QUESTION 76

With Unit 2 operating at 100% Reactor Power, a Normal Supply Breaker has tripped open AND suffered damage due to arcing. In addition to RHR AND Core Spray logic alarms, the following significant alarms result:

- HPCI LOGIC POWER FAILURE, (2-9-8A, Window 3)
- HPCI 120 VAC POWER FAILURE, (2-9-8A, Window 7)
- ADS BLOWDOWN POWER FAILURE, (2-9-3C, Window 32)

Which ONE of the following completes both statements below?

The 250 VDC RMOV Board __(1)__ has been lost.

After manually transferring the 250 VDC RMOV Board to the Alternate Source, the Board is considered __(2)__ in accordance with Tech Spec 3.8.7, Distribution Systems – Operating.

A. (1) 2A
(2) inoperable

B. (1) 2A
(2) OPERABLE

C. (1) 2B
(2) inoperable

D. (1) 2B
(2) OPERABLE

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295004 G2.1.7	
	Importance Rating		4.7

295004 Partial or Complete Loss of D.C. Power
2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Explanation: **A CORRECT:** All of the alarms mentioned above, with the exception of one, occur for either board power loss. The reason that the RHR and Core Spray alarms were written generically is because one will be Loop I and the other would be Loop II. The HPCI 120 VAC failure is the only unique alarm, as they are listed here. HPCI is a DIV II System, but has an "ALPHA" Board Power Supply – this is counterintuitive and often confused. The ARP will have the operators manually transfer the board to its alternate power supply. The Tech Spec Bases for 3.8.7 discusses the fact that the board is considered inoperable, even with power restored; because of single failure considerations.

B- Incorrect. First part correct – as detailed in 'A' above. Second part is incorrect – per the Tech Spec Bases for 3.8.7 (as discussed above) the board is considered inoperable due to single failure considerations

C- . Incorrect. First part incorrect - The presence of the HPCI 120 VAC POWER FAILURE alarm is the designator that this is the Alpha versus the Bravo board. Second part is correct (for either board).

D- Incorrect. . First part incorrect - The presence of the HPCI 120 VAC POWER FAILURE alarm is the designator that this is the Alpha versus the Bravo board. Second part is incorrect – per the Tech Spec Bases for 3.8.7 (as discussed above) the board is considered inoperable due to single failure considerations.

Technical Reference(s): 2-ARP-9-8C window 4 and 11, 2-ARP-9-3C window 32, 2-ARP-9-3F window 3and 7, U2 Tech Spec 3.8.7 page 3.8-87a

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: BFN 1006 #76
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis : X
10 CFR Part 55 Content:	55.43 (2) Facility operating limitations in the technical specifications and their bases.

250V REACTOR MOV BD 2A UV 2-EA-57-94 (4)	Sensor/Trip Point:	
	72N-BA	Normal supply overcurrent.
	72E-BA	Alternate supply overcurrent.
	27EX	Normal supply undervoltage.
	27B	MOV bd undervoltage (7sec TDDO)

(Page 1 of 1)

Sensor Location: 250V RMOV Bd 2A, El 621', R-14 Q-LINE, Shutdown Bd Rm C

Probable Cause:
A. Loss of normal supply (250V Battery Bd 2, Pnl 3, Bkr 302).
B. Overcurrent on normal or alternate supply to the board.
C. Fuse failure.
D. Sensor malfunction.

Automatic Action: None.

Operator Action:
A. **VERIFY** conditions of alarm:
- Loss of HPCI indicating lights on Panel 2-9-3. ☐
- Loss of backup scram valve lights on Panel 2-9-5. ☐

B. **DISPATCH** Personnel to MOV board to check for abnormal conditions: undervoltage, breaker tripped, etc. ☐

NOTE

[II/C] If the mechanism resets (hear click and feel resistance when pushing in), this indicates that an overcurrent condition tripped the breaker.

C. **IF** Normal or Alternate feeder breaker tripped, **THEN** Manually **DEPRESS** mechanical trip/reset mechanism on breaker face to reset Bell Alarm lockout device.[NER/C II-B-92-069] ☐
D. **VERIFY** Bkr 302 closed at Battery Bd Room 2, Panel 3, El 593'. ☐
E. **REFER TO** TS Section 3.8.7. ☐
F. **REFER TO** 0-OI-57D to re-energize or transfer the board. ☐
G. **REFER TO** appropriate OI for recovery or realignment of equipment. ☐

References: 45N620-11 2-45E712-1 45N714-7
TS Section 3.8.7.

250V REACTOR MOV BD 2B UV 2-EA-57-100 (11)	<u>Sensor/Trip Point:</u>	
	72N-BA	Normal supply overcurrent
	72E-BA	Alternate supply overcurrent
	27EX	Normal supply undervoltage
	27B	Rx 2B MOV bd undervoltage (7sec TDDO)

(Page 1 of 1)

Sensor Location: 250V RMOV Bd 2B, El 593', R-14 Q-LINE, Shutdown Bd Rm B

Probable Cause:

A. Loss of normal supply (250V Battery Bd 3, Pnl 3, Bkr 303).
B. Overcurrent on normal or alternate supply to the board.
C. Fuse failure.
D. Sensor malfunction.

Automatic Action: None

Operator Action:

A. **VERIFY** alarm by checking: ☐
- Loss of HPCI and RHR indicating lights (Panel 2-9-3). ☐
- Loss of backup scram valve lights (Panel 2-9-5). ☐

B. **DISPATCH** Personnel to MOV board to check for abnormal conditions: undervoltage, breaker tripped, etc. ☐

> **NOTE**
> [II/C] If the mechanism resets (hear click and feel resistance when pushing in), this indicates that an overcurrent condition tripped the breaker.

C. **IF** Normal or Alternate feeder breaker tripped, **THEN** Manually depress mechanical trip/reset mechanism on breaker face to reset Bell Alarm lockout device. [NER/C II-B-92-069] ☐
D. **VERIFY** bkr 303 closed at Battery Bd Room 3, Panel 3, El 593'. ☐
E. **REFER TO** 0-OI-57D to re-energize or transfer the board. ☐
F. **REFER TO** TS Section 3.8.7. ☐
G. **REFER TO** appropriate OI for recovery or realignment of equipment. ☐

References: 45N620-11 2-45E712-2 45N714-7
TS Section 3.8.7.

ADS BLOWDOWN POWER FAILURE	32

(Page 1 of 1)

Sensor/Trip Point:

Relay 2E-K40	Panel 9-33	De-energized
Relay 2E-K1A	Panel 9-30	De-energized
Relay 2E-K1B	Panel 9-33	De-energized
Relay 2E-K12	Panel 9-30	De-energized
Relay 2E-K32	Panel 2-25-32	De-energized
Relay 2E-K33	Panel 2-25-32	De-energized
Relay 2E-K37	Panel 2-25-32	De-energized
Relay 2E-K38	Panel 2-25-32	De-energized

Sensor Location:

Panel 2-25-32
Backup Control Center
El 621', R-13 Q-LINE

Panel 2-9-30 and 2-9-33
Aux Instrument Rm
El 593'

Probable Cause:

A. Cleared Fuse(s)
B. Loss of 250V DC power supply to panels.
C. Auto Xfer of Logic Bus B Power Supplies.

Automatic Action:

Main steam auto relief valves, PCV-1-22 and 30, auto transfer power supply from 250V DC Rx Mov Bd A to 250 DC Rx Mov Bd B (PCV-1-22) & Bd C (PCV-1-30) on loss of normal power supply.
Logic Bus "B" transferred to Alternate supply (250V Rmov Bd 2A) upon Loss of Normal Supply (250V RMOV Bd 2B) or fuse failure.

Operator Action:

A. **VERIFY** power is available to PCV-1-22 and -30. ☐
B. **IF** annunciator HPCI LOGIC POWER FAILURE, XA-55-3F Window 3, is in alarm, this is indicative of loss of power to 250V DC Rx Mov Bd 2A/2B. **DISPATCH** personnel to check 250V DC Rx Mov Bd 2A Breaker 11A2 and 250V DC Rx MOV Bd 2B Breaker 1B1. ☐
C. **DISPATCH** personnel(s) to check:
   1. Logic Bus A
      a. 250V DC Rx Mov Bd 2B, Breaker 1F1. ☐
      b. Fuses 2-FU2-001-2E-K3 in Panel 9-30. ☐
   2. Logic Bus B
      a. 250V DC Rx Mov Bd 2A, Breaker 9A1. ☐
      b. Fuses 2-FU1-001-2E/K22A and 2-FU1-001-2E/K22B on Panel 9-33. ☐
      c. Fuses 2-FU2-001-2E-K13 in Panel 9-30. ☐
      d. Fuses 2-FU2-1-2E-K11A and 2-FU2-1-2E-K11) on Panel 9-33 (GG Block).

D. **REFER TO** Tech Spec Section 3.5.1. ☐
E. **REFER TO** TRM 3.3.3.4. ☐

References:

2-45N620-2	2-45E712-1, -2 and -3	GE 730E929 -1, -2 and 3
Technical Specifications 3.5.1		TRM 3.3.3.4.

HPCI LOGIC POWER FAILURE 3

(Page 1 of 1)

Sensor/Trip Point:

Relay 23A-K39 (Bus A) Loss of 250V DC
Relay 23A-K44 (Bus B) Control Power
Relay 23A-K44B (Bus B)

Sensor Location: Panel 9-32, Bus A, Aux Instr Rm, El 593' — Panel 9-39, Bus B, Aux Instr Rm, El 593'

Probable Cause:

A. Cleared fuse(s).
B. Loss of 250V DC power supply to panels.

Automatic Action:

A. Logic Bus A failure renders Channel A trip and automatic isolation logic inop. HPCI continues to function.
B. Logic Bus B failure renders Channel B trip, automatic initiation, and automatic isolation logics inop. If HPCI is in service the HPCI TURBINE STOP VALVE, 2-FCV-73-18, closes. HPCI becomes inoperable.

Operator Action:

A. **DETERMINE** which logic bus has failed, **REFER TO** automatic action section. ☐
B. **DISPATCH** personnel to verify source of power failure: ☐
   1. Logic Bus A ☐
      a. Fuses 2-FU2-073-23A-K36 (23A-F19) and 2-FU2-073-23A-K36 (23A-F20), Panel 9-32. ☐
      b. Power supply 250V DC Rx Mov Bd 2B, Breaker 1B1. ☐
   2. Logic Bus B ☐
      a. Fuses 2-FU2-073-0039A and 2-FU2-073-0039B, Panel 9-39. ☐
      b. Power supply 250V DC RMOV Bd 2A, Breaker 11D1. ☐

C. **REFER TO** Tech Spec 3.5.1, 3.5.2, 3.3.5.1, 3.3.6.1, and TRM 3.3.3.4. ☐

References:

2-45E620-1 GE 730E928-2-3 and -4.
Technical Specifications 3.3.5.1, 3.3.6.1,3.5.1,3.5.2, TRM 3.3.3.4

HPCI 120 VAC POWER FAILURE 7

(Page 1 of 1)

Sensor/Trip Point:

Relay 23A-K50 — Loss of the 120 VAC from DIV II ECCS ATU inverter and Loss of power to the HPCI Flow IND Controller (2-FIC-73-33)

Sensor Location: Panel 9-19
Aux Instr Rm, El 593'

Probable Cause:
A. Blown fuses, Fuse 2-FU2-073-0033C, Panel 2-9-82 AA1 & AA2,
B. DIV II ECCS ATU inverter failure.
C. Loss of 250V DC power supply to DIV II ECCS ATU inverter (RMOV BD 2A compt 11A1).

Automatic Action:
A. HPCI controller loses power. HPCI becomes inoperable.
B. If HPCI is in service, the HPCI Turbine Stop Valve, 2-FCV-73-18, closes. HPCI controller loses power. HPCI becomes inoperable.
C. 2-PI-064-67B will lose power and become inop.

Operator Action:
A. **DISPATCH** personnel to CHECK the following: ☐
- Fuses 2-FU2-073-0033C, Panel 2-9-82, AA1 & AA2. ☐
- DIV II ECCS ATU inverter. ☐
- DIV II ECCS ATU inverter breaker, RMOV BD 2A, compt 11A1. ☐

B. 2-PI-064-67B will lose power and become inop.
REFER TO TECH SPEC 3.3.3.1, Table 3.3.3.1-1, TRM 3.3.5. ☐
C. **REFER TO**: Tech Spec 3.5.1. Tech Spec 3.3.3.1, Table 3.3.3.1-1. ☐

References: 2-45E620-1 GE 730E928-2 and -4
Technical Specifications 3.5.1, and 3.3.3.1
Technical Specifications Bases 3.3.3.1, TRM 3.3.5

BASES

LCO (continued)

When 480 V Shutdown Board 2B is aligned to the alternate supply 4.16 kV Shutdown Board C, a LOCA/LOOP with a failure of the Shutdown Board D Battery would disable the normal supply 4.16 kV Shutdown Board D, and would also prevent the 480 V Shutdown Board 2B from load shedding its 480 V loads which would overload the alternate supply Diesel Generator D. This would result in the loss of diesel generators C and D, associated 4.16 kV shutdown boards and RHRSW pumps. Therefore, the restrictions on the associated drawings shall be adhered to whenever 480 V Shutdown Board 2B is on its alternate supply.

The Unit 2 480 V RMOV boards 2A and 2B have an alternate power supply from the other 480 V shutdown board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source would affect both divisions.

The Unit 2 250 V DC RMOV boards 2A, 2B, and 2C have alternate power supplies from another 250 V Unit DC board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source could affect both divisions depending on the board alignment.

If a 4.16 kV or 480 V shutdown board is aligned to its alternate 250 V DC control power source a single failure of the alternate power source could affect both ECCS divisions and common equipment needed to support the other units depending on the board alignment. Therefore, the restrictions on the associated drawings shall be adhered to whenever a 4.16 kV or 480 V shutdown board is on its alternate control power supply.

76. 295004 AA2.01

With Unit 2 operating at 100% Reactor Power, a Normal Supply Breaker has tripped open **AND** suffered damage due to arcing. In addition to RHR **AND** Core Spray logic alarms, th
following significant alarms result:

- HPCI LOGIC POWER FAILURE, (2-9-8A, Window 7)
- HPCI 120 VAC POWER FAILURE, (2-9-8A, Window 7)
- ADS BLOWDOWN POWER FAILURE, (2-9-3C, Window 32)

Which ONE of the following completes the statements?

The 250 VDC RMOV Board __**(1)**__ has been lost.

After manually transferring the 250 VDC RMOV Bd, the Board is considered __**(2)**__ in accordance with Tech Spec 3.8.7, "Distribution Systems – Operating."

A. **(1)** 2A
(2) inoperable

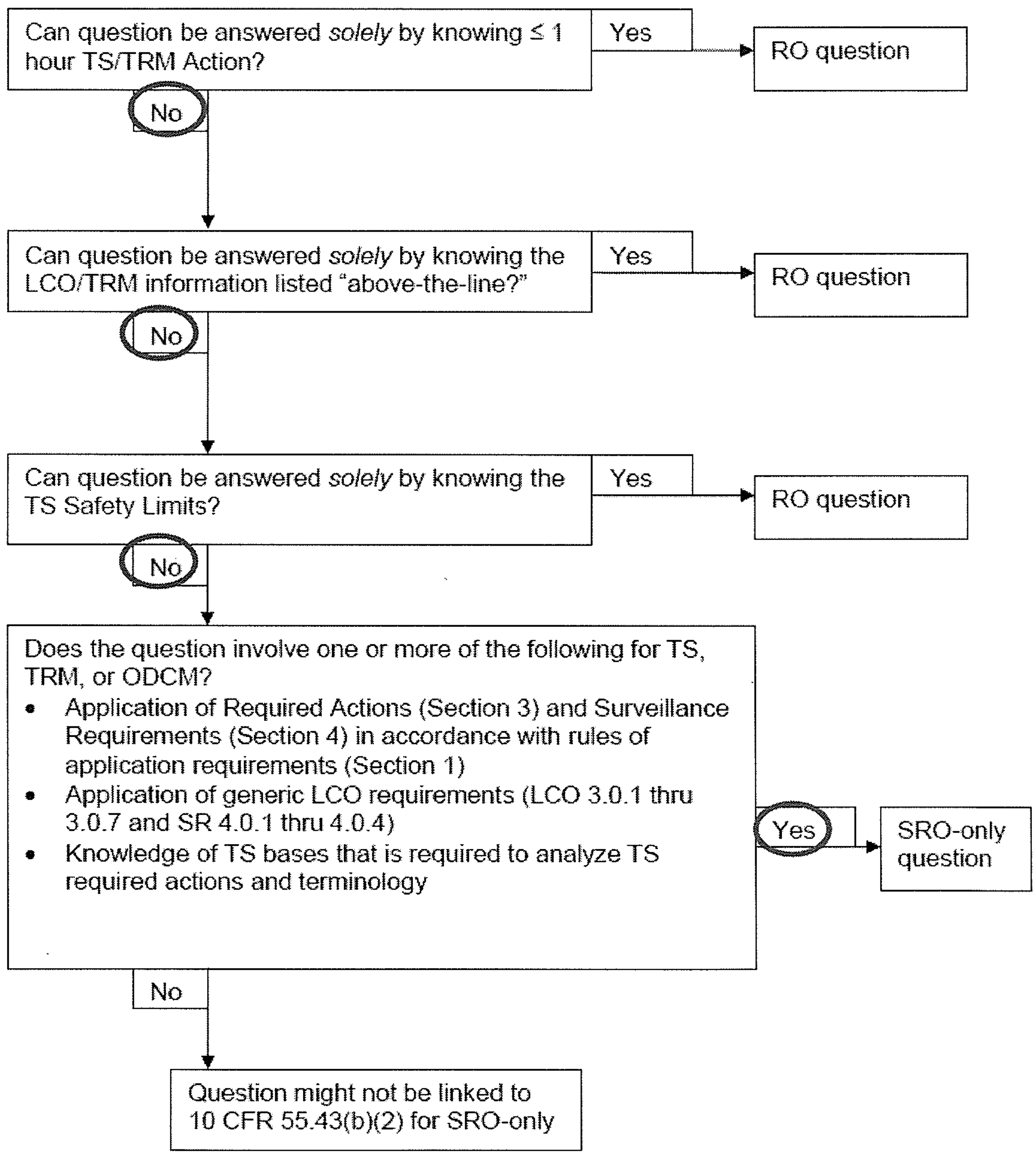
B. **(1)** 2B
(2) inoperable

C. **(1)** 2A
(2) operable

D. **(1)** 2B
(2) operable

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



QUESTION 77

A reactor scram has occurred on Unit 1.

In accordance with 1-AOI-100-1, Reactor Scram, the RO ATC reports the following:

- Mode Switch is in Shutdown
- 12 Rods are at position 24
- RPV water level is (-) 9 inches and slowly recovering
- Reactor Pressure is 955 psig and steady
- MSIV are open
- ATWS Actions are Complete and APRMs are downscale

Which ONE of the following completes both statements below?

When the RO ATC reports "ATWS Actions are Complete and APRMs are downscale" this means that BOTH channels of ARI are initiated and the __(1)__.

Assuming ARI caused NO rod motion, the Unit Supervisor shall __(2)__.

A. (1) Reactor Recirc Pumps are tripped
(2) remain in the RC/L leg of EOI-1, RPV CONTROL

B. (1) Reactor Recirc Pumps are tripped
(2) exit the RC/L leg of EOI-1, RPV CONTROL and enter C-5, LEVEL/POWER CONTROL

C. (1) Reactor Recirc Pumps are at minimum speed
(2) remain in the RC/L leg of EOI-1, RPV CONTROL

D. (1) Reactor Recirc Pumps are at minimum speed
(2) exit the RC/L leg of EOI-1, RPV CONTROL and enter C-5, LEVEL/POWER CONTROL

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295006 G2.1.20	
	Importance Rating		4.6

295006 SCRAM
2.1.20 Ability to interpret and execute procedure steps.

Explanation: **D CORRECT:** When the RO ATC reports "ATWS Actions are Complete" this means that BOTH channels of ARI are initiated and Reactor Recirc Pumps are at minimum speed. Given the conditions in the stem, i.e. it has NOT been determined that the reactor will remain shutdown under all conditions without boron, RC/L should be exited and C5, Level/Power Control should be entered.

A- Incorrect. First part incorrect –Plausible because step RC/Q-3 directs tripping the Recirc pumps under these conditions, however "ATWS Actions are Complete" in AOI-100-1, Reactor Scram, ONLY means that Recirc Pumps have been run back to minimum speed. Second part is incorrect, plausible if the candidate does not recognize that plant conditions meet override RC/L-3 in EOI-1, RPV CONTROL, directing the operator to exit the RC/L leg of EOI-1, RPV CONTROL and enter C-5, LEVEL/POWER CONTROL.

B- Incorrect. First part incorrect –Plausible because step RC/Q-3 directs tripping the Recirc pumps under these conditions, however "ATWS Actions are Complete" in AOI-100-1, Reactor Scram, ONLY means that Recirc Pumps have been run back to minimum speed. Second part is correct.

C- Incorrect. First part correct. Second part is incorrect, plausible if the candidate does not recognize that plant conditions meet override RC/L-3 in EOI-1, RPV CONTROL, directing the operator to exit the RC/L leg of EOI-1, RPV CONTROL and enter C-5, LEVEL/POWER CONTROL.

Technical Reference(s): 1-EOI-1, RPV CONTROL, 1- AOI-100-1, Reactor Scram, ODM 4.20

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X	
Question History:	Previous NRC: None	
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis : X	

10 CFR Part 55 Content: 55.43 (5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

4.2 Subsequent Actions

NOTES

1) The steps in this section are written in general order of importance for most anticipated events; however, they are not required to be performed in order, but as required to maintain stable conditions. Once a step is entered, all associated substeps are required to be completed in order, except those in Step 4.2[33](Return to Service). Steps which are not applicable for this scram should be N/A'd.

2) For Scram Response logic to initiate, all of the following conditions must be met:
   - Scram Response Logic is not inhibited (amber light at SCRAM RESPONSE INHIBIT/RESET switch, 1-HS-46-5 on Panel 1-9-5, is extinguished).
   - REACTOR WATER LEVEL CONTROL PDS, 1-LIC-46-5 on Panel 1-9-5, is in AUTO and at least one individual RFPT Speed Control PDS in AUTO.
   - Either RPS A or B Backup Scram channel activates.
   - Reactor Level (narrow range) falls below 0 inches within 60 seconds of first Backup Scram channel activating.

3) If Programmed Scram Response is initiated, the logic is reset by ANY of the following conditions:
   1. Placing REACTOR WATER LEVEL CONTROL PDS, 1-LIC-46-5 on Panel 1-9-5 in MANUAL.
   2. Reactor level (narrow range) exceeding level setpoint.
   3. Five minutes expire from the time the Scram Response logic was activated.
   4. Depressing SCRAM RESPONSE INHIBIT/RESET Switch, 1-HS-46-5, on Panel 1-9-5.

[1] **ANNOUNCE** Reactor SCRAM over PA system. ☐

[2] **IF** all control rods **CAN NOT** be verified fully inserted, **THEN**

PERFORM the following (otherwise N/A):

➡ [2.1] **INITIATE** ARI by arming and depressing BOTH of the following:

- ARI MANUAL INITIATE, 1-HS-68-119A ☐
- ARI MANUAL INITIATE, 1-HS-68-119B ☐

➡ [2.2] **VERIFY** the Reactor Recirc Pumps (if running) at minimum speed at Panel 1-9-4. ☐

➡ [2.3] **REPORT** "ATWS Actions Complete" and power level. ☐

4.7.3 RPV Control (EOI-1) (continued)

C. Power Leg of flowchart

If the determination is made that the reactor is subcritical by the use of nuclear instrumentation, then the subsequent actions of AOI-100-1 should be directed. If control rods remain withdrawn from the core and EOI appendices have been directed that will insert the control rods prior to the determination of subcriticality, then the appendices should continue to be used until all control rods are fully inserted into the core.

D. ATWS Actions

It is the expectation that IF all control rods CANNOT be verified fully inserted, the OATC should actuate both channels of ARI, run both Recirc pumps to minimum speed and report "*ATWS actions are complete and Reactor Power is* _____", as per AOI-100-1 Hard Card.

During an ATWS, the US should not exit RC/L and enter C-5, LEVEL/POWER CONTROL, until OATC ATWS actions per the AOI-100-1 Hard Card have been completed and Reactor Power has been reported to the US.

When EOI-1, Step RC/Q-9 is reached, IF core oscillations are observed, THEN INITIATE SLC.

When EOI-1, Step RC/Q-10 is reached, IF reactor power is greater than APRM downscale, THEN INITIATE SLC.

RC/L

C1, Alternate Level Control

C1-2, C1-29

C4, RPV Flooding

C4-1

C5, Level/Power Control

C5-1

CAUTION

#1 Ambient temp may affect RPV water lvl indication and trend

MONITOR and **CONTROL** RPV water lvl

RC/L-1

VERIFY each as required:

- PCIS isolations (Groups 1, 2 and 3)
- ECCS
- RCIC

RC/L-2

WHILE EXECUTING THE FOLLOWING STEPS:

	IF	**THEN**
1	It has NOT been determined that the reactor will remain subcritical without boron under all conditions	**EXIT** RC/L and **ENTER** C5, Level/Power Control
	RPV water lvl CANNOT be determined	**EXIT** RC/L and **ENTER** C4, RPV Flooding
	PC water lvl CANNOT be maintained below 105 ft OR Suppr chmbr press CANNOT be maintained below 55 psig	**STOP** inj into the RPV from sources external to the PC NOT required for adequate core cooling

RC/L-3

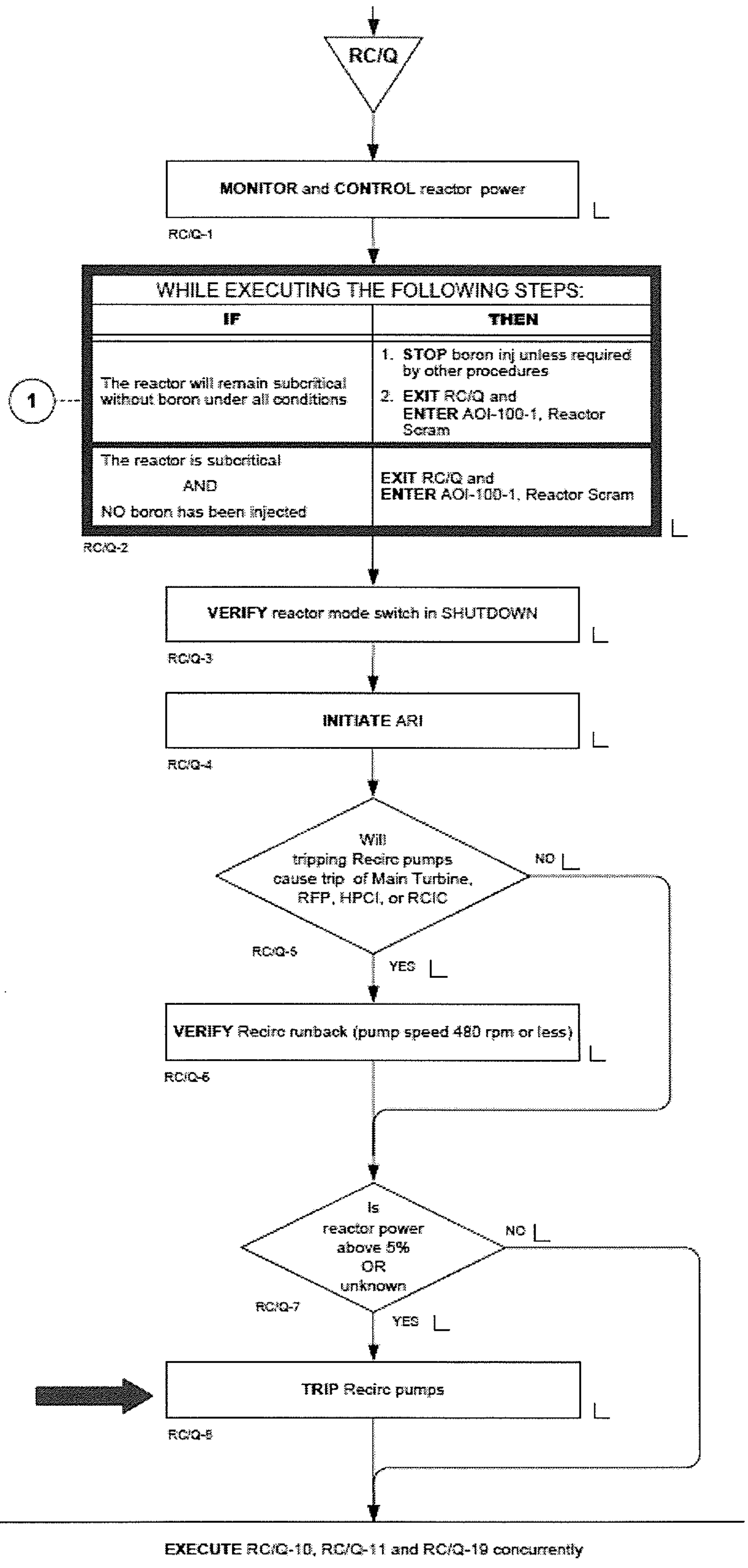
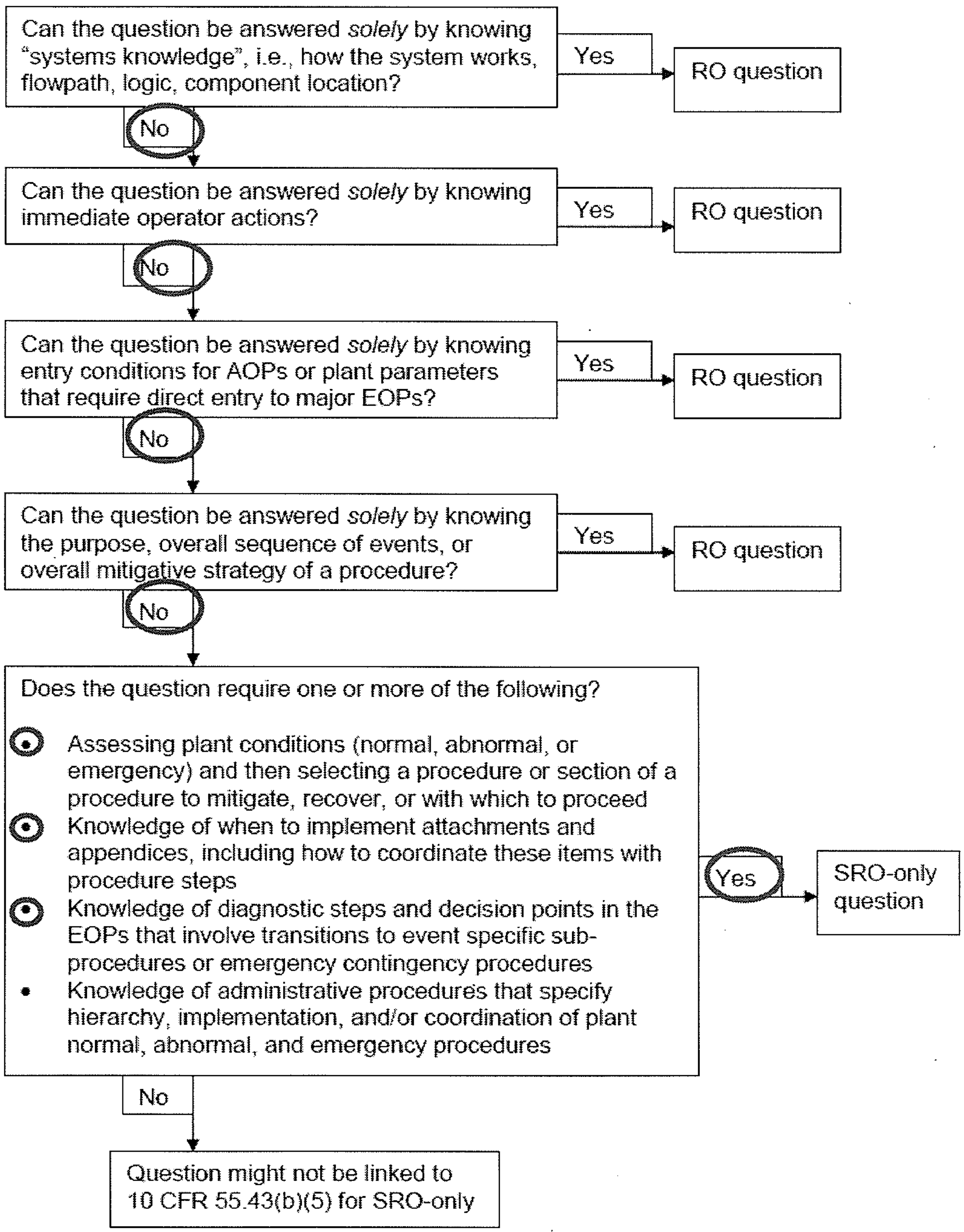

RC/Q
MONITOR and CONTROL reactor power
RC/Q-1
WHILE EXECUTING THE FOLLOWING STEPS:
IF
THEN
1
The reactor will remain subcritical without boron under all conditions
1. STOP boron inj unless required by other procedures
2. EXIT RC/Q and ENTER AOI-100-1, Reactor Scram
The reactor is subcritical AND NO boron has been injected
EXIT RC/Q and ENTER AOI-100-1, Reactor Scram
RC/Q-2
VERIFY reactor mode switch in SHUTDOWN
RC/Q-3
INITIATE ARI
RC/Q-4
Will tripping Recirc pumps cause trip of Main Turbine, RFP, HPCI, or RCIC
NO
YES
RC/Q-5
VERIFY Recirc runback (pump speed 480 rpm or less)
RC/Q-6
Is reactor power above 5% OR unknown
NO
YES
RC/Q-7
TRIP Recirc pumps
RC/Q-8
EXECUTE RC/Q-10, RC/Q-11 and RC/Q-19 concurrently

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



QUESTION 78

Given the following conditions for Unit 2:

- Reactor has scrammed due to a rapid loss of control air header pressure.
- Multiple rods remain out
- Suppression pool temperature is 115° F
- Control Air header pressure is 25 PSIG and lowering
- RPV water level was lowered to (-) 55 inches

Which ONE of the following completes the statements below?

MSIV status is __(1)__.

In order to re-establish the condenser as a heat sink, the MSIV should be re-opened using __(2)__.

NOTE: 2-AOI-32-2, Loss of Control Air
2-EOI APPENDIX 8B, Reopening MSIVs/Bypass Valve Operation

A. (1) all MSIVs CLOSED
(2) 2-AOI-32-2 ONLY

B. (1) all MSIVs CLOSED
(2) 2-EOI-APPENDIX 8B and 2-AOI-32-2

C. (1) inboard MSIVs OPEN, outboard MSIVs CLOSED
(2) 2-AOI-32-2 ONLY

D. (1) inboard MSIVs OPEN, outboard MSIVs CLOSED
(2) 2-EOI-APPENDIX 8B and 2-AOI-32-2

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295019 AA2.02	
	Importance Rating		3.7

295019 Partial or Complete Loss of Instrument AA2.02 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Status of safety-related instrument air system loads (see AK2.1 - AK2.19)

Explanation: **D CORRECT**: First Part- CORRECT: On a rapid loss of control air, the outboard MSIV's close at 45 psig. . Second Part-CORRECT: In order to open MSIV's, 2-EOI-1 directs the use of 2-EOI APPENDIX 8B. However, this appendix does not contain info to align DW control air to the outboard MISV's. 2-AOI-32-2 directs aligning drywell control air to open the Outboard MSIV's. Given the plant conditions in the stem, reopening the MSIVs is appropriate. The candidate must interpret the listed conditions and recognize that the normal air supply to the MSIVs has been lost and that DW air must be aligned to accomplish the goal of the applicable EOI. Thus, a detailed knowledge of procedures needed to open the MSIVs under these conditions is needed.

A- Incorrect: Part (1) Incorrect-It is plausible that all MSIV's would be closed on either a misconception of isolation on level or a loss of control air.. (2) Incorrect-See A.

B- Incorrect: Part (1) Incorrect-See C. (2) CORRECT-See B.

C- Incorrect: Part (1) Correct see A. Part (2) Incorrect-Although, 2-AOI-32-2 directs aligning drywell control air to open the Outboard MSIV's, 2-EOI-1 directs the use of Appendix B to open the MSIV's when re-establishing the Main Condenser as a heat sink.

Technical Reference(s): 2-EOI-1;2-EOI APPENDIX 8B; 2-AOI-32-2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL 171.054 Obj. B.8

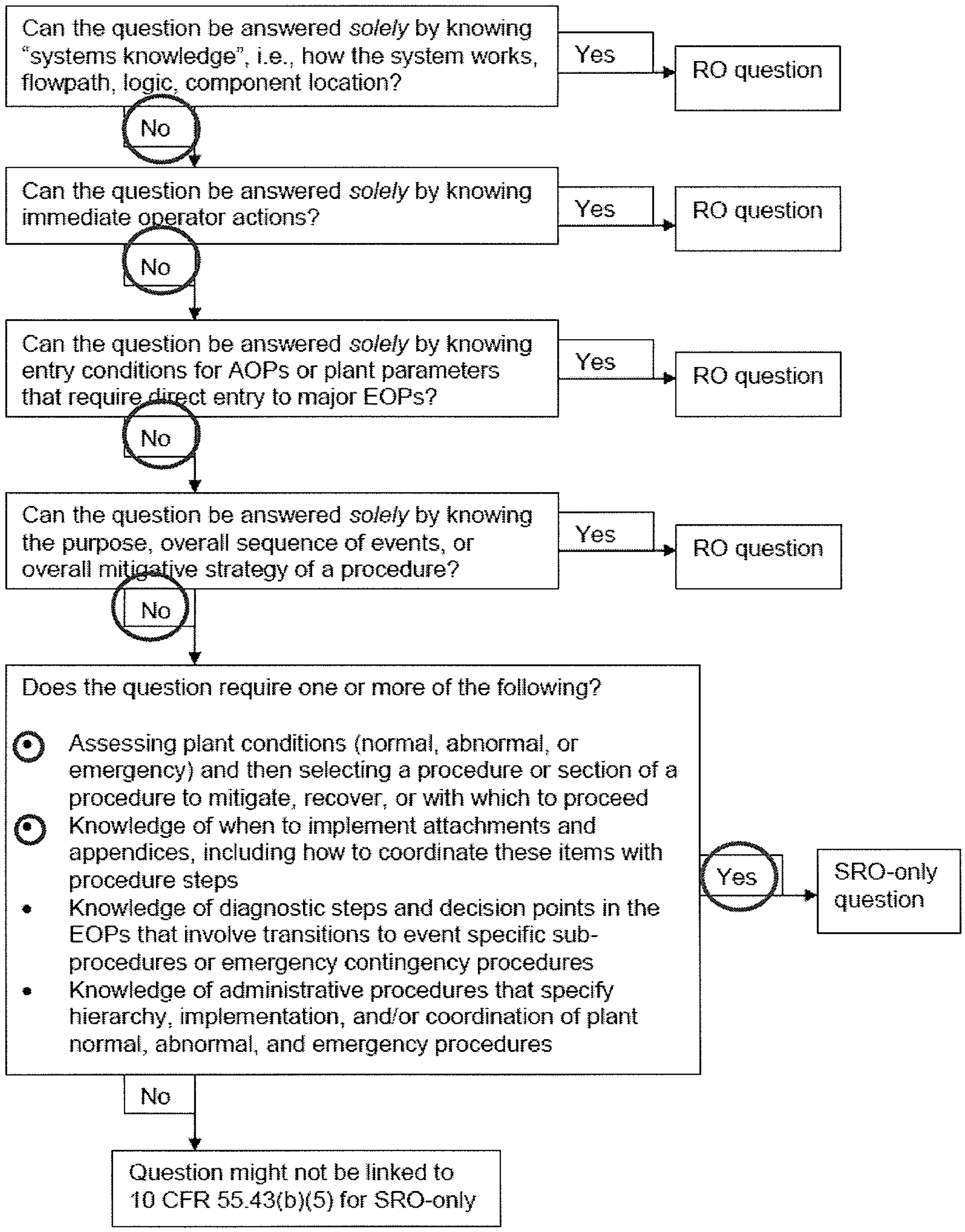
Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC:

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

10 CFR Part 55 Content: 55.43 (5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



WHILE EXECUTING THE FOLLOWING STEPS:	
IF	**THEN**
Steam cooling is required	EXIT RC/P
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 (ok to exceed 100°F/hr cooldown rate)
Suppr pl lvl CANNOT be maintained in the safe area of Curve 4	**MAINTAIN** RPV press in the safe area of Curve 4 (ok to exceed 100°F/hr cooldown rate)

2

RC/P-7

WHILE EXECUTING THE FOLLOWING STEPS:	
IF	**THEN**
DW control air becomes unavailable	**CROSSTIE** CAD to DW control air (APPX 8G)

RC/P-8

WHILE EXECUTING THE FOLLOWING STEPS:	
IF	**THEN**
Boron inj is required AND The main condenser is available AND There has been no indication of a steam line break	**PERFORM** the following to reestablish the main condenser as a heat sink: • **OPEN** MSIVs (APPX 8A Bypass Gp1 PCIS Low RPV Lvl, APPX 8B Reopen MSIVs) • **RESTORE** ventilation as necessary (APPX 8E Bypass Gp 6 PCIS Low RPV Lvl and High DW Press, APPX 8F Restore RF and RB Ventilation)

RC/P-9

A

RC/P-12

STABILIZE RPV press below 1073 psig with the main turbine bypass vlvs (APPX 8B)

RC/P-10

CAUTION	
#3	Elevated suppr chmbr press may trip RCIC
#6	HPCI or RCIC suction temp above 140°F

AUGMENT RPV press control as necessary with ANY of the following:

DEPRESSURIZATION SYSTEM	APPX
MSRVs ONLY when suppr lvl is above 5.5 ft IF "MAIN STEAM RELIEF VLV AIR ACCUM PRESS LOW" annunciator (XA-55-3D-18) is in alarm THEN **PLACE** each MSRV control switch in CLOSE/AUTO AND **PLACE** MSRV auto actuation logic inhibit XS-1-202 to INHIBIT	11A
HPCI with CST suction if available	11C
RCIC with CST suction if available	11B
RFPTs on min flow	11F
Main Steam system drains	11D
Steam seals	11G
SJAEs	11G
Off Gas preheater	11G
HPCI and RCIC drains	11J
RWCU if NO boron has been injected into the RPV	11E

RC/P-11

4.2 **Subsequent Actions (continued)**

[7.2.1] **ESTABLISH** lube oil temperature between 80°F and 100°F using the following TCV BYPASS VALVE(s):

- For A RFP use 2-24-626A or 3-24-627A ☐
- For B RFP use 2-24-626B or 3-24-627B ☐
- For C RFP use 2-24-626C or 3-24-627C ☐

[7.3] **CLOSE** EHC fluid cooler TCV isolation valves 2-24-592 or 2-24-593, **THEN**

ESTABLISH fluid temperature between 85°F and 125°F on TI-47-59 using TCV BYPASS VALVE 2-24-590 or 2-24-591. ☐

[8] **VERIFY** drywell control air system is being supplied by either the Nitrogen System or CAD system. ☐

NOTE
DRYWELL control air can be valved into control air lines for outboard MSIV's.

[9] **IF** Unit Supervisor determines outboard MSIV's need to be opened in order to establish the main condenser as a heat sink, **THEN**

PERFORM Attachment 2. (Otherwise N/A) ☐

1.0 INSTRUCTIONS

LOCATION:	Unit 2 Control Room
ATTACHMENTS	None

[1] **IF** pressure control with bypass valves is desired and MSIVs are open, **THEN**

PROCEED to step 1.0[13]. ☐

[2] **VERIFY** ALL MSIV control switches in CLOSE position. ☐

[3] **RESET** PCIS logic (Panel 2-9-4). ☐

[4] **DEPRESS** the following pushbuttons to trip RFPTs (Panel 2-9-6):

- 2-HS-3-125A, RFPT 2A TRIP ☐
- 2-HS-3-151A, RFPT 2B TRIP ☐
- 2-HS-3-176A, RFPT 2C TRIP. ☐

1.0 INSTRUCTIONS (continued)

[5] **VERIFY CLOSED** the following drain valves (Panel 2-9-3):

NOTE
To prevent auto opening of 2-FCV-1-58, handswitch 2-HS-1-58A must be held in the CLOSE position until main turbine speed decreases to below 1700 RPM.

- 2-FCV-1-58, UPSTREAM MSL DRAIN TO CONDENSER ☐
- 2-FCV-1-59, DOWNSTREAM MSL DRAIN TO CONDENSER. ☐

[6] **VERIFY CLOSED** the following drain valves (Panel 2-9-7):

- 2-FCV-6-100, STOP VALVE 1 BEFORE SEAT DR VLV ☐
- 2-FCV-6-101, STOP VALVE 2 BEFORE SEAT DR VLV ☐
- 2-FCV-6-102, STOP VALVE 3 BEFORE SEAT DR VLV ☐
- 2-FCV-6-103, STOP VALVE 4 BEFORE SEAT DR VLV ☐

[7] **VERIFY CLOSED** the following drain valves (Panel 2-9-6):

- 2-FCV-6-122, RFPT 2A HP STOP VLV ABOVE SEAT DR ☐
- 2-FCV-6-127, RFPT 2B HP STOP VLV ABOVE SEAT DR ☐
- 2-FCV-6-132, RFPT 2C HP STOP VLV ABOVE SEAT DR ☐

[8] **OPEN** the following outboard MSIVs (Panel 2-9-3):

- 2-FCV-1-15, MSIV LINE A OUTBOARD ☐
- 2-FCV-1-27, MSIV LINE B OUTBOARD ☐
- 2-FCV-1-38, MSIV LINE C OUTBOARD ☐
- 2-FCV-1-52, MSIV LINE D OUTBOARD. ☐

QUESTION79

Unit 1 was at 100% power when one Safety Relief Valve failed open and was unable to be closed. The Reactor Mode Switch was placed in SHUTDOWN.

The following conditions exist:

- Reactor power 8% and lowering
- Reactor pressure 900 psig and stable
- Suppression pool temperature 182 °F and rising
- Suppression pool level 16.0 ft and slowly rising

Which ONE of the following completes the statement below?

The Unit Supervisor should direct ____.

REFERENCE PROVIDED

A. lowering reactor pressure within the limits of the cool down rate, in accordance with 1-EOI-1, RPV Control

B. lowering reactor pressure irrespective of cool down rate as required, in accordance with 1-EOI-1, RPV Control

C. lowering reactor pressure by anticipating Emergency Depressurization in accordance with 1-EOI-1, RPV Control

D. immediate Emergency Depressurization of the reactor in accordance with 1-C-2, Emergency RPV Depressurization

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295025 EA2.03	
	Importance Rating		4.1

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

Explanation: **Answer B – CORRECT:** lowering reactor pressure irrespective of cooldown rate is correct for given conditions

A – incorrect – plausible. In that lowering reactor pressure is correct but the requirement is OK to exceed cooldown rate limits

C– incorrect – plausible ED is likely but cannot anticipate ED in an ATWS

D – incorrect – plausible with margin to HCL and RPV pressure at 900 psig ED is not correct for current conditions.

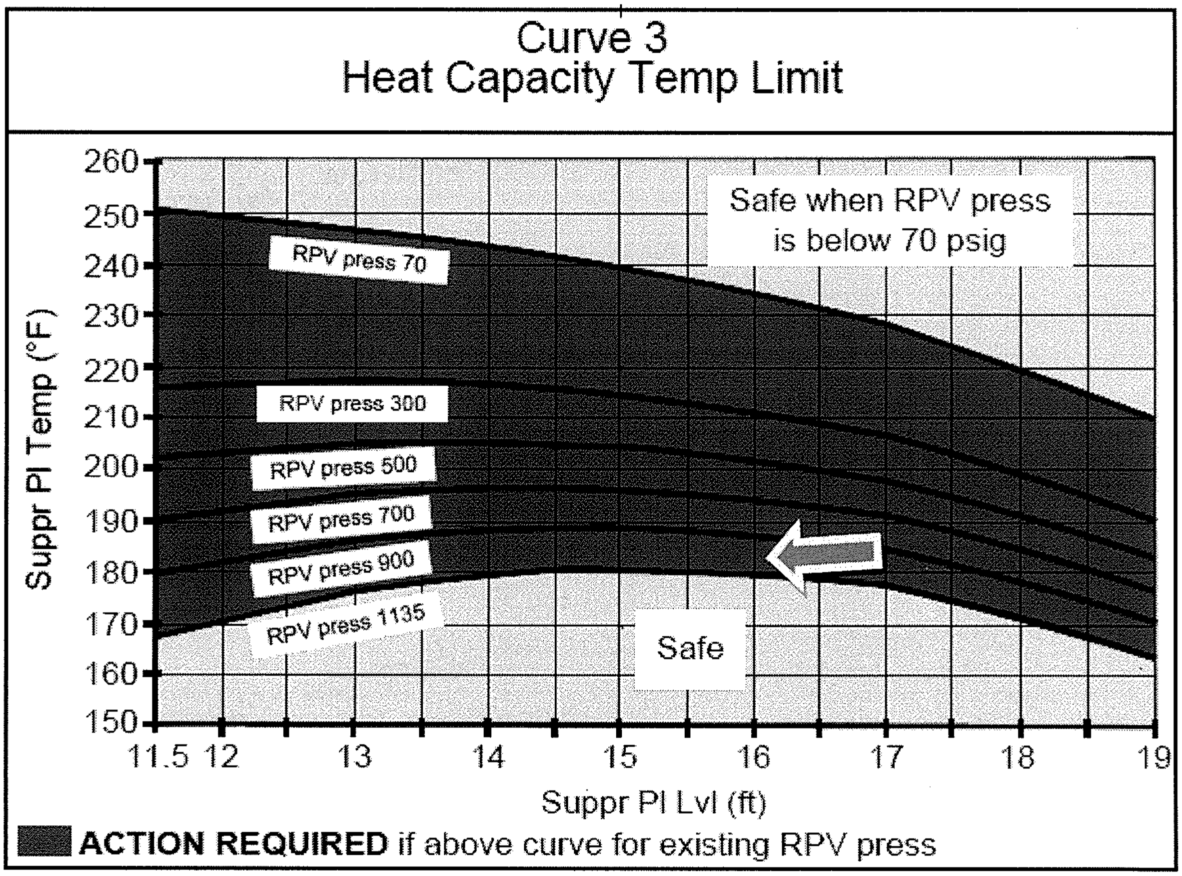
Technical Reference(s): 2-EOI-1and 2-EOI-2 Flowchart

Proposed references to be provided to applicants during examination: Heat Capacity Temperature Limit Curve 3

Learning Objective (As available): OPL171.202 V.B.15

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: Perry NRC 2009 #3 SRO
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis : X

10 CFR Part 55 Content: 55.43 b(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.



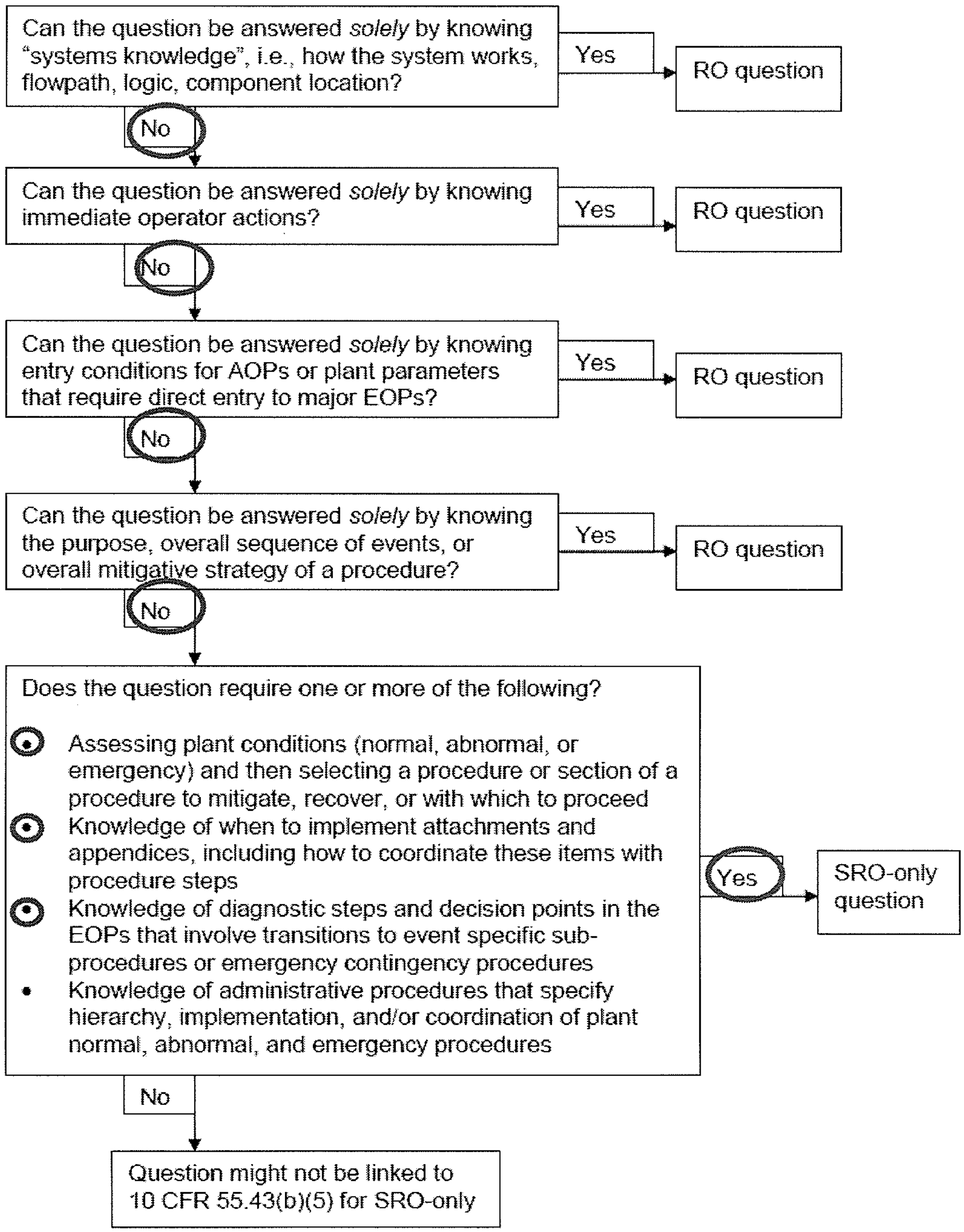
Suppr pl temp and lvl CANNOT be maintained in a safe area of Curve 3 at the existing RPV press	**LOWER** RPV press to maintain suppr pl temp and lvl in a safe area of Curve 3 (ok to exceed 100°F/hr cooldown rate)

WHEN suppr pl temp and lvl CANNOT be maintained within a safe area of Curve 3

SP/T-6

EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
(EOI-1, RC/P-4; C1-1, C1-20; C5-12, C5-14)

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



QUESTION 79 1108 audit high miss 42.9% got it right

Which ONE of the following completes the statements below in accordance with 2-EOI-2, "Primary Containment Control and 2-EOI-1, RPV Control?"

The SRO __**(1)**__ direct exceeding the Tech Spec cooldown limit in order to maintain within the SAFE region of the Heat Capacity Temperature Limit (HCTL).

If the plant enters the UNSAFE region of the Heat Capacity Temperature Limit (HCTL), then the SRO at this time __**(2)**__ allowed to reduce pressure AND return to the SAFE region to avoid an unnecessary Emergency Depressurization.

A. **(1)** may
(2) is

B. **(1)** may
(2) is**NOT**

C. **(1)** may **NOT**
(2) is

D. **(1)** may **NOT**
(2) is**NOT**

ANSWER: **D**

A	INCORRECT: Part 1 correct – See explanation B. Part 2 incorrect – See Explanation C
B	**CORRECT**: Part 1 correct – Reactor Pressure may be reduced as required to maintain within the safe area of HCTL, even in an ATWS. Part 2 correct – Pressure reduction is allowed to prevent from entering the UNSAFE region, but once there the plant is not allowed to restore to the SAFE region except by Emergency Depressurization.
C	**INCORRECT: Part 1 incorrect – Plausible in that the SRO may believe with an ATWS in progress a cooldown may not exceed TS limits or may not be initiated. Part 2 incorrect –Plausible in that some conditions requiring Emergency Depressurization IAW EOIs allow for the parameter to be restored and maintained.**
D	**INCORRECT: Part 1 incorrect – See explanation C. Part 2 correct – See Explanation B.**

QUESTION 80

Given the following Unit 1 conditions:

- A Loss of off-site power has occurred
- OATC reports 26 control rods are at position 02 and the remaining rods are at position 00
- Reactor Power is unknown
- RPV pressure is being maintained between 800 to 1000 psig
- RPV level is (-) 20 inches and steady
- Suppression Pool temperature is 102° F and steady

Which ONE of the following completes both statements below?

In accordance with Emergency Operating Instructions, the RPV Level Band is __(1)__ inches.

Standby Liquid Control (SLC) __(2)__ required to be initiated in accordance with **1-EOI-1**.

A. (1) (+) 51 to (-) 180
(2) is

B. (1) (+) 51 to (-) 180
(2) is NOT

C. (1) (-) 50 to (-) 180
(2) is

D. (1) (-) 50 to (-) 180
(2) is NOT

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295037 EA2.05	
	Importance Rating		4.3

Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: Control rod position

Explanation: **D CORRECT:** No RPS power is available which eliminates the use of APRMs for power level determination. Rod Position indication is available; as it is powered from D/G backed Unit Preferred or Batteries, and based on the conditions stated in the stem, sufficient power is available to use the RPIS function. Subcritical under all conditions is sufficient to determine the power and level control functions of the EOIs. First part: The reactor is NOT subcritical under all conditions. Therefore RPV water level is lowered in step C5-10 to (-)50 inches. A water level band of (-)50 to (-) 180 inches would be correct. SLC would NOT be required because Suppression Pool Temperature is 102° F and steady.

A-Incorrect. First Part: Incorrect. Plausible because this is the normal control band for RPV water level in a subcritical reactor. Second Part: Incorrect. Plausible because thus is still in the RC/Q leg of EOI-1. However suppression Pool temperature is less than 110° F and steady.

B- Incorrect. First Part: Incorrect. Plausible because this is the normal control band for RPV water level in a subcritical reactor. Second Part: Correct.

C- Incorrect. First Part: Correct. Second part: Incorrect. Plausible because thus is still in the RC/Q leg of EOI-1. However suppression Pool temperature is less than 110° F and steady.

Technical Reference(s): 1-EOI-1, 1-EOI-C5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: BFN 0610 (2006) #80

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

10 CFR Part 55 Content: 55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/P-3

Positive confirmation that the reactor will remain subcritical under all conditions is best obtained by determining that no control rod is withdrawn beyond the Maximum Subcritical Banked Withdrawal Position (MSBWP, **A.70**). The MSBWP is the greatest banked rod position at which the reactor will remain shutdown under all conditions. Refer to EOIPM Section 0-II-ZB for discussion of the MSBWP.

1.0 EOI-1, RPV CONTROL BASES (continued)

DISCUSSION: RC/P-3(cont'd)

Other criteria may also be employed to determine that the reactor is shutdown. Possibilities are listed in Note 1. Note 1 identifies the bounding control rod positions that ensure the reactor will remain subcritical without boron under all conditions when Reactor Engineering is not available to support this determination. This instruction requires a positive determination, not only that the reactor *is* subcritical, but that it will *remain* subcritical, without reliance upon boron, under worst-case cold shutdown conditions. The phrase "without boron" does not imply that the condition cannot be met if boron has been injected, but that credit cannot be taken for the negative reactivity contributed by the boron. Control rod insertion alone must provide the necessary shutdown margin.

NOTE	
1	The reactor will remain subcritical without boron under all conditions when: • Any 19 control rods are at position **02** with all other control rods fully inserted OR • All control rods except one are inserted to or beyond position **00** OR • Determined by Reactor Engineering
2	TSC staff may recommend an alternate curve for Station Blackout per 0-AOI-57-1A

WHILE EXECUTING THE FOLLOWING STEPS:	
IF	**THEN**
1 It has NOT been determined that the reactor will remain subcritical without boron under all conditions	**EXIT** RC/L and **ENTER** C5, Level/Power Control
RPV water lvl CANNOT be determined	**EXIT** RC/L and **ENTER** C4, RPV Flooding
PC water lvl CANNOT be maintained below 105 ft OR Suppr chmbr press CANNOT be maintained below 55 psig	**STOP** inj into the RPV from sources external to the PC NOT required for adequate core cooling

C5 Inj System Status		
System	APPX	Inj Stopped and Prevented
CNDS and FW	4	
HPCI	4	
LPCI	4	
CS	4	

STOP and **PREVENT ALL** inj into RPV **EXCEPT** from RCIC, CRD and SLC (APPX 4)

C5-9

WHEN RPV water lvl drops below -50 in.

C5-10

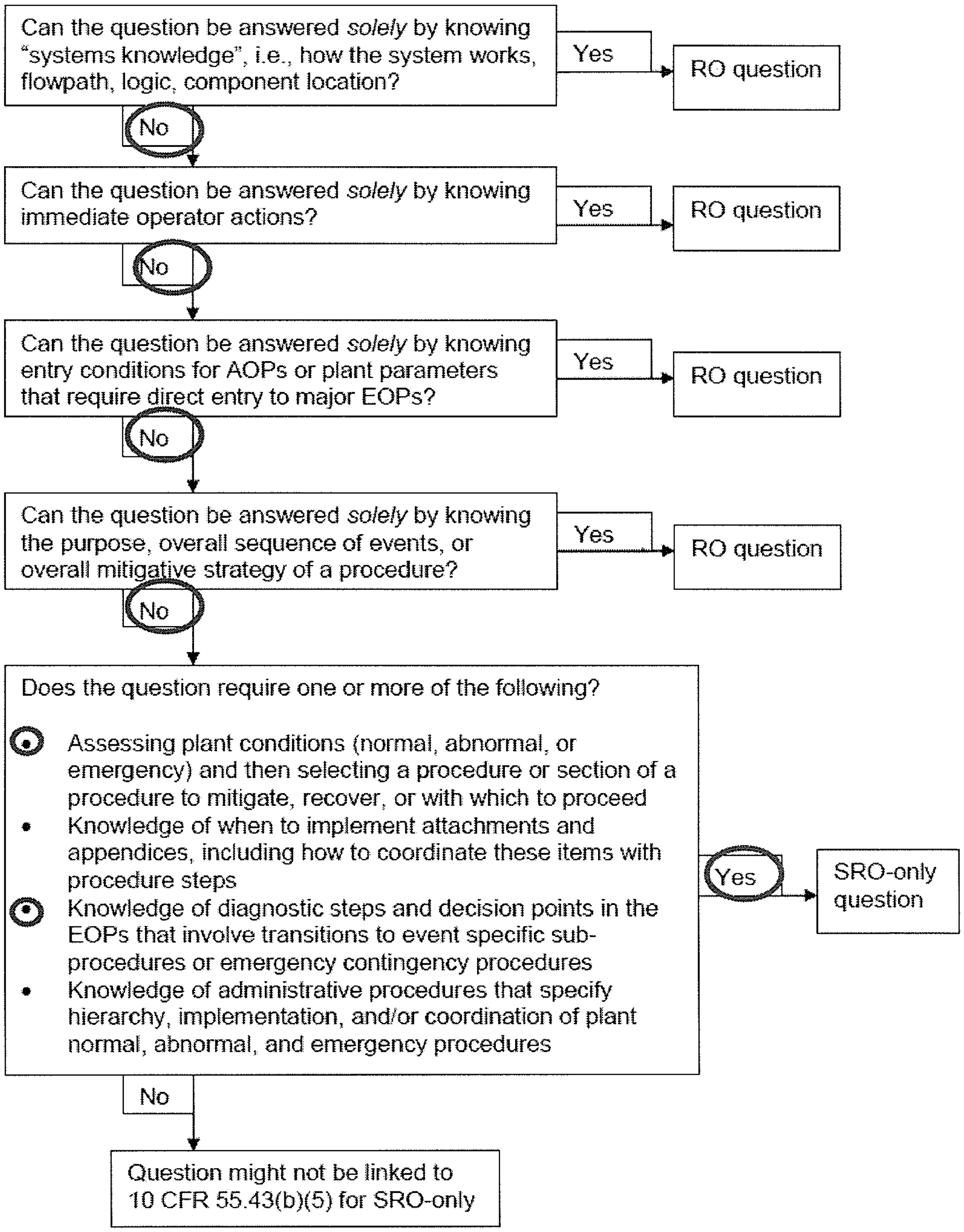
STOP lowering RPV water lvl

C5-11

Record water level for Step C5-11:

________ in.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



80. SRO 295037EA2.05 002

Given the following Unit 1 conditions:

- A Loss of off-site power has occurred.
- OATC reports 26 control rods are at position 02 and the remaining rods are at position 00.
- APRM indications are NOT available.
- RPV pressure is being maintained between 800 to 1000 psig.
- RPV level is (-)20 inches and steady.
- Suppression Pool temperature is 102 °F and steady.

Which ONE of the following describes the appropriate actions to be performed in accordance with Emergency Operating Instructions?

A✓ Level band is (+)2 to (+)51 inches.
SLC should NOT be initiated.
RPV cooldown is permitted.

B. Level band is (-)50 to (-)100 inches.
SLC should be initiated.
RPV cooldown is NOT permitted.

C. Level band is (+)2 to (+)51 inches.
SLC should be initiated.
RPV cooldown is NOT permitted.

D. Level band is (-)50 to (-)100 inches.
SLC should NOT be initiated.
RPV cooldown is permitted.

QUESTION 81

The following conditions exist for Unit 1:

- Mode 3
- Reactor Pressure 75 psig
- Core Spray Loop 2 is INOPERABLE
- RHR Loop 2 is operating in Shutdown Cooling

Which ONE of the following subsystems, if inoperable, would require an Hourly Fire Watch?

[REFERENCE PROVIDED]

A. Preaction System for HPCI, 1-26-37

B. Fire Detection for RCIC on Panel 1-LPNL-25-545

C. Fire Detection for RHR on Panel 1-LPNL-25-545

D. Unit 1 Auxiliary Instrument Room CO2 System

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	600000 AA2.15	
	Importance Rating		3.5

Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Requirements for establishing a fire watch

Explanation: **D CORRECT:** Unit 1 Aux Instrument Room is required to be Operable and with the CO2 system inoperable it is required to establish an hourly fire watch.

A-Incorrect. Plausible in that this would be correct if HPCI was OPERABLE.

B- Incorrect. Plausible in that this would be correct if RCIC was OPERABLE.

C- Incorrect. Plausible because this would require a continuous Fire Watch.

Technical Reference(s): Fire Protection Report Volume 1

Proposed references to be provided to applicants during examination: Fire Protection Report Volume 1

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: Perry 2009 SRO #4

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

10 CFR Part 55 Content: 55.43 (1) Conditions and limitations in the facility license.

9.3.11.D CO_2 SYSTEMS	9.4.11.D CO_2 SYSTEMS
1. The low pressure CO_2 systems protecting the following areas shall be OPERABLE whenever equipment protected by the CO_2 systems is required to be OPERABLE. a. Unit 1 and 2 Diesel Generator Rooms, Auxiliary Board Rooms, Fuel Transfer Pump Rooms b. Unit 3 Diesel Generator Rooms, Auxiliary Board Rooms, and Fuel Transfer Pump Rooms c. Computer Rooms 1, 2, and 3 EL 593, Control Building d. Auxiliary Instrument Rooms 1, 2, and 3 2. With one or more of the above CO_2 systems inoperable, within 1 hour establish an hourly fire watch patrol. NOTE: A delay of fifteen (15) minutes for the start of rounds is acceptable during shift turnover, to facilitate proper turnover with minimal plant impact.	1. Each of the required CO_2 systems shall be demonstrated OPERABLE. a. At least weekly by verifying the CO_2 storage tank level to be greater than 8.5 tons for Units 1 and 2 and 3 tons for Unit 3 and pressure to be greater than 275 psig, and b. At least once per 18 months by verifying: 1. The system, including associated ventilation system fire dampers and fire door release mechanisms, actuates manually and automatically upon receipt of a simulated actuation signal, and 2. Flow from each nozzle during a "Puff Test".

9.3.11.A FIRE DETECTION INSTRUMENTATION	9.4.11.A FIRE DETECTION INSTRUMENTATION
1. As a minimum, the fire detection local control panel shown in Table 9.3.11.A shall be OPERABLE whenever equipment protected by the fire detection Instrument is required to be OPERABLE.	1. Each of the required fire detection instruments shall be demonstrated OPERABLE at least annually by performance of a CHANNEL FUNCTIONAL TEST.
2. With one or more of the above required local control panels inoperable, within one hour establish a continuous fire watch for those areas specifically identified in Table 9.3.11.A; for other areas listed in Table 9.3.11.A, establish an hourly roving fire watch.	2. The supervised circuits associated with alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE semiannually.
a. The fire detection systems heat and smoke detectors for all protected areas shall be OPERABLE. b. If requirement 9.3.11.A.2.a cannot be met, a patrolling fire watch will be established (unless noted otherwise in Table 9.3.11.A) to ensure that each protected fire zone or area with inoperable detectors is checked at intervals no greater than once each hour. NOTE: A delay of fifteen (15) minutes for the start of rounds is acceptable during shift turnover, to facilitate proper turnover with minimal plant impact.	3. The non-supervised circuits associated with alarms of each of the above required instruments shall be Demonstrated OPERABLE at least monthly.

Panel Location (Building - EL)	Local Panel	Area Protected/ Equipment	Detector Type	Function
1. Reactor - 519	1-LPNL-25-545	HPCI	Smoke	Actuate Preaction System
2. Reactor - 519	1-LPNL-25-545	RCIC	Heat/Smoke	Annunciation
*3. Reactor - 519/541	1-LPNL-25-545	RHR	Smoke	Actuate Preaction System
*4. Reactor - 565	1-LPNL-25-545	General Area	Smoke	Actuate Preaction System
*5. Reactor - 593	1-LPNL-25-545	General Area	Smoke	Actuate Preaction System
*6. Reactor - 621	1-LPNL-25-545	General Area	Smoke	Actuate Preaction System
*7. Reactor - 639	1-LPNL-25-545	General Area (South Side) General Area (North Side)	Smoke Smoke	Actuate Preaction System Annunciation

* When one or more of the required fire alarm panels are inoperable, within one hour establish a continuous fire watch for the area(s) of coverage associated with the applicable alarm panel(s). The following is a list of the panels/elevations/systems requiring a continuous fire watch when inoperable:

1-LPNL-25-545 (EL 519/541, 565, 593, 621, 639)/1-26-77

2-LPNL-25-545 (EL 519/541, 565); 2-LPNL-25-546 (EL 593); and/or 2-LPNL-25-547 (EL 621, 639)/2-26-77

3-LPNL-25-545 (EL 565); 3-LPNL-25-546 (EL 593); 3-LPNL-25-547 (EL 621,639)/3-26-77

0-LPNL-25-538(Intake Pumping Station Elevation 550')/0-26-72E

The continuous fire watch will not be stationed in one location, but will move continuously throughout the area(s) normally protected by the alarm panel(s) each hour. The continuous fire watch shall not leave the specified area(s) without a proper relief. Depending on the capability of the fire watch to complete the patrol of the deployment area(s) within the allotted time frame, one fire watch may be responsible for multiple panels/systems located within one or more unit(s) of the Reactor Building. The fire watch for the Intake Pumping Station may not have responsibilities that would require leaving the Intake Pumping Station as part of the patrol.

NOTE: No Compensatory measures are required if a single detector in a fire area/zone is inoperable, unless the detector that is determined inoperable is:

- the only detector in that room/area
- the only detector of a cross-zoned or 2 out of 3 logic actuation system.
- a duct detector

If the detector remains inoperable 30 days or longer, compensatory measures need to be established per section 9.3.11.A.2.b or evaluated by Site Engineering on a case-by-case basis (Reference 2.33).

9.3.11.C SPRAY AND/OR SPRINKLER SYSTEMS	9.4.11.C SPRAY AND/OR SPRINKLER SYSTEMS
1. The spray and sprinkler systems in Table 9.3.11.B shall be OPERABLE whenever equipment protected by the spray and/or sprinkler systems is required to be OPERABLE. Sprinkler and/or spray systems are considered inoperable if their water supply is unavailable.	1. Each of the required spray and systems in Table 9.3.11.B shall be demonstrated OPERABLE: a. Intentionally left blank. b. At least yearly by cycling each testable valve in the flow path Through at least one complete cycle of full travel.
2. With one or more of the above required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch for those areas specifically identified in Table 9.3.11.B; for other areas listed in Table 9.3.11.B, establish a roving hourly fire watch patrol. a. For sprinkler and/or spray systems, the associated sprinkler/spray nozzles for all protected areas shall be OPERABLE. b. If requirement 9.3.11.C.2.a cannot be met, a roving hourly fire watch patrol will be established(unless noted otherwise in Table 9.3.11.B), to ensure that each protected area with inoperable sprinkler/spray nozzles is checked at intervals no greater than once each hour. NOTE: A delay of fifteen (15) minutes for the start of rounds is acceptable during shift turnover, to facilitate proper turnover with minimal plant impact.	c. At least once per 18 months: (1) By performing a system functional test which includes simulated automatic actuation of the system, verifying that the automatic valves in the flow path actuate to their correct positions on a fire alarm test signal. (2) By a visual inspection of the non-air supervised spray and sprinkler headers to verify their integrity. (3) By a visual inspection of each sprinkler or water spray nozzle's spray area to verify that the spray pattern is not obstructed. d. At least once per 3 years, by performing an air flow test through each open head spray header and verifying that each open head spray and sprinkler nozzle is unobstructed.

QUESTION SRO 4

Which one of the following conditions requires an Hourly Fire Watch Patrol?

Reference Provided: PAP-1910 Fire Protection Program Body & Attachment #3

A. RCIC Pump Room Wet-Pipe Sprinkler will not deliver water.

B. Heat Detection for Reactor Recirculation Pump B is out of service.

C. Unit 1 Division 1 Cable Spreading Pre-Action Spray System will not deliver water.

D. General area smoke detectors in Containment are functional but the detection system will not transmit an alarm to SAS.

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

II. Some examples of additional knowledge and abilities as they pertain to an SRO license and the 10 CFR 55.43(b) topics [ES-401, Section D.1.c]:

A. Conditions and limitations in the facility license. [10 CFR 55.43(b)(1)]

Some examples of SRO exam items for this topic include:

- Reporting requirements when the maximum licensed thermal power output is exceeded.
- Administration of fire protection program requirements such as compensatory actions associated with inoperable sprinkler systems, fire doors, etc.
- The required actions for not meeting administrative controls listed in Technical Specification (TS) Section 5 or 6, depending on the facility (e.g., shift staffing requirements).
- National Pollutant Discharge Elimination System (NPDES) requirements, if applicable.
- Processes for TS and FSAR changes.

Note: The analysis and selection of required actions for TS Sections 3 and 4 may be more appropriately listed in the following 10 CFR 55.43 topic.

QUESTION 82

The following conditions exist for Unit 3 after the Reactor scrammed due to a LOCA:

- Reactor water level is (-) 165 inches and lowering
- RHR pump 3A is lined up and injecting to the RPV as the only available injection source
- 0-AOI-57-1E, Grid Instability, has been entered
- RHR pump 3A amps are in the Red band

Which ONE of the following completes both statements below?

RHR pump 3A __(1)__ remain running.

The HIGHEST given water level at which Emergency Depressurization is required is __(2)__ inches.

A. (1) can
   (2) (-) 180

B. (1) can
   (2) (-) 195

C. (1) CANNOT
   (2) (-) 180

D. (1) CANNOT
   (2) (-) 195

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	700000 G2.4.9	
	Importance Rating		4.2

700000 Generator Voltage and Electric Grid Disturbances G2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Explanation: **A CORRECT:** Part 1: CORRECT- RHR pump 3A can remain running as it is required to maintain adequate core cooling. Part 2: CORRECT- Emergency Depressurization would be required prior to (-)180 inches IAW 3-C-1, Alternate Level Control.

B- Incorrect: Part 1: CORRECT- See A. Part 2: Incorrect- This is plausible as Emergency Depressurization is required at (-) 195 inches IAW 3-C-1, Alternate Level Control, while in steam cooling.

C- Incorrect: Part 1: Incorrect- This is plausible as pumps not required for adequate core cooling that are operating in the red band should be secured IAW 0-AOI-57-1E, Grid Instability. Part 2: CORRECT- See A.

D- Incorrect: Part 1: Incorrect- See C. Part 2: Incorrect-See B. In addition if RHR pump A was secured that would place the plant in steam cooling in which (-) 195 inches would be the correct answer.

Technical Reference(s): 3-C-1, Alternate Level Control; 0-AOI-57-1E, Grid Instability

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.205 Rev 10 ILT Obj. 2

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

4.2 Subsequent Action (continued)

[7.4] IF system flowrates are high due to abnormal grid conditions, THEN

ADJUST flow per appropriate OI (within allowed limits) to lower flowrate as desired. ☐

[7.5] IF system pump amps are high due to abnormal grid conditions, THEN

ADJUST flow per appropriate OI to lower pump amperage rates to allowed limits (ensure less than red band on control room ammeters) per the appropriate OI. ☐

NOTE

Pumps that are operating in the control room ammeter red band should be secured unless required to assure adequate core cooling as directed by the EOI's.

[7.6] IF system pump flowrates and/or amps cannot be restored within normal bands, THEN

EVALUATE the need to secure the affected pump/system. ☐

[7.7] IF Diesel Generators are supplying the 4 kV Shutdown Boards, THEN

PERFORM applicable sections of 0-AOI-57-1A in parallel with this instruction. ☐

3-C-1 ALTERNATE LEVEL CONTROL Rev 12

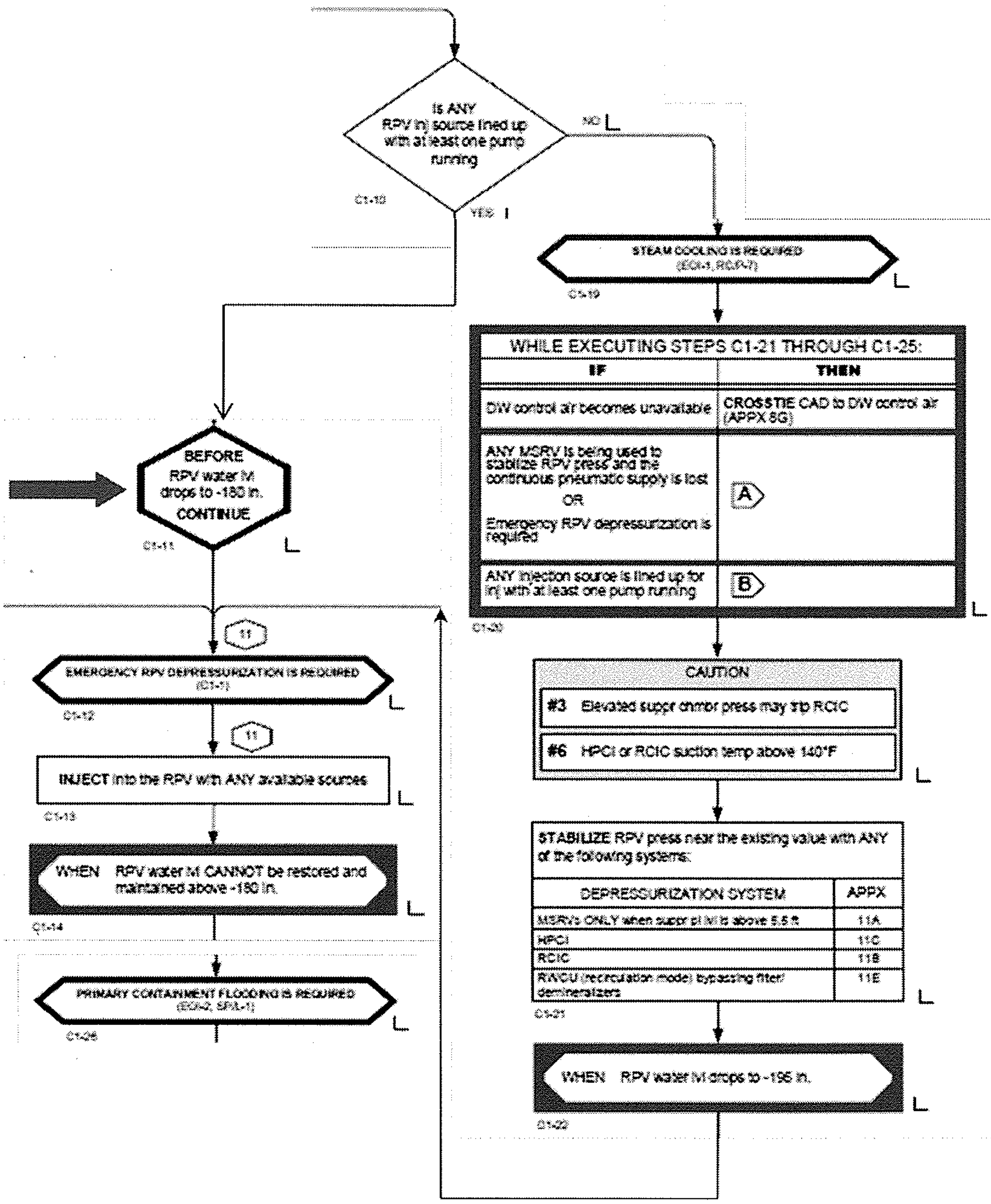
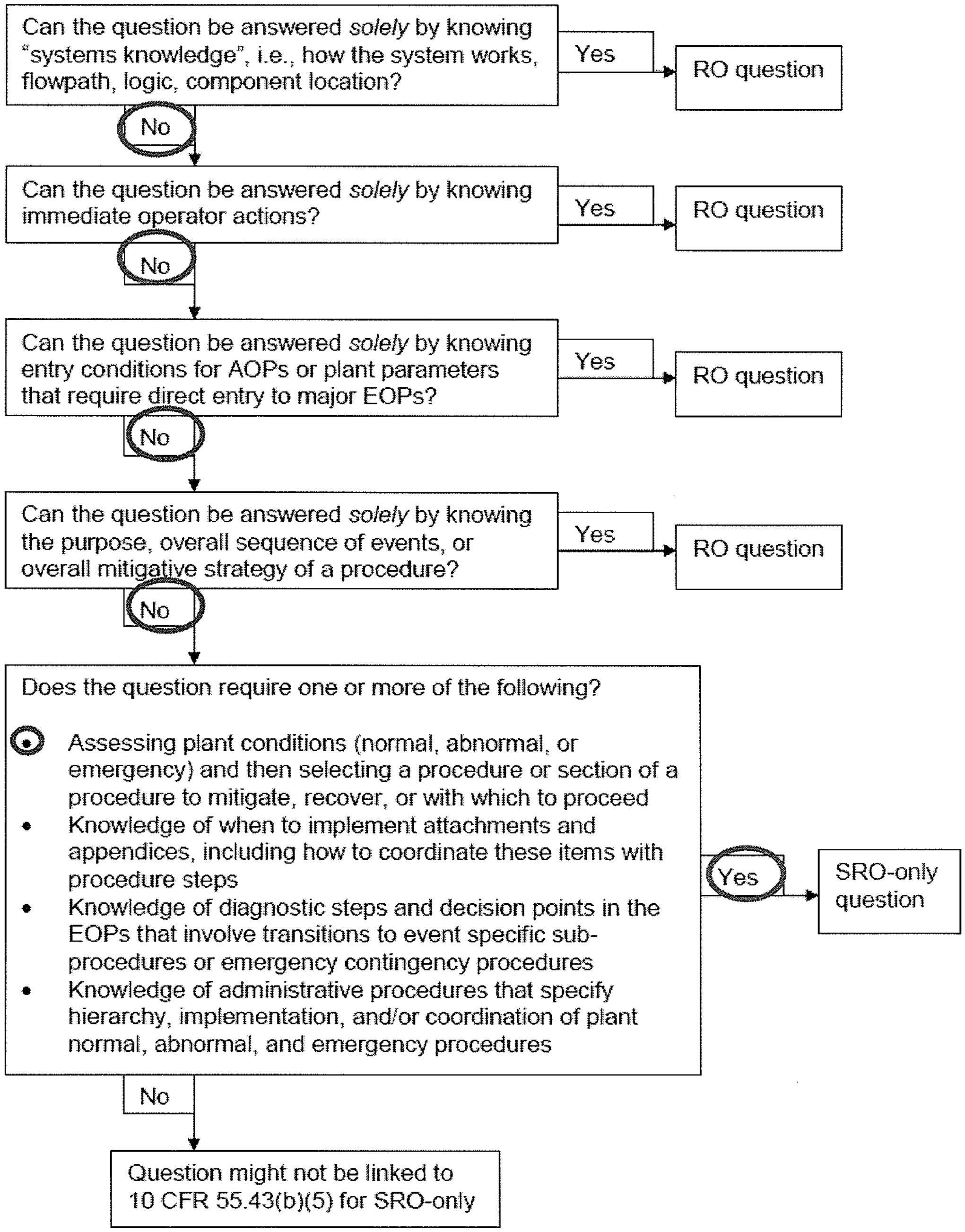


Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



QUESTION 83

During a Unit 2 startup, the following conditions exist:

- REACTOR MODE SWITCH in STARTUP/STANDBY
- Reactor pressure is 855 psig
- Control rod 22-11 is at position 00, its nitrogen accumulator has a cracked weld and is isolated for repair.

The operating Control Rod Drive (CRD) pump trips and the following conditions are noted:

- CRD Charging Water Header Pressure 900 psig
- CRD ACCUM PRESS LOW/LEVEL HIGH (2-9-5A Window 29) annunciator is received for the following rods:

Rod	Position	Accumulator Pressure
18-27	00	900 psig
38-23	48	900 psig

Which ONE of the following are the correct actions to take in accordance with Technical Specification 3.1.5, Control Rod Scram Accumulators?

A. Declare BOTH control rods inoperable immediately AND immediately place the REACTOR MODE SWITCH in SHUTDOWN.

B. Fully insert control rod 38-23 and declare BOTH control rods inoperable within 1 hour. If the first action is NOT met, immediately place the REACTOR MODE SWITCH in SHUTDOWN.

C. Declare ONLY control rod 38-23 inoperable within 1 hour. If any other accumulator becomes inoperable, immediately place the REACTOR MODE SWITCH in SHUTDOWN.

D. Declare control rod 38-23 slow within 1 hour AND if charging header pressure CANNOT be restored to at least 940 psig within 20 minutes, place the REACTOR MODE SWITCH in SHUTDOWN.

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	295022 AA2.01	
	Importance Rating		3.6

Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS : Accumulator pressure

Explanation: **B CORRECT:** For the given condition this is action C, and if C.1 cannot be completed within one hour, then action D requires the mode switch in shutdown.

A-Incorrect. Plausible in that both Control Rods are inoperable but there is no immediate requirement to declare inoperable. In addition the requirement to place the mode switch in shutdown is not immediate unless certain conditions are NOT met within one hour.

C- Incorrect. Plausible in that this is part of Condition B and part of Condition C and part of Condition D.

D- Incorrect. Plausible in that this is the correct answer for Condition B with Reactor Pressure greater than 900 psig.

Technical Reference(s): Unit 2 Tech Spec 3.1.5

Proposed references to be provided to applicants during examination: Unit 2 Tech Spec 3.1.5

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: Perry 2004 #84

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

10 CFR Part 55 Content: 55.41 (2) Facility operating limitations in the technical specifications and their bases.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Control Rod Scram Accumulators

LCO 3.1.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each control rod scram accumulator.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control rod scram accumulator inoperable with reactor steam dome pressure ≥ 900 psig.	A.1	-------------NOTE----------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. --------------------------------	
		Declare the associated control rod scram time "slow."	8 hours
	<u>OR</u>		
	A.2	Declare the associated control rod inoperable.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure $\geq$ 900 psig.	B.1 Restore charging water header pressure to $\geq$ 940 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig
	AND	
	B.2.1 --------------NOTE------------ Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. --------------------------------- Declare the associated control rod scram time "slow."	1 hour
	OR	
	B.2.2 Declare the associated control rod inoperable.	1 hour

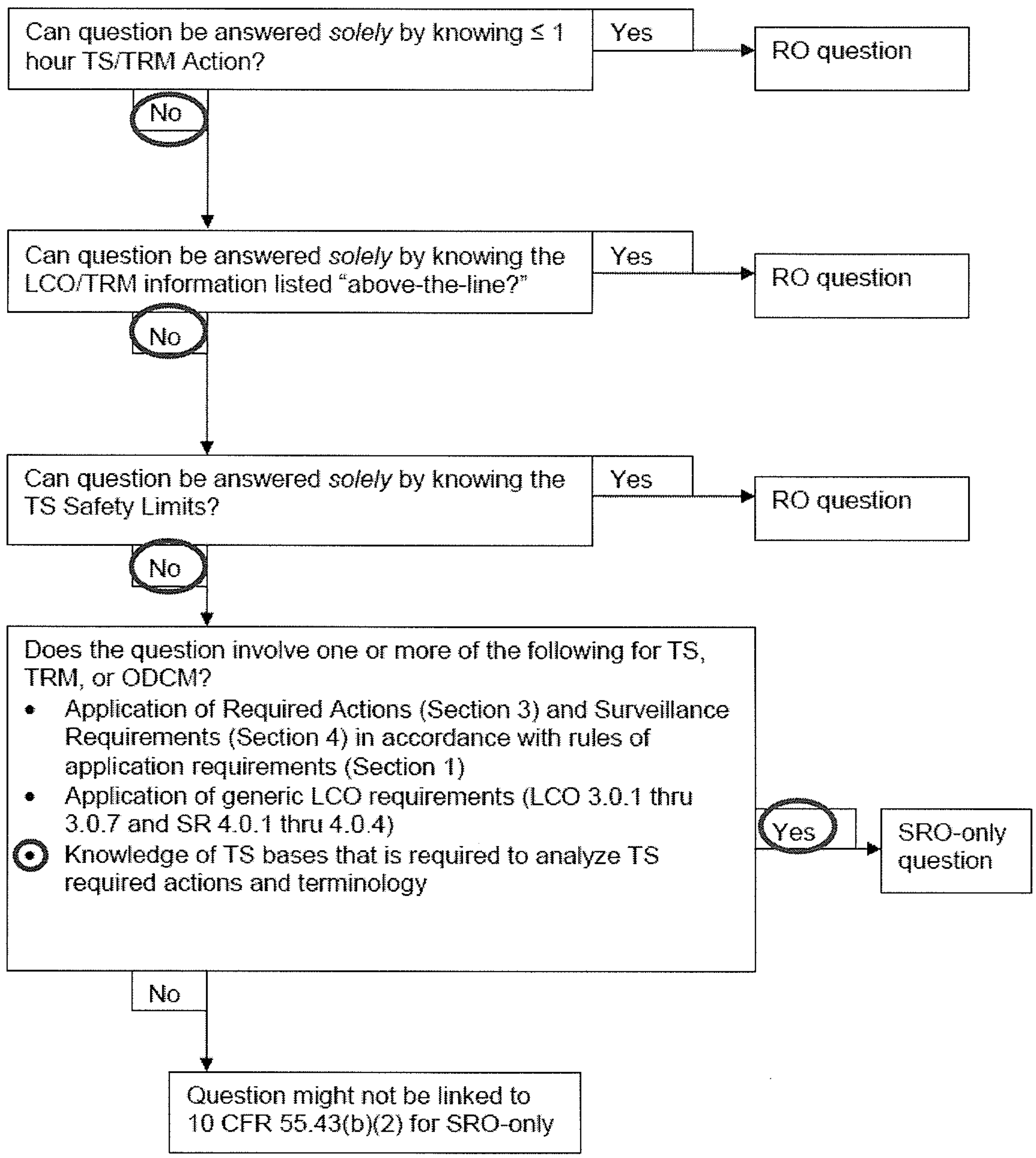
(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 900 psig.	C.1 Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < 940 psig
	<u>AND</u>	
	C.2 Declare the associated control rod inoperable.	1 hour
D. Required Action and associated Completion Time of Required Action B.1 or C.1 not met.	D.1 --------------NOTE------------ Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. --------------------------------- Place the reactor mode switch in the shutdown position.	Immediately

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



Perry 2004 #84

QUESTION 084

During a plant startup, the following conditions exist:

- REACTOR MODE SWITCH in STARTUP/STANDBY
- Reactor pressure is 855 psig.
- Control rod 22-11 is at position 00, its nitrogen accumulator has a cracked weld and is isolated for repair.

The operating Control Rod Drive (CRD) pump trips, CRD Charging Header Pressure indicates 50 psig, and the CRD HCU LEVEL HI/PRESS LO annunciator is received for the following rods:

Rod	Position	Accumulator Pressure
18-27	00	1500 psig
38-23	48	1500 psig

Which ONE of the following should you direct the control room operators to do?

a. Declare both CRD accumulators INOPERABLE and have the Supervising Operator place the REACTOR MODE SWITCH to SHUTDOWN.

b. Declare control rod 38-23 accumulator INOPERABLE; insert and isolate control rod 38-23 within 1 hour, or place the REACTOR MODE SWITCH to SHUTDOWN.

c. If charging header pressure CANNOT be restored to at least 1600 psig within 20 minutes, place the REACTOR MODE SWITCH to SHUTDOWN. Both control rods are still OPERABLE.

d. Declare control rod 18-27 and 38-23 INOPERABLE. Monitor accumulator status. If any other accumulator becomes INOPERABLE, immediately place the REACTOR MODE SWITCH to SHUTDOWN.

ANSWER: **B**

QUESTION 84

Unit 1 has experienced a LOCA and the following containment parameters exist:

- All Control Rods fully inserted
- MSIVs are Open
- Drywell Pressure is 23.4 psig and rising
- Suppression Chamber Pressure is 22 psig and lowering slowly
- Hydrogen concentration in the Drywell is 2.9%
- Suppression Pool Level is 15 feet
- Emergency Depressurization in progress
- Reactor Water Level is (-) 170 inches and rising

Which ONE of the following completes the statements below?

The required procedure to vent primary containment is __(1)__ .

Vent under these conditions __(2)__ .

A. (1) 1-EOIAPPENDIX-12, Primary Containment Venting
(2) irrespective of offsite radioactive release rates

B. (1) 1-EOIAPPENDIX-12, Primary Containment Venting
(2) ONLY if offsite radioactive release rates can be maintained below ODCM limits

C. (1) 1-EOIAPPENDIX-15, RPV Venting for Primary Containment Flooding
(2) irrespective of offsite radioactive release rates

D. (1) 1-EOIAPPENDIX-15, RPV Venting for Primary Containment Flooding
(2) ONLY if offsite radioactive release rates can be maintained below ODCM limits

ANSWER: **B**

	Level:	RO	SRO
	Tier #		1
	Group #		2
Examination Outline Cross-Reference	K/A#	295010 G2.4.20	
	Importance Rating		4.3

295010 High Drywell Pressure
2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.

Explanation: **B CORRECT:** Part 1- CORRECT: 1-EOI-2, requires PC vented when Hydrogen is detected at greater than 2.4% IAW 1-EOI APPENDIX-12, Primary Containment Venting. Part 2- CORRECT- Venting is to be stopped if offsite radioactive release rate reach ODCM limits

A-Incorrect. Part 1: correct Second Part- Incorrect:Plausible because if venting IAW appendix 13 and / or Appendix 15 Off site release rates limits may be exceeded.

C- Incorrect. Part 1- Incorrect, plausible in that this would be correct if the SRO remained in the Primary Containment Flood flow path if RPV water level did not recover above minus 180 inches. Part 2- Incorrect for appendix 15 venting stack release rates would be maintained within table 7 limits and the off Gas Release rate limits may be exceeded

D-Incorrect. Part 1-Incorrect: See C.Part 2- Correct: See B.

The Current Venting procedures are rarely used in training due to the conditions needed to meet the requirements to vent.

Technical Reference(s): 1-EOI-2;1-EOI-C-1, 1-EOIAPPENDIX-12; 1-EOI APPENDIX-15

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X	
Modified Bank:		
	New:	

Question History: Previous NRC: BFN 1006 NRC

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

10 CFR Part 55 Content: 55.41 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

PRIMARY CONTAINMENT VENTING

CAUTION

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

ADJUST 1-FIC-84-19, PATH B VENT FLOW CONT, or 1-FIC-84-20, PATH A VENT FLOW CONT, as applicable, to maintain ALL of the following:

Stable flow as indicated on controller,
AND
□ 1-PA-84-21, VENT PRESS TO SGT HIGH, alarm light extinguished,
AND
□ Release rates as determined below:

iii. IF Venting for ANY other reason than items i or ii above, THEN **MAINTAIN** release rates below

□ Stack release rate of 1.4×10^7 µCi/s
AND
□ 0-SI-4.8.B.1.a.1 release fraction of 1.

EOI-2

PC/H

WHILE EXECUTING THE FOLLOWING STEPS:	
IF	**THEN**
H_2 and O_2 monitoring system is inoperable	**NOTIFY** Chem Lab to sample for DW and suppr chmbr H_2 and O_2
Offsite radioactivity release rate reaches ODCM limits OR H_2 is NO longer detected in PC (2.4% on control room indicators)	**SECURE** PC purge AND **SECURE** PC vent NOT required by other EOI steps

PC/H-1

VERIFY H2/O2 analyzer in service (APPX 19)

PC/H-2

WHEN H_2 is detected in PC (2.4% on control room indicators)

PC/H-3

IF offsite radioactivity release rate is expected to remain below ODCM limits

THEN **VENT** and **PURGE** PC as follows:

1. **NOTIFY** Chem Lab to sample for offsite radioactivity release
2. **VENT** PC (APPX 12)
3. **IF** PC can be vented

   THEN **PURGE** PC with nitrogen makeup (APPX 14A)

PC/H-4

EOI Appendix-15

RPV VENTING FOR PRIMARY CONTAINMENT FLOODING

CAUTION

Off-Gas Release Rate Limits may be exceeded.

PRIMARY CONTAINMENT FLOODING IS REQUIRED
(EOI-2, SP/L-1)

C1-26

C1-33

VENT the RPV. **MAINTAIN** offsite radioactivity release rates within Table 7 limits (APPX 15)

C1-34

Table 7 Max Post LOCA Stack Release Rates	
Time After Core Uncovery (hr)	Noble Gases (µCi/sec)
0 – 0.5	3.12×10^8
0.5 – 2	1.86×10^7
2 – 8	8.35×10^7
8 – 24	7.25×10^7
24 – 48	5.38×10^7
48 - 96	3.96×10^7
96 – 120	3.10×10^7
120 – 240	1.97×10^7
240 – 480	2.89×10^7
480 – 720	4.53×10^6

WHEN PC water lvl reaches 90 ft

C1-35

MAINTAIN PC water lvl between 90 ft and 105 ft with the following inj sources taking suction from sources external to the PC ONLY as required:

INJ SOURCE	APPX	INJ PRESS
CNDS	6A	480 psig
CRD	5B	1640 psig
LPCI	10C, 10D 6B, 6C	320 psig
Stby coolant	7D	160 psig

Question 97 1006 Exam

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 23.4 psig and lowering slowly
- Suppression Chamber Pressure is 22 psig and lowering slowly
- Suppression Pool Level is 15 feet
- MSIVs are OPEN
- Reactor has been Emergency Depressurized
- Reactor Water Level lowered to (-) 180 inches and is now (-) 170 inches and rising

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

In accordance with the EOIs, venting the Primary Containment is required to be performed using __________.

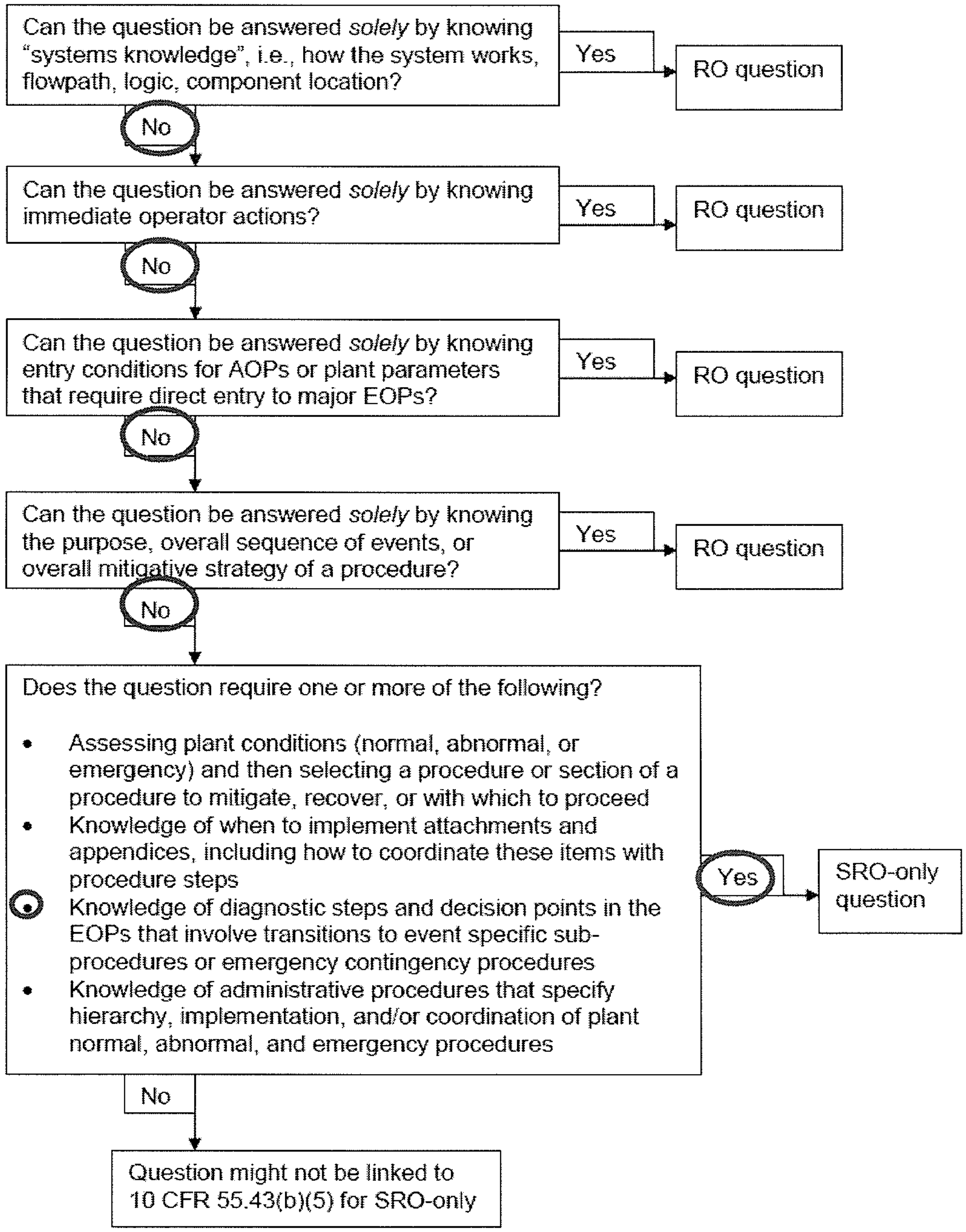
A. 3-EOI APPENDIX-12, “Primary Containment Venting,” irrespective of radioactive release rates

B. 3-EOI-APPENDIX-15,”RPV Venting for Primary Containment Flooding,” 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-15,”RPV Venting for Primary Containment Flooding,” **ONLY** if radioactive release rates can be maintained below ODCM limits

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



QUESTION 85

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

Following the Scram a primary system began discharging into Secondary Containment, with the following alarms and indications:

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

In accordance with 1-EOI-3, Secondary Containment Control, the crew is required to enter __(1)__ AND a potential isolation source is __(2)__.

	(1)	(2)
A.	1-EOI-1, RPV Control	SDV vents and drains
B.	1-EOI-1, RPV Control	RWCU suction and return isolation valves
C.	0-EOI-4, Radioactivity Release Control	SDV vents and drains
D.	0-EOI-4, Radioactivity Release Control	RWCU suction and return isolation valves

Answer: **A**

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

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Cause of high area radiation

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

B – Incorrect – First Part: 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. Second Part: Incorrect, The 565 elevation Northeast has no possible isolation sources listed in EOI-3 table 4.

C– Incorrect – First Part: 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. Second Part: Correct, SDV valves are the correct source choice.

D – Incorrect – First Part: 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. Second Part: 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).

Technical Reference(s) 1-EOI-1, 1-EOI-3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: BFN 0801 #85

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

10 CFR Part 55 Content: 55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

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
1-RE-090-0028	RHR East, Rx Bldg. El 519', R-6 U-Line
1-RE-090-0029	Suppression Pool, Rx Bldg. El 519', R-5 U-Line

Probable Cause:

A. Radiation levels have risen above alarm point.
B. Dry Cask Storage activities in progress (activities could affect rad levels sensed by 1-RE-090-0004, 1-RE-090-0009, 1-RE-090-0014, 1-RE-090-0021).

NOTE

Due to location of the Rad Monitor in relation to the test line in the HPCI quad, the HPCI Room rad alarm may be received when the HPCI flow test is in progress.

C. HPCI Flow Rate Surveillance in progress.

Continued on Next Page

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

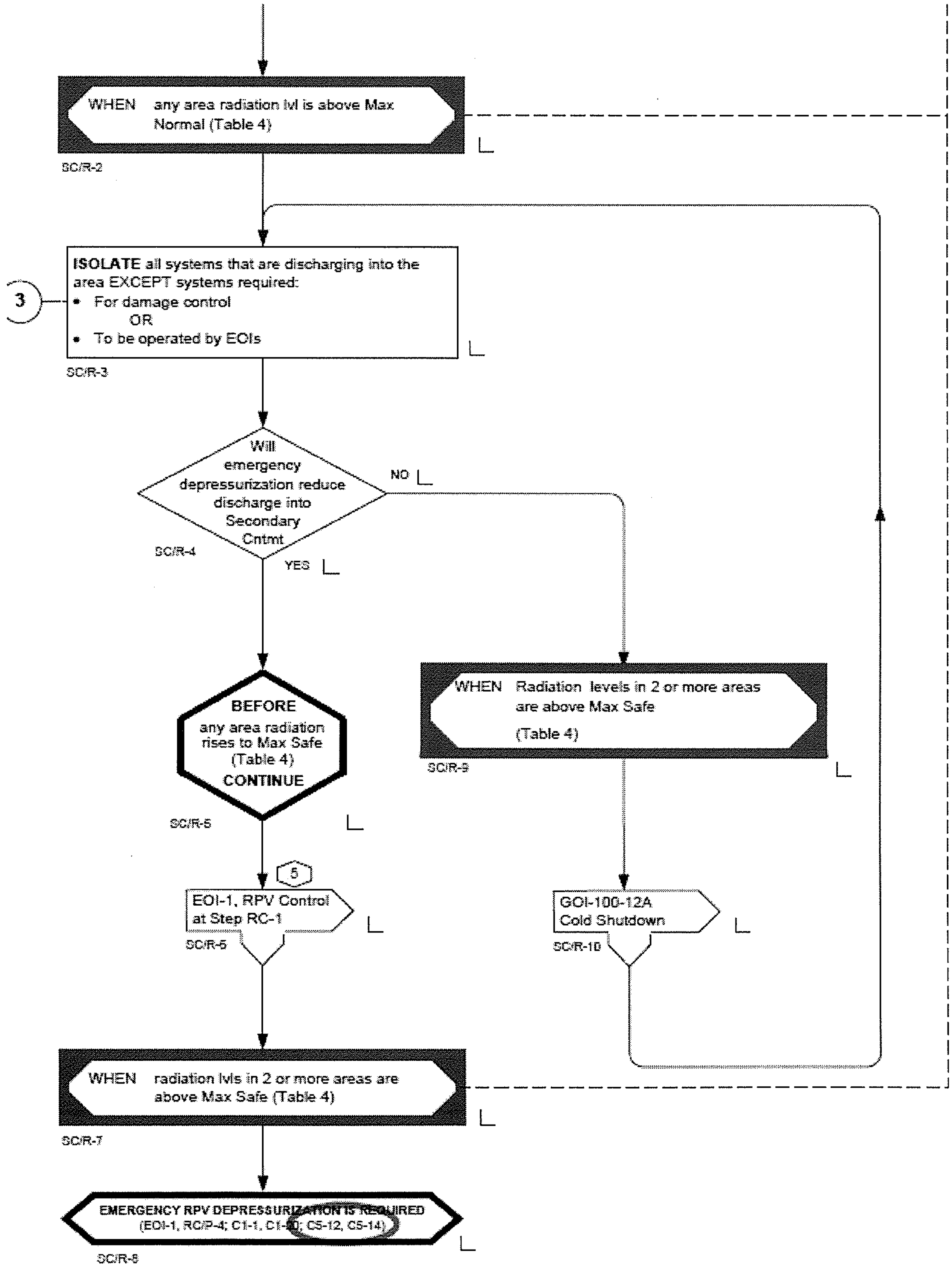
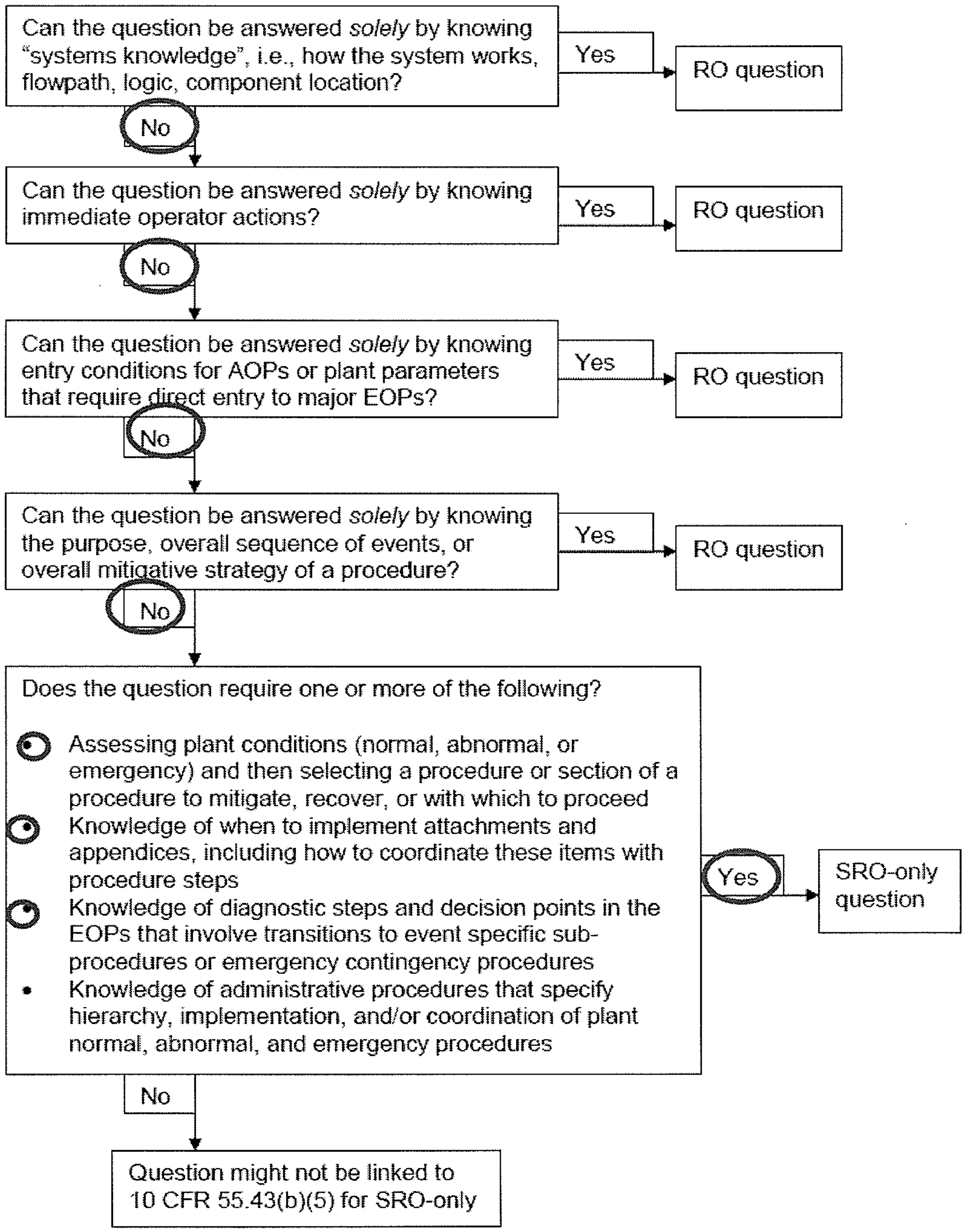

WHEN any area radiation lvl is above Max Normal (Table 4)
SC/R-2
ISOLATE all systems that are discharging into the area EXCEPT systems required:
• For damage control
OR
• To be operated by EOIs
3
SC/R-3
Will emergency depressurization reduce discharge into Secondary Cntmt
SC/R-4
NO
YES
BEFORE any area radiation rises to Max Safe (Table 4) CONTINUE
SC/R-5
WHEN Radiation levels in 2 or more areas are above Max Safe (Table 4)
SC/R-9
5
EOI-1, RPV Control at Step RC-1
SC/R-6
GOI-100-12A Cold Shutdown
SC/R-10
WHEN radiation lvls in 2 or more areas are above Max Safe (Table 4)
SC/R-7
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED (EOI-1, RC/P-4; C1-1, C1-20; C5-12, C5-14)
SC/R-8

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



BFN 0801 #85

Examination Outline Cross-reference:	Level	RO	SRO
295033 High Secondary Containment Area Radiation Levels / 9	Tier #	------	1
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:	Group #	------	2
• Cause of high area radiation	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.

Following the Scram a primary system began discharging into Secondary Containment, with the following alarms and indications:

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

In accordance with 1-EOI-3, "Secondary Containment Control," the crew is required to enter __(1)__ AND a potential isolation source 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.

QUESTION 86

Unit 2 was shutdown due to RHR Loop II being inoperable for 6 days.

- RHR Loop II is tagged out for maintenance on the 2-FCV-74-66, Outboard LPCI Injection Valve
- RHR Loop I is in Shutdown Cooling with RHR pump 2A

Which ONE of the following completes both statements below?

In accordance with Technical Specification 3.5.2, ECCS Shutdown, for these plant conditions, __(1)__ low pressure ECCS injection/spray subsystem(s) is(are) required to be Operable.

While in Shutdown Cooling, RHR pump 2A __(2)__ be considered an Operable ECCS subsystem.

A. (1) one
(2) can

B. (1) one
(2) CANNOT

C. (1) two
(2) can

D. (1) two
(2) CANNOT

ANSWER: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	203000 G2.2.38	
	Importance Rating		4.5

203000 Residual Heat Removal /Low Pressure Coolant Injection: Injection Mode (Plant Specific)
Knowledge of conditions and limitations in the facility license.

Explanation: **C CORRECT:** 3.5.2 BASES requires 2 pumps. It also provides direction on the ability to align (manual / remote) and still maintain operability. Additionally 90 psig is less than the permissive for SDC, therefore it is permitted per TSs.

A- Incorrect. First Part: Incorrect. Plausible because the candidate may be aware of that one subsystem can be in SDC in TS 3.5.2, and therefore may conclude that only one subsystem is required for LPCI. Second Part: Correct.

B- Incorrect. First Part: Incorrect. Plausible because the candidate may be aware of that one subsystem can be in SDC in TS 3.5.2, and therefore may conclude that only one subsystem is required for LPCI. Second Part: Incorrect.

D- Incorrect. First Part: Correct. Second Part: Incorrect. Plausible because 3.5.2 BASES provides direction on the ability to align (manual / remote) and still maintain operability.

Technical Reference(s): TS 3.5.1, TS 3.5.2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: None
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis : X
10 CFR Part 55 Content:	55.41 (2) Facility operating limitations in the TS and their bases.

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

3.5.2 ECCS - Shutdown

LCO 3.5.2 Two low pressure ECCS injection/spray subsystems shall be OPERABLE.

APPLICABILITY: MODE 4,
MODE 5, except with the spent fuel storage pool gates removed and water level ≥ 22 ft over the top of the reactor pressure vessel flange.

BASES (continued)

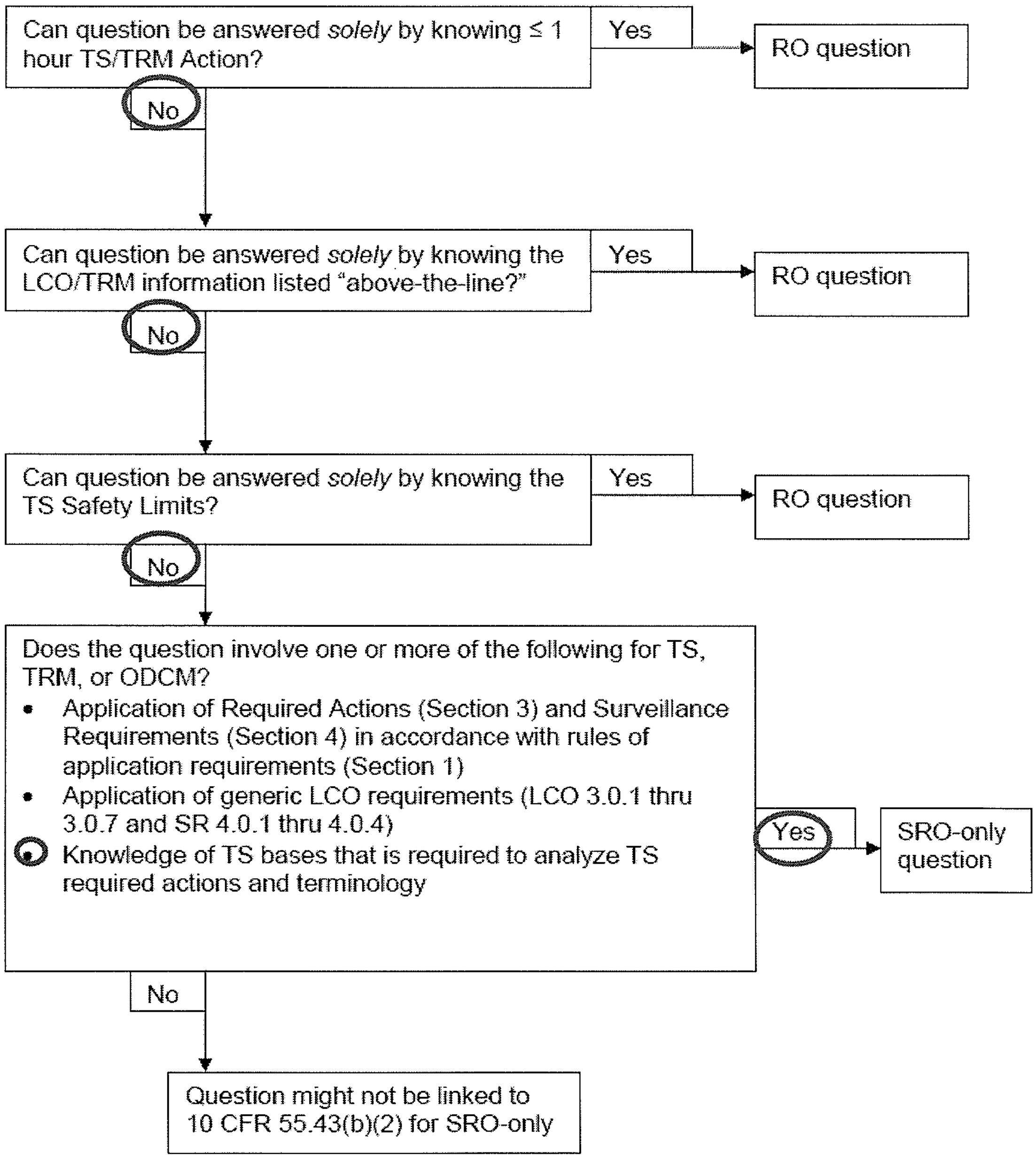
LCO

Two low pressure ECCS injection/spray subsystems are required to be OPERABLE. The low pressure ECCS injection/spray subsystems include CS subsystems and LPCI subsystems. Each CS subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool to the reactor pressure vessel (RPV). Each LPCI subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. In MODES 4 and 5, the LPCI crosstie valve is not required to be closed. The necessary portions of the Emergency Equipment Cooling Water System are also required to provide adequate cooling to each required ECCS subsystem.

An LPCI subsystem may be aligned for decay heat removal and considered OPERABLE for the ECCS function, if it can be manually realigned (remote or local) to the LPCI mode and is not otherwise inoperable. Because of low pressure and low temperature conditions in MODES 4 and 5, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncovery.

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



QUESTION 87

Unit 3 is at 40% power with power ascension in progress with the following conditions:

- Core flow is 55 Mlbm/hr
- APRM #3 has failed downscale and is bypassed

Subsequently,

- Recirc Loop "A" flow transmitter input to APRM #1 fails UPSCALE

Which ONE of the following (1) describes the plant response and (2) the required operator action(s)?

[REFERENCE PROVIDED]

A. (1) Rod Block ONLY
(2) Enter LCO 3.3.1.1 Condition A ONLY.

B. (1) Rod Block ONLY
(2) Enter LCO 3.3.1.1 Conditions A and B.

C. (1) Rod block and an Upscale trip input to ALL 4 voters
(2) Enter LCO 3.3.1.1 Condition A ONLY.

D. (1) Rod block and an Upscale trip input to ALL 4 voters
(2) Enter LCO 3.3.1.1 Conditions A and B.

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	215005 A2.05	
	Importance Rating		3.6

Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of recirculation flow signal

Explanation: **A CORRECT:** Per T.S. table 3.3.1.1-1, 3 APRMs are required. With one less than the required number an APRM must be restored within 12 hours or the channel placed in a tripped condition. A rod block will occur due to >10% difference in total recirc flow signals from the APRMs. APRM #1 summer will have an output > 75%, Other APRM flow summers will indicate approximately 50% at this core flow.

B-Incorrect. First Part: Correct. Incorrect. An APRM Upscale trip will NOT be generated on HIGH recirc flow. The Recirc flow transmitter failing upscale will cause the flow biased scram setpoint to increase. Second Part: Incorrect. LCO 3.3.1.1 B does not apply because it does not affect both trip systems.

C- Incorrect. First Part: Incorrect. An APRM Upscale trip will NOT be generated on HIGH recirc flow. The Recirc flow transmitter failing upscale will cause the flow biased scram setpoint to increase. Second Part: Correct.

D- Incorrect. First Part: Incorrect. An APRM Upscale trip will NOT be generated on HIGH recirc flow. The Recirc flow transmitter failing upscale will cause the flow biased scram setpoint to increase. Second Part: Incorrect. LCO 3.3.1.1 B does not apply because it does not affect both trip systems.

Technical Reference(s): TS LCO 3.3.1.1, OPL171.007, 3-ARP-9-5A window 7

Proposed references to be provided to applicants during examination: Unit 3 TS 3.3.1.1

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: Nine Mile 2 2010 #88
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis : X
10 CFR Part 55 Content:	55.41 (2) Facility operating limitations in the TS and their bases.

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	<u>OR</u>	
	A.2 -------------NOTE------------ Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ---------------------------------	
	Place associated trip system in trip.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -------------NOTE------------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. ------------------------------------ One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	OR		
	B.2	Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours

(continued)

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

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

(continued)

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

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

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

2. Average Power Range Monitor (APRM)/Rod Block Monitor (RBM) System.

    a. The Recirculation Loop flow elements (68-5, 68-81) provide the flow signals to the APRM for flow biased setpoint determination.

    b. The APRM flow signals are then sent to the Rod Block Monitor, where they are evaluated for differences in flow between APRM drawers (5% disparity will result in 'flow compare' alarm on panel drawer).

    c. Failure of one of the two flow transmitters at high power will result in a reduction in the measured flow, with possible rod block or scram signals resulting.

ILT Objective 19b
LOR Objective 9c

CONTROL ROD WITHDRAWAL BLOCK 7 (Page 1 of 2)	Sensor/Trip Point:		
	Relays:	3A-K1	Nuclear Instrumentation
		3A-K2	Refuel Equipment in Use
			High Level in Scram Discharge Volume
			Scram Discharge Volume High Water Level Bypass
			Rx Mode Switch in SHUTDOWN
			PRNM (ANY APRM OPRM or RBM)

Sensor Location: Panel 3-9-28
Elevation 593'
Aux Instr Room

Probable Cause:
A. One or more sensors at or above set point.
B. Malfunction of sensor.
C. Control rod drop accident.

Automatic Action: Rod withdrawal block.

Operator Action:

A. **DETERMINE** initiating condition from corresponding rod withdrawal block alarm(s) and **REFER TO** operator action for alarm(s). ☐
B. **IF** alarm due to inadvertent criticality during incore fuel movements, **THEN**
REFER TO 3-AOI-79-2. ☐
C. **IF** alarm is from a control rod drop, **THEN**
REFER TO 3-AOI-85-1. ☐
D. **IF** NO corresponding alarm exists, **THEN**
   1. AT ICS console, **DETERMINE** if there is a refuel rod block by selecting single Point menu, Single Value display, and typing C602, Return. ☐
   2. **IF** rod block was from Refuel Floor, **THEN**
      CALL Refuel Floor Operator to have dummy plug (Refuel floor between cavity and pool, south side) checked and check jumpers in U-3 Aux Inst. Rm, Panel 3-9-28 Bay 3, if Installed per 3-OI-85 Section 8.34. ☐
   3. **WHEN** IRM switches are below Range 3 with REACTOR MODE SWITCH **NOT** in RUN, **THEN**
      CHECK SRM detectors **NOT** FULL IN. ☐
   4. **WHEN** REACTOR MODE SWITCH is in START-UP position, **THEN**
      CHECK IRM detectors **NOT** FULL IN. ☐

Continued on Next Page

Nine Mile Point Unit 2 2010 NRC SRO Written Examination

Facility: Nine Mile Point Unit 2

Vendor: GE

Exam Date: 2010

Exam Type: S

Examination Outline Cross-reference:	Level		SRO
	Tier #		2
	Group #		1
	K/A #	215005	A2.05
	Importance Rating		3.6

Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of recirculation flow signal

Question: SRO #88

The plant is at 40% power with a power ascension in progress. Core flow is 55 Mlbm/hr. APRM #3 has failed downscale and is bypassed.

With these initial conditions, the Recirc Loop "A" flow transmitter inputting to APRM #1 fails UPSCALE.

Which one of the following describes the plant response and required operator action(s)?

A. Rod block and a ½ scram on RPS "A".
Enter LCO 3.3.1.1 A. No additional action is required for RPS due to existing ½ scram.

B. Rod Block ONLY.
Enter LCO 3.3.1.1 A. If an inoperable APRM channel is not restored within 12 hours, insert an APRM upscale trip.

C. Rod Block ONLY.
Enter LCO 3.3.1.1 C. If an inoperable APRM channel is not restored within 1 hour, insert an APRM upscale trip.

D. Rod block and a ½ scram on RPS "A".
Enter LCO 3.3.1.1 B. No additional action is required for RPS due to existing ½ scram.

Answer: B

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2)
(Tech Specs)

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?

Yes → RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes → RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes → RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- Knowledge of TS bases that is required to analyze TS required actions and terminology

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(2) for SRO-only

QUESTION 88

Given the following plant conditions:

- Unit 1 is starting up currently in Mode 2
- Unit 2 is operating in Mode 1
- Unit 3 is in Mode 5 with fuel movement in progress
- Diesel Generator 3ED is out of service

A Unit Operator reports that while attempting to start SGT Fan 'B' that the fan will NOT start.

Which ONE of the following completes the statement below?

In accordance with Tech Spec 3.6.4.3, Standby Gas Treatment System, ____.

[REFERENCE PROVIDED]

A. fuel movement can continue on Unit 3. Unit 1 AND 2 operation is permitted for 7 days

B. fuel movement can continue on Unit 3. Enter LCO 3.0.3 on Unit 1 AND 2 in 4 hours

C. Unit 2 must be in MODE 3 in 12 hours. Unit 3 fuel movement AND Unit 1 startup can continue for 4 hours

D. IMMEDIATELY suspend fuel movement on Unit 3. IMMEDIATELY enter LCO 3.0.3 on Unit 1 AND 2

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	261000G2.2.4	
	Importance Rating		3.6

261000 SGTS G2.2.4 (multi-unit license) Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility.

Explanation: **CORRECT B** With 3ED D/G inoperable, when 'B' SBGT blower fails to start, the redundant component requirement comes into effect. Therefore LCO 3.8.1.B.3 allows 4 hours to declare the 'C' train of SBGT Inoperable and then enter 3.0.3.

A – Incorrect: Two SBGT trains are Inoperable, and requires entry into TS 3.0.3. Plausible because this is the action for one SBGT Inoperable TS 3.6.4.3 A.1.

C – Incorrect. Units 1 & 2 have 4 hours to enter TS 3.0.3. This is plausible if LCO 3.8.1.B.3 completion time (4 hours to declare the 'C' train of SBGT Inoperable) is missed and TS 3.0.3 is entered immediately.

D – Incorrect. Suspension of fuel handling on Unit 3 is not required. This is plausible if on Units 1 and 2 the LCO 3.8.1.B.3 completion time (4 hours to declare the 'C' train of SBGT Inoperable) is missed and TS 3.0.3 is entered immediately.

Technical Reference(s): 0-OI-65, Unit 1/2/3 TS 3.6.4.3

Proposed references to be provided to applicants during examination: Unit 1/2/3 TS 3.6.4.3 & 3.8.1 (NO BASES)

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	X
	New:	

Question History: Previous NRC: BFN 1006 #88

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

10 CFR Part 55 Content: 55.43 b (2) Facility operating limitations in the technical specifications and their bases.

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)

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.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

(continued)

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)

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

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

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

b. Unit 3 diesel generators (DGs) with two divisions of 480 V load shed logic and common accident signal logic OPERABLE; and

c. Unit 1 and 2 DG(s) capable of supplying the Unit 1 and 2 4.16 kV shutdown board(s) required by LCO 3.8.7, "Distribution Systems - Operating."

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Verify power availability from the remaining OPERABLE offsite transmission network. AND	1 hour AND Once per 8 hours thereafter
		(continued)

ACTIONS

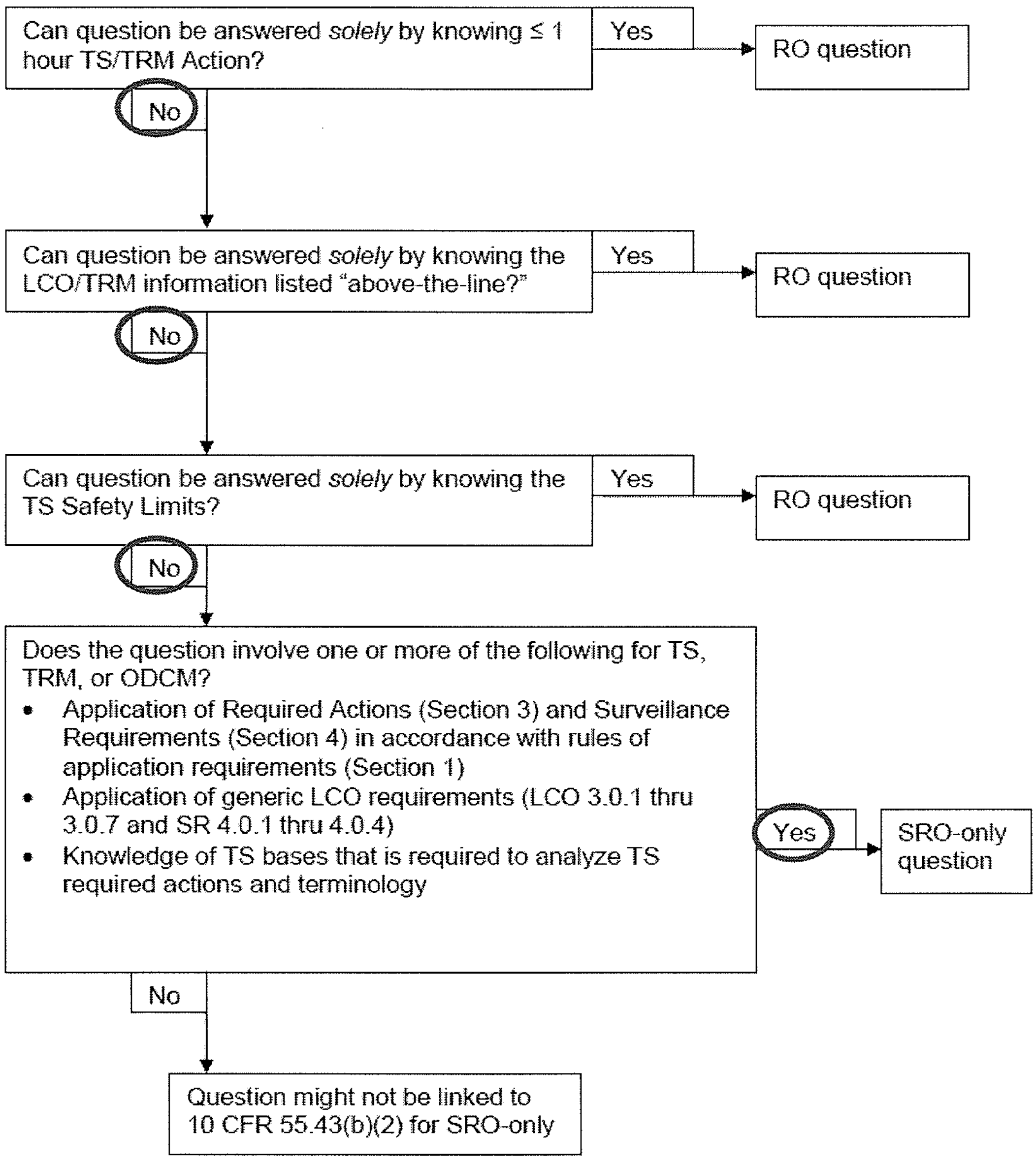
CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one shutdown board concurrent with inoperability of redundant required feature(s)
	AND	
	A.3 Restore required offsite circuit to OPERABLE status.	7 days AND 21 days from discovery of failure to meet LCO
B. One required Unit 3 DG inoperable.	B.1 Verify power availability from the offsite transmission network.	1 hour AND Once per 8 hours thereafter
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Evaluate availability of both temporary diesel generators (TDGs). AND	1 hour AND Once per 12 hours thereafter
	B.3 Declare required feature(s), supported by the inoperable Unit 3 DG, inoperable when the redundant required feature(s) are inoperable. AND	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	B.4.1 Determine OPERABLE Unit 3 DG(s) are not inoperable due to common cause failure. OR	24 hours
	B.4.2 Perform SR 3.8.1.1 for OPERABLE Unit 3 DG(s). AND	24 hours
		(continued)

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



BFN 1006 #88

Examination Outline Cross-reference:
261000 SGTS
G2.2.4 (10CFR 55.43.5 - SRO Only)
(multi-unit license) Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility.

Level	RO	SRO
Tier #	------	2
Group #	------	1
K/A #	261000G2.2.4	
Importance Rating	------	3.6

Proposed Question: # 88

Given the following plant conditions:

- Unit 1 AND Unit 2 are at 100% Reactor Power
- Unit 3 is in Mode 5 with an Operation with Potential to Drain the Vessel (OPDRV) in progress

The Unit 3 Unit Supervisor directs starting of ALL Standby Gas Treatment Subsystems (SGTS).

Which ONE of the following completes the statements?

The PREFERRED location in accordance with 0-OI-65, "Standby Gas Treatments System," is to start SGTS from the __(1)__ Control Room.
SGTS 'A' trips following manual start. In accordance with Tech Spec 3.6.4.3, "Standby Gas Treatment System," __(2)__.

[REFERENCE PROVIDED]

A. (1) Unit 1 OR Unit 2
(2) the OPDRV must be suspended immediately

B. (1) Unit 1 OR Unit 2
(2) SGTS 'A' must be restored to operable within 7 days

C. (1) Unit 3
(2) the OPDRV must be suspended immediately

D. (1) Unit 3
(2) SGTS 'A' must be restored to operable within 7 days

Proposed Answer:
B

Explanation (Optional):

A INCORRECT: Part 1 correct – See Explanation B. Part 2 incorrect – See Explanation C.

B CORRECT: Part 1 correct - Per 0-OI-65, "Standby Gas Treatments System," Precaution and Limitation - Although all three trains of the SGT System can be started from the Unit 3 Control Room, it is recommended that the trains be started from the Units 1 and 2 Control Rooms due to the availability of instrumentation and shutdown capability. Part 2 correct – Although TS 3.6.4.3 is not applicable for current conditions on Unit 3, it is applicable for Units 1 and 2. In accordance with Unit 1 and 2 TS 3.6.4.3 Condition A, SGTS 'A' must be restored within 7 days.

C INCORRECT: Part 1 incorrect – Plausible in that Unit 3 controls are available to start SGTS in the Unit 3 Control Room. Part 2 incorrect – Plausible in that an OPDRV is in progress on Unit 3. However, it is not required immediately.

D INCORRECT: Part 1 incorrect – See Explanation C. Part 2 correct – See Explanation B.

QUESTION 89

Unit 3 is at 100% power and the following condition exists:

- Unit Battery 3 is on a float charge when a loss of ventilation to Battery Room 3 occurs.
- Unit 3 Battery Room Temperature is 79° F and rising.

Which ONE of the following completes both statements below?

The associated battery is __(1)__.

The Unit Supervisor should direct use of temporary ventilation in accordance with __(2)__.

A. (1) OPERABLE
   (2) 0-OI-30F, Common and DG Building Ventilation

B. (1) OPERABLE
   (2) 0-OI-31, Control Bay and Off-Gas Treatment Building Air Conditioning System

C. (1) INOPERABLE BUT Available
   (2) 0-OI-30F, Common and DG Building Ventilation

D. (1) INOPERABLE BUT Available
   (2) 0-OI-31, Control Bay and Off-Gas Treatment Building Air Conditioning System

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	263000 A2.02	
	Importance Rating		2.9

Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of ventilation during charging

Explanation: **B CORRECT:** The batteries are Operable until additional criteria are reached (low volts, Hi temps). 0-OI-31 gives reference to temporary ventilation setup.

A- Incorrect. First Part: Correct. Second Part: Incorrect. Plausible because 0-OI-30F contains the procedure sections for controlling battery Room 3EB ventilation.

C-Incorrect. First Part: Incorrect. The batteries are Operable until additional criteria re reached (low volts, Hi temps). Plausible because it makes sense that a loss of ventilation lineup would require LCO entry. Second Part: Incorrect. Plausible because 0-OI-30F contains the procedure sections for controlling battery Room 3EB ventilation.

D- Incorrect. First Part: Incorrect. The batteries are Operable until additional criteria are reached (low volts, Hi temps). Plausible because it makes sense that a loss of ventilation lineup would require LCO entry. Second Part: Correct.

Technical Reference(s): OPL171.037, 0-OI-31, TRM 3.7.6

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.41 (2) Facility operating limitations in the TS and their bases.

TR 3.7 PLANT SYSTEMS

TR 3.7.6 Electric Board Room Air Conditioning (AC) System

BASES

BACKGROUND

The Unit 3 Electric Board Room AC System provides temperature control for the two Unit 3 Electric Board Rooms on elevations 593' and 621' of the reactor building for both normal operation as well as accidents and plant transients. The Unit 3 Electric Board Room AC System consists of two redundant subsystems that provide cooling of recirculated electric board room air. A subsystem consists of an air handling unit, a condensing unit, ductwork, dampers, piping, and instrumentation and controls to provide for electric board room temperature control. Heat removed by the Electric Board Room AC System is transferred to the environment via the Emergency Equipment Cooling Water System.

A single subsystem provides the required temperature control to maintain the reactor building 593' and 621' elevation electric board rooms temperature within acceptable limits for operation of equipment and for uninterrupted safe occupancy under all plant conditions. The design conditions for the electric board room environment are 104°F and 80% relative humidity. Each subsystem is capable of maintaining the electric board rooms temperature at or below 104°F during abnormal or accident conditions. Alternate methods of cooling the Unit 3 Electric Board Rooms are available. These include, but are not limited to, the use of fans in the doors to the electric board rooms coupled with the Unit 1/2 and Unit 3 Control Room AC Systems and the Relay Room AC Systems. The Electric Board Room AC System operation in maintaining the electric board rooms temperature is discussed in the FSAR, Section 10.12 (Ref.1).

LCO

Two redundant subsystems of the Unit 3 Electric Board Room AC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits unless alternate methods of cooling are implemented. Equipment in the room is declared inoperable whenever the temperature in the room exceeds 104°F.

8.15 Temporary Ventilation for Electrical Equipment Rooms

NOTES

1) This instruction contains the temperature requirements and methods for monitoring and supplying temporary ventilation due to the loss or malfunction of HVAC system to one or more of the following rooms:
   - U1/U2 Main Control Room
   - U3 Main Control Room
   - U1, U2, U3 Auxiliary Instrument Rooms
   - U1, U2, U3 125V/250V Battery and Battery Board Rooms
   - U1, U2, U3 Unit Preferred MG Set Rooms
   - U1, U2, U3 RPS MG Set Rooms
   - U3 SHUTDOWN BOARD ROOMs 3EA, 3EB, 3EC, 3ED
   - U1, U2 250V Shutdown Battery Rooms
   - Relay Room
   - Cable Spreading Room
2) MSI-0-000-PRO005, ELECTRICAL EQUIPMENT ROOM EMERGENCY VENTILATION FOLLOWING AN APPENDIX R FIRE EVENT contains additional guidance for compensatory measures to ensure operability of electrical equipment following an Appendix R fire.
3) Temporary portable fans and generators with necessary equipment for placement are located in the T-Warehouse, Row 6

[1] **IF** ventilation is lost to Battery Room(s), **THEN** (Otherwise NA)

PERFORM (explosive gas) monitoring as follows: ☐

[1.1] **MONITOR** for H_2 buildup daily, **THEN**

RECORD on Illustration 2, Battery Room Atmosphere Readings, (detector(s) can be obtained from Fire Operations Personnel). ☐

7. Battery Room Ventilation Systems

ILT/LOR Obj. 10
NLO/NLOR Obj. 8
FUND

a. Purpose

The various battery room ventilation systems provide adequate room ventilation to prevent an explosive atmosphere due to hydrogen buildup from the batteries.

b. The Unit Battery Rooms 1, 2 and 3 and the Communications Battery Room are supplied air through the door ventilators. Air is exhausted with Battery and Board Room Exhaust Fans 1A and 1B (Battery Room 1 and 2, and communications battery room), and Unit 3 Battery and Board Room Exhaust Fans 3A and 3B (Battery Room 3). Plant/Station Battery Rooms are supplied air via an HVAC unit located outside the rooms to maintain an optimum temperature between 70 and 80 degrees F. A small exhaust fan is located in the ceiling with an off and on switch located on the wall (speed is variable). The purpose of the exhaust fan is to keep hydrogen concentration below 2%. With the exhaust fan off it will take over 8 hours to reach the design limit of 2% hydrogen. Upon loss of the exhaust fan, a "system abnormal" will alarm in the control room. The ceiling also has vent pipes to exhaust the flow of air. Battery Room 4 also required the installation of a new bypass damper with the existing ventilation fan to maintain hydrogen concentration below the design limit.

TR 3.7 PLANT SYSTEMS

TR 3.7.6 Electric Board Room Air Conditioning (AC) System

LCO 3.7.6 Two Unit 3 Electric Board Room AC subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Unit 3 electric board room AC subsystem inoperable.	A.1 Restore Unit 3 electric board room AC subsystem to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two Unit 3 electric board room AC subsystems inoperable.	B.1 Initiate action to restore one Unit 3 electric board room AC subsystem to OPERABLE status.	Immediately
	<u>AND</u>	
	B.2 Place an alternate method of cooling in operation.	12 hours
	<u>AND</u>	
	B.3 Restore one electric board room AC subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Declare the electrical equipment in the electric board room inoperable.	12 hours

(continued)

Nine Mile 1 2008 #90

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	263000 A2.02	
	Importance Rating		2.9

(K&A Statement) Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of ventilation during charging

Proposed Question: SRO 90

The plant is operating at 100% power when Annunciator A3-2-3, LOSS OF VENTILATION BATTERY ROOM 11 OR 12, alarms. The alarm is caused by a failure of the Turbine Building Ventilation to Battery Room 11.

Over the next hour the following events occur:

- Hydrogen levels in Battery Room 11 exceed 2.0%
- Maintenance recommends securing the battery charger for Battery 11

Which one of the following actions is required?

Secure the in-service battery charger (161A or 161B) and...

A. line up MG 167 as an alternate battery charger, TS entry is not required.

B. line up Battery Board 12 to supply Valve Board 11, TS entry is not required.

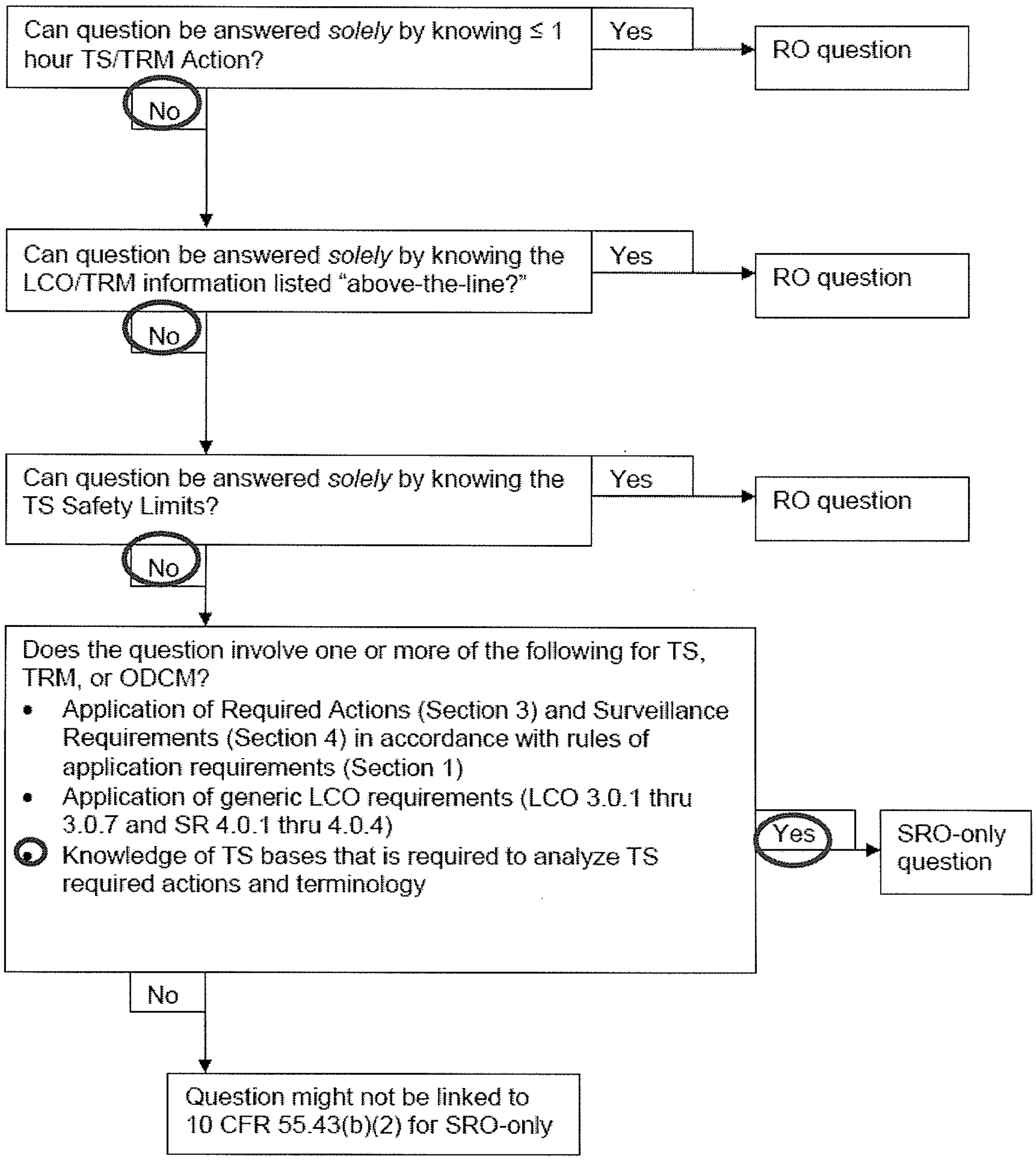
C. return a battery charger to service within 24 hours or then enter TS 3.1.5.b which requires RPV pressure <110 psig within 10 hours.

D. return a battery charger to service within 24 hours or then enter TS 3.0.1 which requires placing the Mode Switch to SHUTDOWN within 1 hour.

Proposed Answer: C.

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2)
(Tech Specs)



QUESTION 90

Unit 1 is Mode 5 with vessel reassembly in progress to the point of installing the Dryer. RHR Pump 1A is operating in Shutdown Cooling.

- RBCCW Pump 1B has tripped
- RBCCW PUMP SUCTION HDR TEMP has increased to 106° F
- Spare RBCCW Pump is UNAVAILABLE

RWCU System AND the Fuel Pool Cooling System have been shutdown as directed by 1-AOI-70-1, Loss of Reactor Building Closed Cooling Water.

NOTE: 1-OI-70, Reactor Building Closed Cooling Water
1-OI-74, Residual Heat Removal System

Which ONE of the following completes the statements below?

The Fuel Pool temperature limit in TRM 3.9.2 is __(1)__.

In accordance with 1-AOI-78-1, Fuel Pool Cleanup System Failure, the Unit Supervisor would direct __(2)__.

A. (1) 125° F
(2) placing EECW in Service to the RBCCW Heat Exchangers, IAW 1-OI-70

B. (1) 125° F
(2) initiation of Supplemental Fuel Pool Cooling with RHR Drain Pump B, IAW 1-OI-74

C. (1) 150° F
(2) placing EECW in Service to the RBCCW Heat Exchangers, IAW 1-OI-70

D. (1) 150° F
(2) initiation of Supplemental Fuel Pool Cooling with RHR Drain Pump B, IAW 1-OI-74

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	400000A2.01	
	Importance Rating		3.4

400000 Component Cooling Water. Ability to (a) predict the impacts of the following on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of CCW Pump

Explanation: **B CORRECT**: Actions to correct for a loss of fuel pool cooling, based on the above conditions are contained within 1-AOI-78-1, Fuel Pool Cleanup System Failure, Subsequent Action [3.7] directs placing RHR in Supplemental Fuel Pool Cooling mode per 1-OI-74, as necessary to maintain Fuel Pool temperature less than 125° F. At this time Shutdown Cooling would not be a viable option to cool the spent fuel pool. This is because the fuel pool gates are installed while the reactor cavity is drained for vessel assembly.

A – Incorrect: First Part: Correct. Second Part: Incorrect. Plausible in that 1-OI-70 will correct fuel pool temperatures for a different type of failure.

C – Incorrect. First Part: Incorrect. Plausible because 150°F is the TRM limit. Second Part: Incorrect. Plausible in that 1-OI-70 will correct fuel pool temperatures for a different type of failure.

D – Incorrect. First Part: Incorrect. Plausible because 150°F is the TRM limit. Second Part: Correct.

Technical Reference(s): 1-AOI-70-1, 1-AOI-78-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.074 V.B.1

Question Source:	Bank: X	
	Modified Bank:	
	New:	
Question History:	Previous NRC: BFN 1006 #90	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis : X	

10 CFR Part 55 Content: 55.43 b(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

4.2 **Subsequent Actions (continued)**

NOTE
Opening RBCCW TCV Bypass valves may cause an EECW pump start due to low pressure.

[9] IF RBCCW PUMP SUCTION HDR TEMP, 1-TIS-70-3 (Panel 1-9-4), cannot be maintained below 105°F, **THEN**

PERFORM the following (otherwise N/A):

- **SLOWLY OPEN** bypass valves around RBCCW TCVs. ☐
- **PLACE** Spare RBCCW Heat Exchanger in service in accordance with 1-OI-70. ☐
- **VERIFY** RWCU System removed from service. ☐

NOTE
It may be necessary to place the RHR System in the Supplemental Fuel Pool Cooling Mode to maintain Fuel Pool temperature below 150°F (Tech. Specs. 3.10.C.2) (TRM 3.9.2).

- **SHUT DOWN** Fuel Pool Cooling System in accordance with 1-OI-78. ☐

[10] IF RBCCW PUMP SUCTION HDR TEMP, 1-TIS-70-3 (Panel 1-9-4), cannot be maintained below 110°F, **THEN**

PERFORM the following (otherwise N/A):

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

REDUCE core flow to between 50-60% (otherwise N/A). ☐

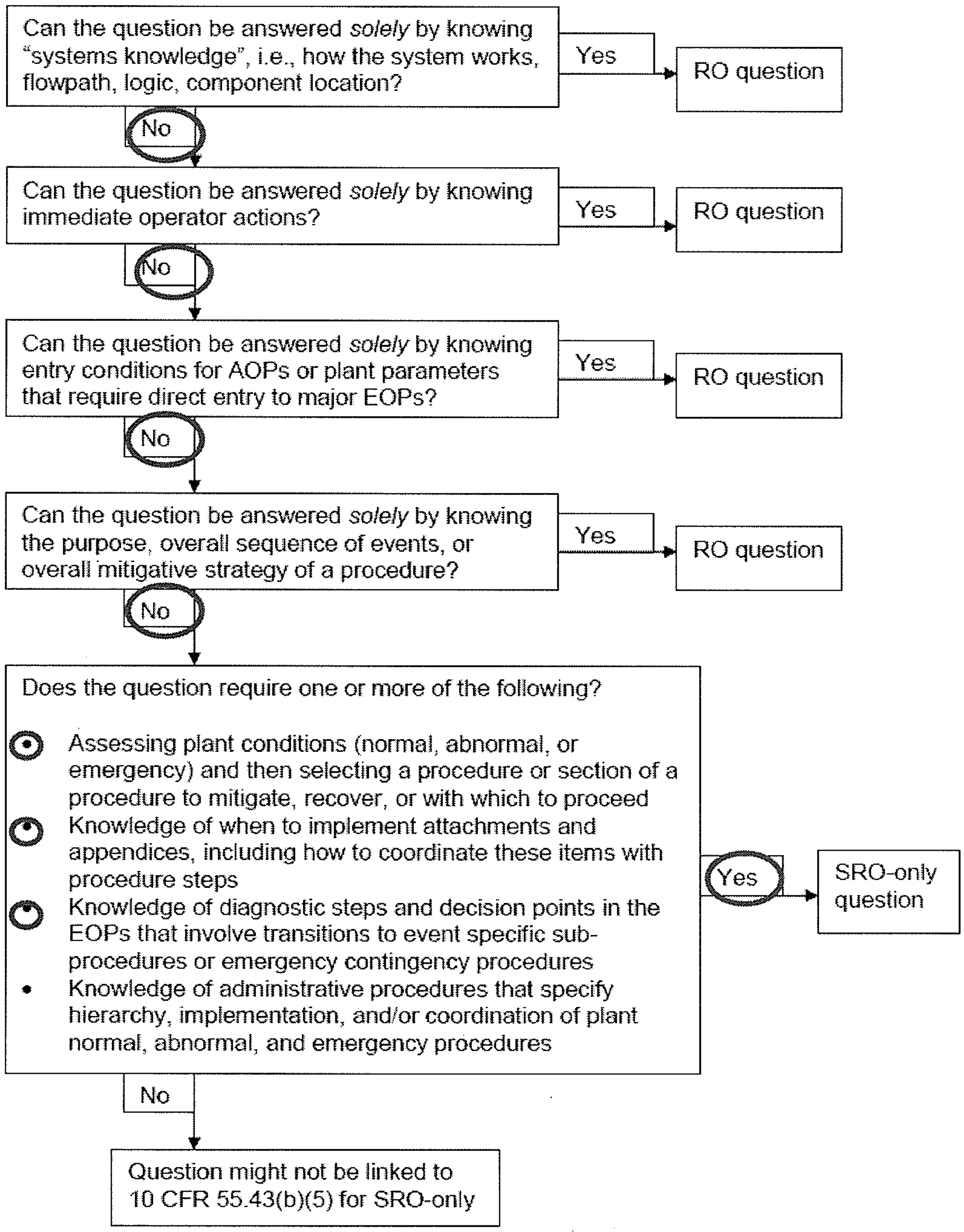
[10.2] **MANUALLY SCRAM** the reactor and **PLACE** Mode Switch to SHUTDOWN. (REFER TO 1-AOI-100-1) ☐

4.2 Subsequent Actions (continued)

[3.6] **ESTIMATE** the time for the Fuel Pool temperature to rise to 125°F, 150°F and 200°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. ☐

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



BFN 1006 #90

Examination Outline Cross-reference:	Level	RO	SRO
400000 Component Cooling Water A2.01 (10CFR 55.43.5 - SRO Only) Ability to (a) predict the impacts of the following on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: • Loss of CCW Pump	Tier #	------	2
	Group #	------	1
	K/A #	400000A2.01	
	Importance Rating	------	3.4

Proposed Question: # 90

Unit 1 is Mode 5 with vessel reassembly in progress to the point of installing the Dryer. RHR Pump 1A is operating in Shutdown Cooling.

- RBCCW Pump 1B has tripped
- RBCCW PUMP SUCTION HDR TEMP has increased to 106°F
- Spare RBCCW Pump is UNAVAILABLE

RWCU System AND the Fuel Pool Cooling System have been shutdown as directed by 1-AOI-70-1, "Loss of Reactor Building Closed Cooling Water."

NOTE: 1-OI-70, Reactor Building Closed Cooling Water
1-OI-74, Residual Heat Removal System

Which ONE of the following completes the statement?

In order to maintain Fuel Pool temperature below the MAXIMUM allowed temperature of __(1)__ , as established in 1-AOI-78-1, "Fuel Pool Cleanup System Failure," the Unit Supervisor is required to enter __(2)__.

A. (1) 125°F
(2) Section 8.14 Initiation of Supplemental Fuel Pool Cooling with RHR Drain Pump B per 1-OI-74

B. (1) 125°F
(2) Section 8.8 Placing EECW in Service to the RBCCW Heat Exchangers per 1-OI-70

C. (1) 150°F
(2) Section 8.14 Initiation of Supplemental Fuel Pool Cooling with RHR Drain Pump B per 1-OI-74

D. (1) 150°F
(2) Section 8.8 Placing EECW in Service to the RBCCW Heat Exchangers per 1-OI-70

Proposed Answer:
A

QUESTION 91

Unit 1 was operating at 75% power when the Reactor Recirc Loop B flow transmitter, 1-FT-68-81C, input to APRM 3 fails downscale.

Which ONE of the following completes both statements below?

The effect on Rod Block Monitor (RBM) A will be a __(1)__.

The Required Action (if any) in accordance with Control Rod Block Instrumentation Technical Specification 3.3.2.1, if any, is __(2)__.

[REFERENCE PROVIDED]

A. (1) Flow Compare (inverse video) alarm ONLY
(2) no action

B. (1) Flow Compare (inverse video) alarm ONLY
(2) restore RBM channel A to OPERABLE status in 24 hours

C. (1) Flow Compare (inverse video) alarm AND a Control Rod Block
(2) no action

D. (1) Flow Compare (inverse video) alarm AND a Control Rod Block
(2) restore RBM channel A to OPERABLE status in 24 hours

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	215002 A2.02	
	Importance Rating		3.3

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: Loss or reduction in recirculation system flow (flow comparator): BWR-3,4,5

Explanation: **A CORRECT**: The only effect on Rod Block Monitor (RBM) A will be a Flow Compare (inverse video) alarm. Since RBM A primary APRM is APRM 1 (which gets its Reactor Recirc flow input from FT-68-81A), the RBM remains unaffected and therefore Operable.

B – Incorrect. First Part: Correct. Second Part: Incorrect. Plausible because this would be the correct required Action per TS 3.3.2.1 if RBM A was Inoperable.

C – Incorrect: First Part: Incorrect. Plausible if the candidate confuses the Reactor Recirc flow inputs to the RBM. Second Part: Correct.

D – Incorrect. : First Part: Incorrect. Plausible if the candidate confuses the Reactor Recirc flow inputs to the RBM. Second Part: Incorrect. Plausible because this would be the correct required Action per TS 3.3.2.1 if RBM A was Inoperable.

Technical Reference(s): OPL171.148, TS 3.3.2.1

Proposed references to be provided to applicants during examination: TS 3.3.2.1

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X	
Question History:	Previous NRC: None	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis : X	
10 CFR Part 55 Content:	55.43 b (2) Facility operating limitations in the technical specifications and their bases.	

INSTRUCTOR NOTES

(4) The STP is also used by the RBM for reference APRM power level.

g. An APRM downscale condition (at≤ 5%) also produces a rod block when the reactor mode switch is in RUN.

Obj. V.B.13.a
Obj V.D.7a

h. Recirculation Flow Monitor

Obj. V.B.10, V.D.6

(1) Each flow monitor channel consists of two flow inputs used to calculate a Total Recirculation Flow, one from Recirculation Loop A and one from Recirculation Loop B.

(2) Each APRM receives the inputs from two (4 to 20 mA) differential pressure (ΔP) transmitters used to measure the recirculation loop flows.

Displays on Input Status display

(3) Each RBM receives the four Total Recirculation Flow values from the APRM channels to determine the status of the flow compare alarm.

(4) The Recirculation Flow Monitor Function provides the following alarm functions for each Total Recirculation Flow level:

- Flow Upscale Alarm (generated by the APRM)
- Flow Compare Alarm (generated by the RBM, display alarm only).

Inputs to flow upscale alarm APRM flow alarm is when flow is >107 % or upscale, generates rod block.
Attention to Detail
Right unit/train/comp

(5) The table below shows the relationship between the flow monitors and each APRM channel.

Discuss effects if Flow transmitters fail

	Loop A	Loop B
APRM 1	FT-68-5A	FT-68-81A
APRM 2	FT-68-5B	FT-68-81B
APRM 3	FT-68-5C	FT-68-81C
APRM 4	FT-68-5D	FT-68-81D

Show drawing
2-47E610-68-1

reactor scram signals

(b)	One transmitter fails	FT-68-82C
	If the B recirc loop signal fails for APRM 3, a flow compare alarm is initiated by the RBM. Since for APRM 3, it is only seeing 1/2 of the total recirc flow signal, this exceeds the flow compare setpoint. Also the indicator on panel 9-4 is lost The APRM channel 3 flow biased alarm is also reduced by this reduction in recirculation flow. Other channels are not affected.	Refer to flow transmitter assignment failure Human Performance - what other instruments are available for use for backup indication?

INSTRUCTOR NOTES

2. General Description

a. RBM consists of two redundant channels for monitoring of reactor power in the immediate vicinity of a control rod selected for movement.

(1) Labeled as 'A' RBM and 'B' RBM.

b. RBM is active above 25% Simulated Thermal Power as determined by the reference APRM for the associated RBM channel. — Obj. V.B.22.d

(1) 'A' RBM receives Simulated Thermal Power (STP) input from APRM #1 with alternate APRM being APRM #3 and second alternate channel being APRM #4. — Obj. V.C.6.b

(a) Alternate APRM is automatically selected when associated primary APRM is bypassed

(2) 'B' RBM receives Simulated Thermal Power (STP) input from APRM #2 with alternate APRM being APRM #4 and second alternate channel being APRM #3. — Obj. V.B.22.b

(a) Alternate APRM is automatically selected when associated primary APRM is bypassed

(3) Both RBM channels are bypassed when APRM reference power is less than 25% STP.

(4) Both RBM channels are also bypassed when a peripheral (edge) rod is selected.

(5) RBM must have an internal control rod selected to be active.

3.3 INSTRUMENTATION

3.3.2.1 Control Rod Block Instrumentation

LCO 3.3.2.1 The control rod block instrumentation for each Function in Table 3.3.2.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2.1-1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One rod block monitor (RBM) channel inoperable.	A.1	Restore RBM channel to OPERABLE status.	24 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two RBM channels inoperable.	B.1	Place one RBM channel in trip.	1 hour
C. Rod worth minimizer (RWM) inoperable during reactor startup.	C.1 <u>OR</u>	Suspend control rod movement except by scram.	Immediately (continued)

(continued) analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, 7, and 12. The standard BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range - Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
b. Intermediate Power Range - Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
c. High Power Range - Upscale	(f),(g)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.8	(e)
d. Inop	(g),(h)	2	SR 3.3.2.1.1	NA
e. Downscale	(g),(h)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	(i)
2. Rod Worth Minimizer	1(c),2(c)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch — Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ≥ 27% and ≤ 62% RTP and MCPR less than the value specified in the COLR.

(b) THERMAL POWER > 62% and ≤ 82% RTP and MCPR less than the value specified in the COLR.

(c) With THERMAL POWER ≤ 10% RTP, except during the reactor shutdown process if the coupling of each withdrawn control rod has been confirmed.

(d) Reactor mode switch in the shutdown position.

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

(f) THERMAL POWER > 82% and < 90% RTP and MCPR less than the value specified in the COLR.

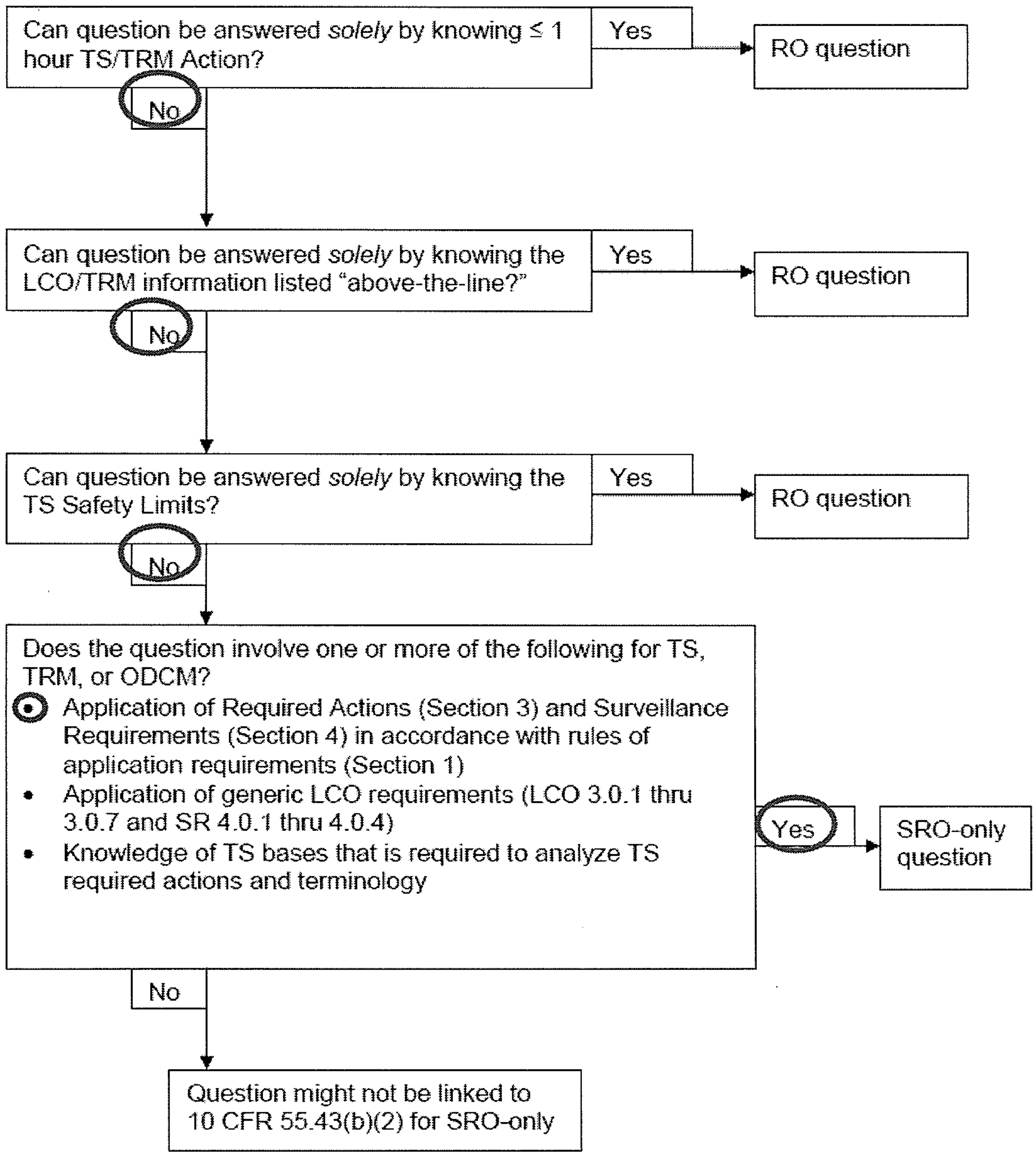
(g) THERMAL POWER ≥ 90% RTP and MCPR less than the value specified in the COLR.

(h) THERMAL POWER ≥ 27% and < 90% RTP and MCPR less than the value specified in the COLR.

(i) Greater than or equal to the Allowable Value specified in the COLR.

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2)
(Tech Specs)



QUESTION 92

Unit 2 Mode Switch is in Startup/Hot Standby, Reactor Power is 8%, and preparations are being made to place the Mode Switch in Run.

Subsequently:

- A failure in the Main Turbine Bypass Valve control causes Reactor Pressure to rise
- Attempts to manually control Main Turbine Bypass Valves are unsuccessful
- The Reactor is manually scrammed

Following the scram:

- All rods are in
- The Mode Switch has been placed in Shutdown
- Reactor Pressure has stabilized at 1060 psig

Which ONE of the following completes both statements below?

For these conditions, the required Technical Specification 3.4.10, Reactor Steam Dome Pressure, actions are __(1)__.

The first Immediate NRC Notification required is a __(2)__ report.

[REFERENCE PROVIDED]

A. (1) no longer applicable
(2) 4-Hour

B. (1) no longer applicable
(2) 8-Hour

C. (1) still required to be completed
(2) 4-Hour

D. (1) still required to be completed
(2) 8-Hour

Correct Answer: **A**

	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	241000 G2.4.30	
	Importance Rating		4.1

241000 Reactor/Turbine Pressure Regulating System:
2.4.30 Knowledge of which events related to system operation/status should be reported to outside agencies.

Explanation: **A CORRECT:** TS 3.4.10 requires Rx Pressure to be less than 1050 psig when Rx is in Mode 1 and 2, TS 3.7.5 requires Turbine Bypass System to be Operable when Rx power is >25%. In the conditions given, Rx is shutdown in Mode 3, no TS Action required. Therefore the Tech Spec actions are no longer applicable. RPS actuation requires a 4-Hour notification when the reactor is critical.

B – Incorrect – First Part: Correct. Second Part: Incorrect, Plausible because RPS actuation requires 8-Hour Notification, unless Rx is critical when actuated, then a 4-Hour notification is required.

C – Incorrect – First Part: Incorrect, Plausible because TS 3.4.10 requires Rx Pressure to be less than 1050 psig when Rx is in Mode 1 and 2, and TS 3.7.5 requires Turbine Bypass System to be Operable when Rx power is >25%. In the conditions given, Rx is shutdown in Mode 3, no TS Action required. Second Part: Correct.

D – Incorrect – First Part: Incorrect, TS 3.4.10 requires Rx Pressure to be less than 1050 psig when Rx is in Mode 1 and 2, TS 3.7.5 requires Turbine Bypass System to be Operable when Rx power is >25%. In the conditions given, Rx is shutdown in Mode 3, no TS Action required. Second Part: Incorrect, Plausible because RPS actuation requires 8-Hour Notification, unless Rx is critical when actuated, then a 4-Hour notification is required.

Technical Reference(s) NPG-SPP-03.5, Tech Spec 3.4.10, Tech Spec 3.7.5

Proposed references to be provided to applicants during examination: NPG-SPP-03.5, Appendix A

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: None
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis : X

10 CFR Part 55 Content: 55.43 (2) Facility operating limitations in the technical specifications and their bases.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1050 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

3.7 PLANT SYSTEMS

3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

Appendix A
(Page 4 of 15)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

d. The actual or attempted introduction of contraband into a protected area, material access area, vital area, or transport.

C. The following criteria require 4-hour notification:

1. §50.72(b)(2)(i) - The initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

2. §50.72(b)(2)(iv)(A) - Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

3. §50.72(b)(2)(iv)(B) - Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

NOTES

1) NPG-SPP-05.14 provides additional instructions regarding addressing and informally communicating events to outside agencies involving radiological spills and leaks.

2) Routine or day-to-day communications between TVA organizations and state agencies typically do not constitute a formal notification to other government agencies that would require a report in accordance with §50.72(b)(2)(xi).

4. §50.72(b)(2)(xi) - Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactive contaminated materials.

D. The following criteria require 8-hour notification:

NOTE
With the exception of "Events or Conditions that Could Have Prevented Fulfillment of a Safety Function," ENS notifications are required for any event that occurred within three years of discovery, even if the event was not ongoing at the time of discovery.

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 Immediate Notification - NRC (continued)

1. §50.72(b)(3)(ii)(A) - Any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded.

2. §50.72(b)(3)(ii)(B) - Any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.

3. §50.72(b)(3)(iv)(A) - Any event or condition that results in valid actuation of any of the systems listed in paragraph (b)(3)(iv)(B) [see list below], except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

   a. The systems to which the requirements of paragraph §50.72(b)(3)(iv)(A) apply are:

NOTE
Actuation of the RPS when the reactor is critical is also reportable under §50.72(b)(2)(iv)(B) above.

(1) Reactor protection system (RPS) including: reactor scram or reactor trip.

(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

Clarification Guidance for SRO-only Questions
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Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2)
(Tech Specs)

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?

Yes → RO question

No ↓

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes → RO question

No ↓

Can question be answered *solely* by knowing the TS Safety Limits?

Yes → RO question

No ↓

Does the question involve one or more of the following for TS, TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- Knowledge of TS bases that is required to analyze TS required actions and terminology

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(2) for SRO-only

QUESTION 93

Unit 2 is operating at 100% power with Offgas System Isolation Valve, 2-FCV-66-28, mechanically restrained open.

SUBSEQUENTLY, a fuel leak results in the following:

- OG POST TRTMT RAD MONITOR HI-HI-HI/INOP (2-9-4C,Window 35) is received
- 2-RM-90-265A, OG POST TRTMT RAD MONITOR, indicates 6.4 x 10^5 cps and rising
- 2-RM-90-266A, OG POST TRTMT RAD MONITOR, indicates 7.2 x 10^5 cps and rising
- Stack Noble Gas (WRGERMS RM-90-306) indicates 2.95 X 10^9 μCi/sec and rising

Which ONE of the following completes both statements below?

The correct actions to mitigate these conditions are to __(1)__.

In accordance with EPIP-1, Emergency Classification Procedure, prior to the emergency declaration based on WRGERMS indication, an assessment of the release rate by another method must be performed, provided the assessment can be accomplished within __(2)__.

[REFERENCE PROVIDED]

A. (1) enter 2-AOI-66-2, Offgas Post Treatment Radiation Hi Hi Hi ONLY
(2) 15 minutes

B. (1) enter 2-AOI-66-2, Offgas Post Treatment Radiation Hi Hi Hi ONLY
(2) 60 minutes

C. (1) enter 2-AOI-66-2, Offgas Post Treatment Radiation Hi Hi Hi AND 0-EOI-4, Radioactivity Release Control
(2) 15 minutes

D. (1) enter 2-AOI-66-2, Offgas Post Treatment Radiation Hi Hi Hi AND 0-EOI-4, Radioactivity Release Control
(2) 60 minutes

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	271000A2.09	
	Importance Rating		2.8

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

Explanation: **CORRECT C** First Part: The Offgas Post Treatment Hi Hi Hi is valid based on the radiation monitor indications. Therefore, the correct actions to mitigate these conditions are to enter 2-AOI-66-2, Off gas Post Treatment Radiation Hi Hi Hi. 2-AOI-66-2 will direct the closure of Offgas System Isolation Valve, 2-FCV-66-28. 0-EOI-4, Radioactivity Release Control is entered at an Emergency Plan declaration of Alert or higher. In this case entry would be required. Second Part: Prior to declaring the ALERT, EPIP-1, Emergency Classification Procedure states that an assessment of the release rate by another method must be performed, provided the assessment can be accomplished within 15 minutes.

A – Incorrect: First Part: Incorrect. 0-EOI-4, Radioactivity Release Control is entered at an Emergency Plan declaration of Alert or higher. In this case entry would be required. Second Part: Correct.

B – Incorrect. First Part: Incorrect. 0-EOI-4, Radioactivity Release Control is entered at an Emergency Plan declaration of Alert or higher. In this case entry would be required. Second Part: incorrect. Plausible because Prior to declaring the NOUE, EPIP-1, Emergency Classification Procedure states that an assessment of the release rate by another method must be performed, provided the assessment can be accomplished within 60 minutes.

D – Incorrect. First Part: Correct. Second Part: incorrect. Plausible because Prior to declaring the NOUE, EPIP-1, Emergency Classification Procedure states that an assessment of the release rate by another method must be performed, provided the assessment can be accomplished within 60 minutes.

Technical Reference(s): EPIP-1, 2-AOI-66-2, 2-ARP-9-4C

Proposed references to be provided to applicants during examination: EPIP-1 Section 4

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X
Question History:	Previous NRC: None
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis : X

10 CFR Part 55 Content: 55.43 b(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

OG POST TRTMT RAD MONITOR HI-HI-HI/INOP 2-RA-90-265C 35

(Page 1 of 1)

Sensor/Trip Point:

2-RM-90-265A	6.2×10^5 cps
2-RM-90-266A	6.2×10^5 cps

Sensor Location: 2-RE-90-265 Panel 2-25-94 Off-Gas Building
2-RE-90-266 Elevation 538.5

Probable Cause:
A. Resin trap failure (RWCU or Condensate demins).
B. Fuel damage.

Automatic Action: OFFGAS SYSTEM ISOLATION VALVE 2-FCV-66-28 closes after a 5 second time delay

Operator Action:

A. **VERIFY** alarm condition on the following ☐
- OFFGAS RADIATION recorder, 2-RR-90-266 on Panel 2-9-2 ☐
- OG POST-TREATMENT CHAN A RAD MON RTMR radiation monitor, 2-RM-90-266A on Panel 2-9-10. ☐
- OG POST-TREATMENT CHAN B RAD MON RTMR radiation monitor, 2-RM-90-265A on Panel 2-9-10. ☐

B. **VERIFY** OFF-GAS SYSTEM ISOLATION VALVE, 2-FCV-66-28 has the Mechanical Restraint **DISENGAGED** and 2-FCV-66-28 is CLOSED. ☐

C. **REFER TO** 2-AOI-66-2. ☐

References:

2-45E620-4	2-45E614-2	2-47E610-90-2	2-115D6410RE-3
GE 2-729E814-6		FSAR Sections 1.6.4.4.6, 7.12.2.2, 7.12.2.3, 7.12.3.3, 9.5.4, and 13.6.2	

4.0 OPERATOR ACTIONS

4.1 Immediate Actions

[1] **IF** scram has not occurred, **THEN**

PERFORM the following:

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

REDUCE core flow to between 50-60%. ☐

[1.2] **MANUALLY SCRAM** the Reactor. (Reference 2-AOI-100-1). ☐

4.2 Subsequent Actions

[1] **IF** OFFGAS SYSTEM ISOLATION VALVE, 2-FCV-066-0028 has been mechanically restrained open due to plant conditions **THEN**

DISENGAGE 2-FCV-066-0028 mechanical restraint by rotating the restraining handwheel fully in the counterclockwise direction, locally at the Stack. (Otherwise N/A) ☐

[2] **VERIFY CLOSED** OFFGAS SYSTEM ISOLATION VALVE, 2-FCV-66-28 on Panel 2-9-53 or locally. ☐

[3] **MONITOR** area radiation levels at Panel: 2-9-11. ☐

[4] **REFER TO** EPIP-1 for emergency classification level and response. ☐

[5] **MONITOR** the following parameters:

A. MAIN STEAM LINE RADIATION, 2-RR-90-135, Panel 2-9-2. ☐

B. OFFGAS RADIATION, 2-RR-90-266, Panel 2-9-2. ☐

C. STACK GAS RADIATION, 0-RR-90-147, Unit 1 Panel 1-9-2. ☐

NOTES

4.1-U Prior to making this emergency classification based upon the WRGERMS indication, assess the release by either of the following:

1. Actual field measurements exceed the limits in table 4.1-U
2. 0-SI 4.8.B.1.a.1 release fraction exceeds 2.0

If neither assessment can be conducted within 60 minutes then the declaration must be made on the valid WRGERMS reading.

4.1-A Prior to making this emergency classification based upon the WRGERMS indication, assess the release by either of the following:

1. Actual field measurements exceed the limits in table 4.1-A
2. 0-SI 4.8.B.1.a.1 release fraction exceeds 200

If neither assessment can be conducted within 15 minutes then the declaration must be made on the valid WRGERMS reading.

4.1-S Prior to making this emergency classification based upon the gaseous release rate indication, assess the release by either of the following methods:

1. Actual field measurements exceed the limits in table 4.1-S.
2. Projected or actual dose assessments exceed 100 mrem TEDE or 500 mrem CDE.

If neither assessment can be conducted within 15 minutes then the declaration must be made based on the valid WRGERMS reading.

4.1-G Prior to making this emergency classification based upon the gaseous release rate indication, assess the release by either of the following methods:

1. Actual field measurements exceed the limits in table 4.1-G.
2. Projected or actual dose assessments exceed 1000 mrem TEDE or 5000 mrem CDE.

If neither assessment can be conducted within 15 minutes then the declaration must be made based on the valid WRGERMS reading.

CURVES/TABLES:

Table 4.1-U
RELEASE LIMITS FOR UNUSUAL EVENT

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	2.88×10^{7} µCi/sec	1 Hour
Gaseous Release Rate	0-SI 4.8.B.1.a.1	Release Fraction 2.0	1 Hour
Site Boundary Radiation Reading	Field Assessment Team	0.10 MREM/HR Gamma	1 Hour

Table 4.1-A
RELEASE LIMITS FOR ALERT

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	2.88×10^{9} µCi/sec	15 Minutes
Gaseous Release Rate	0-SI 4.8.B.1.a.1	Release Fraction 200	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	10 MREM/HR Gamma	15 Minutes

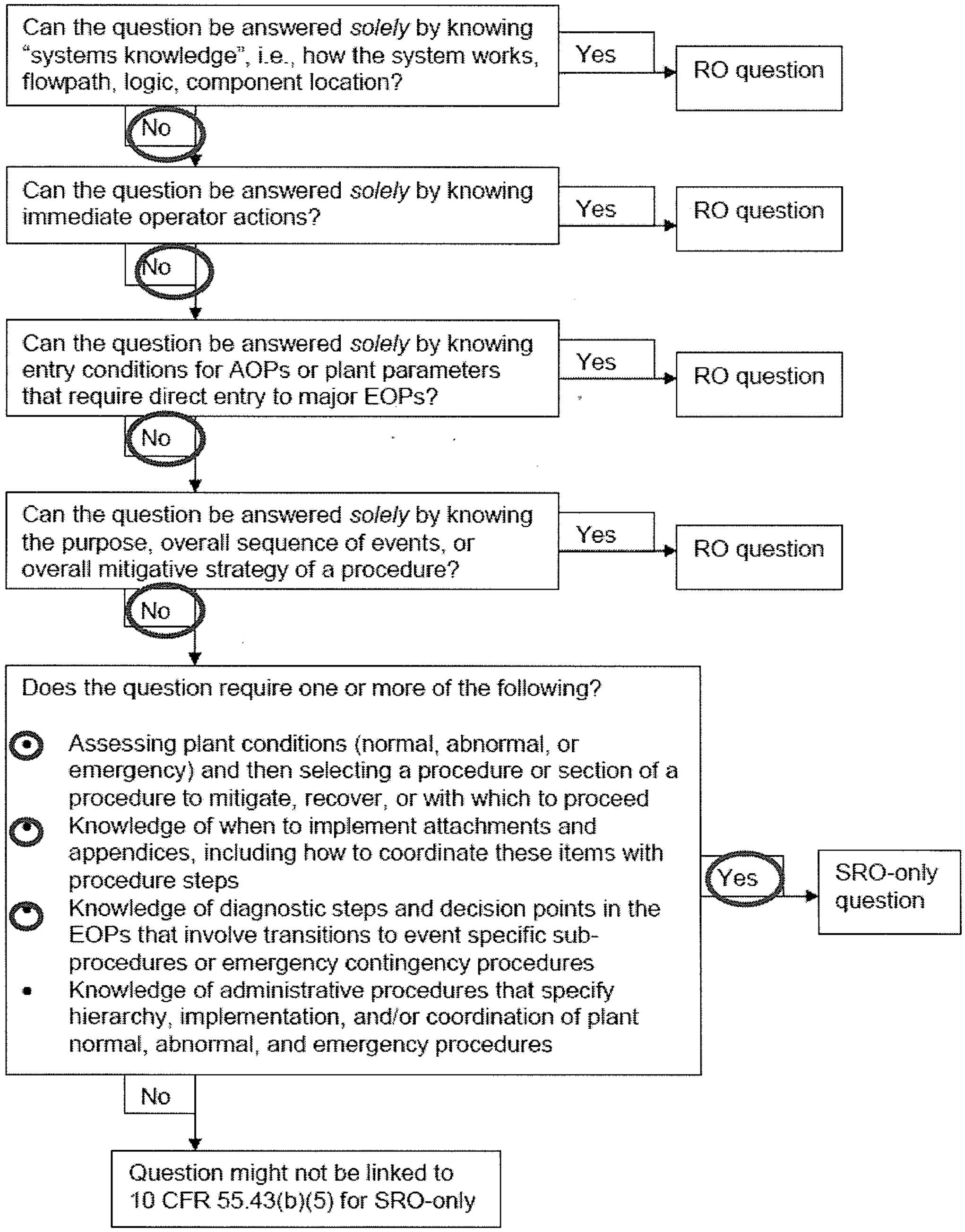
Table 4.1-S
RELEASE LIMITS FOR SITE AREA EMERGENCY

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	5.9×10^{9} µCi/sec	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	100 MREM/HR Gamma	1 Hour
Site Boundary Iodine-131	Field Assessment Team	3.9×10^{-7} µCI /cm^{3}	1 Hour

Table 4.1-G
RELEASE LIMITS FOR GENERAL EMERGENCY

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	5.9×10^{10} µCi/sec	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	1000 MREM/HR Gamma	1 Hour
Site Boundary Iodine-131	Field Assessment Team	3.9×10^{-6} µCI / cm^{3}	1 Hour

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



QUESTION 94

Unit 1 is at 8% power performing 1-GOI-100-1A, Unit Startup, and preparing to transition to MODE 1.

The following alarm and pump seal pressure indications are received:

- RECIRC PUMP 1A NO 1 SEAL LEAKAGE ABN (1-9-4A, Window 25)
- No. 1 Seal Pressure: 980 psig
- No. 2 Seal Pressure: 980 psig

In accordance with Tech Specs, which ONE of the following completes the statements below?

The alarm/indications __(1)__ RCS pressure boundary leakage.

Mode 1 __(2)__ be entered if the 24 hour average TOTAL leakage stabilizes at 31 gpm.

A. (1) represent
   (2) can

B. (1) represent
   (2) CANNOT

C. (1) do NOT represent
   (2) can

D. (1) do NOT represent
   (2) CANNOT

Correct Answer: **D**

	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.1.23	
	Importance Rating		4.4

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Explanation: **D CORRECT:** The alarm/indications given in the stem do NOT represent RCS pressure boundary leakage. LCO 3.0.4: When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made: a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time.

A – Incorrect.- First Part: Incorrect. Plausible because the Recirc Pump Seals do contain the reactor pressure and because the interpretation of what does/does not constitute pressure boundary leakage is located in the Tech Spec Bases. (However, the 1st part of the question is RO knowledge since the RO knows which systems are routed to the equipment drain sump - identified leakage). Second Part: Incorrect. Plausible if the candidate does not interpret LCO 3.0.4, which is not provided as a reference.

B – Incorrect – First Part: Incorrect. Plausible because the Recirc Pump Seals do contain the reactor pressure and because the interpretation of what does/does not constitute pressure boundary leakage is located in the Tech Spec Bases. (However, the 1st part of the question is RO knowledge since the RO knows which systems are routed to the equipment drain sump - identified leakage). Second Part: Correct.

C – Incorrect – First Part: Correct. Second Part: Incorrect. Plausible if the candidate does not interpret LCO 3.0.4, which is not provided as a reference.

Technical Reference(s) 2-GOI-100-1, TS 3.0.4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: BFN 1108 #92
Question Cognitive Level:	Memory or Fundamental Knowledge: Comprehension or Analysis : X

10 CFR Part 55 Content: 55.43 (b)(2) Facility operating limitations in the technical specifications and their bases.

RECIRC PUMP 1A NO 1 SEAL LEAKAGE ABN 1-PA-68-63 25	Sensor/Trip Point:	
	1-PI-068-0063A	≤ 400 psig lowering ≥ 600 psig rising

(Page 1 of 3)

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. ☐
- Plugging of No. 2 RO - No. 2 seal pressure approaches No. 1 seal pressure. ☐
- Failure of No. 1 seal - No. 2 seal pressure is greater than 50% of the pressure of No. 1. ☐
- Failure of No. 2 seal - No. 2 seal pressure is less than 50% of the No. 1 seal. ☐

B. **RECORD** pump seal parameters hourly on Attachment 1, Page 3 of this window response, unless other compensatory methods for recording these parameters is evaluated and approved by Engineering. ☐

C. **IF** single seal failure is indicated, **THEN** **INITIATE** seal replacement as soon as possible. Continued operation is permissible if Drywell Leakrate is with T. S. limits ☐

Continued on Next Page

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE; and

c. ≤ 30 gpm total LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

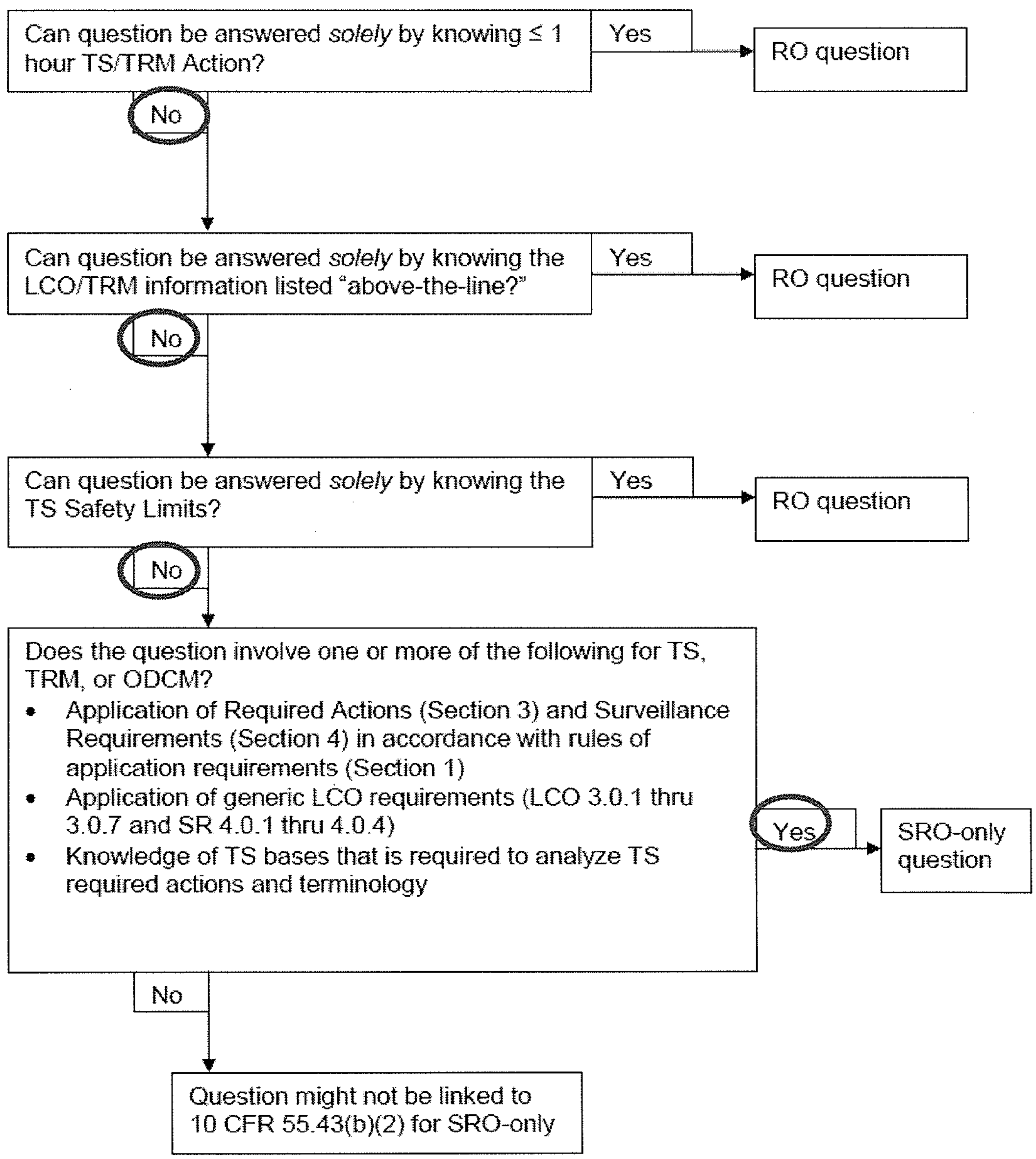
ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Unidentified LEAKAGE not within limit. <u>OR</u> Total LEAKAGE not within limit.	A.1	Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 <u>OR</u>	Reduce LEAKAGE increase to within limits.	4 hours

(continued)

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2)
(Tech Specs)



BFN 1108 NRC #92

92. 202001 G2.2.44 NEW/H

Unit 1 is at 8% power with the Mode switch in the Startup/Hot Standby position.

The following alarm and pump seal pressure indications are received:

- RECIRC PUMP 1A NO 1 SEAL LEAKAGE ABN (1-9-4A, Window 25)
- No. 1 Seal Pressure: 980 psig
- No. 2 Seal Pressure: 980 psig

In accordance with Tech Specs, which ONE of the following completes the statements below?

The alarm/indications __(1)__ RCS pressure boundary leakage.

Mode 1 __(2)__ be entered if the 24 hour average TOTAL leakage stabilizes at 31 gpm.

A. (1) represent
(2) cannot

B. (1) do NOT represent
(2) cannot

C. (1) represent
(2) can

D. (1) do NOT represent
(2) can

Answer: B

QUESTION 95

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One pump inoperable.	A.1	Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

Given the above LCO:

Pump 1 becomes inoperable at 0600 on May 11th. Pump 2 becomes inoperable at 0100 on May 12th.

Which ONE of the following completes the statements below?
(Both pumps are in the same system)

If Pump 1 is restored to OPERABLE at 0800 on May 12th, and if Pump 2 is NOT restored to OPERABLE then Condition B is entered at time __(1)__.

If Pump 2 is restored to OPERABLE at 0800 on May 12th, and if Pump 1 is NOT restored to OPERABLE then Condition B is entered at time __(2)__.

A. (1) 0100 May 19th
(2) 0600 May 18th

B. (1) 0100 May 19th
(2) 0600 May 19th

C. (1) 0100 May 20th
(2) 0600 May 18th

D. (1) 0100 May 20th
(2) 0600 May 19th

Answer: **A**

	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.2.22	
	Importance Rating		4.7

Knowledge of limiting conditions for operations and safety limits.

Explanation: **A CORRECT:** If Pump 1 is restored to OPERABLE at 0800 on May 12th, and if Pump 2 is NOT restored to OPERABLE then Condition B is entered at 0100 May 19th, 7 days for Pump 2 an extension from the original entry of 19 hours. If Pump 2 is restored to OPERABLE at 0800 on May 12th, and if Pump 1 is NOT restored to OPERABLE then Condition B is entered at 0600 May 18th, 7 days for pump 1.

B – Incorrect – First Part: Correct. Second Part: Incorrect. Plausible because the candidate may incorrectly apply the 24 hour extension.

C – Incorrect – First Part: Incorrect. Plausible because the candidate may incorrectly apply the 24 hour extension. Second Part: Correct.

D – Incorrect – First Part: Incorrect. Plausible because the candidate may incorrectly apply the 24 hour extension. Second Part: Incorrect. Plausible because the candidate may incorrectly apply the 24 hour extension.

Technical Reference(s) TS 1.3 Completion Times examples

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.43 (2) Facility operating limitations in the technical specifications and their bases.

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

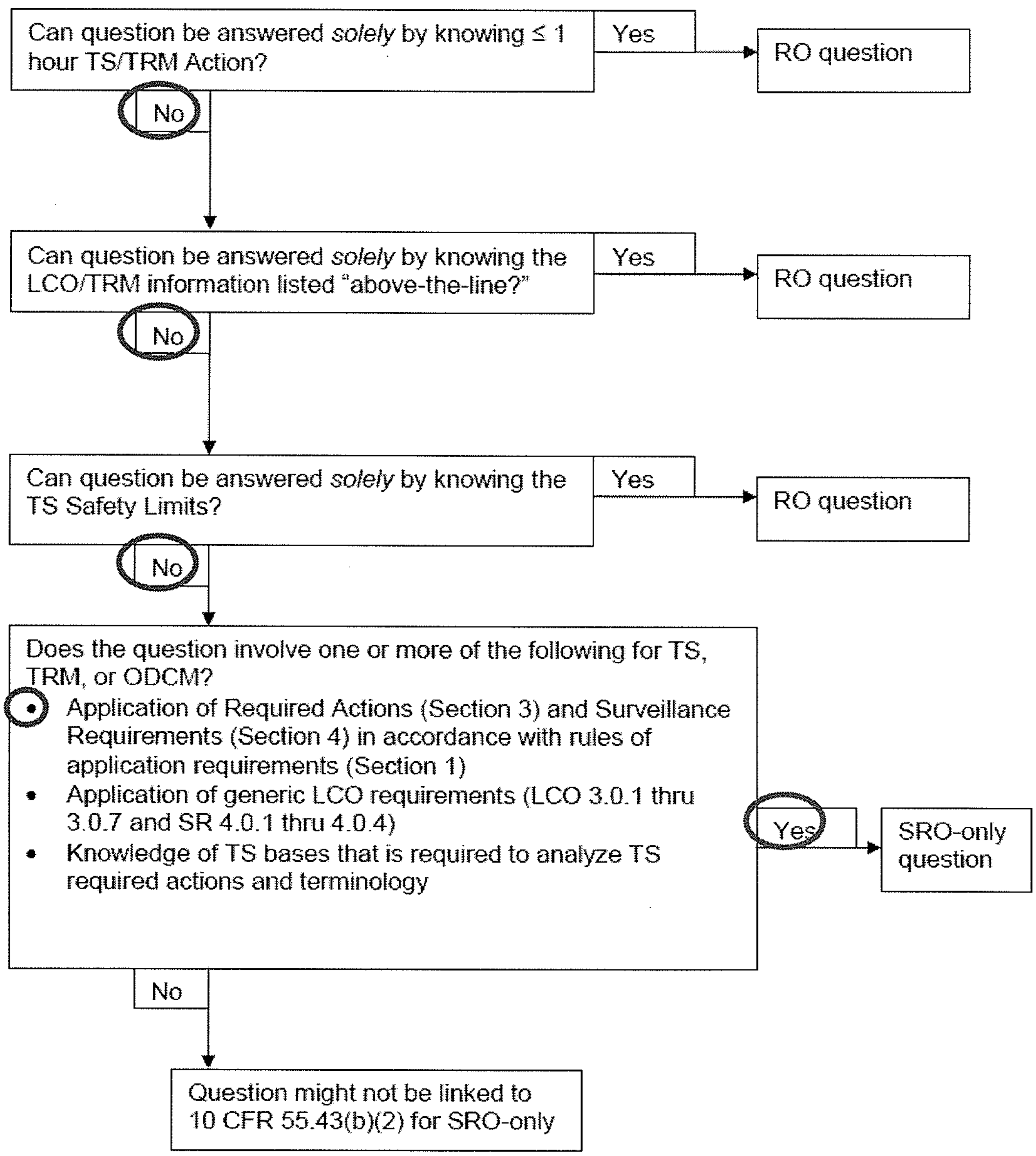
While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(continued)

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



QUESTION 96

Unit 3 is operating at 98% power.

During a control rod surveillance a reactivity event resulted in the following annunciators:

RBM HIGH/INOP (3-9-5, Window 24)
CONTROL ROD WITHDRAWAL BLOCK (3-9-5, Window 7)

The Unit Operator (UO) observes the following values on Powerplex:

MFLCPR 0.925
MAPRAT 0.754
MFDLRX 1.20

Which ONE of the following completes both statements below?

All Tech Spec 3.2, Power Distribution Limits, Limiting Conditions for Operation (LCO) are __(1)__.

The procedure that provides classification criteria for reactivity events is __(2)__.

A. (1) met
(2) OPDP-1, Conduct of Operations

B. (1) met
(2) NPG-SPP 10.4, Reactivity Management Program

C. (1) NOT met
(2) OPDP-1, Conduct of Operations

D. (1) NOT met
(2) NPG-SPP 10.4, Reactivity Management Program

Correct Answer: **D**

	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.2.38	
	Importance Rating		4.5

Knowledge of conditions and limitations in the facility license.

Explanation: **D CORRECT:** All Tech Spec 3.2, Power Distribution Limits, Limiting Conditions for Operation (LCO) are Not met. MFDLRX is greater than 1.0 at 1.20 . The procedure that provides classification criteria for reactivity events is NPG-SPP 10.4, Reactivity Management Program.

A – Incorrect – First Part: Incorrect. Plausible if the applicant does not know that MFDLRX is the same as LHGR. Second Part: Incorrect. Plausible because the severe classification examples (BWR) include rods and Tech Specs.

B – Incorrect- First Part: Incorrect. Plausible if the applicant doesn't understand that MFDLRX, MFLCPR and MAPRAT are ratios of the thermal limit value compared to its maximum, and any value >1 means the thermal limit value has exceeded its maximum. Second Part: Correct.

C– Incorrect – First Part: Correct. Second Part: Incorrect. Plausible because the severe classification examples (BWR) include rods) and Tech Specs.

Technical Reference(s) 3-AOI-85-7, Mispositioned Control Rod, Tech Specs 3.2.3, Linear Heat Generation Rate, NPG-SPP-10.4, Reactivity Management Program

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: BFN 1108 #84

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

10 CFR Part 55 Content: 55.43 (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER $\geq$ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

NPG-SPP 10.4 Reactivity Management

F. Reactivity management issues will be assigned an event category per the following criteria:

There are six different Significance Levels (SLs) for Reactivity Management Issues based on plant impact.

If an issue can be classified at more than one SL, then the highest SL is used. Management discretion can be used to raise the SL of an issue, but not to lower it.

3.2.5 Issue Identification and Trend Analysis (continued)

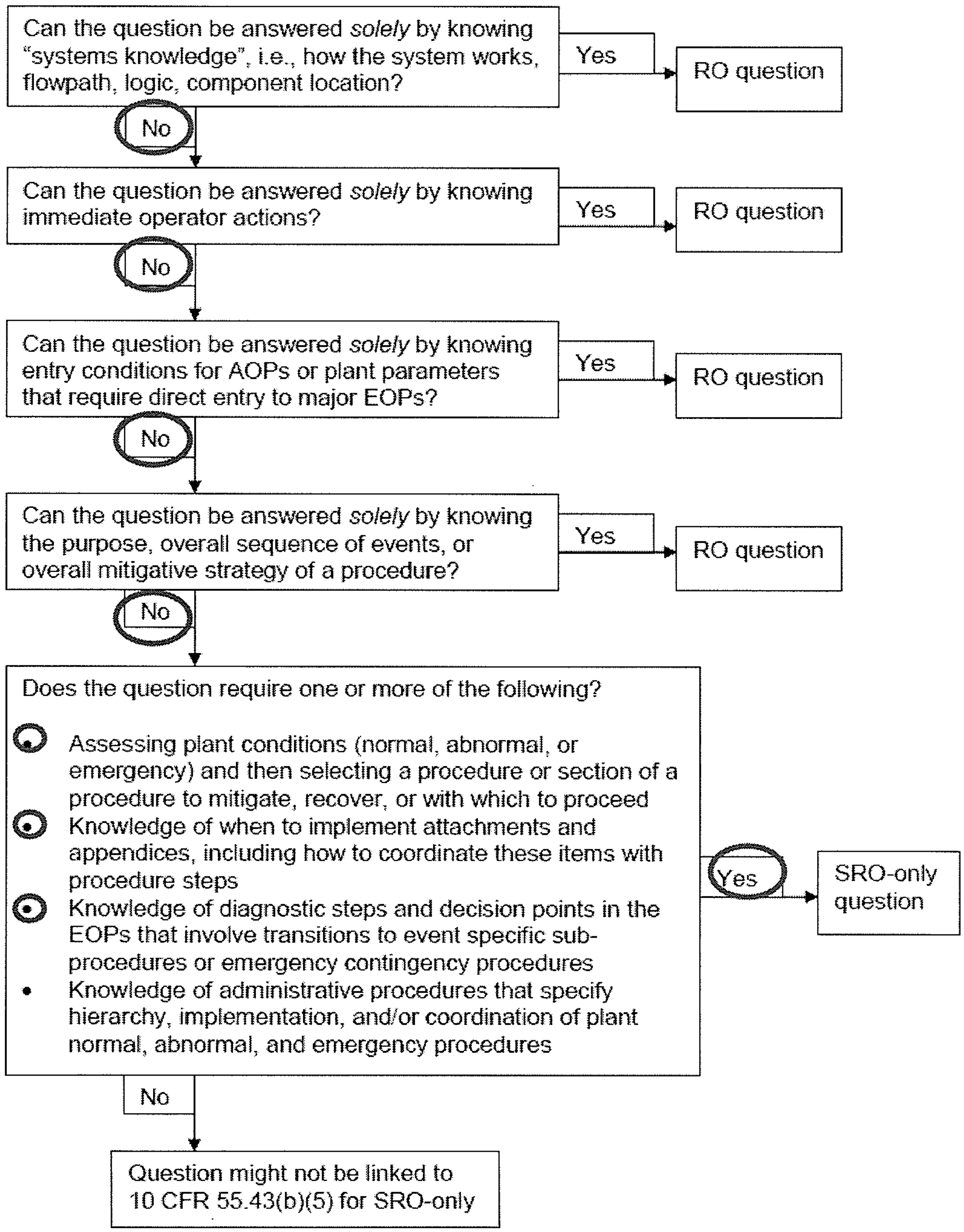
RMRB can raise or lower SL if, in their judgment, it was incorrectly assessed.

SL	Plant Impact	Short Description
1	Highest	Fundamental Organizational Breakdown
2		Violation of Design or Licensing Basis
3		Violation of Process / Procedural Requirements
4		Precursor
5	Lowest	Concern
6	None	No impact; trending purposes only

SL 1 through SL 3 events are lagging indications of performance. SL 4 and SL 5 issues, in general, may be precursors to more significant events and are considered leading indications of performance.

A Reactivity Management Issue that results in a significant plant impact or indicates a high potential for future significant events is classified as a Reactivity Management Event. There are five different classifications for Reactivity Management Events based on severity with the sixth level for issues that do not impact Reactivity Management but are monitored and trended to prevent more significant issues from occurring at a later time. Since almost all Reactivity Management-related activities are protected by at least two barriers, a Reactivity Management Event normally involves the failure of at least two barriers. Barriers include but are not limited to: redundant indications, potential or actual operator actions, procedures, and control systems.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



BFN 1108 #84

84. 295014G2.2.38 001/1/2/SRO/NEW/H/3//MAB

Unit 3 is operating at 98% power.

During a control rod surveillance, the Unit Operator (UO) was required to single notch a control rod from position 14 to 16; however, the UO continuously withdrew the control rod until the following annunciators began alarming:

RBM HIGH/INOP (9-5, W24)
CONTROL ROD WITHDRAWAL BLOCK (9-5, W7)

The control rod's final position is 26 and the Unit Operator (UO) observes the following values on Powerplex:

MFLCPR	0.925
MAPRAT	0.754
MFDLRX	1.20
MFLPD	0.00

Which ONE of the following indicates:

1) whether all Tech Spec 3.2, Power Distribution Limits, Limiting Conditions for Operation (LCO) are met

and

2) the required classification for this event in accordance with NPG-SPP-10.4, Reactivity Management Program?

A. All Tech Spec 3.2 LCOs are met. (no required action statement)
Severe Reactivity Management Event (SL 1)

B. All Tech Spec 3.2 LCOs are NOT met.
Severe Reactivity Management Event (SL 1)

C. All Tech Spec 3.2 LCOs are met. (no required action statement)
Major Reactivity Management Event (SL 2)

D✓ All Tech Spec 3.2 LCOs are NOT met.
Major Reactivity Management Event (SL 2)

QUESTION 97

After a radwaste tank was properly recirculated and sampled,

- The SRO authorized the release in accordance with 0-SI-4.8.A.1-1, Liquid Effluent Permit.

Subsequently,

- During the batch release, the 0-RM-90-130, Radwaste Effluent Radiation Monitor is determined to be inoperable.

Which ONE of the following identifies the required actions in accordance with 0-SI-4.8.A.1-1?

Note: Attachment 1 of 0-SI-4.8.A.1-1 is titled "Bypassing the RM-90-130 Isolation Logic"
Attachment 2 of 0-SI-4.8.A.1-1 is titled "Valve Checklist"

A. The in-progress release may continue provided Attachment 2 is performed and verified.

B. The in-progress release may continue if an independent verification of the release rate calculations and valve alignments are completed.

C. The release would be terminated until Attachment 1 and 2 are performed and the release is re-commenced on the current 0-SI-4.8.A.1-1.

D. The release must be terminated, a new 0-SI-4.8.A.1-1 Liquid Effluent Permit is required to be initiated and approved.

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.3.6	
	Importance Rating		3.8

2.3.6 Ability to approve release permits.

Explanation: **D CORRECT**: If RM-90-130 is declared inoperable during the release, the release must be terminated and a new SI initiated. This is to ensure the requirement for two independent samples, independent verification of the release rate calculations, and independent verification of valve alignment are met.

A – Incorrect. Plausible the RM-90-130 can be inoperable during the release provided certain conditions in the SI are met PRIOR to the release.

B – Incorrect. Plausible the RM-90-130 can be inoperable during the release provided certain conditions in the SI are met PRIOR to the release.

C – Incorrect: Plausible a release may be performed with an inoperable monitor provided attachment 1 and 2 are performed

Technical Reference(s): 0-SI-4.8.A.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: River Bend 2008 #97

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

10 CFR Part 55 Content: 55.43 b (4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

3.0 PRECAUTIONS AND LIMITATIONS

A. A Unit Supervisor must authorize this release and, when required, authorize the bypassing of the 0-RM-90-130 isolation logic.

B. Start first sample counting within 1 hour of sample time.

C. If the Plant Release MODE (i.e., OPEN or HELPER) changes or CCWP(s) are removed from service during the performance of this SI and the dilution flow is decreased, the release must be terminated until a new SI is initiated or until original dilution flow is restored.

D. If the 0-RM-90-130 monitor is declared inoperable during a release, terminate the release and initiate a new SI. This is to ensure the requirement for two independent samples, independent verification of the release rate calculations, and independent verification of valve alignment are met.

3.0 PRECAUTIONS AND LIMITATIONS (continued)

E. If 77-60 is declared inoperable during the release, the release may continue provided the time of inoperability is recorded and flow rate is estimated once every four hours during the release.

F. If 0-RM-90-130 or 77-60 is inoperable per ODCM Requirements, "N/A" may be recorded in the blanks requiring data from these instruments or the instrument readings may be recorded with the understanding the data is taken from ODCM inoperable equipment and is **NOT** intended to meet ODCM requirements for the release.

QUESTION 97 Rev 1

Examination Outline Cross-Reference:	Level	RO☐ SRO☒
	Tier #	3
	Group #	Radiation Control
	K/A #	G2.3.6
	Importance Rating	3.8

Ability to approve release permits.

Proposed Question:

Which of the following is required to discharge an LWS tank to the Mississippi River if RMS-RE107 is INOPERABLE?

A. Two independent samples of the tank are analyzed. One qualified member of the Chemistry staff and one qualified member of the Radwaste staff independently verify the release rate calculations and the discharge valve lineup.

B. A single sample is analyzed by two qualified members of the Chemistry staff independently. Two qualified members of the Radwaste staff independently verify the discharge valve lineup.

C. Two independent samples of the tank are analyzed. Two qualified members of the Chemistry staff independently verify the release rate calculations. Two qualified members of the Radwaste staff independently verify the discharge valve lineup.

D. A single sample of the tank is analyzed. One qualified member of the Chemistry staff verifies the release rate calculation and one qualified member of the Radwaste staff verifies the discharge valve lineup.

Proposed Answer: C.

10CFR55.43 b (4)

D. Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. [10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include:

Process for gaseous/liquid release approvals, i.e., release permits.

Analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.

Analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits.

SRO-only knowledge should not be claimed for questions that can be answered *solely* based on RO knowledge of radiological safety principles; e.g., RWP requirements, stay-time, DAC-hours, etc.

QUESTION 98

Unit 2 is in a refueling outage with the following plant conditions:

- A Refuel Floor overhead crane failure has led to dropping a heavy load into the Unit 2 fuel pool during a core off-load.
- 2-RA-90-1A, Fuel Pool Floor Radiation Monitor, is in alarm AND reading 1000 mr/hr
- NO TVA Emergency Response Facilities are activated

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

A(n) __(1)__ is required to be declared.

The action required to continue assessing plant conditions is to direct the performance of __(2)__.

[REFERENCE PROVIDED]

A. (1) Alert
   (2) CECC EPIP-8, Dose Assessment Staff Activities During Nuclear Plant Radiological Emergencies

B. (1) Alert
   (2) EPIP-13, Dose Assessment

C. (1) Site Area Emergency
   (2) CECC EPIP-8, Dose Assessment Staff Activities During Nuclear Plant Radiological Emergencies

D. (1) Site Area Emergency
   (2) EPIP-13, Dose Assessment

Correct Answer: **B**

	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.3.13	
	Importance Rating		3.8

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Explanation: **B CORRECT:** First Part: the correct classification is an ALERT based on EAL 3.2-A. Second Part: NO TVA EROs are activated, therefore Radcon must do the dose assessment in accordance with EPIP-13.

A – Incorrect – First Part: Correct. Second Part: Plausible in that if ALL required TVA EROs are activated this would be the correct answer.

C – Incorrect – First Part: Incorrect. Plausible in that dose level have reached SAE level in accordance with EAL 3.2-S. However, dose levels must be coupled with an unisolable Primary System leak discharging into Secondary Containment to warrant an SAE. Also, not an applicable operating mode. Second Part: Plausible in that if ALL required TVA EROs are activated this would be the correct answer.

D– Incorrect – –First Part: Incorrect. Plausible in that dose level have reached SAE level in accordance with EAL 3.2-S. However, dose levels must be coupled with an unisolable Primary System leak discharging into Secondary Containment to warrant an SAE. Also, not an applicable operating mode. Second Part: Correct.

Technical Reference(s) EPIP-1, EPIP-3

Proposed references to be provided to applicants during examination: EPIP-1 Section 3

Learning Objective (As available): OPL171.075 V.B.2

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: BFN 1006 #85

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

10 CFR Part 55 Content: 55.43 (4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

NOTES

CURVES/TABLES:

TABLE 3.2
MAXIMUM SAFE OPERATING AREA RADIATION LIMITS

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

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	

SECONDARY CONTAINMENT RADIATION

Description

UNUSUAL EVENT

ALERT

3.2-A				

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

- 1, 2, or 3-RA-90-1A, Fuel Pool Floor Alarm
- 1, 2, or 3-RA-90-250A, Reactor, Turbine, Refuel Exhaust
- 1, 2, or 3-RA-90-142A, Reactor Refuel Exhaust
- 1, 2, or 3-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	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.

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

APPENDIX G
Page 4 of 5
TECHNICAL SUPPORT CENTER
ALERT CLASSIFICATION INSTRUCTION

4.0 Dose Assessment Evaluation

[1] IF emergency circumstances warrant dose assessment, **CONTACT**, TSC Radiation Protection AND **DIRECT** Radiation Protection to implement EPIP-13 "Dose Assessment".

5.0 Notification of the Nuclear Regulatory Commission (NRC)

NOTE
Notification of the NRC is required to be completed as soon as possible not to exceed 60 minutes from classification declaration.

[1] **DIRECT** the TSC NRC Coordinator to implement Appendix D, "Notification of the NRC".

6.0 Maintaining communications with the NRC

NOTE
When the TSC is staffed, the open and continuous line of communications with the NRC is managed by the TSC NRC Coordinator position.

[1] IF **REQUESTED** by the NRC to maintain an open and continuous line of communications, **DIRECT** TSC NRC Coordinator to maintain and or manage an open and continuous line of communications as directed by NRC.

SRO Only – requires analysis of radiation hazards and prescribing a procedure with which to proceed. This question is rated as C/A due to the requirement to assemble, sort, and integrate the parts of the question to predict an outcome. This requires mentally using this knowledge and its meaning to predict the correct outcome.

BFN 1006 #85

Examination Outline Cross-reference:
295033 High Secondary Containment Area Radiation Levels / 9
G2.3.13 (10CFR 55.43.4 - SRO Only)
Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Level	RO	SRO
Tier #	------	1
Group #	------	2
K/A #	295033G2.3.13	
Importance Rating	------	3.8

Proposed Question: # 85

Unit 2 is in a refueling outage with the following plant conditions:

- A Refuel Floor overhead crane failure has led to dropping a loaded Multi Purpose Canister (MPC) into the Unit 2 fuel pool during a core off-load
- RA-90-1A, Fuel Pool Floor Radiation Monitor, is in alarm AND reading 1000 mr/hr
- NO TVA Emergency Response Facilities are activated

Based on the above conditions, which ONE of the following describes the HIGHEST required Emergency Action Level AND the action required to continue assessing plant conditions?

[REFERENCE PROVIDED]

A. Alert, Direct CECC EPIP-8, “Dose Assessment Staff Activities During Nuclear Plant Radiological Emergencies,” be performed.

B. Alert, Direct EPIP-13, “Dose Assessment,” be performed.

C. Site Area Emergency, Direct CECC EPIP-8, “Dose Assessment Staff Activities During Nuclear Plant Radiological Emergencies,” be performed.

D. Site Area Emergency, Direct EPIP-13, “Dose Assessment,” be performed.

QUESTION 99

The Shift Manager / Site Emergency Director (SM/SED) has declared a General Emergency. The Central Emergency Control Center (CECC) is NOT staffed.

Besides classification, which ONE of the following duties can NOT be delegated to another emergency team member by the SM/SED?

A. Making notifications to the state

B. Directing the shutdown of the plant

C. Conducting site accountability actions

D. Determining Protective Action Recommendations

Correct Answer: **D**

	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.4.40	
	Importance Rating		4.5

Knowledge of SRO's responsibilities in emergency plan implementation.

Explanation: **D CORRECT:** Per EPIP-5, General Emergency, The Site Emergency Director must make any required recommendations until the CECC is staffed. This responsibility cannot be delegated until CECC is in operation. Recommendations are required at General Emergency.

A – Incorrect – The Operations Duty Specialist (ODS) should be notified by the SM/SED within five minutes of the event classification. The ODS relays the information to the Emergency Duty Officer (EDO), the State of Alabama, and the CECC Director. The EDO keeps the CECC Director informed of the situation as necessary.

B – Incorrect – If the event is determined to be one of the four emergency classifications, the Shift Manager assumes the responsibility of SED until relieved by the Plant Manager or designee.

C – Incorrect – The Shift Manager or SED shall make the decision to activate the assembly and accountability process. The actions carried out as a result of this decision can be delegated but the decision itself cannot be delegated.

Technical Reference(s) EPIP-5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.075 V.B.4

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: BFN 1006 #99

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

10 CFR Part 55 Content: 55.43 (6) Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

APPENDIX A

Page 1 of 1

GENERAL EMERGENCY INITIAL NOTIFICATION FORM

1. ☐ This is a Drill ☐ This is an Actual Event - Repeat - This is an Actual Event

2. The SED at Browns Ferry has declared a **GENERAL EMERGENCY.**

3. **EAL Designator:** ______________________________(USE ONLY ONE EAL DESIGNATOR)

4. **Radiological Conditions:** (Check one under both Airborne and Liquid column.)

Airborne Releases Offsite	Liquid Releases Offsite
☐ Minor releases within federally approved limits[1]	☐ Minor releases within federally approved limits[1]
☐ Releases above federally approved limits[1]	☐ Releases above federally approved limits[1]
☐ Release information not known	☐ Release information not known
([1]Tech Specs/ODCM)	([1]Tech Specs/ODCM)

5. **Event Declared:** Time: ___________(Central Time) Date: _______________

6. **The Meteorological Conditions are:** (Use 91 meter data from the Met Tower. If data is not available from the MET tower, contact the National Weather Service by dialing 9-1-256-890-8505 or 9-1-205-621-5650. The National Weather Service will provide wind direction and wind speed.)

   Wind Direction is FROM: _____________degrees **(15 min average)** Wind Speed: _____________ m.p.h **(15 min average)**

STEP MUST BE COMPLETED BY THE SITE EMERGENCY DIRECTOR

7. **Provide Protective Action Recommendation utilizing Appendix H: (Check either 1 or 2 or 3)**

☐ Recommendation 1 • EVACUATE LISTED SECTORS (2 mile Radius & 10 miles downwind) • Shelter remainder of 10 mile EPZ. • Consider issuance of POTASSIUM IODIDE in accordance with the State Plan.	RECOMMENDATION-1	WIND FROM DEGREES (Mark wind direction from Step 7)	RECOMMENDATION-2	☐ Recommendation 2 • EVACUATE LISTED SECTORS (2 mile radius & 5 mile downwind) • SHELTER remainder of 10 mile EPZ. • Consider issuance of POTASSIUM IODIDE in accordance with the State Plan.
A2, B2, F2, G2, E5,E10, F5, F10, G5, G10		**From 4° - 40°**		**A2, B2, F2, G2, E5, F5, G5**
A2, B2, F2, G2, F5, F10, G5, G10, H10		From 41°- 73°		A2, B2, F2, G2, F5, G5
A2, B2, F2, G2, G5, G10, H10, I10		**From 74° - 92°**		**A2, B2, F2, G2, G5**
A2, B2, F2, G2, A5, G5, H10, I10, J10, K10		From 93° - 137°		A2, B2, F2, G2, A5, G5
A2, B2, F2, G2, A5, A10, I10, J10, K10		**From 138° - 203°**		**A2, B2, F2, G2, A5**
A2, B2, F2, G2, A5, A10, B5, B10		From 204° - 282°		A2, B2, F2, G2, A5, B5
A2, B2, F2, G2, B5, B10, C10, D10, E5,E10		**From 283°-326°**		**A2, B2, F2, G2, B5, E5**
A2, B2, F2, G2, C10, D10, E5,E10, F5, F10		From 327° - 3°		A2, B2, F2, G2, E5, F5

☐ **Recommendation 3**

- SHELTER all sectors
- CONSIDER issuance of POTASSIUM IODIDE in accordance with the State Plan.

Completed by: ______________________________

Peer Checked by ______________________________

APPENDIX B
Page 2 of 3
Activation of the Emergency Response Organization (ERO)

[3] IF... unable to establish contact with the ODS

THEN... continue to perform Step 1.0 [2] for 5 minutes

[4] IF... unable to establish contact with the ODS after 5 minutes

THEN... **IMPLEMENT** Appendix I, "Activation of the Emergency Paging System" concurrently with this Appendix beginning at Step 2.0.

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

F. Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity. [10 CFR 55.43(b)(6)]

Some examples of SRO exam items for this topic include:

- Evaluating core conditions and emergency classifications based on core conditions.
- Administrative requirements associated with low power physics testing processes.
- Administrative requirements associated with refueling activities, such as approvals required to amend core loading sheets or administrative controls of potential dilution paths and/or activities.
- Administrative controls associated with the installation of neutron sources.
- Knowledge of TS bases for reactivity controls.

BFN 1006 #99

Examination Outline Cross-reference:
G2.4.40 (10CFR 55.43.5 – SRO Only)
Knowledge of SRO's responsibilities in emergency plan implementation.

	RO	SRO
Level	RO	SRO
Tier #	-----	3
Group #	-----	-----
K/A #	G2.4.40	
Importance Rating	-----	4.5

Proposed Question: # 99

The Shift Manager / Site Emergency Director (SM/SED) has declared a General Emergency. The Central Emergency Control Center (CECC) is NOT staffed.

Besides classification, which ONE of the following duties can NOT be delegated to another emergency team member by the SM/SED?

A. Make notifications to the state.

B. Direct the shutdown of the plant.

C. Conduct site accountability actions.

D. Determine Protective Action Recommendations.

Proposed Answer: D

Explanation (Optional):

A INCORRECT: The Operations Duty Specialist (ODS) should be notified by the SM/SED within five minutes of the event classification. The ODS relays the information to the Emergency Duty Officer (EDO), the State of Alabama, and the CECC Director. The EDO keeps the CECC Director informed of the situation as necessary.

B INCORRECT: If the event is determined to be one of the four emergency classifications, the Shift Manager assumes the responsibility of SED until relieved by the Plant Manager or designee.

C INCORRECT: The Shift Manager or SED shall make the decision to activate the assembly and accountability process. The actions carried out as a result of this decision can be delegated but the decision itself cannot be delegated.

D CORRECT: Per EPIP-5, “General Emergency,” The Site Emergency Director must make any required recommendations until the CECC is staffed. This responsibility cannot be delegated until CECC is in operation. Recommendations are required at General Emergency.

QUESTION 100

An ATWS has occurred on Unit 1 with the following conditions:

- An Alert has been declared
- The On-Call SED has assumed the duties of the SED.
- There are no radiological or other hazards in the Reactor Building

Subsequently, it becomes necessary to perform 1-EOI Appendix-1B, Vent and Depressurize the Scram Pilot Air Header.

Which ONE of the following completes both statements below?

The AUO will perform 1-EOI Appendix-1B at the __(1)__.

The AUO required to perform this appendix __(2)__ requested through the TSC.

A. (1) CRD catwalk above Hydraulic Control Units
   (2) is

B. (1) CRD catwalk above Hydraulic Control Units
   (2) is NOT

C. (1) 565' elevation north east at the CRD station
   (2) is

D. (1) 565' elevation north east at the CRD station
   (2) is NOT

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.4.42	
	Importance Rating		3.8

2.4.42 Knowledge of emergency response facilities.

Explanation: **C CORRECT**: First Part: CORRECT- This is the correct location per appendix 1B. Second Part: CORRECT- Once any REP classification has been declared, and the TSC activation is complete, the control room must request an OSC team through the TSC when it becomes necessary to dispatch an operator in to the reactor building. The candidate must know once the On-call SED has assumed responsibilities of the SED that TSC activation is complete.

A – Incorrect: First Part: Incorrect- plausible because this is the location for AUO actions in 1-EOI Appendix 1E for venting CRD over piston area. Second Part: Correct- see C.

B – Incorrect. First Part: Incorrect- See A. Second Part: Incorrect- this is plausible as AUO's are normally directed from the MCR. In addition, the MCR may direct AUO's in to the DG rooms, Electric Board Rooms, and Control Bay as long as there are no radiological or other hazards in those areas

D – Incorrect. First Part: Correct- See C - Second Part: Incorrect- See B.

Technical Reference(s): BFN-ODM-4.2; 1-EOI Appendix-1B; 1-EOI Appendix-1E;EPIP-6

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.206 ILT Obj. 1.a

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: BFN 1108 #75

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

10 CFR Part 55 Content: 55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

1-EOI Appendix-1B Rev 1

BFN UNIT 1	VENTING AND REPRESSURIZING THE SCRAM PILOT AIR HEADER	1-EOI APPENDIX-1B Rev. 1 Page 1 of 3

LOCATION: Unit 1 RB NE, El 565 ft, 1-LPNL-925-0018B

ATTACHMENTS: 1. Tools and Equipment (✓)

1-EOI Appendix-1E Rev 1

BFN UNIT 1	MANUAL INSERTION OF CONTROL RODS BY VENTING THE OVER PISTON AREA	1-EOI APPENDIX-1E Rev. 1 Page 2 of 7

4.a (Continued)

NOTE

A ladder may be required to perform the following steps. Refer to Attachment 1.

b. **UNLOCK** and **CLOSE** 1-SHV-085-615, WITHDRAW RISER SOV. ______

c. **OBTAIN** necessary tools and equipment and **PERFORM** the following from the catwalk area:

1) **CONNECT** quick disconnect end of vent hose to coupling 1-TV-085-623, WITHDRAW RISER VENT BLOCK TEST VLV. ______

BFN-ODM-4.2 Rev 2

1.1 ODM-4.2 DIRECTIVE (continued)

C. Control Room Dispatch of Operators and Area Evacuation Authority following REP Event Classification:

	May Control Room Order Area Evacuation Due To *Radiation Levels?*	May Control Room Order Direct Dispatch Of Operators...	
ANY REP Classification Has Been Declared, AND TSC Activation IS NOT Complete, Or Has NOT Been Ordered	YES	...into ANY Area?	**YES** Control room staff should evaluate all available indications of radiological and other hazard conditions before dispatch.
ANY REP Classification Has Been Declared, AND TSC Activation IS Complete.	**NO** Unless TSC concurs that area evacuation will NOT adversely affect essential OSC team task performance.	...into Control Bay?	**YES** Use operators staged in control bay per this ODM, if possible.
		...into Electric Board Rooms?	**YES** Use operators staged in control bay per this ODM, if possible. If radiological or other hazard conditions exist in or near room, then request OSC team through TSC.
		...into Diesel Generator Rooms?	**YES** Use operators staged in control bay per this ODM, if possible. If radiological or other hazard conditions exist in or near room, including those from *radiological releases outside secondary containment,* then request OSC team through TSC.
		...into ANY Other Radiological Controlled Or Affected Areas (RB, TB, etc.)?	**NO** Request OSC team through TSC.

EPIP-6 Rev 34

1.0 INTRODUCTION

1.1 Purpose

The purpose of this procedure is to describe activation of the Technical Support Center (TSC), define the TSC organization and provide for TSC operations by defining staff responsibilities.

2.0 REFERENCES

2.1 Industry Documents

A. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants"
B. 10 CFR 50.47, Code of Federal Regulations

2.2 Plant Instructions

A. TVA Radiological Emergency Plan
B. Emergency Plan Implementing Procedure (EPIP) - 1, "Emergency Classification Procedure"
C. EPIP - 2, "Notification of Unusual Event"
D. EPIP - 3, "Alert"
E. EPIP - 4, "Site Area Emergency"
F. EPIP - 5, "General Emergency"
G. EPIP - 16, "Termination and Recovery"
H. EPIP-15, "Emergency Exposures"
I. EPIP-11, "Security and Access Control"

3.0 INSTRUCTIONS

3.1 Activation

The TSC is required to be activated at the Alert or higher emergency classification, however, activation can occur at the discretion of the Shift Manager (SM). Once an emergency classification has been declared, the SM becomes the Site Emergency Director (SED). Depending upon the emergency classification declared, steps to activate the TSC are specified in the applicable EPIP for the emergency classification. When the TSC is activated, the on-call SED will obtain a turnover from the SM/SED, ensure that minimum staffing is met for the emergency center, and assume the responsibilities of the SED from the SM/SED. Once the responsibilities of the SM/SED have been assumed by the on-call SED, command and control of the emergency response transfers to the TSC. TSC activation time is defined in the Radiological Emergency Plan.

75. G2.4.35 NEW/H

Unit 1 has experienced an ATWS and RPS cannot immediately be de-energized.

The US has dispatched an AUO to perform 1-EOI Appendix-1B, Vent and Depressurize the Scram Pilot Air Header.

Which ONE of the following completes both statements below?

The AUO will perform 1-EOI Appendix-1B __(1)__.

The AUO will vent the scram header at __(2)__.

A. (1) at the CRD catwalk above Hydraulic Control Units
(2) the pressure switch used for the SCRAM PILOT AIR HEADER PRESS LOW (1-9-5B, Window 28) annunciator.

B. (1) at the 565' elevation north east at the CRD station
(2) one of the 3-way Alternate Rod Insertion (ARI) solenoid valves.

C. (1) at the 565' elevation north east at the CRD station
(2) the pressure switch used for the SCRAM PILOT AIR HEADER PRESS LOW (1-9-5B, Window 28) annunciator.

D. (1) at the CRD catwalk above Hydraulic Control Units
(2) one of the 3-way Alternate Rod Insertion (ARI) solenoid valves.

CORRECT ANSWER C

Tier 3: Generic.
2.4.35. Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. (CFR: 41.10/43.5/45.13) RO IR: 3.8

Plausibility: The 1st part of "A" and "D" is plausible because this is the location for AUO actions in 1-EOI Appendix 1E for venting CRD over piston area. The 2nd part of "B" and "D" is plausible because the ARI valves are an "alternate means" to vent the scram air header.

References
Unit 1 EOI flow chart
1-EOI Appendix 1B
1-EOI Appendix 1D
1-ARP-9-5B
OPL171.206

Lesson Plan Objectives OPL171.206 Objective D.1.b

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)