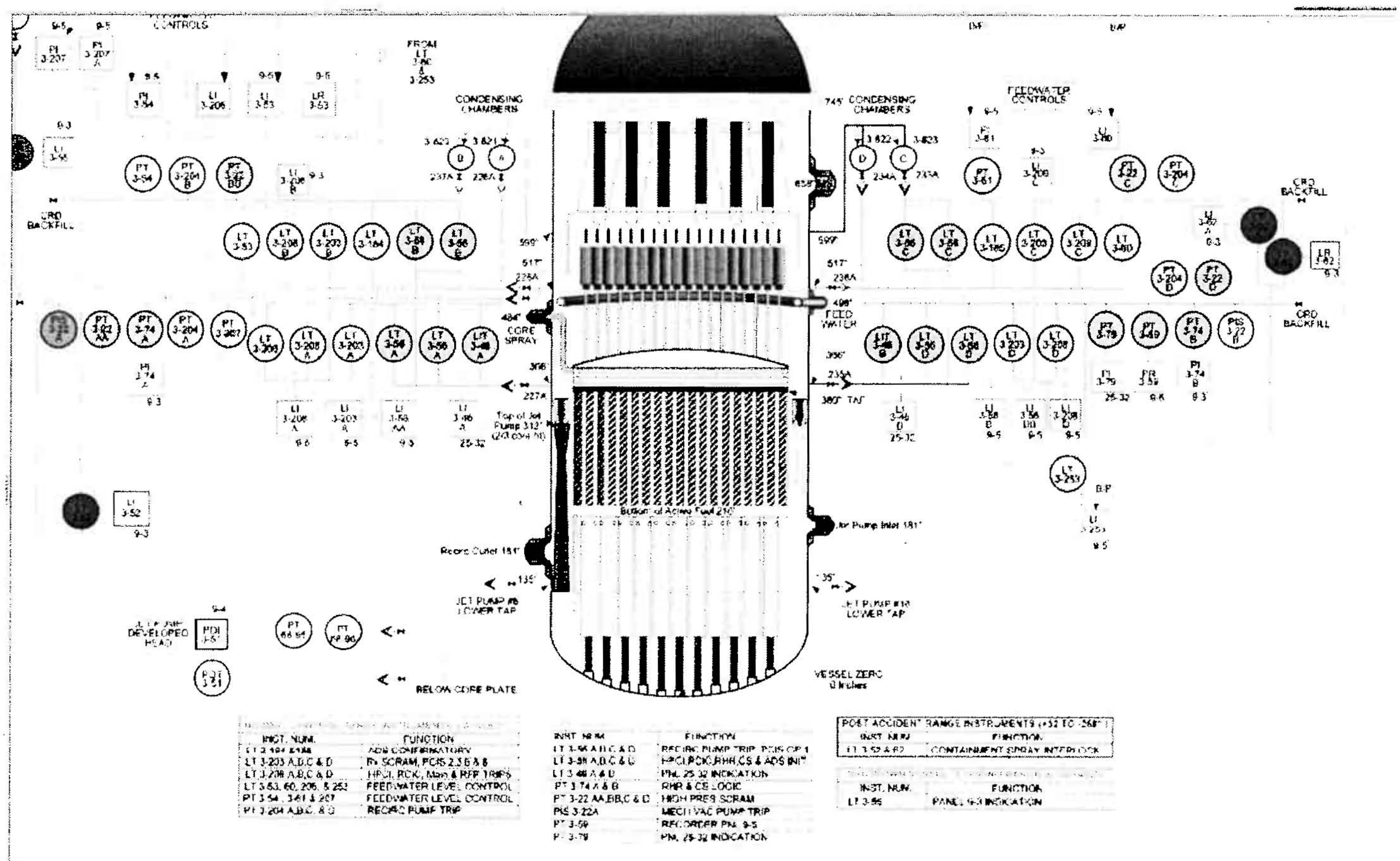
Table B 3.3.3.2-1 (Page 1 of 4)
Backup Control System Instrumentation and Controls

FUNCTION	NUMBER REQUIRED
Instrument Parameter	
1. Reactor Water Level Indication (1-LI-3-46A, -46B)	1
2. Reactor Pressure Indication (1-PI-3-79)	1
3. Suppression Pool Temperature Indication (1-TI-64-55B)	1
4. Suppression Pool Level Indication (1-LI-64-54B)	1
5. Drywell Pressure Indication (1-PI-64-50)	1
6. RHR Flow Indication (1-FI-74-79)	1
7. RCIC Flow Indication (1-FIC-71-36B)	1, note a
8. RCIC Turbine Speed Indication (1-SI-71-42B)	1
9. Drywell Temperature Indication (1-TI-64-52AA)	1
10. RHRSW Header Pressure (0-PI-23-58/2, -59/2)	2

BASES

2. Reactor Vessel Water Level
(LI-3-52, LI-3-62A, LI-3-58A, and LI-3-58B)

Reactor vessel water level is a Category 1 variable provided to support monitoring of core cooling and to verify operation of the ECCS. Two different range water level channels (Emergency Systems and Post-accident Flood Range) provide the PAM Reactor Vessel Water Level Functions. The water level channels measure from 1/3 of the core height to 221 inches above the top of the active fuel. Water level is measured by two independent differential pressure transmitters for each required channel. The output from these channels is indicated on two independent indicators, which is the primary indication used by the operator during an accident. Therefore, the PAM Specification deals specifically with this portion of the instrument channel.



TP-6 Reactor Vessel Level/Pressure Instrumentation

Examination Outline Cross-reference:

295018 Partial or Total Loss of CCW / 8

AA2.01 (10CFR 55.43.5 - SRO Only)

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER:

- Component temperatures

Level		SRO
Tier #		1
Group #		1
K/A #	295018AA2.01	
Importance Rating		3.4

Proposed Question: **# 78**

Unit 1 is operating at 100% Reactor Power when the following occur:

- At 1200 on 5/18/09, a QA auditor reports that Functional Surveillances for **ALL** Division 1 OPRM Channels have not been performed for 40 months. The required Surveillance frequency per TS 3.3.1.1 is 24 months.
- At 1230 on 5/18/09, RECIRC PUMP 1A COOLING WATER FLOW LOW, (1-9-4A, Window 34) alarm is received.
- At 1245 on 5/18/09, the Unit Operator then reports that Recirc Pump Motor 1A-Seal No. 1 Cavity temperature is 210 °F **AND** rising.
- Drywell Temperatures remain stable at normal values.

Based on the above conditions, which ONE of the following describes the required actions to implement?

Immediately __**(1)**__ **AND** enter __**(2)**__.

A. **(1)** shut down Recirculation Pump 1A
(2) 1-AOI-68-1B, "Recirc Pump Trip/Core Flow Decrease."

B. **(1)** insert a Core Flow Runback
(2) 1-AOI-68-1B, "Recirc Pump Trip/Core Flow Decrease."

C. **(1)** shut down Recirculation Pump 1A
(2) 1-AOI-68-1A, "Recirc Pump Trip/Core Flow Decrease OPRMs Operable."

D. **(1)** insert a Core Flow Runback
(2) 1-AOI-68-1A, "Recirc Pump Trip/Core Flow Decrease OPRMs Operable."

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 correct – See explanation C. Part 2 incorrect as detailed in Explanation C.

B INCORRECT: Part 1 incorrect – RBBCW supplies cooling water to the pump. 1-AOI-70-1, Loss of RBCCW, directs tripping the Recirc Pump if ≥ 180°F. 1-9-4A Window 34 directs tripping the Recirc Pump if ≥ 200°F. In either case, the action setpoint has been met, requiring the tripping of the pump. If temp limits were exceeded on both Recirc Pump, this would be the correct action followed by Reactor Scram. Part 2 is incorrect as detailed in Explanation C.

C **CORRECT:** Part 1 correct - RBBCW supplies cooling water to the pump. 1-AOI-70-1, Loss of RBCCW, directs tripping the Recirc Pump if ≥ 180°F. 1-9-4A Window 34 directs tripping the Recirc Pump if ≥ 200°F. In either case, the action setpoint has been met, requiring the tripping of the pump. Part 2 correct- Candidate must recognize that although Surveillance has not been performed within its required completion time, per TS SR 3.0.4, if it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. It has been less than 24 hours so Division 1 OPRM Channels are Operable.

D INCORRECT: Part 1 is incorrect as explained above. Part 2 is correct as explained above.

Technical Reference(s): 1-AOI-70-1 Rev 9 / 1-AOI-68-1A Rev 2 (Attach if not previously provided)
1-9-4A Rev16 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 X

Comments: Reviewed 1-9-4A Rev17 issued 3/25/09. New revision has no impact on this question.

BFN Unit 1	Recirc Pump Trip/Core Flow Decrease OPRMs Operable	1-AOI-68-1A Rev. 0002 Page 3 of 12

1.0 PURPOSE

This instruction provides the symptoms, automatic actions, and operator actions for a core flow decrease or Reactor Recirc Pump trip in one or two loops with OPRMs operable.

BFN Unit 1	Panel 9-4 1-XA-55-4A	1-ARP-9-4A Rev. 0016 Page 46 of 47

RECIRC PUMP 1A COOLING WATER FLOW LOW 1-FA-68-53 34	Sensor/Trip Point: 1-FIS-068-0053 22 gpm

(Page 1 of 1)

Sensor Location: Recirculation Pump A area
Drywell

Probable Cause:
A. Partial or complete loss of RBCCW system.
B. High RBCCW flow through Fuel Pool Cooling heat exchangers.

Automatic Action: None

Operator Action:

A. **CHECK** following on RBCCW system:
- RBCCW PRI CNTMT OUTLET VALVE handswitch, 1-HS-70-47A (1-FCV-70-47) OPEN. ☐
- RBCCW PUMP 1A, 1-HS-70-5A in service. ☐
- RBCCW PUMP 1B, 1-HS-70-8A in service. ☐
- Surge tank level normal indicated by RBCCW SURGE TANK LEVEL LOW 1-LA-70-2B (Window 13 on 1-XA-55-4C) **NOT** illuminated. ☐

B. **REFER TO** 1-AOI-70-1. ☐

C. **IF** the 1-TE-68-61U(T), RECIRC PMP MTR 1A-SEAL NO. 1(2) CAVITY temperature on RECIRC PUMP MTR 1A & 1B WINDING AND BRG TEMP temperature recorder, 1-TR-68-71 on Panel 1-9-21 is ≥200°F, **THEN**
- **SHUT DOWN** Recirculation Pump 1B. ☐
- **REFER TO** 1-AOI-68-1A <u>or</u> 1-AOI-68-1B. ☐

D. **IF** CRD or RBCCW system is lost, **THEN** **REFER TO** 1-OI-68, Precautions and Limitations Sections to determine requirements for continued pump operation. ☐

1-45E620-5-1 1-47E610-68-1 FSAR Section 13.6.2

BFN Unit 1	Loss of Reactor Building Closed Cooling Water	1-AOI-70-1 Rev. 0009 Page 9 of 12

[11] **MONITOR** the following Recirc Pump and Motor 1A(1B) temperatures on 1-TR-68-71, Panel 1-9-21: ☐

- TE-68-61A(73A), RECIRC PMP MTR 1A(1B)-THR BRG UPPER FACE (<190°F)
- TE-68-61C(73C), RECIRC PMP MTR 1A(1B)-THR BRG LOWER FACE (<190°F)
- TE-68-61E(73E), RECIRC PMP MTR 1A(1B)-UPPER GUIDE BRG (<190°F)
- TE-68-61N(73N), RECIRC PMP MTR 1A(1B)-LOWER GUIDE BRG (<190°F)
- TE-68-61G,J,L(73G,J,L), RECIRC PMP MTR 1A(1B)-MOTOR WINDING A,B,C (<255°F)
- TE-68-61T(73T), RECIRC PMP MTR 1A(1B)-NO. 2 CAVITY (<180°F)
- TE-68-61U(73U), RECIRC PMP MTR 1A(1B)-NO. 1 SEAL CAVITY (<180°F)
- TE-68-54(67), RECIRC PMP MTR 1A(1B)-CLG WTR FROM SEAL CLG (<140°F)
- TE-68-57(70), RECIRC PMP MTR 1A(1B)-CLG WTR FROM BRG (<140°F)

[12] **IF** any of the above temperature limits are exceeded on either Recirc Pump, **THEN**

SHUT DOWN the affected Recirc Pump as follows and **REFER TO** 1-AOI-68-1A(1B): (otherwise N/A)

- **DEPRESS** RECIRC DRIVE 1A SHUTDOWN, 1-HS-96-19. ☐
- **DEPRESS** RECIRC DRIVE 1B SHUTDOWN, 1-HS-96-20. ☐

[13] **IF** any of the above temperature limits are exceeded on both Recirc Pumps, **THEN**

PERFORM the following (otherwise N/A):

[13.1] **IF** core flow is above 60%, **THEN**

REDUCE core flow to between 50-60%. ☐

[13.2] **MANUALLY SCRAM** the Reactor and **PLACE** Mode Switch in SHUTDOWN. (**REFER TO** 1-AOI-100-1) ☐

[13.3] **SHUT DOWN** both Recirc Pumps as follows:

- **DEPRESS** RECIRC DRIVE 1A SHUTDOWN, 1-HS-96-19. ☐
- **DEPRESS** RECIRC DRIVE 1B SHUTDOWN,

Examination Outline Cross-reference:	Level		SRO
295023 Refueling Acc Cooling Mode / 8	Tier #		1
AA2.01 (10CFR 55.43.5 - SRO Only)	Group #		1
Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS:	K/A #		295023AA2.01
• Area radiation levels	Importance Rating		4.0

Proposed Question: **# 79**

During refueling operations on Unit 2, an irradiated fuel bundle jams on the Upper Core Guide Plate and the Main Hoist Fuel Grapple fails open. The bundle then falls the entire way into the core location.

The following conditions are subsequently noted:

- Gas bubbles are seen coming from the fuel bundle
- REFUEL ZONE EXHAUST RADIATION HIGH, (2-9-3A, Window 34), is alarming and reading 120 mr/hr
- FUEL POOL FLOOR AREA RADIATION HIGH, (2-9-3A, Window 1), is alarming and reading 100 mr/hr

Which ONE of the following completes the statement?

Based on the above conditions, Secondary Containment must be verified intact as directed by__**(1)**__ . This event would be classified as __**(2)**__ per EPIP-1, "Emergency Classification Procedure."

[REFERENCE PROVIDED]

A. **(1)** 0-EOI-4, "Radioactivity Release Control,"
(2) an Alert

B. **(1)** 0-EOI-4, "Radioactivity Release Control,"
(2) a Site Area Emergency

C. **(1)** 2-AOI-79-1, "Fuel Damage During Refueling,"
(2) an Alert

D. **(1)** 2-AOI-79-1, "Fuel Damage During Refueling,"
(2) a Site Area Emergency

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 is incorrect; The direction to verify Secondary CTMT intact is in 2-AOI-79-1, "Fuel Damage During Refueling". Part 2 is correct; with REFUEL ZONE EXHAUST RADIATION HIGH, (2-9-3A, Window 34),and FUEL POOL FLOOR AREA RADIATION HIGH, (2-9-3A, Window 1) above their alarm set points, and confirmation that fuel damage occurred, an Alert must be declared per EAL 3.2-A

B INCORRECT: Part 1 is incorrect as detailed above. Part 2 is incorrect; area rad levels would have to be above Max Safe levels for a Site Area Emergency.

C **CORRECT:** Part 1 is correct; per 2-AOI-79-1, "Fuel Damage During Refueling," subsequent actions, operating crew must verify Secondary CTMT intact and refer to TS 3.6.4.1. Part 2 is correct; with REFUEL ZONE EXHAUST RADIATION HIGH, (2-9-3A, Window 34),and FUEL POOL FLOOR AREA RADIATION HIGH, (2-9-3A, Window 1) above their alarm set points, and confirmation that fuel damage occurred, an Alert must be declared per EAL 3.2-A

D INCORRECT: Part 1 is correct and Part 2 incorrect as detailed above.

Technical Reference(s): 2-AOI-79-1 Rev 17 (Attach if not previously provided)
EPIP-1 Rev 44
2-9-3A Window 1 rev 37 (Including version / revision number)
2-9-3A Window 34 rev 37

Proposed references to be provided to applicants during examination: **EPIP-1 Event Classification Matrix**

Learning Objective: ______ (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New **X**

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 **X**

Comments:

BFN Unit 2	Fuel Damage During Refueling	2-AOI-79-1 Rev. 0017 Page 5 of 7

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **STOP** all fuel handling. ☐

[2] **EVACUATE** all non-essential personnel from Refuel Floor. ☐

4.2 Subsequent Actions

CAUTION

The release of iodine is of major concern. If gas bubbles are identified at any time, Iodine release should be assumed until RADCON determines otherwise.

[1] **VERIFY** secondary containment is intact. (**REFER TO** Tech Spec 3.6.4.1) ☐

[2] **IF** any EOI entry condition is met, **THEN**

ENTER the appropriate EOI(s). ☐

[3] **VERIFY** automatic actions. ☐

[6] **MONITOR** radiation levels, for the affected areas, using the following radiation recorders and indicators:

A. 2-RR-90-1 (points 1 and 2), 2-MON-90-50 (Address 11), 2-RR-90-142 and 2-RR-90-140 (Panel 2-9-2). ☐

B. 2-RM-90-142, 2-RM-90-140, 2-RM-90-143 and 2-RM-90-141 Detectors A and B (Panel 2-9-10). ☐

C. 2-RI-90-1A and 2-RI-90-2A (Panel 2-9-11). ☐

D. 0-CONS-90-362A (Address 09, 10, 08) for Unit 1, 2, 3-RM-90-250, respectively (Panel 1-9-44). ☐

[7] **IF** possible, **MONITOR** portable CAMs & ARMs.

BROWNS FERRY	EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX	EPIP-1

SECONDARY CONTAINMENT RADIATION

Description

UNUSUAL EVENT

ALERT

3.2-A				

Any of the following high radiation alarms on Panel 9-3:

- RA-90-1A, Fuel Pool Floor Alarm
- RA-90-250A, Reactor, Turbine, Refuel Exhaust
- RA-90-142A, Reactor Refuel Exhaust
- RA-90-140A, Refueling Zone Exhaust

AND

Confirmation by Refuel Floor personnel that irradiated fuel damage may have occurred.

OPERATING CONDITION:
ALL

SITE EMERGENCY

3.2-S			TABLE	

An unisolable Primary System leak is discharging into Secondary Containment

AND

Any area radiation level at or above the Maximum Safe Operating Area radiation limit listed in Table 3.2.

OPERATING CONDITION:
Mode 1 or 2 or 3

GENERAL EMERGENCY

3.2-G			TABLE	US

An unisolable Primary System leak is discharging into Secondary Containment

AND

Any area radiation level at or above the Maximum Safe Operating Area radiation limit listed in Table 3.2.

AND

Any indication of potential or significant fuel cladding failure exists. Refer to Table 3.1-G/3.2-G with RCS Barrier intact inside Primary Containment.

OPERATING CONDITION
Mode 1 or 2 or 3

BROWNS FERRY	EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX	EPIP-1

NOTES

CURVES/TABLES:

TABLE 3.2 MAXIMUM SAFE OPERATING AREA RADIATION LIMITS		
AREA	RAD MONITOR	MAX SAFE VALUE MR/HR
RHR West Room	90-25A	1000
RHR East Room	90-28A	1000
HPCI Room	90-24A	1000
CS/RCIC Room	90-26A	1000
Core Spray Room	90-27A	1000
Suppr Pool Area	90-29A	1000
CRD-HCU West Area	90-20A	1000
CRD-HCU East Area	90-21A	1000
TIP Drive Area	90-23A	1000
North RWCU System Area	90-13A	1000
South RWCU System Area	90-14A	1000
RWCU System Area	90-9A	1000
MG Set Area	90-4A	1000
Fuel Pool Area	90-1A	1000
Service Flr Area	90-2A	1000
New Fuel Storage	90-3A	1000

TABLE 3.1-G/3.2-G INDICATIONS OF POTENTIAL OR SIGNIFICANT FUEL CLADDING FAILURE WITH RCS BARRIER INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1 DRYWELL RADIATION		UNIT 2 DRYWELL RADIATION		UNIT 3 DRYWELL RADIATION	
1-RE-90-272A	≥ 196 R/HR	2-RE-90-272A	≥ 642 R/HR	3-RE-90-272A	≥ 196 R/HR
1-RE-90-273A	≥ 297 R/HR	2-RE-90-273A	≥ 297 R/HR	3-RE-90-273A	≥ 297 R/HR
Reactor Coolant Activity ≥ 300 µCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity ≥ 300 µCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity ≥ 300 µCi/gm Dose Equivalent Iodine 131	

BFN Unit 2	Panel 9-3 2-XA-55-3A	2-ARP-9-3A Rev. 0037 Page 4 of 50

FUEL POOL
FLOOR AREA
RADIATION HIGH

2-RA-90-1A

1

(Page 1 of 1)

Sensor/Trip Point:

RI-90-1B
RI-90-2B — For setpoints
RI-90-3B — **REFER TO** 2-SIMI-90B.

Sensor Location:	RE-90-1B	El 664'	R-11 P-LINE
	RE-90-2B	El 664'	R-10 U-LINE
	RE-90-3B	El 639'	R-10 Q-LINE

Probable Cause:
A. Change in general radiation levels.
B. Refueling accident.
C. Sensor malfunction.

Operator Action:

A. **CHECK** 2-RI-90-1A, 2-RI-90-2A and 2-RI-90-3A on Panel 2-9-11 ☐
B. **NOTIFY** refuel floor personnel. ☐
C. **IF** Dry Cask loading/unloading activities are in progress, **THEN** **NOTIFY** Cask Supervisor. ☐
D. **IF** airborne levels rise by 100 DAC **AND** RAD PRO confirms, **THEN** **REFER TO** EPIP-1. ☐
E. **REFER TO** 2-AOI-79-1 or 2-AOI-79-2 as applicable. ☐
F. **IF** this alarm is not valid, **THEN REFER TO** 0-OI-55. ☐
G. **IF** this alarm is valid, **THEN** **MONITOR** the other parameters that input to it frequently. These other parameters will be masked from alarming while this alarm is sealed in. ☐
H. **ENTER** 2-EOI-3 Flowchart. ☐

References: 0-47E600-13 2-47E610-90-1 2-45E620-3
GE 730E356 Series, TVA Calc NDQ00902005001/EDC63693

BFN Unit 2	Panel 9-3 2-XA-55-3A	2-ARP-9-3A Rev. 0037 Page 47 of 50

REFUELING ZONE EXHAUST RADIATION HIGH
2-RA-90-140A
34

(Page 1 of 2)

Sensor/Trip Point:

2-RE-90-140A	72 MR/HR	
2-RE-90-140B	72 MR/HR	
2-RE-90-141A	72 MR/HR	Required setting of
2-RE-90-141B	72 MR/HR	≤ 100 MR/HR.

Sensor Location: Rx Bldg, El 664' (Refuel Floor), R-10 P-LINE

Probable Cause:
A. Radiation levels have risen above alarm setpoint.
B. Refueling accident.

NOTE
TVA Calc NDQ0090205008 requires these detectors be temporarily shielded during Dry Cask loading/unloading activities.

C. Temporary shielding not in place for monitors during Dry Cask loading/unloading activities.
D. Loss of power to NUMAC drawer.

Automatic Action:
A. Control Room and Refuel Zone ventilation isolates.
B. SGTS initiates.
C. Control Room emergency pressurization units start.

Operator Action:
A. **VERIFY** alarm condition on the following:
1. REACTOR ZONE EXHAUST RADIATION recorder. 2-RR-90-140 on Panel 2-9-2. ☐
2. RX & REFUEL ZONE EXH CH A RAD MON RTMR radiation monitor. 2-RM-90-140/142 on Panel 2-9-10 ☐
3. RX & REFUEL ZONE EXH CH BRAD MON RTMR radiation monitor 2-RM-90-141/143 on Panel 2-9-10 ☐

B. **IF** Dry Cask loading/unloading activities are in progress, **THEN NOTIFY** the Cask Supervisor to place the MPC in a safe condition using MSI-0-079-DCS037 or as directed by RAD PRO. ☐
C. **NOTIFY** Shift Manager, Unit 1 and Unit 3 ☐
D. **IF** the TSC is **NOT** manned, **THEN EVACUATE** personnel from the refuel floor. ☐

E. **IF** the TSC is manned, **THEN NOTIFY** the TSC to evacuate non-essential personnel from affected areas. ☐
F. **ENTER** 2-EOI-3 Flowchart. ☐
G. **REFER TO** 2-AOI-64-2d and, for loss of power to NUMAC drawer, to 2-OI-90, Section 6.0. ☐

H. **REFER TO** 2-AOI-79-1 or 2-AOI-79-2 as applicable. ☐
I. **REFER TO** EPIP-1. ☐
J. **REFER TO** Technical Specification Section 3.3.6.2 and 3.3.7.1 ☐

References:
2-45E620-3 | 2-47E610-90-1 | GE 2-730E927-21
Technical Specifications 3.3.6.2 and 3.3.7.1 | TVA Calc NDQ0090200500 1/EDC63693

Examination Outline Cross-reference:

295025 High Reactor Pressure / 3

EA2.01 (10CFR 55.43.5 - SRO Only)

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE:

- Reactor pressure

Level	[redacted]	SRO
Tier #		1
Group #		1
K/A #	295025EA2.01	
Importance Rating	[redacted]	4.3

Proposed Question: **# 80**

Following a Scram on Unit 2, 2-EOI-1, "RPV Control," 2-EOI-2, "Primary Containment Control," and 2-C-5, "Level/Power Control," are being executed. Although direction had been given, an attempt to establish a level band for 2-C-5 resulted in Reactor Water Level dropping to (-) 125 inches before **ANY** outside action could be completed. Additionally, the following plant conditions exist:

- Four SRVs are OPEN at their setpoint(s)
- Reactor Water Level is currently being controlled at (-) 75 inches
- There is **NO** indication of a steam line break
- Standby Liquid Control (SLC) is injecting with tank level at 83%
- Suppression Pool Level is 17 feet and stable
- You receive a crew update that 2-EOI-Appendix 8A, "Bypassing Group I RPV Low Low Low Level Isolation Interlocks," has just been completed

Which ONE of the following completes the statement?

Based on the above conditions, the correct course of action is to _______.

[REFERENCE PROVIDED]

A. exit the RC/P leg of 2-EOI-1, "RPV Control," **AND** enter into 2-C-2, "Emergency RPV Depressurization."

B. perform the actions necessary to "Anticipate Emergency Depressurization," per the RC/P Override in 2-EOI-1, "RPV Control."

C. execute 2-EOI-Appendix 8B, "Reopening MSIVs Following a Group I Isolation," per the Override in 2-EOI-1, "RPV Control."

D. augment Pressure Control with RCIC **AND** RWCU per 2-EOI-Appendix 11B **AND** 11E, "Alternate RPV Pressure Control Systems (RCIC/RWCU)."

Proposed Answer: **A**

Explanation (Optional):

A CORRECT: With four SRVs lifting at their setpoint, it can be deduced that Reactor Pressure is 1135 psig. Given a corresponding Suppression Pool Level of 17 feet and the curve for SRV Tail Pipe Limit, the candidate can determine that they are in the ACTION REQUIRED portion of the curve. Thus, Emergency Depressurization is required for these conditions. Portions of C-5 will be applicable for the ED, but stem already indicates that candidate is executing C-5. Complexity is compounded by not having the EOI flowchart and the override which is related to the Curve 4, which only says to reduce pressure irrespective of cooldown rate.

B INCORRECT: "Anticipating Emergency Depressurization" is rapidly depressurizing using Turbine Bypass Valves irrespective of cooldown rate. Stem conditions indicate that we dropped below MSIV Isolation setpoint on Low Reactor Water Level before Appendix 8A was completed. Therefore, MSIVs are closed. Additionally, "Anticipating" is not allowed while in C-5.

C INCORRECT: Although this action would be prudent to pursue, the aforementioned SRV Tail Pipe Limit ACTION REQUIRED dictates otherwise; in that an ED is necessary.

D INCORRECT: Augmenting pressure control might also be a prudent choice. But, because SLC is injecting, RWCU is isolated and not available to augment Pressure Control. RCIC would have an initiation signal present.

Technical Reference(s): 2-EOI-1, Rev. 12 (Attach if not previously provided)

2-EOI App. 8A / 11B / 11E Rev 3 / 5 / 4 (Including version / revision number)

Proposed references to be provided to applicants during examination: **2-EOI-1, Curve 4 SRV Tail Pipe**

Learning Objective: ________________ (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

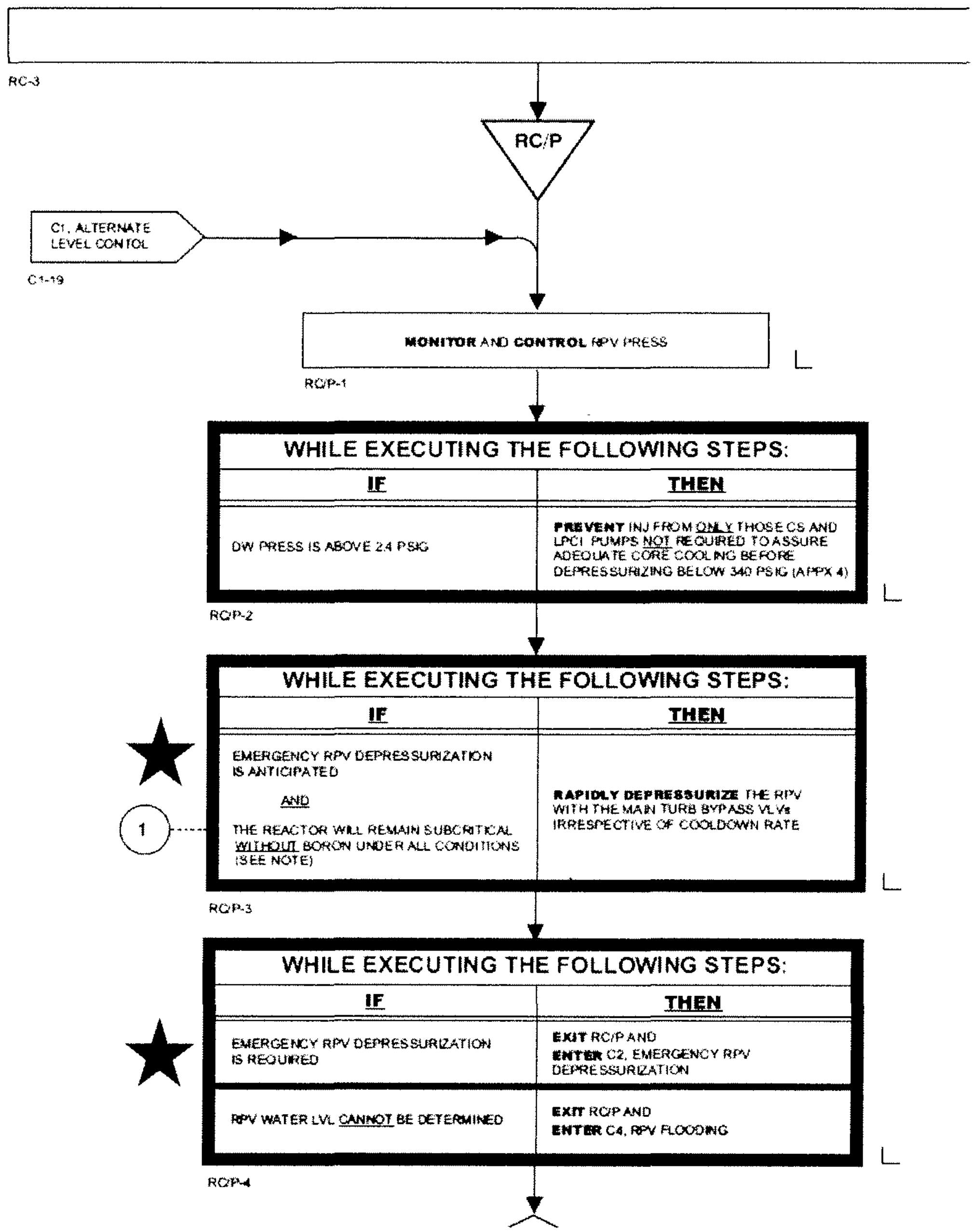
Comprehension or Analysis X

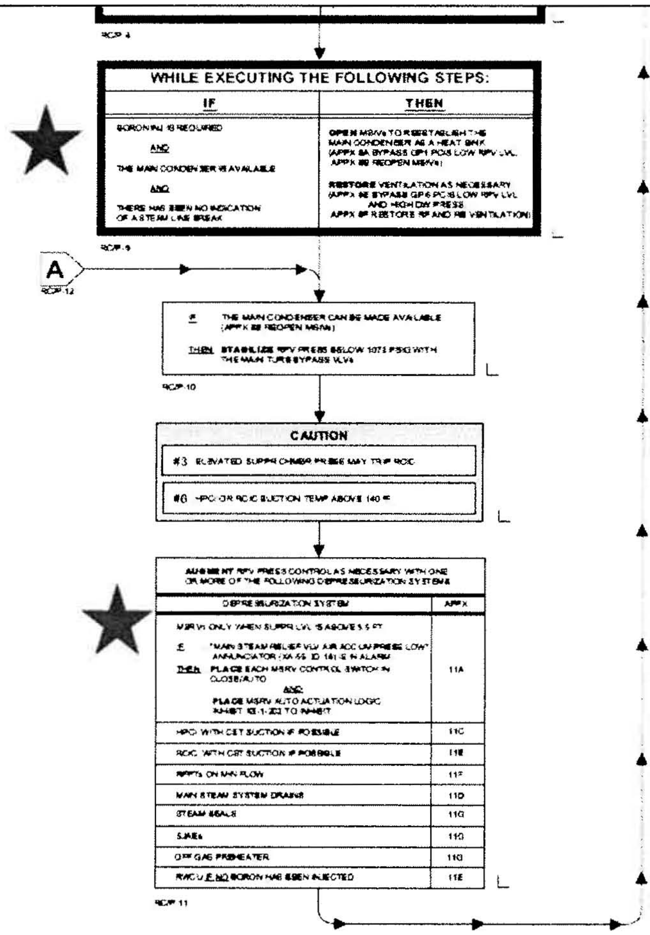
10 CFR Part 55 Content: 55.41

55.43 X

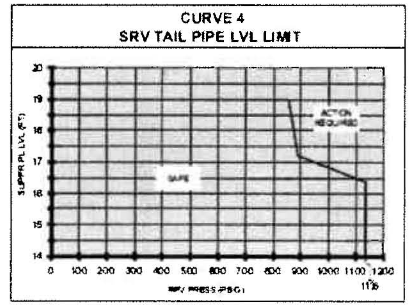
Comments:

2-EOI-1


RC-3
RC/P
C1, ALTERNATE LEVEL CONTOL
C1-19
MONITOR AND CONTROL RPV PRESS
RC/P-1
WHILE EXECUTING THE FOLLOWING STEPS:
IF
THEN
DW PRESS IS ABOVE 2.4 PSIG
PREVENT INJ FROM ONLY THOSE CS AND LPCI PUMPS NOT REQUIRED TO ASSURE ADEQUATE CORE COOLING BEFORE DEPRESSURIZING BELOW 340 PSIG (APPX 4)
RC/P-2
WHILE EXECUTING THE FOLLOWING STEPS:
IF
THEN
EMERGENCY RPV DEPRESSURIZATION IS ANTICIPATED
AND
THE REACTOR WILL REMAIN SUBCRITICAL WITHOUT BORON UNDER ALL CONDITIONS (SEE NOTE)
RAPIDLY DEPRESSURIZE THE RPV WITH THE MAIN TURB BYPASS VLVs IRRESPECTIVE OF COOLDOWN RATE
1
RC/P-3
WHILE EXECUTING THE FOLLOWING STEPS:
IF
THEN
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
EXIT RC/P AND
ENTER C2, EMERGENCY RPV DEPRESSURIZATION
RPV WATER LVL CANNOT BE DETERMINED
EXIT RC/P AND
ENTER C4, RPV FLOODING
RC/P-4



EOI-1



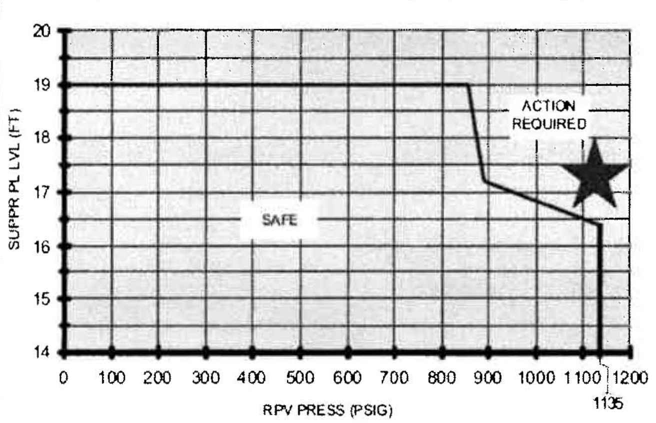
CAUTIONS

CAUTION #1

- AN RPV WATER LVL INSTRUMENT MAY BE USED TO DETERMINE OR TREND LVL ONLY WHEN IT READ[S] 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 (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 RUN TI (FROM TA
LI-3-58A, B	EMERGENCY -155 TO +60	ON SCALE	N/A	BELOW
		-145	N/A	151 TO
		-140	N/A	201 TO
		-130	N/A	251 TO
		-120	N/A	301 TO
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
		+5	N/A	151 TO
		+15	N/A	201 TO
		+20	N/A	251 TO
		+30	N/A	301 TO
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

CURVE 4
SRV TAIL PIPE LVL LIMIT



WHILE EXECUTING THE FOLLOWING STEPS:	
IF	**THEN**
STEAM COOLING IS REQUIRED	**EXIT** RC/P
2 SUPPR PL TEMP AND LVL CANNOT BE MAINTAINED IN A SAFE AREA OF CURVE 3 AT THE EXISTING RPV PRESS	**LOWER** RPV PRESS TO MAINTAIN SUPPR PL TEMP AND LVL IN A SAFE AREA OF CURVE 3, IRRESPECTIVE OF COOL DOWN RATE
SUPPR PL LVL CANNOT BE MAINTAINED IN THE SAFE AREA OF CURVE 4	**MAINTAIN** RPV PRESS IN THE SAFE AREA OF CURVE 4, IRRESPECTIVE OF COOL DOWN RATE

RC/P-7

Examination Outline Cross-reference:

295028 High Drywell Temperature / 5

EA2.01 (10CFR 55.43.5 - SRO Only)

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE:

- Drywell temperature

Level	SRO
Tier #	1
Group #	1
K/A #	295028EA2.01
Importance Rating	4.1

Proposed Question: **# 81**

Unit 1 is operating at 100% Reactor Power when a loss of Drywell Cooling occurs resulting in the following conditions:

- Drywell Temperature is 145 °F and rising
- Drywell Pressure is 1.4 psig and rising

Which ONE of the following completes the statement?

Based on the above conditions, Tech Spec BASES analysis assumptions indicate that if a Design Basis Accident LOCA were to occur at this moment, the design Drywell Temperature limit __**(1)**__ expected to be exceeded; **AND** if **ALL** other Structures, Systems, and Components function as designed, __**(2)**__ will be required.

A. **(1)** is **NOT**
(2) 1-EOI Appendix 17B "RHR System Operation Drywell Sprays," using only those pumps that are **NOT** required for adequate core cooling

B. **(1)** is
(2) 1-EOI Appendix 17B "RHR System Operation Drywell Sprays," using only those pumps that are **NOT** required for adequate core cooling

C. **(1)** is **NOT**
(2) venting the Drywell irrespective of offsite radioactivity release rates per 1-EOI Appendix 13, "Emergency Venting Primary Containment,"

D. **(1)** is
(2) venting the Drywell irrespective of offsite radioactivity release rates per 1-EOI Appendix 13, "Emergency Venting Primary Containment,"

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** Part 1 = correct, Tech Spec Bases (3.6.1.4) document for high drywell temperature states the design drywell temperature of 336 degrees cannot be ensured following a design basis LOCA if drywell temperature is not maintained below 150 degrees. With conditions starting at < 150 degrees, the initial conditions are met; therefore, the DW Temp limit is NOT expected to be exceeded. Part 2 = correct, EOI-2 step DW/T-10 requires using APP 17B to spray the Drywell using ONLY pumps not required for adequate core cooling. This would be achieved under DBA LOCA conditions.

B INCORRECT: Part 1 = incorrect, See above . Part 2 = correct, as explained above.

C INCORRECT: Part 1 = correct. Part 2 = incorrect, PSP is **not** exceeded, based on given conditions, following a DBA LOCA with a functional Torus and although trending up, Drywell Pressure is in normal range. Therefore, Emergency Venting would not be necessary.

D INCORRECT: Part 1 and 2 are incorrect.

Technical Reference(s): T.S.3.6.1.4 / 2.1 BASES, 1-EOI-2 Rev 0 (Attach if not previously provided)

FSAR 14.6.3, EOI APP. 13/17B Rev 0/0 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New X

Question History: Last NRC Exam ____

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

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).
LCO	In the event of a DBA, with an initial drywell average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the maximum allowable temperature. As a result, the ability of primary containment to perform its design function is ensured.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.1 Suppression Pool Average Temperature

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the decay heat and sensible energy released during a reactor blowdown from safety/relief valve discharges or from Design Basis Accidents (DBAs). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment that ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (62 psig). The suppression pool must also condense steam from steam exhaust lines in the turbine driven systems (i.e., the High Pressure Coolant Injection System and Reactor Core Isolation Cooling System). Suppression pool average temperature (along with LCO 3.6.2.2, "Suppression Pool Water Level") is a key indication of the capacity of the suppression pool to fulfill these requirements.

The technical concerns that lead to the development of suppression pool average temperature limits are as follows:

a. Complete steam condensation - the original limit for the end of a LOCA blowdown was 170°F, based on the Bodega Bay and Humboldt Bay Tests;

b. Primary containment peak pressure and temperature - design pressure is 56 psig and design temperature is 281°F (Ref. 1); and

c. Condensation oscillation loads - maximum allowable initial temperature is 110°F.

APPLICABLE SAFETY ANALYSES

The postulated DBA against which the primary containment performance is evaluated is the entire spectrum of postulated pipe breaks within the primary containment. Inputs to the safety analyses include initial suppression pool water volume and suppression pool temperature (Reference 1 for LOCAs and Reference 2 for the pool temperature analyses required by Reference 3). An initial pool temperature of 95°F is assumed for the Reference 1 and Reference 2 analyses. Reactor shutdown at a pool temperature of 110°F and vessel depressurization at a pool temperature of 120°F are assumed for the Reference 2 analyses. The limit of 105°F, at which testing is terminated, is not used in the safety analyses because DBAs are assumed to not initiate during unit testing.

Suppression pool average temperature satisfies Criteria 2 and 3 of the NRC Policy Statement (Ref. 5).

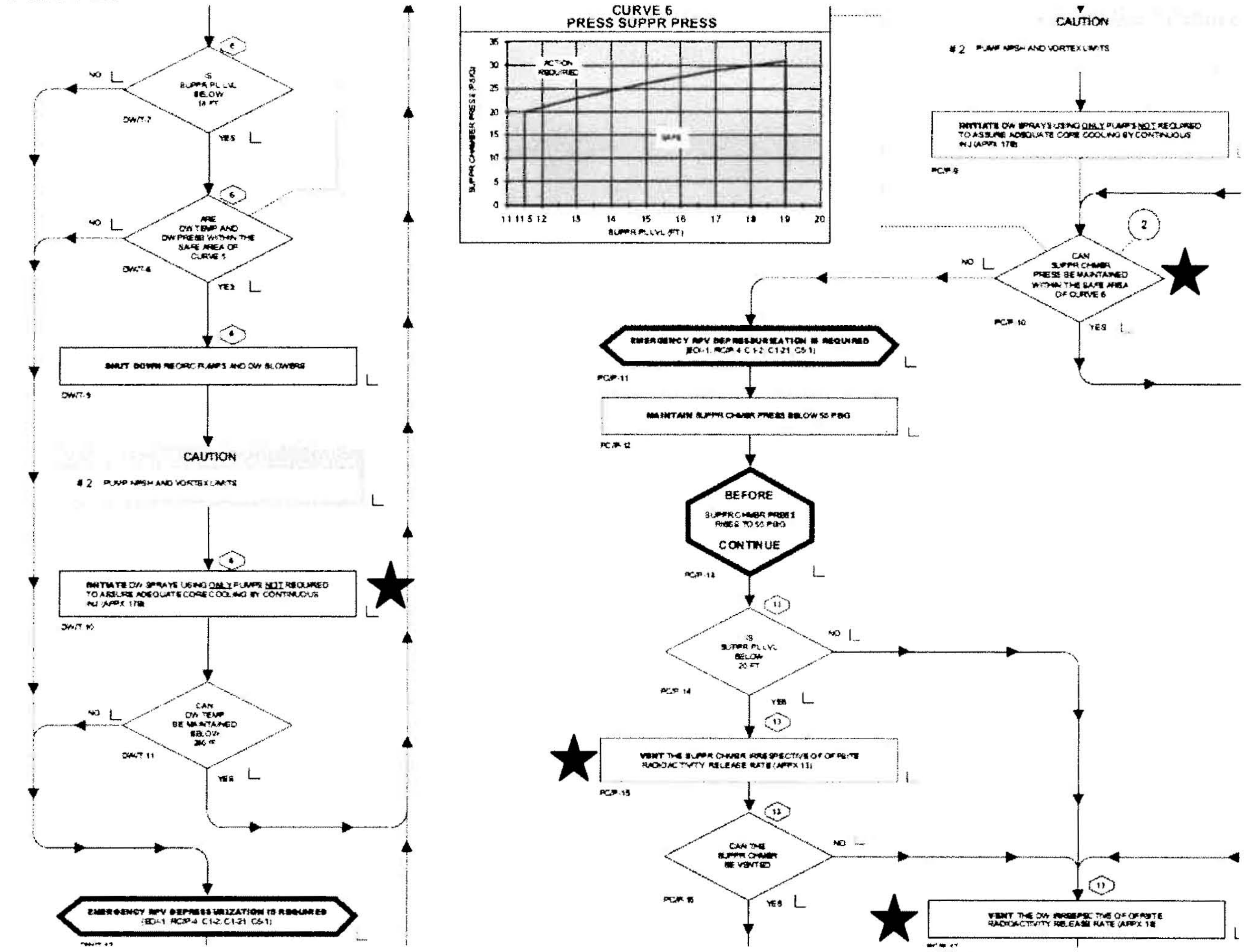
BFN-21

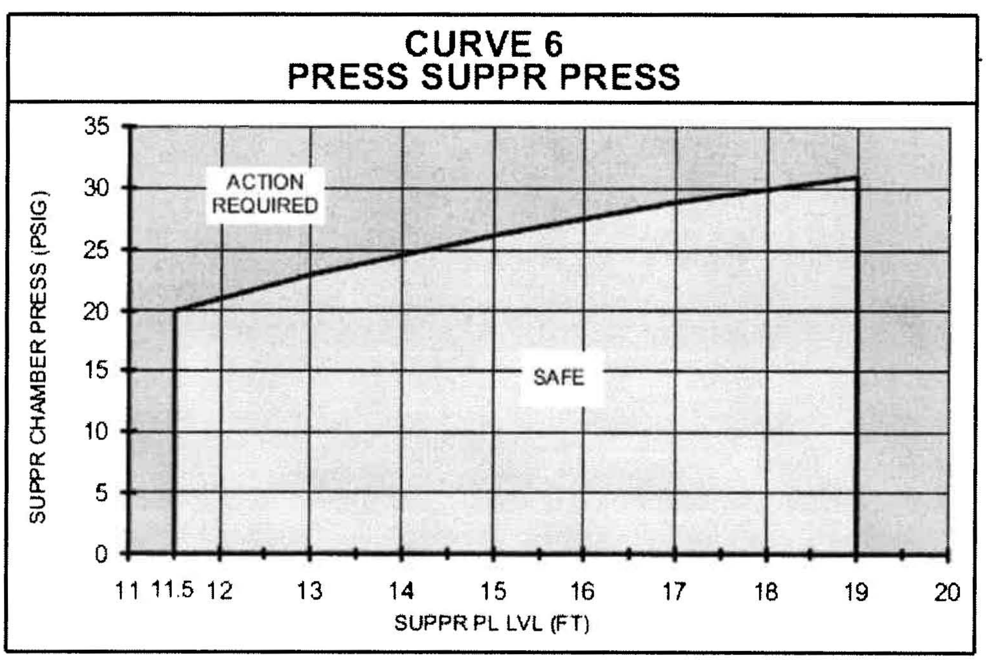
TABLE 14.6-3

SUMMARY OF POWER UPRATE INPUT PARAMETERS USED FOR ALL CONTAINMENT ANALYSES

Parameter	Unit	Analysis Value for Power Uprate
Core Thermal Power 102% of uprated power (3458 MWt)	MWt	3527
Initial Reactor Core Flow (100% rated)	Mlbm/hr	102.5
Vessel dome pressure At 102% of uprated power (3458 MWt)	psia	1053
Initial drywell pressure	psia	17.0/15.1[1]
Initial drywell temperature (Maximum value used to maximize the drywell temp. response)	°F	150
Initial drywell relative humidity (Minimum)	%	20
Initial wetwell pressure	psia	15.9/15.1[1]
Initial wetwell airspace temperature (Maximum)	°F	95
Initial wetwell airspace relative humidity (Maximum)	%	100
Initial pressure suppression pool temperature (Maximum value used to maximize the suppression pool temp. response)	°F	95

1-EOI-2:


CURVE 6
PRESS SUPPR PRESS
CAUTION
BEFORE
CONTINUE


CURVE 6
PRESS SUPPR PRESS
ACTION REQUIRED
SAFE
SUPPR CHAMBER PRESS (PSIG)
35
30
25
20
15
10
5
0
11
11.5
12
13
14
15
16
17
18
19
20
SUPPR PL LVL (FT)

Examination Outline Cross-reference:

700000 Generator Voltage and Electric Grid Disturbances / 6

G2.4.31 (10CFR 55.43.5 - SRO Only)

Knowledge of annunciator alarms, indications, or response procedures.

Level		SRO
Tier #		1
Group #		1
K/A #	700000G2.4.31	
Importance Rating		4.1

Proposed Question: **# 82**

Unit 1 is shut down for a Refueling Outage. Unit 2 is shut down for a Forced Outage. Unit 3 is operating at 100% Reactor Power. An electrical grid disturbance results in the following plant conditions:

- Loss of Limestone **AND** Trico 500kV offsite lines
- Grid Frequency is 59.9 Hz
- Grid Voltage is 514 kV
- Unit 3 remains on line

Which ONE of the following completes the statement?

Based on current conditions, the operating crew must __**(1)**__ Reactive Power (MVARS) as directed by __**(2)**__.

A. **(1)** lower
(2) 0-AOI-57-1B, "Loss of 500kV."

B. **(1)** raise
(2) 0-AOI-57-1B, "Loss of 500kV."

C. **(1)** lower
(2) 0-AOI-57-1E, "Grid Instability."

D. **(1)** raise
(2) 0-AOI-57-1E, "Grid Instability."

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 and 2 are incorrect. See Explanation D.

B INCORRECT: Part 1 = correct, As explained in D Part 2 = incorrect, although there is a partial loss of 500 kV, 0-AOI-57-1B, "Loss of 500kV" provides guidance for complete loss of 500 kV distribution.

C INCORRECT: Part 1 = incorrect, As explained in D. Plausible in that this would be correct action if system voltage was > 540 kV Part 2 = correct, as explained in D.

D **CORRECT:** Parts 1 and 2 are correct - Per 0-AOI-57-1E, "Grid Instability", if system voltage is less than 515 kV, raise reactive power until system voltage returns to 520 kV

Technical Reference(s): OPL171.036 Rev 11 (Attach if not previously provided)

0-AOI-57-1E Rev 7 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

BFN Unit 0	Grid Instability	0-AOI-57-1E Rev. 0007 Page 7 of 18

4.2 Subsequent Action (continued)

[6] **IF** grid instability is characterized by system voltage being maintained outside the normal limits of 525 + 5 KV, **THEN**

PERFORM the following steps:

[6.1] **IF** system voltage is greater than 540KV, **THEN**

[6.1.1] **LOWER** reactive power to system voltage returns to 530KV, **OR UNTIL** Generator Reactive power reaches -150 MVAR. ☐

[6.1.2] **CHECK** 161KV Cap Banks are Out of Service and **EVALUATE** conditions to determine appropriate actions. **REFER TO** 0-GOI-300-4. ☐

[6.2] **IF** system voltage is lower than 515KV, **THEN**

PERFORM the following:

[6.3] **RAISE** reactive power to system voltage returns to 520KV **OR UNTIL** Generator Reactive Power reaches +200 MVAR, ☐

[6.4] **CHECK** 161KV Cap Banks are In Service and **EVALUATE** conditions to determine appropriate actions. **REFER TO** 0-GOI-300-4. ☐

[6.5] **EVALUATE** as applicable, entry into Technical Specifications 3.8.1, 3.8.2, 3.8.7 and 3.8.8. ☐

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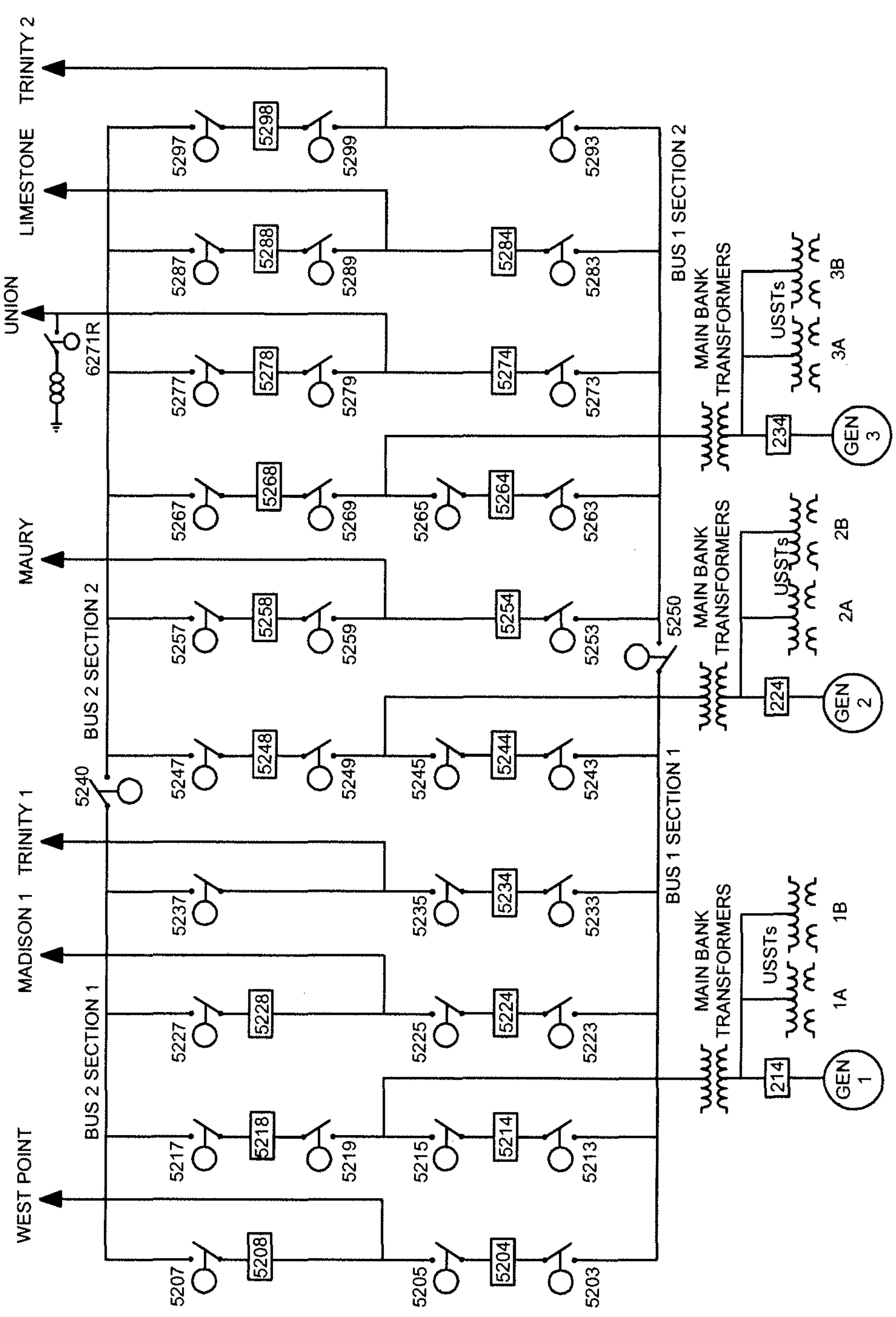
Appendix F Transparencies



Examination Outline Cross-reference:	Level	[redacted]	SRO
295012 High Drywell Temperature / 5	Tier #		1
G2.4.34 (10CFR 55.43.5 - SRO Only) Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	Group #		2
	K/A #	295012G2.4.34	
	Importance Rating	[redacted]	4.1

Proposed Question: **# 83**

A LOCA has occurred on Unit 1 with the following plant conditions:

- Reactor Level is being maintained with Feedwater at (+) 5 inches and stable
- Drywell Temperature is 245 °F and rising slowly
- Suppression Pool Level is 15 feet
- Suppression Chamber Pressure is 1.5 psig and rising slowly
- Suppression Pool Temperature is 125 °F
- Recirc Pumps and Drywell Coolers have been secured
- All except one Control Rod fully inserted
- Both RHR Loop I Pumps are tagged out
- RHR SYS II DW SPRAY OUTBD VLV, 1-FCV-74-74 breaker is tripped and will not reset

Based on the conditions identified, which ONE of the following identifies the next direction to be given by the Unit Supervisor?

A. Initiate Standby Liquid Control per 1-EOI Appendix 3A, "SLC Injection."

B. Inhibit ADS and direct operator to perform 1-EOI Appendix 5C, "Injection System Lineup RCIC."

C. Dispatch operators to line up, then initiate Suppression Chamber Spray on RHR Loop I using Fire Protection per 2-EOI Appendix 17C, "RHR System Operation Suppression Chamber Sprays."

D. Dispatch a Reactor Operator to manually open RHR SYS II DW SPRAY OUTBD VLV, 1-FCV-74-74, and initiate Drywell Spray per 2-EOI Appendix 17B, "RHR System Operation Drywell Sprays."

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: SLC Injection is required if Reactor will not be assured of staying subcritical under all conditions without Boron and before SP Temp 110° F. Also, SLC is not needed for level control with level (+) 5 inches and stable and all high pressure injection available. Plausible in that one control rod did not insert and SP Temp is above 110° F.

B INCORRECT: Inhibiting ADS is the required action if level drops below (-) 120 inches or ADS timer has initiated or if C-1,"Alternate Level Control". There is no indication that level has dropped below (-) 120 inches and conditions are not met to start ADS Timer. Also, there would be no reason to enter C-1,"Alternate Level Control" with level stable and all high pressure injection systems available. Plausible in that this is a normally exercised step in response to LOCA conditions and performing Appendix 5C would be appropriate action to ensure RCIC available if Feedwater lost.

C INCORRECT: 1-EOI-2 requires initiation of Suppression Chamber Spray before 12 psig. Although pressure is rising and expected to continue to rise, with Suppression Chamber Pressure at 1.5 psig it would be inappropriate to initiate Suppression Chamber Spray. Plausible in that pressure is rising and expected to continue to rise with LOCA.

D **CORRECT:** Per 1-EOI-2, before Drywell Temperature rises to 280° F and with SP Level below 18 feet, Initiate Drywell Spray using pumps not required for adequate core cooling. With the Drywell Temp rise being slow, the Loop II Drywell Spray Valve being readily accessible to manually open and adequate core cooling assured with other systems, this is the required action.

Technical Reference(s): 1-EOI-2 Rev 0 (Attach if not previously provided)

(Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

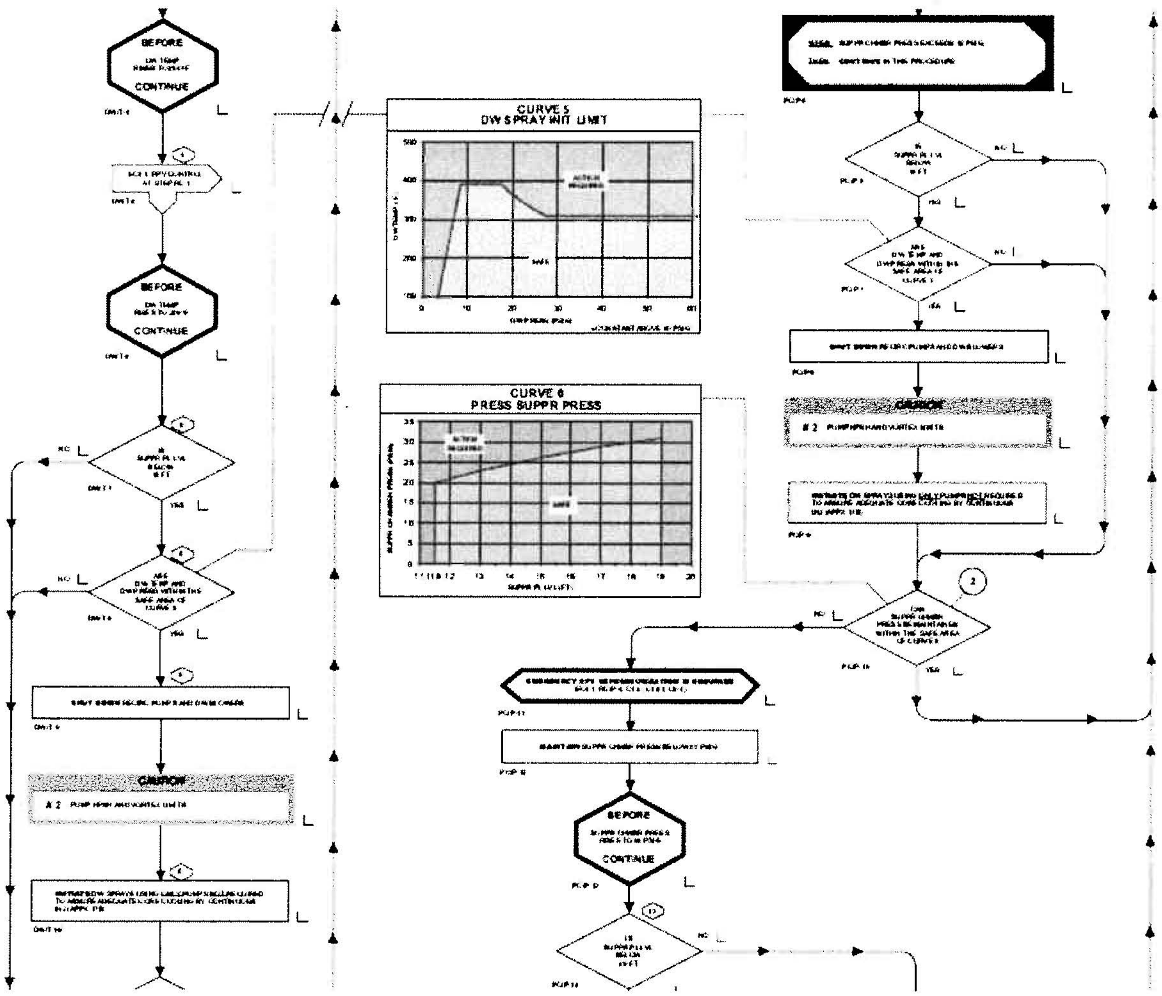
Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:



1-EOI-2 PAGE 1 OF 1
PRIMARY CONTAINMENT CONTROL
UNIT 1 BROWNS FERRY NUCLEAR PLANT
REV 0

Examination Outline Cross-reference:	Level	RO	SRO
295013 High Suppression Pool Temp. / 5	Tier #	------	1
AA2.02 (10 CFR 55.43.5 - SRO Only)	Group #	------	2
Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE:	K/A #	295013AA2.02	
• Localized heating/stratification	Importance Rating	------	3.5

Proposed Question: **# 84**

Unit 2 is operating at 100% Reactor Power, when MSRV, 2-PCV-1-4, inadvertently OPENS. The following conditions exist:

- Suppression Pool Bulk Average Temperature is 89 °F
- Suppression Pool Single Element, “Bay 2, TE-64-162A,” is 96 °F
- Suppression Pool Level is 15 feet
- 2-AOI-1-1, “Relief Valve Stuck Open,” has been entered

Which ONE of the following completes the statement?

If the MSRV were unable to be closed in an expeditious manner **AND** Suppression Pool (SP) Cooling was delayed in being placed in service, there would be concern for __**(1)**__ **AND** direction to place SP Cooling into service per __**(2)**__shall be given.

A. **(1)** damage to the coating on the MSRV Tailpipe
(2) 2-OI-74, “Residual Heat Removal System,”

B. **(1)** free release of steam to the Torus air space
(2) 2-OI-74, “Residual Heat Removal System,”

C. **(1)** damage to the coating on the MSRV Tailpipe
(2) 2-EOI APPENDIX-17A, “RHR System Operation Suppression Pool Cooling,”

D. **(1)** free release of steam to the Torus air space
(2) 2-EOI APPENDIX-17A, “RHR System Operation Suppression Pool Cooling,”

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Part 1 = incorrect, the MSRVs have no coating that would be of concern, this applies to the coating of the Suppression Pool inner lining. Part 2 = correct, The correct procedure is addressed, based on NO EOI entry condition exists (average is < 95° F).

B **CORRECT:** Part 1 = correct, localized heating could cause steam to be generated, thus impacting the torus air space. Part 2 = correct, The correct procedure is addressed, based on NO EOI entry condition exists (average is < 95° F).

C INCORRECT: Part 1 = incorrect, the MSRVs have no coating that would be of concern, this applies to the coating of the Suppression Pool inner lining. Part 2 = incorrect, NO EOI conditions exist, (based on average temp not single point) AOI / OI actions apply. Plausible in that the candidate may confuse what constitutes EOI entry.

D INCORRECT: Part 1 = correct, localized heating could cause steam to be generated, thus impacting the torus air space. Part 2 = incorrect, NO EOI conditions exist, (based on average temp not single point) AOI / OI actions apply. Plausible in that the candidate may confuse what constitutes EOI entry.

Technical Reference(s): OPL171.009 Rev 10, 2-OI-74 Rev 141

U2 TS 3.6.2.1, EOI Program Manual 0-V-D Rev 0, (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

4. Relief valves Operation:

   a. Operators should alternate relief valves in the recommended sequence to heat the suppression pool evenly.

   b. Failure to alternate use may result in suppression pool localized overheating, potentially causing:

      (1) Damage the coating on the inner surface of the suppression pool.

      (2) Possible release free steam to the torus, causing pressurization of the air space.

3.6 CONTAINMENT SYSTEMS

3.6.2.1 Suppression Pool Average Temperature

LCO 3.6.2.1 Suppression pool average temperature shall be:

a. ≤ 95°F when any OPERABLE intermediate range monitor (IRM) channel is > 70/125 divisions of full scale on Range 7 and no testing that adds heat to the suppression pool is being performed;

b. ≤ 105°F when any OPERABLE IRM channel is > 70/125 divisions of full scale on Range 7 and testing that adds heat to the suppression pool is being performed; and

c. ≤ 110°F when all OPERABLE IRM channels are ≤ 70/125 divisions of full scale on Range 7.

APPLICABILITY: MODES 1, 2, and 3.

DISCUSSION: ENTRY CONDITIONS EOI-2 (Continued)

Primary containment hydrogen concentration above <A.11>, Minimum Dectectable Hydrogen Concentration.

Controlling primary containment hydrogen concentration prevents failure of primary containment due to pressure/temperature increases associated with ignition of combustible gases.

Suppression pool temperature above <A.31>, Technical Specification Suppression Pool Temperature LCO

Controlling suppression pool temperature: 1) maintains the pressure suppression function of primary containment, 2) maintains adequate NPSH requirements for pumps that take suction on the suppression pool, and 3) prevents exceeding suppression pool/chamber design limits.

Examination Outline Cross-reference:

295033 High Secondary Containment Area Radiation Levels / 9

EA2.03 (10 CFR 55.43.5 - SRO Only)

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS:

- Cause of high area radiation

Level	SRO
Tier #	1
Group #	2
K/A #	295033EA2.03
Importance Rating	4.2

Proposed Question: **# 85**

Unit 1 was operating at 100% Reactor Power when an inadvertent Reactor Scram occurred.

The below listed alarms were received following the Scram:

- RX BLDG AREA RADIATION HIGH, (1-9-3A, Window 22)
- RX BLDG, TURB BLDG, RF ZONE EXH RADIATION HIGH, (1-9-3A, Window 4)

1-EOI-3, "Secondary Containment Control," has been entered based on the following radiation levels in Unit 1 Reactor Building:

- Elevation 565 East ARM meter indicating off-scale high
- Elevation 565 Northeast ARM meter indicating 600 mr/hr and stable

Which ONE of the following completes the statement?

Based on the above conditions, the required action is to execute __(1)__ **AND** a possible isolation source for the primary system release is __(2)__?

	__(1)__	__(2)__
A.	1-EOI-1, "RPV Control,"	FCV 69-1, 2, 12.
B.	0-EOI-4, "Radioactivity Release Control,"	FCV 69-1, 2, 12.
C.	1-EOI-1, "RPV Control,"	SDV vents and drains.
D.	0-EOI-4, "Radioactivity Release Control,"	SDV vents and drains.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 = correct, EOI-1 is correct to enter, but with info given on Hi rads on both east / west, rules out RWCU (69-1, 2 12). They are ONLY applicable to the west side alarm. Step SC/R-2 is answered YES, 1050 mr/hr is > MAX SAFE. Part 2 = incorrect, The 565 elevation Northeast has no possible isolation sources listed in EOI-3 table 4.

B INCORRECT: Part 1 = incorrect, Given conditions indicate that there is ONLY one source > MAX SAFE, EOI-4 is entered based on off-site dose. No indications are given as to indications of exceeding any. Part 2 = incorrect, RWCU (69-1, 2 12) are NOT a possible leakage source based on given alarm locations, only on west side (no reference to rad alarms given in stem).

C **CORRECT:** Part 1 = correct, EOI-1 is correct to enter. Step SC/R-2 is answered YES, 1050 mr/hr is > MAX SAFE. Part 2 = correct, SDV valves are correct source choice, with rad levels elevated on both the east / west sides.

D INCORRECT: Part 1 = incorrect, ONLY **one** MAX SAFE has been exceeded. EOI-4 is entered based on off-site dose. No indications are given as to indications of exceeding any. Part 2 = correct, SDV valves are the correct source choice.

Technical Reference(s): 1-EOI-1 flow chart Rev 0, 1-9-3A Rev 38 (Attach if not previously provided)

1-EOI-3 flow chart / table 4 Rev 0 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank #

Modified Bank # 0606 NRC SRO 295017AA2.04 (Note changes or attach parent)

New

Question History: Last NRC Exam 0606 NRC SRO

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments: Distractors provide choices that either are correct for different values and symptoms in secondary containment or provide possible sources that do not result in the rad levels associated with the given symptoms.

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0038 Page 34 of 53

RX BLDG AREA RADIATION HIGH 1-RA-90-1D

22

(Page 1 of 2)

Sensor/Trip Point:

RI-90-4A	10 MR/HR	RI-90-24A	10 MR/HR
RI-90-8A	10 MR/HR	RI-90-25A	100 MR/HR
RI-90-9A	10 MR/HR	RI-90-26A	30 MR/HR
RI-90-13A	10 MR/HR	RI-90-27A	30 MR/HR
RI-90-14A	10 MR/HR	RI-90-28A	60 MR/HR
RI-90-20A	40 MR/HR	RI-90-29A	110 MR/HR
RI-90-21A	80 MR/HR		
RI-90-22A	1500 MR/HR		
RI-90-23A	10 MR/HR		
RI-90-23A	10 MR/HR		

SENSOR LOCATION:	1-RE-090-0004	MG Set Area Rx Bldg. El 639', R-5 S-Line
	1-RE-090-0008	Main Control Room. Rx Bldg. El 617', R-7 P-Line
	1-RE-090-0009	Clean-up System, Rx Bldg El 621', R-6 T-Line
	1-RE-090-0013	North Clean-up Sys. Rx Bldg. El 593', R-6 P-Line
	1-RE-090-0014	South Clean-up Sys. Rx Bldg. El 593', R-6 S-Line
	1-RE-090-0020	CRD-HCU West. Rx Bldg El 565', R-2 R-Line
	1-RE-090-0021	CRD-HCU East. Rx Bldg El 565', R-6 R-Line
	1-RE-090-0022	Tip Room, Rx Bldg El 565', R-5 P-Line
	1-RE-090-0023	Tip Drive, Rx Bldg. El 565', R-5 P-Line
	1-RE-090-0024	HPCI Room, RX Bldg. El 519', R-1 U-Line
	1-RE-090-0025	RHR West. Rx Bldg. El 519', R-2 U-Line
	1-RE-090-0026	Core Spray-RCIC, Rx Bldg. El 519', R-3 U-Line
	1-RE-090-0027	Core Spray. Rx Bldg. El 519', R-6 U-Line

BFN Unit 1	Panel 9-3 XA-55-3A	1-ARP-9-3A Rev. 0038 Page 9 of 53

RX BLDG,TURB BLDG, RF ZONE EXH RADIATION HIGH 1-RA-90-250A

4

(Page 1 of 1)

Sensor/Trip Point:

1-RM-90-250

Gas — HIGH ALARM - 6594 CPM

ALERT - 3297 CPM

Sensor Location: El 664' Refuel Floor R-4 P-Line

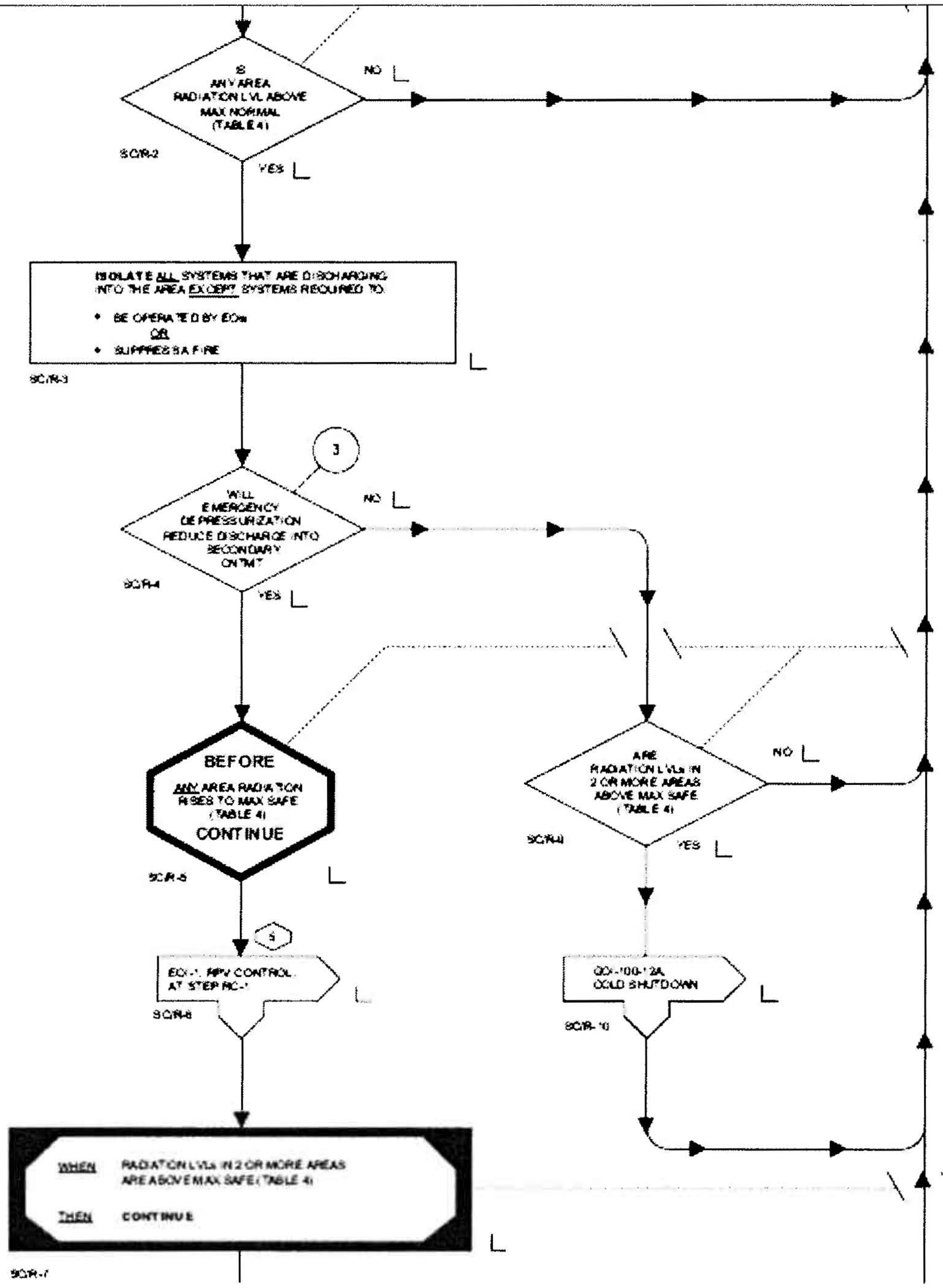
Probable Cause:
A. Daily source check.
B. High radiation in the Reactor Building, Turbine Building, Refuel Zone exhaust ventilation ducts
C. Dry Cask storage activities in progress.

Automatic Action: None

Operator Action:
A. **CHECK** 1-RM-90-250 on Panel 1-9-2 (0-MON-90-361) and **MONITOR** activity levels on recorder AIR PARTICULATE MONITOR CONTROLLER 1-MON-90-50 on Panel 1-9-2. ☐
B. **IF** high activity is conformed, **THEN** **NOTIFY** RAD PRO ☐
C. **REQUEST** Chemistry perform analysis to determine source. ☐
D. **IF** Dry Cask storage activities are in progress, **THEN** **NOTIFY** CASK Supervisor. ☐
E. **IF** the TSC is **NOT** manned, **THEN** **EVACUATE** personnel from affected areas. ☐

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



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 EOPs
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
ARE RADIATION LVLs IN 2 OR MORE AREAS ABOVE MAX SAFE (TABLE 4)
NO
YES
EOI-1, RPV CONTROL, AT STEP RC-1
SC/R-6
COLD SHUTDOWN
SC/R-10
WHEN RADIATION LVLs IN 2 OR MORE AREAS ARE ABOVE MAX SAFE (TABLE 4)
THEN CONTINUE
SC/R-7

Original Question: NRC 0606 SRO

Unit 1 was operating at full power when the following indications were received:

- Reactor Building Area High Rad alarm
- RWCU Area High Temperature element 69-835A is in alarm
- Reactor Building Ventilation Abnormal alarm

Radcon reports that radiation levels in Unit 1 Reactor building elevation 565 east are 950 mr/hr and rising. Radiation levels at elevation 565 west are 800 mr/hr and stable.

REFERENCE PROVIDED

Which ONE of the following describes the required actions for the given conditions and a possible isolation source for the radiation release?

A. Enter EOI-1, FCV 74-47, 48

B. Enter EOI Contingency C2, FCV 69-1, 2, 12

C. Enter EOI-1, SDV vents and drains

D. Enter EOI Contingency C2, SDV vents and drains

Examination Outline Cross-reference:		
206000 HPCI	Level	SRO
A2.16 (10CFR 55.43.5 - SRO Only)	Tier #	2
Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	Group #	1
	K/A #	206000A2.16
• High drywell pressure: BWR-2,3,4	Importance Rating	4.1

Proposed Question: **# 86**

Unit 1 is at 100% Reactor Power.

- At 0647 on 5/14/09 ECCS HIGH DRYWELL PRESSURE INSTRUMENT CHANNEL 'A', 1-PIS-64-58A (Associated Relay 14A-K5B), was removed from service for a calibration surveillance
- At 0720 on 5/14/09 ECCS HIGH DRYWELL PRESSURE INSTRUMENT CHANNEL 'B', 1-PIS-64-58B (Associated Relay 14A-K5A), failed downscale

Based on these conditions, which ONE of the following completes the statement for required actions associated with the failed pressure channel **AND** HPCI in accordance with TS 3.3.5.1, "ECCS Instrumentation"?

ECCS HIGH DRYWELL PRESSURE INSTRUMENT CHANNEL 'B', 1-PIS-64-58B, must be placed in trip no later than _________.

[REFERENCE PROVIDED]

A. 0647 on 5/15/09 **ONLY**.

B. 0720 on 5/15/09 **ONLY**.

C. 0647 on 5/15/09 **AND** declare HPCI inoperable no later than 0820 on 5/14/09.

D. 0720 on 5/15/09 **AND** declare HPCI inoperable no later than 0820 on 5/14/09.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Incorrect as explained in D. Plausible in that this would be the correct completion time if separate condition entry was not allowed and the candidate failed to recognize that HPCI has lost initiation capability on High Drywell Pressure.

B INCORRECT: Plausibility based on this would be the correct answer if HPCI had not lost initiation capability on High Drywell Pressure.

C INCORRECT: Plausible in that this would be the correct completion time if separate condition entry was not allowed for each channel.

D **CORRECT**: With less than the required number of Drywell Pressure High Channels Operable, TS 3.3.5.1 Conditions A and B must be entered. The failed Drywell Pressure channel must be placed in trip 24 hours from the time that it failed per Condition B.3 since separate condition entry allowed for each channel. With A and B channels inoperable, HPCI has lost initiation capability on Drywell Pressure High requiring that HPCI be declared inoperable within one hour per Condition B.2

Technical Reference(s): U1 TS 3.3.5.1 (Attach if not previously provided)
OPL171.042, Rev. 19 (Including version / revision number)
1-730E928

Proposed references to be provided to applicants during examination: **TS 3.3.5.1 / HPCI Initiation Logic Diagram**

Learning Objective: VB.2 / VB.3 / VB.10 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

3.3 INSTRUMENTATION

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

LCO 3.3.5.1 The ECCS instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more channels inoperable.	A.1	Enter the Condition referenced in Table 3.3.5.1-1 for the channel.	Immediately

(continued)

ACTIONS (continued)

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

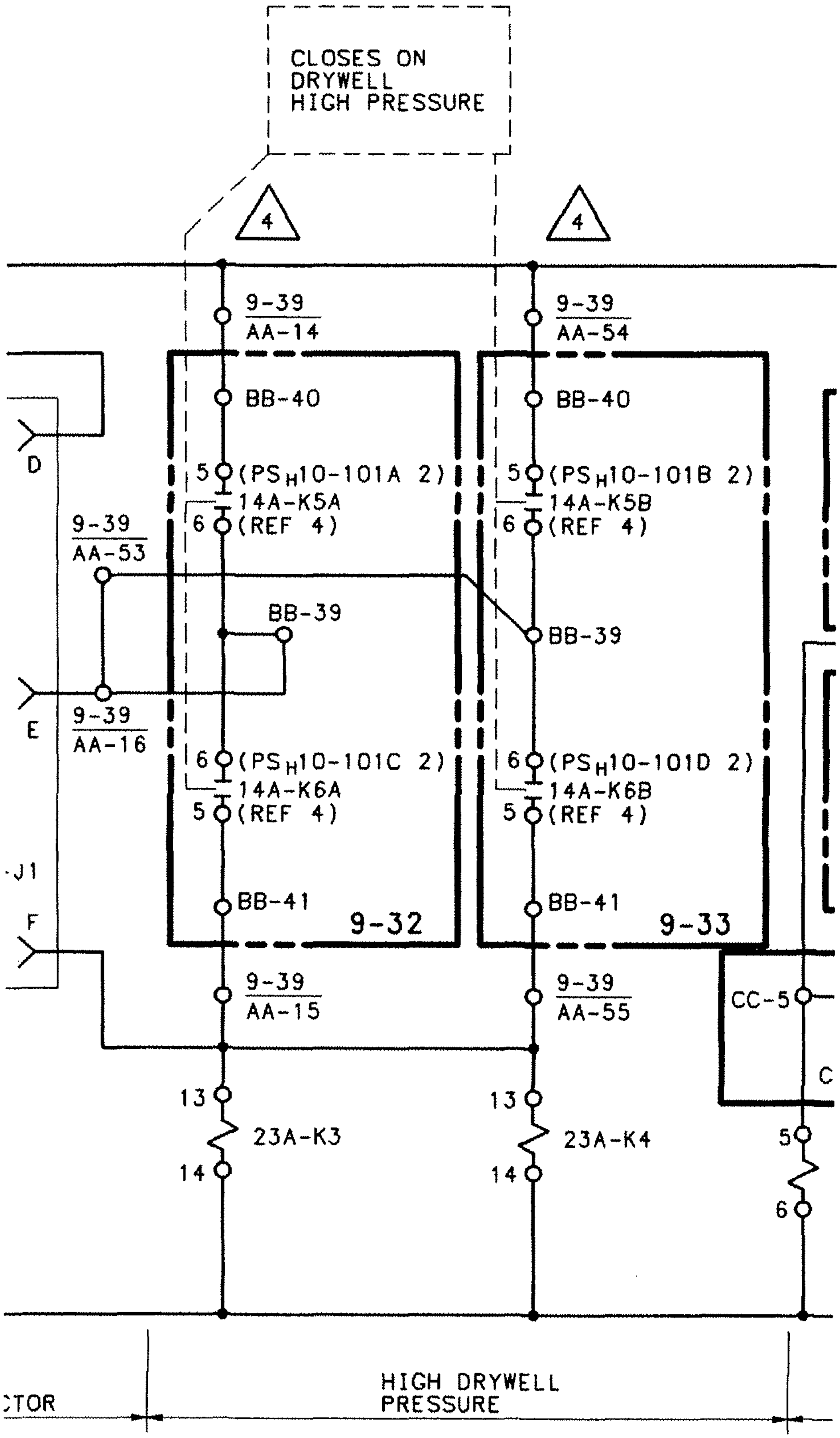
(continued)

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level - Low Low, Level 2(e)	1, 2(d), 3(d)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 470 inches above vessel zero
b. Drywell Pressure - High(e)	1, 2(d), 3(d)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Vessel Water Level - High, Level 8	1, 2(d), 3(d)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 583 inches above vessel zero
d. Condensate Header Level - Low	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ Elev. 551 feet
e. Suppression Pool Water Level - High	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≤ 7 inches above instrument zero
f. High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 671 gpm
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level - Low Low Low, Level 1(e)	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero

(continued)

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


CLOSES ON
DRYWELL
HIGH PRESSURE
4
4
9-39
AA-14
9-39
AA-54
BB-40
BB-40
D
5 (PS H10-101A 2)
14A-K5A
6 (REF 4)
5 (PS H10-101B 2)
14A-K5B
6 (REF 4)
9-39
AA-53
BB-39
BB-39
E
9-39
AA-16
6 (PS H10-101C 2)
14A-K6A
5 (REF 4)
6 (PS H10-101D 2)
14A-K6B
5 (REF 4)
J1
F
BB-41
9-32
BB-41
9-33
9-39
AA-15
9-39
AA-55
CC-5
C
13
23A-K3
14
13
23A-K4
14
5
6
TOR
HIGH DRYWELL
PRESSURE

Examination Outline Cross-reference:			
212000 RPS	Level	[redacted]	SRO
A2.20 (10CFR 55.43.5 - SRO Only)	Tier #	[redacted]	2
Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM (RPS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	Group #	[redacted]	1
	K/A #		212000A2.20
• Full system activation (full-SCRAM)	Importance Rating	[redacted]	4.2

Proposed Question: **# 87**

Following a Reactor Scram on Unit 1 from 100% Reactor Power, the following conditions are observed:

- Reactor Power is 22%
- Reactor Level is (-) 100 inches and stable
- CONTROL AIR DRYER DISCH PRESSURE LOW, (1-9-20B, Window 32), is in alarm
- Control Air Header Pressure is 0 psig
- Control Rod Position indication is available but Control Rods can **NOT** be driven.
- Suppression Pool Temperature is 155 ° F and rising
- Suppression Pool Level 15 feet
- Reactor Pressure 950 psig

Which ONE of the following completes the statement?

The Unit Supervisor must direct execution of 1-EOI __**(1)**__ to insert Control Rods. The current conditions require a __**(2)**__ be declared in accordance with EPIP-1, "Emergency Classification Procedure."

[REFERENCE PROVIDED]

A. **(1)** Appendix 1F, "Manual Scram."
(2) General Emergency

B. **(1)** Appendix 1F, "Manual Scram."
(2) Site Area Emergency

C. **(1)** Appendix 1E, "Manual Insertion of Control Rods By Venting The Over Piston Area."
(2) General Emergency

D. **(1)** Appendix 1E, "Manual Insertion of Control Rods By Venting The Over Piston Area."
(2) Site Area Emergency

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Part 1 incorrect - Plausible in that this is normal method of for inserting Control with a Hydraulic Lock ATWS which is present based on conditions. However, with the loss of control air, the scram can not be reset and SDV can not be drained. Part 2 incorrect – Per EPIP-1 1.S-2, Scram Failure, Failure of automatic scram, manual scram, and ARI to bring the reactor subcritical requires SAE be declared. GE would be required if HCTL was in the unsafe or if level could not be maintained > (-) 180 inches.

B INCORRECT: Part 1 incorrect as explained above. Part 2 is correct.

C INCORRECT: Part 1 is correct. Part 2 is incorrect as explained above.

D **CORRECT:** Part 1 is correct. Venting the over piston area with the Hydraulic Lock conditions present will result in control rod insertion. Part 2 correct – Per EPIP-1 1.S-2, Scram Failure, Failure of automatic scram, manual scram, and ARI to bring the reactor subcritical requires SAE be declared.

Technical Reference(s): OPL171.202 Rev 8 (Attach if not previously provided)
1-EOI Appendix 1E Rev 0 (Including version / revision number)
EPIP-1 Rev 44

Proposed references to be provided to applicants during examination: EPIP-1 Classification Matrix

Learning Objective: V.B.13 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

a.	**Vent control rod drive overpiston volumes**	Safety Awareness
	This method maximizes differential pressure across the control rod drive piston. This method is performed locally at the HCUs, is time consuming, and should be performed when the HCU area is accessible.	EOI Appendix 1E provides step-by-step guidance to vent CRD overpiston volumes.

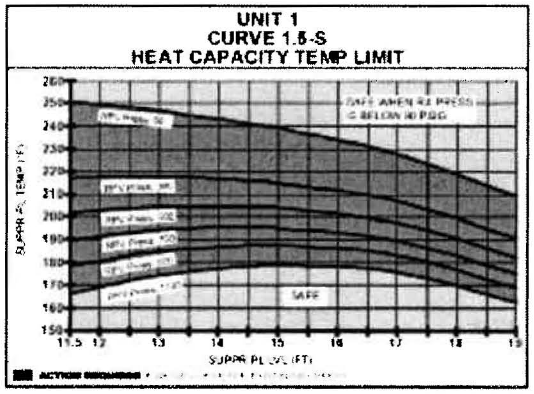
BROWNS FERRY	EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX	EPIP-1

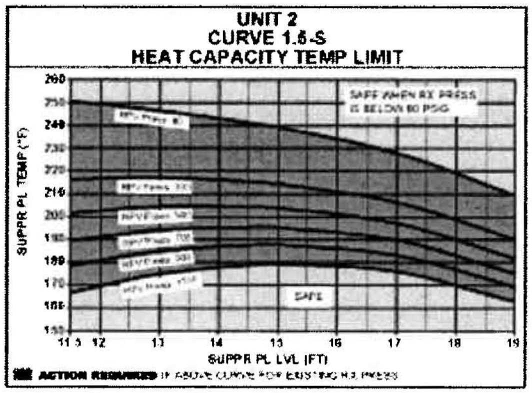
	SCRAM FAILURE	REACTOR COOLANT ACTIVITY
	Description	Description
UNUSUAL EVENT		**1.3-U** Reactor coolant activity exceeds 26 µCi/gm dose equivalent I-131 (Technical Specification Limits) as determined by chemistry sample. OPERATING CONDITION ALL
ALERT	**1.2-A** NOTE Failure of RPS automatic scram functions to bring the reactor subcritical AND Manual scram or ARI (automatic or manual) was successful. OPERATING CONDITION: Mode 1 or 2	**1.3-A** Reactor coolant activity exceeds 300 µCi/gm dose equivalent Iodine-131 as determined by chemistry sample. OPERATING CONDITION: Mode 1 or 2 or 3
SITE EMERGENCY	**1.2-S** NOTE Failure of automatic scram, manual scram, and ARI to bring the reactor subcritical. OPERATING CONDITION: Mode 1	
GENERAL EMERGENCY	**1.2-G** CURVE US Failure of automatic scram, manual scram, and ARI. Reactor power is above 3% AND Either of the following conditions exists: • Suppression Pool temp exceeds HCTL. Refer to Curve 1.2-G. • Reactor water level can NOT be restored and maintained at or above -180 inches. OPERATING CONDITION: Mode 1 or 2	

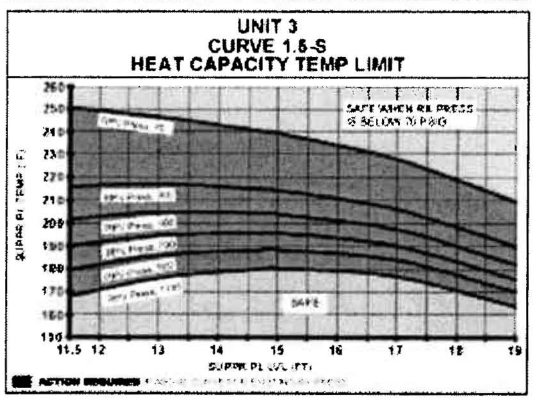
BROWNS FERRY	EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX	EPIP-1

NOTES

CURVES/TABLES:







Examination Outline Cross-reference:	Level	[redacted]	SRO
217000 RCIC	Tier #	[redacted]	2
G2.4.41 (10CFR 55.43.5 - SRO Only)	Group #	[redacted]	1
Knowledge of the emergency action level thresholds and classifications.	K/A #	217000G2.4.41	
	Importance Rating	[redacted]	4.6

Proposed Question: **# 88**

The following initial conditions exist on Unit 2:

- 100% Reactor Power
- HPCI is out of service

Subsequently a loss of Off Site Power occurs, causing a Reactor Scram.

The following conditions exist following the Scram:

- The Reactor is shutdown with **ALL** Control Rods inserted
- Reactor Water Level is (-) 120 inches and slowly lowering
- Reactor Pressure is 725 psig and slowly lowering
- Drywell Pressure is 2.6 psig and slowly rising
- DRYWELL DIV 1/2 RAD MONITOR DOWNSCALE/INOP, (2-9-7C, Window 12/13), in alarm
- RCIC is the **ONLY** system injecting
- A break occurs in the RCIC Steam Supply line, as evidenced by 2-TE-71-41B reading 180 °F **AND** 2-RM 90-26A reading up-scale high
- RCIC fails to AUTOMATICALLY **AND** MANUALLY isolate

Based on the above conditions, which ONE of the following describes the HIGHEST Emergency Action Level which must be declared?

[REFERENCE PROVIDED]

A. A General Emergency.

B. A Site Area Emergency.

C. An Alert.

D. An Unusual Event.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: No evidence of fuel failure exists, based on given DW rad Downscale alarm. The candidate may confuse this with hi rad conditions if read incorrectly, creating plausibility to enter a General Emergency.

B **CORRECT**: SAE per 3.1-S for unisolated / discharging into secondary ctmt + > max safe (rad level) **or** if the candidate decides that -162" cannot be maintained, enter SAE on 1.1-S.1. The temperature is < max safe for U2. The rad level is > max safe. 2-RM-90-26A scale reads 0.1 - 1000 mr/hr. The candidate will have to recall from memory the Max safe value of 1000 mr/hr.

C INCORRECT: Plausible in that the candidate may act on the 2.45# DW pressure value (2.1-A) and think the leak is in the DW

D INCORRECT: Plausible in that the candidate may act upon the lowering level (1.1-U.1), which is only applicable in mode 5.

Technical Reference(s): EPIP-1 Rev 44 (Attach if not previously provided)

2-EOI-3 Rev 11 (Including version / revision number)

Proposed references to be provided to applicants during examination: **EPIP-1 Event Classification Matrix + table 3.1**

Learning Objective: ____________ (As available)

Question Source: Bank #

Modified Bank # OPL171.075 36 (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

OPL171.075 36

Selected 0

Select this one

Multiple choice: single 4 1

Comp/Mem	Tier/Group	Sys/Procedure #	Objective	KA #
2	REP			S000EM21

Unit 2 is at 100% power.
A loss of feedwater flow occurs.
The reactor scrams on low water level.
All rods fully insert.
HPCI and RCIC automatically initiate on low-low water level.
A break occurs in the HPCI steam supply line, as evidenced by TE 73-55A reading 270 deg. F and RM 90-24A reading 300 mr/hr.
The automatic system isolation fails, and attempts to manually isolate the leak from the control room are unsuccessful.

SELECT the proper event classification.

A. Notification of Unusual Event

B. Alert

C.✔ Site Area Emergency

D. General Emergency

PRIMARY CONTAINMENT PRESSURE	PRIMARY CONTAINMENT HYDROGEN	
Description	Description	
		UNUSUAL EVENT
2.1-A TABLE		
Drywell pressure at or above 2.45 psig AND Indication of Primary System leakage into Primary Containment. Refer to Table 2.1-A. OPERATING CONDITION: Mode 1 or 2 or 3		ALERT

BROWNS FERRY	EMERGENCY CLASSIFICATION PROCEDURE EVENT CLASSIFICATION MATRIX	EPIP-1

WATER LEVEL

Description	Description	
1.1-U1 NOTE Uncontrolled water level decrease in Reactor Cavity with irradiated fuel assemblies expected to remain covered by water. OPERATING CONDITION: Mode 5	**1.1-U2** Uncontrolled water level decrease in Spent Fuel Pool with irradiated fuel assemblies expected to remain covered by water. OPERATING CONDITION ALL	UNUSUAL EVENT
1.1-A1 NOTE Uncontrolled water level decrease in Reactor Cavity expected to result in irradiated fuel assemblies being uncovered. OPERATING CONDITION: Mode 5	**1.1-A2** Uncontrolled water level decrease in Spent Fuel Storage Pool expected to result in irradiated fuel assemblies being uncovered. OPERATING CONDITION: ALL	ALERT
1.1-S1 NOTE Reactor water level can NOT be maintained above -162 inches. (TAF) OPERATING CONDITION: ALL	**1.1-S2** Reactor water level can NOT be determined. OPERATING CONDITION: Mode 1 or 2 or 3	SITE EMERGENCY
1.1-G1 Reactor water level can NOT be restored and maintained above -180 inches. OPERATING CONDITION: Mode 1 or 2 or 3	**1.1-G2** NOTE TABLE US Reactor water level can NOT be determined AND Either of the following exists: • The reactor will remain subcritical without boron under all conditions, and ➢ Less than 4 MSRVs can be opened, or ➢ Reactor pressure can NOT be restored and maintained above Suppression Chamber pressure by at least ❖ UNIT 1 – 90 psi ❖ UNIT 2 – 80 psi ❖ UNIT 3 – 70 psi • It has NOT been determined that the reactor will remain subcritical without boron under all conditions and unable to restore and maintain MARFP in Table 1.1-G2. OPERATING CONDITION: Mode 1 or 2 or 3	GENERAL EMERGENCY

CURVES/TABLES:

TABLE 3.1 MAXIMUM SAFE OPERATING AREA TEMPERATURE LIMITS				
AREA	APPLICABLE PANEL 9-21 TEMPERATURE ELEMENTS (UNLESS OTHERWISE NOTED)	MAX SAFE OPERATING VALUE ^{0}F		
		UNIT 1	UNIT 2	UNIT 3
RHR A/C Pump Room	74-95A	215	150	155
RHR B/D Pump Room	74-95B	150	210	215
HPCI Turbine Area	73-55A	275	270	270
CS A/C Pump and RCIC Turbine Area	71-41A	190	190	190
RCIC Steam Supply Area	71-41B, 41C, 41D	195	200	250
HPCI Steam Supply Area	73-55B, 55C, 55D	245	240	240
RHR A/C Pump Supply Area	74-95H	245	240	240
RHR B/D Pump Supply Area	74-95G	190	240	240
Main Steam Line Leak Detection High	(XA-55-3D-24) Panel 9-3 TIS-1-60A	315	315	315
RHR Valve Room	74-95E	175	170	175
RWCU Isol Logic Channel A/B Temp High	(XA-55-5B-32/33) Panel 9-5 69-835A, B, C, D Aux Inst Room	175	170	175
RWCU Outbd Isol Vlv Area	69-29F	220	220	220
RWCU Hx Area	69-29G	220	220	220
RWCU Hx Exh Duct	69-29H	220	220	220
RWCU Recirc Pump A Area	69-29D	215	215	215
RWCU Recirc Pump B Area	69-29E	215	215	215
RHR A/C Hx Room	74-95C	210	195	200
RHR B/D Hx Room	74-95D	210	195	200
FPC Hx Area	74-95F	160	155	155

CURVES/TABLES:

TABLE 3.2 MAXIMUM SAFE OPERATING AREA RADIATION LIMITS		
AREA	RAD MONITOR	MAX SAFE VALUE MR/HR
RHR West Room	90-25A	1000
RHR East Room	90-28A	1000
HPCI Room	90-24A	1000
CS/RCIC Room	90-26A	1000
Core Spray Room	90-27A	1000
Suppr Pool Area	90-29A	1000
CRD-HCU West Area	90-20A	1000
CRD-HCU East Area	90-21A	1000
TIP Drive Area	90-23A	1000
North RWCU System Area	90-13A	1000
South RWCU System Area	90-14A	1000
RWCU System Area	90-9A	1000
MG Set Area	90-4A	1000
Fuel Pool Area	90-1A	1000
Service Flr Area	90-2A	1000
New Fuel Storage	90-3A	1000

TABLE 3.1-G/3.2-G INDICATIONS OF POTENTIAL OR SIGNIFICANT FUEL CLADDING FAILURE WITH RCS BARRIER INTACT INSIDE PRIMARY CONTAINMENT					
UNIT 1 DRYWELL RADIATION		UNIT 2 DRYWELL RADIATION		UNIT 3 DRYWELL RADIATION	
1-RE-90-272A	$\geq$ 196 R/HR	2-RE-90-272A	$\geq$ 642 R/HR	3-RE-90-272A	$\geq$ 196 R/HR
1-RE-90-273A	$\geq$ 297 R/HR	2-RE-90-273A	$\geq$ 297 R/HR	3-RE-90-273A	$\geq$ 297 R/HR
Reactor Coolant Activity $\geq$ 300 μCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity $\geq$ 300 μCi/gm Dose Equivalent Iodine 131		Reactor Coolant Activity $\geq$ 300 μCi/gm Dose Equivalent Iodine 131	

3.1-S | | | **TABLE** | **US** An unisolable Primary System leak is discharging into Secondary Containment **AND** Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3.1. OPERATING CONDITION: Mode 1 or 2 or 3	**SITE EMERGENCY**
3.1-G | | | **TABLE** | **US** An unisolable Primary System leak is discharging into Secondary Containment **AND** Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3.1 **AND** Any indication of potential or significant fuel cladding failure exists. Refer to Table 3.1-G/3.2-G with RCS Barrier intact inside Primary Containment. OPERATING CONDITION Mode 1 or 2 or 3	**GENERAL EMERGENCY**

	T
3.2-A | | | | Any of the following high radiation alarms on Panel 9-3: • RA-90-1A, Fuel Pool Floor Alarm • RA-90-250A, Reactor, Turbine, Refuel Exhaust • RA-90-142A, Reactor Refuel Exhaust • RA-90-140A, Refueling Zone Exhaust **AND** Confirmation by Refuel Floor personnel that irradiated fuel damage may have occurred. OPERATING CONDITION: ALL	**ALERT**
3.2-S | | | **TABLE** | An unisolable Primary System leak is discharging into Secondary Containment **AND** Any area radiation level at or above the Maximum Safe Operating Area radiation limit listed in Table 3.2. OPERATING CONDITION: Mode 1 or 2 or 3	**SITE EMERGENCY**
3.2-G | | | **TABLE** | **US**	

BROWNS FERRY	EMERGENCY CLASSIFICATION PROCEDURE TECHNICAL BASIS	EPIP-1

WATER LEVEL 1.1-S1

SITE AREA EMERGENCY

EAL: Reactor water level can NOT be maintained above -162 inches. (TAF).

OPERATING CONDITION: ALL

BASIS: If reactor water level cannot be maintained above TAF the potential exist for fuel cladding damage. Events most likely to result in coolant inventory loss to this extent are RCS boundary degradation events or station blackout events. For this event to be declared, RPV water level must have decreased or be trending to a value that, in the opinion of the Site Emergency Director, has resulted in or will result in some actual core uncovery. Additionally, the Site Emergency Director must have evidence that Reactor level has been or can be recovered to above TAF.

This event classification also applies in Mode 5 when the Reactor Vessel head is installed. Inadvertent draining of the Reactor Vessel is possible under these conditions due to valving errors associated with the RHR system or failures associated with isolation valves during alignment changes of systems connected to the Reactor Vessel below the normal water level.

The fact that the transient was severe enough to result in inability to maintain RPV level coupled with the anticipatory nature of this event classification as a precursor to more serious event warrants the Site Area Emergency event classification.

For events that occur during operation, escalation to General Emergency is based on inability to assure adequate core cooling by restoring and maintaining RPV water level following transients that have resulted in extreme RPV water level decrease. For events that occur during shutdown or Mode 5, escalation is by radioactive release event classifications.

SECONDARY CONTAINMENT RADIATION 3.2-G

GENERAL EMERGENCY

EAL: An unisolable Primary System leak is discharging into Secondary Containment

AND

Any area radiation level at or above the Maximum Safe Operating Area radiation limit listed in Table 3.2.

AND

Any indication of potential or significant fuel cladding failure exists. Refer to Table 3.1-G/3.2-G with RCS Barrier intact inside Primary Containment.

OPERATING CONDITION: Mode 1 or 2 or 3

BASIS: Secondary Containment radiation levels of this magnitude are indicative of significant inability to contain or control radioactive materials within the primary system and Primary Containment. If the primary system is the source then these indications represent loss of RCS pressure boundary and Primary Containment pressure boundary. Table 3.1-G/3.2-G provides guidance for determining if significant fuel failure should be assumed.

This event classification represents loss or potential loss of all three barriers designed to contain fission products during accidents; therefore, the General Emergency classification is appropriate.

REFERENCES: Reg Guide 1.101 Rev. 3, (NUMARC-FG)
EOI Program Manual, Section V-E
Calculation ND-N0090-930055 R12

Examination Outline Cross-reference:

239002 SRVs

G2.4.45 (10CFR 55.43.5 - SRO Only)

Ability to prioritize and interpret the significance of each annunciator or alarm.

Level	[redacted]	SRO
Tier #	[redacted]	2
Group #	[redacted]	[redacted]
K/A #	239002G2.4.45	
Importance Rating	[redacted]	4.3

Proposed Question: **# 89**

The Unit 2 is currently operating steady state at approximately 10% Reactor Power. During the performance of 2-SR-3.4.3.2, "Main Steam Relief Valves Manual Cycle Test," the following events occur:

- At 1945: SRV 2-PCV-1-179 (Panel vertical) fails to open
- At 2030: SRV 2-PCV-1-19 (Panel apron) fails to open
- At 2200: SRV 2-PCV-1-42 (Panel vertical) fails to open
- At 2205: SRV Testing is secured to troubleshoot
- At 2240: HPCI TURBINE EXH RUPTURE DISC PRESSURE HIGH, (2-9-3F, Window 17), annunciator is received

Given that any related automatic actions occur, which ONE of the following sets of Technical Specification actions are required for the above conditions?

[REFERENCE PROVIDED]

A. At 1945: Enter LCO 3.4.3 Condition 'A'.
At 2030: Enter LCO 3.5.1 Condition 'E' **AND** Re-enter LCO 3.4.3 Condition 'A'.
At 2200: Re-enter LCO 3.4.3 Condition 'A'.

B. At 2030: Enter LCO 3.4.3 Condition 'A' **AND** LCO 3.5.1 Condition 'E'.
At 2240: Enter LCO 3.5.1 Conditions 'C' **AND** 'H'.
At 2240: Enter LCO 3.03 concurrently with **ALL** other related LCOs

C. At 2030: Enter LCO 3.4.3 Condition 'A' **AND** LCO 3.5.1 Condition 'E'.
At 2200: Re-enter LCO 3.4.3 Condition 'A'.
At 2240: Enter LCO 3.0.3 **AND** exit **ALL** other related LCOs.

D. At 1945: Enter LCO 3.5.1 Condition 'E'.
At 2030: Enter LCO 3.4.3 Condition 'A'.
At 2200: Enter LCO 3.5.1 Condition 'G'.
At 2240: Enter LCO 3.5.1 Condition 'C'.

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: 1945 - 3.4.3, have 13 SRVs and only require 12 SRVs. While entry into 3.5.1 'E' is required at time 2030, re-entry into 3.4.3. 'A' would only be being entered for the first time.

B **CORRECT:** At time 1945, you still have 12 (of 13) required SRVs, and no Tech Spec entry is required. At time 2030, 2 SRVs are Inop = (3.4.3 'A') + 1 is ADS = 3.5.1 'E'. At time 2200, you have a 3rd SRV Inop (not ADS), but there is no NOTE allowing separate entry into 3.4.3, so you do not re-enter. 2240 HPCI Inop = 3.5.1 'C' AND 'H' = Entry into LCO 3.03. In LCO 3.03, you do not exit other LCOs, so their clocks would run concurrently.

C INCORRECT: Time 2030 actions are correct. At time 2200, you have a 3rd SRV Inop (not ADS), but there is no NOTE allowing separate entry into 3.4.3, so you do not re-enter. After entering LCO 3.0.3, you do not exit other LCOs.

D INCORRECT: If candidate confuses which valves are ADS versus standard SRVs, this may be the approach they would take.

Technical Reference(s): T.S. 3.4.3 and 3.5.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **T.S. 3.4.3 and 3.5.1 No Bases or SRs**

Learning Objective: ____________ (As available)

Question Source: Bank #

Modified Bank # Fitz D New Q 81 (Note changes or attach parent)

New

Question History: Last NRC Exam 2005 Fitz NRC

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **x**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

BFN Unit 2	Unit Startup and Power Operation	2-GOI-100-1A Rev. 0139 Page 105 of 169

5.0 INSTRUCTION STEPS (continued)

[72] **VERIFY** HPCI operable within 12 hours after Reactor pressure is greater than or equal to 950 psig, but less than or equal to 1040 psig, AND at least two turbine bypass valves are full open. **COMPLETE** 2-SR-3.5.1.7 OR **VERIFY** current (N/A if HPCI surveillance is going to be performed in Mode 1).

(R) ________ ________ ________
Initials Date Time

NOTE

2-SR-3.4.3.2, Main Steam Relief Valves Manual Cycle Test, is performed once per operating cycle. Tech Specs SR 3.4.3.2 requires that each S/RV opens when manually actuated, however it is not required to be performed until 12 hours after Reactor steam pressure and flow are adequate to perform the test. Adequate pressure at which this test is to be performed is greater than 935 psig. Adequate steam flow is represented by at least 3 main turbine bypass valves full open. A check with Work Control will determine whether this SR should be performed at this time.

[73] **WHEN** Reactor pressure is greater than or equal to 935 psig **AND** three (3) Turbine bypass valves are fully open, **THEN**

PERFORM the following:

- **ENTER** 12 hour LCO for Main Steam Relief Valve Operability. (Tech Specs LCO 3.4.3). (N/A, if 2-SR-3.4.3.2 is not required)

  (R) ________ ________ ________
  Initials Date Time

- **RECORD** Time LCO entered. (N/A if LCO entry not required.)

  Date ________ Time ________

  (R) ________ ________ ________
  Initials Date Time

- **IF** 2-SR-3.4.3.2 is required to be performed and Reactor pressure is greater than or equal to 935 psig with 3 turbine bypass valves full open, **THEN**

  PERFORM 2-SR-3.4.3.2. (Otherwise N/A)

BFN	Panel 9-3	2-ARP-9-3F
Unit 2	2-XA-55-3F	Rev. 0027
		Page 20 of 39

HPCI TURBINE
EXH RUPTURE DISC
PRESSURE HIGH
2-PA-73-20

17

(Page 1 of 1)

Sensor/Trip Point:

PS-73-20A	10 psig
PS-73-20B	10 psig
PS-73-20C	10 psig
PS-73-20D	10 psig

Sensor Location: El 519', Column R-14 U-LINE
Rx Bldg Panel 2-25-63

Probable Cause:
A. Inner diaphragm ruptured.
B. Sensor malfunction.

Automatic Action:
A. HPCI TURBINE STOP VALVE, 2-FCV-73-18, closes.
B. HPCI PUMP MIN FLOW VALVE, 2-FCV-73-30, closes.
C. The following HPCI steam supply valves close:
- HPCI STEAM LINE INBD ISOL VALVE, 2-FCV-73-2
- HPCI STEAM LINE OUTBD ISOL VALVE, 2-FCV-73-3

NOTE

TACF 2-08-002-073 electrically disabled 2-FCV-73-81 at breaker with valve closed.

- HPCI STEAM LINE WARM-UP VALVE, 2-FCV-73-81

D. The following HPCI Suppression Pool suction valves close:
- HPCI SUPPR POOL INBD SUCT VLV, 2-FCV-73-26
- HPCI SUPPR POOL OUTBD SUCT VLV, 2-FCV-73-27

E. The following amber lights indicating auto isolation seal-in will illuminate:
- HPCI AUTO ISOL LOGIC A, 2-IL-73-58A

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (S/RVs)

LCO 3.4.3 The safety function of 12 S/RVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required S/RVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

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

3.5.1 ECCS - Operating

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

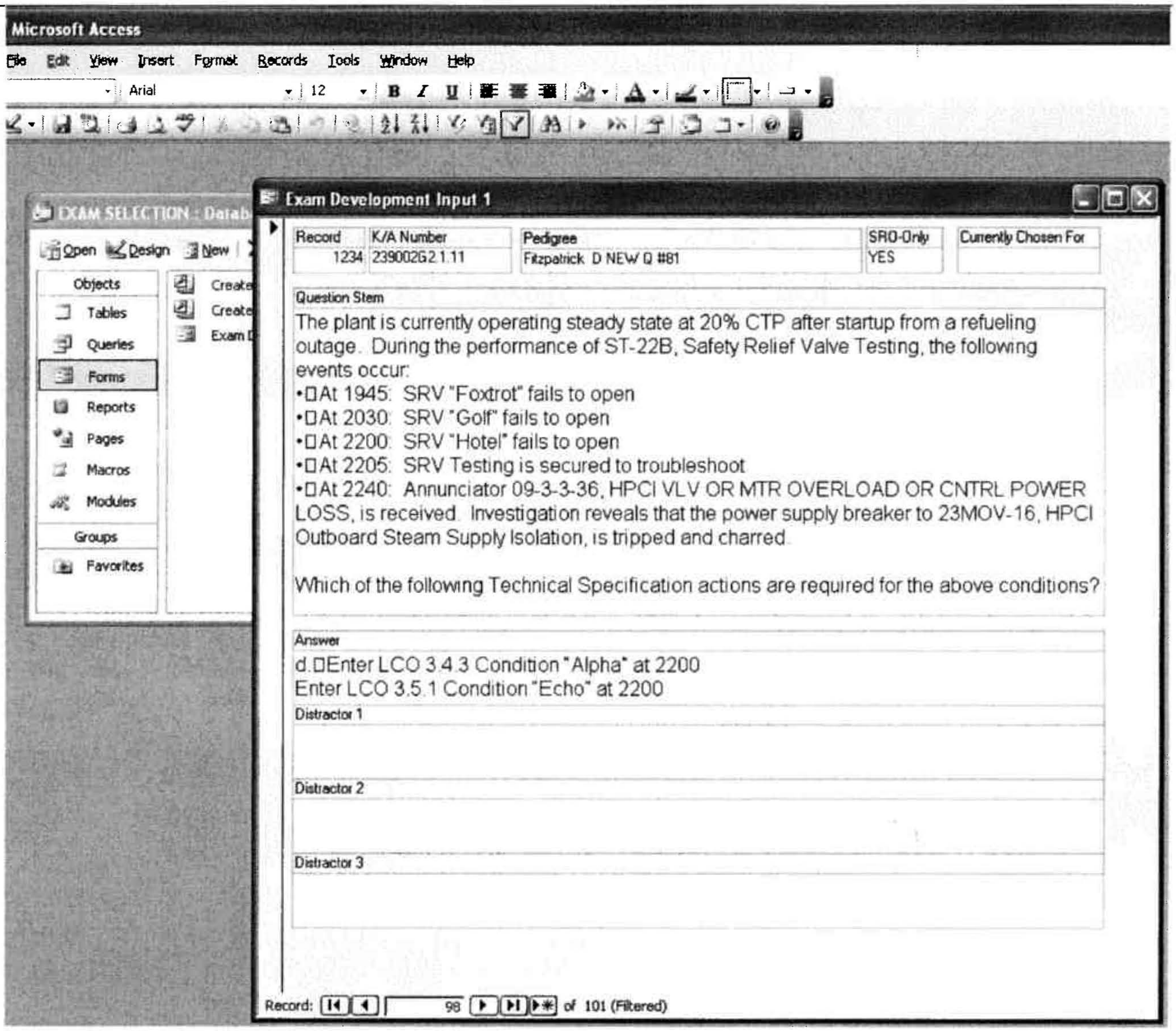
APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

-----------------------------------NOTE-----------------------------------
LCO 3.0.4.b is not applicable to HPCI.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. OR One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days(1)

(continued)



Record	K/A Number	Pedigree	SRO-Only	Currently Chosen For
1234	239002G2.1.11	Fitzpatrick D NEW Q #81	YES	

Question Stem

The plant is currently operating steady state at 20% CTP after startup from a refueling outage. During the performance of ST-22B, Safety Relief Valve Testing, the following events occur:

- At 1945: SRV "Foxtrot" fails to open
- At 2030: SRV "Golf" fails to open
- At 2200: SRV "Hotel" fails to open
- At 2205: SRV Testing is secured to troubleshoot
- At 2240: Annunciator 09-3-3-36, HPCI VLV OR MTR OVERLOAD OR CNTRL POWER LOSS, is received. Investigation reveals that the power supply breaker to 23MOV-16, HPCI Outboard Steam Supply Isolation, is tripped and charred.

Which of the following Technical Specification actions are required for the above conditions?

Answer

d. Enter LCO 3.4.3 Condition "Alpha" at 2200
Enter LCO 3.5.1 Condition "Echo" at 2200

Distractor 1

Distractor 2

Distractor 3

Examination Outline Cross-reference:	Level	SRO
261000 SGTS	Tier #	2
A2.11 (10CFR 55.43.5 - SRO Only)	Group #	1
Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM (SGTS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	K/A #	261000A2.11
• High containment pressure	Importance Rating	3.3

Proposed Question: **# 90**

The following conditions exist:

- Units 1 **AND** 3 are at 100% Reactor Power
- Unit 2 is in Mode 3, with cooldown to mode 4 in progress
- Standby Gas Treatment System 'A' was removed from service at 1000 on 3/2/09 for planned maintenance

At 1300 on 3/3/09 a coolant leak in the Drywell on Unit 2 results in the following plant conditions:

- Drywell Pressure is 2.85 psig
- Reactor Water Level is being controlled (+) 2 to (+) 51 inches with RCIC
- Standby Gas Treatment 'B' Blower tripped immediately upon initiation **AND** cannot be repaired for two weeks

Standby Gas Treatment System 'A' is restored to Operable at 1500 on 3/3/09.

Which ONE of the following identifies the latest time / date that Units 1 **AND** Unit 3 must be in Mode 3?

[REFERENCE PROVIDED]

A. 0200 on 3/4/09.

B. 2200 on 3/9/09.

C. 2200 on 3/10/09.

D. 0100 on 3/11/09.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: This would be the correct time if SGTS A remained inoperable at 0200 on 3/4. However, since SGTS A was returned to Operable at 1100 on 3/3, Condition D for TS 3.6.4.3 and TS 3.0.3 are exited.

B INCORRECT: This would be the correct completion time based initial Inop time for SGTS A per TS 3.6.4.3 Conditions A and B. However, since the subsequent inoperability existed concurrent with the first inoperability and remained inoperable after the first inoperability was resolved, Completion Times may be extended in accordance with TS Section 1.3, "Completion Times" making this answer incorrect.

C **CORRECT**: This completion time is based on initial inoperability plus additional 24 hours. Since the subsequent inoperability existed concurrent with the first inoperability and remained inoperable after the first inoperability was resolved, Completion Times may be extended in accordance with TS Section 1.3, "Completion Times". The completion time extension will be the more restrictive of initial entry plus an additional 24 hours or completion time as measured from discovery of the subsequent inoperability. Since this completion time is more restrictive it is the correct answer.

D INCORRECT: This would be the correct completion time as measured from discovery of the subsequent inoperability. However, TS Section 1.3, "Completion Times" states that the completion time will be the more restrictive of either initial entry plus an additional 24 hours or completion time as measured from discovery of the subsequent inoperability. Since initial entry plus 24 hours is more restrictive, this is incorrect.

Technical Reference(s): TS 1.3 (Attach if not previously provided)

TS 3.6.4.3 , TS 3.0.3 (Including version / revision number)

Proposed references to be provided to applicants during examination: **U1/2/3-TS 3.6.4.3 (NO Bases)**

Learning Objective: ________________ (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **X**

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

1.3 Completion Times

DESCRIPTION (continued)

Once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

However, when a subsequent division, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each division, subsystem, component or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Three SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during OPDRVs.	C.1	Place two OPERABLE SGT subsystems in operation.	Immediately
	<u>OR</u>		
	C.2	Initiate action to suspend OPDRVs.	Immediately
D. Two or three SGT subsystems inoperable in MODE 1, 2, or 3.	D.1	Enter LCO 3.0.3.	Immediately

(continued)

LCO 3.0.3 When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 2 within 10 hours;

b. MODE 3 within 13 hours; and

c. MODE 4 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, and 3.

Examination Outline Cross-reference:		
202001 Recirculation	Level	SRO
A2.10 (10CFR 55.43.5 - SRO Only)	Tier #	2
Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	Group #	2
• Recirculation pump seal failure	K/A #	202001A2.10
	Importance Rating	3.9

Proposed Question: **# 91**

Unit 1 Reactor Recirculation Pump 1A was removed from service due to the following indications:

- Number 2 seal pressure is 800 psig and rising slowly
- Recirculation Pump 1A Controlled Leakage is 1.4 gpm and rising slowly

Which ONE of the following completes the statement?

Recirculation Pump 1A parameters indicate a degraded Seal __**(1)**__. Per Tech Spec 3.4.1, "RPS Instrumentation," setpoints for Single Loop Operation must be incorporated within __**(2)**__.

A. **(1)** Number 1
(2) 12 hours

B. **(1)** Number 1
(2) 24 hours

C. **(1)** Number 2
(2) 12 hours

D. **(1)** Number 2
(2) 24 hours

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: Part 1 is correct. Seal Cavity #2 pressure approaching Seal Cavity #1 pressure is indicative of #1 Seal failure. The elevated controlled leakage rules out plugging of #2 RO. Part 2 is incorrect. RPS Instrumentation set points for Single Loop Operation must be incorporated within 24 hours of entering SLO per TS 3.4.1. The 12 hour time is recognizable as the time required to place an Inop channel in trip per RPS Instrumentation TS.

B **CORRECT**: Part 1 is correct. Seal Cavity #2 pressure approaching Seal Cavity #1 pressure is indicative of #1 Seal failure. The elevated controlled leakage rules out plugging of #2 RO. Part 2 is correct. RPS Instrumentation set points for Single Loop Operation must be incorporated within 24 hours of entering SLO per TS 3.4.1.

C INCORRECT: Part 1 is incorrect. Seal Cavity #2 pressure would be less than half of Seal Cavity #1 pressure if #1 Seal failed. Part 2 is incorrect. RPS Instrumentation set points for Single Loop Operation must be incorporated within 24 hours of entering SLO per TS 3.4.1. The 12 hour time is recognizable as the time required to place an Inop channel in trip per RPS Instrumentation TS.

D INCORRECT: Part 1 is incorrect. Seal Cavity #2 pressure would be less than half of Seal Cavity #1 pressure if #1 Seal failed. Part 2 is correct. RPS Instrumentation set points for Single Loop Operation must be incorporated within 24 hours of entering SLO per TS 3.4.1.

Technical Reference(s): 1-OI-68 Rev 14 / TS 3.4.1 (Attach if not previously provided)

1-ARP-9-4A Rev 16 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______ (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 X

Comments: Reviewed 1-9-4A Rev17 issued 3/25/09. New revision has no impact on this question.

BFN	Panel 9-4	1-ARP-9-4A
Unit 1	1-XA-55-4A	Rev. 0016
		Page 33 of 47

RECIRC PUMP 1A
NO 1 SEAL
LEAKAGE ABN

1-FA-68-62

25

(Page 1 of 3)

Sensor/Trip Point:

1-FIS-068-0062 ≥ 0.9 gpm rising
≤ 0.5 gpm lowering

Sensor Location: Recirculation Pump 1A
Drywell Elevation 549.2

Probable Cause:
A. Plugging of No. 1 and/or No. 2 RO (controlled pressure breakdown orifice).
B. Failure of no. 1 seal.
C. Reactor Pressure < 450 psig (Alarm resets at > 650 psig).

Automatic Action: None

Operator Action:
A. **DETERMINE** initiation cause by comparing No.1 and 2 seal cavity pressure indicators on 1-9-4 or ICS. ☐
- Plugging of No. 1 RO - No. 2 seal cavity pressure indicator drops toward zero, and control leakage lowers to ≤ 0.5 gpm. ☐
- Plugging of No. 2 RO - No. 2 seal pressure approaches No. 1 seal pressure and control leakage lowers to ≤ 0.5 gpm. ☐
- Failure of No. 1 seal - No. 2 seal pressure is greater than 50% of the pressure of No. 1. The controlled leakage will be ≥ 0.9 gpm. ☐
- Failure of No. 2 seal - No. 2 seal pressure is less than 50% of the No. 1 seal. ☐

BASES (continued)

ACTIONS

A.1

With the requirements of the LCO not met, the recirculation loops must be restored to operation with matched flows within 24 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. The loop with the lower flow must be considered not in operation. Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to the operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

Examination Outline Cross-reference:

215002 RBM

A2.05 (10CFR 55.43.5 - SRO Only)

Ability to (a) predict the impacts of the following on the ROD BLOCK MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

- RBM high or inoperable: BWR-3,4,5

Level	[redacted]	SRO
Tier #		2
Group #		2
K/A #	215002A2.05	
Importance Rating	[redacted]	3.3

Proposed Question: **# 92**

Unit 3 is at 75% Reactor Power for a sequence exchange. Control Rod 26-59 is being withdrawn from position 00 to 48 when the following annunciators / indications are received:

- RBM HI/INOP, (3-9-5A, Window 24)
- CONTROL ROD WITHDRAW BLOCK, (3-9-5A, Window 7)
- LPRM HI status light adjacent to Control Rod 26-59 is blinking on/off
- APRM Power indication momentarily spiked then stabilized at 78%

Based on the above conditions, which ONE of the following completes the statement?

The crew must execute __**(1)**__ **AND** reduce __**(2)**__.

A. **(1)** 3-AOI-85-1, "Rod Drop Accident,"
   (2) Core Flow to 50 to 60%.

B. **(1)** 3-AOI-85-7, "Mispositioned Control Rod,"
   (2) Core Flow to 50 to 60%.

C. **(1)** 3-AOI-85-1, "Rod Drop Accident,"
   (2) Reactor Power by 10%.

D. **(1)** 3-AOI-85-7, "Mispositioned Control Rod,"
   (2) Reactor Power by 10%.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: Part 1 = correct – See D. Part 2 = incorrect - 3-AOI-85-1, "Rod Drop Accident" does not direct lowering Core Flow to 50 to 60%. This direction is plausible and recognizable as subsequent action in other AOIs.

B INCORRECT: Part 1 and 2 = incorrect as detailed in C.

C **CORRECT**: Part 1 = correct - Based on the indications received, the Candidate should recognize the symptoms of a Rod Drop Accident and enter 3-AOI-85-1, "Rod Drop Accident." Part 2 = correct – Subsequent action 4.2 of 3-AOI-85-1, "Rod Drop Accident" directs if no Scram occurred to reduce Reactor Power by 10% from the power prior to the event.

D INCORRECT: Part 1 = incorrect – Symptoms described clearly indicate a Rod Drop Accident has occurred. Although the dropped control rod is mispositioned, the appropriate guidance for this event is provided in 3-AOI-85-1, "Rod Drop Accident". Part 2 = correct as detailed above.

Technical Reference(s): 3-AOI-85-1 Rev 6 (Attach if not previously provided)

(Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ______________ (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

BFN Unit 3	Rod Drop Accident	3-AOI-85-1 Rev. 0006 Page 4 of 8

1.0 PURPOSE

This abnormal operating instruction provides symptoms, automatic actions and operator actions for a control rod drop accident.

2.0 SYMPTOMS

A. Reactor power rise.

B. LPRM HI status lights around dropping control rod in alarm on Full Core Display.

C. Reactor high neutron flux scram with any of the following annunciators on Panel 3-9-4 and 3-9-5 in alarm:

1. IRM CH A,C,E,G HI-HI/INOP (3-XA-55-5A, Window 33)
2. IRM CH B,D,F,H HI-HI/INOP (3-XA-55-5A, Window 34)
3. APRM HIGH/INOP OR OPRM TRIP(3-XA-55-5A, Window 25)
4. REACTOR CHANNEL A AUTO SCRAM (3-XA-55-5B, Window 1)
5. REACTOR CHANNEL B AUTO SCRAM (3-XA-55-5B, Window 2)
6. NEUTRON MONITORING SYS HALF SCRAM (3-XA-55-4A, Window 16)
7. CONTROL ROD WITHDRAWAL BLOCK (3-XA-55-5A, Window 7)

BFN Unit 3	Rod Drop Accident	3-AOI-85-1 Rev. 0006 Page 6 of 8

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

None

4.2 Subsequent Actions

[1] **VERIFY** automatic actions. ☐

[2] **IF** NO scram occurs, **THEN**

REDUCE Reactor power by 10% from the power prior to event. ☐

[3] With concurrence of SM,

Examination Outline Cross-reference:		Level	SRO
233000 Fuel Pool Cooling/Cleanup		Tier #	2
G2.4.35 (10CFR 55.43.5 - SRO Only)		Group #	2
Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.		K/A #	233000G2.4.35
		Importance Rating	4.0

Proposed Question: **# 93**

Unit 1 is commencing a Refueling outage, with the following conditions existing:

- Shutdown Cooling in service on RHR Loop II
- Fuel Pool temperature 126 °F
- RHR Heat Exchanger outlet temperature 145 °F and steady
- Fuel Pool / Reactor Cavity gates are installed

Which ONE of the following completes the statement?

The Unit Supervisor is required to direct the Reactor Building AUO to __**(1)**__, which will __**(2)**__.

A. **(1)** assist in connecting the Unit 2 Fuel Pool Cooling to Unit 1, per 1-AOI-78-1, "Fuel Pool Cleanup System Failure,"
(2) introduce cooler water into the Fuel Pool Cooling System.

B. **(1)** raise the skimmer weirs inside the Spent Fuel Pool locally, per 1-OI-78, "Fuel Pool Cooling and Cleanup System,"
(2) allow more Fuel Pool Cooling System flow.

C. **(1)** verify FPC F/D BYP VLV 1A, 1-FCV-78-66, is closed locally, per 1-AOI-78-1, "Fuel Pool Cleanup System Failure,"
(2) raise Fuel Pool Cooling System flow.

D. **(1)** assist in establishing EECW makeup flow to the Spent Fuel Pool, per 1-OI-78, "Fuel Pool Cooling and Cleanup System,"
(2) introduce makeup water into the Fuel Pool Cooling System.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** part 1 = correct, 1-AOI-78-1, step 4.2[3.8] directs this activity. The gates are not removed, so SDC cannot assist in cooling the SFP. Part 2 = correct, temperature control assistance from Unit 2 will begin once the ½ Transfer gates are removed.

B INCORRECT: part 1 = incorrect, this is a subset step for 1-AOI-78-1, step 4.2[3.8]. They would be lowered to promote more flow. Part 2 = incorrect, raising would reduce flow.

C INCORRECT: part 1 = incorrect, 1-AOI-78-1, step 4.2[3.4] directs **opening** / bypassing the Demins, to raise system flow. Part 2 = correct, the intent is to raise system flow, but having the B/P closed does not achieve this.

D INCORRECT: part 1 = incorrect, used if SFP level is lowering. 1-AOI-78-1, step 4.2[2.6]. Part 2 = incorrect, makeup water is not needed, cooling is. This would provide some relief, due to dilution. Eventually would need to drain water to prevent overflow.

Technical Reference(s): 1-AOI-78-1 Rev 14 (Attach if not previously provided)

0-GOI-100-3A Rev 53 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source:	Bank #		
	Modified Bank #	295023G2.4.4	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	0407A	

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

BFN Unit 1	Fuel Pool Cleanup System Failure	1-AOI-78-1 Rev. 0014 Page 8 of 22

4.2 Subsequent Actions (continued)

[2.6] **IF** level cannot be restored, **THEN**

PERFORM Attachment 2 to provide EECW makeup flow through hoses to Unit 1 Spent Fuel Pool. ☐

[2.7] **PERFORM** visual inspection of fuel pool piping on all levels of reactor building, inspecting for leaks. ☐

[2.8] **IF** a line break is detected, **THEN**

UNLOCK and **CLOSE** the following on the refuel floor:

- POOL POOL DIFFUSER A INL VLV, 1-SHV-078-0524 ☐
- POOL POOL DIFFUSER B INL VLV, 1-SHV-078-0525 ☐

[3] **IF** fuel pool cooling system failure is from loss of cooling, **THEN**

PERFORM the following:

[3.1] **START** idle Fuel Pool Cooling Pump 1B(1A). ☐

[3.2] **ATTEMPT** to re-start the tripped Fuel Pool Cooling Pump 1A(1B). ☐

[3.3] **VERIFY** RBCCW System is operating and **REFER TO** 1-OI-70. ☐

[3.4] **BYPASS** fuel pool filter demineralizer to raise flow by performing the following:

THROTTLE OPEN FPC F/D 1A BYP VLV A(B), 1-FCV-078-0066(0065), using local control switch 1-HS-078-0066B(0065B), to maintain pump discharge pressure greater than 130 psig as indicated on FPC PMP 1A(1B) DISCH PRESS LOW, 1-PIS-078-0011(0016), on FUEL POOL PUMP PANEL, 1-LPNL-925-0016. ☐

[3.5] **IF** Fuel Pool/Reactor Cavity gates are removed, **THEN**

PERFORM the following:

- **VERIFY** RWCU System in service in accordance with 1-OI-69. ☐
- **PLACE** RHR System in Shutdown Cooling mode in accordance with 1-OI-74. ☐

[3.6] **DIRECT** the STA to ESTIMATE the time for the fuel pool temperature to rise to 125°F and 150°F, using the heat-up rates as provided in Attachment 1, Table 1 at least once per shift UNTIL Fuel Pool cooling is restored. ☐

[3.7] **PLACE** RHR supplemental fuel pool cooling mode in operation and **REFER TO** 1-OI-74 as necessary to maintain fuel pool temperature less than 125°F as indicated on RHR/FUEL POOL CLG TEMPERATURE recorder, 1-TR-74-80, on Panel 9-21. ☐

Original Question:

295023G2.4.4	1	Selected 0 / Select this one	Multiple choice: single	4	1	1	

RO Tier	SRO Tier	Keyword	Cog Level	Source	Exam	Test	Author/Reviewer
T1/G1		REFUEL	C/A 4.0/4.3	295023G2.4.4	BF05301	R	TCK/RM

Arial 12

Unit 2 is in a refueling outage and a fuel shuffle has just been completed. The following conditions exist at this time:

- Shutdown cooling in service on Loop II
- Approximately half the core unloaded to the Spent Fuel Pool
- Fuel pool temperature 126°F
- RHR Heat Exchanger outlet temperature 145°F and steady
- All SGT Systems have just been declared INOPERABLE

Based on the above conditions, which ONE of the following procedures should be implemented?

A. 2-AOI-30B-1, Reactor Building Ventilation Failure.

B. 2-AOI-74-1, Loss of Shutdown Cooling.

C✓ 2-AOI-78-1, Fuel Pool Cleanup System Failure.

D. 2-AOI-79-1, Fuel Damage during Refueling.

Examination Outline Cross-reference:

G2.1.13 (10CFR 55.43.5 – SRO Only)

Knowledge of facility requirements for controlling vital/controlled access.

Level		SRO
Tier #		3
Group #		
K/A #	G2.1.13	
Importance Rating		3.2

Proposed Question: **# 94**

Given the following plant conditions:

- An after hours emergency has occurred at BFN
- The Shift Manager (SM) has determined that a Maintenance individual with special skills is required inside the Protected Area of the plant
- When contacted the individual informs the SM that he has been drinking alcohol within the past 5 hours but feels good enough to report to work

Which ONE of the following describes the decision made by the SM?

A. The individual shall **NOT** be called in.

B. The individual can be called in **AND NOT** tested under special exemption of SPP-1.2, "Fitness-For-Duty."

C. The individual can be called in **AND** shall be tested when on site. Blood alcohol level must be less than 0.02% to allow the individual access to the Protected Area.

D. The individual can be called in **AND** shall be tested when on site. Blood alcohol level must be less than 0.04% to allow the individual access to the Protected Area.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: Only if the individual declares that he/she is unfit, will the call-in attempt be aborted. FFD010, section FFD-09 provides guidance.

B INCORRECT: This procedure does not give guidance to exempting the test.

C INCORRECT: 0.02% is the limit for **scheduled** work. Per section FFD-09, 0.04% is the cutoff for unscheduled work.

D **CORRECT:** Per section FFD-09, 0.04% is the cutoff for unscheduled work.

Technical Reference(s):	FFD-10 Rev 10	(Attach if not previously provided)
	SPP-1.2 Rev 12	(Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________________ (As available)

Question Source: Bank #
Modified Bank # X (Note changes or attach parent)
New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

FFD010
Revision 10
Page 7 of 36

FFD-09 **FFD POLICY**

TVAN's FFD policy dictates that you **SHALL**:

- Report to work fit for duty, unimpaired from alcohol or drugs. All employees and workers are prohibited from reporting to a TVAN work location under the influence of illegal drugs or alcohol at or above the limits described in SSP-1.2.

- Abstain from alcohol for at least five (5) hours preceding regularly scheduled work and long enough to ensure blood alcohol content (BAC) is less than 0.02 percent. BAC levels of 0.02-0.039 are prohibited by TVAN policies and procedures. A BAC of 0.04 will be considered a positive alcohol test result. See Table of Minimum Penalties for Violations of the FFD Program for more specific information.

- BAC levels of 0.01 – 0.019 will be reviewed by Nuclear Security, Nuclear Access Services, to determine if further actions are necessary.

- Every employee is expected to report to work **FIT FOR DUTY**.

  - Consumption of alcohol is prohibited:

    * If a person leaves work with the intent to return that day, or shift.
    * If a person will be driving a TVA vehicle.
    * If a person is scheduled to report to work within five hours.

◊ <u>Policy Exception</u>

* Employees at the sites or projects when called in for unscheduled work must be asked if they are fit for duty **<u>AND</u>** if they have consumed alcohol within the past five hours.

* Emergency Response Center personnel who are called by automated electronic systems are responsible for advising the center if they believe they are unfit to report for duty **<u>AND</u>** if they have consumed alcohol within the last five hours.

* The section/department manager is responsible for ensuring these questions are asked and the answers are documented by the assigned caller on a form SPP-1.2-1 or similar type form. Documentation of responses in day timers (planners) is not acceptable.

* During emergencies, as defined by the responsible supervisor, the abstinence period may be waived provided a determination of fitness for duty is made using a saliva test or breath analysis by Nuclear Security or by other personnel trained to administer the test as approved by Nuclear Security.

* In most cases where an employee has consumed alcohol within five hours of reporting, but feels that he or she is fit for duty, the person should not be asked to report.

* After careful consideration, a manager may determine that there is a <u>critical</u> need for a specific individual who feels he/she is fit for duty even though alcohol has been consumed within 5 hours of the requested report time.

* Any individual indicating that he/she is not fit for duty when called shall not be requested to report for unscheduled work.

* No sanctions shall be applied for a positive breath analysis when an employee is called in for unscheduled work if the employee reported the alcohol use at the time he/she was called in.

* **An individual with a blood alcohol concentration of greater than or equal to 0.04% WOULD NOT be considered fit for duty.**

NPG Standard Programs and Processes	Fitness-For-Duty	SPP-1.2 Rev. 0012 Page 53 of 74

3.15 Call-in for Unscheduled Work

3.15.1 General

NOTE
Contractor employees will not be requested to report if alcohol has been consumed within the past five (5) hours except in true emergency situations.

A. All individuals are expected to not consume alcohol for at least five (5) hours prior to reporting for SCHEDULED work and to report fit and within FFD guidelines. If called for unscheduled work the individual's suitability for work must be determined.

The following must be done whenever an individual is being called in for unscheduled work.

1. The caller will ask and will document (on Form SPP-1.2-1 or similar-type form) the individual's responses to BOTH the following questions.

   a. Are you fit to report to work/FFD?

   b. Have you consumed alcohol within the past five (5) hours?

2. The individual must advise the caller and the supervisor if he or she believes that he or she is unfit to report for work and if he or she has consumed alcohol within the past five (5) hours.
3. If the answer to the "Are you fit to report to work/FFD?" question is "no", then the individual should not be requested to report for unscheduled work. The caller will document the individual's stated reason(s) for being unfit to report.
4. If the answer to the "Have you consumed alcohol within the past five (5) hours?" question is "yes", the caller will document how much alcohol was consumed and when. The caller and supervisor will then decide whether or not to have the individual report for work.
5. If the answer to alcohol consumption question is "yes" (reference 4 above) and the caller and supervisor request that the individual report for work, then Nuclear Security on site shall be notified and be requested to administer a saliva test to the individual. This test shall be administered as soon as the individual arrives on site. The test shall be administered **outside** the Protected Area.
6. If the test results are 0.039 or below, the individual's responsible supervisor shall determine if the individual can be permitted to work. The individual will not be subject to disciplinary action.
7. If the results are 0.040 or above, the individual will NOT be permitted to work. This will NOT be considered a positive test for FFD purposes.

Original Question: 0610 Audit SRO Only,

Given the following plant conditions:

- An emergency has occured at BFN requiring entry into the EPIPs
- The SM has determined that a maintenance individual with special skills is required onsite.
- When contacted the individual informs the SM that he has been drinking alcohol within the past 5 hours but feels good enough to report to work

Which ONE of the following describes the decision made by the SM?

A. The indiviual shall not be called in.

B. The indiviual can be called in. They shall tested when they arrive onsite, if their blood alcohol level is below .04% they may be allowed access to the controlled area.

C✓ The indiviual can be called in. They shall tested when they arrive onsite, if their blood alcohol level is below .02% they may be allowed access to the controlled area.

D. The indiviual can be called in without testing with special exemption allowed in 'SPP-1.3 Plant Access and Security' and '10 CFR 73.56, NRC Order for Compensatory Measures'.

Examination Outline Cross-reference:

G2.1.4 (10CFR 55.43.2 – SRO Only)

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Level	RO	SRO
Tier #		3
Group #		
K/A #	G2.1.4	
Importance Rating		3.8

Proposed Question: **# 95**

You are a Shift Manager (SM) that has been on sick leave. You have returned to the site and have commenced OPDP-10, "License Status Maintenance, Reactivation and Proficiency for Non-Licensed Operators," Appendix-E, "Browns Ferry Nuclear Plant Requirements for Returning an Inactive License to Active Status."

Which ONE of the following sets of conditions correctly describes the elements required for re-activation, if you are being assigned to a crew?

1. Perform 40 hours break-in under the current SM.
2. As part of the plant tour, discuss focus areas with the Operations Superintendent.
3. Prior to the break-in period, verify completion of a physical within the last 12 months.
4. Obtain signed approval from the Operations Manager to resume licensed activities.
5. Anytime during the break-in period, verify a simulator evaluation has been performed within the last 12 months.
6. Prior to the break-in period, meet with the Operations Training Manager **AND** Operations Superintendent to discuss standards.

A. **ONLY** 1, 3, 6.

B. **ONLY** 1, 4, 5.

C. **ONLY** 2, 3, 6.

D. **ONLY** 2, 4, 5.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (1) correct, OPDP-10, 4.0.B.1.a directs current SM (3) correct, OPDP-10, Appendix F. (6) correct, OPDP-10, Appendix E, 4.0.A.1, OTM & Ops Superintendent required.

B INCORRECT: (1) correct, OPDP-10, 4.0.B.1.a directs current SM. (4) incorrect, OPDP-10, Appendix F, Plant Manager not Ops Manager. (5) incorrect, OPDP-10, 4.0.A.1, prior to, not during.

C INCORRECT: (2) incorrect, OPDP-10, 4.0.10.c, SM not Ops Superintendent. (3) correct, OPDP-10, Appendix F. (6) correct, OPDP-10, Appendix E, 4.0.A.1, OTM & Ops Superintendent required.

D INCORRECT: (2) incorrect, OPDP-10, 4.0.10.c SM not Ops Superintendent. (4) incorrect, OPDP-10, Appendix F, Plant Manager not Ops Manager. (5) incorrect, OPDP-10, 4.0.A.1, prior to, not during.

Technical Reference(s): OPDP-10 rev 0 (Attach if not previously provided) (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source:
Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41
55.43 X

Comments:

NPG Standard Department Procedure	License Status Maintenance, Reactivation and Proficiency for Non-Licensed Positions	OPDP-10 Rev. 0000 Page 15 of 30

Appendix C
(Page 2 of 3)

Browns Ferry Nuclear Plant Requirements for Maintaining Active License Status

4.0 INSTRUCTIONS

Appendix E
(Page 1 of 5)

Browns Ferry Nuclear Plant Requirements for Returning an Inactive License to Active Status

1.0 PURPOSE

This appendix is intended to provide additional guidance, to return a licensed individual to an active status.

4.0 INSTRUCTIONS

A. The following guidelines are to be used when reactivating a license:

1. Prior to standing the minimum of 40 hours of shift functions, the licensed individual shall meet with the Operation Training Manager and the Operations Superintendent to discuss his/her current status and any standards and/or expectations. For certain individuals, additional requirements may be imposed (greater than those required by law) if directed by the Operations Superintendent.

Appendix E
(Page 4 of 5)

Browns Ferry Nuclear Plant Requirements for Returning an Inactive License to Active Status

4.0 INSTRUCTIONS (continued)

10. As a minimum, the following shall be completed to satisfy the plant tour requirement:

a. Review of Control Room logs and equipment status in order to ascertain current plant status and configuration.

b. Review of radiological conditions in the plant.

c. Tour of accessible plant areas where significant modifications have occurred or major maintenance activities are occurring, with special attention if safety-related systems are involved.

(1) Prior to beginning the tour, a discussion should be held with the Shift Manager to obtain guidance on which areas to focus on during the plant tour.

(2) Document areas discussed on Appendix H and have the Shift Manager sign that the discussion was held.

(3) The plant tour will be performed by the individual accompanied by a Licensed Reactor Operator or a Senior Reactor Operator, as applicable, and logged in the Narrative Log.

11. Additionally, the following are considerations for performing the plant tour:

a. ALARA will be considered when deciding which areas of the plant to tour.

b. The individual should walkdown additional areas, as he/she deems appropriate, to ensure he/she is comfortable with plant conditions.

B. Returning an Inactive Shift Manager to active Status

1. Before resumption of independent Shift Manager duties, the Plant Manager or designee will certify the following:

a. The individual has completed 40 hours of break-in under the current Shift Manager.

b. Prior to a Shift Manager being assigned to an on-shift crew, that individual should attend simulator training with the other licensed members of that crew.

c. Documentation of completion shall be forwarded to Operations Training Manager for retention.

Appendix F
(Page 1 of 1)

Return to Active License Status Certification (BFN)

To: Operations Training Manager

From: Operations Superintendent ________ Date

NAME: ______________________________

A. Licensee requalification training is current, including a simulator evaluation within the past 12 months in the position(s) to be assumed and the licensee has had a physical in the last two years. (To be verified prior to standing the 40 hours of shift functions under instruction.)

______________________________ Operational Training Manager Date: ____ / ____ / ____

B. The qualifications and status of the licensed individual listed above are current and valid, and Standards and Expectations have been discussed, prior to standing the 40 hours of shift functions under instruction.

______________________________ Operational Superintendent Date: ____ / ____ / ____

C. If the licensee has a medical restriction requiring corrective lenses, the licensee will verify that he/she has the proper corrective lenses required to Don SCBA available while performing license duties (N/A if corrective lenses are not required).

______________________________ Licensee Date ____ / ____ / ____

D. The above licensed individual has completed at least 40 hours of shift functions under the direction of an operator or senior operator, as appropriate, including a complete tour of the plant accompanied by a licensed RO or SRO, as applicable, and review of all required shift turnover procedures.

______________________________ Licensee Date: ____ / ____ / ____

______________________________ Shift Manager Date: ____ / ____ / ____

______________________________ Operations Superintendent Date: ____ / ____ / ____

______________________________ Operations Manager Date: ____ / ____ / ____

E. The above licensed individual is authorized to resume licensed activities.

______________________________ Plant Manager Date: ____ / ____ / ____

F. Complete and Attach OPDP-10-1, Licensee Documentation Form (SRO & RO) as the cover sheet for this documentation.

______________________________ Licensee Date ____ / ____ / ____

cc: Operations Manager
Training File
BROWNS FERRY NUCLEAR PLANT
REQUIREMENTS FOR RETURNING AN INACTIVE LICENSE TO ACTIVE STATUS

Examination Outline Cross-reference:

G2.2.19 (10CFR 55.43.5 – SRO Only)

Knowledge of maintenance work order requirements.

Level		SRO
Tier #		3
Group #		
K/A #	G2.2.19	
Importance Rating		3.4

Proposed Question: **# 96**

You are about to review electronic Work Orders (WOs) that have just been generated.

Which ONE of the following sets of conditions correctly describes the elements that must be performed by you?

1. Review existing WOs for duplicate deficiencies.
2. Evaluate each WO within 24 hours of initiation.
3. Determine the type of maintenance (corrective, elective, etc.).
4. Review WOs that deal with material procurement.
5. Determine if work can be performed as 'Toolpouch maintenance'.
6. Consider whether the equipment should be restricted.

A. **ONLY** 2, 3, 6.

B. **ONLY** 1, 3, 5.

C. **ONLY** 2, 4, 6.

D. **ONLY** 1, 4, 5.

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT:** (2) correct, Per SPP-6.1Step 3.3 A [1], requires SRO process WO w/I 24 hours. (3) correct, Per SPP-6.1Step 3.3.A [2] Non-emergency (6) correct, SPP-6.1 Step 3.3 A [3] SM/Designee performs.

B INCORRECT: (1) incorrect, Per SPP-6.1 Step 3.2 [C], WO initiator performs review for duplicate. (3) correct, Per SPP-6.1Step 3.3.A [2] Non-Emergency (5) incorrect, Per SPP-6.1 Step 3.2 [A], WO initiator determines 'toolpouch maintenance' not SRO.

C INCORRECT: (2) correct, Per SPP-6.1Step 3.3 A [1], requires SRO process WO w/I 24 hours. (4) incorrect, Per SPP-6.1 3.0 note, does not require OPS review. (6) correct, SPP-6.1 Step 3.3 A [3]SM/Designee performs.

D INCORRECT: (1) incorrect, Per SPP-6.1 Step 3.2 [C], WO initiator performs review for duplicate. (4) incorrect, Per SPP-6.1 3.0 note, does not require OPS review. (5) incorrect, Per SPP-6.1 Step 3.2 [A], WO initiator determines 'toolpouch maintenance' not SRO.

Technical Reference(s): SPP-6.1 Rev 6 (Attach if not previously provided)

(Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

NPG Standard Programs and Processes	Work Order Process Initiation	SPP-6.1 Rev. 0006 Page 6 of 19

3.0 INSTRUCTIONS

NOTE
WOs created strictly for the purpose of material procurement require only the information on Screen 1 of the WO to be loaded and do not require Supervisor or Operations reviews. These WOs should be loaded with a maintenance type of MT (Material Tracking Only).

3.2 **Initiation of Work**

Initiator

A. Before initiating a WO, the employee should evaluate the work to determine if it can be performed as "toolpouch maintenance." Refer to Appendix A for the definition of minor maintenance, criteria for evaluation, and performance of toolpouch maintenance. If the work meets the toolpouch maintenance criteria, perform the activity in accordance with Appendix A. No documentation or approval is required.

B. If the work cannot be performed as "toolpouch maintenance," initiate a WO in accordance with this procedure.

C. If EMPAC is available, the WO will be initiated as follows:

1. Review the existing open WOs in the system by entering the UNID in EMPAC to determine if the symptom, deficiency, or work requested already has an open WO. If no open WO exists, proceed to initiate the WO by entering the following information into the system. The following data must be entered as a minimum: If an open WO exists, exit the process.

2. Initiate the WO by entering the following information into the system. The following data must be entered as a minimum:

SM or Designee

A. General Requirements

1. WOs shall be evaluated and prioritized within 24 hours.

2. Evaluate the WO for Technical Specifications and for Independent Spent Fuel Storage Installation Certificate of Compliance (ISFSI CoC) operability considerations. If the WO describes a deficiency that causes immediate entry into a Technical Specifications or ISFSI CoC Limiting Condition of Operation (LCO), take appropriate action to enter and comply with the LCO.

3. Consider whether equipment service should be restricted because of abnormal operating conditions.

4. If the WO describes a deficiency that requires notification of the NRC or other outside government agency such as the Environmental Protection Agency or another federal, state, or local government agency, make the necessary notifications.

SM or Designee

A. Non-Emergency WOs

1. If the work described justifies a WO, prioritize the WO in accordance with SPP-7.1.

2. Determine the type of maintenance (corrective, elective, enhancement, other, preventive, inspection and testing or material tracking) and route to the appropriate maintenance group and to scheduling in parallel for processing in accordance with SPP-7.1. Refer to the definitions in SPP-7.1 for determining maintenance type.

3. Refer to Appendix A for the definition of minor maintenance.

4. If the work described does not justify a WO, disapprove the WO. The reason for disapproval should be indicated.

Examination Outline Cross-reference:	Level	[redacted]	SRO
G2.2.23 (10CFR 55.43.2 – SRO Only)	Tier #	[redacted]	3
Ability to track Technical Specification limiting conditions for operations.	Group #	[redacted]	[redacted]
	K/A #	G2.2.23	
	Importance Rating	[redacted]	4.6

Proposed Question: **# 97**

Unit 3 is operating at 10% Reactor Power, with power ascension in progress following a Refueling outage.

The following discoveries were made by the Reactor Engineer:

- Control Rod 30-31 Scram time to position 06 was 3.37 seconds
- Control Rod 26-27 Scram time to position 06 was 7.1 seconds
- Control Rod 22-23 Scram time to position 46 was 0.47 seconds

Assuming **NO** Operator actions have yet been taken **AND** based on the above information, which ONE of the following describes the correct Tech Spec applicability?

[REFERENCE PROVIDED]

A. **ALL** three Rods are SLOW; Tech Spec 3.1.4 Condition 'A' entry is applicable.

B. **ALL** three Rods are SLOW; Tech Spec 3.1.4 is applicable, with Condition 'A' **NOT** required.

C. Rods 22-23 **AND** 30-31 are SLOW; an Information LCO for Tech Spec 3.1.4 is applicable. Rod 26-27 is INOPERABLE, Tech Spec 3.1.3 Condition 'D' entry is applicable.

D. Rods 22-23 **AND** 30-31 are SLOW; Tech Spec 3.1.4 is applicable, with Condition 'A' **NOT** required. Rod 26-27 is INOPERABLE, Tech Spec 3.1.3 Condition 'C' entry is applicable.

Proposed Answer: **D**

Explanation (Optional):

A INCORRECT: 22-23 and 30-31 are SLOW. 26-27 is INOPERABLE (see note 2 in TS 3.1.4). These rods are adjacent, but with 26-27 being INOPERABLE per T.S.3.1.4 Table 1 [note], the 2 SLOW rods are separated. Therefore LCO 3.1.4 'b' is NOT applicable.

B INCORRECT: 30-31 and 22-23 are SLOW. 26-27 is INOPERABLE (see note 2 in TS 3.1.4). These rods are adjacent, but with 26-27 being INOPERABLE per T.S.3.1.4 Table 1 [note], the 2 SLOW rods are separated. Therefore LCO 3.1.4 'b' is NOT applicable. An Information LCO is inappropriate. An Active LCO would be generated for each Rod, but with < 13, and NO 2 SLOW are adjacent, NO actions are required, once < 13 and NO 2 SLOW are adjacent are verified.

C INCORRECT: 22-23 and 30-31 are SLOW. 26-27 is INOPERABLE (see note 2 in TS 3.1.4). These rods are adjacent, but with 26-27 being INOPERABLE per T.S. 3.1.4 Table 1 [note], the 2 SLOW rods are separated. Therefore LCO 3.1.4 'b' is NOT applicable. Power is at 10%, but NO indication of being out of pattern is given, therefore Condition 'D' of 3.1.3 is inappropriate to enter. Concerning T.S.3.1.3 Condition D, the candidate may think the SLOW rods may not comply with BPWS. If the actions of 3.1.3 'C' were taken, 'D' would apply, but the stem referenced no actions have yet been taken.

D **CORRECT:** 22-23 and 30-31 are SLOW. 26-27 is INOPERABLE (see note 2 in TS 3.1.4). These rods are adjacent, but with 26-27 being INOPERABLE per T.S. 3.1.4 Table 1 [note], the 2 SLOW rods are separated. Therefore LCO 3.1.4 'b' is NOT applicable. An Active LCO would be generated for each Rod, but with < 13, and NO 2 SLOW are adjacent, NO actions are required, once < 13 and NO 2 SLOW are adjacent are verified. T.S.3.1.3 Condition C is appropriate for 26-27.

Technical Reference(s): T.S.3.1.3 , T.S.3.1.4 (Attach if not previously provided)

3-OI-85 Rev 65 (Including version / revision number)

Proposed references to be provided to applicants during examination: **Tech Spec 3.1.3, 3.1.4 (NO BASES) & 3-OI-85 Illustration 3**

Learning Objective: ____________ (As available)

Question Source: Bank #

Modified Bank # 0707 Audit # 92 (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis **X**

10 CFR Part 55 Content: 55.41

55.43 **X**

Comments:

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4

a. No more than 13 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and

b. No more than 2 OPERABLE control rods that are "slow" shall occupy adjacent locations.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

---NOTES---

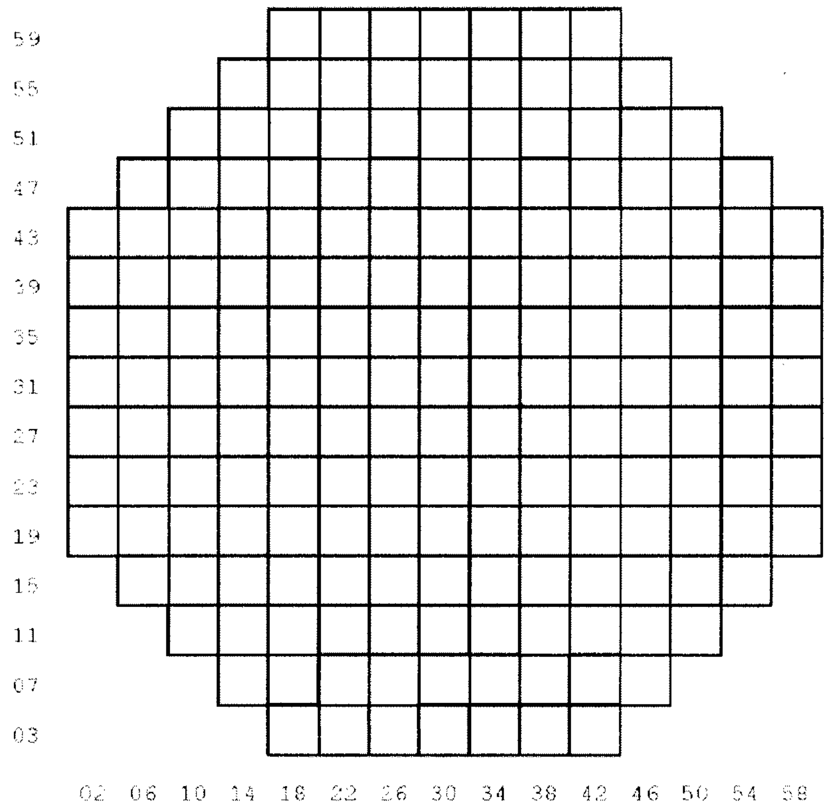
1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 06. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES[a][b] (seconds) REACTOR STEAM DOME PRESSURE $\geq$ 800 psig
46	0.45
36	1.08
26	1.84
06	3.36

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure, when < 800 psig are within established limits.



NAME (print) INITIALS

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	<u>AND</u>		
	A.4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	-------------NOTE------------ RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ---------------------------------	
		Fully insert inoperable control rod.	3 hours
	<u>AND</u>		
	C.2	Disarm the associated CRD.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. -------------NOTE------------ Not applicable when THERMAL POWER > 10% RTP. -------------------------------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1 Restore compliance with BPWS. OR D.2 Restore control rod to OPERABLE status.	4 hours 4 hours
E. Required Action and associated Completion Time of Condition A, C, or D not met. OR	E.1 Be in MODE 3.	12 hours

Comp/Mem	Tier/Group	Sys/Procedure#	Objective	KA#	RO/SRO KA Rating	RO/SRO Level	Date Create/Modify
COMP	2 / 2	CRDM / 085		2.2.40	3.4 / 4.7	SRO	7/14/08-BKC

Unit 3 is operating at 40% power, with power ascension in progress following a Refueling outage.

The following discoveries were made by the Reactor Engineer:

- Control Rod 30-31 Scram time to position 06 was 3.37 seconds.
- Control Rod 18-23 Scram time to position 06 was 7.1 seconds.
- Control Rod 18-19 Scram time to position 46 was 0.45 seconds.

Based on the above information, select the applicable Tech Spec actions to apply?

REFERENCE PROVIDED

A✓ Information LCO for Tech Spec 3.1.4 on Rod 30-31.
Active LCO for Tech Spec 3.1.3 on Rod 18-23.

B. Active LCO for Tech Spec 3.1.4 on Rod 30-31.
Information LCO for Tech Spec 3.1.3 on Rod 18-23.

C. Information LCO for Tech Spec 3.1.4 on Rod 18-19.
Information LCO for Tech Spec 3.1.3 on Rod 18-23.

D. Active LCO for Tech Spec 3.1.4 on Rods 18-19 and 18-23.
Information LCO for Tech Spec 3.1.4 on Rod 30-31.

Examination Outline Cross-reference:

G2.3.11 (10CFR 55.43.4 – SRO Only)

Ability to control radiation releases.

Level	RO	SRO
Tier #		3
Group #		
K/A #	G2.3.11	
Importance Rating		4.3

Proposed Question: **# 98**

Unit 3 was operating at 100% Reactor Power, when a coolant leak in the Drywell caused a Reactor Scram. The following conditions are noted:

- **ALL** Control Rods fully inserted
- Drywell Pressure is 25.4 psig and rising slowly
- Suppression Chamber Pressure is 24 psig and rising slowly
- Suppression Pool Level is 15 feet
- MAIN STEAM LINE RADIATION HIGH-HIGH, (3-9-3A, Window 27), is in alarm

Given these conditions, which ONE of the following completes the statement?

Venting of the Primary Containment is accomplished per __________.

A. 3-EOI APPENDIX-12, “Primary Containment Venting,” irrespective of radioactive release rates.

B. 3-EOI-APPENDIX-13,”Emergency Venting Primary Containment,” irrespective of radioactive release rates.

C. 3-EOI APPENDIX-12, “Primary Containment Venting,” **ONLY** if radioactive release rates can be maintained below ODCM limits.

D. 3-EOI-APPENDIX-13, “Emergency Venting Primary Containment,” **ONLY** if radioactive release rates can be maintained below ODCM limits.

Proposed Answer: **C**

Explanation (Optional):

A INCORRECT: With given Suppression Chamber Pressure < 55#, App-12 is appropriate, but must maintain release rate < ODCM limit.

B INCORRECT: With given Suppression Chamber Pressure < 55#, App-12 is appropriate, but must maintain release rate < ODCM limit.

C **CORRECT:** Release rates are requires to be controlled per step 12 of APPENDIX-12.

D INCORRECT: With given Suppression Chamber Pressure < 55#, App-12 is appropriate, but must maintain release rate < ODCM limit. Plausible in that the candidate may incorrectly evaluate the given conditions

Technical Reference(s): 3-EOI-2 Rev 7 (Attach if not previously provided)

3-EOI Appendix-12 Rev 3

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ____________ (As available)

Question Source:		
	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	X
Question History:	Last NRC Exam	

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

3-EOI APPENDIX-12

PRIMARY CONTAINMENT VENTING

LOCATION: Unit 3 Control Room

ATTACHMENTS: 1. Vent System Overview
2. Post-LOCA Release Rate Table (√)

****

CAUTION

Stack release rates exceeding 1.4 x 10^7 µCi/s, or 0-SI-4.8.B.1.a.1 release fraction above 1.0 will result in ODCM release limits being exceeded.

****

3-EOI APPENDIX-12
Rev. 3
Page 2 of 8

****

CAUTION

Venting Primary Containment during CAD addition is outside the CAD system FSAR design basis.

****

5. IF ... While executing this procedure, CAD addition per SAMG-2, Step G-4 OR G-9, is to begin,
   THEN . BEFORE CAD is initiated, **PERFORM** Step 13 to secure the vent path. ____

NOTE: Venting may be accomplished using EITHER:

- 3-FIC-84-19, PATH B VENT FLOW CONT,

  OR

- 3-FIC-84-20, PATH A VENT FLOW CONT.

NOTE: Unless the TSC recommends otherwise, venting the Drywell DIRECTLY should be performed ONLY if the Suppression Chamber can NOT be vented.

- Release rates as determined below:

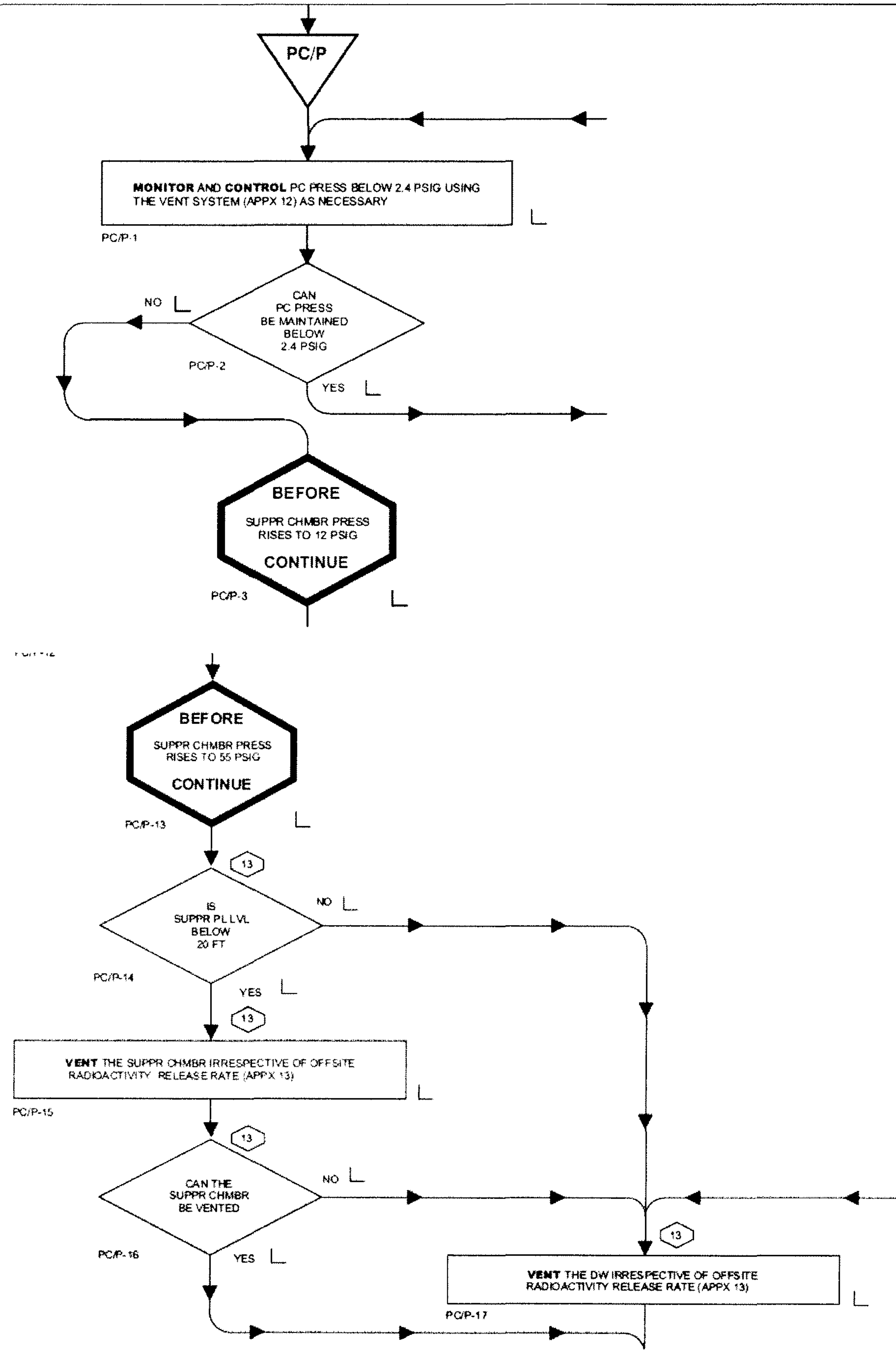
  i. IF. . .PRIMARY CONTAINMENT FLOODING per C-1, Alternate Level Control, is in progress,
     THEN. .**MAINTAIN** release rates below those specified in Attachment 2. ____

  ii. IF. . .Severe Accident Management Guidelines are being executed,
      THEN. .**MAINTAIN** release rates below those specified by the TSC SAM Team. ____

  iii. IF. . .Venting for ANY other reason than items i or ii above,
       THEN. .**MAINTAIN** release rates below
       - Stack release rate of 1.4 x 10^7 µCi/s ____

         AND

       - 0-SI-4.8.B.1.a.1 release fraction of 1. ____


PC/P
MONITOR AND CONTROL PC PRESS BELOW 2.4 PSIG USING THE VENT SYSTEM (APPX 12) AS NECESSARY
PC/P-1
CAN PC PRESS BE MAINTAINED BELOW 2.4 PSIG
NO
YES
PC/P-2
BEFORE SUPPR CHMBR PRESS RISES TO 12 PSIG CONTINUE
PC/P-3
BEFORE SUPPR CHMBR PRESS RISES TO 55 PSIG CONTINUE
PC/P-13
13
IS SUPPR PL LVL BELOW 20 FT
NO
YES
PC/P-14
VENT THE SUPPR CHMBR IRRESPECTIVE OF OFFSITE RADIOACTIVITY RELEASE RATE (APPX 13)
PC/P-15
CAN THE SUPPR CHMBR BE VENTED
NO
YES
PC/P-16
VENT THE DW IRRESPECTIVE OF OFFSITE RADIOACTIVITY RELEASE RATE (APPX 13)
PC/P-17

Examination Outline Cross-reference:

G2.3.7 (10CFR 55.43.4/5 – SRO Only)

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Level		SRO
Tier #		3
Group #		
K/A #	G2.3.7	
Importance Rating		3.6

Proposed Question: **# 99**

In accordance with RCDP-3, "Administration of Radiation Work Permits," which ONE of the following completes the statement?

The__**(1)**__ may authorize short term deviation from RWP requirements. The__**(2)**__must authorize immediate entry into areas during emergency situations **AND** approval requirements of RWP will be waived.

A. **(1)** Radiation Protection Supervisor
(2) Shift Manager

B. **(1)** Operations Unit Supervisor
(2) Shift Manager

C. **(1)** Radiation Protection Supervisor
(2) Radiation Protection Manager

D. **(1)** Operations Unit Supervisor
(2) Radiation Protection Manager

Proposed Answer: **A**

Explanation (Optional):

A **CORRECT**: Part 1 = correct – Per RCDP-3, "Administration of Radiation Work Permits", RADCON Supervision may authorize short term deviations (excluding regulatory and procedural deviations) from RWP requirements without revising the RWP. Part 2 = correct - Per RCDP-3, "Administration of Radiation Work Permits", in emergency situations where the Shift Manager authorizes immediate entry to an area, the prior approval requirements of a RWP will be waived.

B INCORRECT: Part 1 = incorrect but plausible in that Unit Supervisor is an SRO and member of shift management. Part 2 = correct for reasons detailed in A.

C INCORRECT: Part 1 = correct, as detailed in A. Part 2 = incorrect for reasons detailed in A and plausible in that RPM is the senior RP management on site.

D INCORRECT: Both parts are incorrect for reasons detailed in A.

Technical Reference(s): RCDP-3 Rev 2 (Attach if not previously provided)

(Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: ________ (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 X

Comments:

TVAN STANDARD DEPARTMENT PROCEDURE	ADMINISTRATION OF RADIATION WORK PERMITS	RCDP-3 Rev. 2 Page 6 of 11

3.6.5 RWPs describe the minimum requirements for performing radiological work. RADCON job coverage personnel or supervision may verbally require additional protective requirements for certain aspects of a work activity without revising the RWP. RADCON supervision may also authorize short-term deviations (excluding regulatory and procedural deviations) from RWP requirements without revising the RWP. Any deviations shall be documented in the RADCON Computer System RWP logbook.

3.6.7 The use of the RADCON Computer System to log RWP entries and exits may be suspended during emergency conditions. In emergency situations where the Shift Manager authorizes immediate entry to an area, the prior approval requirements of a RWP will be waived. If the RWP approval requirement is waived, RADCON and the personnel escorted by RADCON must comply with radiation protection procedures for entry into high radiation areas (i.e., RADCON individual is equipped with radiation dose rate monitoring device and provides positive control over activities within the area to include protective recommendations for the personnel being escorted for the duration of the emergency). Radiation surveillance by virtue of RADCON escort is considered to be continuous coverage. The RWP must be completed when the emergency entry is completed or the emergency is over. At the completion of the exempt work, actions will be taken to document (in the RADCON Computer System) the work, entries, exits, dose accrued, etc. Per WBN Tech specs and FSAR, this step does not apply to WBN.

Examination Outline Cross-reference:

G2.4.8 (10CFR 55.43.5 – SRO Only)

Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Level		SRO
Tier #		3
Group #		
K/A #	G2.4.8	
Importance Rating		4.5

Proposed Question: **# 100**

Unit 1 was operating in Mode 1 when an inadvertent Scram occurred. 1-EOI-1, "RPV Control," **AND** 1-C-5, "Level / Power Control," are currently being executed.

The following conditions exist:

- 1-EOI APPENDIX -1D, "Insert Control Rods Using Reactor Manual Control System," is being executed
- Reactor Water Level **AND** Pressure control are still being established
- Drywell Pressure is 4 psig and rising slowly
- MSIVs are closed
- Boron Injection is required **AND** Standby Liquid Control Pump 1A is operating
- DRYWELL RADIATION HIGH, (1-9-7C, Window 15), is alarming **AND** verified valid

The Unit Operator now reports that **ALL** control rods are inserted to position 00.

Upon exiting 1-C-5, which ONE of the following describes the correct set of steps to execute?

A. Stop Boron injection,
Exit the RC/Q leg of 1-EOI-1, **AND**
Enter 1-AOI-100-1, "Reactor Scram."

B. Continue Boron injection,
Exit the RC/Q leg of 1-EOI-1, **AND**
Enter 1-AOI-100-1, "Reactor Scram."

C. Stop Boron injection,
Exit 1-EOI-1, **AND**
Enter 1-GOI-100-12A,"Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations."

D. Continue Boron injection,
Exit 1-EOI-1, **AND**
Enter 1-GOI-100-12A,"Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations."

Proposed Answer: **B**

Explanation (Optional):

A INCORRECT: part 1= incorrect, RC/Q-2 step directs the stopping of Boron Injection if: The reactor will remain subcritical without Boron under all conditions and NOT required by other procedures. ARP 1-9-7C-W15 directs the continuation of Boron injection. Part 2 = correct, Only the RC/Q leg is exited and 1-AOI-100-1 is entered.

B **CORRECT:** part 1= correct, RC/Q-2 step directs the stopping of Boron Injection if: The reactor will remain subcritical without Boron under all conditions and NOT required by other procedures. ARP 1-9-7C-W15 directs the continuation of Boron injection. Part 2 = correct, Only the RC/Q leg is exited and 1-AOI-100-1is entered.

C INCORRECT: part 1= incorrect, RC/Q-2 step directs the stopping of Boron Injection if: The reactor will remain subcritical without Boron under all conditions and NOT required by other procedures. ARP 1-9-7C-W15 directs the continuation of Boron injection. Part 2 = incorrect, Only the RC/Q leg is exited and 1-AOI-100-1is entered. 1-GOI-100-12A is directed from other flow charts which are not applicable to the above set of conditions. The candidate is required to have sufficient knowledge of all EOI flow charts, thus distinguishing which chart directs what procedure(s) to enter.

D INCORRECT: part 1= incorrect, RC/Q-2 step directs the stopping of Boron Injection if: The reactor will remain subcritical without Boron under all conditions and NOT required by other procedures. ARP 1-9-7C-W15 directs the continuation of Boron injection. This level would be appropriate to use if all rods not full-in. Part 2 = incorrect, Only the RC/Q leg is exited and 1-AOI-100-1 is entered. 1-GOI-100-12A is directed from other flow charts which are not applicable to the above set of conditions. The candidate is required to have sufficient knowledge of all EOI flow charts, thus distinguishing which chart directs what procedure(s) to enter.

Technical Reference(s): 1-EOI-1 flowchart Rev 0, 1-9-7C Rev 20 (Attach if not previously provided)

OPL171.201 Rev 7 (Including version / revision number)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: V.B.7 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

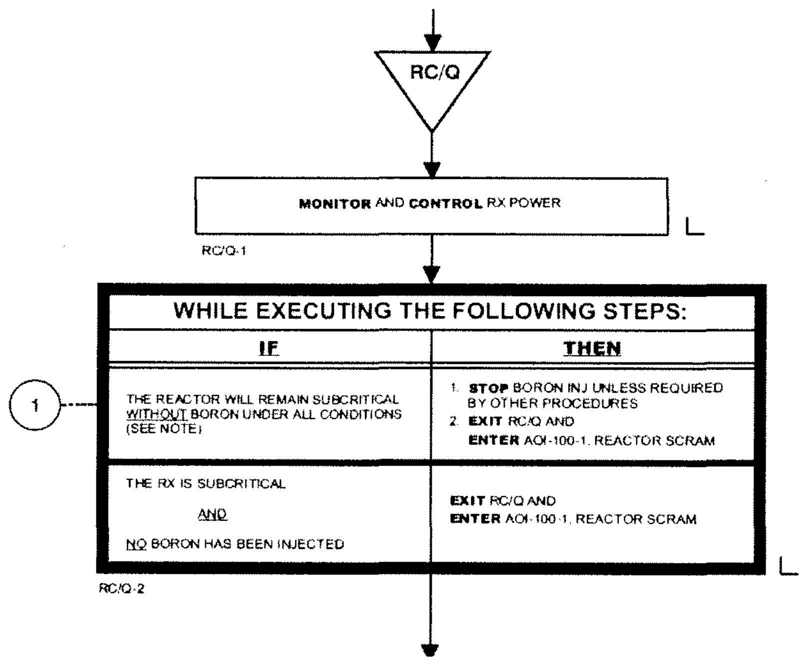
55.43 X

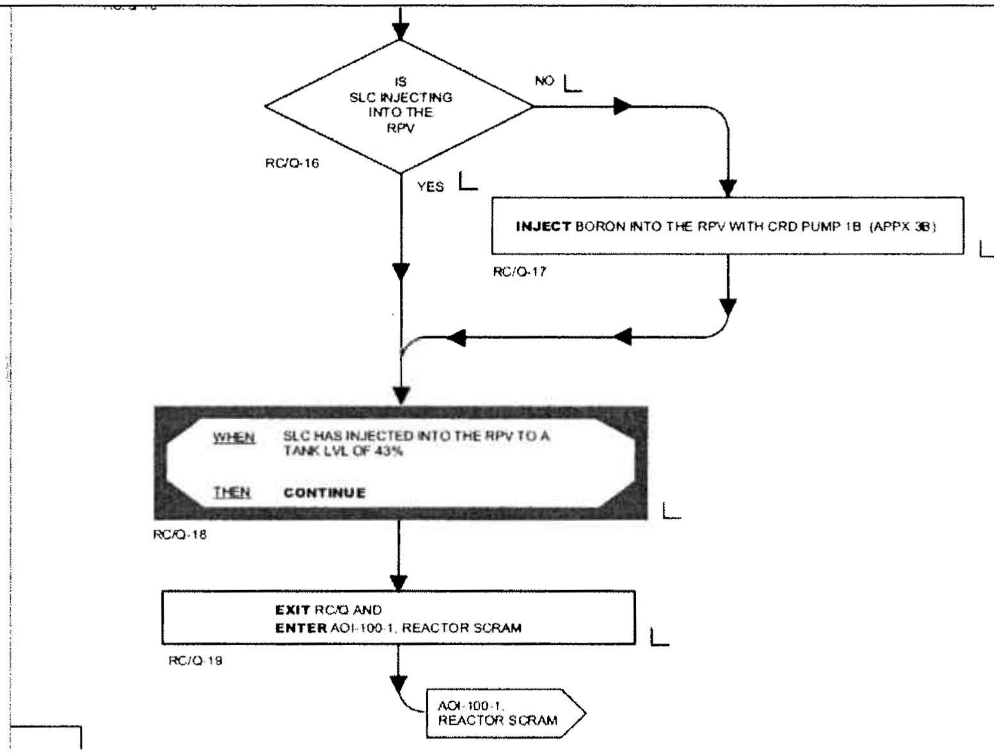
Comments:

OPL171.201 r7

1. Coordination of the EOIs and Other Plant Procedures

    a. Other procedures, such as AOIs, ARPs, EPIPs, etc., have event specific entry conditions and may be used to supplement EOIs. In some instances the EOIs will direct the operators to the unit operating procedures (OIs, GOIs, and AOIs) for completion of specific tasks. Usually, the EOIs direct the operators to specific EOI Appendices. The Appendices are specific task related procedures written to satisfy directives given within the EOIs.





NOTES

(1) THE REACTOR WILL REMAIN SUBCRITICAL <u>WITHOUT</u> BORON UNDER ALL CONDITIONS WHEN:

- ALL CONTROL RODS ARE INSERTED TO OR BEYOND POSITION 02

  <u>OR</u>

- ALL CONTROL RODS <u>EXCEPT</u> <u>ONE</u> ARE INSERTED TO OR BEYOND POSITION 00

  <u>OR</u>

- DETERMINED BY REACTOR ENGINEERING

(2) TSC STAFF MAY RECOMMEND AN ALTERNATE CURVE FOR STATION BLACKOUT PER 0-AOI-57-1A

DRYWELL RADIATION HIGH 1-RA-90-272, Window 15
(Page 2 of 2)

Operator Action: (Continued)

E. **IF** ALL the following conditions exist:
- Alarm is determined to be valid. ☐
- The reactor will remain subcritical 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 E, **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