

Question: RO- 01

Examination Outline Cross-reference: 01

Revision: 1

Date: 04/25/2022

Tier: 1

Group: 1

K/A Number: 295001 AK 3.07

Level of Difficulty: 3

RO Importance Rating: 3.3

K/A Description: Knowledge of the reasons for the following responses or actions as they apply to Partial or Complete Loss of Forced Core Flow Circulation: Recirculation pump discharge/suction valve manipulation.

CGS is operating at 100% power.

RRC-P-1A trips.

The CRS directs closing RRC-V-67A, RRC Pump A Discharge valve, in accordance with ABN-RRC-LOSS.

What is the basis for closing this valve and what other automatic action will occur?

RRC-V-67A is closed to prevent (1). When this valve is closed, breaker (2) will automatically open.

- A. (1) cooldown of the idle RRC loop
(2) RRC-CB-RPT3A
- B. (1) cooldown of the idle RRC loop
(2) RRC-CB-RPT4A
- C. (1) or eliminate reverse flow in the non-operating loop
(2) RRC-CB-RPT3A
- D. (1) or eliminate reverse flow in the non-operating loop
(2) RRC-CB-RPT4A

Answer: C

K/A Match:

Requires student to know the bases for closing the pump discharge valve on a loss of RRC pump.

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Requires knowledge of reasons for ABN response along with an understanding of system response to closing a RRC pump discharge valve.

10 CFR Part 55 Content: 55.41 10 _____

Comments / Reference: ABN-RRC-LOSS	Rev:	Major: 19	Minor: N/A
<p>NOTE: CB-RPT-3A will auto trip when closing RRC-V-67A or RRC-V-23A.</p> <p>NOTE: RRC-V-67A has an alternate operating station at E-MC-7BA (RB 522) to provide control if the valve will not close from the MCR.</p>			
4.2.8	<p><u>IF</u> operation of RRC-V-67A is available in the Main Control Room, <u>THEN PERFORM</u> the following to prevent reverse rotation of RRC-P-1A:</p> <p>a. CLOSE RRC-V-67A (Pump Discharge Valve) (H13-P602). _____</p> <p>b. LOG the time RRC-V-67A was closed in the electronic logging system. _____</p>		
4.2.5.	<p>If the initial RRC Drive Flow is LT 33,000 gpm then it is desirable to raise the flow if possible to improve core stability and prevent cool down of the idle RRC loop. A rise in Drive Flow is only allowed if the current CTP is GT 25% and the current Drive Flow is LT 33,000 gpm and GT 4173 gpm. The Drive Flow is always GT 4173 gpm, even at minimum RRC pump speed. The initial Drive Flow may be LT 33,000 gpm if the plant was at reduced power when the RRC pump tripped off. This step will allow the core flow and the CTP to be raised to a power level less than the CTP prior to the RRC pump trip. If raising the loop flow is allowed then it should not be raised to GT 57.5 Mlb/hr. Raising RRC pump speed above 15 Hz at less than 25% CTP is outside of normal plant startup procedure guidance and is therefore not allowed. {R-6.15}</p>		
4.2.8,9	<p>The directions to manipulate RRC-V-67A are provided to prevent/eliminate reverse flow through RRC Loop A.</p>		

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Comments / Reference: SD000178	Rev:	Major: 19	Minor: 3
COLUMBIA SYSTEMS RRC		March 2021 SD000178, r19 mr3	
5. RRC-CB-RPT3A(B) Circuit Breaker	The RRC-CB-RPT3A(B) breakers are interlocked to enforce the following recirculation system safety features.		LO-11780a LO-11780f RRC-12c RRC-12d RRC-12m RRC-12n
a) Breaker Close Permissives	(1) Pump Suction valve full open (2) Pump Discharge valve full open (3) CB-RRA(B) lockout reset [RRC-RLY-86/RRA(RRB)]		
b) Breaker Trips	(1) Pump Suction valve < 90% open (2) Pump Discharge valve < 90% open		

Question: RO- 02

Examination Outline Cross-reference: 2

Revision: 0 Date: 5/31/22

Tier: 1

Group: 1

K/A Number: 295003.AA2.04

Level of Difficulty: 2

RO Importance Rating: 3.6

K/A Description: Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of AC Power:
System lineups

CGS is in Mode 1.

A Station Blackout (SBO) has occurred.

The CRS enters PPM 5.6.1, SBO/ELAP.

Which of the following is the minimum set of diesel generators that must be unavailable to classify this event as an Extended Loss of AC Power (ELAP)?

- (1) DG1
- (2) DG2
- (3) DG3
- (4) DG4

- A. (1) or (2)
- B. (1) and (2)
- C. (1) and (2) and (3)
- D. (1) and (2) and (3) and (4)

Answer: C

K/A Match:

Requires an understanding of the system lineups required to classify a Loss of AC Power as an ELAP.

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Explanation:

A. Incorrect. Plausible since a loss of (1) or (2) is required for an ELAP. However, both of these DGs and DG3 must be unavailable to consider the event an ELAP.

B. Incorrect. Plausible since a loss of (1) or (2) is required for an ELAP. However, both of these DGs and DG3 must be unavailable to consider the event an ELAP.

C. Correct. In accordance with PPM 5.6.1, entry conditions, DG1, DG2, and DG3 must be unavailable to classify this event as an ELAP.

D. Incorrect. Plausible since (1), (2) and (3) must be unavailable to classify the event as an ELAP. However, DG4 is not required.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.6.1, SBO/ELAP	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 16261 - Given plant conditions and EOP flowcharts, evaluate plant conditions and determine the appropriate actions according to EOP 5.6.1

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires the candidate to know the system lineup required to classify an event as an ELAP.

10 CFR Part 55 Content: 55.41 10

Question: RO- 02

Comments / Reference: PPM 5.6.1	Rev:	Major:031	Minor:000
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Loss of ALL of the following:

- Main generator
- Offsite 230 KV startup power (TR-S)
- Offsite 115 KV backup power (TR-B)
- Diesel Generator 1
- Diesel Generator 2
- Diesel Generator 3 (Required for ELAP)

Question: RO-03

Examination Outline Cross-reference: RO-03

Revision: 0

Date: 5/02/2022

Tier: 1

Group: 1

K/A Number: 295004 AK1.04

Level of Difficulty: 3

RO Importance Rating: 3.9

K/A Description: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Partial or Complete Loss of DC Power: (CFR: 41.8 to 41.10) AK1.04 Battery capacity

CGS is operating at 100% power.

E-C1-1A experiences a failure and cannot be restored.

The crew entered ABN-ELEC-125VDC.

Attempts to start E-C1-1B have been unsuccessful.

At 1100 E-B1-1 voltage is 115VDC and lowering at 1VDC/min.

What is the earliest time that failures of affected remotely operated circuit breakers will occur?
How will these breakers fail?

Circuit breaker failures will begin to be seen at (1).

The affected circuit breakers will (2).

- A. (1) 1105
(2) automatically trip
- B. (1) 1105
(2) be unable to be operated from the main control room
- C. (1) 1110
(2) automatically trip
- D. (1) 1110
(2) be unable to be operated from the main control room

Answer: D

K/A Match:

Requires student to understand the effect that a discharging battery will have on the busses that it supplies.

Question: RO-03

Explanation:

A. Incorrect. (1) Incorrect. Plausible if the student believes that operability is affected at 110VDC. This voltage is plausible since ABN-ELEC-125VDC gives direction for operability determinations at 110VDC
(2) Incorrect. Plausible if the student believes that the DC circuit breakers are deenergize to actuate coils in which case the breakers would trip when DC voltage was lost. Incorrect because ABN-ELEC-125VDC states that the ability to trip or close remotely operated circuit breakers may occur at battery voltages of LT 105 VDC.

B. Incorrect. (1) Incorrect. Plausible if the student believes that operability is affected at 110VDC. This voltage is plausible since ABN-ELEC-125VDC gives direction for operability determinations at 110VDC
(2) Correct. ABN-ELEC-125VDC states that when battery voltage is LT105VDC the inability to remotely operate circuit breakers may occur.

C. Incorrect. (1) This is correct. With battery voltage lowering at 1VDC/min battery voltage will lower to 105VDC in 10 minutes (time 1110). ABN-ELEC-125VDC states that when battery voltage is LT105VDC the inability to remotely operate circuit breakers may occur.
(2) Incorrect. Plausible if the student believes that the DC circuit breakers are deenergize to actuate coils in which case the breakers would trip when DC voltage was lost. Incorrect because ABN-ELEC-125VDC states that the ability to trip or close remotely operated circuit breakers may occur at battery voltages of LT 105 VDC.

D. Correct. (1) This is correct. With battery voltage lowering at 1VDC/min battery voltage will lower to 105VDC in 10 minutes (time 1110). ABN-ELEC-125VDC states that when battery voltage is LT105VDC the inability to remotely operate circuit breakers may occur.
(2) ABN-ELEC-125VDC states that when battery voltage is LT105VDC the inability to remotely operate circuit breakers may occur.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-ELEC-125VDC	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 6839 - Describe the reason for the following step associated with a 125 V DC Distribution System Failure: [ABN-ELEC-125VDC]
 a. Breaker remote operation at less than 105V DC

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

Question History: Last NRC Exam: N/A

Question: RO-03

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires student to calculate the time battery voltage will reach the value where operability is affected.

10 CFR Part 55 Content: 55.41 10

Comments / Reference: ABN-ELEC-125VDC	Rev:	Major: 17	Minor:	N/A
<p><u>CAUTION</u></p> <p>Loss of the ability to trip or close Div. 1 remotely operated circuit breakers may occur at battery voltages of LT 105 VDC. Plant status should be evaluated and appropriate actions taken prior to reaching this voltage.</p>				
4.2.8	<p><u>WHEN</u> the on line charger E-C1-1A or E-C1-1B, or E-B1-1 battery has experienced a failure or shutdown, <u>THEN DEMONSTRATE</u> 125 VDC Div 1 battery E-B1-1 operability by satisfactory completion of one of the following: N/A the other.</p>			
	<ul style="list-style-type: none"> <u>IF</u> 125 VDC Div 1 battery voltage remained GT 110 VDC during the 125 VDC battery or battery charger outage, <u>THEN PERFORM</u> ESP-BAT-W101. 			
	<ul style="list-style-type: none"> <u>IF</u> 125 VDC Div 1 battery voltage decreased to LE 110 VDC during the 125 VDC battery or battery charger outage, <u>THEN PERFORM</u> ESP-B11-Q101. 			

Question: RO-04

Examination Outline Cross-reference: RO-04

Revision: 0

Date: 5/10/2022

Tier: 1

Group: 1

K/A Number: 295005 AA2.09

Level of Difficulty: 2

RO Importance Rating: 2.9

K/A Description: Ability to determine and/or interpret the following as they apply to Main Turbine Generator Trip: AA2.09
Turbine limitations.

CGS is operating at 20%.

A failure occurs in the air removal system that causes backpressure to degrade.

When is the soonest you would expect a trip of the Main Turbine? What action needs to be taken IAW ABN-BACKPRESSURE?

The turbine will trip when condenser backpressure reaches (1).Before this setpoint is reached the crew will (2).

- A. (1) 5.5"Hg
(2) Trip the main turbine ONLY.
- B. (1) 5.5"Hg
(2) Scram the reactor, trip the main turbine, and trip the main generator.
- C. (1) 8.0"Hg
(2) Trip the main turbine ONLY
- D. (1) 8.0"Hg
(2) Scram the reactor, trip the main turbine, and trip the main generator.

Answer: A

K/A Match:

Question requires student to know and apply Main Turbine Generator Trip limits and the actions required when exceeding these limits.

Explanation:

- A. Correct. With Reactor Power at 20% Generator loading is ~200MWe. With loading at 200MWe the main turbine will trip at 5.5"Hg. Since reactor power is LT 25% only a main turbine trip is required per ABN-BACKPRESSURE.
- B. Incorrect. Plausible because: (1) is correct. (2) Incorrect but plausible because this is the action directed by ABN-BACKPRESSURE if reactor power is above 25%.

Question: RO-04

- C. Incorrect. Plausible because: (1) Plausible because 8.0”Hg is the trip setpoint if main turbine loading is GT 835MWe. Incorrect because with reactor power at 20% MT loading is ~200%. (2) is correct.
- D. Incorrect. Plausible because: (1) Plausible because 8.0”Hg is the trip setpoint if main turbine loading is GT 835MWe. Incorrect because with reactor power at 20% MT loading is ~200%. (2) Incorrect but plausible because this is the action directed by ABN-BACKPRESSURE if reactor power is above 25%.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-BACKPRESSURE	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: Reference or N/A

Learning Objective: 6788 – Given a loss of Main Condenser Vacuum, identify those automatic actions that may have occurred. [ABN-BACKPRESSURE]

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Question requires student to determine MT trip setpoints based on plant conditions and determine when a backpressure trip will occur.

10 CFR Part 55 Content: 55.41 10

Comments / Reference: ABN-BACKPRESSURE	Rev:	Major: 8	Minor: N/A
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Question: RO-04

2.0 AUTOMATIC ACTIONS

2.1 Main Turbine trip

- For turbine loads LT 560 MWe: 5.5" Hg back pressure.
- For Turbine loads 560-835 MWe: linear rise from 5.5 – 8.0" Hg back pressure.
- For Turbine loads GE 835 MWe: 8.0" Hg back pressure.

3.1 IF Reactor power is GT 25%
AND a Main Condenser back pressure trip is imminent (setpoints in section 2.0),
THEN **SCRAM the Reactor**
AND **TRIP the Main Turbine**
AND **TRIP the Main Generator.**

4.1 IF a Main Turbine back pressure limit is exceeded (Attachment 7.1),
AND Reactor power is GT 25%,
THEN **VERIFY** the following occur:

4.1.1 Main Turbine trips.

4.1.2 Reactor SCRAM occurs.

4.2 IF a Main Condenser back pressure trip is imminent,
AND Reactor power is LT 25%,
THEN **TRIP the Main Turbine.**

Question: RO-05

Examination Outline Cross-reference: 5

Revision: 0

Date: 5/31/22

Tier: 1

Group: 1

K/A Number: 295006.AK1.02

Level of Difficulty: 4

RO Importance Rating: 3.6

K/A Description: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to SCRAM: Shutdown margin

CGS is in Mode 1.

An event causes the crew to perform a manual reactor scram.

During the scram report, RO1 reports:

- APRM readings are downscale.
- Several control rods are full out.

The CRS enters PPM 5.1.1, RPV Control (Modes 1-3).

The CRS determines that ATWS conditions exist and transitions to PPM 5.1.2, RPV Control – ATWS.

- Reactor power is 3% and stable.

Which of the following actions should be performed?

- (1) Inhibit ADS
- (2) Trip both RRC pumps
- (3) Initiate SLC

A. (1) ONLY

B. (2) ONLY

C. (1) and (3) ONLY

D. (2) and (3) ONLY

Answer: A

K/A Match:

Requires knowledge of the required actions for insufficient shutdown margin on a reactor scram.

Question: RO-05

Explanation:

A. Correct. Step RC-1 of PPM 5.1.2 direct operators to Inhibit ADS regardless of reactor power.

B. Incorrect. Plausible since step Q-5 of PPM 5.1.2 requires tripping both RRC pumps if reactor power is GT 5%. However, the stem of the question places reactor power at 3%. Therefore, tripping RRC pumps is not required.

C. Incorrect. Plausible since (1) is correct and (3) is required if reactor power is GT 5%, neutron flux oscillations are GT 25%, and prior to reaching 207°F in the wetwell. However, these conditions are not met and initiating SLC is not required.

D. Incorrect. Plausible since (2) and (3) are required if reactor power is GT 5%. However, with the conditions given in the question stem, these actions are not required.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.1.1, RPV Control	
PPM 5.1.2, RPV Control - ATWS	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 13568 - Given a copy of PPM 5.1.2 RPV Control - ATWS and an event requiring entry into the EOPs, execute the strategies of this PPM in accordance with procedure use standards and expectations without error.

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

Question History: Last NRC Exam: N/A

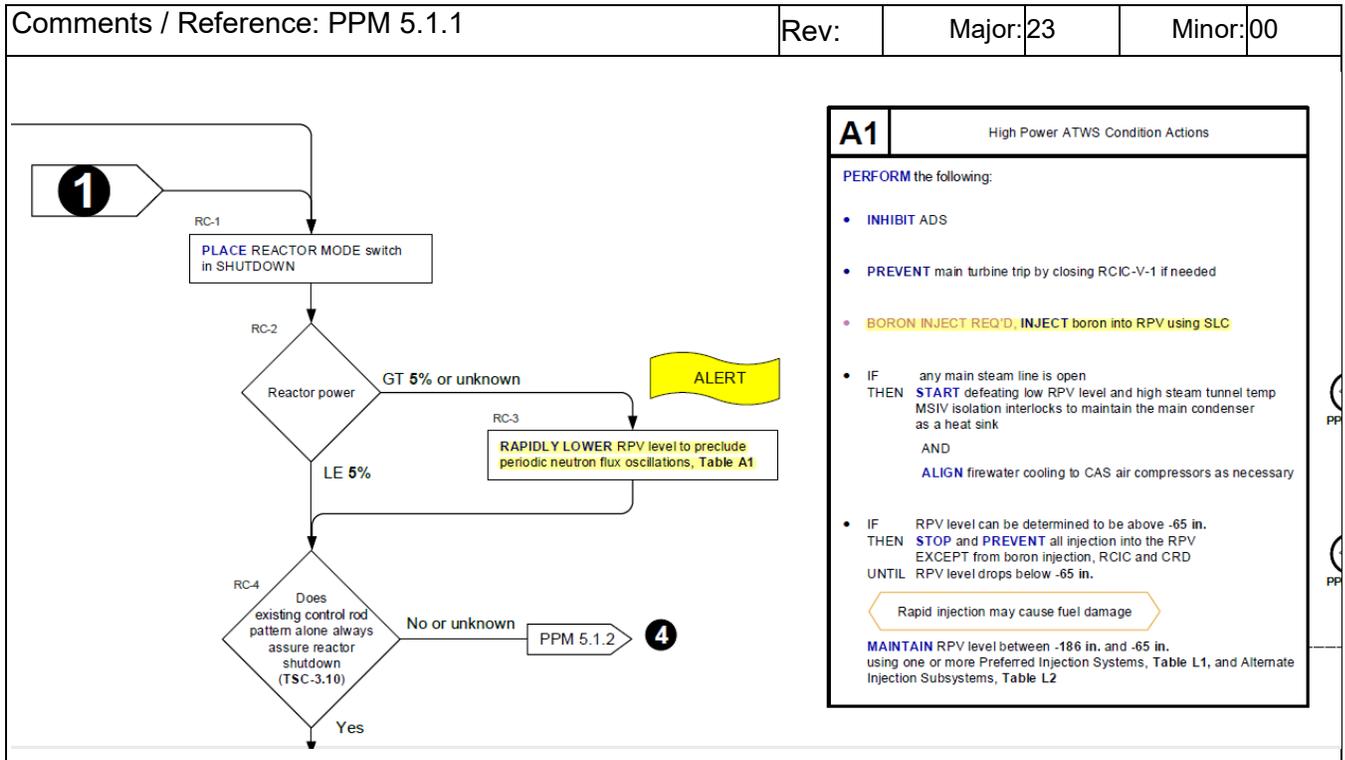
Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize an understanding of the strategy for combating ATWS conditions in PPM 5.1.1 and PPM 5.1.2

10 CFR Part 55 Content: 55.41 10

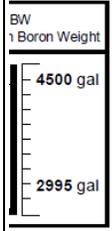
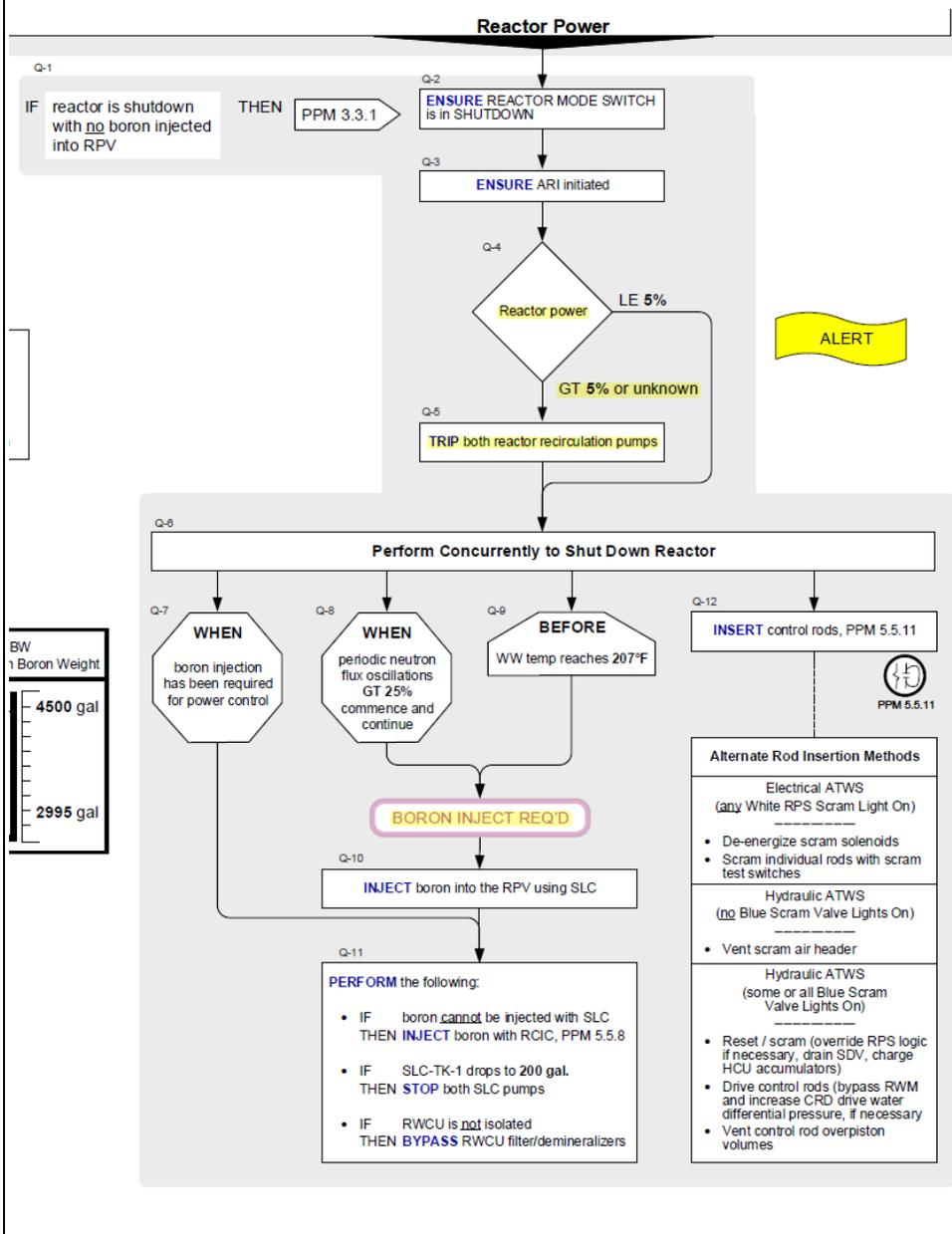
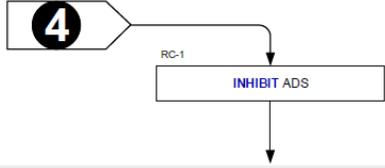
Question: RO-05



Question: RO-05

Comments / Reference: PPM 5.1.2	Rev:	Major: 27	Minor: 00
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- RPV level below +13 in.
- RPV pressure above 1060 psig
- Drywell pressure above 1.68 psig
- Both:
 - a reactor scram is required
 - AND
 - reactor power is above 5% or cannot be determined



ALERT

Examination Outline Cross-reference: 06

Revision: 0

Date: 05/12/22

Tier: 1

Group: 1

K/A Number: 295016 AA1.16

Level of Difficulty: 3

RO Importance Rating: 3.3

K/A Description: Ability to operate and/or monitor the following as they apply to Control Room Abandonment: Cooling water systems

The Main Control Room has been evacuated due to a fire.

- All immediate and subsequent actions of ABN-CR-EVAC have been completed.

You have been directed to start SW-P-1B per attachment 7.8 of ABN-CR-EVAC.

- All prerequisites to start SW-P-1B have been met.
- SW-P-1B is NOT already operating.
- SW-V-2B and SW-V-12B indicate closed.

What is the expected response to starting SW-P-1B per ABN-CR-EVAC Attachment 7.8?

The control switch for SW-P-1B is placed to **START** at the (1).

SW-V-12B starts to automatically open and SW-P-1B automatically starts.

SW-V-2B (2).

- A. (1) Remote Shutdown Panel
(2) automatically opens when pump discharge pressure is GT 50psig
- B. (1) Remote Shutdown Panel
(2) control switch is held in the open position until SW-V-2B indicates full open
- C. (1) Alternate Remote Shutdown Panel
(2) automatically opens when pump discharge pressure is GT 50psig
- D. (1) Alternate Remote Shutdown Panel
(2) control switch is held in the open position until SW-V-2B indicates full open

Answer: B

K/A Match:

Operation of SW from the remote shutdown panel.

Explanation:

- A. Incorrect – Plausible (1) Correct – Operation, monitoring and control of SW-P-1B is performed at the Remote Shutdown Panel. (2) Incorrect SW-V-2B does not automatically open from the RSD panel, plausible if the operator confused operations from the MCR with operations from the RSD panel, In the main control room SW-V-2B automatically open when SW-P-1B is started and there is 50 psig discharge pressure.
- B. Correct - (1) Correct – Correct – Operation, monitoring and control of SW-P-1B is performed at the Remote Shutdown Panel. (2) Correct – SW-V-2B does not maintain its automatic opening feature from the RSD panel, the control switch must be held in the open position until the valve indicates fully opened per ABN-CR-EVAC.
- C. Incorrect – Plausible (1) Incorrect – SW-P-1A Operation, monitoring and control is performed from the Alternate Remote Shutdown Panel .(2) Incorrect SW-V-2B does not automatically open from the RSD panel, plausible if the operator confused operations from the MCR with operations from the RSD panel, In the main control room SW-V-2B automatically open when SW-P-1B is started and there is 50 psig discharge pressure.
- D. Incorrect – Plausible (1) Incorrect – SW-P-1A Operation, monitoring and control is performed from the Alternate Remote Shutdown Panel (2) Correct – SW-V-2B does not maintain its automatic opening feature from the RSD panel, the control switch must be held in the open position until the valve indicates fully opened per ABN-CR-EVAC.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-CR-EVAC	
SD000210 RSD	
SD000204 SSW	
SOP-SW-START	

Proposed references to be provided during examination: N/A

Learning Objective: SW-04 Describe the function, operation, interlocks, trips, physical location, and power supplies of the following Service Water System components.

- b. SW-P-1A(B)
- d. SW-V-2A(B)
- e. SW-V-12A(B)

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

For the given plant conditions the operator must use plant and procedure knowledge to determine the expected plant response for operation of the SW pump and the location that it is operated for ABN-CR-EVAC.

Comments / Reference: ABN-CR-EVAC			Rev:	Major: 45	Minor: N/A
7.8	<u>SW-P-1B Operation</u> (RSD Panel) (CRO1/2/3)				
	NOTE: SW-P-1B required to be started within 10 minutes of DG2 start.				
7.8.1	CHECK the discharge pressure of SW-P-1B on SW-PI-32BR (195-225 psig). _____				
	NOTE: SW-P-1B control switch is spring return to normal.				
★	7.8.2	<u>IF</u> SW-P-1B is not already operating, <u>THEN</u> START SW-P-1B as follows:			
	a.	<u>IF</u> SW-V-2B is open, <u>THEN</u> CLOSE SW-V-2B. _____			
	b.	PLACE the control switch for SW-P-1B to START . _____			
	c.	VERIFY SW-V-12B auto OPENS . _____			
	d.	<u>WHEN</u> SW-V-12B starts to OPEN, <u>THEN</u> VERIFY SW-P-1B STARTS . _____			
	e.	PLACE and HOLD the control switch for SW-V-2B in the OPEN position until SW-V-2B indicates FULL OPEN (after timing sequence). _____			
	f.	VERIFY adequate discharge pressure on SW-PI-32BR (195 – 225 psig). _____			
	g.	VERIFY SW-FIS-15 is indicating GE 1650 gpm. (SW flow in DG-2 room, x2618). _____			
7.13.4	<u>IF</u> SW-P-1A is not operating, <u>THEN</u> START SW-P-1A as follows:				
	a.	<u>IF</u> SW-V-2A is open, <u>THEN</u> CLOSE SW-V-2A. _____			
	b.	PLACE the control switch for SW-P-1A in START . _____			
	c.	PLACE the control switch for SW-V-12A to the OPEN position. _____			
	d.	<u>WHEN</u> SW-V-12A indicates intermediate, <u>THEN</u> VERIFY SW-P-1A STARTS . _____			
	e.	PLACE the control switch for SW-V-2A in the OPEN position. _____			
	f.	VERIFY SW-V-2A FULLY OPENS (after timing sequence). _____			
	g.	VERIFY adequate discharge pressure on SW-PI-32AR. _____			

Comments / Reference: SD000210 RSD			Rev:	Major: 10	Minor: 03
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- 6. SW-P-1A functions:
 - a) To start, SW-V-2A must be closed and SW-V-12A must be off the closed seat.
 - b) Pump will trip when SW-V-12A is closed.
- 7. SW-V-12B will open on SW-P-1B start and close on SW-P-1B stop.
- 3. PUMP B DISCHARGE SW-V-2B
 3-position switch, CLOSE/NEUT/OPEN
 - CLOSE - Valve closes
 - NEUT - Valve stops open travel in first minute of stroke
 - OPEN - Valve opens when held for at least one minute
- 4. SW LOOP B RETURN TO SPRAY POND A SW-V-12B
 3-position switch, CLOSE/NORM/OPEN, returns to NORMAL
 - CLOSE - Valve closes
 - NORMAL - Valve opens on SW-P-1B start, closes on pump stop
 - OPEN - Valve opens

Comments / Reference: SD000204 Standby Service Water	Rev:	Major:20	Minor:01
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A. Control Room Controls

1. SW-P-1A(B), Board A(B)

3-position switch, STOP/NORM/START (spring return to NORM)

- a) STOP Pump stops and close signal goes to Pump Discharge valve SW-V-2A (B).
- b) NORM After a 20 second time delay, if SW-V-2A (B) is closed then SW-V-12A (B) starts to open and then SW-P-1A(B) starts, from the following auto start signals. The 20 second time delay allow ECCS pumps to sequence on to the bus to prevent overloading in the event of an ECCS automatic initiation:
 - 1) SW-P-1A
 - (a) RHR-P-2A started (52a contact)
 - (b) LPCS Pump started (52a contact)
 - (c) DG1 started and no under voltage trip on SM-7.
 - 2) SW-P-1B
 - (a) RHR-P-2B or 2C started (52a contact)
 - (b) DG2 started and no under voltage trip on SM-8
 - (c) RCIC-V-45 fully opens
- c) START if SW-V-2A (B) is closed, SW-V-12A(B) starts to open, then SW-P-1A(B) starts.

2. SW-V-2A (B), Pump Discharge Valves, Board A(B)

3-position switch, CLOSE/AUTO/OPEN (spring return to AUTO, Throttling Valve)

- a) CLOSE Valve closes regardless of auto signals.
- b) AUTO Valve auto opens if SW-P-A(B) has started (52a contact), and discharge pressure exceeds 50 psig. Valve opens for 14 seconds (to approximately 20% open) and stops for 48 seconds, then continues to travel to its full-open position. Valve auto closes when pump is stopped.
- c) OPEN Valve opens with same sequence as AUTO. Control switch must be held in OPEN position until valve indicates fully open (50 psig press. signal is not required).

NOTE: SW-V-2B may also be controlled from the Remote Shutdown Panel and SW-V-2A may also be controlled from the Alternate Remote Shutdown Panel.

3. SW-V-12A (B), Spray Pond Isolation, Board A (B)

3-position switch CLOSE/AUTO/OPEN (spring return to AUTO).

- a) CLOSE Valve closes, when the valves closed limit switch is engaged, the SW pump will trip.
- b) AUTO Valve auto opens if the respective SW pump switch is turned to START, or if an auto start signal occurs. Valve auto closes if respective pump stops.
- c) OPEN Valve opens. The valve will open and remain open even if pump is not running.

Comments / Reference: SOP-SW-START

Rev:

Major:08

Minor:01

5.2 SW Loop B Manual Initiation

NOTE: If Supervisory System Trouble alarm is lit on the B SSW BISI panel, then SW-PI-32B may not be accurate. System pressure may be checked using local gauge SW-PI-32BG.

NOTE: SW-P-1B does not start until SW-V-12B starts to open.

5.2.1 **HOLD** SW-RMS-P/1B in **START** (SW-P-1B control switch). _____

5.2.2 **VERIFY** SW-V-12B **OPENING** (Loop B Return to Pond A). _____

5.2.3 WHEN SW-P-1B starts,
THEN **RELEASE** SW-RMS-P/1B. _____

NOTE: At approximately 50 psig pump discharge, SW-V-2B opens for approximately 14 seconds (approximately 20%), pauses for approximately 48 seconds, and then continues to go full open.

5.2.4 **VERIFY** the following: _____

- SW-V-2B **OPEN**. _____
- SW-V-12B **OPEN**. _____
- RHR-HX-1B SW flow between 6900 to 7600 gpm (H13-P601). {C-9448} _____

Examination Outline Cross-reference: 07

Revision: 0

Date: 04/26/22

Tier: 1

Group: 1

K/A Number: 295018 AK2.03

Level of Difficulty: 3

RO Importance Rating: 3.5

K/A Description: Knowledge of the relationship between Partial or Complete Loss of Component Cooling Water and the following systems or components: (CFR: 41.7 / 45.8) AK2.03 Reactor water cleanup system

CGS is at 100% power.

CRD-P-1B is in service.

RCC-P-1A and RCC-P-1B are in service.

RCC-P-1C is in Pull to Lock and is tagged out for motor replacement.

When:

- SL-71 Experiences an overcurrent trip.
- Reactor Operators have taken their immediate actions.
- CRS has entered ABN-ELEC-SM1/SM7 and ABN-RCC.

What action should be taken?

- A. Start CRD-P-1A per ABN-ELEC-SM1/SM7.
- B. Stop RWCU-P-1A(1B) and throttle open RWCU-V-104.
- C. Scram the Reactor and Stop both Reactor Recirculation Pumps.
- D. Reduce RWCU flow to maintain non regenerative heat exchanger outlet temp LT 130°F.

Answer: B

K/A Match:

Knowledge of the RCC system response to abnormal plant conditions and the effect on the RWCU system.

Explanation:

Stem of the question states that immediate operator actions have been taken (This is per ABN-ELEC-SM3/SM8). This results in RCC-P-1B and RCC-P-1C taken to PTL, With only one RCC pump still in operation as sensed by breaker position (RCC-P-1A), RCC-V-6 RW/RB supply will be closed.

- A. Incorrect – Plausible –Start of CRD-P-1B is required for a loss or momentary loss of SM-7 per ABN-ELEC-SM1/SM7, this is NOT required for a loss of SL-71.
- B. Correct – For the given plant conditions, RCC-V-6 will be closed. Due to RCC-P-1C being tagged out for motor replacement and immediate operator actions, RCC-P-1B will be running and RCC-

P-1A will be in PTL. Stop RWCU-P-1A(B) and throttle open RWCU-V-104 is correct direction from the CRS for the given plant conditions per ABN-RCC step 4.2.3.

- C. Incorrect – Plausible - If the candidate fails to diagnose the current plant conditions (Number of RCC pumps running or the status of RCC-V-6) and believes that there is a complete loss of RCC cooling flow to the drywell regardless of immediate operator actions, Scram of the reactor and stopping of both RRC pumps would be correct per ABN-RCC for a complete loss of RCC flow or a loss of flow to the drywell.
- D. Incorrect – Plausible - If the candidate fails to diagnose the current plant conditions (Number of RCC pumps running or the status of RCC-V-6) and believes that there is still “some” cooling flow to the non regenerative heat exchanger following immediate operator actions, Note that in this case RCC-P-1C CAN NOT be placed in auto after Start due to being tagged out for motor replacement. If RCC-V-6 was still open this would be a correct action directed by the CRS per ABN-RCC.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-RCC	
ABN-ELEC-SM1/SM7	
SD000196 Reactor Closed Cooling Water	
SD000190 Reactor Water Cleanup	

Proposed references to be provided during examination: N/A

Learning Objective: RWCU-09 Describe the systems supporting the Reactor Water Cleanup System, the nature of the support, and the impact a loss (or malfunction) of the system would have on the Reactor Water Cleanup System.
 a. Reactor Closed Cooling Water systems (RCC)

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Must use multiple pieces of system inter-relationship and ABN strategy knowledge to diagnose plant conditions and determine the correct course of action directed by the CRS.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: ABN-RCC	Rev: Major: Maj	Minor: Min
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4.0 SUBSEQUENT OPERATOR ACTIONS

NOTE: Step 4.2 provides direction for a partial loss of RCC cooling capacity.

4.1 IF a complete loss of RCC flow occurs
OR a loss of RCC flow to the Drywell occurs,
THEN **PERFORM** the following:

4.1.1 **SCRAM the Reactor** per PPM 3.3.1. _____

4.1.2 IF NOT in ATWS GT 5% reactor power,
THEN **STOP** both Reactor Recirculation Pumps (RRC-P-1A and RRC-P-1B). _____

4.2 IF a partial loss of RCC cooling capability occurs,
THEN **PERFORM** the following:

NOTE: If RCC flow is lost to the Radwaste Building, OG-RF-20A(B)(C) may trip.

4.2.1 IF an RCC pump has tripped,
THEN **VERIFY** the standby RCC pump has started. _____

4.2.2 IF two RCC pumps are running,
THEN **VERIFY** RCC-V-6 is **OPEN**. _____

CAUTION

Closing RWCU-V-4 without throttling open RWCU-V-104 will result in RWCU-RV-3 lifting, if CRD seal purge is not isolated.

4.2.3 IF RCC-V-6 (RW/RB Supply) is closed,
THEN **PERFORM** the following.

a. **STOP** RWCU-P-1A(1B). _____

b. **THROTTLE OPEN** RWCU-V-104. _____

c. **CLOSE** RWCU-V-4 (RWCU Suction Outboard Isolation). _____

4.2.5 IF RWCU non-regenerative heat exchanger outlet temperature is GT 130°F (RWCU-TI-607, point 3, H13-P602),
THEN **REDUCE** RWCU flow,
OR **SECURE** RWCU per SOP-RWCU-SHUTDOWN. _____

Comments / Reference: ABN-ELEC-SM1/SM7

Rev: Major: 24

Minor: 001

3.0 IMMEDIATE OPERATOR ACTIONS

3.1 IF E-SM-7 is de-energized and NOT restored by DG-1 or TR-B,
AND/OR E-SL-71 is deenergized,
THEN **PERFORM** the following:

3.1.1 IF RCC-P-1A was operating,
THEN **PLACE** RCC-RMS-P/1A in **PTL**. _____

3.1.2 IF RCC-P-1B is not operating,
THEN **PLACE** RCC-RMS-P/1B to **AUTO** after **START**. _____

3.1.3 IF RCC-P-1C is not operating,
THEN **PLACE** RCC-RMS-P/1C to **AUTO** after **START**. _____

4.2 Momentary or Complete Loss of SM-7

4.2.1 IF E-SM-7 has transferred to either E-TR-B or DG-1,
THEN **PERFORM** the following:

CAUTION

Do not run all three Steam Tunnel Supply Fans at the same time in Mode 1, 2, or 3. Running all three fans may prevent the Leak Detection System from performing its design function. All three fans may be operated at the same time in Mode 4 or 5, provided the CRS/Shift Manager approves. Momentary operation of 3 fans is allowed during the process of shifting fans.

- **VERIFY** two steam tunnel fans are operating (N/A fan not running):
 - RRA-FN-21 (E-PP-3DAC from SL-31) _____
 - RRA-FN-9 (E-MC-8B) _____
 - RRA-FN-8 (E-MC-7C) _____

Number: ABN-ELEC-SM1/SM7	Use Category: CONTINUOUS	Major Rev: 024 Minor Rev: 001 Page: 8 of 29
Title: SM-1, SM-7, SM-75, SM-72, SL-71, SL-73 & SL-11 Distribution System Failures		

- IF CRD-P-1A tripped,
THEN **RESTART** a CRD pump per ABN-CRD. _____

Comments / Reference: SD0000196

Rev: Major: 16 Minor: Min

B. Abnormal

1. Loss of RCC (ABN-RCC)

- a) Reactor/Radwaste Building isolation valve RCC-V-6 automatically isolates the RCC cooling supply to all loads except those inside the Primary Containment, during a sustained RCC low-flow condition of GT 10 seconds (LT two RCC pumps running as sensed by breaker position).

IX. POWER SUPPLIES

RCC-P-1A SL-71
RCC-P-1B SL-81
RCC-P-1C SL-81

RADWASTE/RX BLDG SUPPLY RCC-V-6 Three-position switch:
 CLOSE, NOR, OPEN, Spring return to NOR

CLOSE - the valve closes

NOR - the valve closes automatically if less than two pumps are online as sensed by breaker position (plus a 10 second time delay).

OPEN - the valve opens if at least two pumps are online as sensed by breaker position

This valve closes if any of the following actions are done:

- Removing the control power fuses from RCC-P-1A or from valve RCC-V-6 itself
- Closing the RCC-V-6 breaker

Comments / Reference: SD0000190	Rev: Major: 16	Minor: Min
<p>Loss of RCC To RWCU Pump Motor Coolers</p> <p>Because RCC is the primary cooling medium to the RWCU pumps, these pumps should be tripped on a loss of RCC. To avoid thermal shock to the motors and heat exchangers, RCC cooling should be reestablished in a slow and controlled manner.</p> <p>D. <u>Reactor Closed Cooling Water (RCC)</u></p> <p>RWCU-P-1A(B) motors are cooled by RCC. Water is continuously circulated through the pump motor windings, bearing and heat exchangers (HX-3A & 3B) by an auxiliary impeller</p>	<p>RWCU-02a RWCU-12a</p> <p>RWCU-09a</p>	

Examination Outline Cross-reference: 08

Revision: 1 Date: 10/25/22

Tier: 1

Group: 1

K/A Number: 295019.AA1.03

RO Importance Rating: 3.4

Level of Difficulty: 2

K/A Description: Ability to operate and/or monitor the following as they apply to Partial or Complete Loss of Instrument Air:
Backup air supply

CGS is in Mode 1.

A leak on the Containment Instrument Air (CIA) system occurs.

CIA pressure lowers sufficiently to cause the CIA Programmers to place their nitrogen banks in service.

The crew is taking actions in accordance with ABN-CIA.

The CIA leak is stopped and CIA pressure is restored.

The CRS directs operators to restore the CIA system in accordance with SOP-CIA-OPS, Containment Instrument Air System Operation.

How are the CIA Programmers restored to standby?

The CIA Programmers are restored by...

- A. manually rotating the programmers to step 1.
- B. manually rotating the programmers to step 20.
- C. verifying the programmers automatically rotate to step 1.
- D. verifying the programmers automatically rotate to step 20.

Answer: B

K/A Match:

Requires knowledge of the procedure to reset CIA programmers following actuation on a loss of CIA.

Explanation:

- A. Incorrect. Plausible since the programmers must be manually aligned. However, they are aligned to step 20.
- B. Correct. In accordance with the CIA System Description, the standby position of the CIA programmers is step 20. When actuated, the programmers automatically rotate to step 1 to place the first nitrogen bottle in the bank in service. The programmer will continue to automatically rotate to place additional bottle in service as required. SOP-CIA-OPS, step 5.6.3 directs operators to reset the programmers by manually rotating them to step 20.
- C. Incorrect. Plausible since the CIA programmers automatically rotate to step 1 when actuated. However, to reset the programmers to standby, they must be manually rotated to step 20.
- D. Incorrect. Plausible since step 20 is the standby position of the CIA programmers. However, the programmers must be manually rotated to place them in standby.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-CIA, Containment Instrument Air System Failure	
SOP-CIA-OPS, Containment Air System Operation	
SD000156, CIA System Description	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 15736 - Given plant annunciation and indications, describe all contingencies associated with the subsequent operator actions of ABN-CIA.

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Candidate must know the proper reset position of the CIA programmers in accordance with the direction of ABN-CIA.

10 CFR Part 55 Content: 55.41 10

Comments / Reference: ABN-CIA	Rev:	Major:008	Minor:N/A
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4.10 WHEN main header pressure is restored to normal (180 – 190 psig),
THEN **ENSURE** CIA-V-39A(B) opens.
AND **RESTORE** the system per SOP-CIA-OPS.

Comments / Reference: SOP-CIA-OPS	Rev:	Major:006	Minor:N/A
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5.6 Restoring CIA Programmers to Service

5.6.1 **VERIFY** nitrogen pressure restored. _____

5.6.2 IF restoring CIA-PROG-1A to service
THEN VERIFY CIA-PROG-1A at Position 20. (IR-67) _____

5.6.3 IF RE-ENERGIZING CIA-PROG-1A
THEN PERFORM one of the following
(N/A the step that was not performed.)

a. **RESTORE** power to CIA-PROG-1A by installing fuse F-11 in TB-IR-67-2
(Auxiliary box) (Next to IR-67). _____

Simultaneous Verification _____

b. **CLOSE** the following breakers:

• E-DISC-PP7AE/39 (CIA-SPV-1A thru 9A) (R 471 SW) _____

• E-DISC-PP7AE/37 (CIA-SPV-10A thru 15A) (R 471 SW) _____

▲ c. IF CIA-PROG-1A moves from Position 20,
THEN MANUALLY ROTATE programmer back to Position 20.
{P-133881} _____

Comments / Reference: SD000156	Rev:	Major: 11	Minor: 2
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H. Programmers

The programmers (located on the Reactor Building 548' elevation) control the nitrogen bottle banks. CIA-PROG-1A controls a bank of 15 nitrogen bottles; CIA-PROG-1B controls a bank of 19 nitrogen bottles. On an initiation signal the programmers place their respective banks in service.

LO-11755f
NLO-12373c

The normal position for the programmers is on Step 20 (all bottle valves closed). The step programmer function is initiated if at least 2 out of 3 of the following conditions occur:

LO-5150

1. Pressure at CIA-PS-21A (B) less than 160 psig
2. Pressure at CIA-PS-22A (B) less than 156 psig
3. CIA-V-39A(B) indicates closed

When an initiation signal occurs the programmer functions as follows:

1. The programmer advances to step 1, opening first bank bottle isolation.
2. If the initiation signal clears within 30 seconds (pressure restored), the system stays in this alignment until the first bottle is depleted. When the first bottle is depleted and causes another initiation signal or if the initiation signal never cleared within 30 seconds, the programmer advances to step 2 placing the second bottle in-service in parallel with the first. This sequence repeats for the rest of the bottles in the bank until all bottles are in service in parallel.
3. When all bottles have been placed in service and an initiation signal exists, CIA-PROG-1B advances to step 20, all its solenoid valves close. Since CIA-PROG-1A controls only 15 bottles, when it reaches step 16 the programmer automatically advances to step 20, closing the solenoid valves. The programmers will then begin through the cycle again as long as an initiation signal exists.

LO-5151

LO-5152

The programmers are reset after normal system lineup and pressure is restored by manually rotating the timer to position 20, all valves closed. Then the bottle isolation valve indication is reset at the local panels.

Examination Outline Cross-reference: RO-09

Revision: 1

Date: 11/03/22

Tier: 1

Group: 1

K/A Number: 295021 2.4.14

Level of Difficulty: 2

RO Importance Rating: 3.8

K/A Description: Loss of Shutdown Cooling. Knowledge of general guidelines for emergency and abnormal operating procedures.

The following plant conditions exist:

- RHR B is in shutdown cooling.
- Reactor coolant temperature is 170°F.
- Time to boil is 3 hours.

An event occurs resulting in reactor water level dropping to +7”.

What procedures will the CRS enter directly to mitigate this transient?

The CRS enters...

- A. PPM 5.1.1 RPV Control (Mode 1-3) and ABN-RHR-SDC-LOSS.
- B. PPM 5.1.1 RPV Control (Mode 1-3) and ABN-ADHR.
- C. PPM 5.1.9 RPV Control (Mode 4) and ABN-RHR-SDC-LOSS.
- D. PPM 5.1.9 RPV Control (Mode 4) and ABN-ADHR.

Answer: C

K/A Match:

Procedure entry conditions and mitigation for a given set of plant conditions.

Explanation:

This question requires understanding of procedure entry and steps for the conditions provided to mitigate this event.

Shutdown cooling is isolated and lost automatically when RPV level is below +13 inches.

- A. Incorrect –Plausible – Based on the indications provided the candidate may incorrectly determine that we are in mode 3 which would then require entry into PPM 5.1.1 RPV Control. This is incorrect based on a reactor coolant temperature of 170F, the plant is currently in Mode 4. ABN-ADHR would only be entered as a mitigation strategy when it was determined that a RHR shutdown cooling loop could not be returned to service. ABN-RHR-SDC-LOSS would be the correct ABN to mitigate this event after actions were taken per PP 5.1.9 RPV Control.

- B. Incorrect – Plausible – Based on the indications provided the candidate may incorrectly determine that we are in mode 3 which would then require entry into PPM 5.1.1 RPV Control. This is incorrect based on a reactor coolant temperature of 170F, the plant is currently in Mode 4. Part 2:ABN-ADHR would not be used to mitigate this event until it has been determined that a RHR SDC loop could not be restored. Based on information provided SDC can be restored using ABN-FAZ and ABN-RHR-SDC-LOSS.
- C. **CORRECT** – Entry conditions for PPM 5.1.9 have been and would be used to mitigate the level transient. Based on information provided SDC can be restored using ABN-RHR-SDC-LOSS once level has been restored via PPM 5.1.9.
- D. Incorrect – Plausible - Entry conditions for PPM 5.1.9 have been met. Part 2:ABN-ADHR would not be used to mitigate this event until it has been determined that a RHR SDC loop could not be restored. Based on information provided SDC can be restored using ABN-RHR-SDC-LOSS once level has been restored via PPM 5.1.9

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.1.1 RPV Control (Modes 1 to 3)	
Technical Specifications	
ABN-SDC-LOSS	
PPM 5.1.9 RPV Control (Mode 4)	
ABN-ADHR	

Proposed references to be provided during examination: N/A

Learning Objective: RHR-12 Evaluate given key Residual Heat Removal System parameters, if needed determine a course of action to correct or mitigate the following abnormal condition(s):
 t. System isolation
 v. Reactor low water level

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Candidate must interpret plant conditions and determine procedures based on mitigation strategy and procedure hierarchy.

10 CFR Part 55 Content: 55.41 10

Comments / Reference: PPM 5.1.1 RPV Control	Rev:	Major: 23	Minor: Min
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<h1>5.1.1</h1>	RPV CONTROL (Mode 1-3)	 ENERGY NORTHWEST Columbia Generating Station
	Revision 23	Date 06/01/2021

Applicability is mode 1-3

- RPV level below +13 in.
- RPV pressure above 1060 psig
- Drywell pressure above 1.68 psig
- Both:
 - a reactor scram is required
 - AND
 - reactor power is above 5% or cannot be determined

Comments / Reference: PPM 5.1.9 RPV Control	Rev:	Major: 23	Minor: Min
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<h1>5.1.9</h1>	RPV CONTROL (Mode 4)	 ENERGY NORTHWEST Columbia Generating Station
	Revision 0	Date 06/01/2021

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Applicability is mode 4

- RPV level cannot be maintained above +13 in.
- RPV water temperature cannot be maintained below 200°F

Comments / Reference: ABN-RHR-SDC-LOSS	Rev:	Major: 7	Minor: Min
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Number: ABN-RHR-SDC-LOSS	Use Category: CONTINUOUS
Title: Loss of Shutdown Cooling	

1.0 ENTRY CONDITIONS

Loss of an in-service RHR shutdown cooling (SDC) loop.

2.0 AUTOMATIC ACTIONS

2.1 RHR valves (RHR-V-8, 9, 23, 53A, 53B) close on the following signals:

- 2.1.1 High RHR room area ΔT (GT 55°F)
- 2.1.2 High reactor steam dome pressure (125 psig)
- 2.1.3 **Low reactor vessel water level (LT 13")**
- 2.1.4 RHR system excess flow (125% of two loop flow)
- 2.1.5 High RHR room temp (GT 130 - 150°F)

4.1 WHEN a loss of SDC occurs,
THEN PERFORM the following:

- 4.1.1 **MONITOR** reactor coolant temperatures per CRS directions. _____
- 4.1.2 **REFER** to "Time to Boil" Curves from the Shutdown Protection Plan. _____

NOTE: The following step should be performed concurrently with shutdown cooling restoration.

NOTE: If during the performance of the following step, Shutdown Cooling is restored before reactor coolant temperature reaches 200°F, then the step may be exited as directed by the CRS.

- 4.1.3 IF Time to Boil is LE 2 hours,
OR reactor coolant temperature is expected to exceed 200°F,
THEN PERFORM the following: {AR-203695}
 - a. **EVACUATE** all personnel from the Drywell and Wetwell _____
 - b. **CLOSE** all Containment hatches and doors. {P-247034} _____
 - c. **ESTABLISH** Primary Containment integrity _____
 - d. **ESTABLISH** Secondary Containment integrity _____
 - e. **EVALUATE** Emergency Classification per PPM 13.1.1. _____
- 4.1.4 **IDENTIFY** and **CORRECT** the cause. _____

4.1.6 IF possible,
THEN START one RHR SDC cooling loop per SOP-RHR-SDC
(lost loop preferred). _____

Comments / Reference: ABN-ADHR	Rev:	Major:002	Minor:N/A
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While the entry conditions are technically met for entry into ABN-ADHR, Attachment 7.1 is only performed if RHR-SDC loop cannot be restored. Based on conditions provided PPM 5.1.9 can be used to mitigate level and RHR SDC can be restored.

1.0 ENTRY CONDITIONS

All of the following Condition must be met:

- Mode 3, 4, or 5
- One Shutdown Cooling system is not available.
- CRS/Shift Manager has determined Alternate Decay Heat Removal operation may be required.
- An Alternate Shutdown Cooling method is in use

2.0 AUTOMATIC ACTIONS

None

3.0 IMMEDIATE OPERATOR ACTIONS

None

4.0 SUBSEQUENT OPERATOR ACTIONS

NOTE: Alternate Shutdown Cooling (ABN-RHR-SDC-ALT) can be performed concurrently with this procedure if normal Shutdown Cooling cannot be restored.

4.1 IF one RHR shutdown cooling loop is not available and cannot be restored, THEN PERFORM Attachment 7.1 to set up for Alternate Decay Heat Removal while performing Alternate Shutdown Cooling.

Comments / Reference: PPM 1.3.1	Rev:	Major: 134	Minor: N/A
<p>4.8.1 Procedure Hierarchy</p> <p>a. The Volume 5 Emergency Operating Procedures (EOPs) and the actions specified therein have priority/precedence over all Volume 2, 3, and 4 Procedures when an emergency exists (EOP entry condition(s) is/are met).</p> <p>b. During a transient, and as plant conditions continue to degrade, the flow path of procedure usage is from the Volume 2 and 3 Procedures to the Volume 4, Abnormal Procedures, and then to the EOPs.</p>			

Comments / Reference: Technical Specifications	Rev:	Major: 251	Minor: Min
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Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTOR MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	Run	NA
2	Startup	Refuel ^(a) or Startup/Hot Standby	NA
3	Hot Shutdown ^(a)	Shutdown	> 200
4	Cold Shutdown ^(a)	Shutdown	≤ 200
5	Refueling ^(b)	Shutdown or Refuel	NA

(a) All reactor vessel head closure bolts fully tensioned.

(b) One or more reactor vessel head closure bolts less than fully tensioned.

Question: RO-10

Examination Outline Cross-reference: 10

Revision: 0

Date: 6/2/22

Tier: 1

Group: 1

K/A Number: 295023.AK1.03

Level of Difficulty: LOD

RO Importance Rating: 3.8

K/A Description: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Refueling Accidents: Inadvertent criticality

CGS is in Mode 5.

A core reload is in progress.

- A fuel bundle is being inserted into the core.

The Reactor Operator reports that there has been an unexpected doubling of the average SRM count rate.

What action should be taken?

The fuel bundle...

- insertion may continue as long as the average SRM count rate does not double again.
- should be immediately returned to the Spent Fuel Pool until the SNE has determined margin to criticality.
- insertion should be stopped until SRM count rates stabilize. The bundle may then be inserted slowly while closely monitoring subcritical multiplication.
- should be placed in a temporary setdown location identified on the Nuclear Components Transfer List (NCTL) until the Reactivity Manager approves resumption of CORE ALTERATIONS.

Answer: C

K/A Match:

Requires knowledge of required steps to prevent inadvertent criticality during fuel handling evolutions.

Question: RO-10

Explanation:

A. Incorrect. Plausible since, for the conditions given, the fuel move may continue slowly once SRM count rates stabilize. However, this is not dependent on average count rates doubling twice.

B. Incorrect. Plausible since returning the fuel bundle to the SFP is an action that is taken if the fuel bundle cannot be placed in the core due to debris or mechanical interference. However, for the conditions given, the fuel move should be stopped until SRM count rates stabilize.

C. Correct. In accordance with PPM 6.3.2, Fuel Movement with the Refueling Bridge, section 5.2.4.e, if the average SRM count rate unexpectedly doubles, the bundle insertion should be stopped until SRM count rates stabilize.

D. Incorrect. Plausible since placing the fuel bundle in a temporary set down location is the correct action if the fuel bundle cannot be transferred to the SFP due to debris or mechanical interference. However, for the conditions given, the fuel move should be stopped until SRM count rates stabilize.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 6.3.2, Fuel Movement with the Refueling Bridge	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 13286 - Given initial SRM count rate data, determine the count rates that would require a fuel shuffle to be suspended.

Question Source: Bank #: Bank #

Modified Bank #: Mod Bank #. (Note changes or attach parent)

New

Question History: Last NRC Exam: N/S

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize a knowledge of the SRM parameters that must be met to suspend a fuel move with the actions required when those parameters are exceeded.

10 CFR Part 55 Content: 55.41 10

Question: RO-10

Comments / Reference: PPM 6.3.2	Rev:	Major: 027	Minor: N/A
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Number: 6.3.2	Use Category: REFERENCE	Major Rev: 027
Title: Fuel Movement with the Refueling Bridge		Minor Rev: N/A
		Page: 10 of 25

5.1.5 IF a fuel bundle cannot be placed into the NCTL specified destination location due to some issue such as debris in bottom of cell/rack location, mechanical interference with blade guides or control blades, etc THEN:

- a. If the destination location is the SFP, then place the bundle in one of the temporary set down locations specified on the NCTL, notify the on duty SNE of change/issue to evaluate downstream impact on change, and continue with fuel movements.
- b. If the destination location is in the core, then return the fuel bundle to its originating location if from the SFP. ELSE, place the bundle in a temporary setdown location in the SFP as specified on the NCTL.

NOTE: Prior to initiating CORE ALTERATIONS, the SRM count rate readings should be recorded on a form similar to Attachment 7.3. Attachment 7.3 should be periodically updated/validated, as needed.

NOTE: Inserting a bundle near an SRM may have a strong effect on the SRM count rate. An increase in SRM count rate may occur when any bundle is inserted within two face adjacent locations from an SRM. If any other bundle insertion causes a substantial SRM count rate increase, as described below, it is considered unexpected.

- e. The Reactor Operator should periodically review SRM readings during CORE ALTERATIONS and MONITOR for the following unexpected conditions:
 - 1) A doubling in average SRM count rate occurs, or
 - 2) Two doublings of any single SRM count rate occurs.
 - 3) IF either of the above unexpected SRM behavior is observed, **REQUEST** the bridge crew to **STOP** bundle insertion until SRM count rates stabilize, **THEN**
 - a) Continue slow insertion of the fuel bundle while closely monitoring subcritical multiplication.
 - b) After the bundle has been seated, REQUEST the duty SNE support for determining whether a substantial reduction in margin to criticality has occurred. If warranted, the SNE may recommend performing a subcritical check as per reference 2.11.
 - c) Resumption of CORE ALTERATIONS requires the concurrence of the Reactivity Manager.

Examination Outline Cross-reference: RO-11

Revision: 1

Date: 11/03/2022

Tier: 1

Group: 1

K/A Number: 295024 EK3.01

Level of Difficulty: 3

RO Importance Rating: 4.4

K/A Description: Knowledge of the reasons for the following responses or actions as they apply to High Drywell Pressure:
Drywell spray (Mark I, II)

The mode switch is in shutdown due to a leak in Drywell.

- Drywell pressure is 13 psig and up slow.
- The Reactor operator has verified Drywell temperature is below DSIL.
- Wetwell level has been verified below 50 feet.

You are initiating drywell spray from step P-11 of PPM 5.2.1, Primary Containment Control.

Drywell spray (1) designed to prevent the over pressurization of the containment.

Initiation of Drywell sprays prevents chugging by (2).

- A. (1) is
(2) reducing drywell temperature.
- B. (1) is
(2) allowing non-condensable gasses to enter the drywell from the wetwell.
- C. (1) is NOT
(2) reducing drywell temperature.
- D. (1) is NOT
(2) allowing non-condensable gasses to enter the drywell from the wetwell.

Answer: D

K/A Match:

Requires understanding of the reasons for drywell spray during high Drywell pressure conditions.
Includes reasons stated in both the FSAR and PPM 5.0.10a.

Explanation:

A. Incorrect.

Plausible because initiating drywell spray reduces drywell pressure. Incorrect because the FSAR states that containment spray is not required to prevent the overpressurization of containment.

Plausible because drywell spray lowers both pressure and temperature. Temperature is related to the cyclic condensation of steam that creates the chugging condition. Incorrect because reducing temperature does not prevent chugging. PPM 5.0.10a states that drywell pressure must be reduced below the setpoint for the vacuum breakers to open allowing non-condensable gases to flow into the drywell from the wetwell.

B. Incorrect.

Plausible because initiating drywell spray reduces drywell pressure. Incorrect because the FSAR states that containment spray is not required to prevent the overpressurization of containment.

Plausible because allowing non-condensable gasses to enter the drywell from the wetwell is the correct answer. PPM 5.0.10a states that chugging is prevented by allowing non-condensable flow.

C. Incorrect.

Plausible because this portion of the answer is correct. Containment spray is not required to prevent overpressurization of containment per the FSAR.

Plausible because drywell spray lowers both pressure and temperature. Temperature is related to the cyclic condensation of steam that creates the chugging condition. Incorrect because reducing temperature does not prevent chugging. PPM 5.0.10a states that drywell pressure must be reduced below the setpoint for the vacuum breakers to open allowing non-condensable gases to flow into the drywell from the wetwell.

D. Correct.

Containment spray is not required to prevent the overpressurization of containment per the FSAR.

PPM 5.0.10a states that chugging is prevented by allowing non-condensable gas flow from the wetwell to the drywell.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.0.10A	
FSAR Section 6.5.2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 5636 - Describe the function, purpose and design features of the following Primary Containment System components:

- Wetwell to drywell vacuum breakers (include opening pressure)
- Three spray spargers (12 inch and 6 inch)

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires student to recall the reasons for drywell spray as required in step P-11 of PPM 5.2.1.

10 CFR Part 55 Content: 55.41 10

Comments / Reference: 5.0.10A	Rev:	Major:0	Minor:N/A
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h. Steps P-8 through P-11:

- 1) Drywell sprays are initiated to preclude primary containment damage from chugging—the cyclic condensation of steam at the downcomer openings. Scale model tests have demonstrated that chugging can occur only if the drywell atmosphere is mostly steam, with the noncondensable fraction less than 1%. Chugging can thus be prevented by operating Drywell sprays; the resulting drywell pressure reduction opens the Wetwell-to-Drywell vacuum breakers, drawing noncondensibles from the Wetwell back into the Drywell. Initiating Drywell sprays when suppression chamber pressure exceeds the WSIP ensures the drywell noncondensable content remains above the chugging threshold with an added conservatism. The WSIP is the lowest Wetwell pressure which can occur when 95% of the noncondensibles in the Drywell have been transferred to the Wetwell.

6.5.2 CONTAINMENT SPRAY SYSTEM

Design Bases

The containment spray system is capable of reducing containment pressure during the postaccident period of a LOCA through condensation of steam in the drywell and through cooling of the noncondensable gases in the free volume above the suppression pool. Containment spray is not required to prevent overpressurization of the containment.

Question: RO-12

Examination Outline Cross-reference: RO-12

Revision: 0

Date: 09/01/22

Tier: 1

Group: 1

K/A Number: 295025 EA2.04

Level of Difficulty: 3

RO Importance Rating: (4.0/3.4)

K/A Description: Ability to determine and/or interpret the following as they apply to High Reactor Pressure: (CFR: 41.10 / 43.5 / 45.13) EA2.04 Suppression Pool Level

CGS is at 100% power.

A DEH failure results in a high pressure scram.

10 Minutes later:

- All rods are in, RPV Pressure and RPV Level are stable.
- Level control is in automatic on the startup flow control valves.
- RO2 is controlling Reactor pressure with SRVs.
- P601 A11 and A12, 2-3 SUPP POOL LEVEL HIGH/LOW remains in alarm.

What action will the crew take to mitigate Suppression Pool water level?

Crew will perform (1) to adjust Suppression Pool level as procedurally directed by (2).

- A. (1) PPM 5.5.23 Emergency Suppression Pool Makeup
(2) PPM 5.2.1 Primary Containment Control
- B. (1) PPM 5.5.23 Emergency Suppression Pool Makeup
(2) ABN-PRESSURE
- C. (1) SOP-RHR-SPC
(2) PPM 5.2.1 Primary Containment Control
- D. (1) SOP-RHR-SPC
(2) ABN-PRESSURE

Answer: C

K/A Match:

Actions required and guidance to reduce Suppression pool level following a high reactor pressure condition.

Question: RO-12

Explanation:

While is not stated that the Crew is in PPM 5.2.1 Primary containment control, understanding of plant design with a scram from 100% power and SRV's cycling followed by the RO controlling SRV's to control pressure for 10 minutes would result in high WW level. Fundamental part of answering this question revolves around the operators knowledge of the plant response to high pressure -> SRV's cycling even for a short period of time will result in EOP entry conditions for PPM 5.2.1 on HIGH suppression pool level and mitigation of high suppression pool level per PPM 5.2.1 is SOP-RHR-SDC to lower level.

- A. Incorrect – Plausible – (1) Incorrect - PPM 5.5.23 provides guidance for ONLY making up to the suppression pool on LOW level and does not contain guidance for lowering suppression pool level. This action is taken in PPM 5.2.1 Primary Containment Control when WW level is LT -2". Based on plant conditions WW level will by GT +2" and not less. (2) Correct - PPM 5.2.1 will be entered based on plant conditions. Based on simulator testing a high pressure scram with a DEH failure results in multiple SRV's Cycling, the entry conditions for PPM 5.2.1 on Suppression Pool HIGH level will be reached in less than a minute. PPM 5.2.1 Primary Containment Control step L-1 directs SOP-RHR-SPC to lower level
- B. Incorrect – Plausible -(1) Incorrect - PPM 5.5.23 provides guidance for ONLY making up to the suppression pool on LOW level and does not contain guidance for lowering suppression pool level. This action is taken in PPM 5.2.1 Primary Containment Control when WW level is LT -2". Based on plant conditions suing SRV's for pressure control WW level will by GT +2" and not less. (2) Incorrect – ABN-PRESSRE will be entered based on plant conditions, however the guidance in ABN-PRESSURE contains steps to mitigate the reactor pressure transient and does not contain guidance for lowering WW level.
- C. Correct – PPM 5.2.1 will be entered based on plant conditions. Based on simulator testing a high pressure scram with a DEH failure results in multiple SRV's Cycling, the entry conditions for PPM 5.2.1 on Suppression Pool HIGH level will be reached in less than a minute. PPM 5.2.1 Primary Containment Control step L-1 directs SOP-RHR-SPC to lower level.
- D. Incorrect – Plausible – (1) Correct SOP-RHR-SPC is used to mitigate high WW level it is just not directed from ABN-PRESSURE (2) Incorrect – ABN-PRESSRE will be entered based on plant conditions, however the guidance in ABN-PRESSURE contains steps to mitigate the reactor pressure transient and does not contain guidance or a reference to lowering WW level using SOP-RHR-SPC.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.5.23	
PPM 5.2.1 Primary Containment Control	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 8375 Describe the purpose of attempting to maintain

Question: RO-12

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

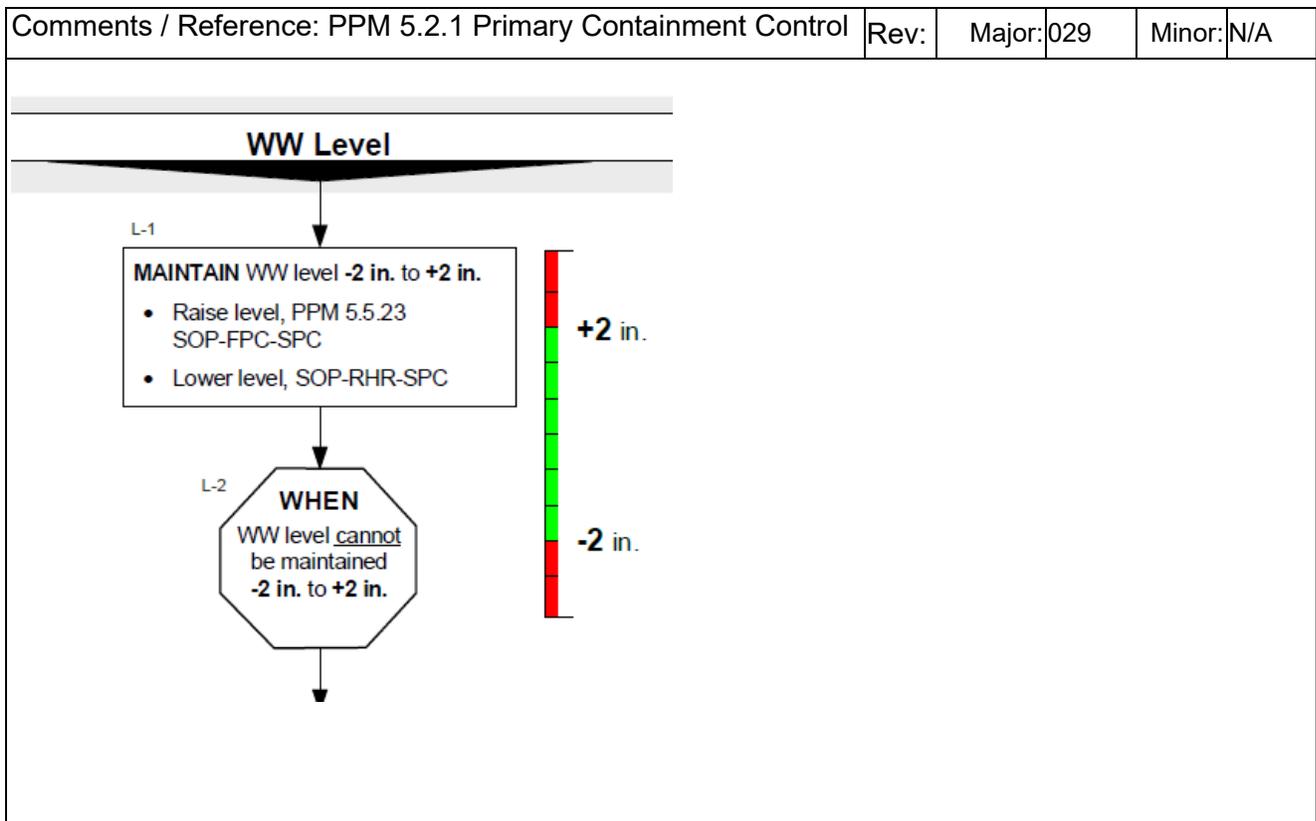
Justification for Cognitive Level

Must evaluate plant conditions and determine the correct procedure that will mitigate wetwell level increase.

10 CFR Part 55 Content: 55.41 7

Question: RO-12

Comments / Reference: PPM 5.5.23	Rev:	Major: 008	Minor: N/A
Number: 5.5.23		Use Category: CONTINUOUS	
Title: EMERGENCY SUPPRESSION POOL MAKEUP			
<p>1.0 <u>PURPOSE</u></p> <p>Provide the direction necessary to makeup to the Suppression Pool.</p>			



Comments / Reference: SOP-RHR-SPC	Rev:	Major: 008	Minor: 004
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Question: RO-12

Number: SOP-RHR-SPC	Use Category: CONTINUOUS	Major F
Title: Suppression Pool Cooling/Spray/Discharge/Mixing		Minor F Page: 4

1.0 PURPOSE

Provide instructions for Residual Heat Removal System operations in Suppression Pool Cooling, Spray, Discharge and Mixing.

5.3.15 IF discharging Suppression Pool Water to Radwaste,
THEN PERFORM the following:

Question: RO-13

Examination Outline Cross-reference: RO-13

Revision: 0

Date: 11/13/22

Tier: 1

Group: 1

K/A Number: 295026 EK2.10

Level of Difficulty: 4

RO Importance Rating: 3.7

K/A Description: Knowledge of the relationship between Suppression Pool High Water Temperature and the following systems or components: LPCS

CGS is in Mode 1.

An event occurs that causes an automatic scram and Wetwell to heat up.

As the Wetwell temperature increases, what action(s) will be performed?

- (1) When Wetwell temperature is above 90°F, enter PPM 5.2.1, Primary Containment Control.
- (2) When Wetwell temperature cannot be maintained below 135°F, enter PPM 5.1.1, RPV Control (Mode 1-3).
- (3) Before Wetwell temperature reaches 225°F, secure LPCS if not needed for core cooling.

- A. (1) ONLY.
- B. (2) ONLY.
- C. (1) and (3) ONLY
- D. (2) and (3) ONLY

Answer: C

K/A Match:

Requires a knowledge that Suppression Pool High Water Temperature affects the net positive suction head for LPCS which can lead to pump damage due to cavitation within the pump.

Question: RO-13

Explanation:

A. Incorrect: Plausible because Choice 1) is the wetwell temperature that requires entry into PPM 5.2.1, Primary Containment Control, and is correct. However, Choice 3) is also correct. The expectation is that LPCS should be operated, if possible, only when Wetwell temperature is below the NPSH Limit. If EOP/SAG objectives associated with providing core cooling cannot otherwise be achieved, however, operation with Wetwell temperature above the NPSH Limit is permitted.

B. Incorrect: Entry into PPM 5.1.1 due to Wetwell temperatures occurs before 110°F. Plausible because this is the temperature for entry into PPM 5.2.1 due to Drywell heat up.

C. Correct: Both Choice 1) and Choice 3) are correct. See A. above.

D. Incorrect: Plausible because Choice 3) is correct, however, Choice 2) is incorrect. See A. and B. above.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
5.0.10, Flowchart Training Manual Volume I, Introduction and References	
5.0.10A, Flowchart Training Manual Volume II, EOPs-Hot Bases	
5.2.1, Primary Containment Control (Mode 1-3)	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: RO-5.2.1-01 Maintain wetwell temperature IAW PPM 5.2.1.

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

Question History: Last NRC Exam: N/A

Question: RO-13

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

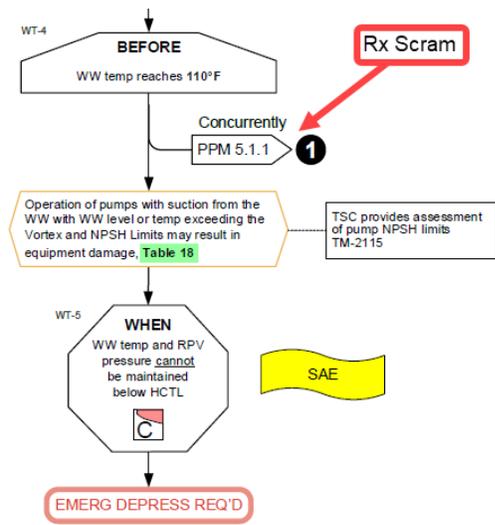
Justification for Cognitive Level

Requires the student to have a fundamental understanding of actions and entry conditions for EOP Flowchart 5.2.1, Primary Containment Control. Additionally, the student must synthesize a knowledge that temperature increases to pump suction supplies affects the saturation temperature within the pump which can lead to cavitation and damage to the pump. The student must also understand that if core cooling is needed, the pump should be operated regardless to damage.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: 5.2.1	Rev:	Major: 029	Minor: N/A
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Excerpt from PPM 5.2.1 Flowchart:



System	WW Level (ft)	WW Temp (°F)
RCIC with suction from WW	17.5	210
LPCS	14.5	225
RHR-C	14.5	225
RHR-A, B	14.0	225
HPCS with suction from WW	5.5	200

Question: RO-13

Comments / Reference: 5.0.10	Rev:	Major: 026	Minor: N/A
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Number: 5.0.10	Use Category: INFORMATION	Major Rev: 026
Title: Flowchart Training Manual Volume I, Introduction and References		Minor Rev: N/A
		Page: 157 of 173

9.27 NPSH Limit

18	Vortex and NPSH Limits	
<u>System</u>	<u>WW Level (ft)</u>	<u>WW Temp (°F)</u>
RCIC with suction from WW	17.5	210
LPCS	14.5	225
RHR-C	14.5	225
RHR-A, B	14.0	225
HPCS with suction from WW	5.5	200

- 9.27.1 **The NPSH Limit is the highest wetwell temperature which provides adequate net positive suction head for RCIC or an ECCS pump taking suction on the wetwell.** The NPSH Limit is a function of pump flow and wetwell overpressure (airspace pressure plus hydrostatic head over the RCIC or ECCS suction). It is utilized to preclude RCIC or ECCS damage due to pump cavitation.
- 9.27.2 As pump flow increases, the suction line head loss increases, reducing the margin to saturation at the pump inlet. The NPSH Limit therefore decreases with pump flow. As overpressure increases, the static pressure and margin to saturation at the pump inlet also increase. The NPSH Limit therefore increases with overpressure.
- 9.27.3 Wetwell temperatures associated with NPSH limits for pumps taking suction from the wetwell are listed in Table 18. Since NPSH limits are nearly constant over the range of pump flows, **the NPSH limits can be expressed as the single most limiting wetwell temperature for each pump.** The Table 18 temperatures thus conservatively bound the NPSH limit curves given in TM-2115 Tab 6 of the TSC Technical Support Guidelines (TSGs). TM-2115 Tab 6 may be consulted if additional margin is desired.

Question: RO-13

Comments / Reference: 5.0.10A		Rev:	Major: 000	Minor: N/A
Number: 5.0.10A		Use Category: INFORMATION		Major Rev: 000
Title: Flowchart Training Manual Volume II, EOPs-Hot Bases			Minor Rev: N/A	Page: 190 of 247
<p>3.8.3 Flowchart Technical Discussion</p> <p>a. Entry:</p> <p>The conditions which require entry into PPM 5.2.1 are symptoms of an emergency or conditions which, if not corrected, could degrade into an emergency. This set of entry conditions is sufficient to assure the flowchart will be entered for transients and accidents which are within the design and licensing basis for Columbia, and for additional events which have been evaluated as significant to emergency response.</p> <p>b. Entry conditions based on plant parameters are the following:</p> <ul style="list-style-type: none"> • Bulk Wetwell temperature above 90°F: This temperature is the most limiting Wetwell temperature LCO. • Average drywell temperature above 135°F: This temperature is the most limiting drywell temperature LCO. • Drywell pressure above 1.68 psig: This pressure corresponds to the high drywell pressure scram setpoint. • Wetwell water level above 31 ft 1-3/4 in. (+2 in.): This water level corresponds to the maximum Wetwell water level LCO. • Wetwell water level below 30 ft 9-3/4 in. (-2 in.): This water level corresponds to the minimum Wetwell water level LCO. • Primary containment hydrogen concentration above 3.56%: This concentration is the high hydrogen alarm setpoint. 				

Question: RO-13

Comments / Reference: 5.0.10A	Rev:	Major: 000	Minor: N/A
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Number: 5.0.10A	Use Category: INFORMATION	Major Rev: 000
Title: Flowchart Training Manual Volume II, EOPs-Hot Bases		Minor Rev: N/A Page: 195 of 247

d. Step WT-4:

- 1) The temperature value of 110°F is the Wetwell temperature at which Technical Specifications require a reactor scram.
- 2) The logic term "BEFORE" permits initiation of a reactor scram at any temperature up to 110°F. The symptom based approach to emergency response provides the latitude of delaying reactor scram until Wetwell temperature reaches ~110°F or initiating a reactor scram well before Wetwell temperature reaches 110°F.
- 3) Entry to PPM 5.1.1 assures that, if possible, the reactor is scrammed and shutdown before the requirement for boron injection is reached. Entry to the RPV Control procedures is explicitly stated because conditions requiring entry to PPM 5.2.1 do not necessarily require entry to PPM 5.1.1. Therefore, a scram may not have yet been initiated.
- 4) Directing that PPM 5.1.1 be entered rather than explicitly stating "initiate a reactor scram" in the Wetwell temperature control flowpath coordinates actions currently being performed if the RPV Control procedures have already been entered.

e. Caution before Step WT-5:

- 1) The NPSH Limit is the highest wetwell temperature which provides adequate net positive suction head for RCIC or an ECCS pump taking suction on the wetwell. The NPSH Limit is a function of pump flow and wetwell overpressure (airspace pressure plus hydrostatic head over the RCIC or ECCS suction). It is utilized to preclude RCIC or ECCS damage due to pump cavitation. Wetwell temperatures associated with NPSH Limits for pumps taking suction from the wetwell are listed in Table 18. Since NPSH Limits are nearly constant over the range of pump flows, the NPSH Limits can be expressed as the single most limiting wetwell temperature for each pump. The Table 18 temperatures thus conservatively bound the NPSH Limit curves given in TM-2115 Tab 6 of the TSC Technical Support Center Guidelines (TSCs). TM-2115 Tab 6 may be consulted if additional margin is desired.
- 2) The EOPs do not explicitly state when it is permissible to operate pumps irrespective of the NPSH Limits. The expectation is that pumps should be operated, if possible, only when Wetwell temperature is below the NPSH Limit. If EOP/SAG objectives associated with providing core cooling or avoiding primary containment failure cannot otherwise be achieved, however, operation with Wetwell temperature above the NPSH Limit is permitted.
- 3) Refer to Volume I, Section 9, of this document for discussion of the NPSH Limits.

Question: RO-14

Examination Outline Cross-reference: RO-14

Revision: 0

Date: 8/1/22

Tier: 1

Group: 1

K/A Number: 295028 2.4.31

Level of Difficulty: 3

RO Importance Rating: 4.2

K/A Description: Knowledge of annunciator alarms, indications, or response procedures: High Drywell Temperature (Mark I and Mark II only)

CGS is at 100% power.

- All five drywell forced coolers are in service.

An event occurs:

- Both P601.A11 1-3, Drywell/Suppression Pool Temperature High, and P601.A12 1-3, Drywell/Suppression Pool Temperature High, annunciators are in alarm.
- There are no leaking or open SRVs.
- Alarm Group 1 - Average Drywell Temperature High, (Point A01) on both CMS-TR-5 and CMS-TR-6 are verified to be causing the alarms and read similarly.
- Drywell temperature continues to rise.

Based on the above conditions, what is the **lowest** drywell temperature where PPM 5.2.1, Primary Containment Control, is entered?

- A. 90°F.
- B. 131°F.
- C. 135°F.
- D. 285°F.

Answer: C

K/A Match:

Requires a knowledge of when to enter PPM 5.2.1 based on the high drywell alarm response procedure.

Question: RO-14

Explanation:
A. <u>Incorrect: Plausible because this is the specified wetwell temperature to enter PPM 5.2.1.</u>
B. <u>Incorrect: Plausible because this is the alarm setpoint for Drywell/Suppression Pool Temperature High annunciator alarms.</u>
C. <u>Correct: PPM 5.2.1 is entered when A01 is GT 135°F when all five coolers are operating.</u>
D. <u>Incorrect: Plausible because this is the expected drywell temperature per OI-15 to initiate drywell sprays.</u>

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
5.2.1, Primary Containment Control	
OI-15, EOP and EAL Clarifications	
4.601.A11, P601.A11 Annunciator Panel Alarms	
4.601.A12, P601.A12 Annunciator Panel Alarms	

Proposed references to be provided during examination: N/A

Learning Objective: RO-5.2.1-02 Maintain drywell temperature IAW PPM 5.2.1.

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level:

Requires the student to synthesize a knowledge of PPM 5.2.1 entry conditions with alarm conditions and Alarm Response Procedure actions.

10 CFR Part 55 Content: 55.41 10

Question: RO-14

Comments / Reference: 5.2.1	Rev:	Major: 029	Minor: N/A
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Excerpt from PPM 5.2.1:

- WW temp above 90°F
- Drywell temp above 135°F
- Drywell pressure above 1.68 psig
- WW level above +2 in. or below -2 in.
- PC hydrogen above 3.56%

5.2.1

PRIMARY CONTAINMENT CONTROL
(Mode 1-3)

ENERGY NORTHWEST
Columbia Generating Station

Revision 29 Date 06/01/2021

CONTROLLED COPY

Perform Concurrently to Monitor and Control:

PPM 5.2.1 Entry Conditions

Comments / Reference: 4.601.A11	Rev:	Major: 027	Minor: N/A
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Excerpt from PPM 4.601.A11:

Number: 4.601.A11	Use Category: CONTINUOUS	Major Rev: 027
Title: 601.A11 Annunciator Panel Alarms		Minor Rev: N/A
		Page: 6 of 41

1-3 DRYWELL/SUPPRESSION POOL TEMPERATURE HIGH

1-3 WINDOW	SOURCE	AUTOMATIC ACTIONS
DRYWELL/ SUPP POOL TEMP HIGH	CMS-TR-6	None

1. **CHECK** for a leaking or open SRV.
2. **DETERMINE** which Alarm Group on CMS-TR-6 caused the alarm.
3. **IF** in Modes 1 and 2,
AND instrument operability is in doubt
THEN REFER to LCS 1.3.3.1.

Alarm Group 1 - AVG D/W TEMP HI (Pt A01 - 131°F)

1. **DETERMINE** which input(s) caused the alarm.
2. **COMPARE** the Average Drywell Temperature on CMS-TR-6 with the associated reading on CMS-TR-5.

3. **ENTER** PPM 5.2.1, Primary Containment Control, under the following conditions: {1}

- A01 is GT 135°F when all five coolers are operating.
- GT 135°F based on an average temperature obtained from a manual calculation of the active fan inlet temperatures.
- A01 is GT 135°F with inlet temperature signals from the inoperable drywell cooling fan inputs removed from the A01 average.

Question: RO-14

Comments / Reference: 4.601.A12	Rev:	Major: 029	Minor: 001
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Excerpt from PPM 4.601.A12:

Number: 4.601.A12	Use Category: CONTINUOUS	Major Rev: 029
Title: 601.A12 Annunciator Panel Alarms		Minor Rev: 001
		Page: 6 of 34

1-3 DRYWELL/SUPPRESSION POOL TEMPERATURE HIGH

1-3 WINDOW	SOURCE	AUTOMATIC ACTIONS
DRYWELL/ SUPP POOL TEMP HIGH	CMS-TR-5	None

1. **CHECK** for a leaking or open SRV.
2. **DETERMINE** which Alarm Group on CMS-TR-5 caused the alarm.
3. IF instrument operability is in doubt,
AND in Modes 1 and 2,
THEN REFER to LCS 1.3.3.1.

Alarm Setpoint

Alarm Group 1 - AVG D/W TEMP HI (Pt A01 - 131°F)

PPM 5.2.1
Entry Condition

1. **DETERMINE** which input(s) caused the alarm.
2. **COMPARE** the Average Drywell Temperature on CMS-TR-5 with the associated reading on CMS-TR-6.
3. **ENTER** PPM 5.2.1, Primary Containment Control under the following conditions: {1}
 - A01 is GT 135°F when all five coolers are operating
 - GT 135°F based on an average temperature obtained from a manual calculation of the active fan inlet temperatures
 - A01 is GT 135°F with inlet temperature signals from the inoperable drywell cooling fan inputs removed from the A01 average

Comments / Reference: OI-15	Rev:	Major: 037	Minor: N/A
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Excerpt from PPM OI-15:

- Step DT-3: Operation of the Drywell sprays is directed before Drywell temperature increases to 330°F. The expectation is that Drywell sprays will be initiated at 285°F.

Examination Outline Cross-reference: RO-15

Revision: 0

Date: 8/10/22

Tier: 1

Group: 1

K/A Number: 295030 2.4.46

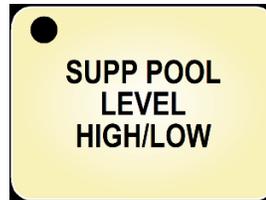
Level of Difficulty: 2

RO Importance Rating: 4.2

K/A Description: Ability to verify that the alarms are consistent with the plant conditions: Low Suppression Pool Water Level

CGS is at 100% power.

The following annunciator is in alarm:



- (1) How is this alarm verified to be consistent with plant conditions?
- (2) What is the significance of the black dot in the annunciator window?

- A. (1) Determine level on CMS-LR-4 only.
(2) Potential EOP entry.
- B. (1) Determine level on CMS-LR-3 and CMS-LR-4.
(2) Potential EOP entry.
- C. (1) Determine level on CMS-LR-4 only.
(2) Potential EAL entry.
- D. (1) Determine level on CMS-LR-3 and CMS-LR-4.
(2) Potential EAL entry.

Answer: D

K/A Match:

Requires a knowledge of how to verify that the alarms are consistent with the plant conditions, specifically Suppression Pool Water Level.

Explanation:

- A. (1) Incorrect: Plausible because CMS-LR-4 is the source of the annunciator alarm.
(2) Incorrect: Plausible because annunciator windows are marked with an orange triangle for potential EOP entries.
- B. (1) Correct: Alarm annunciator procedure 4.601.A11 2-3, Suppression Pool Level High/Low, directs level to be determined on CMS-LR-3 and CMS-LR-4.
(2) Incorrect: See A(2) above.
- C. (1) Incorrect: See A(1) above.
(2) Correct: An annunciator window marked with a black dot indicates a potential EAL entry.
- D. (1) Correct: See B(1) above.
(2) Correct: See C(2) above.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
OI-45, Color Banding of Control Room Instrumentation	
4.601.A11, P601.A11 Annunciator Panel Alarms	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 2.1.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup

Question Source: Bank #: _____ Bank #
 Modified Bank #: _____ Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level:

Requires the student to synthesize a knowledge of Alarm Response Procedure actions with a knowledge of Annunciator Window markings to determine appropriate course of action.

10 CFR Part 55 Content: 55.41 10

Excerpt from 2-3:

Number: 4.601.A11	Use Category: CONTINUOUS	Major Rev: 027
Title: 601.A11 Annunciator Panel Alarms		Minor Rev: N/A
		Page: 14 of 41

2-3 SUPPRESSION POOL LEVEL HIGH/LOW

2-3 WINDOW	SOURCE	AUTOMATIC ACTIONS
SUPP POOL LEVEL HIGH/LOW	CMS-LR-4 (GT +0.73" OR LT -0.73 ")	None

NOTE: Instrument zero is 466' 2.75" for Suppression Pool level.

NOTE: CMS-LT-2 reference leg will fill, causing downscale indication on Suppression Pool Narrow Range indication when Wide Range Indication increases to approximately 33 ft.

1. DETERMINE level on CMS-LR-3 and CMS-LR-4.

Excerpt from PPM OI-45:

3.1.2 Annunciator/EPN markings are as follows:

Orange triangle Potential EOP entry

Black dot Potential EAL entry

Blue dot Potential PAM TS or LCS entry

Examination Outline Cross-reference: RO-16

Revision: 0

Date: 7/26/22

Tier: 1

Group: 1

K/A Number: 295031 EK2.11

Level of Difficulty: 3

RO Importance Rating: 4.3

K/A Description: Knowledge of the relationship between Reactor Low Water Level and the following systems or components: Reactor Protection System

CGS is at 100% power.

An event occurs:

- Both Reactor Protection System (RPS) trip systems deenergize due to Reactor Low Water Level.
- All immediate actions of PPM 3.3.1, Reactor Scram QC, have been completed.
- All rods are in.

The RPS trip occurred at a level of (1) and the **NEXT** procedure to be entered is (2).

- A. (1) +13"
(2) PPM 5.1.1, RPV Control (Mode 1-3)
- B. (1) +13"
(2) ABN-RPS, Loss of RPS
- C. (1) -50"
(2) PPM 5.1.1, RPV Control (Mode 1-3)
- D. (1) -50"
(2) ABN-RPS, Loss of RPS

Answer: A

K/A Match:

Requires a knowledge of the reactor low water level that will cause an RPS trip.

Explanation:

- A. (1) Correct: The RPS low reactor level trip occurs at +13".
(2) Correct: Entry conditions for PPM 5.1.1 have been met and Annunciator Response Procedures (ARP) 4.603.A7 2-3 and 4.603.A8 2-3 specify entry into PPM 5.1.1 after PPM 3.3.1.
- B. (1) Correct: See A(1) above.
(2) Incorrect: Plausible because ABN-RPS provides direction for resetting RPS trip logic, however, the entry conditions for ABN-RPS have not been met. This procedure provides direction in the event the output from one or both RPS MG Sets is lost.
- C. (1) Incorrect: Plausible because -50" is the low reactor water level trip point for NSSSS isolation.
(2) Correct: See A(2) above.
- D. (1) Incorrect: See C(1) above.
(2) Incorrect: See B(2) above.

Technical Reference(s)	
3.3.1-QC, Reactor Scram Quick Card	Attached w/ Revision # See Comments / Reference
5.1.1, RPV Control (Mode 1-3)	
ABN-RPS, Loss of RPS	
1.3.1, Operating Policies, Programs, and Practices	
4.601.A11, P601.A11 Annunciator Panel Alarms	
4.601.A12, P601.A12 Annunciator Panel Alarms	
4.603.A7, P603.A7 Annunciator Panel Alarms	
4.603.A8, P603.A8 Annunciator Panel Alarms	

Proposed references to be provided during examination: N/A

Learning Objective: RO-5.2.1-01 Maintain wetwell temperature IAW PPM 5.2.1.

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level:

Requires the student to synthesize a knowledge of RPS Trip setpoints and procedure entry conditions with an understanding of procedure hierarchy.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: 4.603.A7	Rev:	Major: 060	Minor: N/A
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Excerpt from PPM 4.603.A7:

Number: 4.603.A7	Use Category: CONTINUOUS	Major Rev: 060 Minor Rev: N/A Page: 18 of 73
Title: 603.A7 Annunciator Panel Alarms		

2-3 REACTOR PRESSURE VESSEL LEVEL LOW TRIP +13"

2-3 WINDOW	SOURCE	AUTOMATIC ACTIONS
RPV LEVEL LOW TRIP +13"	MS-LIS-24A (LE +13") (RPS-RLY-K6A)	Half Scram
	OR MS-LIS-24C (LE +13") (RPS-RLY-K6C)	

1. IF RPV level is verified to be below +13"
THEN **PERFORM** the following:
 - a. **VERIFY** a Reactor Scram has occurred
 - b. **REFER** to PPM 3.3.1, Reactor Scram.
 - c. **ENTER** Emergency Operating Procedure 5.1.1, RPV Control.

Comments / Reference: 4.603.A8 Rev: Major: 045 Minor: 001

Excerpt from PPM 4.603.A8:

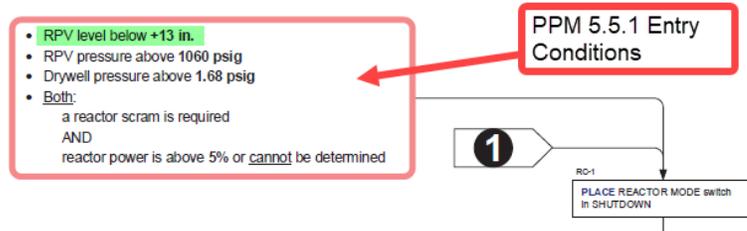
Number: 4.603.A8	Use Category: CONTINUOUS	Major Rev: 045 Minor Rev: 001 Page: 19 of 68
Title: 603.A8 Annunciator Panel Alarms		

2-3 REACTOR PRESSURE VESSEL LEVEL LOW TRIP +13"

2-3 WINDOW	SOURCE	AUTOMATIC ACTIONS
RPV LEVEL LOW TRIP +13"	MS-LIS-24B (LE +13") (RPS-RLY-K6B)	Half Scram
	OR MS-LIS-24D (LE +13") (RPS-RLY-K6D)	

1. IF RPV level LT +13",
THEN **PERFORM** the following:
 - a. **VERIFY** a Reactor Scram has occurred.
 - b. **REFER** to PPM 3.3.1, Reactor Scram.
 - c. **ENTER** Emergency Operating Procedure 5.1.1, RPV Control.

Excerpt from PPM 5.1.1 Flowchart:



Excerpt from ABN-RPS:

Number: ABN-RPS	Use Category: CONTINUOUS	Major Rev: 016
Title: Loss of RPS		Minor Rev: N/A
		Page: 3 of 14

1.0 ENTRY CONDITIONS

- Loss of either or both divisions of RPS power

Excerpts from PPM 1.3.1:

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 134
Title: Operating Policies, Programs, and Practices		Minor Rev: N/A
		Page: 23 of 102

4.8 Procedure Usage

NOTE: Use and adherence to approved Plant Procedures is controlled by SWP-PRO-01.

4.8.1 Procedure Hierarchy

a. The Volume 5 Emergency Operating Procedures (EOPs) and the actions specified therein have priority/precedence over all Volume 2, 3, and 4 Procedures when an emergency exists (EOP entry condition(s) is/are met).

b. During a transient, and as plant conditions continue to degrade, the flow path of procedure usage is from the Volume 2 and 3 Procedures to the Volume 4, Abnormal Procedures, and then to the EOPs.

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 134
Title: Operating Policies, Programs, and Practices		Minor Rev: N/A
		Page: 25 of 102

4.8.4 Abnormal Procedures

NOTE: These procedures are not Emergency procedures, but may be executed concurrently with EOPs if, following diagnosis of plant conditions, the Abnormal Procedure is applicable. If any conflict in the required operator actions exist between the Abnormal Procedure and the EOPs, the actions specified in the EOPs should have priority/precedence. If a Control Room evacuation is required, ABN-CR-EVAC will supersede the EOPs, once the Control Room is evacuated.

NOTE: Refer to PPM 1.3.82 for reportability requirements. If a plant transient or equipment issue has resulted in a change in CGS real output generating capability.

Excerpt from 4.601.A11 3-3:

Number: 4.601.A11	Use Category: CONTINUOUS	Major Rev: 027
Title: 601.A11 Annunciator Panel Alarms		Minor Rev: N/A Page: 19 of 41

3-3 NSSSS ISOLATION RPV LEVEL LOW (-50")

3-3 WINDOW	SOURCE	AUTOMATIC ACTIONS
NSSSS ISOL RPV LEVEL LOW -50"	RPS-RLY-K39C (-50" RPV Level) (MS-LS-300C) (147.3" WC)	Single switch actuation <ul style="list-style-type: none"> • Half Isolation • Initiates LPCI-B/C LOCA Time Delay Relay Logic
	OR RPS-RLY-K39D (-50" RPV Level) (MS-LS-300D) (147.3" WC)	Both switches actuating closes: <ul style="list-style-type: none"> • RWCU-V-1 • RRC-V-19 • FDR-V-3 • EDR-V-19

Comments / Reference: 4.601.A12 1-2

Rev: Major: 029 Minor: N/A

Excerpt from 4.601.A12 1-2:

Number: 4.601.A12	Use Category: CONTINUOUS	Major Rev: 029
Title: 601.A12 Annunciator Panel Alarms		Minor Rev: 001 Page: 5 of 34

1-2 NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM ISOLATION RPV LEVEL LOW (-50")

1-2 WINDOW	SOURCE	AUTOMATIC ACTIONS
NSSSS ISOL RPV LEVEL LOW -50"	<ul style="list-style-type: none"> • RPS-RLY-K39A (-50" RPV Level) (MS-LS-300A) (147.3" WC) 	<ul style="list-style-type: none"> • Single switch actuation <ul style="list-style-type: none"> • Half Isolation • Initiates LPCS/LPCI-A LOCA Time Delay Relay Logic
	OR <ul style="list-style-type: none"> • RPS-RLY-K39B (-50" RPV Level) (MS-LS-300B) (147.3" WC) 	<ul style="list-style-type: none"> • Both switches actuating closes: <ul style="list-style-type: none"> • RWCU-V-4 • RRC-V-20 • FDR-V-4 • EDR-V-20

Question: RO-17

Examination Outline Cross-reference: RO-17

Revision: 0

Date: 5/12/2022

Tier: 1

Group: 1

K/A Number: 295037 EK3.07

Level of Difficulty: 3

RO Importance Rating: 4.2

K/A Description: Knowledge of the reasons for the following responses or actions as they apply to SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown: Alternate control rod insertion methods.

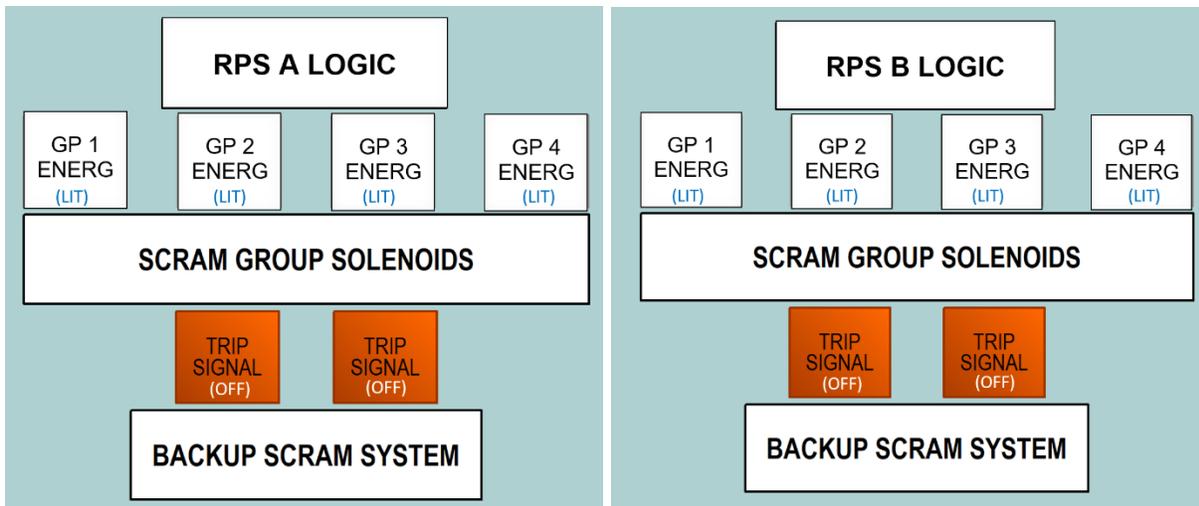
CGS is operating at 100% power.

A transient occurs that requires a reactor scram.

All immediate actions for a reactor scram have been completed.

- There is no rod motion following the scram.

The following indications are observed:



The crew is performing PPM 5.5.11 – Alternate Control Rod Insertions.

What method of control rod insertion will be attempted first to shutdown the reactor?

- Vent the scram air header.
- Vent control rod overpiston volumes.
- Remove fuses to de-energize scram solenoids.
- Scram individual control rods with scram test switches.

Question: RO-17

Answer: C

K/A Match:

Requires student to evaluate plant status and apply knowledge of various alternate control rod insertion methods.

Explanation:

A. Incorrect. Plausible because with no blue scram lights on this method could be used to insert control rods. Incorrect because removing fuses to de-energize scram solenoids will be done before this step IAW PPM 5.5.11.

B. Incorrect. Plausible because this method is directed by PPM 5.5.11 to insert control rods and this method would be used if removing fuses to de-energize scram solenoids is not successful. Incorrect because per PPM 5.5.11 fuses will be removed prior to venting overpiston areas.

C. Correct. Per PPM 5.5.11 with any white RPS Scram Light on the first action taken will be to Remove fuses to de-energize scram solenoids.

D. Incorrect. Plausible because this method is directed by PPM 5.5.11 to insert control rods with any white RPS scram light on and this method would be used if removing fuses to de-energize scram solenoids is not successful. Incorrect because per PPM 5.5.11 fuses will be removed prior to scrambling individual rods with scram test switches.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.1.2	
PPM 5.5.11	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: Reference or N/A

Learning Objective: 16160 - Given the following, explain the purpose, general content, hierarchy, and significant precautions and limitations of the following Emergency Support Procedures: d. Alternate Control Rod Insertion

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

Question History: Last NRC Exam: N/A

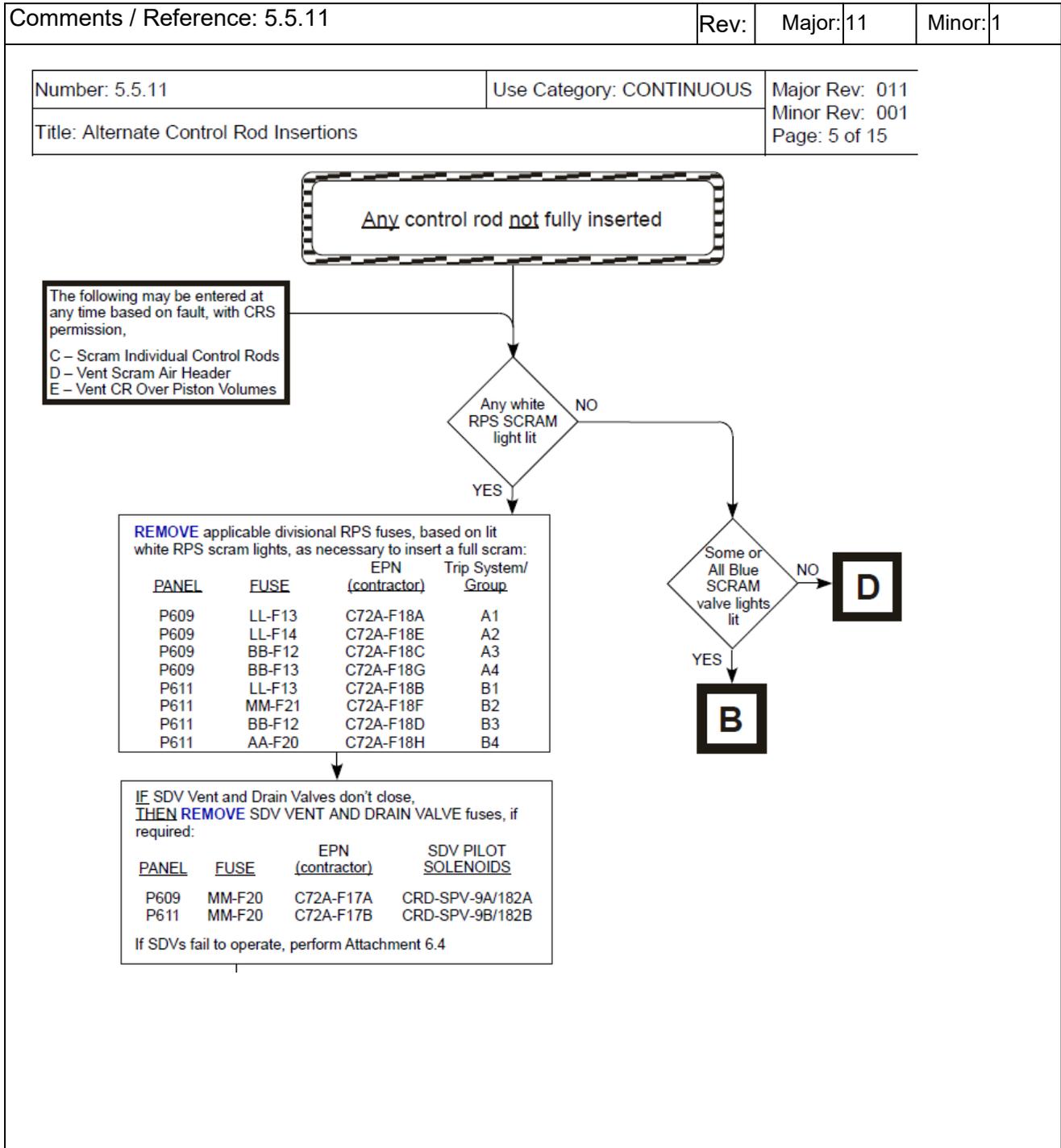
Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Question: RO-17

Justification for Cognitive Level

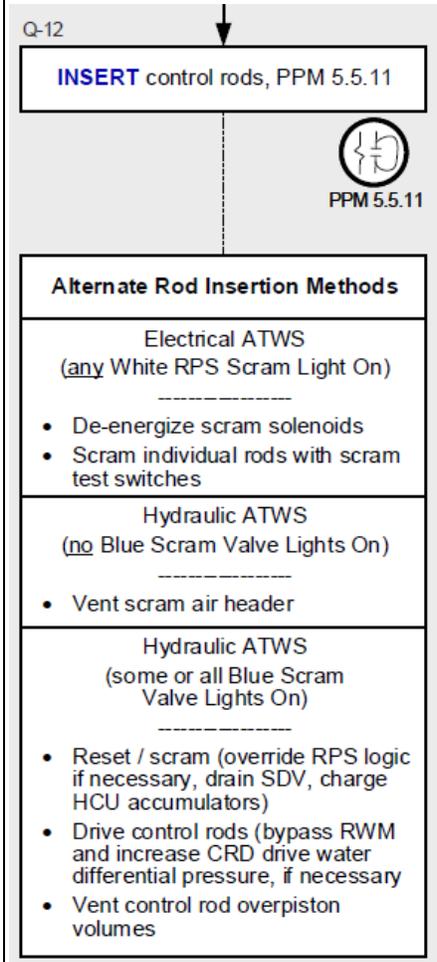
Requires student to interpret plant conditions and determine the appropriate actions.

10 CFR Part 55 Content: 55.41 10



Question: RO-17

PPM 5.1.2 Major Rev 27 – Minor – N/A



Examination Outline Cross-reference: 18

Revision: 0 Date: 08/10/22

Tier: 1

Group: 1

K/A Number: 295038 EA1.07

Level of Difficulty: 4

RO Importance Rating: 3.7

K/A Description: Ability to operate and/or monitor the following as they apply to High Offsite Radioactivity Release Rate: Control room heating, ventilation, and air conditioning

CGS is at 100% power.

- There is a confirmed release of radioactivity from the DOE 618-11 burial site.
- Control room habitability may be threatened.

In accordance with ABN-RAD-RELEASE, what actions are taken regarding the alignment of Control Room HVAC?

Control Room HVAC...

- A. Remote Air Intake No. 1 (NW) is isolated.
- B. is placed in Manual Recirculation mode of operation.
- C. air handling units are aligned to Standby Service Water cooling.
- D. Emergency Filter Unit Fans are manually placed in service.

Answer: B

K/A Match:

Manual alignment of CR HVAC that is required per ABN-RAD-RELEASE and plant conditions.

Explanation:

- A. Incorrect – Plausible since isolation of remote air intakes is allowed for maintenance activities and DOE Burial Site 618-11 is located NW of the control room. However, intake isolation is only allowed during maintenance activities and is not part of the ABN-RAD-RELEASE response.
- B. Correct – Per ABN-RAD-RELEASE section 4.2. If control room habitability is threatened the Control Room is placed in Recirculation Mode of operation per SOP-HVAC/CR-OPS thus eliminating the outside air intakes from the flowpath.
- C. Incorrect – Plausible – The standby service water system is the emergency cooling source for the WMA Air handling units. However, under the given conditions there is no valid reason or strategy associated with swapping the cooling water source.
- D. Incorrect – Plausible: Under circumstances that result in a “Z” signal (13mR detected in RB exhaust plenum) Emergency filter unit fans are automatically aligned in service with a group 3 NS4 isolations signal. Emergency units on a Z signal would be aligned to the remote air intakes.

For a situation where habitability is impacted by the DOE burial grounds (Outside source vrs inside source) this alignment would not be desired nor is it procedurally directed.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SOP-HVAC/CR-OPS	
ABN-RAD-RELEASE	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: CR-HVAC-12 Evaluate given key Control Room HVAC System parameters, if needed determine a course of action to correct or mitigate the following abnormal condition(s);
 a. Initiation/reconfiguration
 b. Extreme environmental conditions (fire, toxic gas, smoke, radiation, etc.)

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Based on plant conditions, determine strategy / configuration for CR HVAC system

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SOP-HVAC/CR-OPS	Rev:	Major:030	Minor:N/A	
Number: SOP-HVAC/CR-OPS	Use Category: CONTINUOUS	Major Rev: 030	Minor Rev: N/A	
Title: Control, Cable, and Critical Switchgear Rooms HVAC Operation		Page: 33 of 52		
5.15 <u>Isolation of Remote Air Intake No. 1 (NW)</u>				
<table border="1"> <tr> <td>NOTE: This section should only be performed as part of a maintenance activity.</td> </tr> </table>				NOTE: This section should only be performed as part of a maintenance activity.
NOTE: This section should only be performed as part of a maintenance activity.				

Comments / Reference: ABN-RAD-RELEASE	Rev:	Major:010	Minor:001
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Number: ABN-RAD-RELEASE	Use Category: CONTINUOUS	Major Rev: 010 Minor Rev: 001 Page: 5 of 7
Title: Abnormal Release <u>Of</u> Radioactivity		

4.2 **Indicated release from DOE 618-11 burial site**

- 4.2.1 **EVACUATE** the affected area as follows:
 - a. **REFER** to PPM 13.5.1 for Localized Evacuation. _____
 - b. **SOUND** the alerting tone for 5 seconds. _____
 - c. **ANNOUNCE** an evacuation or sheltering of all non-emergency personnel from the affected area. _____
 - d. **REPEAT** the above two ~~substeps~~. _____
- 4.2.2 **CLASSIFY** the emergency per PPM 13.1.1. _____
- 4.2.3 **DIRECT** Health Physics/Field Teams to perform surveys to locate areas with high airborne radioactivity. _____
- 4.2.4 IF Control Room habitability may be threatened, THEN **PLACE** the Control Room HVAC System in Recirculation Mode per SOP-HVAC/CR-OPS. _____
- 4.2.5 **PERFORM** PPM 13.8.1 to determine the site boundary dose rates. _____

Question: RO-19

Examination Outline Cross-reference: 19

Revision: 0

Date: 4/25/22

Tier: 1

Group: 1

K/A Number: 600000.AK2.06

Level of Difficulty: 3

RO Importance Rating: 3.8

K/A Description: Knowledge of the relationship between Plant Fire on Site and the following systems or components: Fire pumps

CGS is in Mode 1.

The following alarm activates:

- FCP2.5-3, SYS 57 DELUGE MAIN XFMR TR-M1 – FIRE

The CRS enters ABN-FIRE, Fire.

- Fire Header pressure is 118 psig, down slow.

An operator is performing ABN-FIRE, step 4.7, “VERIFY at least one fire pump is running”.

(1) Which fire pump will be running?

(2) What additional action should be taken in accordance with ABN-FIRE?

- A. (1) FP-P-2A
(2) Start a second fire pump.
- B. (1) FP-P-2A
(2) Dispatch safe shutdown operators to perform Appendix R fire area actions.
- C. (1) FP-P-2B
(2) Start a second fire pump.
- D. (1) FP-P-2B
(2) Dispatch safe shutdown operators to perform Appendix R fire area actions.

Answer: A

K/A Match:

Requires an understanding of fire pump automatic start feature and required actions of ABN-FIRE with respect to fire pump operation.

Question: RO-19

Explanation:

A. Correct. Fire pump FP-P-2A automatically starts at 120 psig header pressure. In accordance with ABN-FIRE, step 4.8, if an automatic sprinkler/deluge system has been activated, a second fire pump should be started.

B. Incorrect. Plausible since FP-P-2A will be running. Plausibility is enhanced since dispatching safe shutdown operators is an action required by ABN-FIRE for a fire in a However,

C. Incorrect. Plausible since starting a second fire pump is the next required action in accordance with ABN-FIRE. However, FP-P-2B automatic start setpoint is 110 psig, and header pressure is still above the setpoint.

D. Incorrect. Plausible since determining the reporting location for the Hanford Fire Department is step 4.9 in ABN-FIRE. Plausibility is enhanced if the automatic start setpoints for FP-P-2A and FP-P-2B are reversed. However, since an automatic sprinkler/deluge system has been activated, the next step is to start second fire pump (step 4.7).

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000177, Fire Protection System Description	
ABN-FIRE, Fire	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: FP-07 Given a Fire Protection System Annunciator, describe:

- a. The condition causing the annunciator
- b. Any automatic actions
- c. Any operational implications

Question Source: Bank #: Bank #

Modified Bank #: Mod Bank #. (Note changes or attach parent)

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

Justification for Cognitive Level

Requires synthesizing a knowledge of fire pump automatic start features with an understanding of actions required based on fire pump operating status.

10 CFR Part 55 Content: 55.41 7

Question: RO-19

Comments / Reference: SD000177	Rev: Major: Maj Minor: Min
<p>COLUMBIA SYSTEMS FIRE PROTECTION</p> <p>AUTO Auto start at LE 110 psig after 15 second time delay (20 seconds for loss of controller power)</p> <p>START Starts Diesel Pump</p> <p>2. Electric Fire Pump FP-P-2A (FCP-3)</p> <p>2 position momentary spring return to AUTO</p> <p>AUTO Auto start at LE 120 psig with no time delay</p> <p>START Starts pump</p> <p>3. Electric Fire Pump FP-P-2B (FCP-3)</p> <p>2 position momentary spring return to AUTO</p> <p>AUTO Auto start at LE 110 psig. (10 second time delay)</p> <p>START Starts pump</p>	<p>February 2022 SD000177, r18 nr0</p>

Question: RO-19

Comments / Reference: ABN-FIRE	Rev:	Major: 043	Minor: 001
4.7 VERIFY at least one fire pump is running.			_____
4.8 IF the fire has actuated an automatic sprinkler/deluge system, THEN START a second fire pump.			_____
<p>NOTE: An "Appendix R Fire" is defined as a fire or explosion in a Fire Protection (FP) vital area where safe shutdown system parameters indicate degraded performance or there is visible damage to vital plant equipment or cabling. If uncertain if the fire is Appendix R, then treat the fire as an Appendix R fire. Refer to Attachment 13.2 for Fire Areas identified as FP Vital Areas.</p>			
4.15 IF the fire is an Appendix R Fire, OR becomes an Appendix R Fire, THEN PERFORM Section 5.0 concurrently with Section 6.0, and Section 8.0.			_____
★ 5.1.3 DISPATCH safe shutdown Operators to perform the applicable actions in Attachment 13.1 (based on the location of the fire).			_____

Question: RO-19

FIRE AREA	PSFF DIV LOST	FIRE LOCATION	FP VITAL AREA
SW-1	1	Standby Service Water Pump House 1A	Yes
SW-2	2	Standby Service Water Pump House 1B	Yes
TG-1	N	Turbine Generator Building General Areas 441-501 (including E-W corridors adjacent RW & RB)	No
Zone TG-2	N	TB 441 East Turbine Oil Storage Room	No
TG-3	N	TB 441-518 East Stair A1	No
TG-4	N	TB 441-518 East Elevator 3	No
Zone TG-5	N	TB 441 Auxiliary Boiler Room	No
TG-6	N	TB 441-501 North Stair A3	No
Zone TG-7	N	TB 441 Hydrogen Seal Oil Room	No
TG-8	N	TB 441-501 NW Stair A4	No
Zone TG-9	N	TB 471 Turbine Oil Reservoir Room	No
Zone TG-10	N	TB 441 West Transformer Vault	No
Zone TG-11	N	TB 441 East Transformer Vault (by Aux. Boiler)	No
Zone TG-12	2	441 Corridors between RW & RB, DG & RB (includes Rooms C121, D104 and D113)	Yes

NOTE: Plant areas not listed do not contain post-fire safe shutdown equipment or cables and can be considered non-FP vital.

Examination Outline Cross-reference: 20

Revision: 1 Date: 10/25/22

Tier: 1

Group: 1

K/A Number: 70000.AK1.02

Level of Difficulty: 3

RO Importance Rating: 3.1

K/A Description: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Generator Voltage and Electric Grid Disturbances: Over-excitation

CGS is in Mode 1.

A plant startup is in progress in accordance with PPM 3.1.2, Reactor Plant Startup.

Operators are synchronizing the Main Turbine with the Grid in accordance with SOP-MT-START.

Current plant conditions:

- Main Turbine speed is 1801 rpm.
- Generator Sync Voltage is matched with Grid voltage.
- The synchroscope is moving SLOW in the FAST direction.

Prior to closing ASHE Breaker NO. 1 (BPA BKR 4885), the following annunciator alarms:

- 800.C3.8-7: MAIN GENERATOR OVEREXCITATION

Generator Overexcitation Relay, E-RLY-59/81G1, reads 1.32 V/HZ, which will cause a generator trip in approximately 10 minutes.

What is the next action that should be taken?

- A. Immediately trip the Main Generator Exciter Field Breaker, E-RMS-41.
- B. Raise Main Turbine frequency at 25 rpm/min with DEH until alarm clears.
- C. Continue with Main Generator synchronization and close ASHE Breaker NO. 1.
- D. Lower the Main Generator Exciter Voltage Adjuster, E-RMS-90V, until alarm clears.

Answer: D

K/A Match:

Requires knowledge of actions required for an overexcited generator during synchronization.

Explanation:

- A. Incorrect. Plausible since SOP-MT-START directs tripping the field breaker if there is no output voltage from the generator when the field breaker is initially shut. However, with the given conditions, generator voltage should be lowered until alarm clears.
- B. Incorrect. Plausible since raising speed will clear the alarm. However, turbine speed is normal. The cause of the alarm is generator voltage
- C. Incorrect. Plausible since this action would be correct if the generator overexcitation relay output was LT 1.32 V/HZ. However, since the relay is at 1.32 V/HZ, the cause of the alarm (high voltage) must be addressed prior to continuing synchronization.
- D. Correct. With the generator overexcitation relay at 1.32 V/HZ , the generator will trip in 10 minutes. 4.800.C3.8-7 ARP lowering the terminal voltage to clear the alarm.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
4.800.C3, Annunciator Panel Alarms	
SOP-MT-START, Main Turbine Start	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 5520 - List the Main Generator Trips (device numbers not required).

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Candidate must synthesize knowledge of main generator trips with an understanding of actions required when the main generator is overexcited

10 CFR Part 55 Content: 55.41 Choose.

Comments / Reference: 4.800.C3	Rev: Major: 021	Minor: N/A
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8-7 MAIN GENERATOR OVEREXCITATION

8-7 WINDOW	SOURCE	AUTOMATIC ACTIONS
MAIN GENERATOR OVEREXCITATION	E-RLY-59/81G1(1.26 V/HZ)	GE 1.32 V/Hz causes a generator trip after a time delay.

NOTE: This alarm is activated by a volts/hz relay and indicates either the generator voltage is high or the frequency is low.

NOTE: This alarm is expected during station backfeed per SOP-ELEC-BACKFEED.

NOTE: When synchronizing the unit, it may be necessary to have BPA lower grid voltage to minimize the occurrence of this alarm.

NOTE: If system frequency is normal, then this relay is set to alarm at 105% (26,250 V, 1.26 V/Hz) of generator rated voltage, and a generator trip at 110% (27,500 V, 1.32 V/Hz) of generator rated voltage with a 10 minute time delay. This gives the Main Generator Overexcitation alarm, which is not a problem as long as the generator voltage is not kept above the trip set point (1.32V/Hz) for over 10 minutes, without synchronizing the unit (see table below).

1. **READ** the digital V/Hz value on E-RLY-59/81G1 to confirm V/Hz alarm (Bd F).
2. **IF** this alarm occurs during Main Turbine Generator Synchronizing with Grid or Main Turbine Generator normal shutdown,
THEN VERIFY indication on E-RLY-59/81G1 is LT 1.32 V/HZ (H13-P842),
AND CONTINUE with SOP-MT-START, Main Turbine Start or SOP-MT-SHUTDOWN, Main Turbine Shutdown.
3. **DETERMINE** if the alarm is due to high voltage or low frequency.
4. **IF** the terminal voltage is high,
THEN LOWER the voltage using either the Main Generator Exciter Voltage Adjuster or the Main Generator Exciter Base Adjuster.

5. IF the frequency is low,
THEN ADJUST the speed of the Main Turbine to 1800 rpm. (N/A if generator is synchronized)

	V/Hz	Required TD	Tolerance
Alarm Point SA	1.26	60 Sec	± 1 Sec
Trip Point S1	1.32	600 Sec	± 7 Sec
Trip Point S2	1.35	300 Sec	± 4 Sec
Trip Point S3	1.38	120 Sec	± 2.2 Sec
Trip Point S4	1.41	50 Sec	± 2.5 Sec
Trip Point S5	1.44	6 Sec	± 1.18 Sec
Trip Point S6	1.50	2 Sec	± 1.06 Sec

Comments / Reference: SOP-MT-START		Rev:	034	Minor: N/A
5.5.9	CHECK Main Generator voltage.			_____
5.5.10	<u>IF no</u> Main Generator voltage (0 KV) is observed after the Main Generator Exciter Field Breaker is closed, <u>THEN PERFORM</u> the following:			
	a. IMMEDIATELY trip E-RMS-41 (Main Generator Exciter Field Breaker).			_____
	b. INVESTIGATE why no Generator voltage was observed (e.g., P.T. fuses, wiring, etc.) prior to re-exciting the Generator.			_____

Examination Outline Cross-reference: RO-21

Revision: 0

Date: 8/17/22

Tier: 1

Group: 2

K/A Number: 295009 AK1.05

Level of Difficulty: 3

RO Importance Rating: 3.5

K/A Description: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Low Reactor Water Level: Natural circulation

CGS has scrambled:

- Several rods remained full out.
- Rx Power is GT 5%.
- RPV level is -100" down slow.

(1) What EOP Chart is entered?

(2) When is RPV level restored and maintained +13" to +54"?

A. (1) PPM 5.1.2 directly.

(2) When HSBW is injected into RPV to promote natural circulation mixing of Boron.

B. (1) PPM 5.1.1 then transition to PM 5.1.2.

(2) When HSBW is injected into RPV to promote natural circulation mixing of Boron.

C. (1) PPM 5.1.2 directly.

(2) When CSBW is injected into RPV to allow concurrent depressurization of RPV.

D. (1) PPM 5.1.1 then transition to PM 5.1.2.

(2) When CSBW is injected into RPV to allow concurrent depressurization of RPV.

Answer: B

K/A Match:

Requires a knowledge that lowering RPV water level during an ATWS reduces reactor power by reducing natural circulation core flow in the core. Subsequent raising of RPV water level after HSBW addition increases natural circulation core flow and distributes Boron throughout the core.

Explanation:

- A. (1) Incorrect: Plausible because PPM 5.1.2 is the EOP chart utilized for ATWS conditions, however, it is entered from transition 4 of PPM 5.1.1.
(2) Correct: When HSBW has been injected, RPV water level is restored +13" to +54" to reestablish natural circulation flow and mix the boron throughout the core.
- B. (1) Correct: PPM 5.1.2 is entered from transition 4 of PPM 5.1.1 when it is determined that reactor power is GT 5% and control rod pattern alone does not assure reactor is shutdown. Prior to transition 4, RPV level is rapidly lowered to a best practice band of -140" to -80" to preclude neutron flux oscillations.
(2) Correct: See A(2) above.
- C. (1) Incorrect: See A(1) above.
(2) Incorrect: When CSBW has been injected, RPV depressurization and cooldown may proceed. This is independent of RPV level.
- D. (1) Correct: See B(1) above.
(2) Incorrect: See C(2) above.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.0.10A, Flowchart Training Manual Volume II, EOPs-Hot Bases	
PPM 5.1.1, RPV Control (Mode 1-3)	
PPM 5.1.2, RPV Control ATWS (Mode 1-3)	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 8017 - Given plant conditions, recognize an EOP entry condition(s) and enter the appropriate flow chart. (PPM 5.0.10)

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level:

Requires the student to synthesize a knowledge of EOP entry conditions with a knowledge of SLC injection to determine appropriate course of action to ensure Boron is evenly distributed in the core during an ATWS.

10 CFR Part 55 Content: 55.41 8

Excerpts from 5.0.10A:

Number: 5.0.10A	Use Category: INFORMATION	Major Rev: 000
Title: Flowchart Training Manual Volume II, EOPs-Hot Bases		Minor Rev: N/A Page: 78 of 247

13) Detailed instructions for injecting boron through SLC are provided in operating procedures. Injection normally continues until the entire usable contents of the SLC tank have been injected. Actions in other steps of the EOPs-Hot, however, are conditioned upon lesser amounts of boron:

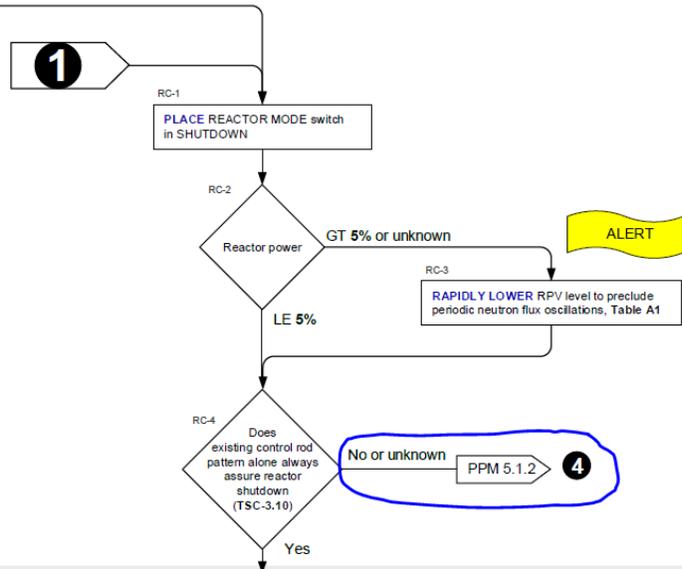
- When the Hot Shutdown Boron Weight (HSBW) has been injected, RPV water level may be restored above the low level scram setpoint in Step L-6.
- When the Cold Shutdown Boron Weight (CSBW) has been injected, RPV depressurization and cooldown may be initiated in accordance with Step P-3, PPM 5.1.5 Step P-3, or Step 5.1.6 Step F-13.

Number: 5.0.10A	Use Category: INFORMATION	Major Rev: 000
Title: Flowchart Training Manual Volume II, EOPs-Hot Bases		Minor Rev: N/A Page: 55 of 247

4) Lowering RPV water level in accordance with Override L-2 reduces reactor power by reducing natural circulation core flow. However, the reduction in natural circulation flow is also expected to reduce boron mixing efficiency when injecting boron with RCIC. While reactor power will be reduced by the flow reduction, the in-core boron concentration may remain below the value required to actually shut down the reactor. Once the Hot Shutdown Boron Weight has been injected, Step L-6 therefore restores RPV water level to reestablish natural circulation flow and distribute boron throughout the core. This strategy minimizes the integrated primary containment heatup by maintaining reactor power as low as possible during the time of boron injection.

Excerpts from PPM 5.1.1

- RPV level below +13 in.
- RPV pressure above 1060 psig
- Drywell pressure above 1.68 psig
- Both:
 - a reactor scram is required
 - AND
 - reactor power is above 5% or cannot be determined



A1	High Power ATWS Condition Actions
<p>PERFORM the following:</p> <ul style="list-style-type: none"> • INHIBIT ADS • PREVENT main turbine trip by closing RCIC-V-1 if needed • BORON INJECT REQ'D, INJECT boron into RPV using SLC • IF any main steam line is open THEN START defeating low RPV level and high steam tunnel temp MSIV isolation interlocks to maintain the main condenser as a heat sink AND ALIGN firewater cooling to CAS air compressors as necessary • IF RPV level can be determined to be above -65 in. THEN STOP and PREVENT all injection into the RPV EXCEPT from boron injection, RCIC and CRD UNTIL RPV level drops below -65 in. <div style="border: 1px solid black; padding: 5px; margin: 10px 0;"> <p style="text-align: center;">Rapid injection may cause fuel damage</p> </div> <p>MAINTAIN RPV level between -186 in. and -65 in. using one or more Preferred Injection Systems, Table L1, and Alternate Injection Subsystems, Table L2</p>	



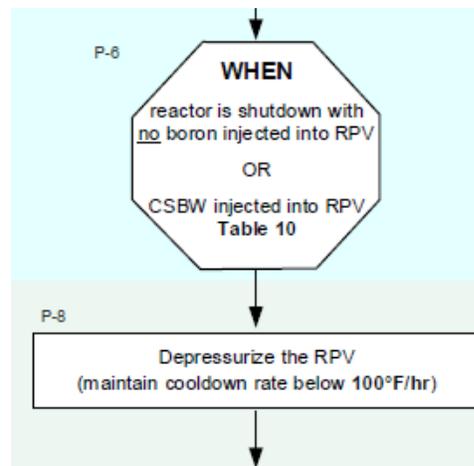
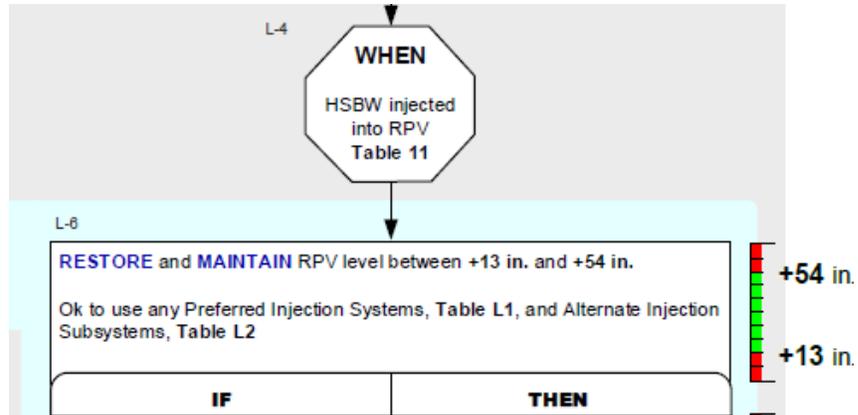
PPM 5.5.6



PPM 5.5.1

Best practice control band: -140 in. to -80 in.

Excerpts from PPM 5.1.2



Examination Outline Cross-reference: RO-22

Revision: 0

Date: 8/17/22

Tier: 1

Group: 2

K/A Number: 295010 AK2.08

Level of Difficulty: 3

RO Importance Rating: 3.2

K/A Description: Knowledge of the relationship between High Drywell Pressure and the following systems or components:
Plant ventilation systems

CGS is in Mode 1.

An event occurs that causes high drywell pressure (F Signal).

The CRS directs verifying that ABN-FAZ Automatic Actions have occurred.

When verifying, what is the expected condition of plant ventilation systems?

- SGT Systems start and maintain secondary containment pressure at (1).
- CR Emergency Filtration systems start and align (2).

A. (1) -0.8" wg
(2) in pressurization mode

B. (1) -0.8" wg
(2) to remote intakes

C. (1) -1.7" wg
(2) in pressurization mode

D. (1) -1.7" wg
(2) to remote intakes

Answer: D

K/A Match:

Requires a knowledge of the effect high drywell pressure has on plant ventilation systems.

Explanation:

- A. (1) Incorrect: Plausible because -0.8" wg is the setpoint for RB/HVAC.
(2) Incorrect: Plausible because pressurization mode is a valid mode of operation for the CR/HVAC and is directed by ABN-FIRE, however, it must be aligned manually.
- B. (1) Incorrect: See A(1) above.
(2) Correct: CR Emergency Filtration systems start and align to remote intakes.
- C. (1) Correct: SGT Systems start and maintain secondary containment pressure at -1.7" wg.
(2) Incorrect: See A(2) above.
- D. (1) Correct: See B(1) above.
(2) Correct: See C(2) above.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-FAZ, FAZ	
SOP-HVAC/CR-OPS, Control, Cable, Critical Switchgear Rms Ops	
SOP-HVAC/RB-OPS, Rx Bldg Ventilation System Ops	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 6916 - Given an F, A or Z Signal Actuation identify those automatic actions that may have occurred. [ABN-FAZ]

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level:

Requires the student to know ABN-FAZ automatic actions.

10 CFR Part 55 Content: 55.41 7

Excerpt from ABN-FAZ, Attachment 7.3, Automatic Actions:

Number: ABN-FAZ	Use Category: CONTINUOUS	Major Rev: 021
Title: FAZ		Minor Rev: N/A
		Page: 30 of 30

Equipment	F	A	Z	✓
Div 1 and 2 SW start (H13-P840 & H13-P820)	•			
RCC pumps trip (H13-P840 & H13-P820)	•	•		
Div 1 and 2 Diesel Generators start (H13-P800)	•			
CB-75/72, CB-85/82 trip(H13-P800)	•	•		
MC-7C, 7E, 8C, and 8E trip: <ul style="list-style-type: none"> • RWCU pumps trip • MSL Tunnel Cooling fan trips • FPC Suppression Pool Cleanup mode isolates and pump trips • RWHV Exhaust Fans trip • CJW-P-1A trips 	•	•		
Reactor Building EDR and FDR sump pump discharge headers isolate (H13-P632)	•	•	•	
TIP withdraws and isolates(H13-P607)	•	•		
SGT Systems start and maintain secondary containment pressure at -1.7" wg.(H13-P811 & H13-P827)	•	•	•	
RBHV isolates and fans trip (H13-P812)	•	•	•	
Reactor Building Emergency Room Coolers start (H13-P812)	•	•	•	
Reactor Building Lighting quenches (H13-P812)	•	•	•	
CSP/CEP isolate (H13-P813)	•	•	•	
Containment Nitrogen makeup isolates (H13-P813)	•	•	•	
CR and TSC Emergency Filtration systems start and align to remote intakes(H13-P826)	•	•	•	
RCC Containment isolation (H13-P825)	•	•		
CMS-SR-20 and 21 isolate (BD-RAD-22 & 23)	•	•		

END

Attachment 7.3, Automatic Actions

Excerpt from SOP-HVAC-OPS:

Number: SOP-HVAC/CR-OPS	Use Category: CONTINUOUS	Major Rev: 030
Title: Control, Cable, and Critical Switchgear Rooms HVAC Operation		Minor Rev: N/A Page: 26 of 52

5.12 Initiating Control Room HVAC Manual Pressurization Mode

5.12.1 IF starting both trains of Control Room ventilation,
THEN VERIFY both Remote Air Intakes are unisolated.

<u>NOTE:</u> Unless otherwise indicated, all control switches and annunciators are located on H13-P826.
<u>NOTE:</u> Both trains of Control Room ventilation may be placed in manual pressurization mode at the same time, as determined by the CRS/Shift Manager, or as directed by ABN-FIRE.

Excerpt from SOP-HVAC-OPS:

Number: SOP-HVAC/RB-OPS	Use Category: CONTINUOUS	Major Rev: 012
Title: Reactor Building Ventilation System Operation		Minor Rev: N/A Page: 8 of 26

<u>NOTE:</u> REA-DPR-1A(B) should be controlled at -0.8" WC to ensure the -0.25" WC limit is not exceeded.
--

5.1.5 ADJUST Reactor Building pressure controller to stabilize Reactor Building pressure at approximately -0.8" WC. (N/A controller not used)

- REA-DPIC-1A
- REA-DPIC-1B

Examination Outline Cross-reference: RO-23

Revision: 0

Date: 8/18/22

Tier: 1

Group: 2

K/A Number: 295012 AK3.01

Level of Difficulty: 3

RO Importance Rating: 3.8

K/A Description: Knowledge of the reasons for the following responses or actions as they apply to High Drywell
Temperature: Increased drywell cooling

CGS is in Mode 1.

An event causes Drywell temperature to rise.

The CRS enters PPM 5.2.1, Primary Containment control.

When is drywell spray required to be initiated and what is the reason for this limit?

Drywell sprays should be initiated prior to reaching (1) in the drywell to prevent (2).

- A. (1) 330°F
(2) exceeding ADS design temperature
- B. (1) 330°F
(2) primary containment damage from chugging
- C. (1) 340°F
(2) exceeding ADS design temperature
- D. (1) 340°F
(2) primary containment damage from chugging

Answer: A

K/A Match:

Requires a knowledge of the reason to initiate drywell sprays to increase drywell cooling when high drywell temperatures occur.

Explanation:

- A. Correct. In accordance with step DT-3 of PPM 5.2.1, drywell sprays are initiated prior to reaching 330°F. PPM 5.0.10A states that drywell sprays are initiated at this point prior to the ADS design temperature.
- B. Incorrect. Plausible since (1) is correct. However, PPM 5.0.10A states that drywell sprays are initiated at this point prior to the ADS design temperature.
- C. Incorrect. Plausible since (2) is correct. However, drywell sprays are initiated prior to reaching 330°F.
- D. Incorrect. Plausible since 340°F is the primary containment design temperature. However, drywell sprays are initiated prior to reaching 330°F.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.2.1, Primary Containment Control (Mode 1-3)	
PPM 5.0.10A, Flowchart Training Manual Volume II, EOPs-Hot Bases	
PPM 4.601.A12, H13-P601.A12 Annunciator Panel Alarms	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 8318 - Describe the reason for attempting to maintain drywell temperature below 330 °F. (PPM 5.2.1)

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

Question History: Last NRC Exam: Exam year or N/A

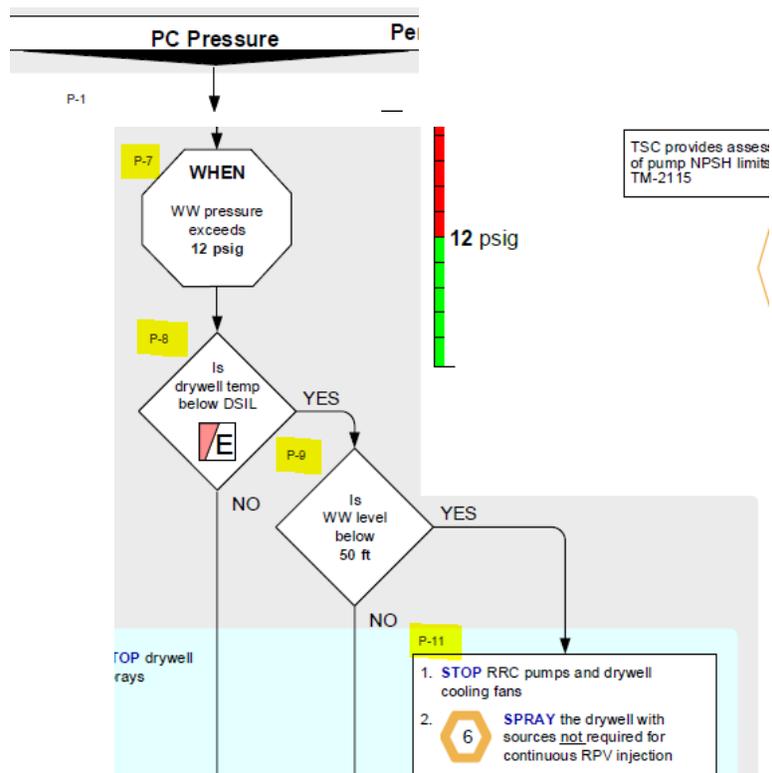
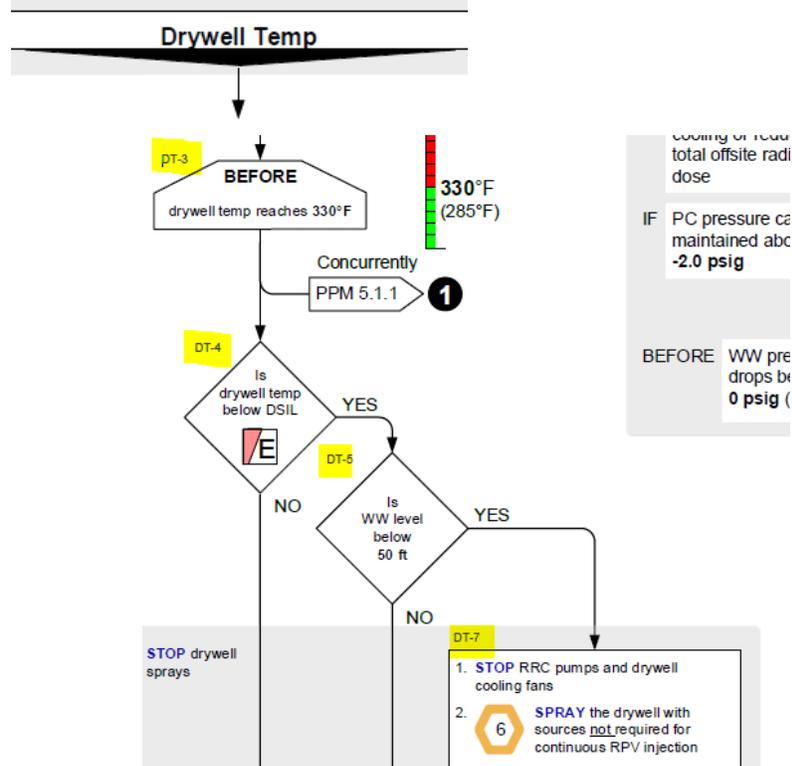
Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level:

Requires the student to know the reason to initiate drywell sprays when high drywell temperatures occur.

10 CFR Part 55 Content: 55.41 5

Excerpts from PPM 5.2.1



Excerpts from PPM 5.0.10A:

Number: 5.0.10A	Use Category: INFORMATION	Major Rev: 000
Title: Flowchart Training Manual Volume II, EOPs-Hot Bases		Minor Rev: N/A
		Page: 197 of 247

c. Steps DT-3 through DT-7:

- 1) If operation of all available drywell cooling is unable to stop increasing drywell temperature, drywell sprays are initiated to effect the required drywell temperature reduction (status of governing plant conditions permitting) before the ADS design temperature is reached (330°F).

Number: 5.0.10A	Use Category: INFORMATION	Major Rev: 000
Title: Flowchart Training Manual Volume II, EOPs-Hot Bases		Minor Rev: N/A
		Page: 209 of 247

h. Steps P-8 through P-11:

- 1) Drywell sprays are initiated to preclude primary containment damage from chugging—the cyclic condensation of steam at the downcomer openings. Scale model tests have demonstrated that chugging can occur only if the drywell atmosphere is mostly steam, with the noncondensable fraction less than 1%. Chugging can thus be prevented by operating Drywell sprays; the resulting drywell pressure reduction opens the Wetwell-to-Drywell vacuum breakers, drawing noncondensibles from the Wetwell back into the Drywell. Initiating Drywell sprays when suppression chamber pressure exceeds the WSIP ensures the drywell noncondensable content remains above the chugging threshold with an added conservatism. The WSIP is the lowest Wetwell pressure which can occur when 95% of the noncondensibles in the Drywell have been transferred to the Wetwell.

Comments / Reference: 4.601.A12 Rev: Major: 029 Minor: 001

Excerpt from 4.601.A12 1-3:

Number: 4.601.A12	Use Category: CONTINUOUS	Major Rev: 029
Title: 601.A12 Annunciator Panel Alarms		Minor Rev: 001
		Page: 10 of 34

1-3 DRYWELL/SUPPRESSION POOL TEMPERATURE HIGH
(CONTINUED FROM PREVIOUS PAGE)

Alarm Group 5 - AVG D/W TEMP HI (POST ACCIDENT) (Pt 110 - 333.4°F)

NOTE: This alarm indicates approaching the maximum design temperature for containment structure (340°F).

Question: RO-24

Examination Outline Cross-reference: RO-24

Revision: 0

Date: 8/18/22

Tier: 1

Group: 2

K/A Number: 295013 2.1.7

Level of Difficulty: 3

RO Importance Rating: 4.4

K/A Description: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation: High Suppression Pool Water Temperature

CGS is at 100% power.

- Wetwell temperature is increasing steadily.
- Both Annunciator P601.A11 1-3, Drywell/Supp Pool Temp High, and P601.A12 1-3, Drywell/Supp Pool Temp High, alarm.

What operational decisions must be made based on wetwell temperature trend?

- Enter PPM 5.2.1, Primary Containment Control (Mode 1-3), when wetwell temperature is above (1).
- Scram before wetwell temperature exceeds (2).

A. (1) 80°F
(2) 110°F

B. (1) 80°F
(2) 135°F

C. (1) 90°F
(2) 110°F

D. (1) 90°F
(2) 135°F

Answer: C

K/A Match:

Shows ability to evaluate increasing high suppression pool temperature and based on that trend, make operational judgements on when to enter EOPs and perform plant scram.

Question: RO-24

<p>Explanation:</p> <p>A. <u>(1) Incorrect: Plausible because 80°F is in the tolerance band for when the annunciators actuate. The annunciators, however, are not marked with an orange triangle indicating potential EOP entry.</u> <u>(2) Correct: The temperature value of 110°F is the Wetwell temperature at which Technical Specifications require a reactor scram. In Step WT-4 of PPM 5.2.1, the logic term "BEFORE" permits initiation of a reactor scram at any temperature up to 110°F. The symptom-based approach to emergency response provides the latitude of delaying reactor scram until Wetwell temperature reaches ~110°F or initiating a reactor scram well before Wetwell temperature reaches 110°F.</u></p> <p>B. <u>(1) Incorrect: See A(1) above.</u> <u>(2) Incorrect: Plausible because above 135°F is the drywell temperature for entry into PPM 5.2.1.</u></p> <p>C. <u>(1) Correct: Above 90°F is the wetwell temperature for entry into PPM 5.2.1.</u> <u>(2) Correct: See A(2) above</u></p> <p>D. <u>(1) Correct: See C(1) above.</u> <u>(2) Incorrect: See B(2) above.</u></p>

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.2.1, Primary Containment Control (Mode 1-3)	
PPM 5.0.10A, Flowchart Training Manual Volume II, EOPs-Hot Bases	
PPM 4.601.A12, H13-P601.A12 Annunciator Panel Alarms	
CMS-TR-5 Instrument Master Data Sheet	

Proposed references to be provided during examination: N/A

Learning Objective: 8300 - Describe the reason for entering PPM 5.1.1, "RPV Control", before wetwell temperature reaches 110°F. (PPM 5.2.1))

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level:

Requires the student to synthesize an understanding of an increasing trend and based on that trend, make operational judgements on when to enter EOPs and perform plant scram.

10 CFR Part 55 Content: 55.41 5

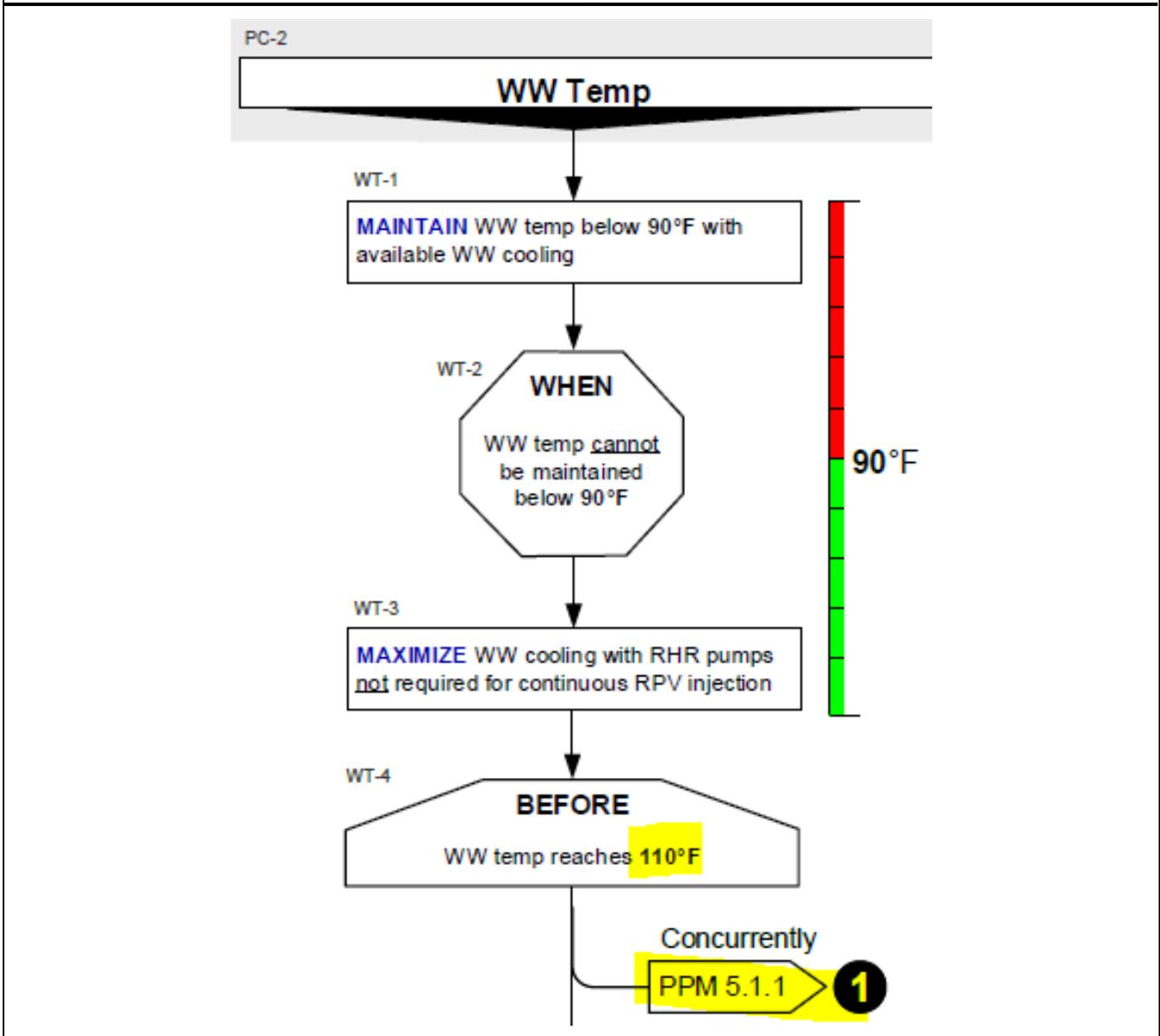
Question: RO-24

Comments / Reference: PPM 5.2.1	Rev:	Major: 029	Minor: N/A
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Excerpts from PPM 5.2.1:

Entry Conditions:

- WW temp above 90°F
- Drywell temp above 135°F
- Drywell pressure above 1.68 psig
- WW level above +2 in. or below -2 in.
- PC hydrogen above 3.56%



Question: RO-24

Comments / Reference: PPM 5.0.10A	Rev:	Major: 000	Minor: N/A
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Excerpts from PPM 5.0.10A:

Number: 5.0.10A	Use Category: INFORMATION	Major Rev: 000
Title: Flowchart Training Manual Volume II, EOPs-Hot Bases		Minor Rev: N/A
		Page: 190 of 247

3.8.3 Flowchart Technical Discussion

a. Entry:

The conditions which require entry into PPM 5.2.1 are symptoms of an emergency or conditions which, if not corrected, could degrade into an emergency. This set of entry conditions is sufficient to assure the flowchart will be entered for transients and accidents which are within the design and licensing basis for Columbia, and for additional events which have been evaluated as significant to emergency response.

b. Entry conditions based on plant parameters are the following:

- Bulk Wetwell temperature above 90°F: This temperature is the most limiting Wetwell temperature LCO.
- Average drywell temperature above 135°F: This temperature is the most limiting drywell temperature LCO.

Number: 5.0.10A	Use Category: INFORMATION	Major Rev: 000
Title: Flowchart Training Manual Volume II, EOPs-Hot Bases		Minor Rev: N/A
		Page: 195 of 247

d. Step WT-4:

- 1) The temperature value of 110°F is the Wetwell temperature at which Technical Specifications require a reactor scram.
- 2) The logic term "BEFORE" permits initiation of a reactor scram at any temperature up to 110°F. The symptom based approach to emergency response provides the latitude of delaying reactor scram until Wetwell temperature reaches ~110°F or initiating a reactor scram well before Wetwell temperature reaches 110°F.
- 3) Entry to PPM 5.1.1 assures that, if possible, the reactor is scrammed and shutdown before the requirement for boron injection is reached. Entry to the RPV Control procedures is explicitly stated because conditions requiring entry to PPM 5.2.1 do not necessarily require entry to PPM 5.1.1. Therefore, a scram may not have yet been initiated.
- 4) Directing that PPM 5.1.1 be entered rather than explicitly stating "initiate a reactor scram" in the Wetwell temperature control flowpath coordinates actions currently being performed if the RPV Control procedures have already been entered.

Question: RO-24

Comments / Reference: 4.601.A12	Rev:	Major: 029	Minor: 001
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Excerpt from 4.601.A12 1-3:

Number: 4.601.A12	Use Category: CONTINUOUS	Major Rev: 029
Title: 601.A12 Annunciator Panel Alarms		Minor Rev: 001
		Page: 7 of 34

1-3 DRYWELL/SUPPRESSION POOL TEMPERATURE HIGH
(CONTINUED FROM PREVIOUS PAGE)

Alarm Group 2 - AVG SUPP POOL TEMP HI (UPPER LVL) (Pt 220 - 81.4°F)

1. DETERMINE which input(s) caused the alarm.
2. COMPARE the Average Suppression Pool Temperature on CMS-TR-5 with the associated reading on CMS-TR-6.

NOTE: Upper temperature elements will be uncovered if Suppression Pool level is at approximately 30'3" on wide range or -9 inches on narrow range.

3. IF temperature alarm is due to uncovered upper thermocouple, THEN EVALUATE Suppression Pool temperature using a lower thermocouple.
4. IF the temperature rise has been gradual, THEN EVALUATE both CMS-TR-5 and CMS-TR-6 points 220 and A02 to determine if thermal stratification exists.
5. IF the upper and lower thermocouple differential temperatures will support lowering the average Suppression Pool temperature, THEN PLACE Fuel Pool Cooling in Suppression Pool Cleanup per SOP-FPC-SPC, FPC Suppression Pool Operations.

Comments / Reference: CMS-TR-5 IMDS	Rev:	Major: 10/25/17	Minor:
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Excerpt from CMS-SR-5 IMDS - Setpoint = 81.4°F – see above:



Columbia Generating Station
EC 16790 INSTRUMENT MASTER DATA SHEET DIC: 1801.1

Page 1 of 9 Design Engineer: *[Signature]* System Engineer: *[Signature]* 10/25/2017

EPN: CMS-TR-5	SERVICE: Drywell/Suppression Pool Tem/Avg. Temp.
Loop: See Addenda	Rev. #: 18 Date: 10/25/2017
Work Order:	Input From : See Addenda
Cal Procedure: ISP-CMS-X310	Output To: Alarm

Process Input		Output (± tolerance)		
Cal. Input		Desired		
VDC	TYPE °T	Chart Paper	T/C Input Indication	VDC Input Indication
1	50 °F	50 ± 3.5°F	50 ± 1.4°F	50 ± .3°F
2	137.5 °F	137.5 ± 3.5°F	137.5 ± 1.4°F	137.5 ± .3°F
3	225 °F	225 ± 3.5°F	225 ± 1.4°F	225 ± .3°F
4	312.5 °F	312.5 ± 3.5°F	312.5 ± 1.4°F	312.5 ± .3°F
5	400 °F	400 ± 3.5°F	400 ± 1.4°F	400 ± .3°F

Bi-Stable Parameters - Limits

Switch ID	Setpoint/Tolerance	Action At Setpoint	Reset/ Tolerance	Logic Affected
Addenda 19	Setpoint ± 1.4°F Setpoint ± 0.003 VDC	Addenda 19	Addenda 19	P601.A12-1.3 Alarm

Question: RO-25

Examination Outline Cross-reference: RO-25

Revision: 0

Date: 8/19/22

Tier: 1

Group: 2

K/A Number: 295014 AA1.11

Level of Difficulty: 3

RO Importance Rating: 3.9

K/A Description: Ability to operate and/or monitor the following as they apply to Inadvertent Reactivity Addition: Feedwater system

CGS is at 100% power.

Annunciators H13-P840.A2 7-2 HIGH PRESSURE HEATER 6A LEVEL HIGH TRIP, and 8-2, HIGH PRESSURE HEATER 6A LEVEL HIGH, alarm.

- All automatic actions occur.

What procedure is entered and what action is taken?

Enter (1) and take action to (2).

- A. (1) ABN-CORE
(2) SCRAM the Reactor per PPM 3.3.1
- B. (1) ABN-POWER
(2) SCRAM the Reactor per PPM 3.3.1
- C. (1) ABN-CORE
(2) REDUCE reactor power with RRC flow to 74 Mlbm/hr core flow at 5% per minute.
- D. (1) ABN-POWER
(2) REDUCE reactor power with RRC flow to 74 Mlbm/hr core flow at 5% per minute.

Answer: D

K/A Match:

Shows an understanding of when an inadvertent reactivity addition has occurred due to the feedwater system and the ability to select appropriate procedure and to operate equipment to mitigate the inadvertent reactivity addition.

Question: RO-25

Explanation: A #6 Feed Water trip causes lower temperature feedwater to be injected into the reactor. The inadvertent reactivity addition will cause an increase in reactor power.

A. (1) Incorrect: Plausible because annunciator procedure 4.840.A2 7-2 contains instructions to enter ABN-CORE if reactor power is starts to oscillate. The stem does not state that reactor power is starting to oscillate.
(2) Incorrect: Plausible because both ABN-CORE and ABN-POWER contain instructions to scram the reactor under certain conditions. Loss of a #6 feed water heater is not one of the conditions.

B. (1) Correct: #5 or #6 Feed Water Heater Trip Indication is an entry condition to ABN-POWER.
(2) Incorrect: See A(2) above.

C. (1) Incorrect: See A(1) above.
(2) Incorrect: Plausible because ABN-CORE does have instructions to adjust flow under certain conditions. Loss of a #6 feed water heater is not one of the conditions.

D. (1) Correct: See B(1) above.
(2) Correct: In ABN-POWER, a single #6 feed water heater trip is considered an unplanned feedwater temperature reduction. At 100% power, RRC flow is ~108 Mlbm/hr. Per ABN-POWER, if RRC flow is GT 74 Mlbm/hr, REDUCE reactor power with RRC flow to 74 Mlbm/hr core flow at 5% per minute. .

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 4.840.A2, H13-P840.A2 Annunciator Panel Alarms	
ABN-POWER, Unplanned Reactor Power Change	
ABN-CORE, Unplanned Core Operating Conditions	
ABN-FWH-HI/LEVEL-TRIP, Feedwater Heater High Level Trip	

Proposed references to be provided during examination: N/A

Learning Objective: 6742 -Given plant annunciation and indications, evaluate conditions for entry into ABN-POWER due to an unplanned loss of feedwater heating.

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level:

Requires the student to synthesize an understanding of when an inadvertent reactivity addition has occurred and the ability to select appropriate procedure and to operate equipment to mitigate the inadvertent reactivity addition.

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Question: RO-25

Comments / Reference: 4.840.A2	Rev:	Major: 019	Minor: N/A
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Excerpts from PPM 4.840.A2:

Number: 4.840.A2	Use Category: CONTINUOUS	Major Rev: 019
Title: 840.A2 Annunciator Panel Alarms		Minor Rev: N/A Page: 40 of 54

7-2 HIGH PRESSURE HEATER 6A LEVEL HIGH TRIP

7-2 WINDOW	SOURCE	AUTOMATIC ACTIONS
HP HEATER 6A LEVEL HI TRIP	HD-LS-6AHT (5 inches below heater centerline)	<ul style="list-style-type: none"> BS-V-6A closes (HP Htr. 6A extraction steam isol) BS-V-7A closes (HP Htr. 6A non-return valve) BS-DV-6A opens (HP Htr. 6A extraction steam dump valve) HD-V-31A and 31B close (MSR drain tanks 3A and 3B non-return valves) HD-V-32A and 32B close (MSR drain tanks 2A and 2B non-return valves)
	OR	
	HD-LS-6AIT (5 inches below heater centerline)	

NOTE: These controllers transfer to the automatic state after a voltage transient (spike) on the plant electrical buses.

1. VERIFY all automatic actions have occurred.
2. IF in FFTR (Final Feedwater Temperature Reduction, PPM 3.1.11), THEN CLOSE the following:
 - RFW-V-109
 - COND-V-144
3. REFER to ABN-POWER, Unplanned Reactor Power Change.
4. IF Reactor power starts to oscillate, THEN REFER to ABN-CORE, Unplanned Core Operating Conditions.
5. REFER to ABN-FWH-HI/LEVEL-TRIP, Feedwater Heater High Level Trip.

Comments / Reference: ABN-FWH-HI/LEVEL-TRIP	Rev:	Major: 011	Minor: 002
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Excerpts from PPM ABN-FWH-HI/LEVEL-TRIP:

Number: ABN-FWH-HILEVEL/TRIP	Use Category: CONTINUOUS	Major Rev: 011
Title: Feedwater Heater High Level Trip		Minor Rev: 002 Page: 8 of 86

5.0 BASES

1.0 A Feedwater Heater Hi Level Trip results in isolating bleed steam to the affected feedwater heater, causing a lower temperature of the feedwater injected into the reactor. This would cause an increase in reactor power. The severity of this event worsens with higher plant power and the condensate flowpath sequence of the heater lost. The later in the heating process that the heating loss occurs the more severe the effect. A loss of heating to the 6th stage heaters cannot be compensated and will result in lower feedwater injection temperatures. The assumption of 5th stage heating load by the 6th stage is possible at below rated conditions when a loss of 5th stage heating occurs. Isolation of bleed steam to the one of the earlier stages of feedwater heating will stress the later stages but may be within the capability of the later stages to maintain the heating load.

Question: RO-25

Comments / Reference: ABN-POWER	Rev:	Major: 018	Minor: N/A
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Excerpts from ABN-POWER:

Number: ABN-POWER	Use Category: CONTINUOUS	Major Rev: 018
Title: Unplanned Reactor Power Change		Minor Rev: N/A
		Page: 3 of 17

1.0 **ENTRY CONDITIONS**

1.1 Unplanned Reactor Power Change

A noticeable unplanned reactor power change is indicative of changing plant conditions. This procedure should not be entered for normal fluctuations of reactor power observed on the APRMs due to fluctuations in the boiling boundary. This procedure should also not be entered for a reactor power change associated with a reactor scram. This procedure should not be entered for a planned reactor power change directed by a Reactivity Control Plan (RCP). Typically, small power changes are not identifiable on the APRMs but can be identified on PPCRS. Power changes can be caused by a number of things including:

- RPV inlet temperature change
- #5 or #6 Feed Water Heater Trip Indication

Number: ABN-POWER	Use Category: CONTINUOUS	Major Rev: 018
Title: Unplanned Reactor Power Change		Minor Rev: N/A
		Page: 9 of 17

4.3 Unplanned Feedwater Temperature Reduction

CAUTION

Operating with feedwater heaters out of service raises the possibility of thermal hydraulic oscillations in the AIA (see ABN-CORE).

CAUTION

The rate of power reduction should be limited to prevent tripping the main turbine and reactor feedwater pumps on RPV level 8 (+54.5").

NOTE: The limit of 6°F on feedwater temperature is not inclusive of temperature changes due to operator action following entry into the procedure. The 6°F change in inlet temperature is in reference to a single event that causes a change in inlet temperature of GE 6°F.

4.3.1 IF feedwater inlet temperature experiences an unplanned drop of GE 6°F, OR a single number 5 or 6 Feed Water Heater trips, THEN PERFORM the following:

- a. IF RRC flow is GT 74 Mlbm/hr, THEN REDUCE reactor power with RRC flow to 74 Mlbm/hr core flow at 5% per minute.

Question: RO-26

Examination Outline Cross-reference: 26

Revision: 0

Date: 5/23/22

Tier: 1

Group: 2

K/A Number: 295022.AA2.01

Level of Difficulty: 3

RO Importance Rating: 4.0

K/A Description: Ability to determine and/or interpret the following as they apply to Loss of Control Rod Drive Pumps:
Accumulator pressure

CGS is in Mode 1.

The following events occur:

<u>Time</u>	<u>Event</u>
1159	The operating Control Rod Drive (CRD) pump, CRD-P-1A, trips.
1200	Charging Header Pressure (CRD-PI-15) is 938 psig, down slow.
1203	Control Rod Scram Accumulator CRD-HCU-3027 is declared inoperable due to low accumulator pressure.
1205	Control Rod Scram Accumulator CRD-HCU-4635 is declared inoperable due to low accumulator pressure.
1230	Charging Header Pressure (CRD-PI-15) is 895 psig, down slow.

Based on this timeline, what is the earliest time that a reactor scram is required to be initiated to meet technical specification requirements?

- A. 1220
- B. 1223
- C. 1225
- D. 1230

Answer: C

K/A Match:

Requires knowledge of required actions for low accumulator pressure on a loss of CRD pumps.

Question: RO-26

Explanation:

- A. Incorrect. Plausible if it is believed that a reactor scram must be initiated 20 minutes after the CRD pump trips with two inoperable scram accumulators. However, the 20 minute time limit starts when the second scram accumulator becomes inoperable with steam dome pressure ≥ 900 psig.
- B. Incorrect. Plausible since a reactor scram is required 20 minutes after 2 control rod scram accumulators become inoperable with CRD d/p below 940 psig. However, the second control rod scram accumulator does not become inoperable until 1205. Therefore, a scram is not required at 1223.
- C. Correct. Limiting Condition for Operation (LCO) 3.1.5, Control Rod Scram Accumulators, Condition B states that with “Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 900 psig”, operators are required to “Restore charging water header pressure to ≥ 940 psig” with a required completion time (B.1) of “20 minutes from discovery of Condition B concurrent with charging water header pressure < 940 psig.” For the conditions given, the clock for completing the required actions of Condition B starts at 1205, when the second scram accumulator becomes inoperable. Condition D of the LCO requires placing the Reactor Mod Switch in SHUTDOWN immediately if the completion time for required action B.1 is not met. Therefore, the earliest time when a scram is required is 20 minutes from entering Condition B, or 1225.
- D. Incorrect. Plausible since a reactor scram is required if steam dome pressure is < 900 psig and a single scram accumulator is inoperable with charging water pressure < 940 psig. However, the question stem states that at 1230, charging water pressure is < 900 psig, NOT steam dome pressure. With reactor power at 100%, steam dome pressure is well above 900 psig.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
TS LCO 3.1.5, Control Rod Scram Accumulators	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 11768 – Discuss when the reactor must be manually scrammed due to Control Rod Drive Hydraulic system malfunctions

Question Source: Bank #: 64695
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: 2021

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Examinee must synthesize a knowledge of the < 1hour actions required by LCO 3.1.5 along with an understanding of when the plant conditions given in the question stem meet the requirements to take Tech Spec actions.

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Question: RO-26

Comments / Reference: TS LCO 3.1.5	Amendment: 265		
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Control Rod Scram Accumulators
3.1.5

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 \geq 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

Control Rod Scram Accumulators
3.1.5

ACTION

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 900 psig.	C.1 Verify the associated control rod is fully inserted.	Immediately upon discovery of charging water header pressure < 940 psig 1 hour
	<u>AND</u> C.2 Declare the associated control rod inoperable.	
D. Required Action B.1 or C.1 and associated Completion Time 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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each control rod scram accumulator pressure is \geq 940 psig.	In accordance with the Surveillance Frequency Control Program

Examination Outline Cross-reference: 27

Revision: 0

Date: 5/26/22

Tier: 2

Group: 1

K/A Number: 203000.K5.02

Level of Difficulty: 4

RO Importance Rating: 4.2

K/A Description: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the RHR/LPCI: Injection Mode: core cooling methods

CGS is in Mode 1.

A large LOCA occurs in containment.

An Emergency Depressurization was performed at -100" on RPV level.

Current plant conditions:

- RPV level: -140 inches, down 10"/min.
- Drywell Pressure: 21psig.
- Wetwell Pressure 16psig.
- Wetwell Temperature is 97F and up slow.

The following system lineups exist:

- LPCS-P-1 is injecting @4000gpm due to LPCS-V-5 being stuck mid position.
- RHR-P-2B is spraying the Wetwell and cooling the suppression pool.
- RHR-P-2A is spraying the Drywell.
- RHR-P-2C is injecting into the core.
- HPCS-P-1 tripped on overcurrent.

3 minutes later, what is the next action to take to maintain adequate core cooling?

- A. Maintain RHR systems in current configuration. Adequate Core Cooling will be maintained with LPCS-P-1 in spray cooling.
- B. Secure all sprays and suppression pool cooling. Line up RHR-P-2A and RHR-P-2B to inject per PPM 5.5.26 via the Shutdown cooling line to prevent fuel damage.
- C. Secure all sprays and suppression pool cooling. Use all available systems to keep RPV level above -161".
- D. Secure Drywell sprays and use RHR-P-2A to assist with RPV level. Maintain RHR-P-2B in current lineup for containment parameters.

Answer: C

K/A Match:

Requires knowledge of RHR pump NPSH limits when used to inject into the RPV.

Explanation:

- A. Correct. In accordance with PPM 5.2.1, Table 18, the NPSH limit of pump (1) is 225°F. Pumps (2) and (3) have NPSH temperature limits below the given wetwell temperature. Therefore, they cannot be used without exceeding this limit.
- B. Incorrect. Plausible since Wetwell level is within the limit for the RCIC pump. Plausibility is enhanced since the RCIC pump could be operated with its suction lined up to the CST. However, conditions given in the question stem state that the RCIC pump suction is lined up to the Wetwell. Additionally, RCIC pump NPSH limit is below the Wetwell temperature given.
- C. Incorrect. Plausible since (1) is correct. However, pump (3) NPSH limit is 200°F, which is below the Wetwell temperature given in the question stem.
- D. Incorrect. Plausible since Wetwell level is within the limits for both pump (2) and (3). However, the NPSH temperature limits for both pumps is below the Wetwell conditions given in the question stem.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.2.1, Primary Containment Control	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: RHR-12 Evaluate given key Residual Heat Removal System parameters, if needed determine a course of action to correct or mitigate the following abnormal condition(s): a. Inadequate net positive suction head

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize knowledge of the NPSH limiting parameters (both Wetwell Temperature and Level) for RHR pumps along with the limits for other ECCS pumps

Comments / Reference: PPM 5.2.1	Rev:	Major: 29	Minor: N/A
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18	Vortex and NPSH Limits		
<u>System</u>	<u>WW Level (ft)</u>	<u>WW Temp (°F)</u>	
RCIC with suction from WW	17.5	210	
LPCS	14.5	225	
RHR-C	14.5	225	
RHR-A, B	14.0	225	
HPCS with suction from WW	5.5	200	

Examination Outline Cross-reference: RO-28

Revision: 0

Date: 8/19/22

Tier: 2

Group: 1

K/A Number: 205000 A2.08

Level of Difficulty: 4

RO Importance Rating: 4.0

K/A Description: Ability to (a) predict the impacts of the following on the Shutdown Cooling System (RHR Shutdown Cooling Mode) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of heat exchanger cooling

CGS is in Mode 5.

RHR-B is in fuel pool cooling assist mode with suction from the Skimmer Surge tanks.

Plant conditions:

- Fuel pool gates are removed.
- Fuel pool level is GE 22' above irradiated fuel.
- Division 1 is OOS for maintenance for the next 3 days.

An event causes the following:

- RHR-V-48B receives a LPCI initiation signal.
- An NS⁴ Group 6 isolation signal is generated.

What is the impact to shutdown cooling and what action is required to mitigate the consequences?

- A. A loss of shutdown cooling has occurred. Manually open RHR-V-8 and RHR-V-9.
- B. A loss of shutdown cooling has occurred. Evacuate all personnel from the drywell and wetwell.
- C. The reactor cavity and the fuel pool will heat up. Throttle RHR-V-48B closed after 10 minutes to increase heat exchanger cooling.
- D. The reactor cavity and the fuel pool will cool down. Throttle RHR-V-48B open after 10 minutes to decrease heat exchanger cooling.

Answer: C

K/A Match:

Requires the student to predict what will happen to shutdown cooling when heat exchanger cooling is decreased and understand the actions necessary to mitigate the consequences.

Explanation:

- A. Incorrect: Plausible because an NSSSS Gp. 6 isolation closes RHR-V-8 and RHR-V-9, the normal suction valves, and cause a loss of shutdown cooling. In this case, however, RHR-B is in FPC assist mode with suction from the skimmer surge tanks and is unaffected by a Gp. 6 isolation. Manually opening RHR-V-8 and RHR-V-9 is a normal recovery action but is unnecessary in this case.
- B. Incorrect: Plausible because an NSSSS Gp. 6 isolation normally results in a loss of shutdown cooling (see above), however, not in this case. Evacuate all personnel from the drywell and wetwell is a normal action for a loss of shutdown cooling and the time to boil is LE 2 hrs. In this case, the time to boil GT 2 hrs because of the large mass of water (flooded with fuel pool gates removed).
- C. Correct: RHR-V-48 is the RHR HX bypass valve and will fully open for 10 minutes after a LPCI initiation signal. This will cause less flow through the heat exchanger resulting in a partial loss of heat exchanger cooling and heat up of the reactor cavity and the spent fuel pool.
- D. Incorrect: Plausible because RHR-V-53B cannot be opened for 10 minutes after a LPCI initiation signal.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SOP-RHR-ASSIST-ALT, Alternate Fuel Pool Cooling Assist	
RHR System Description	
Tech Ref 1	
Tech Ref 1	

Proposed references to be provided during examination: N/A

Learning Objective: RHR-04 - RHR-04 Describe the function, operation, interlocks, trips, physical location, and power supplies of the following Residual Heat Removal System components: RHR-V-8/9, SDC isolation valves, RHR-V-48A/48B, HX shell side bypass valves, and RHR-53A/53B, shutdown cooling return valves.

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

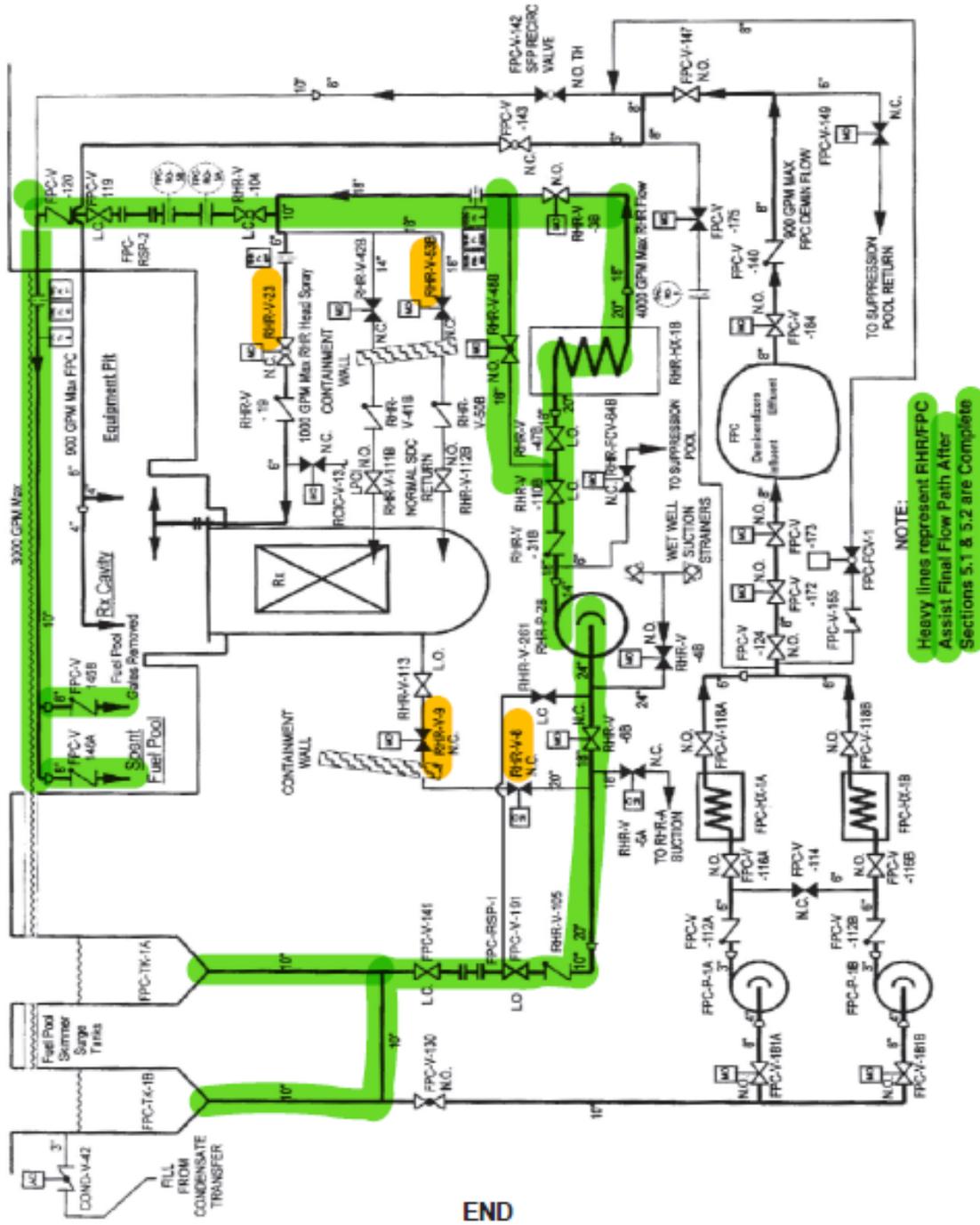
Requires the student to synthesize an understanding of what occurs in a Group 6 Isolation and how RHR-V-48B is affected by LPCI Initiation with a knowledge of how those actions will affect heat exchanger cooling and ultimately shutdown cooling.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SOP-FPC-ASSIST-ALT	Rev:	Major: 014	Minor: N/A
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Excerpt from SOP-FPC-ASSIST-ALT showing lineup after Section 5.1 and 5.2 are complete and RHR-B is in fuel pool cooling assist mode with suction from the Skimmer Surge tanks.

SUPPLEMENTAL DECAY HEAT REMOVAL SIMPLIFIED FLOW DIAGRAM



Comments / Reference: SD000198

Rev: Major:020 Minor:0

Excerpts from RHR System Manual.

- b) Group 6
 - Isolation Signals
 - +13" Reactor Water Level
 - 125 PSIG Reactor Pressure
 - Shutdown Cooling High Flow, 6.3 PSID
 - RHR Area High Temperature
 - RHR Area High ΔT
 - (1) Valves Affected
 - RHR-V-8
 - RHR-V-9
 - RHR-V-53A
 - RHR-V-53B
 - RHR-V-23

31. RHR-V-48B HX Shell Side Bypass

3-position switch CLOSE/AUTO/OPEN (spring return to AUTO, throttle)

- | | |
|-------|--|
| CLOSE | The valve closes if the switch is held in this position. Will remain closed when switch is released, if 10 minutes have elapsed since an initiation signal occurred. |
| AUTO | The valve opens if closed at the time of LPCI initiation and will auto open if attempted to be closed within ten minutes after the initiation. |
| OPEN | The valve opens if the switch is held in this position. |

35. RHR-V-53B Shutdown Cooling Return

3-position switch CLOSE/AUTO/OPEN (spring return to AUTO)

- | | |
|-------|--|
| CLOSE | The valve closes. |
| AUTO | The valve will close if any of the following conditions exist: <ul style="list-style-type: none">a) RPV low level - Level 3 (+13 inches)b) Shutdown cooling excess flowc) RHR Area high temperatured) RHR area ventilation high delta temperature.e) RPV pressure GT 160 psig downstream of RHR-V-8 and RHR-V-9. (RPV steam dome pressure GT about 125 psig) |
| OPEN | The valve opens if conditions listed above do not exist or a LPCI initiation signal exists for GT 10 minutes. |

Question: RO-29

Examination Outline Cross-reference: RO-29

Revision: 0

Date: 4/26/22

Tier: 2

Group: 1

K/A Number: 209001 K3.02

Level of Difficulty: 2

RO Importance Rating: 4.1

K/A Description: (209001 SF4 LPCS Low-Pressure Core Spray System) – K3.02 - Knowledge of the effect that a loss or malfunction of the Low-Pressure Core Spray System will have on the following systems or system parameters: ADS Logic

CGS is at 100% power.

CGS experiences a transient:

- large break LOCA.
- SM-8 overcurrent trip.
- all available ECCS systems respond as expected.
- no operator actions taken.

Current plant conditions:

- Reactor level -135" and stable for last 5 minutes.
- Reactor pressure 200 psig.

LPCS-V-1, Pump Suction from Suppression Pool, spuriously shuts.

How does this affect the plant?

LPCS-P-1 (1). When LPCS-P-1 discharge pressure drops below 145 psig, ADS-SRVs will (2).

- A. (1) trips
(2) shut
- B. (1) trips
(2) remain open
- C. (1) cavitates
(2) shut
- D. (1) cavitates
(2) remain open

Answer: D

K/A Match:

Question: RO-29

Requires knowledge that a malfunction of the pump suction valve does not cause a trip of the pump but will starve the pump causing output pressure to drop and suction cavitation. The output pressure signal, along with the output pressure signal from RHR-A, input into the Division 1 ADS logic.

Question: RO-29

Explanation: A large break LOCA will cause a significant loss of inventory in the RPV requiring recovery by ECCS systems. The SM-8 lockout will prevent RHR-B and RHR-C from operating. All available ECCS systems (HPCS, LPCS, RHR-A, and ADS) respond as expected. With no operator action, HPCS will start and inject at -50", LPCS and RHR-A will start at -129", and when level has been less than -129" for longer than 105 sec, Division 1 ADS logic will be satisfied and all 7 ADS-SRVs will open via the "A" solenoids. With the RPV at 220 psig, both LPCS and RHR-A will be injecting. When LPCS-V-1 spuriously closes, it will not cause a trip of the pump but will starve the pump causing cavitation and output pressure to drop near zero. The loss of output pressure from LPCS-P-1 does not cause the 7 ADS-SRVs to close because either pump, LPCS or RHR-A, output pressure signal is needed to satisfy the Division 1 ADS logic and keep open the 7 ADS-SRVs.

- A. Incorrect: (1) Incorrect but plausible in that the student may think a loss of suction will trip the pump similar to RHR-A. (2) Incorrect but plausible in that the student may not understand that the ADS logic is satisfied with either LPCS or RHR-A output pressure signal satisfactory.
- B. Incorrect (1) Incorrect but plausible in that the student may think a loss of suction will trip the pump similar to RHR-A. (2) Correct, see explanation above.
- C. Incorrect (1) Correct, see explanation above. (2) Incorrect but plausible in that the student may not understand that the ADS logic is satisfied with either LPCS or RHR-A output pressure signal satisfactory.
- D. Correct (1) Correct, LPCS-P-1 will run and cavitate due to loss of suction. (2) Correct, ADS-SRVs will remain open even though LPCS output pressure drops below 145 psig because RHR-A output pressure is satisfied.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
EWD-1E-018, ADS Div 1 Initiation Logic	
Nuclear Power for Everyone (www.nuclear-power.com)	
LPCS System Manual, SD000192	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 11878 - Describe the effect that a loss or malfunction of the following will have on the Automatic Depressurization System:

- a. RHR/LPCI system pressure
- b. Low Pressure Core Spray system pressure

Question: RO-29

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level
 Cog Level Justification.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SD000192	Rev:	Major: 15	Minor: 0
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Excerpt from System Manual:

COLUMBIA SYSTEMS
 LOW PRESSUE CORE SPRAY

February 2022
 SD000192, r15 nr0

V. CONTROL THEORY AND INTERLOCKS

A. Control Room Controls

1. Control Switches

a) LPCS-P-1 LPCS Pump

(Three position switch, spring return to Auto)

STOP - Pump stops. The amber MANUAL OVERRIDE light comes on when the initiation logic is sealed in and until the override is cleared the pump must be started manually. The override is cleared by resetting the initiation logic.

AUTO - Pump starts on initiation, if not running

START - Pump starts

b) LPCS-V-1 LPCS Pump Suction Valve

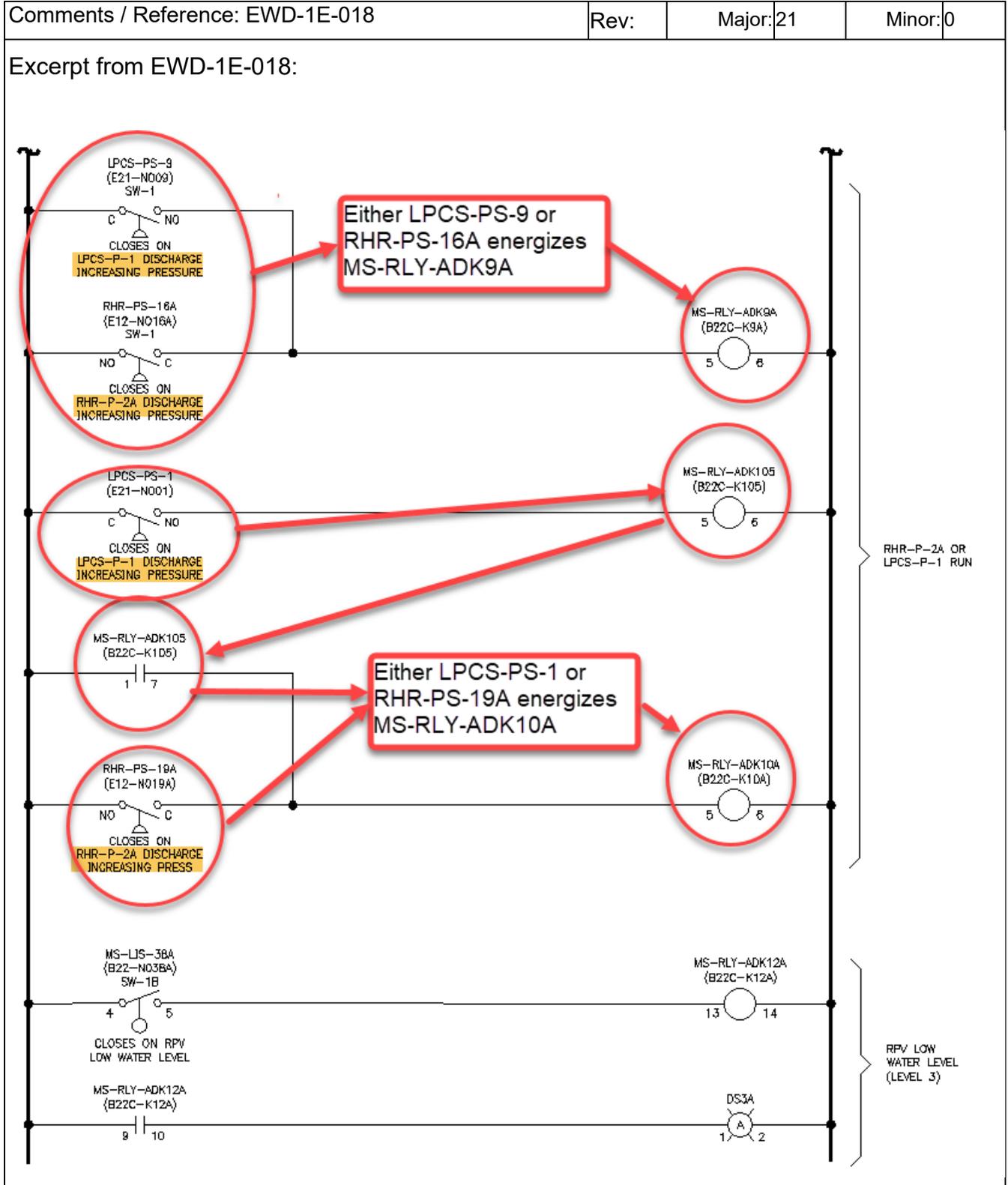
(Two position keylock switch key removable in open)

CLOSE - LPCS-V-1 Closes

OPEN- LPCS-V-1 Opens

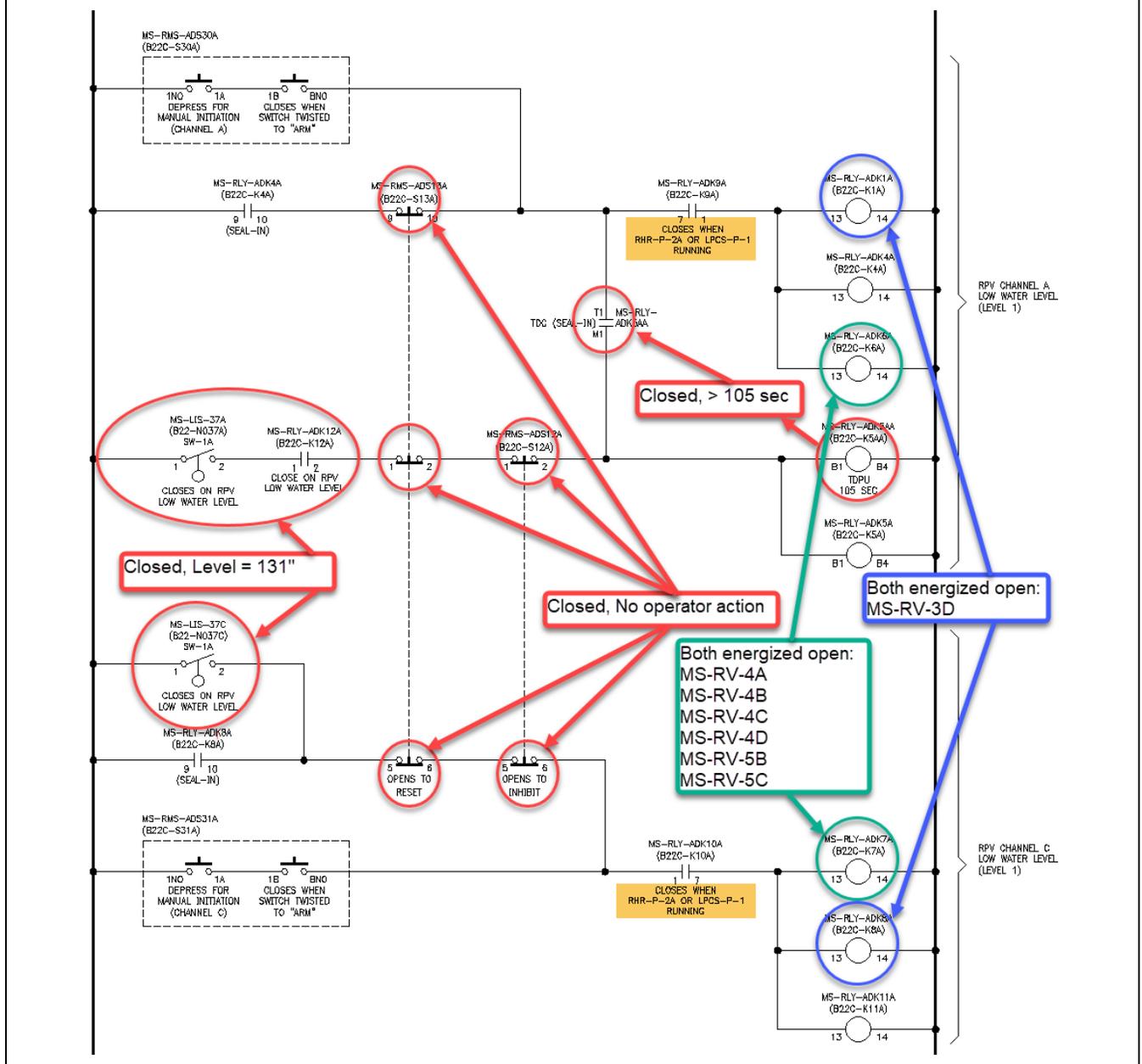
NOTE: LPCS-P-1 has no suction valve interlock to prevent pump operation with the suction valve closed.

Question: RO-29



Question: RO-29

Excerpt from EWD-1E-018 (Cont):



Question: RO-30

Examination Outline Cross-reference: RO-30

Revision: 0

Date: 5/3/22

Tier: 2

Group: 1

K/A Number: 209002 K2.02

Level of Difficulty: 2

RO Importance Rating: 3.9

K/A Description: (209002 SF4 HPCS High-Pressure Core Spray System (BWR 5, 6)) – K2.02 - Knowledge of electrical power supplies to the following: Valves

What is the electrical power supply to HPCS-V-4, HPCS Injection Valve?

- A. E-MC-7BA
- B. E-MC-8BA
- C. E-MC-S2/1A
- D. E-MC-4A

Answer: D

K/A Match:

Requires knowledge of the electrical power supplies to HPCS valves.

Explanation:

- A. Incorrect: Plausible in that E-MC-7BA is a critical bus that is the power supply for LPCS-V-5, LPCS Injection Valve, and the student make confuse this power source for HPCS-V-4.
- B. Incorrect: Plausible in that E-MC-8BA is a critical bus that is the power supply for RHR-V-42C, RHR-C Injection Valve, and the student may confuse this power source for HPCS-V-4.
- C. Incorrect: Plausible in that E-MC-S2/1A is a critical DC bus that is the power supply for RCIC-V-13, RCIC Injection Valve, and the student may confuse this power source for HPCS-V-4.
- D. Correct: E-MC-4A is the electrical power supply for HPCS-V-4.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
E503-9, Auxiliary One Line Diagram	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Question: RO-30

Learning Objective: 209002-4 Describe the function, operation, interlocks, trips, physical location, and power supplies of the following High Pressure Core Spray System components: HPCS-V-4

Question Source: Bank #: LX00769/2131
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

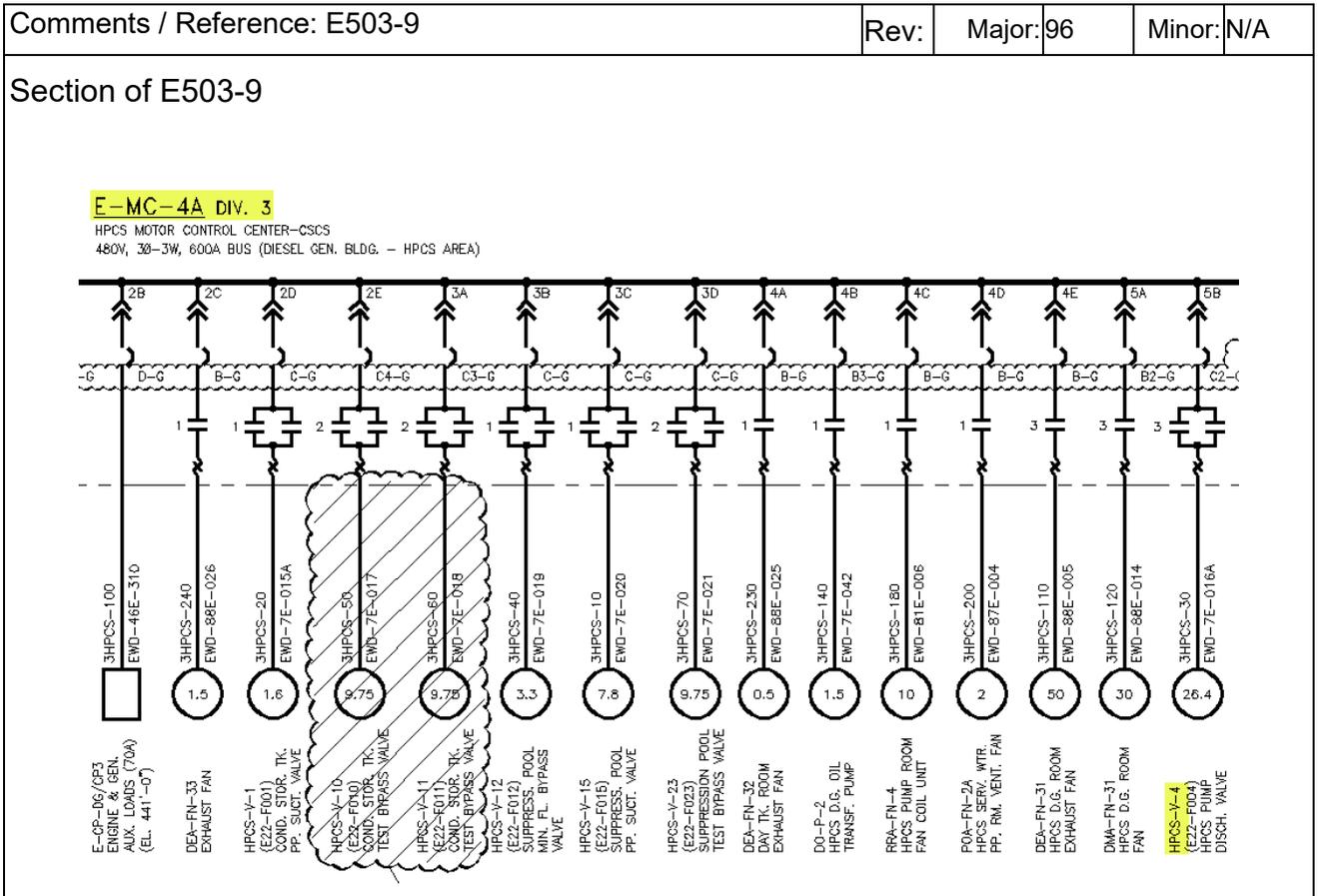
Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires a fundamental knowledge of the electrical power supplies to HPCS valves.

10 CFR Part 55 Content: 55.41 7



Question: RO-31

Examination Outline Cross-reference: RO-31

Revision: 2

Date: 5/3/22

Tier: 2

Group: 1

K/A Number: 211000 K1.06

Level of Difficulty: 2

RO Importance Rating: 3.3

K/A Description: (211000 SF1 SLCS Standby Liquid Control System) – K1.06 - Knowledge of the physical connections and/or cause and effect relationships between the Standby Liquid Control System and the following systems: Reactor vessel internals

During SLC injection, how is the boron solution normally routed to the reactor?

The boron solution is normally routed from the SLC Storage tank, through the SLC pumps, squib valves, and into the...

- A. LPCS sparger
- B. HPCS sparger
- C. RCIC reactor head spray
- D. RFW spargers

Answer: B

K/A Match:

Requires a knowledge of how SLC is routed to the RPV vessel internals.

Explanation:

- A. Incorrect: Plausible in that LPCS and HPCS have similar spargers and the student may confuse LPCS with HPCS.
- B. Correct: Boron injection with SLC is through the HPCS sparger directly into the core region.
- C. Incorrect: Plausible in that the RCIC reactor head spray can be used to rout the boron solution to the RPV but RCIC is an alternate method when the normal method (SLC) is unavailable.
- D. Incorrect: Plausible in that RFW has spargers in the reactor vessel internals and the student may have knowledge that other systems such as RWCU are routed to the RPV via RFW and confuse SLC as being one of those systems.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
5.0.10A, Flowchart Training Manual Vol II, EOPs-Hot Bases	
M522, Flow Diagram SLC System Rx Bldg	
Tech Ref 3	

Question: RO-31

Tech Ref 4	
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Proposed references to be provided during examination: N/A

Learning Objective: 5922 Describe the following SLC System flowpaths: Normal Injection

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires the student to have a fundamental knowledge SLC System flowpaths

10 CFR Part 55 Content: 55.41 8

Question: RO-31

Comments / Reference: 5.0.10A		Rev:	Major: 000	Minor: N/A
Number: 5.0.10A		Use Category: INFORMATION		Major Rev: 000
Title: Flowchart Training Manual Volume II, EOPs-Hot Bases				Minor Rev: N/A
				Page: 55 of 247
<p>2) Small transient reactor power increases are expected as injection into the RPV is initially increased and natural circulation core flow is re-established. These transient power increases will be reversed as boron is mixed. The phrase "commences and continues to increase" excludes such transients, referring only to a sustained rise in reactor power indicative of insufficient boron in the core.</p> <p>3) The magnitude of the transient reactor power increase is event specific, dependent on factors such as control rod density, temperature of water entering the core, etc. However, the magnitude of the power increase is less important than observation that reactor power peaks and then decreases as RPV water level rises. Once core flow is re-established, the in-core boron concentration is expected to increase quickly enough to outweigh the increase in moderation, resulting in a net decrease in reactivity. It is therefore expected that reactor power will begin to decrease shortly after RPV water level begins to increase and will continue to decrease until the reactor is fully shutdown.</p> <p>I. Step L-6:</p> <p>1) The Hot Shutdown Boron Weight (HSBW, Table 11) is the least weight of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under hot standby conditions. Refer to Volume I for discussion of the HSBW and Table 11.</p> <p>2) Boron injection with SLC is through the HPCS sparger directly into the core region. it may be desirable to raise RPV water level slowly to avoid displacement of boron from the core.</p> <p>3) If RCIC must be used as an alternate boron injection method, boron is injected into the RPV upper head and flows downward into the downcomer region where natural circulation carries the boron through the lower plenum where it enters the core. If RPV water level has been lowered, natural circulation core flow may have stagnated, and boron may not have yet reached the core. It may be desirable to raise RPV water level relatively quickly to distribute boron into the core region and shut down the reactor.</p>				

Question: RO-32

Examination Outline Cross-reference: RO-32

Revision: 3

Date: 5/18/22

Tier: 2

Group: 1

K/A Number: 212000 K4.07

Level of Difficulty: 3

RO Importance Rating: 4.2

K/A Description: (212000 SF7 RPS Reactor Protection System) – K4.07 - Knowledge of Reactor Protection System design features and/or interlocks that provide for the following: Manual SCRAM

Which of the following will trip the appropriate Manual Scram Relays (RPS-RLY-K15A thru K15D) resulting in a full reactor scram?

1. Single SRM Hi-Hi trip with shorting links removed
2. Depressing manual scram pushbuttons (A1 and A2) **OR** (B1 and B2)
3. Placing mode select switch in Shutdown
4. Depressing manual scram pushbuttons (A1 or A2) **AND** (B1 or B2)

A. 1, 2, and 3 ONLY

B. 1, 2, and 4 ONLY

C. 1, 3, and 4 ONLY

D. 2, 3, and 4 ONLY

Answer: C

K/A Match:

Requires the student to have knowledge of the RPS manual scram logic and interlocks.

Question: RO-32

Explanation: RPS trip logic consists of two trip systems, RPS trip system "A" and RPS trip system "B", with each trip system comprised of two redundant trip channels:

- RPS "A" trip system consists of "A1" and "A2" trip channels.
- RPS "B" trip system consists of "B1" and "B2" trip channels.

A full scram occurs when at least one trip channel in both trip systems deenergizes. The combination of the "one-out-of-two" logic for the trip channels and the "two-out-of-two" logic for the trip systems is called "one-out-of-two-taken-twice" logic.

Each RPS trip channel has a Manual Scram Relay (RPS-RLY-K15A for "A1", K15B for "B1", K15C for "A2", and K15D for "B2") that, when deenergized, will deenergize the trip channel. The following cause the manual scram relays to deenergize:

- All four manual scram relays deenergize when the Mode Select Switch is placed in Shutdown causing a full scram.
- When the shorting links are removed, any single SRM or IRM trip will cause a full scram because the trip deenergizes two manual scram relays, one in RPS "A" trip system and one in RPS "B" trip system.
- Each manual scram pushbutton (A1, A2, B1, and B2) deenergizes the corresponding manual scram relay when depressed. One manual scram pushbutton in each trip system, (A1 or A2) and (B1 or B2), must be depressed to cause a full scram. Depressing manual scram pushbuttons (A1 and A2) or (B1 and B2) will not result in a full scram. Only one of the trip systems is deenergized and a 1/2 Scram will occur.

- A. Incorrect: See explanation
- B. Incorrect: See explanation
- C. Correct: See explanation
- D. Incorrect: See explanation

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
EWD-15E-020, RPS Trip System A Scram Trip Logic	
EWD-15E-022, RPS Trip System B Scram Trip Logic	
EWD-15E-060, RPS Trip System A Scram Relay	
EWD-15E-061, RPS Trip System B Scram Relay	

Proposed references to be provided during examination: N/A

Learning Objective: RPS-02 Describe the Reactor Protection System design feature(s) or interlocks that provide for the following: Manual SCRAM

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

Question History: Last NRC Exam: Exam year or N/A

Question: RO-32

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

Justification for Cognitive Level

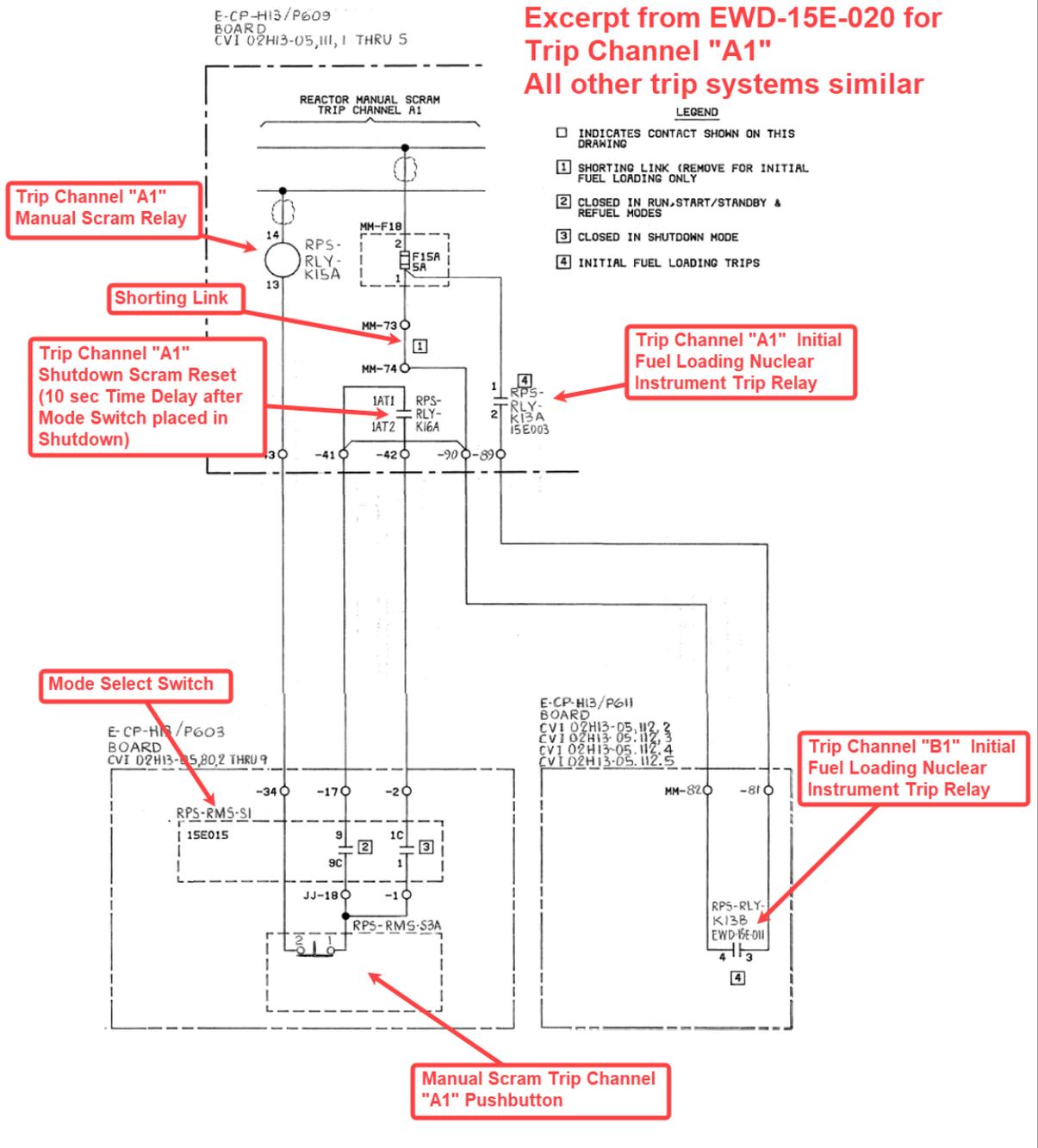
Requires a knowledge of the RPS system and how the design features and interlocks provide for a manual SCRAM.

10 CFR Part 55 Content: 55.41 7

Question: RO-32

Comments / Reference: EWD-15E-020	Rev:	Major: 15	Minor: N/A
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From EWD-15E-020



Examination Outline Cross-reference: 33

Revision: 2

Date: 5/16/22

Tier: 2

Group: 1

K/A Number: 215003.K4.10

Level of Difficulty: 3

RO Importance Rating: 3.6

K/A Description: Knowledge of Intermediate Range Monitor System design features and/or interlocks that provide for the following: Automatically bypassing IRM rod block signals

CGS is in Mode 2.

A reactor plant startup is in progress.

- The Mode Switch is in STARTUP/HOT STBY.

Several annunciators are in alarm, including:

- 4.603.A7.2-7 – ROD OUT BLOCK

IRMs read as follows:

- IRM-D: 110/125, Range 8
- All other IRMs: 105/125, Range 8

Which of the following actions, when considered separately, will clear the rod block signal?

- (1) Place the Reactor Mode switch in RUN.
- (2) Place the IRM-D Mode/Test switch (on the IRM-D drawer) out of OPERATE.
- (3) Place the IRM Bypass switch on P603 to "D".

- (1) ONLY
- (2) ONLY
- (1) and (3) ONLY
- (2) and (3) ONLY

Answer: C

K/A Match:

Requires knowledge of the actions that will automatically bypass IRM Rod Block signals.

Explanation:

- A. Incorrect. Plausible since (1) is correct. However, (3) is also correct..
- B. Incorrect. Plausible since taking the IRM Mode/Test switch out of OPERATE makes the channel Inoperable. However, this will automatically generate a Rod Block and scram signal from the associated channel with the Mode Switch not in RUN.
- C. Correct. In accordance with 4.603.A7.2-7, IRM Rod block signals are bypassed when the Mode Switch is taken to RUN. In accordance with SD000138, taking the IRM Bypass Joystick to "D" will bypass the rod block signal.
- D. Incorrect. Plausible since taking the IRM Mode/Test switch out of OPERATE makes the channel Inoperable. Plausibility is enhanced since withdrawing IRM-D from the core will lower the detector output below the Rod Block setpoint of 108/125. However, an automatic Rod Block signal is inserted if the actions of (2) or (3) are performed.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000138, IRM System Description	
4.603.A7.2-7, Annunciator 603.A7.2-7 actions	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: IRM-6 Discuss the Intermediate Range Monitoring System automatic functions/interlocks including purpose, signals, set points, sensing points, when bypassed, how/when they are: b. Automatically bypassing IRM rod block signals

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires knowledge of actions that will automatically clear IRM Rod Block signals.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SD000138	Rev:	Major: 11	Minor: 0
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C. TRIPS/INTERLOCKS

Setpoint	Result	Setpoint
5/125	Rod Block	5/125
108/125	Rod Block	108/125
120/125	Scram	120/125
Detector Not full in	Rod Block	Detector Not full in
S000	Rod Block	S000
MU	Scram	MU
HVL	(IRM INOP)	HVL
NVL		NVL

S000 - Switch out of Operate; **MU** - Module Unplugged; **HVL** - High Voltage Power Supply Low; **NVL** - Negative Voltage Power Supply Lost

4. IRM Channel Bypass Joysticks

- a) There are two IRM Channel Bypass joysticks located on the vertical section of P603. Each bypass switch can only bypass one IRM channel at a time. One bypass switch can bypass channel A, C, E, or G and the other bypass switch can bypass channel B, D, F, or H.
- b) **The following items are bypassed when an IRM is taken to BYPASS:**
 - (1) RPS scram trips from the associated IRM channel
 - (2) RMCS rod blocks from the associated IRM channel**
 - (3) Annunciator inputs from the associated IRM channel
- c) The operation of the associated IRMs back panel trip lights is unaffected by the Bypass joystick.

Comments / Reference: 4.603.A7.2-7	Rev:	Major: 059	Minor: 002
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2-7 ROD OUT BLOCK

2-7 WINDOW	SOURCE	AUTOMATIC ACTIONS
<p>ROD OUT BLOCK</p>	<p style="text-align: center;">Mode Switch - <u>ALL POSITIONS</u></p> <ul style="list-style-type: none"> • APRM Inop Trip <ul style="list-style-type: none"> • Critical Self-Test Fault • Watchdog timer has timed out • Loss of input power to the APRM • APRM Keylock Switch in INOP position • APRM <ul style="list-style-type: none"> • LT 20 LPRM inputs per APRM • LT 3 LPRM inputs per level per APRM • APRM Flow Upscale 105% flow • RWM violation • RPIS malfunction • RBM upscale <ul style="list-style-type: none"> • GT 118.2% of reference in LPSP • GT 113.2% of reference in IPSP • GT 108.2% of reference in HPSP • RBM Inop <ul style="list-style-type: none"> • Nulling (Inhibit) • Failure to complete gain calculation within 2 seconds. • Critical Self-Test Fault • Too few inputs (LT 50% LPRM) • Watchdog timer has timed out • Loss of input power to the RBM • More than one rod selected • RBM Keylock Switch in INOP position • SDV high level <ul style="list-style-type: none"> • (CRD-LS-613E - GE 527' 1" elev.) • (CRD-LS-613F - GE 527' 1" elev.) 	<p>Prevents withdrawal of any Control Rods</p>

2-7 ROD OUT BLOCK
(CONTINUED FROM PREVIOUS PAGE)

2-7 WINDOW	SOURCE	AUTOMATIC ACTIONS	
ROD OUT BLOCK	Mode Switch - <u>NOT IN RUN</u>	Prevents withdrawal of any Control Rods	
	• SRM detector wrong position		
	• SRM high (GE 1 x 10 ⁵ cps)		
	• SRM Inop <ul style="list-style-type: none"> • Function Switch out of OPERATE • Module unplugged • High Voltage low • Negative Voltage low 		
	• SRM downscale (LE 0.7 cps)		
	• IRM detector wrong position		
	• IRM high (GE 108/125)		
	• IRM Inop <ul style="list-style-type: none"> • Function Switch out of OPERATE • Module unplugged • High Voltage low • Negative Voltage low 		
	• IRM downscale (LE 5/125)		
	• APRM neutron flux high (GE 12%)		
	Mode Switch - <u>RUN</u>		
	• APRM STP upscale alarm (GE 108%) (GE 0.62W + 57.1%)		
	• APRM downscale (LE 5%)		
	• RBM downscale (LE 5%)		
	• Anytime a full scram is in		

(CONTINUED ON FOLLOWING PAGE)

2-7 ROD OUT BLOCK
(CONTINUED FROM PREVIOUS PAGE)

2-7 WINDOW	SOURCE	AUTOMATIC ACTIONS
ROD OUT BLOCK	<p style="text-align: center;">Mode Switch - <u>START/HOT STBY</u></p> <ul style="list-style-type: none"> • Refuel Platform over core <p><u>NOTE:</u> The Service Platform has been removed and the rod block jumpered.</p> <ul style="list-style-type: none"> • Service Platform Hoist loaded <p style="text-align: center;">Mode Switch - <u>REFUEL</u></p> <ul style="list-style-type: none"> • Refuel platform over core and any one of the following: <ul style="list-style-type: none"> • load on fuel grapple • load on frame mounted hoist • load on trolley mounted hoist <p><u>NOTE:</u> The Service Platform has been removed and the rod block jumpered.</p> <ul style="list-style-type: none"> • Service Platform Hoist loaded • SDV Bypass Switch in BYPASS <p style="text-align: center;">Mode Switch - <u>SHUTDOWN</u></p> <ul style="list-style-type: none"> • Rod block directly off the mode switch in the SHUTDOWN position • SDV Bypass Switch in BYPASS 	Prevents withdrawal of any Control Rods

NOTE: The RBM Inop input may activate due to the reference APRMs being gaffed, if indicated power is raised to GT 26%, with a non-peripheral rod selected. {P-202748}

1. **STOP** control rod withdrawal.
2. **INVESTIGATE** the cause of the Rod Out Block.
3. IF CRS directs, **THEN PERFORM** the following:
 - a. **INITIATE** a MON run to verify the reactor is in compliance with the fuel thermal limits.
 - b. **BYPASS** the rod block.

REFERENCES: CVI 02C12-05,14,9 (RMCS)
 CVI 02C12-05,81,20 (RMCS)
 EWD-13E-012, 013A, 023
 EWD-15E-016 {P-202748}

Question: RO-34

Examination Outline Cross-reference: RO-34

Revision: 0

Date: 5/23/22

Tier: 2

Group: 1

K/A Number: 2.1.45

Level of Difficulty: 3

RO Importance Rating: 4.3

K/A Description: (215004 - SF7 SRMS Source Range Monitor System) – 2.1.45 - Ability to identify and interpret diverse indications to validate the response of another indication

How is Source Range Monitor to Intermediate Range Monitor (SRM/IRM) overlap verified?

1. Verify the SRM detectors are fully inserted
2. Verify a reactor period of GT 60 seconds
3. Verify at least 3 operable IRMs per trip system are on scale and increasing
4. Verify at least 3 operable SRMs read LT 3×10^5 cps

A. 1, 2, and 3 ONLY

B. 1, 2, and 4 ONLY

C. 1, 3, and 4 ONLY

D. 2, 3, and 4 ONLY

Answer: C

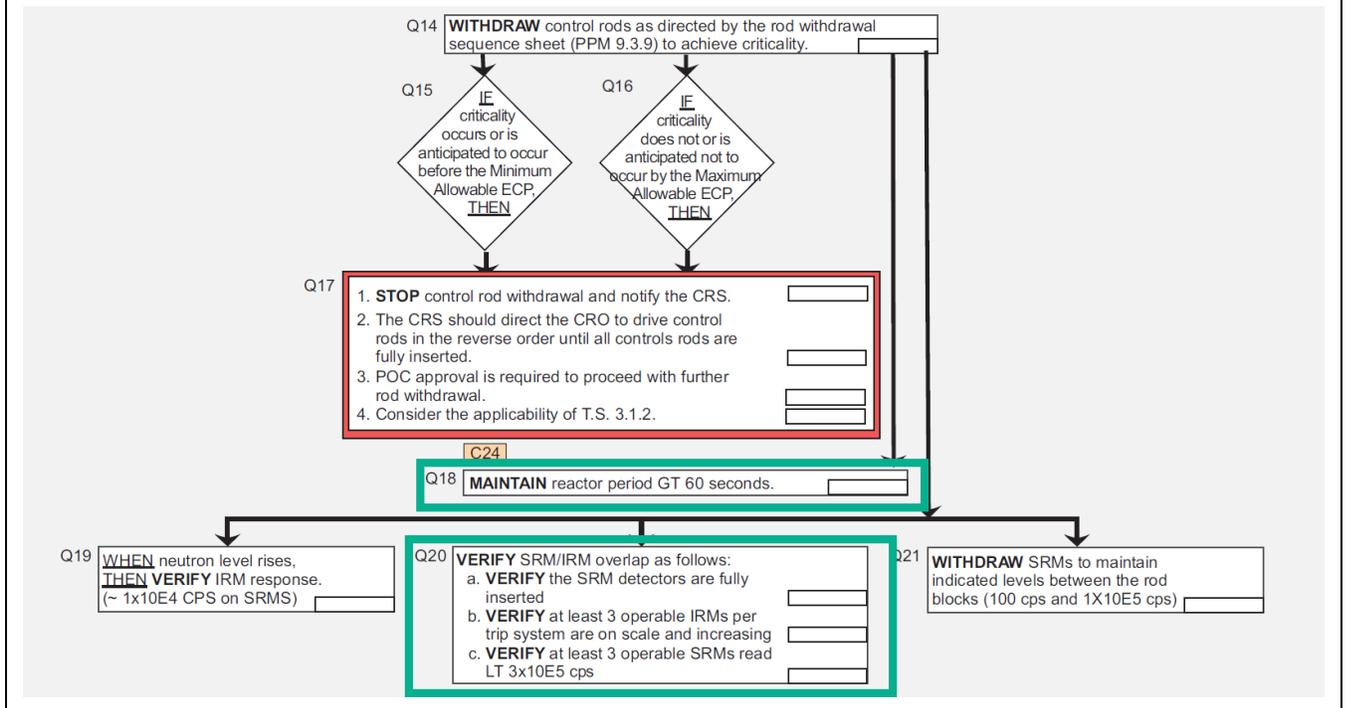
K/A Match:

Requires the student to identify and interpret the indications of SRMs to validate the response of IRMs by understanding when proper SRM/IRM overlap has occurred.

Question: RO-34

Comments / Reference: PPM 3.1.2	Rev:	Major: 093	Minor: N/A
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From Attachment 7.3, Start Up Flow Chart, of PPM 3.1.2, Reactor Plant Startup



Examination Outline Cross-reference: RO-35

Revision: 2

Date: 05/05/2022

Tier: 2

Group: 1

K/A Number: 215005 A1.03

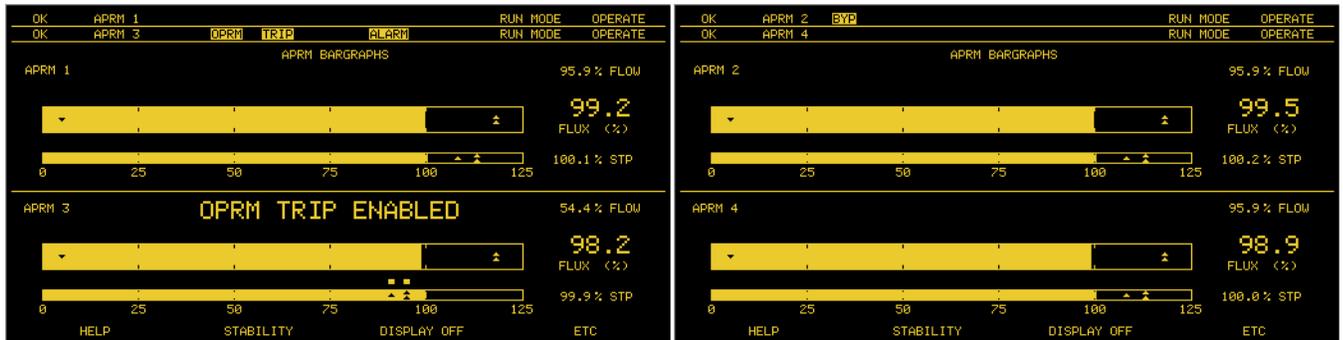
Level of Difficulty: 2

RO Importance Rating: 3.9

K/A Description: ARPM - Ability to predict and/or monitor changes in parameters associated with operation of the Average Power Range Monitor/Local Power Range Monitor System, including: (CFR: 41.5 / 45.5)
A1.03 Control rod block status

CGS is operating at 100% power.

A failure occurs that results in the following indications on the PRNM system.



With the provided indications what is the expected plant response?

A Rod Block Signal (1). A ½ Scram signal (2).

- A. (1) exists
(2) does NOT exist
- B. (1) exists
(2) exists
- C. (1) does NOT exist
(2) does NOT exist
- D. (1) does NOT exist
(2) exists

Answer: A

K/A Match:

Requires knowledge of APRM rod block setpoints and the ability to interpret plant status to determine if one of these setpoints has been exceeded.

Explanation:

- A. Correct. The provided indication shows the plant at 100% power with a failure of an RRC Flow element for the 'C' channel of PRNM. With this failure the APRM Upscale Thermal Scram and rod block setpoints have been exceeded. This results in a rod block and 1 vote sent to the voter but no half scram.
- B. Incorrect. Plausible because the APRM Upscale Thermal scram and rod block setpoints have both been exceeded for APRM 3. This does result in a rod block on APRM 3, but it only results in 1 vote being sent to the 2-out-of-4 voters.
- C. Incorrect. Plausible because the APRM Upscale Thermal scram and rod block setpoints have both been exceeded for APRM 3. This does result in a rod block on APRM 3, but it only results in 1 vote being sent to the 2-out-of-4 voters. A single vote does not result in a half scram, this is still plausible though if the student believes that a scram signal from APRM 3 will result in a ½ scram on RPS A and the student believes that the bypassed APRM 2 results in another vote causing a full scram to occur.
- D. Incorrect. Plausible because the APRM Upscale Thermal scram and rod block setpoints have both been exceeded for APRM 3. This does result in a rod block on APRM 3, but it only results in 1 vote being sent to the 2-out-of-4 voters. A single vote does not result in a half scram, this is still plausible though if the student believes that a scram signal from APRM 3 will result in a ½ scram on RPS B.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD001819	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: PRNM ODAs provided in stem.

Learning Objective: PRNM-04 - Describe the function, operation, interlocks, trips, physical location, and power supplies of the following Power Range Neutron Monitoring (PRNM) System components.
 b. APRM Chassis (channels)

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires interpreting plant indications and determining if a scram and/or rod block signal is generated.

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APRM ROD BLOCK SIGNALS		
Rod Block	Setpoint	Bypassed When
Upscale STP Setdown	GE 12% power	Reactor Mode Switch in RUN
Upscale Thermal	0.62Wd + 57.1% [Two Loop Operation] 0.62Wd + 56.0% [Single Loop Operation] (Wd = Recirc Drive Flow) (For both TLO and SLO clamped at 108%)	Reactor Mode Switch NOT in RUN

APRM SCRAM SIGNALS (Votes)		
Trip	Setpoint	Bypassed When
Upscale Neutron Flux Setdown	GT 15% power	Reactor Mode Switch in RUN
	118%	Reactor Mode Switch NOT in RUN
Upscale Thermal (STP)	0.62Wd + 60.9% [Two Loop Operation] 0.62Wd + 56.1% [Single Loop Operation] (Wd = Recirc Drive Flow) (For both TLO and SLO clamped at 112.9%)	Never

5. APRM Bypass Switch (Panel P603)

One fiber optic joystick type bypass switch provides the ability to bypass one of the four APRM channels at a time. Use of this bypass switch automatically bypasses all functions associated with that APRM. These include:

- OPRM trips
- Flow signal and upscale alarms
- Inputs from APRM to RBM

2. APRMs

A minimum of three channels out of four must be operable for startup and power operation. During startup (mode switch in startup), monitor IRMs and APRMs to ensure approximately one-decade overlap.

The APRM inputs to RPS will initiate scram trip signals under all operational conditions if the previously discussed set points are exceeded. If the two-out-of-four voter logic (two-out-of-three if one channel bypassed) input to the RPS System is satisfied, a reactor scram will occur. The APRMs do not operate in the non-coincidence mode like the SRMs/IRMs if the shorting links are removed.

- Any one APRM can initiate a rod block, regardless of reactor mode switch position.

Question: RO-36

Examination Outline Cross-reference: RO-36

Revision: 0

Date: 6/1/22

Tier: 2

Group: 1

K/A Number: 217000 K3.05

Level of Difficulty: 3

RO Importance Rating: 3.3

K/A Description: (217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling) – K3.05 - Knowledge of the effect that a loss or malfunction of the Reactor Core Isolation Cooling System will have on the following systems or system parameters: Suppression pool level.

CGS has scrammed and entered PPM 5.1.1, RPV Control (Modes 1-3).

- HPCS and RFW are unavailable.
- RCIC is manually initiated.

What occurs on initiation?

RCIC-V-13, RPV Injection, opens (1).

If RCIC-V-13 fails to open, Suppression Pool Level will (2).

- A. (1) immediately
(2) increase
- B. (1) when RCIC-V-45, Steam to RCIC Turbine Inlet Valve, starts to open
(2) increase
- C. (1) immediately
(2) remain the same
- D. (1) when RCIC-V-45, Steam to RCIC Turbine Inlet Valve, starts to open
(2) remain the same

Answer: B

K/A Match:

Requires the student to understand the normal line-up of RCIC, the interlocks that will cause the injection valve to open, and the effect on the suppression pool level if the injection valve fails to open.

Question: RO-36

Explanation:

In a normal line-up for RCIC:

- RCIC-V-45 is closed.
- RCIC-V-1, RCIC Turbine Trip & Throttle Valve, is open.
- RCIC-V-13 is closed.
- RCIC-V-10, RCIC-P-1 Condensate Storage Tank (CST) Suction Valve, is open.
- RCIC-V-31, RCIC-P-1 Suppression Pool (SP) Suction Valve, is closed.
- RCIC-V-19, RCIC-P-1 Min Flow Valve, is closed.
- RCIC-V-68, RCIC Turbine Exhaust to Suppression Pool, is open.

On initiation of RCIC:

- RCIC-V-45 opens if RCIC-V-68 is open.
- RCIC-V-13 opens if RCIC-V-45 and RCIC-V-1 are not full closed.
- RCIC-V-10 remains open and RCIC-V-31 remains closed (i.e., no swap over).

If RCIC-V-13 fails to open:

- RCIC-V-19 opens when RCIC-P-1 pressure is GT 125 psig and flow is LT 75 gpm.
- Flow path for RCIC-P-1 is from the CSTs to the SP resulting in an increase of SP level.

- A. Incorrect: (1) Incorrect but plausible because RCIC-V-13 does open on initiation, however, RCIC-V-45 is shut at the time of initiation and must start opening before RCIC-V-13 can open.
(2) Correct
- B. Correct: (1) Correct.
(2) Correct.
- C. Incorrect: (1) Incorrect but plausible because RCIC-V-13 does open on initiation, however, RCIC-V-45 is shut at the time of initiation and must start opening before RCIC-V-13 can open.
(2) Incorrect but plausible because the credited source is the SP, however, the preferred source is the CSTs and swap over doesn't occur until CST low level of 1'10". This results in a flow path from CSTs to SP, not a flow path from SP to SP.
- D. Incorrect: (1) Correct
(2) Incorrect but plausible because the credited source is the SP, however, the preferred source is the CSTs and swap over doesn't occur until CTS low level of 1'10". This results in a flow path from CSTs to SP, not a flow path from SP to SP.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
EWD-6E-049, RCIC-V-13 Electrical Wiring Diagram	
EWD-6E-053, RCIC-V-19 Electrical Wiring Diagram	
SD000180, RCIC System Manual	

Question: RO-36

Proposed references to be provided during examination: N/A

Learning Objective: RCIC-11 Evaluate the following RCIC System indications/responses and determine if the indication/ response is expected and normal: RCIC flow, Suppression pool level

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

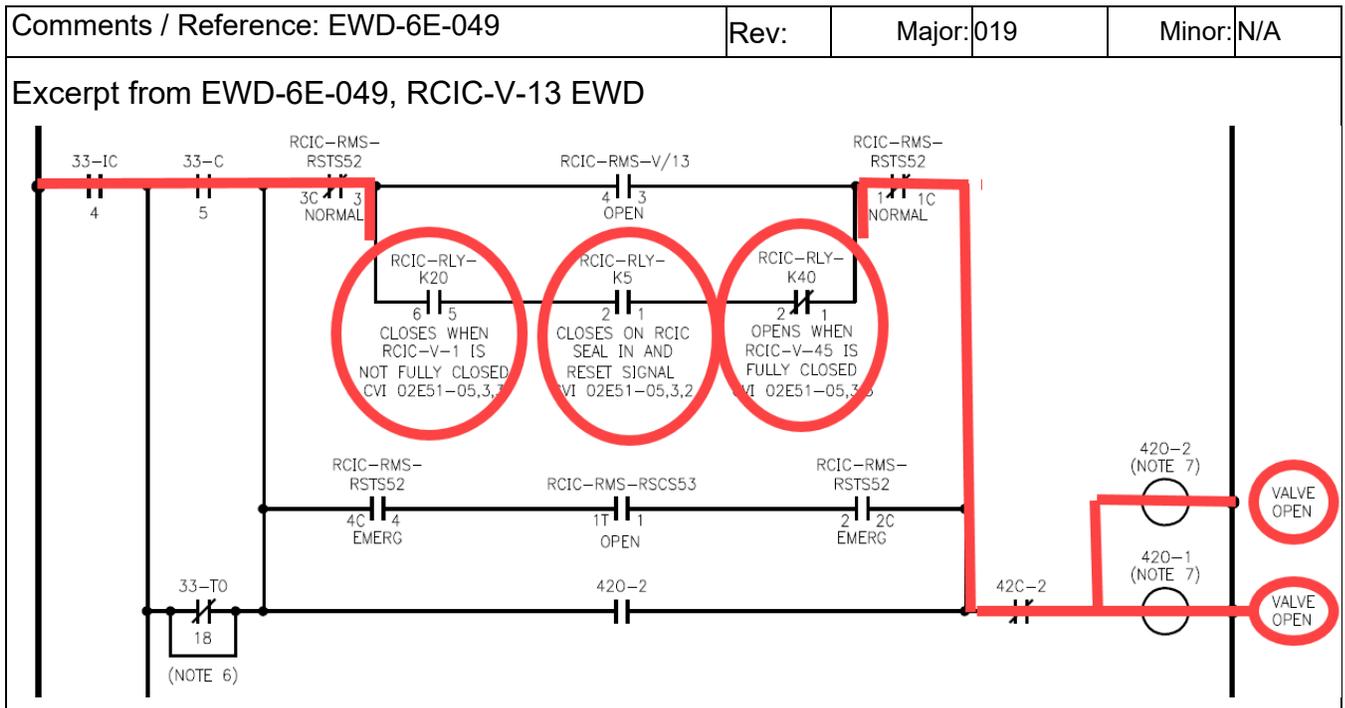
Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires a synthesis of the knowledge on RCIC normal line-up and interlocks with expected responses to malfunctions when RCIC is initiated.

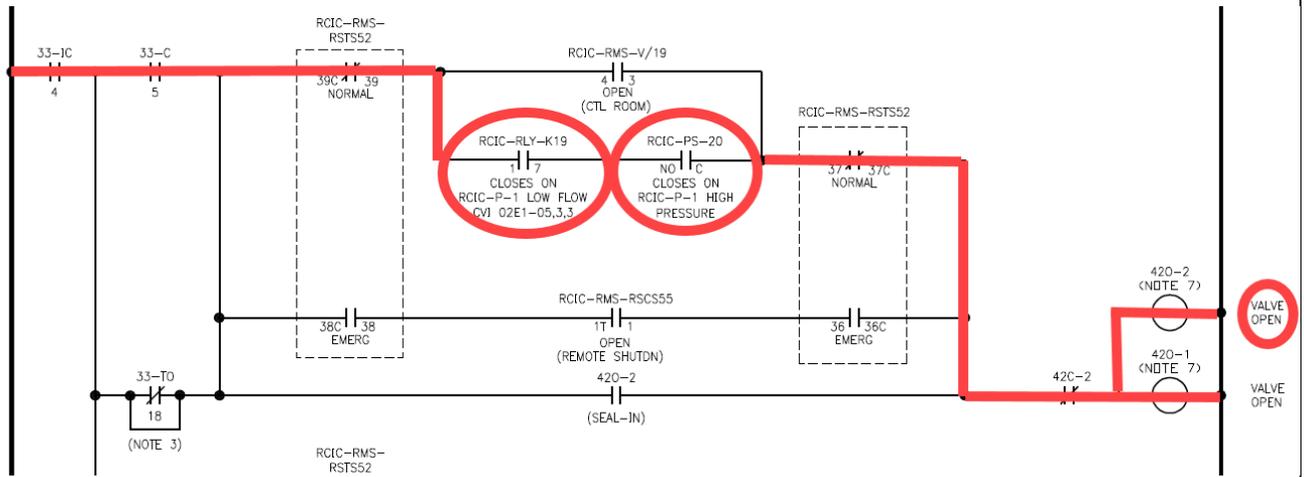
10 CFR Part 55 Content: 55.41 7



Question: RO-36

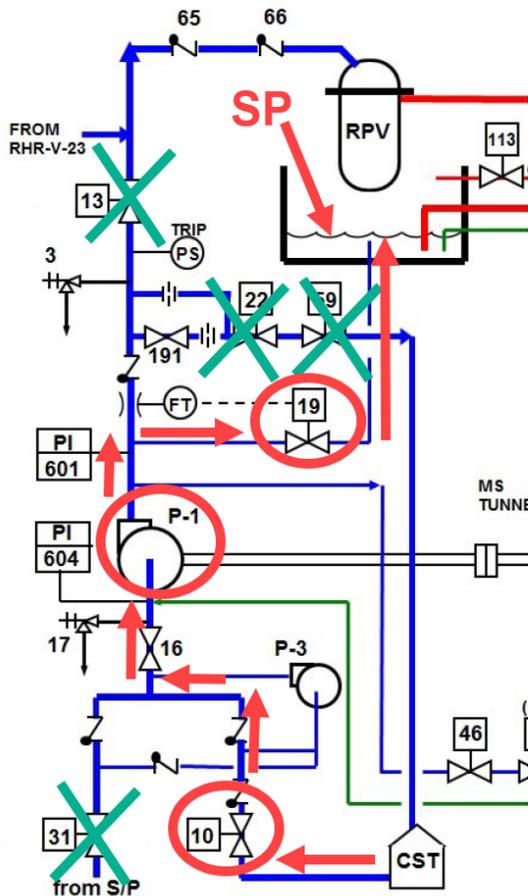
Comments / Reference: EWD-6E-053	Rev:	Major: 019	Minor: N/A
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Excerpt from EWD-6E-053, RCIC-V-19 EWD



Comments / Reference: SD000180	Rev:	Major: 019	Minor: 0
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Excerpt from SD000180, Flow path if RCIC-V-13 is closed:



Question: RO-37

Examination Outline Cross-reference: 37

Revision: 0

Date: Rev Date

Tier: 2

Group: 1

K/A Number: 218000.A4.03

Level of Difficulty: 2

RO Importance Rating: 3.9

K/A Description: Ability to manually operate and/or monitor in the control room: ADS logic reset

Which of the following operator actions will cause ALL Automatic Depressurization System (ADS) valves to close after a valid initiation signal is received in BOTH ADS trip systems?

- A. Place either ONE of the ADS inhibit switches to the INHIBIT position.
- B. Depress BOTH ADS "Reactor Vessel Low Level/Timer Seal In" RESET pushbuttons.
- C. Place the control switches for the ADS valves to the OFF position on panel H13-P601.
- D. Secure (shutdown) BOTH low pressure emergency core cooling system (ECCS) pumps in EITHER division 1 OR division 2.

Answer: B

K/A Match:

Requires knowledge of the actions that will manually reset ADS logic.

Explanation:

- A. Incorrect. Plausible since placing BOTH ADS Inhibit switches to INHIBIT will cause all ADS SRVs to close. However, operating only one switch will not cause all valves to close.
- B. Correct. Depressing both ADS RESET pushbuttons will close the SRVs and reset the 105 second timer.
- C. Incorrect. Plausible if it is believed that the SRV control switches will close the SRVs in all conditions. However, the control switches control the 'C' solenoids on the SRVs while the ADS circuitry opens and closes the SRVs via the 'A' and 'B' solenoids. Therefore, the control switches will not close the ADS SRVs that have been opened via the automatic depressurization system.
- D. Incorrect. Plausible since ADS valves will close if LPCI pump pressure is < 125 psig and LPCS pump pressure is < 145 psig. However, since BOTH ADS trip systems have actuated, Securing ECCS pumps on a single train will not close the ADS valves since the opposite train ECCS pumps are still operating.

Question: RO-37

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000186 – ADS System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 5076 - State how to override or reset the ADS logic before and after the timer has timed out.

Question Source: Bank #: 90168
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires knowledge of actions that will reset ADS

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SD000186	Rev:	Major: 14	Minor: 0
COLUMBIA SYSTEMS AUTOMATIC DEPRESSURIZATION SYSTEM		April 2022 SD000186, r14 mr0	
VII. <u>SYSTEM OPERATIONAL SUMMARY</u> (ADS-05, 06, 10, 12)			
A. <u>Normal</u>			
Operation of ADS is designed to be automatic. During normal reactor plant operation the ADS SRVs are closed, the logic is deenergized, and ADS initiation signals are not present.			
B. <u>Abnormal</u>			
1. When in the Emergency Operating Procedures (EOP's), the automatic operation of ADS is prevented in PPM 5.1.1 using the ADS DIV 1(2) INHIBIT switches, and manual actions are taken to depressurize the reactor when required. During failure to scram conditions (ATWS) PPM 5.1.2 directs the ADS function be prevented early in the event using the ADS DIV 1(2) INHIBIT switches to preclude injection of large amounts of cold unborated water which may result in core damage.			
2. ADS valves will close after initiation when any one of the following occurs:			
a) Both ADS INHIBIT switches are taken to INHIBIT.			
b) Both ADS REACTOR VESSEL LOW LEVEL/TIMER SEAL IN reset pushbuttons are depressed (this closes the valves and resets the 105 sec. time delay).			
c) ECCS pump pressure is LT 125 psig (LPCI) or 145 psig (LPCS).			
d) Loss of Division 1 and 2 125VDC.			
e) Depletion of the SRV nitrogen supply.			

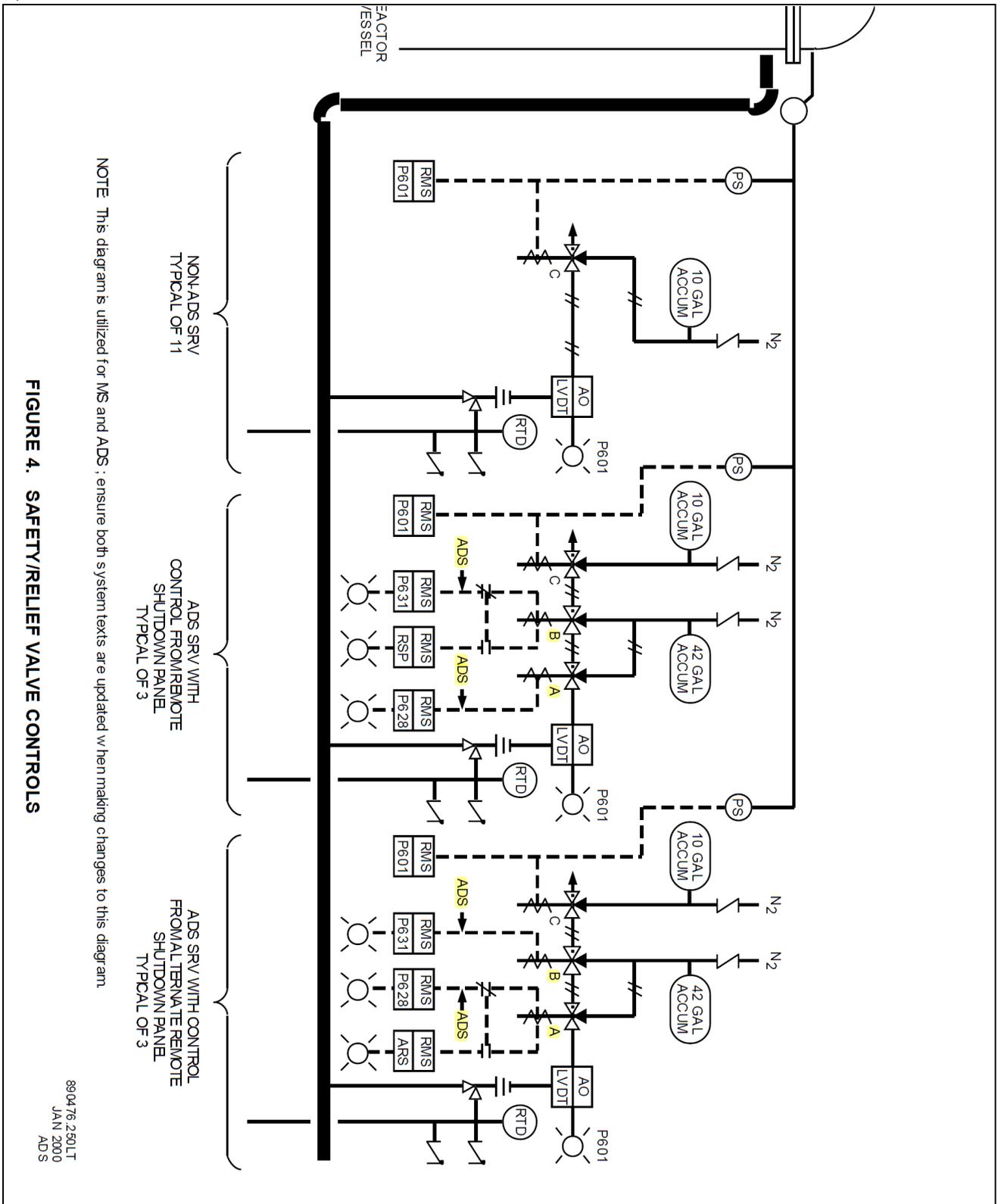


FIGURE 4. SAFETY/RELIEF VALVE CONTROLS

Question: RO-38

Examination Outline Cross-reference: 38

Revision: 3

Date: 5/16/22

Tier: 2

Group: 1

K/A Number: 223002.A4.03

Level of Difficulty: LOD

RO Importance Rating: 4.0

K/A Description: Primary Containment Isolation System/Nuclear Steam Supply Shutoff: Ability to manually operate and/or monitor in the control room: Reset of system isolations

How is the plant affected when the NS4 ISOLATION LOGIC C AND D RESET pushbutton at P601 is depressed?

Depressing the NS4 ISOLATION LOGIC C AND D RESET pushbutton at P601 resets (1) seal in signals (2) CR-HVAC dampers and fans.

- A. (1) INBOARD
(2) except
- B. (1) INBOARD
(2) including
- C. (1) OUTBOARD
(2) except
- D. (1) OUTBOARD
(2) including

Answer: A

K/A Match:

Requires knowledge of operation of NS4 reset logic in the control room.

Explanation:

- A. Correct. In accordance with SD000173, NS4 System Description, depressing the ISOLATION LOGIC C AND D RESET pushbuttons Resets Inboard (Groups 1-7) seal in signals except CR-HVAC dampers and fans. CR-HVAC dampers and fans are reset using the FAZ RESET pushbuttons on RC-1 and 2 after resetting NS4 logic on P601.
- B. Incorrect. Plausible since (1) is correct. However, CR-HVAC dampers and fans are reset using the FAZ RESET pushbuttons on RC-1 and 2 after resetting NS4 logic on P601.
- C. Incorrect. Plausible since (2) is correct. However, LOGIC A and B control OUTBOARD isolation valves.
- D. Incorrect. Plausible if it is believed that LOGIC C and D operate the OUTBOARD NS4 isolation valves. However, LOGIC A and B control OUTBOARD isolation valves. Additionally, CR-HVAC

Question: RO-38

dampers and fans are reset using the FAZ RESET pushbuttons on RC-1 and 2 after resetting NS4 logic on P601.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000173 – NS4 System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 5602 - Explain the purpose of the NS4 ISOLATION LOGIC C AND D RESET and the ISOLATION LOGIC A AND B RESET pushbuttons.

Question Source: Bank #: 64303
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires recall of operation of NS4 Logic Reset pushbuttons.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SD000173	Rev:	Major: 15	Minor: 0
<p>COLUMBIA SYSTEMS NS4</p> <p>B. <u>NS4 RESET Pushbuttons P601</u> (Figure 9, 17)</p> <p><u>ISOLATION LOGIC C AND D RESET pushbutton</u></p> <p>Resets <u>Inboard</u> (Groups 1-7) seal in signals except <u>CR-HVAC dampers and fans</u>.</p> <p><u>ISOLATION LOGIC A AND B RESET pushbutton</u></p> <p>Resets <u>Outboard</u> (Groups 1-7) seal in signals except <u>CR-HVAC dampers and fans</u>.</p> <p><u>NOTE: All MSIV control switches must be in CLOSE when the RESET pushbuttons are depressed to reset the Group 1 logic. CR-HVAC dampers and fans are reset using the FAZ RESET pushbuttons on RC-1 and 2 after resetting NS4 logic on P601. The "Z" signal, Rx. Bldg. Exhaust Rad. Monitors can be reset at P606 and P633 without depressing the RESETS on P601. (Figure 31, 31)</u></p>		<p>February 2021 SD000173, r15 mr0</p> <p>NSSSS-05c</p>	

Question: RO-39

Examination Outline Cross-reference: 39

Revision: 0

Date: 5/16/22

Tier: 2

Group: 1

K/A Number: 239002.K2.01

Level of Difficulty: 2

RO Importance Rating: 3.7

K/A Description: Knowledge of electrical power supplies to the following: SRV solenoids

What is the power supply to the 'B' solenoid for SRV MS-RV-4C when operated from the control room?

- A. DP-S1-1A
- B. DP-S1-1F
- C. DP-S1-2A
- D. DP-S1- 2D

Answer: C

K/A Match:

Requires knowledge of the power supplies to SRV solenoids

Explanation:

- A. Incorrect. Plausible since the 'A' and 'C' solenoids are powered from DP-S1-1A. However, the 'B' solenoid is powered from DP-S1-2A.
- B. Incorrect. Plausible since the power supply to the 'A' solenoid of SRVs that can be operated from the Alternate Remote Shutdown Panel (ARSP) shifts to DP-S1-1F when the valve is operated from the ARSP. However, the 'B' solenoid is powered from DP-S1-2A.
- C. Correct. When operated from the control room, the power supply to the 'B' solenoid of MS-RV-4C is DP-S1-2A.
- D. Incorrect. Plausible since the power supply to the 'B' solenoid of SRVs that can be operated from the Remote Shutdown Panel (RSD) shifts to DP-S1-1F when the valve is operated from the RSD. However, the 'B' solenoid is powered from DP-S1-2A when operated from the control room.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000128, Main Steam System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: MS-9 Describe the systems supporting the Main Steam System, the nature of the support, and the impact a loss (or malfunction) of the system would have on the Main Steam System. c. DC Distribution

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires recall of power supplies to SRV solenoids.

10 CFR Part 55 Content: 55.41 7

Question: RO-39

Comments / Reference: SD000128		Rev:	Major: 15	Minor: 0
COLUMBIA SYSTEMS MAIN STEAM		November 2021 SD000128, r15 mr0		
IV. POWER SUPPLIES				
<u>SRV's</u>	<u>"A" SPV</u>	<u>"B" SPV</u>	<u>"C" SPV</u>	
MS-RV-2A	N/A	N/A	DP-S1-1A	
MS-RV-3A	N/A	N/A	DP-S1-1A	
MS-RV-2C	N/A	N/A	DP-S1-1A	
MS-RV-2D	N/A	N/A	DP-S1-1A	
MS-RV-1B	N/A	N/A	DP-S1-1A	
MS-RV-2B	N/A	N/A	DP-S1-1A	
MS-RV-3B	N/A	N/A	DP-S1-1A	
MS-RV-3C	N/A	N/A	DP-S1-1A	
MS-RV-1A	N/A	N/A	DP-S1-1A	
MS-RV-1D	N/A	N/A	DP-S1-1A	
MS-RV-1C	N/A	N/A	DP-S1-1A	
** MS-RV-4C	DP-S1-1A	DP-S1-2A	DP-S1-1A	
* MS-RV-5C	DP-S1-1A	DP-S1-2A	DP-S1-1A	
MS-RV-4D	DP-S1-1A	DP-S1-2A	DP-S1-1A	
** MS-RV-4B	DP-S1-1A	DP-S1-2A	DP-S1-1A	
** MS-RV-4A	DP-S1-1A	DP-S1-2A	DP-S1-1A	
* MS-RV-5B	DP-S1-1A	DP-S1-2A	DP-S1-1A	
* MS-RV-3D	DP-S1-1A	DP-S1-2A	DP-S1-1A	
* When these SRVs are controlled from the Alternate Remote Shutdown Panel (ARSP) the power supply to their "A" solenoid shifts to DP-S1-1F. The "B" and "C" solenoids cannot be activated from the Alternate Remote Shutdown Panel.				
** When these SRVs are controlled from the Remote Shutdown Panel (RSD) the power supply to their "B" solenoid shifts to DP-S1-2D. The "A and "C" solenoids cannot be activated from the Remote Shutdown Panel.				

Question: RO-40

Examination Outline Cross-reference: 40

Revision: 0

Date: 5/17/22

Tier: 2

Group: 1

K/A Number: 259002.A2.07

Level of Difficulty: 3

RO Importance Rating: 3.0

K/A Description: Ability to (a) predict the impacts of the following on the Reactor Water Level Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of comparator bias signal

CGS is in Mode 1.

Reactor power is 100%.

A failure occurs in the Feedwater Level Control (FWLC) system which removes all feed flow/steam flow comparator bias input.

How does this failure affect RPV level control and what actions should be taken?

- A. RPV level will lower until a reactor scram occur. Reduce RRC flow to 74 Mlb/hr to restore RPV level.
- B. RPV level will not initially change but both RFTs will automatically transfer to MDEM. Manually control RFT speed to match feed flow with steam flow.
- C. RPV level will rise until both RFTs trip. Scram the reactor prior to RPV level reaching the RFT trip setpoint.
- D. RPV level will not change but level control will be more sluggish during transients. Verify that the FWLC system automatically transfers to 1-element mode.

Answer: D

K/A Match:

Requires knowledge of the results of a loss of comparator bias and the required actions.

Question: RO-40

Explanation:

A. Incorrect. Plausible since some FWLC system failures could cause RPV level to lower. Plausibility is enhanced since ABN-LEVEL directs lowering RRC flow if unable to match feed flow and steam flow. However, a loss of comparator bias signal will not cause steady state RPV level to lower.

B. Incorrect. Plausible since steady state RPV level will not change. Plausibility is enhanced since RFTs will automatically transfer to MDEM on a complete loss of Narrow Range RPV level input. However, a loss of comparator bias will not cause RFTs to automatically transfer to MDEM.

C. Incorrect. Plausible since some FWLC system failures could cause RPV level to rise. Plausibility is enhanced since ABN-LEVEL directs manually scrambling the reactor prior to RPV level reaching the level to automatically trip the RFTs. However, a loss of comparator bias signal will not cause steady state RPV level to rise.

D. Correct. The comparator bias signal in the FWLC system provides anticipatory control of RPV level during a transient to allow adjustments to RFT speed prior to experiencing a large level deviation. However, the FWLC system is level dominant so that RPV level will be maintained within the 3 inch deadband. The FWLC system will automatically transfer to 1-element control on a loss of feed flow/steam flow bias signal. Operators should verify that this occurs.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000157, Feedwater Level Control System Description	
ABN-LEVEL, Unplanned RPV Water Level Change	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: FWLC-12 - Evaluate given key Feedwater Level Control System parameters, if needed determine a course of action to correct or mitigate the following abnormal condition(s): g. Loss of comparator bias signal

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize an understand of the effects of a loss of FWLC comparator bias with a knowledge of the appropriate actions for these conditions.

Comments / Reference: SD000157

Rev:

Major: 23

Minor: 0

V. CONTROL THEORY AND INTERLOCKS

A. Control Room Controls

1. Three Element and Single Element Control Mode (Figure 1)

The purpose of the three element control mode of the FWLC system is to provide a more stable reactor level control (compared to single element control) during normal operation and transients. To accomplish this, the PLC must anticipate changes in reactor level and take mitigating action before they occur. This is done by monitoring steam flow/feed flow mismatch and adjusting feedwater flow accordingly.

The three element control mode is level dominant. The steam flow/feed flow mismatch signal is only used to modify the indicated level signal. The system is designed such that it can respond very quickly to significant short term steam flow/feed flow mismatches but in the long run, the reactor level signal will dominate.

In three element control, total steam flow is compared to total feed flow within the PLC and any difference is used to bias the indicated level signal used to compare to desired level to determine a desired feedflow signal. During steady-state operation, total steam flow equals total feed flow and the indicated level signal is passed through unbiased and used to determine desired feed flow signal.

When power is increased, total steam flow will initially increase and exceed total feed flow. In three element control, this results in a signal from the steam flow / feed flow comparator software program that subtracts from the level signal making indicated water level look lower. When this biased indicated level signal is compared to the level setpoint, the mismatch causes the PLC to increase desired feed flow signal and therefore to increase the speed of the RFP turbines to increase feed flow in anticipation of the expected decrease in level. Loss of this comparator bias would result in more sluggish level control response (It would be like single element control).

Question: RO-40

When power is decreased, total steam flow will initially decrease and become less than total feed flow. In three element control, this results in a signal from the steam flow / feed flow comparator software program that adds to the indicated level signal making water level look higher. When this biased indicated level signal is compared to the level setpoint, it causes the PLC to decrease desired feedwater flow signal and therefore the speed of the RFP turbines to decrease feed flow in anticipation of the expected increase in level.

In Single Element Control, the level signal is not affected by either steam flow or feed flow and the FWLC system must wait until there is an actual mismatch between desired setpoint and level before the PLC causes a change in RFP turbine speed or change in startup FCV position. This control method is less responsive but adequate at low power levels or steady state high power conditions.

B. Abnormal

1. Steam Flow Transmitter Malfunction

Steam flow validation logic within the PLC detects a steam flow measurement fault and automatically selects 1-element mode before RPV/L changes more than 50% of the normal operating range (3 inches).

This validation logic detects the following:

- Gross Failure Low/High due to power loss or open/shorted circuit
- Excessive Steamline to Steamline channel deviations (instrument drift)
- Excessive total steam flow to main turbine steam flow deviations
- Excessive total steam flow to feedwater flow deviations

The validation logic does allow the isolation of one main steam line without automatically shifting to 1-element. This permits the testing of MSIVs at power without the FWLC system auto transferring to 1-element.

The net effect is that RPV/L will not change but level control will be more sluggish due to the loss of the anticipatory function of the steam flow/feed flow mismatch. Several annunciators will alarm at P603 to alert the operator of this abnormal condition.

2. Feed Water Flow Transmitter Malfunction

Similar to the steam flow validation logic the feedwater flow validation logic within the PLC detects a feedwater flow measurement fault and automatically selects 1-element mode before RPV/L changes more than 50% of the normal operating range (3 inches).

This validation logic detects the following:

- Gross Failure Low/High due to power loss or open/shorted circuit
- Excessive Feedline to Feedline channel deviations (instrument drift)
- Excessive total feedwater flow to steam flow deviations

As with the steam flow validation logic, the net effect is that RPV/L will not change but level control will be more sluggish due to the loss of the anticipatory function of the steam flow/feed flow mismatch. Several annunciators will alarm at P603 to alert the operator of this abnormal condition.

Question: RO-40

Comments / Reference: ABN-LEVEL		Rev:	Major: 007	Minor: 003
Number: ABN-LEVEL		Use Category: CONTINUOUS		Major Rev: 007
Title: Unplanned RPV Water Level Change				Minor Rev: 003
				Page: 8 of 17
<p>4.2 <u>HIGH RPV LEVEL</u></p> <p>4.2.1 <u>IF</u> the reactor has scrammed, <u>THEN</u> PERFORM PPM 3.3.1. _____</p> <p>4.2.2 IF RPV level GT +54.5" is imminent, AND reactor power is GE 5%, <u>THEN</u> SCRAM the Reactor per PPM 3.3.1]. _____</p> <p>4.2.3 <u>IF</u> RPV level GT +108" is imminent, <u>THEN</u> PERFORM the following: _____</p> <p> a. <u>IF</u> the Reactor Mode Switch is in STARTUP or RUN, <u>THEN</u> SCRAM the Reactor per PPM 3.3.1. _____</p> <p> b. CLOSE the inboard MSIVs. _____</p> <p> c. CLOSE RCIC-V-63 and RCIC-V-76. {P-26365} _____</p>				

Number: ABN-LEVEL	Use Category: CONTINUOUS	Major Rev: 007
Title: Unplanned RPV Water Level Change		Minor Rev: 003 Page: 5 of 17

4.0 SUBSEQUENT OPERATOR ACTIONS {P-28670}

NOTE: Section 4.1 provides actions for LOW RPV level.
Section 4.2 provides actions for HIGH RPV level

NOTE: FWLCS Channel A and RFW-LI-606A receive a signal from RFW-DPT-4A.
FWLCS Channel B and RFW-LI-606B receive a signal from RFW-DPT-4B.
FWLCS Channel C and RFW-LI-606C(D) receive a signal from RFW-DPT-4C.

4.1 LOW RPV LEVEL

- 4.1.1 IF RPV level is LE +13"
THEN **SCRAM the Reactor** per PPM 3.3.1,
AND **REFER** to PPM 5.1.1. _____
- 4.1.2 IF the selected FWLCS instrument is not consistent with other RPV level instruments,
THEN **SELECT** a new FWLCS RPV level channel at H13-P603. _____
- 4.1.3 IF total feedwater flow is LT total steam flow,
THEN **PERFORM** the following:
 - a. **INCREASE** feedwater flow until total feedwater flow is GE total steam flow. _____
 - b. **VERIFY** RFW injection flow path to RPV is **OPEN**. _____
- 4.1.4 IF feedwater flow cannot be raised to meet or exceed steam flow,
THEN **REDUCE** RRC flow to approximately 74 Mlb/hr per PPM 3.2.4 as required to restore RPV level to +36". _____

Question: RO-41

Examination Outline Cross-reference: RO-41

Revision: 2

Date: 8/20/22

Tier: 2

Group: 1

K/A Number: 261000 K6.01

Level of Difficulty: 3

RO Importance Rating: 3.5

K/A Description: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Standby Gas Treatment System: AC Distribution

CGS is at 100% power.

- E-CB-71/7B trips on overcurrent.

What is the effect of this malfunction on Standby Gas Treatment?

- A. SGT-A Lag Fan and SGT-B Lag Fan will lose power.
- B. SGT-A Lead Fan and SGT-B Lag Fan will lose power.
- C. SGT-A Lag Fan and SGT-B Lead Fan will lose power.
- D. SGT-A Lead Fan and SGT-B Lead Fan will lose power.

Answer: B

K/A Match:

Shows a knowledge of the effect a loss of AC power will have on Standby Gas Treatment.

Question: RO-41

Explanation: A trip of E-CB-71/7B, 480V FEEDER BKR FOR E-MC-7B, will de-energize E-MC-7B. E-MC-7B supplies to E-DISC-7B3C, DISCONNECT FOR E-MC-7BB. If E-MC-7BB will also de-energize. All answers are plausible because the sources of power to the fans and which are lead and lag must be known. The answers cannot be derived.

- SGT-FN-1A1 is the SGT-A lead fan and is powered by E-MC-7BB.
- SGT-FN-1A2 is the SGT-A lag fan and is powered by E-MC-8BB
- SGT-FN-1B1 is the SGT-B lag fan and is powered by E-MC-7BB
- SGT-FN-1B2 is the SGT-B lead fan and is powered by E-MC-8BB

A. Incorrect: See explanation above

B. Correct: See explanation above.

C. Incorrect: See explanation above

D. Incorrect: See explanation above

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SOP-ELEC-AC-LU, AC Electrical Distribution System Breaker Lineup	
Asset Suite	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 11955 - State the power supplies to the following Standby Gas Treatment System components: Fans

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level:

Requires the student to know which fans are lead or lag and what the power sources are to each fan.

10 CFR Part 55 Content: 55.41 7

Question: RO-41

Comments / Reference: SOP-ELEC-AC-LU	Rev:	Major:	091	Minor:	N/A
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Question: RO-41

Excerpts from SOP-ELEC-AC-LU:

SL-71 (E502-2)			
REA-CB-FN1A	RW 467	REA-FN-1A	RI
E-CB-71/7B	RW 467	E-MC-7B	RI/CL
E-CB-71/7C	RW 467	E-MC-7C	RI/CL
E-CB-71/7AB	RW 467	E-MC-7AB	RI/CL
RCC-CB-P1A	RW 467	RCC-P-1A	RI
CCH-CB-CR1A	RW 467	CCH-CR-1A	RI

E-MC-7B (E503-8)			
SGT-DISC-7B1BL	RB 522	SGT-ESH-1A	ON
SGT-DISC-7B1BR	RB 522	SGT-ESH-2A	ON
CRA-42-7B1C	RB 522	CRA-FN-5C	ON
E-DISC-7B1D	RB 522	E-MC-7BA	ON
CRA-42-7B2A	RB 522	CRA-FN-5A	ON
CRA-42-7B2B	RB 522	CRA-FN-4A	ON
RRA-42-7B2C	RB 522	RRA-FN-11	ON
CRA-42-7B2D	RB 522	CRA-FN-1A2	ON
CRA-42-7B3A	RB 522	CRA-FN-3A	ON
E-42-7B3B	RB 522	Spare	OFF
E-DISC-7B3C	RB 522	E-MC-7BB	ON

E-MC-7BB (E503-12)			
E-DISC-7BB1C	RB 572	Spare	OFF
RRA-42-7BB1E	RB 572	RRA-FN-13	ON
RRA-42-7BB1F	RB 572	RRA-FN-15	ON
SGT-DISC-7BB2AL	RB 572	SGT-EHC-1B1	ON
SGT-DISC-7BB2AR	RB 572	SGT-EHC-1A1	ON
CMS-DISC-7BB2B	RB 572	CMS-SR-20	ON
SGT-42-7BB2C	RB 572	SGT-FN-1A1	ON
SGT-42-7BB2D	RB 572	SGT-FN-1B1	ON

E-MC-8BB (E503-12)			
SGT-DISC-8BB1BL	RB 572	SGT-EHC-1A2	ON
SGT-DISC-8BB1BR	RB 572	SGT-EHC-1B2	ON
E-DISC-8BB1CL	RB 572	Spare	OFF
E-DISC-8BB1CR	RB 572	Spare	Locked OFF
RRA-42-8BB1E	RB 572	RRA-FN-14	ON
SGT-42-8BB1F	RB 572	SGT-FN-1B2	ON
SGT-42-8BB2B	RB 572	SGT-FN-1A2	ON

Question: RO-41

Comments / Reference: Asset Suite						Rev:	Major:000	Minor:N/A
Excerpt from Asset Suite:								
System	Equip	Equip	Comp	Comp	Equip	Name		
SGT	FN	1A1			AC...	SGT-FU-1A EXHAUST LEAD FAN TO ELEVATED DUCT		
SGT	FN	1A1+			NO...	DELETED		
SGT	FN	1A2			AC...	SGT-FU-1A EXHAUST LAG FAN TO ELEVATED RELEASE DUCT		
SGT	FN	1A2+			NO...	NONEQUIP (DUPLICATE OF 2-SGT-FN-1A2)		
SGT	FN	1B1			AC...	SGT-FU-1B EXHAUST LAG FAN TO ELEVATED RELEASE DUCT		
SGT	FN	1B1+			NO...	NONEQUIP (DUPLICATE OF SGT-FN-1B1)		
SGT	FN	1B2			AC...	SGT-FU-1B EXHAUST LEAD FAN TO ELEVATED RELEASE DUCT		
SGT	FN	1B2+			NO...	NONEQUIP (DUPLICATE OF 2-SGT-FN-1B2)		

Question: RO-42

Examination Outline Cross-reference: 42

Revision: 3

Date: 5/19/22

Tier: 2

Group: 1

K/A Number: 262001.K5.01

Level of Difficulty: 3

RO Importance Rating: 3.9

K/A Description: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the AC Electrical Distribution: Paralleling AC sources

CGS is in Mode 5.

A special test is in progress that powers SM-4 from DG-3.

The special test is complete and operations is transferring SM-4 from DG-3 to TR-S

The CRO places the E-CB-4/2 Sync Selector switch to BUS 2 and notices that the associated synchroscope is rotating FAST in the SLOW direction.

How should the CRO align the system for the transfer of SM-4?

The CRO should adjust DG-3 (1) so that the synchroscope is rotating SLOW in the (2) direction.

- A. (1) Voltage Reg Control switch
(2) SLOW
- B. (1) Voltage Reg Control switch
(2) FAST
- C. (1) Electric Governor Control switch
(2) SLOW
- D. (1) Electric Governor Control switch
(2) FAST

Answer: C

K/A Match:

Requires knowledge of process to parallel DG-3 with TR-S

Explanation:

A. Incorrect. Plausible since (2) is correct. Plausibility is enhanced since, prior to paralleling DG3 to TR-S, DG3 voltage should be adjusted slightly lower than SM-4 voltage. However, the DG3 Electric Governor Control switch is adjusted to adjust the synchroscope.

B. Incorrect. Plausible since the synchroscope should be adjusted SLOW in the FAST direction when shifting SM-4 from TR-S to DG3. However, since the incoming frequency inputs to the synchroscope are constant, DG3 speed should be adjusted so that the synchroscope is moving SLOW in the SLOW direction to ensure that TR-S picks up real load on the transfer.

C. Correct. The DG3 Electric Governor Control switch is used to adjust DG3 speed (frequency) so that DG3 frequency is slightly below TR-S frequency to ensure that TR-S picks up real load on the transfer. This is represented by the synchroscope moving SLOW in the SLOW direction.

D. Incorrect. Plausible since (1) is correct. However, since the incoming frequency inputs to the synchroscope are constant, DG3 speed should be adjusted so that the synchroscope is moving SLOW in the SLOW direction to ensure that TR-S picks up real load on the transfer.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SOP-ELEC-4160V-OPS, 4160 Volt AC Electrical Power Distribution System Operation	
SOP-DG3-START, High Pressure Core Spray Diesel Generator Start	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: EDG-011 Evaluate the following Emergency Diesel Generators indications/responses and determine if the indication/ response is expected and normal. a. Describe manual starting, synchronizing and loading of an EDG to a critical bus

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Question: RO-42

Requires candidate to synthesize knowledge of synchroscope operation with an understanding of the procedure for paralleling the DG with an AC source.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SOP-ELEC-4160V-OPS	Rev:	Major:	021	Minor:	001
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Number: SOP-ELEC-4160V-OPS	Use Category: CONTINUOUS	Major Rev: 021
Title: 4160 Volt AC Electrical Power Distribution System Operation		Minor Rev: 001 Page: 44 of 49

5.25 Transfer of SM-4 from DG-3 to TR-S

- 5.25.1 **VERIFY** SM-2 is energized from TR-S. _____
- 5.25.2 IF E-CB-2/4 is open,
THEN **PERFORM** the following:
 - a. **PLACE** the E-CB-2/4 Sync Selector switch to **MAN**. _____
 - b. **CLOSE** E-CB-2/4. _____
 - c. **PLACE** E-CB-2/4 Sync Selector switch in **OFF**. _____
- 5.25.3 **PLACE** E-CB-4/2 Synchronizing Selector switch in **BUS 2**. _____
- 5.25.4 **PLACE** the HPCS-RMS-DG3/DROP switch to **PARA** (E-CP-DG/RP3). _____
- 5.25.5 **VERIFY** the Diesel Generator Mode Selector switch in **CR** (Control Room). _____

CAUTION

Failure to parallel in the described manner may result in a reverse power trip of the diesel.

NOTE: Due to synchroscope selector switch wiring, the synchroscope is required to rotate **SLOWLY** in the **SLOW** direction when transferring loads from the running diesel to TR-S.

- 5.25.6 **ADJUST** Diesel Engine Governor Control switch (diesel speed) until the synchroscope is moving **SLOW** in the **SLOW** direction (CCW). _____
- 5.25.7 **ADJUST** Diesel Generator Voltage Reg Control switch until BUS 4 voltage is slightly higher than INCOMING voltage. _____
- 5.25.8 WHEN the sync scope is ~5 minutes before the 12 o'clock position,
THEN **CLOSE** E-CB-4/2. _____
- 5.25.9 **VERIFY** E-CB-4/2 **CLOSED**. _____
- 5.25.10 **IMMEDIATELY VERIFY** KVARs are zero or slightly positive.
(Adjust Diesel Generator Voltage Reg control switch, if necessary) _____
- 5.25.11 **IMMEDIATELY VERIFY** KW is GE 100 KW.
(Adjust Electric Governor control switch, if necessary). _____

Comments / Reference: SOP-DG3-START	Rev:	Major: 034	Minor: N/A
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Number: SOP-DG3-START	Use Category: CONTINUOUS	Major Rev: 034 Minor Rev: N/A Page: 26 of 42
Title: High Pressure Core Spray Diesel Generator Start		

5.4 Paralleling DG-3

NOTE: Due to Diesel Generator voltage regulator design, it is possible to have indicated generator output voltage near bus voltage and, when paralleled, have the Diesel Generator in an overexcited condition with the KVAR meter deflected GT 2000 KVAR right of zero. This results from continuing to attempt to raise generator output voltage beyond the no load capacity of the static exciter voltage regulator.

To recover, the Voltage Regulator Control Switch should be placed in the LOWER position until KVAR indication comes back to the normal operating range. To minimize the possibility of overexciting the Diesel Generator, the operator should not attempt to raise generator output voltage after an increase in generator output voltage is no longer observed.

5.4.1 IF paralleling from the CONTROL ROOM,
THEN PERFORM the following:

- a. **VERIFY** Diesel Generator Mode Selector is in **CR** (H13-P601). _____
- b. **PLACE** HPCS-RMS-DG3/DROOP in **PARA** (E-CP-DG/RP3). _____
- c. **VERIFY** CB-4DG3 Synchronizing Selector Switch is in **DGEN/BUS** (H13-P601). _____
- d. **ADJUST** frequency using Electric Governor control switch, until synchroscope is moving slowly in the fast (clockwise) direction (H13-P601). _____

NOTE: Diesel Generator Voltage Reg Control switch response time may be slower than expected due to temperature sensitivity. The switch may be initially sluggish when cold. DG-3 is considered operable, provided the voltage can be adjusted {AR-2.23}

NOTE: PDIS signals X125 (SM-4 voltage) and X108 (DG-3 voltage) should be used to determine the incoming voltage is slightly higher than the running voltage as well as noting when an increase in generator output voltage is no longer observed.

- e. **ADJUST** voltage using Diesel Generator Voltage Reg control switch, until generator (incoming) voltage is slightly higher than bus (running) voltage (H13-P601). _____
- f. IF available,
THEN VERIFY PDIS signal X108 (DG-3 voltage) is slightly higher than X125 (SM-4 voltage). _____

Examination Outline Cross-reference: 43

Revision: 0

Date: 8/15/22

Tier: 2

Group: 1

K/A Number: 262002.K6.01

Level of Difficulty: 3

RO Importance Rating: 3.4

K/A Description: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Uninterruptable Power Supply (AC/DC): AC electrical distribution

CGS is in Mode 1.

A condition on the normal AC input to E-IN-1 causes the static switch to automatically swap to its Alternate AC source.

The condition that caused the swap has cleared.

E-IN-1 Auto Retransfer is enabled.

Which of the following conditions is/are required to allow the static switch to automatically transfer the load back to the inverter?

- (1) All alarms on E-IN-1 are reset.
- (2) The Alternate AC Source and Inverter are In-Sync.
- (3) E-IN-1 AC output current is LE 100% of inverter rating.

- A. (1) ONLY
- B. (2) ONLY
- C. (1) and (3) ONLY
- D. (2) and (3) ONLY

Answer: D

K/A Match:

Requires knowledge of requirements for automatic retransfer following an issue with the AC input to UPS inverter.

Explanation:

- A. Incorrect. Plausible since some of the alarms on E-IN-1 represent conditions that will prohibit auto retransfer. However, all alarms do not need to be reset to allow auto retransfer.
- B. Incorrect. Plausible since (2) is required to allow auto retransfer. However, (3) is also required.
- C. Incorrect. Plausible since (3) is correct. However, (1) is not correct.
- D. Correct. In accordance SD000194, UPS System Description, both (2) and (3) are required for the E-IN-1 Static Switch to auto retransfer loads back to the inverter.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000194, UPS System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 5891 State the purpose and various functions of the following with respect to E-IN-1 and E-IN-5:
 c. Static Switch
 (i) Transfer Criteria

Question Source: Bank #: LO04063 (65800)
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires knowledge of requirements for auto retransfer

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SD000194	Rev:	Major: 17	Minor: 1
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e) Static Switch Retransfer Criteria:

LO-5891

(1) The static switch will automatically retransfer the load to the inverter following an automatic static switch transfer assuming all of the following conditions are true.

- (a) Auto Retransfer is enabled.
- (b) Alternate Source and Inverter are In-Sync.
- (c) Auto transfer condition cleared.
- (d) AC output current is less than or equal to 100% of system rating.
- (e) Auto Retransfer time delay has expired.

If any of these conditions is not met, the load will not automatically transfer back to the inverter.

Question: RO-44

Examination Outline Cross-reference: 44

Revision: 0

Date: 5/17/22

Tier: 2

Group: 1

K/A Number: 263000.A3.02

Level of Difficulty: 2

RO Importance Rating: 3.5

K/A Description: Ability to monitor automatic operation of the DC Electrical Distribution, including: Breaker trips

CGS is in Mode 1.

125 VDC battery chargers are aligned as follows:

- Charger E-C1-1A is powering bus S1-1.
- Charger E-C1-2B is powering bus S1-2.

The supply breaker to MC-8A trips open.

With no operator action, what is the status of the 125 VDC buses 2 minutes later?

S1-1 is powered from its (1) and S1-2 is powered from its (2).

- A. (1) battery
(2) battery
- B. (1) battery
(2) battery charger
- C. (1) battery charger
(2) battery
- D. (1) battery charger
(2) battery charger

Answer: C

K/A Match:

Requires knowledge of automatic actions (bus supplied by battery) on a tripped breaker.

Question: RO-44

Explanation:

- A. Incorrect. Plausible since (2) is correct. Plausibility is enhanced if it is believed that battery chargers for each DC bus are cross-tied between divisions and automatically swap on a loss of power to one charger. However, battery chargers are not cross-tied between divisions and the standby charger must be manually started. Since both battery chargers for bus S1-2 are powered from MC-8A, S1-2 will be powered from its respective battery.
- B. Incorrect. Plausible if it is believed that battery chargers for bus S1-1 are powered from MC-8A. However, Division 1 chargers are powered from MC-7A while Division 2 chargers are powered from MC-8A.
- C. Correct. Each Division of 125VDC has two battery chargers. Both divisional chargers are powered from the same supply; Division 1 from MC-7A and Division 2 from MC-8A. One charger is supplying the bus while the other is in standby. The standby charger must be manually aligned. On a loss of MC-8A, both Division 2 chargers lose the ability to supply bus S1-2. Therefore, the bus is powered from its respective battery. Division 1 chargers are not affected and the online charger will continue to power bus S1-1.
- D. Incorrect. Plausible since (1) is correct. Plausibility is enhanced enhanced if it is believed that battery chargers for each DC bus are cross-tied between divisions and automatically swap on a loss of power to one charger. However, battery chargers are not cross-tied between divisions and the standby charger must be manually started.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000188, DC Distribution System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: DC-02 Describe the DC Distribution system design feature(s) or interlocks that provide for the following:
 d. Battery charging methods
 f. Divisional separation

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

Question History: Last NRC Exam: N/A

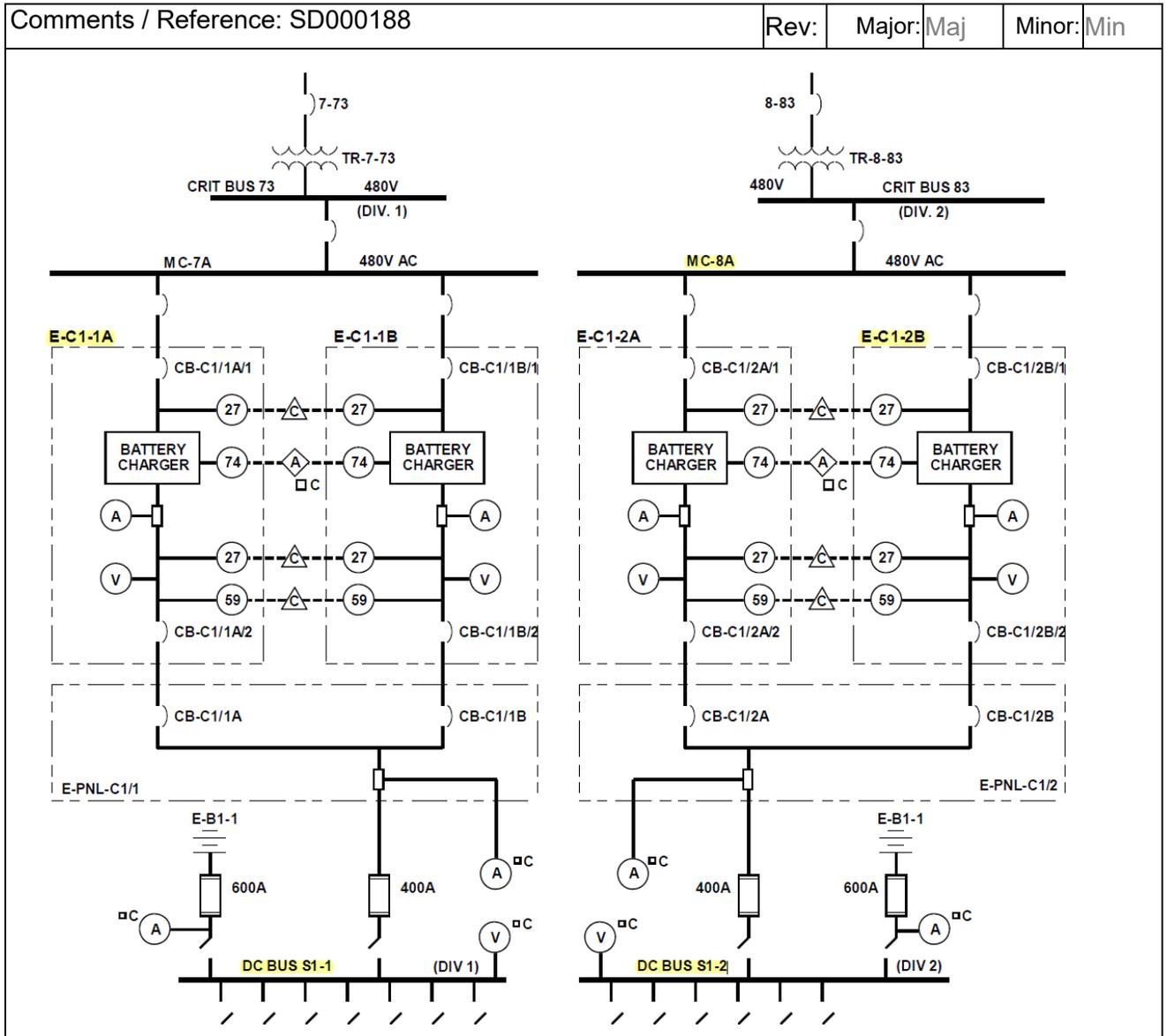
Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Question: RO-44

Justification for Cognitive Level

Requires candidate to synthesize a knowledge of the methods of powering DC buses and when each is used with a knowledge of battery charger power supplies.

10 CFR Part 55 Content: 55.41 7



Question: RO-45

Examination Outline Cross-reference: 45

Revision: 0

Date: 5/17/22

Tier: 2

Group: 1

K/A Number: 264000.A3.04

Level of Difficulty: 2

RO Importance Rating: 4.0

K/A Description: Ability to monitor automatic operation of the Emergency Generators, including: Frequency and voltage control

CGS is in Mode 1.

The Monthly Operability Test of DG-1 is in progress in accordance with OSP-ELEC-M701.

- The Excitation Mode Selector Switch is in PARA.
- The Engine Speed Selector Switch is in IDLE.
- DG-1 is running at Idle Speed (400 rpm).
- DG-1 Output Breaker, CB-DG1/7 is OPEN.

A Loss of Offsite Power (LOOP) occurs concurrent with a Main Turbine trip.

How does DG-1 respond and what actions should be taken?

DG-1 will...

- A. REMAIN at idle speed UNTIL the Engine Speed Selector Switch is placed in RATED. DG-1 WILL THEN automatically ramp to rated speed and CB-DG1/7 will automatically CLOSE.
- B. automatically RAMP to rated speed with NO operator action. CB-DG1/7 will automatically CLOSE. Place the Engine Speed Selector switch in RATED.
- C. automatically RAMP to rated speed with NO operator action. CB-DG1/7 will NOT automatically CLOSE. Placing the Engine Speed Selector switch in RATED allows CB-DG1/7 to automatically CLOSE.
- D. REMAIN at idle speed. Place the Engine Speed Selector Switch in RATED. MANUALLY RAMP DG-1 to rated speed. Manually CLOSE CB-DG1/7.

Answer: B

K/A Match:

Requires knowledge of DG automatic speed (frequency) control during a Loss of Offsite Power.

Question: RO-45

Explanation:

A. Incorrect. Plausible since DG speed will normally remain at 400 rpm with the Speed Selector switch in IDLE. Plausibility is enhanced since the DG output breaker will automatically close. However, during a LOOP, the DG will automatically go to rated speed regardless of Speed Selector switch position.

B. Correct. A Loss of Offsite Power (LOOP) will cause an undervoltage condition on SM-7. This will send a start signal to DG-1 (already running). DG-1 will come up to rated speed and voltage regardless of Speed Selector switch position. CB-DG1/7 will automatically close to reenergize SM-7. OSP-ELEC-M701 directs operators to place the Speed Selector switch in RATED to follow plant conditions.

C. Incorrect. Plausible since DG-1 will come up to rated speed regardless of Speed Selector switch position. However, CB-DG1/7 will automatically close without taking the Speed Selector switch to RATED.

D. Incorrect. Plausible since DG speed will normally remain at 400 rpm with the Speed Selector switch in IDLE and it is normally required to manually close CB-DG1/7. However, on a LOOP, DG-1 will come up to rated speed and voltage regardless of Speed Selector switch position. CB-DG1/7 will automatically close to reenergize SM-7.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000200, DG System Description	
OSP-ELEC-M701, Diesel Generator 1 – Monthly Operability Test	
SD000182, AC Distribution System Description	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: EDG-04 Describe the function, operation, interlocks, trips, physical location, and power supplies of the following Emergency Diesel Generators components.

- a. EDG1(2) and HPCS DG trips/emergency stop
- b. EDG 1(2) and HPCS DG Voltage Regulator control switch
- c. EDG 1(2) and HPCS DG Governor control switch

Question Source: Bank #: 65905
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize an understanding of the DG-1 response to a LOOP with the knowledge of required operator actions for the conditions given.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SD000200	Rev:	Major:	15	Minor:	0
<p>b) In IDLE, High Drywell Pressure or Low RPV level has no effect on idle speed; but with UV on bus, the engine ramps to rated speed automatically. If in IDLE & UV occurs, personnel are directed to take switch to RATED (follow indication only). IDLE feature allows warm-up on tests.</p>					

Comments / Reference: SD000182	Rev:	Major:24	Minor:0
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The following is a sequential listing of the automatic actions that would occur for a loss of offsite power and main generator trip at time $T=0$ seconds with TR-S and TR-B not available:

T=0 seconds

- a) Trip and lock out the 500 KV breakers
- b) Trip and lock out the "N" breakers
- c) "S" breakers prevented from closing
- d) Trip non-critical/critical bus tie breakers 1-7, 2-4, 3-8
- e) Trip 4160/480 volt tie breakers 1-11, 2-21, 3-31
- f) Trip stub bus tie breakers 75-72 & 85-82
- g) Trip SM-2 loads, RRC, TSW and TMU pumps
- h) Start DG1 and 2

T=2 seconds

Trip breaker 4-2

T=3 seconds

- a) Start DG3
- b) Trip breakers 7-1 & 8-3
- c) Trip SM-7 and 8 loads

T = 5 seconds

a) Trip SM-1 and 3 loads

T = 10 seconds (approx.- based on DG start time)

Close DG1(2) breakers, DG1-7(DG2-8) to supply SM-7 & 8

T = 13 seconds (approx.- based on DG start time)

Close DG3 breaker, 4-DG3

T = 20 seconds (10 sec. after bus re-energized)

Start standby TSW pump

T = 30 seconds (20 sec. after bus re-energized)

Start SW pump A and B

Comments / Reference: OSP-ELEC-M701	Rev:	Major:063	Minor:001
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Number: OSP-ELEC-M701	Use Category: CONTINUOUS	Major Rev: 063
Title: Diesel Generator 1 - Monthly Operability Test		Minor Rev: 001
		Page: 8 of 45

4.6 If a loss of power occurs while the Engine Speed Selector switch is in the IDLE position, the Diesel Generator will automatically ramp up to rated speed with no operator action. The Engine Speed Selector switch should be placed in the RATED position on loss of power to match the actual (desired) condition.

Question: RO-46

Examination Outline Cross-reference: 46

Revision: 0

Date: 6/2/22

Tier: 2

Group: 1

K/A Number: 300000.2.1.27

Level of Difficulty: 2

RO Importance Rating: 3.9

K/A Description: Instrument Air: Knowledge of system purpose and/or function.

CGS is in Mode 2.

A reactor plant startup is in progress in accordance with PPM 3.2.1, Reactor Plant Startup.

CAS compressor configuration is:

- CAS-C-1A in RUN.
- CAS-C-1B in STANDBY.
- CAS-C-1C tagged out.

Due to startup activities, CAS header pressure drops to 98 psig.

- CAS-C-1B automatically starts.

15 minutes after CAS-C-1B starts, CAS header pressure has returned to the normal band.

How does the CAS system respond to this transient?

- A. The UNLOAD light will illuminate when CAS header pressure rises to 105 psig.
- B. The UNLOAD light will illuminate if CAS-C-1B discharge pressure rises to 125 psig.
- C. CAS-C-1B will run until CAS header pressure reaches 105 psig and will then automatically turn off.
- D. CAS-C-1B will continue to run until the control room operator manually stops the compressor.

Answer: B

K/A Match:

Requires knowledge of how the CAS system performs its function to maintain sufficient air pressure for its loads.

Question: RO-46

Explanation:

A. Incorrect. Plausible since once started, the standby compressor will run until CAS header pressure reaches 105 psig and the compressor has been running for 30 minutes. However, the UNLOAD light will only illuminate if the compressor is running unloaded (GE 125 psig).

B. Correct. As CAS pressure rises, the standby CAS that started will continue to run until CAS header pressure is GE 105 psig and the compressor has run for at least 30 minutes. Since only 15 minutes has elapsed, the compressor will continue to run. The compressor will run in the unloaded condition when header pressure reaches 125 psig. In this condition the UNLOAD light will illuminate.

C. Incorrect. Plausible since the compressor that started will not turn off until header pressure reaches 105 psig. However, the compressor must be running for at least 30 minutes before it will automatically turn off.

D. Incorrect. Plausible if it is believed that once started, the standby compressor must be manually reset. However, the compressor will automatically turn off once header pressure has reached 105 psig and the compressor has been running for at least 30 minutes.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000205, Control and Service Air System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 5870 - Describe the features associated with each position of the Control Air Compressor control switch (LOCAL, RUN, STANDBY, and STOP).

Question Source: Bank #: 36722
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize knowledge of the requirements for a running CAS compressor to automatically secure with an understanding of the conditions that will cause the CAS UNLOAD light to illuminate.

Question: RO-46

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SD000205	Rev:	Major: 13	Minor: 0
COLUMBIA SYSTEMS CAS		March 2021 SD000205, r13 m0	
V. CONTROL THEORY AND INTERLOCKS (See also Table 1, Local Components, Controls and Indications)		5870	
A. Compressor CAS-C-1A(1B)[1C]			
1. Remote Controls Compressor CAS-C-1A(1B)[1C]			
Four position switch; LOCAL/STOP/STANDBY/RUN maintained contacts			
LOCAL: Allows local starting			
STOP: Stops the compressor			
STANDBY: System pressure switches (CAS-PS-AUTO/1A(1B)[1C]) automatically start the compressor at LE 100 psig. The compressor runs until header pressure returns to 105 psig and for a minimum of 30 minutes.			
RUN: Starts the compressor			
2. Local Controls Compressor CAS-C-1A(1B)[1C]			
Three push buttons: START/RESET/STOP			
START: Starts the compressor when the compressor control switch is in the LOCAL position.			
RESET: Resets the safe shutdown circuit if an auto trip has occurred and the condition has cleared.			
STOP: Stops the compressor regardless of the Control Room switch position. It must be pulled out to allow operation of the compressor from either control position.			
3. CAS Compressor trips			
The compressor automatically trips on:		5872	
High Discharge Pressure > 135 psig		12245	
High discharge air temperature > 240°F (5 second time delay)			
Low Oil Pressure < 10 psig			
Low Cooling Water Pressure < 10 psig			
Motor Overload (compressor or fan)			
The above trips are reset at the local control cabinet. If the motor overload trip is present the reset pushbutton inside the CAS enclosure needs to be reset first.		12246	
NOTE: If the local STOP pushbutton is depressed, the compressor always stops.			
Above each control switch is a COMP UN-LOADED lamp (backlit white). This lamp is lit when the compressor is running unloaded.		5873	

Question: RO-46

4. CAS Compressor Control Theory

a) Start Mode - 0 to 50 psig (0 to 345 kPa)

When the compressor is started, pressure will quickly rise from 0 to 50 psig. During this period, both the pressure regulator valve and the pilot valve are closed. The inlet flow control valve is open and the compressor operates at full capacity. The compressor discharge is isolated from the service line by the minimum pressure check valve, set at approximately 50 psig.

12243a

b) Pressurization Mode - 50 to 115 psig (Normal Operating Mode)

When discharge air pressure rises above 50 psig, the minimum pressure check valve opens and delivers compressed air to the air header. The pressure regulator valve and the pilot valve remain closed keeping the inlet flow control valve open.

12243b

c) Modulating Mode - 110 to 125 psig

When the line pressure reaches 110 psig, pressure regulating valve PRV-10A/B/C opens to supply air to the inlet flow control valve positioner. This reduces the amount of air entering the compressor, until pressure stabilizes. The control system functions continually in this manner between 110 and 125 psig.

12243c

d) Unload Mode - Greater than 125 psig

When there is little or no air demand line pressure will rise until the compressor "unloads" at 125 psig. At this time the pilot valve deenergizes, closing the compressor inlet valve and opening the sump blow down valve. Sump pressure drops to approximately 40 psig.

12243d

When line pressure drops below the reset pressure of 115 psig, the pilot valve energizes, the sump blow down valve closes, and the inlet valve can modulate to control line pressure.

Examination Outline Cross-reference: 47

Revision: 1

Date: 11/7/22

Tier: 2

Group: 1

K/A Number: 400000.K1.02

Level of Difficulty: 2

RO Importance Rating: 3.8

K/A Description: Knowledge of the physical connections and/or cause and effect relationships between the Component Cooling Water System and the following systems: Loads cooled by CCW

CGS is in Mode 1.

- RCC-P-1A is running.
- RCC-P-1C is running.
- RCC-P-1B is in standby.

All control power is lost to RCC-P-1A due to blown fuses.

How does the RCC system respond to this event?

RCC-P-1A will (1) and RCC-V-6, Radwaste/Reactor Building Supply Valve, will (2).

- A. (1) remain running
(2) remain open
- B. (1) remain running
(2) close
- C. (1) stop
(2) remain open
- D. (1) stop
(2) close

Answer: B

K/A Match:

Requires knowledge of the effect on RCC loads on a loss of RCC pump control power.

Explanation:

- A. Incorrect. Plausible since RCC-P-1A will remain running. However, RCC-V-6 will close is control power to RCC-P-1A is lost.
- B. Correct. RCC-P-1A will remain running if control power is lost. In accordance with SD000196, RCC System Description, RCC-V-6 will close if less than 2 RCC pumps are running, as sensed by pump breaker position. When RCC-P-1A control power is lost, breaker position for that pump is lost, although the breaker is still closed. RCC-V-6 senses only one pump running and will close.
- C. Incorrect. Plausible if it is believed that RCC-P-1A stops on a loss of control power. Plausibility is enhanced since RCC-P-1B would start if RCC-P-1A stops and this would cause RCC-V-6 to remain open. However, RCC-P-1A does not stop on a loss of control power, but RCC-V-6 will close (see explanation B above).
- D. Incorrect. Plausible since (2) is correct. Plausibility is enhanced if it is believed that a loss of control power to RCC-P-1A will cause the pump to stop. However, RCC-P-1A will not stop on a loss of control power.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000196, RCC System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: RCC-02 Describe the Reactor Closed Cooling Water system design feature(s) or interlocks that provide for the following:

- a. Automatic start of standby pump
- b. Primary Containment isolation

Question Source: Bank #: LO02969/31227
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize an understanding of RCC pump operation with respect to control power with a knowledge of the operation of RCC-V-6.

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Comments / Reference: SD000196	Rev:	Major:	Maj	Minor:	Min
<p>b) RADWASTE/RX BLDG SUPPLY RCC-V-6 Three-position switch: CLOSE, NOR, OPEN, Spring return to NOR</p> <p>CLOSE - the valve closes</p> <p>NOR - the valve closes automatically if less than two pumps are online as sensed by breaker position (plus a 10 second time delay).</p> <p>OPEN - the valve opens if at least two pumps are online as sensed by breaker position</p> <p>This valve closes if any of the following actions are done:</p> <ul style="list-style-type: none">• Removing the control power fuses from RCC-P-1A or from valve RCC-V-6 itself• Closing the RCC-V-6 breaker					

Examination Outline Cross-reference: 48

Revision: 0

Date: 6/6/22

Tier: 2

Group: 1

K/A Number: 510000.A1.05

Level of Difficulty: 3

RO Importance Rating: 3.3

K/A Description: Ability to predict and/or monitor changes in parameters associated with operation of the Service Water System, including: Service water flow

CGS is in Mode 1.

SW System B is running for peroxide addition.

RHR-V-68B (RHR-HX-1B Service Water outlet valve) is inadvertently closed.

SW-FI-9B (SW Loop B) will indicate (1) rated flow.

RHR-V-68B receives and open signal **DIRECTLY** on (2) .

- A. (1) normal
(2) high Drywell pressure
- B. (1) normal
(2) SW-P-1B start
- C. (1) less than
(2) high Drywell pressure
- D. (1) less than
(2) SW-P-1B start

Answer: D

K/A Match:

Requires knowledge of changes in service water flow with a system failure.

Explanation:

- A. Incorrect. Plausible (1) if the candidate believes that the RHR Heat Exchanger only receives flow when RHR is in service. Many plant system heat exchangers have cooling water isolated when not in service. If flow through the HX was bypassed when RHR is not in service it would not impact of change flow through the system when the outlet valve to the RHR HX was closed..
- B. Incorrect. Plausible (1) if the candidate believes that the RHR Heat Exchanger only receives flow when RHR is in service. Many plant system heat exchangers have cooling water isolated when not in service. If flow through the HX was bypassed when RHR is not in service it would not impact of change flow through the system when the outlet valve to the RHR HX was closed. (2) Correct, RHR-V_68B receives an open signal from the start of SW-P-1B.
- C. Incorrect. Plausible (1) Correct - SW to RHR heat exchangers is not dependent on RHR system being in service.
- D. Correct - (1) Correct - SW to RHR heat exchangers is not dependent on RHR system being in service. (2) Correct, RHR-V_68B receives an open signal from the start of SW-P-1B.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000204, Standby Service Water System Description	
OSP-SW/IST-Q701, Standby Service Water Loop A Operability	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 117847 - Describe the Service Water System design feature(s) or interlocks that provide for the following:

- a. SW-P-1A(B), HPCS-P-2 Automatic pump starts
- b. SW-V-2A(B), 12A(B), RHR-V-68A(B) Automatic valve alignments

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

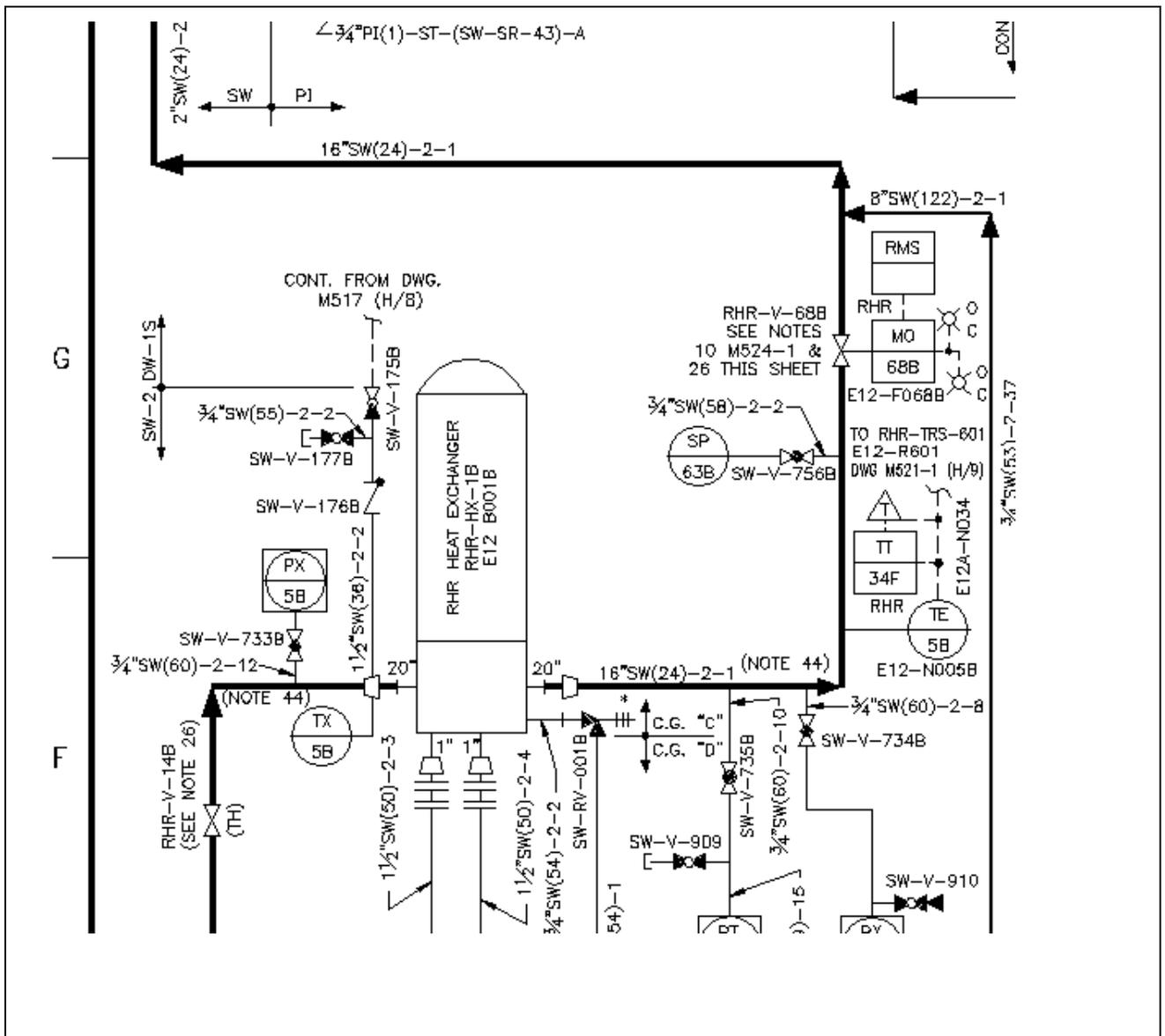
Justification for Cognitive Level

Requires the candidate to synthesize a knowledge of the SW system flowpaths during a SW pump operation with an understanding of the effects of a failure on SW flow.

10 CFR Part 55 Content: 55.41 5

Comments / Reference: SD000204	Rev:	Major:	15	Minor:	N/A

Comments / Reference: M524-2	Rev:	Major:	124	Minor:	N/A
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Question: RO-49

Examination Outline Cross-reference: 49

Revision: 0

Date: 6/6/22

Tier: 2

Group: 1

K/A Number: 203000.A3.08

Level of Difficulty: 2

RO Importance Rating: 4.0

K/A Description: Ability to monitor automatic operation of the RHR/LPCI Injection Mode, including: System initiation sequence.

CGS is in Mode 1.

A LOCA occurs.

How does the RHR-C system respond?

RHR-P-2C will start (1) after Drywell pressure reaches 1.68 psig. LPCI injection valve, RHR-V-42C will open (2).

- A. (1) immediately
(2) immediately after RHR-P-2C starts
- B. (1) immediately
(2) when RPV pressure is LT 470 psig
- C. (1) 5 seconds
(2) immediately after RHR-P-2C starts
- D. (1) 5 seconds
(2) when RPV pressure is LT 470 psig

Answer: B

K/A Match:

Requires knowledge of the LPCI initiation sequence.

Question: RO-49

Explanation:

A. Incorrect. Plausible since RHR-P-2C will start immediately upon an 'F' signal (DW pressure GT 1.68 psig). However, RHR-V-42C will not open until RPV pressure is LT 470 psig.

B. Correct. RHR-P-2C will start immediately upon RPV Level LT -129 inches or DW pressure GT 1.68 psig. RHR-V-42C does not open until RPV pressure is LT 470 psig.

C. Incorrect. Plausible since RHR-P-2A and 2B start 5 seconds after an 'F' signal. However, RHR-P-2C starts immediately after an 'F' signal. Additionally, RHR-V-42C does not open until RPV pressure is LT 470 psig.

D. Incorrect. Plausible since RHR-V-42C will not open until RPV pressure is LT 470 psig. Plausibility is enhanced since RHR-P-2A and 2B start 5 seconds after an 'F' signal. However, RHR-P-2C starts immediately after an 'F' signal.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000198, RHR System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: RHR-02 – Describe the Residual Heat Removal System design feature(s) or interlocks that provide for the following: a. Automatic system initiation/injection.

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize a knowledge of RHR pump start timing for LPCI injection with an understanding of LPCI injection valve interlocks.

10 CFR Part 55 Content: 55.41 7

Question: RO-49

Comments / Reference: SD000198	Rev:	Major: 19	Minor: 0
<p>III. GENERAL DESCRIPTION (Figure 1) (RHR-02, 03, 08, 09, 10, 11, 12)</p> <p>A. <u>LPCI mode</u> is used to automatically restore vessel water level in a loss of coolant accident (LOCA). RHR loops are normally lined up with suction from the suppression pool, heat exchanger valves open, and LPCI injection valves RHR-V-42A, RHR-V-42B & RHR-V-42C shut. Upon an initiation RHR-P-2A, RHR-P-2B, & RHR-P-2C start and when reactor pressure drops below 470 psig the injection valves open (See Figure 1A). Injection flow begins at about 220 psid between the reactor and Wetwell air space. A loss of the LPCI mode of RHR could result in the inability to maintain RPV level above TAF, which could result in a loss of adequate core cooling. A malfunction in the LPCI mode of RHR could result in not developing discharge pressure high enough to satisfy the ADS system logic. This would result in a condition where RPV level cannot be maintained with LPCI when it is required and the ability to reject water from the suppression pool would be lost.</p> <p>3. RHR-P-2C 3-position switch STOP/AUTO/START (spring return to AUTO)</p> <p>STOP Stops pump. If an initiation signal is in, the amber MANUAL OVERRIDE lamp will come on and until the override is cleared, the pump must be started manually. The override condition is cleared by resetting the initiation logic.</p> <p>AUTO RHR-P-2C will start immediately when SM-8 is not powered from TR-S and after a 9.5 second delay when SM-8 is powered from TR-S.</p> <p>RHR Pump C will trip on:</p> <p>a) SM-8 LT 2870 Volts for 2 sec. or LT 3640 Volts for 8 sec.</p> <p>b) RHR-RLY-86/P2C Trip Lockout Relay.</p> <p>c) There is no pump trip associated with RHR-V-4C closure.</p>			

Question: RO-49

27. RHR-V-42C LPCI Injection

3-position switch CLOSE/AUTO/OPEN (spring return to AUTO)

CLOSE The valve closes. If the initiation signal (LPCI) is present, a MANUAL OVERRIDE amber lamp will come on and until the override is cleared, the valve must be manually opened. The override will clear when the initiation logic is reset or when reactor pressure rises above the injection valve open permissive setpoint of 470 psig.

AUTO **The valve will open if:**

- a) **Respective initiation signal is present; and**
- b) **RPV pressure is less than 470 psig.**

OPEN The valve will open if RPV pressure is less than 470 psig.

Examination Outline Cross-reference: 50

Revision: 0

Date: 4/25/22

Tier: 2

Group: 1

K/A Number: 215003

Level of Difficulty: 2

RO Importance Rating: 3.9

K/A Description: Ability to manually operate and/or monitor in the control room: IRM range switches

CGS is in Mode 2.

A reactor plant startup is in progress.

All IRMs are on Range 6, indicating as follows:

IRM "A" - 70/125 scale

IRM "B" - 65/125 scale

IRM "C" - 45/125 scale

IRM "D" - BYPASSED

IRM "E" - 25/125 scale

IRM "F" - 42/125 scale

IRM "G" - 65/125 scale

IRM "H" - 20/125 scale

IRM "C" is placed on Range 5.

What is effect on the plant?

IRM "C" indicates (1) and a rod block (2) active.

A. (1) 14.5/40 scale
(2) is

B. (1) 14.5/40 scale
(2) is NOT

C. (1) UPSCALE
(2) is

D. (1) UPSCALE
(2) is NOT

Answer: C

K/A Match:

Requires an understanding of the effect on the plant when IRM range switches are operated.

Explanation:

Incorrect. Plausible since (2) is correct. Plausibility is enhanced since 14.5/40 is the same percent of scale as 45/125. However, the 0-40 scale is the same as the 0-40 indication of the 0-125 scale but expanded for easier viewing. Therefore, since 45/125 is beyond the 0-40 scale, Range 5 (0-40 scale) will indicate UPSCALE.

Incorrect. Plausible since 14.5/40 is the same percent of scale as 45/125. Plausibility is enhanced if it is believed that the rod block is only active when the Range Switch is on an even scale since the setpoint for the IRM UPSCALE rod block is 108/125. However, the 0-40 scale is the same as the 0-40 indication of the 0-125 scale but expanded for easier viewing. Therefore, since 45/125 is beyond the 0-40 scale, Range 5 (0-40 scale) will indicate UPSCALE. Additionally, the rod block and scram setpoints are based on the percentage of scale. A setpoint of 108/125 equates to 34.5/40. Since IRM "C" reads UPSCALE, a rod block is active.

Correct. The 0-40 scale is the same as the 0-40 indication of the 0-125 scale but expanded for easier viewing. Therefore, since 45/125 is beyond the 0-40 scale, Range 5 (0-40 scale) will indicate UPSCALE. The rod block and scram setpoints are based on the percentage of scale. A setpoint of 108/125 equates to 34.5/40. Since IRM "C" reads UPSCALE, a rod block is active.

Incorrect. Plausible since (1) is correct. Plausibility is enhanced if it is believed that the rod block is only active when the Range Switch is on an even scale since the setpoint for the IRM UPSCALE rod block is 108/125. However, the rod block and scram setpoints are based on the percentage of scale. A setpoint of 108/125 equates to 34.5/40. Since IRM "C" reads UPSCALE, a rod block is active.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000138, IRM System Description	
4.603.A7, 603.A7 Annunciator Panel Alarms	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: IRM-2 - Describe the Intermediate Range Monitoring System design feature(s) or interlocks that provide for the following: b. Ranging IRMs

Question Source: Bank #: 16378 (LO02651)
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

• Comprehension or Analysis

Justification for Cognitive Level

Candidate must synthesize a knowledge of the requirements for an IRM rod block with an understanding of the relationship between 0-40 and 0-125 ranges of IRMs.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SD000138	Rev: Major: 11 Minor: 0
<p>V. <u>CONTROL THEORY AND INTERLOCKS</u> (Obj. 2, 4, 5, 6, 9, 10, 11, 12)</p> <p>A. <u>Control Room Controls</u></p> <p>1. IRM Channel A (B, C, D, E, F, G, H) Range Switch (Figure 9)</p> <p>a) There is one range switch on the P603 for each IRM Channel. Each range switch corresponds to a recorder on the vertical portion of panel P603. Each recorder has a split linear scale of 0 - 40 and 0 - 125, which represent an equivalent range of reactor power dependent upon which range is selected. Odd numbered ranges (1, 3, 5, 7, and 9) use the 0 - 40 scale while the even numbered ranges use the 0 - 125 scale. Faulty range switches have resulted in half scrams being generated when shifting ranges.</p>	

- d) The 0-40 scale is the same as the 0-40 indication of the 0-125 scale but expanded for easier viewing. The 0-40 scale for each IRM recorder allows movement between IRM even ranges while avoiding rod blocks and scrams. The following table illustrates the results of ranging between ranges 7 and 10. This same relationship exists between all ranges.
- e) A failure of an IRM recorder must be evaluated against Tech Specs to ensure that the minimum number of required IRM channels are available to monitor power.

IRM Range	Indication	Scale	Actual % Power
10	7.5	0 - 125	3%
9	7.5	0 - 40	3%
8	75	0 - 125	3%
7	OFF SCALE HI	0 - 40	3%

Number: 4.603.A7	Use Category: CONTINUOUS	Major Rev: 059 Minor Rev: 002 Page: 25 of 73
Title: 603.A7 Annunciator Panel Alarms		

2-7 ROD OUT BLOCK
(CONTINUED FROM PREVIOUS PAGE)

2-7 WINDOW	SOURCE	AUTOMATIC ACTIONS
ROD OUT BLOCK	Mode Switch - <u>NOT IN RUN</u>	Prevents withdrawal of any Control Rods
	<ul style="list-style-type: none"> • SRM detector wrong position • SRM high (GE 1 x 10⁵ cps) • SRM Inop <ul style="list-style-type: none"> • Function Switch out of OPERATE • Module unplugged • High Voltage low • Negative Voltage low • SRM downscale (LE 0.7 cps) • IRM detector wrong position • IRM high (GE 108/125) • IRM Inop <ul style="list-style-type: none"> • Function Switch out of OPERATE • Module unplugged • High Voltage low • Negative Voltage low • IRM downscale (LE 5/125) • APRM neutron flux high (GE 12%) 	
	Mode Switch - <u>RUN</u>	
	<ul style="list-style-type: none"> • APRM STP upscale alarm (GE 108%) (GE 0.62W + 57.1%) • APRM downscale (LE 5%) • RBM downscale (LE 5%) • Anytime a full scram is in 	

Question: RO-51

Examination Outline Cross-reference: 51

Revision: 0

Date: 8/24/22

Tier: 2

Group: 1

K/A Number: 215005.A2.10

Level of Difficulty: 4

RO Importance Rating: 51

K/A Description: 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 operations: Changes in void concentration

CGS is in Mode 1.

A DEH fault causes RPV pressure to slowly lower.

How does this affect Average Power Range Monitor (APRM) indicated power and what action should be taken?

APRM indicated power will (1). Enter ABN-Pressure, take manual control of DEH Throttle Pressure (TP) Control and (2) demand.

- A. (1) lower
(2) lower
- B. (1) lower
(2) raise
- C. (1) rise
(2) lower
- D. (1) rise
(2) raise

Answer: A

K/A Match:

Requires knowledge of the effects of changing void concentration on APRM output and the necessary actions to mitigate this effect.

Explanation:

- A. Correct. When pressure is lowered by DEH, the boiling boundary is lowered, causing an increase in void concentration. Void coefficient \uparrow (negative) and negative reactivity is added to the core. Therefore APRM output \downarrow . This will cause boiling boundary to rise to a steady state level that is below the original level. In accordance with step 4.1.2 of ABN-Pressure, if reactor pressure is dropping and a DEH failure is suspected, Throttle Pressure (TP) Control is taken to MANUAL and demand is lowered. In manual TP control directly controls governor valve demand signal. Lowering demand will cause the governor valves to close and RPV pressure will rise.
- B. Incorrect. Plausible since (1) is correct. However, in manual, TP control directly controls governor valve demand signal. Raising demand will cause RPV pressure to lower faster, exacerbating the low pressure condition.
- C. Incorrect. Plausible since (2) is correct. However, when pressure is lowered by DEH, the boiling boundary is lowered, causing an increase in void concentration. Void coefficient \uparrow (negative) and negative reactivity is added to the core. Therefore APRM output \downarrow .
- D. Incorrect. Plausible if it is believed that lowering pressure causes reactor power to \uparrow due to increased steam demand. Plausibility is enhanced since (2) is the correct action for conditions that would cause (1) to rise. However, when pressure is lowered by DEH, the boiling boundary is lowered, causing an increase in void concentration. Void coefficient \uparrow (negative) and negative reactivity is added to the core. Therefore APRM output \downarrow . This will cause boiling boundary to rise to a steady state level that is below the original level. In accordance with step 4.1.2 of ABN-Pressure, if reactor pressure is dropping and a DEH failure is suspected, Throttle Pressure (TP) Control is taken to MANUAL and demand is lowered. In manual TP control directly controls governor valve demand signal. Lowering demand will cause the governor valves to close and RPV pressure will rise.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-PRESSURE, Unplanned Reactor Pressure Change	
SD001819, Power Range Neutron Monitor System Description	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: PRNM-12 - Evaluate given key Power Range Neutron Monitoring (PRNM) System parameters, if needed determine a course of action to correct or mitigate the following abnormal condition(s): i. Changes in void concentration

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

Question History: Last NRC Exam: N/A

Question: RO-51

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

Justification for Cognitive Level

Requires candidate to synthesize knowledge of the effects of void concentration on LPRMs/APRMs with a knowledge of actions required when void concentration changes due to RPV pressure changes.

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Comments / Reference: ABN-PRESSURE	Rev:	Major:017	Minor:001
<p>4.1.2 IF Reactor pressure continues to drop, AND a DEH failure is suspected, THEN PERFORM the following (Menu, Turbine Start-Up, Reactor Start):</p> <div style="border: 1px solid green; padding: 5px; margin: 10px 0;"> <p>NOTE: Throttle pressure control in manual directly controls governor valve and/or bypass valve demand signal. Raising the demand signal will cause the valve(s) to open and lowering the demand signal will cause the valve(s) to close. There is no feedback in this mode. This mode is very difficult to control pressure in and would generally not be used.</p> </div> <p>a. SELECT TP AUTO/MANUAL. _____</p> <p>b. SELECT TP MANUAL. _____</p> <p>c. SELECT YES. _____</p> <p>d. SELECT DEMAND LOWER. _____</p>			

Comments / Reference: SD001819	Rev:	Major:5	Minor:2
<p>A. Normal</p> <p>1. LPRM</p> <p>The LPRMs are utilized for the detection and indication of local neutron/heat flux throughout the core while operating in the power range (0 125% of rated reactor power). The output from the LPRM System is also used by the APRM, OPRM, and RBM Systems to institute protective trips, such as scrams and rod blocks.</p> <p>LPRM readings are influenced by reactor conditions since they are located in the core. Two of the larger effects are voids and control rod motion. Detectors located high in the core (C and D level), tend to see reduced thermal flux because of the higher void fraction in this region during high power operations. During low power operations, when there are conditions of high control rod density, the detectors see reduced thermal flux; this time due to control rods absorbing neutrons.</p>			

Question: RO-52

Examination Outline Cross-reference: 52

Revision: 0

Date: 8/11/22

Tier: 2

Group: 1

K/A Number: 262002.A1.03

Level of Difficulty: 4

RO Importance Rating: 3.1

K/A Description: Ability to predict and/or monitor changes in parameters associated with operation of the Uninterruptable Power Supply (AC/DC), including: Lights and alarms

Inverter E-IN-3 is in the following condition:

- INVERTER SUPPLYING LOAD light is LIT.
- IN SYNC light is OUT.

What is the effect on the operation of E-IN-3?

If E-RMS-IN3/S202, ALT SOURCE TO LOAD TRANSFER PB is depressed, the static switch (1) transfer to the Alternate AC source. Transferring E-IN-3 loads to the Alternate AC source using the Manual Bypass Switch, E-MBS-IN/3, is (2) by procedure.

- A. (1) will
(2) allowed
- B. (1) will
(2) NOT allowed
- C. (1) will NOT
(2) allowed
- D. (1) will NOT
(2) NOT allowed

Answer: D

K/A Match:

Requires knowledge of operational limits when the IN SYNC light is not lit.

Question: RO-52

Explanation:

A. Incorrect. Plausible since other inverters (E-IN-1/E-IN-5) will transfer under certain conditions even if the Alternate AC source is not in sync with the inverter. Plausibility is enhanced since the Manual Bypass Switch (MBS) will transfer loads in the given conditions. However, the E-IN-3 static switch will not transfer if the Alternate AC source is not synched. Additionally, transferring loads with the MBS is not allowed in accordance with SOP-ELEC-IN3-OPS.

B. Incorrect. Plausible since (2) is correct. However, However, the E-IN-3 static switch will not transfer if the Alternate AC source is not synched.

C. Incorrect. Plausible since (1) is correct. However, transferring loads with the MBS is not allowed in accordance with SOP-ELEC-IN3-OPS.

D. Correct. If the Alternate AC source is not synchronized with the output of E-IN-3, transfer is prevented. Transferring loads with the MBS is not allowed in accordance with SOP-ELEC-IN3-OPS.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000194, UPS System Description	
SOP-ELEC-INV-OPS, IN3 Operations	
E-IN-3 placard	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: UPS-12 - Evaluate given key Uninterruptible Power System parameters, if needed determine a course of action to correct or mitigate the following abnormal condition(s): b. Frequency changes in the system

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize an understanding of the operation of E-IN-3 with a knowledge of procedural requirements for inverter operation.

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Comments / Reference: SD000194	Rev:	Major: 17	Minor: 1
<p>b) Inverters E-IN-2 and E-IN-3 convert 125 VDC power into 60 Hz, 120/240 volt single phase AC power. Each has only one DC input to their respective inverter. The inverter section of E-IN-2 (E-IN-2A and E-IN-2B) is fed from DC distribution panel S1-2 and the inverter section of E-IN-3 (E-IN-3A and E-IN-3B) is fed from DC distribution panel S1-1.</p>			LO-5896
<p>c) The controls maintain a proper phase relationship between the inverter output and the alternate AC source so that the static switch can transfer between two sources in phase.</p>			LO-5896
<p>(1) The control circuit will match the inverter output frequency with the alternate AC source frequency as long as the alternate AC source frequency remains at 60 ± 1 Hz. The frequency of the alternate AC source is monitored by a circuit which prevents a transfer whenever the frequencies are not synchronized. Additionally, the logic will prevent a transfer from an energized power source to a de-energized power source.</p>			UPS-012B
<p>5. <u>Manual Bypass Switch</u></p>			
<p>a) The Manual Bypass Switch aligns the output to either the static switch output or the alternate source. This switch is a make-before-break, so it is important to ensure power supplies are synchronized before operating the switch.</p>			LO-5893e NLO-12570e

Comments / Reference: SOP-ELEC-IN3-OPS	Rev:	Major:	15	Minor:	N/A
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Number: SOP-ELEC-IN3-OPS	Use Category: CONTINUOUS	Major Rev: 015
Title: IN3 Operations		Minor Rev: N/A Page: 5 of 20

4.0 PRECAUTIONS AND LIMITATIONS

4.1 Observe standard electrical safety precautions when energizing or de-energizing electrical equipment.

4.2 If Out Of Sync light is illuminated, operation of Static Switch Manual Transfer Switch (mechanical make-before-break), could result in paralleling of unsynchronized sources.

5.3 Transfer from E-IN-3A to the Alternate Source

NOTE: This section is for transferring to the Alternate Source and will leave E-IN-3A operating. If transferring to the Alternate Source and shutting down E-IN-3A, then use SOP-ELEC-IN3-SHUTDOWN.

NOTE: H13.P800.C1-8.3, INVERTER IN-3A/3B TROUBLE, will remain locked in when E-IN-2A is transferred to Alternate Power Source through the static switch. While this annunciator is locked in, any additional inverter alarm LED will not cause a reflash of the annunciator.

NOTE: When in Mode 1, 2, or 3, transferring E-IN-3A to Alternate Source to Load, for other than shifting inverters is considered a high risk evolution due to loss of redundant power supplies to E-PP-7AA.

5.3.1 **REFER** to PPM 1.3.76, Integrated Risk Management. _____

5.3.2 **VERIFY IN SYNC LED illuminated (E-PNL-IN/3).** _____

Question: RO-52

Comments / Reference: E-IN-3 Placard	Rev:	Major:	Maj	Minor:	Min
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Question: RO-53

Examination Outline Cross-reference: 53

Revision: 0

Date: 8/18/22

Tier: 2

Group: 2

K/A Number: 201003.A4.01

Level of Difficulty: 4

RO Importance Rating: 3.0

K/A Description: Ability to manually operate and/or monitor in the control room: CRD mechanism temperature

CGS is in Mode 1.

A field operator reports that CRDM temperature for Rod 22-31 is GT 350°F.

The crew is taking actions in accordance with ABN-ROD, Rod Control Faults.

Control Rod 22-31 has been declared "slow" per ABN-ROD.

How does this affect Rod 22-31 operation?

Rod 22-31 scram times will increase due to (1). Normal scram time (2) be exhibited once the CRDM temperature returns to normal.

- A. (1) CRDM seal degradation
(2) will
- B. (1) CRDM seal degradation
(2) will NOT
- C. (1) fluid flashing in the CRDM
(2) will
- D. (1) fluid flashing in the CRDM
(2) will NOT

Answer: C

K/A Match:

Requires knowledge of the operational changes that occurs on a CRDM high temperature.

Question: RO-53

Comments / Reference: ABN-ROD	Rev:	Major: 035	Minor: N/A
<p>4.9.2 <u>IF</u> control rod drive temperature remains GE 350°F, <u>THEN</u> PERFORM the following: {P-159950}</p> <p>a. NOTIFY the on call SNE. _____</p> <div style="border: 1px solid green; padding: 5px; margin: 10px 0;"> <p>NOTE: Further actions in this section need not be taken if the drive temperature can be reduced to LT 350°F within 8 hours.</p> </div> <p>b. <u>IF</u> TS LCO 3.1.4 is met, <u>THEN</u> DECLARE the associated control rod "slow". _____</p> <p>c. <u>IF</u> TS LCO 3.1.4 cannot be met for "slow" control rods, <u>THEN</u> DECLARE the control rod(s) inoperable. _____</p> <p>d. <u>IF</u> the control rod associated with the hot control rod drive was previously declared "slow" for reasons other than high temperature, <u>THEN</u> DECLARE the control rod inoperable. _____</p> <p>e. REFER to TSP-CRD-C101 for setting control rod operability status in the AST Analysis. _____</p>			
<p>4.9.2 This step addresses the problem of increased scram times in control rods which are GT 350°F. The high temperature scram delay is attributed to a fluid flashing phenomenon rather than a seal degradation, therefore normal scram performance is exhibited once the temperature returns to normal operating temperatures. Depending upon SCRAM time evaluations rods may need to be declared inoperable and inserted as directed by the on-call SNE. Steps b. through f. are preceded by a note that allows up to 8 hours to correct the high drive temperature condition (e.g. to see if previous actions taken per this procedure have been effective) before declaring the control rod slow and making a Technical Specification Action Statement entry. This is consistent with the time provided by Technical Specification Action Statement 3.1.5A (for an inoperable control rod scram accumulator) which allows 8 hours to make the "slow" declaration. A hot control rod drive mechanism and inoperable accumulator are the only two conditions (other than actual scram time results) that can result in declaring a control rod "Slow". A drive temperature above 350°F may cause slower scram times due to flashing of water in the over piston volume and piping, reference BWROG Technical Paper TP18-1-231, "Generic Response to GE SIL 173, Supplement 1, Revision 1, High Temperature Control Rod Drives." TSP-CRD-C101 contains explicit instructions for setting control rod status in the AST ANALYSIS program.</p>			

Question: RO-54

Examination Outline Cross-reference: 54

Revision: 0

Date: 8/22/22

Tier: 2

Group: 2

K/A Number: 201004.K1.01

Level of Difficulty: 3

RO Importance Rating: 3.5

K/A Description: Knowledge of the physical connections and/or cause and effect relationships between the Rod Sequence Control System and the following systems: Reactor manual control system

CGS is in Mode 1.

Reactor power is 15%.

Rod 23-18, a control rod in the Rod Worth Minimizer (RWM) latched group, is withdrawn one notch beyond the withdrawal limit.

The RWM displays the following:

- 2 INSERT Errors
- 1 WITHDRAW Error

How is subsequent rod movement affected?

Control rod (1) is blocked on (2).

- A. (1) withdrawal ONLY
(2) rod 23-18 ONLY
- B. (1) withdrawal ONLY
(2) all control rods
- C. (1) withdrawal and insertion
(2) rod 23-18 ONLY
- D. (1) withdrawal and insertion
(2) all control rods

Answer: B

K/A Match:

Requires knowledge of the cause and effect relationship between the Rod Worth Minimizer and the Reactor Manual Control System

Question: RO-54

Explanation:

A. Incorrect. Plausible since (1) is correct. However, the withdrawal block will prevent out motion on all control rods.

B. Correct. When the RWM detects 1 withdrawal error, it will not send a signal necessary to allow the RMCS to withdraw control rods. This is applicable to any control rod that is selected.

C. Incorrect. Plausible since the RWM has 2 INSERTION errors. However, 3 INSERTION errors are required to block rod insertion. Additionally, rod motion restrictions from the RWM are applied to all control rods.

D. Incorrect. Plausible since (2) is correct. Plausibility is enhanced since since the RWM has 2 INSERTION errors. However, 3 INSERTION errors are required to block rod insertion.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000154, Rod Worth Minimizer System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: RWM-02 Describe the Rod Worth Minimizer design feature(s) or interlocks that provide for the following:

- b. Insert blocks/errors
- c. Withdraw blocks/errors

Question Source: Bank #: LO00995 (3805)

Modified Bank #: Mod Bank #. (Note changes or attach parent)

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

Justification for Cognitive Level

Requires knowledge of RWM requirements for withdrawal blocks.

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Question: RO-54

Comments / Reference: SD000154	Rev:	Major: 19	Minor:	N/A
<p>Actual rod position is compared to the desired position as established by the rod sequence. If these rod sequences are violated, the RWM generates rod block signals received by the Rod Drive Control System (within the Reactor Manual Control System) that prevent further rod motion until the rod misposition is corrected. Refer to Attachment 1 for a typical rod withdrawal sequence.</p> <p>9. Withdraw Error. A withdraw error occurs either when a rod contained in the currently latched group or any lower group is withdrawn past the withdraw limit for the group, or if a rod contained in a group higher than the one that is latched is withdrawn past the insert limit for the higher group.</p> <p>10. Insert Error. An insert error occurs when a rod contained in the currently latched group is inserted past the insert limit for this group, or if a rod contained in a group lower than the one that is latched is inserted past the withdraw limit for the lower group.</p> <p>11. Withdraw Block. Withdrawal of all control rods is "blocked" by the absence of a withdraw permissive signal to the reactor manual control system (RMCS). A withdraw block can be caused by rod mispositioning or an RWM program that is different from the pullsheets. A Withdraw Block is imposed when power level is below the LPSP and:</p> <ul style="list-style-type: none"> a) a withdraw error is present, or b) an insert block exists, and a rod not identified as an insert error is selected <p>12. Insert Block. Insertion of all control rods is "blocked" by the absence of an insert permissive signal to the reactor manual control system (RMCS). An insert block occurs when three or more insert errors exist and the RWM is below the LPSP. The number of errors must be reduced to two or less to remove the insert block.</p>				

Question: RO-55

Examination Outline Cross-reference: 55

Revision: 0

Date: 8/23/22

Tier: 2

Group: 2

K/A Number: 202002.291003.K1.03

Level of Difficulty: 3

RO Importance Rating: 3.7

K/A Description: Recirculation Flow Control - Function and operation of flow controller in manual and automatic modes

CGS is in Mode 1.

Both RRC Individual Loop Controllers are in AUTO.

RRC-A Individual Controller HZ DEMAND is lowered by 2 Hz.

How is the RRC Flow Control System affected?

RRC-A Individual Loop Controller Bias will (1) and Master Controller Demand will (2).

- A. (1) remain constant
(2) remain constant
- B. (1) remain constant
(2) lower by 2 Hz
- C. (1) lower (deflect to the left) by 2 Hz
(2) remain constant
- D. (1) lower (deflect to the left) by 2 Hz
(2) lower by 2 Hz

Answer: C

K/A Match:

Requires knowledge of RRC flow controllers in Auto.

Explanation:

A. Incorrect. Plausible since (2) is correct. However, the Individual Loop Controller bias will lower by 2 Hz to indicate that the Individual controller demand is 2 Hz below the Master Controller demand

B. Incorrect. Plausible since one of the features of these controllers is when the Individual Controllers are in AUTO, Individual Controller demand will automatically track Master Controller demand when the Master Controller is adjusted to allow a “bumpless” transfer to MANUAL. However, this automatic action does not occur when Individual Controller demand is changed.

C. Correct. When the Individual Controller Demand is adjusted while in AUTO, Individual Controller input to RRC speed control is altered and the pump will change speed. Master Controller demand will not change. The Bias indication on the Individual Controller will change accordingly to record the difference between Individual Controller and Master Controller demand signals.

D. Incorrect. Plausible since (1) is correct. Plausibility is enhanced since one of the features of these controllers is when the Individual Controllers are in AUTO, Individual Controller demand will automatically track Master Controller demand when the Master Controller is adjusted to allow a “bumpless” transfer to MANUAL. However, this automatic action does not occur when Individual Controller demand is changed.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000184, Reactor Recirculation Flow Control (ASD) System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 9676 - Describe the operation of the following controls on the RRC individual controllers, including expected indications and system response:
a. LOWER Pushbutton

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Knowledge of the operation of the RRC Master and Individual Loop Controllers

Question: RO-55

10 CFR Part 55 Content: 55.41 3

Comments / Reference: SD000184	Rev:	Major: 20	Minor: 3
<p>1. Individual Loop Controller (located on P602 in the Control Room) LO</p> <p>a) Deviation / Bias meter</p> <p>When the Individual Loop Controller is in AUTO, this meter indicates the “Bias” applied to the Demand Signal from the Master Controller before it is passed to the ASD MEM units to control pump speed.</p> <p>When the Individual Loop Controller is in MANUAL, this meter indicates the mismatch or “Deviation” between the Individual and Master Controllers.</p> <p>(5) While in AUTO, pump speed can also be changed by adjusting the Bias with the RAISE/LOWER pushbuttons on the Individual Loop Controller by using the RAISE/LOWER pushbuttons. A maximum of ± 6 Hz Bias can be applied when in the AUTO mode.</p> <p>(6) Note that the Demand Frequency signal displayed on the Individual Loop Controller will ALWAYS correspond to the FANUC control signal sent to the MEM units in the ASD Building.</p> <p>(7) For example, in AUTO, if the Master Controller signal is raised by 2 Hz, the Individual Controller tracks and will rise by 2 Hz (Bias will be unchanged). If the Individual Controller signal is raised by 4 Hz, the Bias will rise (deflect to the right) by 4 Hz (Master signal will be unchanged).</p> <p>(8) In MANUAL, if the Master Demand signal is raised by 2 Hz, the Deviation will decrease (deflect to the left) by 2 Hz and the Individual Controller signal will not change. If the Individual Controller signal is raised by 4 Hz, Deviation will also rise by 4 Hz (Master signal will be unchanged).</p> <p>(9) In all cases, pump speed will correspond to the Demand signal displayed on the Individual Controller - this is why all transfers, AUTO to MANUAL or MANUAL to AUTO, are bumpless.</p>			

Question: RO-56

Examination Outline Cross-reference: RO-56

Revision: 0

Date: 8/20/22

Tier: 2

Group: 2

K/A Number: 214000 A2.04

Level of Difficulty: 2

RO Importance Rating: 3.3

K/A Description: Ability to (a) predict the impacts of the following on the Rod Position Information System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Power supply loss

CGS is operating at 65% power for economic dispatch.

- A complete loss E-PP-US occurs.

What is one impact and what actions are necessary to mitigate the impact?

- A. RCC-P-1A trips if running. Refer to ABN-RCC for a complete loss of RCC.
- B. Loss of power to RPIS. Enter ABN-RPIS and determine control rod position from an alternate method.
- C. Loss of power to inboard MSIV solenoid valves. Scram the plant per PPM 3.3.1 prior to MSIV automatic closure.
- D. Loss of power to TDAS computer. Switch from TDAS to PPC as Primary per SOP-COMPUTER-OPS, Plant Process Computer (PPC).

Answer: B

K/A Match:

Requires a knowledge to recognize a loss of power has occurred to the RPIS and the ability to identify the appropriate procedure and actions to mitigate the consequences.

Question: RO-56

Explanation:
A. <u>Incorrect: Plausible because this is an impact and action required for a loss of E-IN-3</u>
B. <u>Correct: A loss of power occurs to the RPIS. ABN-RPIS is entered and rod position is determined from alternate means.</u>
C. <u>Incorrect: Plausible because this is an impact and action required for a loss of E-IN-2.</u>
D. <u>Incorrect: Plausible because this is an impact and action required for a loss of E-IN-5.</u>

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-ELEC-INV, 120 VAC Critical Distribution System Failures	
Tech Ref 4	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 7782 - Predict the effect(s) a failure of IN1 will have on loads supplied.

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level:

Requires the student to synthesize a knowledge of the power supply with the assessment of the actions needed to mitigate the consequences.

10 CFR Part 55 Content: 55.41 7

Question: RO-56

Comments / Reference: ABN-ELEC-INV	Rev:	Major:020	Minor:N/A
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Excerpts from ABN-ELEC-INV for Loss of IN-1:

2.1.2 If the inverter failed to auto transfer, the following actions will occur, due to a loss of E-PP-US.

- Feedwater heater controllers lose power and the valves fail open
- Power is lost to the Full Core Display.
- DEH will lose Primary Power Supply
- Loss of E-PP-US/D results in closure of AR-V-2A and AR-V-2B.
- Power is lost to OG-RIS-601A
- Loss of power to multiple rad monitors on Rad Board 24
- Power is lost to RDCS preventing normal rod motion except by scram.
- Loss of RPIS power supply (rod position indication is lost)
- Loss of power to RWM.

4.1.7 REFER to ABN-RPIS.

4.1.7 Entry into ABN-RPIS is necessary due to the loss of the Full Core Display. ABN-RPIS provides instructions for obtaining control rod position using alternate indications.

Excerpts from ABN-ELEC-INV for Loss of IN-2:

2.2.2 If the inverter fails to auto transfer, the following actions will occur, due to a loss of E-PP-8AA:

- Loss of power to E-MC-8C, (E-CB-81/8C trips).
- Loss of power to E-MC-8E, (E-CB-83/8E trips).
- Closure of Inboard MSIVs
- Reactor Scram on MSIV closure
- Trip of RCC-P-1B & 1C (if running)

4.2.5 IF power cannot be maintained to E-PP-8AA, THEN **SCRAM the Reactor** per PPM 3.3.1.

4.2.5 Should a condition exist in which the control room staff determines power cannot be maintained to E-PP-8AA then a SCRAM is initiated prior to the automatic MSIV closure SCRAM signal.

Question: RO-56

Excerpts from ABN-ELEC-INV for Loss of IN-3:

2.3.2 If the inverter fails to auto transfer, the following actions will occur due to a loss of E-PP-7AA:

- Loss of power to E-MC-7C, (E-CB-71/7C trips).
- Loss of power to E-MC-7E, (E-CB-73/7E trips).
- BOP OUTBOARD isolation (E-PP-7AA supplies power to relays in RC-1)
- RCC flow to Containment loads is isolated due to closure of RCC-V-5 and RCC-V-21 (loss of RC-1).
- Trip of RCC-P-1A (if running)
- Rise in Drywell Temperature and Pressure from a loss of cooling

4.3.2 IF power is lost to E-PP-7AA,
THEN REFER to ABN-RCC for a complete loss of RCC.

4.3.2 This directs the Control Room Operator to reference ABN-RCC, due to trip of RCC-P-1A, and closure of RCC-V-6 on loss of power to E-PP-7AA.

Excerpts from ABN-ELEC-INV for Loss of IN-5:

2.4.2 If the inverter fails to transfer, power is lost to the following:

- Loss of E-PP-US/5 results in closure of AR-V-2C and AR-V-2D.
- DEH Secondary Power Supply
- TDAS computer
- PDIS computer
- PASS panel power

4.4.6 IF necessary,
THEN SWITCH from TDAS to PPC as Primary per SOP-COMPUTER-OPS,
Plant Process Computer (PPC).

Examination Outline Cross-reference: 57

Revision: 0

Date: 8/24/22

Tier: 2

Group: 2

K/A Number: 216000.K2.01

Level of Difficulty: 2

RO Importance Rating: 3.6

K/A Description: Nuclear Boiler Instrumentation - Knowledge of electrical power supplies to the following: Analog trip system

What is the power supply to RFW-DPT-4A, Reactor Water Level (Narrow Range) A transmitter?

- A. RPS-A
- B. RPS-B
- C. PP-7A-A
- D. PP-8A-A

Answer: C

K/A Match:

Requires knowledge of the power supply to a NBI level transmitter that provides an input to the RPV low level trip system.

Explanation:

- A. Incorrect. Plausible since this is the power supply to the wide range Reactor Water Level transmitters MS-LT-61A/C.
- B. Incorrect. Plausible since this is the power supply to the wide range Reactor Water Level transmitters MS-LT-61B/D.
- C. Correct. In accordance with SD000126, the power supply to RFW-DPT-4A is PP-7A-A.
- D. Incorrect. Plausible since this is the power supply to Fuel Zone RPV level transmitter MS-LT-44B.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000126, Nuclear Boiler Instrumentation System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 7640 - Predict the effect(s) that these failures will have on the NBI: e. AC Electrical Distribution.

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires fundamental knowledge of power supplies to NBI transmitters.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SD000126	Rev:	Major: 13	Minor: 2
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IX. POWER SUPPLIES

<u>INSTRUMENT</u>	<u>DETECTOR</u>	<u>POWER SUPPLY</u>
• <u>H13-P601 Pressure</u>		
Post Accident Monitor	MS-PT-51A	PP-7A-A
Recorder (MS-LR/PR-623A)	MS-LT-26A	PP-7A-A
Reactor Level (Fuel Zone Range) (LI-610)	MS-LT-44B	PP-8A-A
Reactor Level (Fuel Zone Range recorder) (LR-615)	MS-LT-44A	PP-7A-A
Post Accident Monitor (Recorder) (MS-LR/PR-623B)	MS-PT-51B	PP-8A-A
	MS-LT-26D	PP-8A-A
Reactor Pressure (Digital) MS-PI-9	MS-PT-51A	PP-7A-A
• <u>H13-P602</u>		
RPV Level (S/D Range) (MS-LI-605)	MS-LT-27	PP-7A-Z
Cal Flow JP-5 (MS-FI-609A)	MS-FT-33A	PP-8A-Z
Cal Flow JP-10 (MS-FI-609B)	MS-FT-33B	PP-8A-Z
JP Loop A Flow (MS-FI-611A)	MS-SUM-33B	PP-8A-Z
Cal Flow JP 15 (MS-FI-609C)	MS-FT-33C	PP-8A-Z
Cal Flow JP 20 (MS-FI-609D)	MS-FT-33D	PP-8A-Z
JP Loop B Flow (MS-FI-611B)	MS-SUM-602B	PP-8A-Z
RPV Level (RFW-LI-606D) (Narrow Range)	RFW-DPT-4C	DP-S1-1
• <u>H13-P603</u>		
Core Pressure Drop & Total Flow (Recorder)(MS-DPR/FR-613)	MS-DPT-32	120V In P603 from P602
Reactor Water Level A (Narrow Range)(RFW-LI-606A)	RFW-DPT-4A	PP-7A-A
Reactor Water Level B (Narrow Range)(RFW-LI-606B)	RFW-DPT-4B	DP-S1-2

<u>INSTRUMENT</u>	<u>DETECTOR</u>	<u>POWER SUPPLY</u>
• <u>H13-P603</u> (cont'd)		
Reactor Water Level C (Narrow Range)(RFW-LI-606C)	RFW-DPT-4C	DP-S1-1
Reactor Pressure (MS-PI-605)	MS-PT-5	PP-7A-A
Reactor Water Level (Wide Range) (MS-LI-604)	MS-LT-26C	PP-1A-Z
Narrow and Upset Range	RFW-DPT-4A/B	PP-7A-A
Reactor Levels (Recorder)(RFW-LR-608)	RFW-DPT-17	PP-7A-A
Reactor pressure and turbine steam flow MS-PR/FR-609	MS-PT-7 & 808	PP-8A-Z
• <u>H13-P609/P611</u>		
Reactor Water Level (Wide Range)`	MS-LT-61A/C	RPS-A
Reactor Water Level (Wide Range)`	MS-LT-61B/D	RPS-B

Question: RO-58

Examination Outline Cross-reference: 58

Revision: 0

Date: 8/23/22

Tier: 2

Group: 2

K/A Number: 219000.A3.01

Level of Difficulty: 2

RO Importance Rating: 3.9

K/A Description: Ability to monitor automatic operation of the RHR/LPCI: Torus/Suppression Pool Cooling Mode, including:
Valve operation

CGS is in Mode 1.

- Reactor Power is 100%.

RHR-B is in Suppression Pool Cooling.

A LOCA occurs.

Plant conditions:

- Drywell pressure: 1.72 psig, up slow.
- RPV level: -70 inches, down slow.
- RPV pressure is 800 psig, down slow.

What is the effect on the RHR-B valve line-up?

RHR-V-24B, Suppression Pool Cooling/Test Return valve, will (1) and RHR-V-42B, LPCI Injection valve, will (2).

- A. (1) remain OPEN
(2) remain CLOSED
- B. (1) remain OPEN
(2) OPEN
- C. (1) CLOSE
(2) remain CLOSED
- D. (1) CLOSE
(2) OPEN

Answer: C

K/A Match:

Requires knowledge of the automatic operation of RHR Suppression Pool Cooling valves.

Explanation:

- A. Incorrect. Plausible since (2) is correct. Plausibility is enhanced if it is believed that LPCI initiation only occurs if RPV level goes below -129 inches. However, a Drywell pressure GT 1.68 psig will initiate LPCI and RHR-V-24B will CLOSE.
- B. Incorrect. Plausible if it is believed that RHR-V-42B will open regardless of RPV pressure. Plausibility is enhanced if it is believed that LPCI initiation only occurs if RPV level goes below -129 inches. However, a Drywell pressure GT 1.68 psig will initiate LPCI and RHR-V-24B will CLOSE. Additionally, RHR-V-42B will only open if RPV pressure is LT 470 psig.
- C. Correct. A LPCI initiation signal will be present if RPV level is LT -129 inches OR Drywell pressure is GT 1.68 psig. An active LPCI initiation signal will cause RHR-V-24B to CLOSE. RHR-V-42B will only open on an active LPCI initiation signal AND RPV pressure LT 470 psig. Therefore, for the conditions given, RHR-V-42B will remain CLOSED.
- D. Incorrect. Plausible since (1) is correct. However, since RPV pressure is GT 470 psig, RHR-V-42B will remain CLOSED.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000198, RHR System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: RHR-03 Describe the following major flow path(s) for the Residual Heat Removal System:
 b. LPCI injection
 d. Suppression Pool Cooling mode

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires Candidate to synthesize an understanding of automatic operations of RHR Suppression Pool valves with a knowledge of requirements to automatically initiate LPCI injection.

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SD000198	Rev:	Major:	19	Minor:	0
<p>20. RHR-V-24B, Suppression Pool Cooling/Test Return</p> <p>3-position switch CLOSE/AUTO/OPEN (spring return to AUTO, throttleable)</p> <p>CLOSE The valve closes if the switch is held in this position (throttleable).</p> <p>AUTO The valve closes on LPCI initiation.</p> <p>OPEN The valve opens if the switch is held in this position with RHR-V-6B shut. If initiation signal is present, RHR-V-24B cannot be opened unless RHR-V-42B is closed. MANUAL OVERRIDE amber lamp will come on and until the override is cleared, the valve must be manually closed. The override can be cleared by resetting the initiation logic.</p> <p>26. RHR-V-42B LPCI Injection</p> <p>3-position switch CLOSE/AUTO/OPEN (spring return to AUTO)</p> <p>CLOSE The valve closes. If the initiation signal (LPCI) is present, a MANUAL OVERRIDE amber lamp will come on and until the override is cleared, the valve must be manually opened. The override will clear when the initiation logic is reset or when reactor pressure rises above the injection valve open permissive setpoint of 470 psig.</p> <p>AUTO The valve will open if:</p> <p> a) Respective initiation signal is present; and</p> <p> b) RPV pressure is less than 470 psig.</p> <p>OPEN The valve will open if RPV pressure is less than 470 psig.</p>					

Question: RO-58

1. LPCI Initiation

A reactor pressure vessel level of LE minus 129 inches (-129"), a high drywell pressure of GE 1.65 PSIG (1.65#) or manual initiation* will automatically align the RHR system for the LPCI mode except when the system is initially in the shutdown cooling mode. Each RHR pump will be at rated speed and injection

Examination Outline Cross-reference: 59

Revision: 0

Date: 8/23/22

Tier: 2

Group: 2

K/A Number: 230000.K5.02

Level of Difficulty: 3

RO Importance Rating: 3.3

K/A Description: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the RHR/LPCI: Torus/Suppression Pool Spray Mode: Pump cavitation/vortex

CGS is in Mode 1.

A major earthquake results in a Reactor Coolant system rupture in the Drywell and a RHR 'A' suction line rupture from the Wetwell.

- Both ruptures are unisolable.

Current plant conditions:

- An emergency depressurization has been performed.
- Wetwell level is 12 feet, down slow.
- Wetwell temperature is 215°F.
- RHR pump 'C' is running to maintain RPV level above TAF. RHR 'C' is the ONLY injection system available.

A spurious trip of RHR pump 'C' occurs

(1) What is the concern with restarting RHR pump 'C'?

AND

(2) What is the effect on subsequent RHR pump 'C' operation?

- A. (1) Pump restart without venting of the system could cause system damage due to entrained gases in system high points.
(2) Restart RHR Pump 'C' and continue with LPCI injection.
- B. (1) Pump restart without venting of the system could cause system damage due to entrained gases in system high points.
(2) Keep RHR Pump 'C' secured until TAF is reached to minimize potential damage of the only injection source.
- C. (1) Pump restart could result in pump cavitation due to excessively high Wetwell temperature.
(2) Restart RHR Pump 'C' and continue with LPCI injection.
- D. (1) Pump restart could result in pump cavitation due to excessively high Wetwell temperature.
(2) Keep RHR Pump 'C' secured until TAF is reached to minimize potential damage of the only injection source.

Answer: A

K/A Match:

Requires knowledge of the operational implications of operating RHR pumps below the vortex limit.

Explanation:

- A. Correct. With the given plant conditions the Vortex Limits specified in PPM 5.2.1 must be taken into account. Specifically, Wetwell level is outside the Vortex Limit for RHR Pump 'C' (14.5 feet) while Wetwell temperature remains within the NPSH Limit for RHR Pump 'C' (225°F). With Wetwell level outside the Vortex Limit, there is a concern for the accumulation of entrained gases in the system high points, especially following a trip and restart of the affected pump. PPM 5.0.10, Flowchart Training Manual, states *"If a system is operated at water levels below the Vortex Limit and subsequently secured, the entrained gases could collect at the system high points. System restart without prior venting of the system could cause water hammer of piping and other components. Prolonged operation with high concentrations of entrained gases could result in reduced system performance and accelerated pump wear. Unless core cooling and primary containment failure concerns exist, pump operation is stopped and prevented if possible to avert potential system damage."* Since RHR 'C' is the only injection source available to maintain RPV level above TAF, the pump needs to be restarted and LPCI injection re-established.
- B. Incorrect. Plausible since (1) is correct. Plausibility is enhanced if it is believed that ECCS pumps cannot be operated below the Vortex limit. However, the potential damage to the core due to lowering RPV level is a greater concern. Therefore, RHR pump 'C' should be restarted to maintain RPV level.
- C. Incorrect. Plausible since (2) is correct. Plausibility is enhanced if it is believed that RHR pump 'C' NPSH temperature has been exceeded. However, the NPSH temperature for RHR pump 'C' has not been exceeded.
- D. Incorrect. Plausible if it is believed that RHR pump 'C' NPSH temperature has been exceeded. Plausibility is enhanced if it is believed that ECCS pumps cannot be operated above the NPSH limit. However, the NPSH limit for RHR pump 'C' is 225°F and has not been exceeded. Additionally, the potential damage to the core due to lowering RPV level is a greater concern. Therefore, RHR pump 'C' should be restarted to maintain RPV level.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.0.10, Flowchart Training Manual	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: RHR-12 Evaluate given key Residual Heat Removal System parameters, if needed determine a course of action to correct or mitigate the following abnormal condition(s): a. Inadequate net positive suction head

Question Source: Bank #: LO03471/36120
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize a knowledge of ECCS pump vortex limits with an understanding of ECCS pump operation when vortex limits are exceeded.

10 CFR Part 55 Content: 55.41 10

Comments / Reference: PPM 5.0.10	Rev:	Major: 026	Minor: N/A
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9.31 Vortex Limit

18	Vortex and NPSH Limits		
	<u>System</u>	<u>WW Level (ft)</u>	<u>WW Temp (°F)</u>
	RCIC with suction from WW	17.5	210
	LPCS	14.5	225
	RHR-C	14.5	225
	RHR-A, B	14.0	225
	HPCS with suction from WW	5.5	200

- 9.31.1 The Vortex Limit is the lowest Wetwell water level above which air entrainment is not expected to occur in an ECCS taking suction from the Wetwell. This Wetwell water level is used to preclude ECCS damage due to air entrainment. Separate limits are provided for each ECCS suction.
- 9.31.2 If a system is operated at water levels below the Vortex Limit and subsequently secured, the entrained gases could collect at the system high points. System restart without prior venting of the system could cause water hammer of piping and other components. Prolonged operation with high concentrations of entrained gases could result in reduced system performance and accelerated pump wear. Unless core cooling and primary containment failure concerns exist, pump operation is stopped and prevented if possible to avert potential system damage.
- 9.31.3 The Vortex Limit is determined assuming that the threshold for air entrainment is defined by a constant Froude Number for the ECCS suction inlet when the Wetwell water level is at least one pipe diameter above the elevation of the centerline of the inlet.

9.27 NPSH Limit

18 Vortex and NPSH Limits		
<u>System</u>	<u>WW Level (ft)</u>	<u>WW Temp (°F)</u>
RCIC with suction from WW	17.5	210
LPCS	14.5	225
RHR-C	14.5	225
RHR-A, B	14.0	225
HPCS with suction from WW	5.5	200

- 9.27.1 The NPSH Limit is the highest wetwell temperature which provides adequate net positive suction head for RCIC or an ECCS pump taking suction on the wetwell. The NPSH Limit is a function of pump flow and wetwell overpressure (airspace pressure plus hydrostatic head over the RCIC or ECCS suction). It is utilized to preclude RCIC or ECCS damage due to pump cavitation.
- 9.27.2 As pump flow increases, the suction line head loss increases, reducing the margin to saturation at the pump inlet. The NPSH Limit therefore decreases with pump flow. As overpressure increases, the static pressure and margin to saturation at the pump inlet also increase. The NPSH Limit therefore increases with overpressure.
- 9.27.3 Wetwell temperatures associated with NPSH limits for pumps taking suction from the wetwell are listed in Table 18. Since NPSH limits are nearly constant over the range of pump flows, the NPSH limits can be expressed as the single most limiting wetwell temperature for each pump. The Table 18 temperatures thus conservatively bound the NPSH limit curves given in TM-2115 Tab 6 of the TSC Technical Support Guidelines (TSGs). TM-2115 Tab 6 may be consulted if additional margin is desired.

Question: RO-60

Examination Outline Cross-reference: 60

Revision: 0

Date: 8/22/22

Tier: 2

Group: 2

K/A Number: 239001.A1.10

Level of Difficulty: 2

RO Importance Rating: 4.1

K/A Description: Ability to predict and/or monitor changes in parameters associated with operation of the Main and Reheat Steam System, including: Reactor power

CGS is in Mode 1.

- Reactor power is 100%.

Feedwater Heater 6B Running Vent valve (HD-V-6B) is inadvertently CLOSED.

What is the predominant effect on the plant over time?

Reactor power will...

- A. rise due to increased feedwater subcooling.
- B. lower due to increased bleed steam flow through FWH 6B.
- C. rise due to increased core flow from improved RRC NPSH.
- D. lower due to reduced thermal loss through the running vents.

Answer: A

K/A Match:

Requires knowledge of the effects of Heater Drain faults on reactor power.

Explanation:

A. Correct. With no running vents, non-condensable gases will build up in the FWH. This will reduce the FWH heat transfer coefficient and increase feedwater subcooling. Since it takes additional energy to heat the feedwater and maintain a constant steam flow out, reactor power must increase.

B. Incorrect. Plausible if it is believed that reducing flow out of the running vent will increase heat transfer in the FWH. However, the flow increase is minimal and does not overcome the reduction in heat transfer coefficient caused by an buildup of non-condensable gases.

C. Incorrect. Plausible since reactor power will rise and RRC pump NPSH may rise slightly. However, the major effect is due to additional heating required to maintain a constant steam flow.

D. Incorrect. Plausible since thermal loss through the running vents occurs. However, this effect is overcome by the reduction of heat transfer coefficient due to non-condensable gas buildup.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000169, Heater Vents and Drains System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 7607 - Determine the affect of a Feedwater Heater Vents and Drains System loss or malfunction will have on the following: e. Reactor Power

Question Source: Bank #: LO03790 (43670)
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize a knowledge of the effect on reactor power on the closure of a heater vent with an understanding of the reason behind this effect.

10 CFR Part 55 Content: 55.41 5

Comments / Reference: SD000169	Rev:	Major: 13	Minor: 001
<p>G. <u>Running Vent Valves</u></p> <p>The purpose of these valves are to provide a means of continuously venting off entrained air and other non-condensable gasses from the shell side of each feedwater heater during heater operation. Each heater has three vent connections each containing a manual needle-type control valve, which will be set initially at some throttled position and will remain fixed during subsequent heater operation. These vent connections are manifolded to a common header containing a motor-operated globe type isolation valve.</p>			

Question: RO-61

Examination Outline Cross-reference: 61

Revision: 0

Date: 08/29/22

Tier: 2

Group: 2

K/A Number: 268000 K3.04

Level of Difficulty: 3

RO Importance Rating: 3.0

K/A Description: Knowledge of the effect that a loss or malfunction of the Radwaste System will have on the following systems or system parameters: (CFR: 41.5 / 45.3)
K3.04 Primary containment drain sumps

CGS is at 100% power.

- P602-A13 2-2 REACTOR BUILDING FLOOR SUMP R3 LEVEL HI-HI is in alarm.
- Radwaste Control Room reports that R3 sump pumps FDR-P-4A and FDR-P-4B will NOT start.

RO will direct a field operator to investigate Sump R3 in the (1) pump room.

Drainage from (2) floor drains can cause an increase in Sump R3 water level.

- A. (1) CRD
(2) ONLY the Reactor Building
- B. (1) CRD
(2) the Drywell and Reactor Building
- C. (1) HPCS
(2) ONLY the Reactor Building
- D. (1) HPCS
(2) the Drywell and Reactor Building

Answer: D

K/A Match:

Direct correlation to the impact of sump pump failures on pump room water levels affecting Table 25 room water levels for PPM 5.3.1 Secondary Containment Control.

Explanation:

- A. Incorrect – Plausible – (1) Incorrect, Sump R3 is located in the HPCS pump room, Sump R5 is located in the CRD pump room. (2) Incorrect, Sump R3 receives drains from BOTH Reactor building floor drains and drywell floor drains

Question: RO-61

- B. Incorrect – Plausible - (1) incorrect Sump R3 is located in the HPCS pump room, Sump R5 is located in the CRD pump room. (2) Correct Sump R3 receives drains from BOTH the reactor building and Drywell floor drains.
- C. Incorrect – Plausible – (1) Correct Sump R3 is located in the HPCS pump room. (2) Incorrect, Sump R3 receives drains from BOTH Reactor building floor drains and drywell floor drains
- D. Correct – Sump R3 is located in the HPCS pump room, it receives drains from BOTH the reactor building and Drywell floor drains.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000130 Plant Drains	
4.602.A13	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 5328 State the purpose of each of the following system components:
c. Floor Drain Sumps and Pumps

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Candidate must know sump locations and what drains input to sump R3

10 CFR Part 55 Content: 55.41 7

Comments / Reference: SD000130	Rev:	Major: 13	Minor: 01
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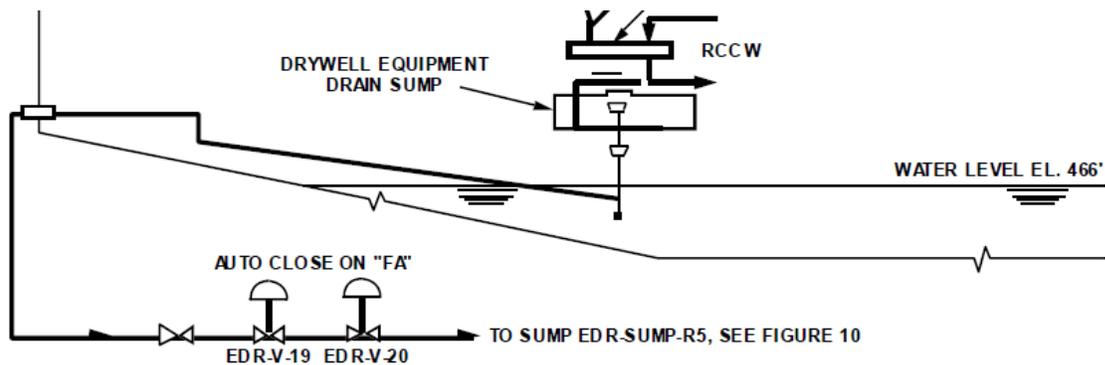
Question: RO-61

1. Reactor Building Equipment Drains

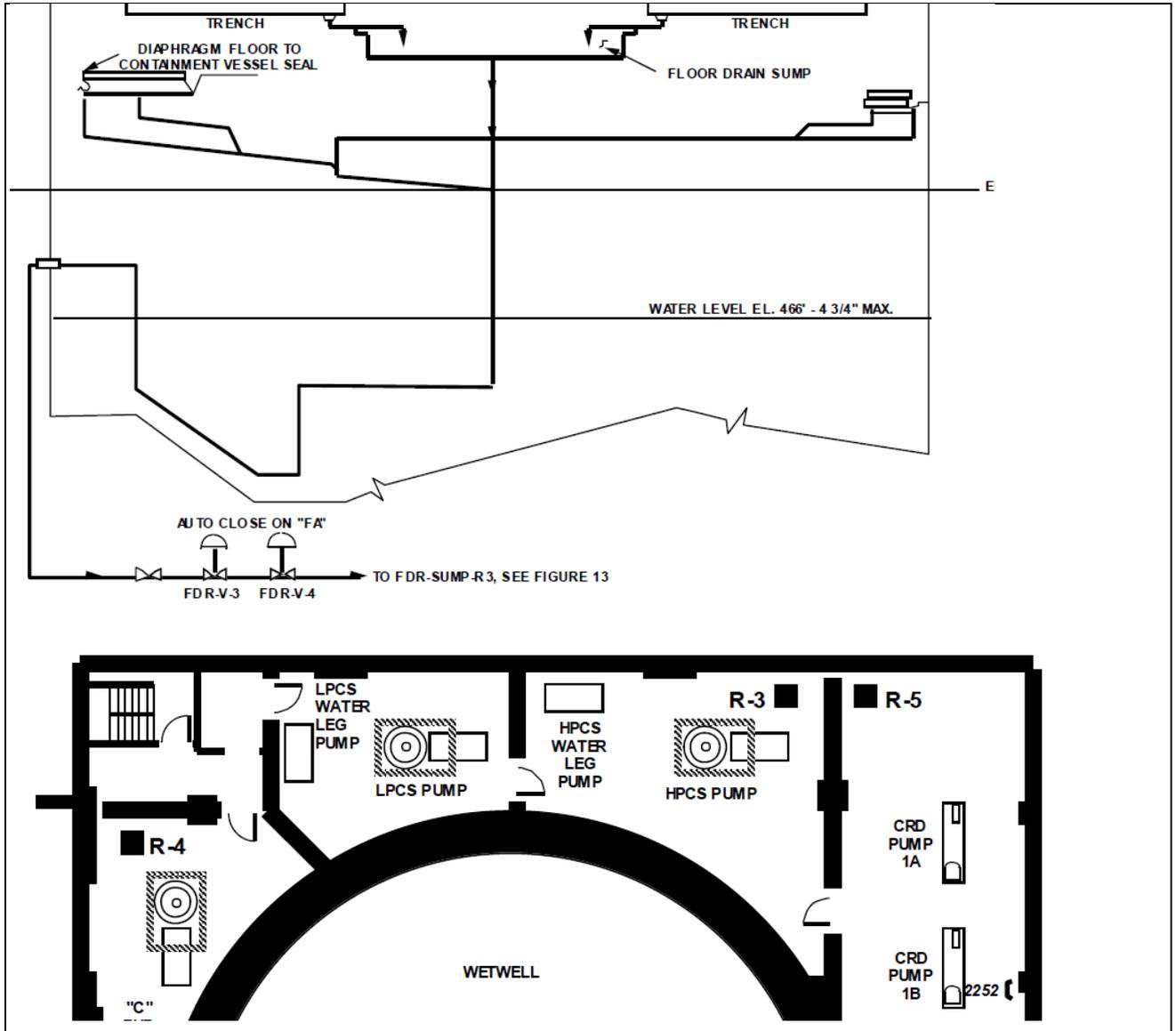
- a) The drywell is equipped with two equipment drain headers which merge and direct the liquid through a heat exchanger (EDR-HX-1) to the drywell equipment drain sump. The liquid is gravity drained through a loop seal, containment isolation valves (EDR-V-19 & 20), and a flow transmitter into the Reactor Building Equipment drain sump EDR-SUMP-R5 located in the CRD Pump Room, 422' elev. (see
- f) The Reactor Building equipment drains are routed to EDR-SUMP-R5 via two 4" drain headers.

1. Reactor Building Floor Drains

- a) Drywell floor drains are collected in trenches and directed into the drywell floor drain sump (see Figure 12). The water is then joined by drains from the diaphragm floor to containment vessel seal and flows through a loop seal, two containment isolation valves (FDR-V-3 & 4), and a flow transmitter into Reactor Building drain sump FDR-SUMP-R3 (see Figure 13).
- b) The Reactor Building floor drains are collected in four drain headers, each leading to a separate floor drain sump (FDR-SUMP-R1, R2, R3 & R4).



Question: RO-61



Comments / Reference: 4.602.A13	Rev:	Major: 30	Minor: N/A
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Question: RO-61

2-2 REACTOR BUILDING FLOOR SUMP R3 LEVEL HIGH-HIGH

2-2 WINDOW	SOURCE	AUTOMATIC ACTIONS
REACTOR BLDG FLOOR SUMP R3 LEVEL HI-HI	FDR-LS-4B (GE 418'10")	Closes FDR-V-608, Sump Inlet Valve to prevent flooding in HPCS and CRD pump rooms.

1. **VERIFY** one of the following are operating (R3 Sump pumps) (Radwaste Control Room):
 - FDR-P-4A
 - FDR-P-4B
2. **CHECK** R3 Sump level (HPCS pump room).

Question: RO-62

Examination Outline Cross-reference: 62

Revision: 0

Date: 10/18/22

Tier: 2

Group: 2

K/A Number: 271000.K4.11

Level of Difficulty: 2

RO Importance Rating: 3.3

K/A Description: Knowledge of Offgas System design features and/or interlocks that provide for the following: Elevated release point.

Which of the following radiation monitors can cause the OG System Discharge to the Elevated Release Stack, OG-V-60, to close?

- A. OG-RIS-601A/B, Offgas Post-Treatment
- B. OG-RIS-612, SJAЕ Condenser Outlet
- C. MS-RIS-610A/B/C/D, Main Steam Line
- D. OG-RIS-611, Offgas Charcoal Bed Vault Leak Detection

Answer: A

K/A Match:

Requires knowledge of the interlocks that will automatically isolate the offgas elevated release point.

Question: RO-62

Explanation:

A. Correct. In accordance with the Process Radiation Monitors System Description, SD000147, If both OG-RIS-601A and B are HI-HI-HI, downscale, or a combination of these two states, OG-V-60, Off-Gas System Isolation, closes.

B. Incorrect. Plausible since this radiation monitor alarms at HI and HI-HI. However, it does not provide an Offgas isolation signal.

C. Incorrect. Plausible since steam is supplied to Offgas Preheaters. However, the MS radiation monitors do not provide an Offgas isolation signal.

D. Plausible since OG-RIS-611 provides a HI alarm. However, it does not provide an Offgas isolation signal.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000147, Process Radiation Monitors System Description.	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: Learning Objective.

Question Source: Bank #: LO04682/90165
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level
 Requires knowledge of radiation monitor functions

10 CFR Part 55 Content: 55.41 7

Question: RO-62

Comments / Reference: SD000147	Rev:	Major: 16	Minor: 0
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V. CONTROL THEORY AND INTERLOCKS

A. Radiation Monitors

1. **Offgas Post Treatment Rad Monitors (OG-RIS-601A/601B) (Figure 5)**
 - a) When the countrate meter is in "CURRENT ACTIVITY" the display will show a bar graph and current activity in $\mu\text{Ci/cc}$.
 - b) When a HI or HI-HI-HI trip signal is reached the display will indicate "ALERT" or "HIGH" respectively.
 - c) The countrate meters in the control room drives the DOWNSCALE, HI, HI-HI-HI, and TROUBLE annunciators.
 - d) Each RMS output signal is sent to Recorder OG-RR-601 on Panel H13-P600. This recorder drives the annunciator for the HI-HI signal annunciator.
 - e) Either High trip results in:
 - OG-V-45 closes (charcoal filter bypass)
 - OG-V-51 A, B, C & D open (charcoal filter isolation)
 - f) **Both HI-HI-HI or HI-HI-HI and downscale or both downscale result in:**
 - OG-V-60 closes (Off-Gas System Isolation)**
 - OG-V-23, 34A & B, and 54 close (Off-Gas Drains to EDR)

I

Examination Outline Cross-reference: 63

Revision: 0

Date: 8/30/22

Tier: 2

Group: 2

K/A Number: 290003.K6.03

Level of Difficulty: 2

RO Importance Rating: 2.9

K/A Description: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Control Room Ventilation: Component cooling water system

Which of the following may be lined up to Control Room Emergency Cooling Coil, WMA-CC-51B1, when Radwaste Chill Water (WCH) is not available?

- (1) Plant Service Water (TSW)
- (2) Standby Service Water (SW)
- (3) Control Room Emergency Chill Water (CCH)

- A. (1) ONLY
- B. (2) ONLY
- C. (1) and (3) ONLY
- D. (2) and (3) ONLY

Answer: D

K/A Match:

Requires knowledge of available backup cooling water systems when normal cooling water system is not available.

Explanation:

- A. Incorrect. Plausible since Plant Service Water provides cooling water to the Critical Switchgear Rooms emergency cooling coil. However, it cannot be lined up to the Control Room Air Handling Units.
- B. Incorrect. Plausible since (2) is correct. However, it is not the only correct source listed.
- C. Incorrect. Plausible since (3) is correct. However, (1) is not correct.
- D. Correct. The Control Room Air Handling Unit is normally supplied from Radwaste Chill Water. The unit has an emergency cooling coil that can be supplied by either Control Room Emergency Chill Water or Standby Service Water.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SD000201, CR-HVAC System Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: CR-HVAC-09 Describe the systems supporting the Control Room HVAC System, the nature of the support, and the impact a loss (or Malfunction) of the system would have on the Control Room HVAC System:

- b. Standby Service Water (SW)
- f. Radwaste Chilled Water
- h. Plant Service Water (TSW)

Question Source: Bank #: LO02685 (100882)
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires knowledge of available backup sources of cooling water to Control Room Air Handlers.

10 CFR Part 55 Content: 55.41 4

Comments / Reference: SD000201	Rev:	Major: 20	Minor: N/A
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B. Control Room Air Handling Units (Figure 9)

The CR Air Handling Units (AHU) WMA-AH-51A/B (Div. 1 & 2) supply 21,000 cfm of recirculated and fresh air at 75°F ± 3°F to the CR during normal operation. The normal cooling coil is supplied with chilled water from Radwaste Building Chilled Water (WCH). Temperature Control Valve WMA-TCV-12A/B adjusts chilled water flow to maintain CR temperature. The emergency cooling coil is supplied by either emergency chilled water (CCH) or Standby Service Water (SW).

Emergency cooling coils WMA-CC-51-A1(B1) are normally lined up to CCH with the ability to line up cooling coils directly to SW. Increased heat load on CR-HVAC system will eventually result in SW being unable to support the CR-HVAC during a DBE.

D. Critical Switchgear Area Air Handling Units (Figure 8)

The CSA AHUs (WMA-AH-53A (28,100 cfm), 53B (22,800 cfm)) supply recirculated and fresh air to the Critical Switchgear rooms. The AHU normal cooling coil is supplied by WCH, or by TSW, which is normally isolated. The emergency-cooling coil is supplied by the SW system. The CSA AHUs also have roll filters (WMA-FL-53A (B)) installed on their inlets, like the filters in the cable spreading room AHUs.

Examination Outline Cross-reference: 64

Revision: 0

Date: 4/21/22

Tier: 3

Group: N/A

K/A Number: 2.1.4

Level of Difficulty: 3

RO Importance Rating: 3.3

K/A Description: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status.

The plant is at full power.

One of the on-coming EOs required for minimum shift complement (filling a fire brigade position), calls in sick.

Which of the following is correct?

After crew turnover...

- A. all off-going fire brigade members can leave as long as a replacement is on-site within two hours.
- B. one off-going EO fire brigade member must stay but can leave when relieved by another qualified EO.
- C. one off-going fire brigade member must stay, and once logged as a fire brigade member, must complete the entire watch.
- D. all off-going fire brigade members can leave as long as the oncoming EO filling the RWCR Watch also fills the fire brigade position.

Answer: B

K/A Match:

Requires knowledge of shift staffing requirements.

Explanation:

- A. Incorrect. Plausible since the shift crew composition may be one less than the minimum requirements for a period not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members. However, all shift personnel are required to remain on duty until properly relieved by a qualified individual.
- B. Correct. The on shift fire brigade member is required to remain on shift until properly relieved.

- C. Incorrect. Plausible since the on shift fire brigade member must remain on shift. However, they may leave after being relieved by a qualified individual without completing the next shift.
- D. Incorrect. Plausible since the fire brigade member may leave after being relieved by a qualified individual. However, the stem notes that the oncoming crew is at minimum EO staffing. Therefore, the oncoming RWCR Watch will be filling the other required fire brigade role or a safe shutdown position, and cannot cover the gapped fire brigade position.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 1.3.1, Operating Policies, Programs and Practices	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 6097 - State the minimum staffing requirements of the Plant Fire Brigade.

Question Source: Bank #: 8335
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires knowledge of staffing requirements listed in PPM 1.3.1.

10 CFR Part 55 Content: 55.41 10

Comments / Reference: PPM 1.3.1	Rev: Major: 133 Minor: N/A
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Number: 1.3.1	Use Category: INFORMATION	Major Rev: 133 Minor Rev: N/A Page: 31 of 103
Title: Operating Policies, Programs and Practices		

4.10 Shift Complement {R-2.3}, {R-2.7}, {R-2.11}, {R-2.14},
{R-2.19}, {R-2.20}, {P-2.49}

NOTE: The shift complement shall always be sufficient to meet the minimum crew composition as set forth in Section 13.1 of the FSAR and the Emergency Plan. The shift crew composition may be one less than the minimum requirements specified for a period not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to the minimum specified, using the call-in process, if necessary. In the case of relief personnel absence, the on-duty personnel shall remain on-shift until relieved by a qualified individual. Additional personnel may be assigned to shifts, as necessary.

POSITION	REQUIRED STAFFING	
Shift Manager	1	(1, 7, 14, 15)
Control Room Supervisor	1	(1, 7, 10, 15)
Control Room Operators	3	(2, 7, 15)
Incident Advisor	1	(1, 7, 9, 11, 14, 15)
Equipment Operator	4	(4, 6, 7, 13, 15)
Shift Technical Advisor Function	1	(1, 7, 10, 12, 14, 15)
Fire Brigade Team	5	(7, 8)
First Responder Team	2	(8)
Health Physics	3	
Chemistry	1	
Electrician/I&C Maintenance	2	(5, 6)
Mechanical Maintenance	2	(6)
SCC Duty Officer	1	
Security Responder	1	
Plant Laborer	1	(3)
Emergency Notification System (ENS) Communicator	1	
Offsite Notifier (EALs and PARs)	1	(12)

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 133 Minor Rev: N/A Page: 32 of 103
Title: Operating Policies, Programs and Practices		

1. An STA shall be on shift in Modes 1, 2 and 3, and must be available within 90 minutes in Modes 4 or 5. The STA function may be staffed by a dedicated individual (licensed or non-licensed), or by an individual filling a dual role position as the CRS, SM, or SSS. When the CRS or Shift Manager is filling a dual role as the STA, an extra SRO is required to be on shift to support emergency response.
2. The third Control Room Operator is an EOP commitment. Two are required in Modes 4 or 5. {P-2.49}
3. On shift as part of the Fire Brigade team.
4. Two Equipment Operators for Plant Fire Brigade. One of the Equipment Operators will be the Fire Brigade Leader. Two Equipment Operators for Plant Operations Response and, if needed, emergency maintenance. One additional Equipment Operator can be scheduled on day shift to assist in TMU rounds (total of five). {R-2.30}
5. Either an Electrician or an I&C Technician is required.
6. This ERO function may be filled by an on-shift Equipment Operator when Maintenance is not present.

NOTE: Fire Brigade members should be able to dress out and be at the scene within 15 minutes.

7. Fire Brigade shall consist of at least five (5) qualified members. The Fire Brigade shall not include the Shift Manager, Control Room Supervisor, three Control Room Operators, and two Equipment Operators necessary for operator actions to ensure safe shutdown of the unit. At a minimum, the Fire Brigade Leader and two members shall have sufficient knowledge of plant fire safe shutdown equipment. (Qualification as Equipment Operator, Chemistry Technician, or Health Physics Technician meets this knowledge requirement). Fire Brigade team normally consists of two Equipment Operators, one Chemistry Technician, one Health Physics Technician and one Laborer. The Shift Technical Advisor and the Incident Advisor (on shift when the STA qualified individual is filling a dual role as the CRS or SM) must report to the Control Room and, thus, cannot be a Fire Brigade member. {R-2.11}
8. Two individuals on shift should be qualified and designated as First Responders. These individuals may also be members of the Fire Brigade but cannot be safe shutdown personnel.
9. Administrative Requirement. This position is not required for technical specification or FSAR minimum staffing. This position is only required for ERO staffing, if filling the requirements of the Incident Advisor.
10. An additional SRO should be assigned to the Control Room during Mode 4, 5 to assist with administrative duties.

Number: 1.3.1

Use Category: INFORMATION

Major Rev: 133

Minor Rev: N/A

Page: 68 of 103

Title: Operating Policies, Programs and Practices

4.27 Shift Turnover

NOTE: Electronic turnover checklists are generated in eSOMS, and should be used for conducting turnover. If eSOMS is unavailable (e.g., eSOMS outage, refueling floor activities), a hard-copy turnover checklist attachment should be used.

NOTE: Electronic turnover checklists in eSOMS will meet or exceed the requirements of the Shift Turnover Checklists (attachment 6.9 thru 6.23). The appearance and order of electronic turnover checklist content may vary from the attachments due to software limitations.

4.27.1 All Shift Personnel

{R-2.19}

- a. All Shift Operations personnel are required to remain on duty with full responsibility for their workstation until properly relieved by a qualified individual.
- b. No person shall permit a relief person to take the shift if there is doubt the relief is alert, coherent, and fully capable of responsible performance; no person shall take a shift position unless physically and mentally fit.
- c. Each shift position shall be relieved by a person who is properly licensed, if required, and qualified to take the responsibilities of the position.

Examination Outline Cross-reference: 65

Revision: 0

Date: 4/21/22

Tier: 3

Group: N/A

K/A Number: 2.1.20

Level of Difficulty: 2

RO Importance Rating: 4.6

K/A Description: Ability to interpret and execute procedure steps.

CGS is in Mode 1.

Control room operators are performing a Continuous Use procedure when they come to the following step:

4.3 **IF** the reactor is in Mode 3, **THEN** perform the following...

What is required to mark this step "N/A"?

- A. No additional documentation or supervisor approval is required.
- B. Document the reason next to the step, and obtain supervisor initials for approval.
- C. Document the reason next to the step; supervisor initials are not required for approval.
- D. Obtain supervisor initials next to the step for approval; a documented reason is not required.

Answer: A

K/A Match:

Requires understanding of the process for executing conditional procedure steps.

Explanation:

Reference: SWP-PRO-01, Procedure and Work Instruction Use and Adherence, section 4.7.1

- A. Correct. When a conditional step does not pertain to the current condition of the plant, it may be marked "N/A". No additional justification or approval is required.
- B. Incorrect. Plausible since documenting a reason and supervisor initials are required to N/A non-conditional steps. However, for conditional steps that do not apply to the current plant conditions, documenting a reason and supervisor initials are not required.
- C. Incorrect. Plausible since documenting a reason is required to N/A non-conditional steps. However, for conditional steps that do not apply to the current plant conditions, documenting reasons and supervisor approvals are not required.

D. Incorrect. Plausible since supervisor initials are required to N/A non-conditional steps. However, for conditional steps that do not apply to the current plant conditions, no other approval is required.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
SWP-PRO-01, Procedure and Work Instruction Use and Adherence	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 3031 - Describe when a procedure step may be marked "N/A" without additional approval in lieu of performing the step. [SWP-PRO-01].

Question Source: Bank #: 43907
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: 2017

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level
Requires candidate to recall procedure information.

10 CFR Part 55 Content: 55.41 10

Comments / Reference: SWP-PRO-01	Rev: Major: 034	Minor: N/A
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Number: SWP-PRO-01	Use Category: INFORMATION	Major Rev: 034
Title: Procedure and Work Instruction Use and Adherence		Minor Rev: N/A
		Page: 18 of 35

4.7.2 Use of Not Applicable (NA) for **Conditional Step(s)** or Specific Condition Statements in Continuous or Reference Use Procedures

NOTE: NA is not to be used to bypass step(s) that are inadequately or improperly written and is not to be used in lieu of an approved change.

NOTE: During a pre-job brief, discussion of conditional steps known to not be applicable and whether a peer check is required for other conditional steps should occur.

- a. **IF** marking a single step NA,
THEN PERFORM the following:

- 1) **MARK** the step NA.
- 2) **INITIAL** the step.

NOTE: A consecutive group of steps are substeps contained within a conditional step.

- b. **IF** marking a consecutive group of steps NA,
THEN PERFORM the following:

- 1) **PLACE** NA in the first step.
- 2) **PLACE** NA in the last step.

CAUTION

Pay careful attention to drawing the line mentioned below to minimize the potential for inadvertently lining out a necessary step. All steps, notes, cautions, and warnings contained within the vertical line should be reviewed to ensure the correct steps are marked NA.

- 3) **DRAW** a vertical line through the steps between the first and last step involved.
- 4) **INITIAL** the NAs.
- 5) **IF** desired,
THEN OBTAIN a peer check.

- c. **IF** a group of steps spans more than one page,
THEN PERFORM the following:

- 1) **PLACE** NA in the first step on each page.
- 2) **PLACE** NA in the last step on each page.

Number: SWP-PRO-01	Use Category: INFORMATION	Major Rev: 034 Minor Rev: N/A Page: 19 of 35
Title: Procedure and Work Instruction Use and Adherence		

CAUTION

Pay careful attention to drawing the line mentioned below to minimize the potential for inadvertently lining out a necessary step. All steps, notes, cautions, and warnings contained within the vertical line should be reviewed to ensure the correct steps are marked NA.

3) **DRAW** a vertical line through the steps.

4) **INITIAL** the step.

5) **IF** desired,
THEN OBTAIN a peer check.

4.7.3 Use of Not Applicable (NA) for **Non-Conditional Step(s)** or Specific Condition Statements in Continuous or Reference Use Procedures

a. **VERIFY** ALL the following criteria are met prior to marking the step NA:

- The step or data entry to be NA'd is not a critical step or data entry (required for the successful completion of the procedure).
- The step is not needed for the current mode, condition or configuration of the plant.
- The intent (method of operation or the results) of the step(s) or sections does not change.
- An unsafe condition is not created.
- The initial conditions, precautions, or prerequisites sections are not violated; however, the Shift Manager may NA prerequisites that are not applicable.
- The method by which processes are performed that may affect quality (safety) structures, systems, and components is not changed without Shift Manager approval.

b. **DOCUMENT** the reason on the front cover or next to the step.

NOTE: The purpose of the asterisk with the number is to tie the particular step to the associated comment on the front page.

c. **IF** documenting the reason on the front cover,
THEN PLACE an asterisk with a number by the step being NA'd
AND PLACE an asterisk with the same number on the cover page
AND INCLUDE the reason.

Number: SWP-PRO-01

Use Category: INFORMATION

Major Rev: 034

Title: Procedure and Work Instruction Use and Adherence

Minor Rev: N/A

Page: 20 of 35

NOTE: Managers and Supervisors are responsible for ensuring they are technically knowledgeable of the task, or obtain concurrence from an individual who is technically knowledgeable, ensuring the explanation for marking the step(s) as NA is documented and appropriate.

- d. **OBTAIN** supervisor initials to show approval, next to the explanation for the step being marked NA.
- e. **IF** the procedure or work instruction is Continuous Use, pertain to Safety-Related equipment, Technical Specifications, or Licensee Controlled Specification procedure, **THEN OBTAIN** Shift Manager initials to show approval next to the explanation.

Examination Outline Cross-reference: 66

Revision: 0

Date: 4/21/22

Tier: 3

Group: N/A

K/A Number: 2.2.22

Level of Difficulty: 2

RO Importance Rating: 4.0

K/A Description: Knowledge of limiting conditions for operation and safety limits.

Which of the following is the lowest reactor steam dome pressure that results in exceeding Technical Specification Safety Limits?

- A. 1310 psig
- B. 1320 psig
- C. 1330 psig
- D. 1340 psig

Answer: C

K/A Match:

Requires knowledge of the Technical Specification Safety Limits.

Explanation:

- A. Incorrect. Plausible since this pressure is above the RPS High RPV Pressure Scram setpoint and within 15 limit is 1325 psig, which is above this pressure.
- B. Incorrect. Plausible since this pressure is within 5 psig of the Safety Limit. However, the Safety Limit is 1325 psig, which is above this pressure.
- C. Correct. 1330 psig is the lowest pressure listed that exceeds the Safety Limit of 1325 psig.
- D. Incorrect. Plausible since this pressure exceeds the Safety Limit. However, it is not the lowest pressure listed that exceeds the safety limit.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
Technical Specifications , 2.1, Reactor Coolant System Pressure Safety Limits	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 10304 - Describe each of the Safety Limits and state what actions are required if a Safety Limit is violated

Question Source: Bank #: 16305
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: 2021

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires knowledge of the Reactor Pressure Safety Limit

10 CFR Part 55 Content: 55.41 10

Comments / Reference: TS Safety Limit 2.1	Rev: Major: 265 Minor: N/A
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SLs
2.0

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 686 psig or core flow < 10% rated core flow:

THERMAL POWER shall be $\leq 25\%$ RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 686 psig and core flow $\geq 10\%$ rated core flow:

The MCPR shall be ≥ 1.07 .

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

Question: RO-67

Examination Outline Cross-reference: RO-67

Revision: 0

Date: 5/19/22

Tier: 3

Group: N/A

K/A Number: 2.2.43

Level of Difficulty: 3

RO Importance Rating: 3.0

K/A Description: Knowledge of the process used to track inoperable alarms.

CGS is in Mode 1.

A control room annunciator has failed.

- Annunciator repairs will be completed prior to the end of shift.

What action is required to be completed in accordance with PPM 1.3.1, Operating Policies, Programs and Practices?

- Place a Caution Clearance on the annunciator.
- Initiate a Component Status Change Order (CSCO).
- Establish a temporary log to monitor the redundant indication.
- Notify the Assistant Operations Manager (AOM-OPS).

Answer: C

K/A Match:

Requires knowledge of process to track inoperable alarms.

Explanation:

- Incorrect. Plausible since a caution clearance is used on a disabled annunciator. However, it is only required to be placed if the annunciator is disabled GT one shift.
- Incorrect. Plausible since a Component Status Change Order (CSCO) is used to track components not in their normal configuration. However, in accordance with PPM 1.3.1, Operating Policies, Programs and Practices, a CSCO is not to be used on a disabled annunciator
- Correct. In accordance with PPM 1.3.1, Operating Policies, Programs and Practices, if a failed annunciator is to be disabled, the operations crew *must* "Establish a temporary log to monitor the redundant indication at a frequency determined by the CRS".
- Incorrect. Plausible since the AOM-OPS should be notified if an annunciator is disabled. However, this is only required when the annunciator is disabled for GT one shift.

Question: RO-67

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 1.3.1, Operating Policies, Programs and Practices	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: Reference or N/A

Learning Objective: 6087 - State the CRO's responsibility at the beginning of each shift concerning all Control panel annunciators that are in an alarm state.

Question Source: Bank #: 100907
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: 2019

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Candidate is required to synthesize information given in the question with a knowledge of the requirements for disabling an alarm and when each requirement applies. Additionally, the candidate must demonstrate an understanding of the application of Component Status Change Orders (CSCOs).

10 CFR Part 55 Content: 55.41 10

Comments / Reference: PPM 1.3.1	Rev:	Major: 133	Minor: N/A
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Number: 1.3.1	Use Category: INFORMATION	Major Rev: 133
Title: Operating Policies, Programs and Practices		Minor Rev: N/A Page: 44 of 103

4.13 Equipment Configuration and Status Control {R-2.33}

4.13.1 Whenever a plant component is suspected of being mis-configured, prompt actions should be taken to ensure the component is in the correct alignment. {P-2.63}

NOTE: Component Status Change Orders (CSCO) are used for short-term (LE 7 days) configuration control. The process should be utilized any time a component such as a valve (including instrument isolation valves) or circuit breaker is aligned out of its normal position and the alignment and realignment is not controlled by procedures, checklists, Clearance Order, or Work Order instructions. This process should not be used to circumvent the requirement for a permanent procedure change, or the need for a Work Request. {R-2.5}, {R-2.33}

NOTE: The use of the CSCO is not limited to Operations and Maintenance. Any organization realigning plant components as described above should process a CSCO.

NOTE: A CSCO tag is used to identify component positions altered by a CSCO and are uniquely identified to a specific CSCO.

NOTE: A CSCO or Caution Tag should be used to identify the status of equipment that may be controlled by a procedure when that procedure is not actively in use. {P-2.62}

NOTE: Use of a CSCO should not be used to circumvent the requirement for a permanent procedure change or the need for a Work Request.

4.13.2 Implementing a Component Status Change Order (CSCO)

- a. A CSCO is not to be used for deactivating annunciators. When deactivating annunciators, use a Caution clearance.
- b. A Caution clearance should be used instead of a CSCO unless there is a reasonable expectation that the CSCO can be cleared in LE 7 days.

NOTE: The CRS/Shift Manager may authorize temporary changes in component status after the CSCO tag is hung without requiring the CSCO to be released or the component status change to be documented each time the component is manipulated. This allows alignment for operational support. Configuration control is maintained by verifying proper alignment when the CSCO is cleared.

- c. The CRS obtains the next CSCO Serial Number from Attachment 6.2, Component Status Change Order Index. The number sequence is 2 followed by the year, a dash and a sequential number (e.g., 201-001, 201-002).

Number: 1.3.1

Use Category: INFORMATION

Major Rev: 133

Minor Rev: N/A

Page: 47 of 103

Title: Operating Policies, Programs and Practices

4.13.6 **Disabling a failed annunciator**

CAUTION

If a locked in annunciator is a valid alarm, then do not disable the annunciator

- a. Check redundant indication.
- b. Obtain permission from the Shift Manager or above, prior to disabling the annunciator. {AR-2.82}
- c. Disable the annunciator such that no other system/component is affected.
- d. Verify no other FSAR described SSCs are affected by disabling the failed annunciator (CRS and Shift Manager). (Refer to 50.59 Screen 12-0065).
- e. IF the annunciator is Fire Protection related,
THEN evaluate the need to initiate a fire impairment.
- f. Log the disabled annunciator in the Control Room Log. {AR-2.82}
- g. **Establish a temporary log to monitor the redundant indication at a frequency determined by the CRS.**
- h. IF applicable,
THEN initiate a Work Request.
- i. Initiate a Condition Report.
- j. **IF the annunciator is disabled GT one shift,
THEN perform the following:**
 - 1) **Notify the Assistant Operations Manager of the disabled annunciator.**
 - 2) **Place a Caution Clearance containing the following information on the disabled annunciator:** {R-2.22}, {AR-2.82}
 - The annunciator taken out of service.
 - Why the annunciator was taken out of service.
 - What actions or plant conditions are required to recover the annunciator and any compensatory actions being taken.
 - When the annunciator is expected to be returned to service.
- k. Independently verify no other FSAR described SSCs are affected by disabling the failed annunciator (Engineering).

Examination Outline Cross-reference: 68

Revision: 0

Date: 5/19/22

Tier: 3

Group: N/A

K/A Number: 2.3.12

Level of Difficulty: 4

RO Importance Rating: 3.2

K/A Description: Knowledge of radiological safety principles and 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, or alignment of filters.

An Equipment Operator (EO) was briefed to hang and lift a clearance order in a Locked High Radiation Area (LHRA).

- EO annual dose prior to hanging the clearance order: 1678 mrem TEDE.

The EO hangs the clearance order.

- EO dose received while hanging the clearance order: 239 mrem.

The work is complete and the clearance order is ready to be lifted.

- Estimated dose for lifting the clearance order: 180 mrem.

Which of the following is correct in accordance with GEN-RPP-06, Dosimetry Program Description?

- A. A Planned Special Exposure should be completed and approved before the EO is allowed to lift the tags.
- B. The EO will not exceed any dose limits or hold points, and may lift the tags without a dose extension.
- C. Dose extensions are not allowed for clearance orders per GEN-RPP-06. A different EO must clear the tags.
- D. An Administrative Dose Extension should be completed and approved before the EO is allowed to lift the tags.

Answer: D

K/A Match:

Requires knowledge of requirements for exceeding dose administrative hold points.

Explanation:

- A. Incorrect. Plausible since a Planned Special Exposure (PSE) is required for exceeding regulatory dose limits (5000 mrem/year TEDE). However, for the conditions given, only the TEDE administrative hold point has been exceeded. This requires an Administrative Dose Extension (ADE)
- B. Incorrect. Plausible since the regulatory dose limit has not been exceeded. However, the administrative dose hold point has been exceeded. This requires an ADE.
- C. Incorrect. Plausible since a dose extension beyond the regulatory dose limit would not normally be granted except for “extraordinary circumstances. However, for the conditions given, only the administrative hold point is exceeded, which requires an ADE.
- D. Correct. After completing the clearance order hang, The EO’s total annual dose is 1678 mrem + 239 mrem = 1917 mrem TEDE. After lifting the clearance order, the EO’s projected total dose is 1917 mrem + 180 mrem = 2097 mrem TEDE. This is above the administrative dose hold point of 2000 mrem TEDE, but below the regulatory dose limit of 5000 mrem TEDE. In accordance with GEN-RPP-06, section 4.2, An Administrative Dose Extension (ADE) should be initiated and approved prior to allowing the individual to exceed an Energy Northwest administrative dose hold point.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
GEN-RPP-06, Dosimetry Program Description	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 11258 - Knowledge of facility ALARA program.

Question Source: Bank #: LO02041/13124
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires synthesis of a knowledge of regulatory dose limits/administrative dose hold points, and an understanding of the requirements to allow personnel to exceed their administrative dose hold point.

10 CFR Part 55 Content: 55.41 12

4.1.3 Approval for an individual to exceed an administrative dose hold point is authorized and controlled through an Administrative Dose Extension (ADE), as explained in Section 4.2.

4.1.4 Energy Northwest may authorize an adult worker to receive doses in excess of the regulatory annual occupational dose limits only for extraordinary circumstances under the provisions of a Planned Special Exposure (PSE). Refer to GEN-RPP-08 for the criteria and requirements for PSEs.

4.2 Administrative Dose Extensions

An Administrative Dose Extension (ADE) (Form 26258) should be initiated and approved prior to allowing the individual to exceed an Energy Northwest administrative dose hold point, as outlined below:

PERSONNEL DOSE LIMITS AND HOLD POINTS



	Regulatory Occupational Dose Limits (rem)	Administrative Dose Hold Points (rem)	Approval Authority to Exceed Administrative Dose Hold Point
Adults			
TEDE	5.000	2.000	Dept Manager + RPM (+ Plant General Manager to exceed 3.000 rem)

Question: RO- 69

Examination Outline Cross-reference: 69

Revision: 0

Date: 6/9/22

Tier: 3

Group: N/A

K/A Number: 2.4.12

Level of Difficulty: 3

RO Importance Rating: 4.0

K/A Description: Knowledge of operating crew responsibilities during emergency and abnormal operations

CGS is in Mode 1.

An event occurs.

The CRS directs the EAL Notifier to come to the control room at 1200.

In accordance with PPM 1.3.1, Operating Policies, Programs and Practices, which of the following is the longest time that will meet the requirements for the EAL Notifier to report to the control room?

- A. 1204
- B. 1209
- C. 1214
- D. 1224

Answer: B

K/A Match:

Requires knowledge of the EAL Notifier reporting responsibilities during an emergency.

Explanation:

- A. Incorrect. Plausible since 1204 is within the required reporting time. However, 1209 is the longest time listed that meets the reporting time of 10 minutes.
- B. Correct. In accordance with PPM 1.3.1, section 4.10.15, the EAL Offsite Notifier is required to report to the control room within 10 minutes.
- C. Incorrect. Plausible since EAL classification time is 15 minutes. However, the Notifier is required to be in the control room within 10 minutes.
- D. Incorrect. Plausible since EAL classification time is 15 minutes and EAL notification time is an additional 15 minutes. However, the Notifier is required to be in the control room within 10 minutes.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 1.3.1, Operating Policies, Programs and Practices	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 13511 - State the time requirement for the EAL Notifier to report to the main control room following an emergency classification.

Question Source: Bank #: 36755
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires recall of EAL Notifier report time requirements.

10 CFR Part 55 Content: 55.41 10

Comments / Reference: PPM 1.3.1		Rev:	Major:	133	Minor:	N/A
Number: 1.3.1		Use Category: INFORMATION		Major Rev: 133		
Title: Operating Policies, Programs and Practices				Minor Rev: N/A		
				Page: 33 of 103		
<p>11. May be assigned to shift personnel who may be assigned other functions.</p> <p>12. One individual performing function of Offsite Notifications. This may be an SRO, IA, STA, field supervisor, qualified NLO or qualified RO.</p> <p>13. The Safe Shutdown Equipment Operator(s) should not be assigned duties that would prevent them from completing Post Fire Safe Shutdown actions within the specified time.</p> <p>14. A minimum of 3 Supervisors is required on each shift, unless authorized by Operations Management on a case-by case basis. Use of relief week and overtime is authorized to cover vacancies. One of the Supervisors may be a supervisor in the field. This Supervisor will spend time outside of the Control Room observing Equipment Operators.</p> <p>15. If unable to maintain required staffing, Operations Management should be notified immediately.</p> <p>Category 1 - report to a station, no action required within response time of 10 minutes. Be in the control room within 10 minutes from anywhere. Risk judgment needed to leave the protected area. If Category 1 personnel are allowed to leave the protected area, log the permission in the Control Room Log.</p> <ul style="list-style-type: none"> • Offsite Notifier • STA/IA 						

Examination Outline Cross-reference: 70

Revision: 0

Date: 05/11/22

Tier: 4

Group: G

K/A Number: 292005 K1.04

Level of Difficulty: 2

RO Importance Rating: 3.5

K/A Description: Predict direction of change in reactor power for a change in control rod position

CGS is operating at steady state 90% power.

A central control rod is moved from notch position 12 to notch position 24.

Assuming reactor recirculation flow remains constant, reactor power will...

- A. increase and stabilize at a higher value.
- B. increase temporarily, then return to the original value.
- C. decrease and stabilize at a lower value.
- D. decrease temporarily, then return to the original value.

Answer: A

K/A Match:

Direct match, changing control rod position (rod insert) and its effect on reactor power.

Explanation:

- A. Correct – Rod being moved from position 12 to position 24 is a rod withdrawal which results in positive reactivity being added to the core. Power will stabilize at a higher value.. .
- B. Incorrect – Plausible – If the candidate confuses reactor power response for at power conditions with response on approach to criticality. On approach to criticality, count rates will change while withdrawing rods but SRM power levels will return to original values following a rod withdrawal.
- C. Incorrect – Plausible – If the candidate confuses a rod withdrawal with a rod insert, a rod insert would result in a reactor power decrease which would stabilize at a lower value.
- D. Incorrect – Plausible – if the candidate confuses a rod withdrawal with a rod insert and neutronic response for at power conditions vrs response on approach to criticality. On approach to criticality, count rates will change while withdrawing rods but SRM power levels will return to original values following a rod withdrawal.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
General Physics	
Energy Northwest EN GFE Study Guide	

Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 292005 K1.04 Predict direction of change in reactor power for a change in control rod position

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Memory of the effects of withdrawing a control rod on reactor power.

10 CFR Part 55 Content: 55.41 1

Comments / Reference: General Physics Cooperation	Rev:	Major:4	Minor:Min
<p>BWR / REACTOR THEORY / CHAPTER 5 / CONTROL RODS</p> <p>By removing a control rod from the core, the control rod's macroscopic cross section decreases absorption ($\Sigma_a^{\text{control rods}}$). This decreases the denominator and increases the overall fraction. The result is more neutrons available for absorption in the fuel. Since the fuel is absorbing more neutrons, the core receives more positive reactivity, and reactor power increases.</p> <p>Conversely, inserting a control rod into the core increases $\Sigma_a^{\text{control rod}}$. This means fewer neutrons are available for absorption in the fuel. Since fewer neutrons are available to cause fission, negative reactivity is added to the core. This negative reactivity causes reactor power to decrease.</p>			

BWR / REACTOR THEORY / CHAPTER 8
/REACTOR OPERATIONAL PHYSICS

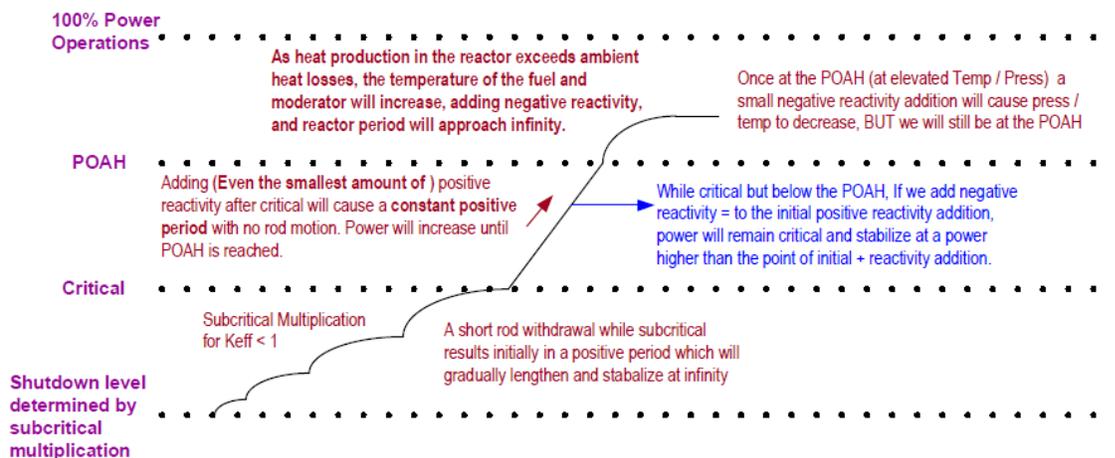
When a control rod is moved out of the core, the SRM period meters show a positive period (its magnitude depends upon the flux that the rod moves through at any given time). Simultaneously, the core average neutron population increases as indicated on the SRM count rate meters and recorders. When rod movement stops, the period meters gradually return to infinity and the neutron population levels off at a higher value.

As k_{eff} approaches 1.0 (criticality), each time a rod is withdrawn it requires a longer time for the period meters to reach infinity and the count level to reach equilibrium. When close to criticality, this time delay can be several minutes.

Comments / Reference: EN GFE Study Guide	Rev:	Major:4	Minor:Min
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Reactor Theory – Control Rods

Changes in Rx Power on start up.



Question: RO-71

Examination Outline Cross-reference: 70

Revision: 0

Date: 05/23/22

Tier: 4

Group: N/A

K/A Number: 292006 K1.01

RO Importance Rating: 3.8

Level of Difficulty: 2

K/A Description: Define fission product poison

Fission fragments or daughters that have substantial neutron absorption cross section and are not fissionable are called...

- A. fissile materials
- B. fission product poisons
- C. fissionable nuclides
- D. burnable poisons

Answer: B

K/A Match:

Define fission product poisons

Explanation:

- A. Incorrect – Plausible -several fission products have appreciable thermal neutron absorption cross sections; however, none can fission. Those fuel types that fission simply due to the neutron binding energy are known as fissile materials.
- B. Correct -the term fission product poison is used to denote a fission product that has a substantial neutron absorption cross section and does not fission.
- C. Incorrect – Plausible - - several fission products have appreciable thermal neutron absorption cross sections; however, none can fission. Those fuels that require additional energy (above the binding energy of the neutron, in the form of neutron kinetic energy) to cause fission are called fissionable materials
- D. Incorrect – Plausible - burnable poisons are intentionally loaded with the fuel to offset excess reactivity. They are designed to absorb a neutron and burn up. The resultant nuclei have low neutron absorption cross sections, and therefore, have no further effect on the neutron flux. Such materials are called burnable poisons.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
NRC GFE B558	
General Physics	

Question: RO-71

Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: Reference or N/A

Learning Objective: Learning Objective.

Question Source: Bank #: NRC GFE B558
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level
 Cog Level Justification.

10 CFR Part 55 Content: 55.41 1

Comments / Reference: NRC GFE Bank	Rev:	Major:	Maj	Minor:	Min
<p>TOPIC: 292006 KNOWLEDGE: K1.01 [2.7/2.8] QID: B558</p> <p>Fission fragments or daughters that have a substantial neutron absorption cross section and are <u>not</u> fissionable are called...</p> <p>A. fissile materials. B. fission product poisons. C. fissionable nuclides. D. burnable poisons.</p> <p>ANSWER: B.</p>					

Comments / Reference: General Physics

Rev:

Major: 4

Minor: Min

BR06Sr4_Fission_Products 2012 March

FISSION PRODUCT POISONING

The term fission product poison is used to denote a fission product that has a substantial neutron absorption cross section and does not fission. Figure 6-1 shows the fission product yield curve as seen previously in Chapter 1.

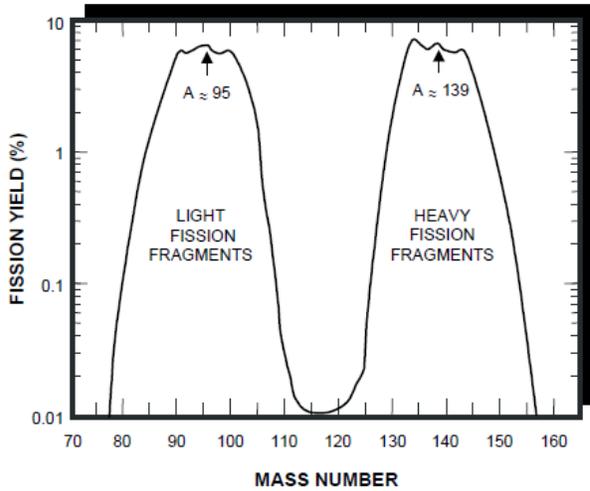


Figure 6-1 Fission Product Yield Curve

Several fission products have appreciable thermal neutron absorption cross sections; however, none can fission. This means that a thermal neutron that is captured by a fission product nucleus is lost from the neutron life cycle. Fission product poison buildup reduces the thermal utilization factor.

Question: RO-72

Examination Outline Cross-reference: 72

Revision: 0

Date: 6/30/22

Tier: 4

Group: N/A

K/A Number: 292008.K1.02

Level of Difficulty: 2

RO Importance Rating: 3.8

K/A Description: List reactivity control mechanisms that exist for plant conditions during the approach to criticality

CGS is in Mode 2.

A reactor plant startup is in progress in accordance with PPM 3.1.2, Reactor Plant Startup.

- The reactor is sub-critical.

Which of the following method(s) is/are used to add positive reactivity during the approach to criticality?

- (1) Control rods
- (2) Steaming rate
- (3) Recirculation flow

A. (1) ONLY

B. (2) ONLY

C. (1) and (3) ONLY

D. (2) and (3) ONLY

Answer: A

K/A Match:

Requires knowledge of reactivity addition methods used during approach to criticality

Question: RO-72

Explanation:

A. Correct. In accordance with PPM 3.1.2, Reactor Plant Startup, control rods are withdrawn to add reactivity to the core.

B. Incorrect. Plausible since steaming rate may be used to add reactivity during reactor operations in Mode 1. However, in Mode 2, with the reactor sub-critical, no steam is produced. Therefore, steaming rate cannot add positive reactivity.

C. Incorrect. Plausible since (1) is correct. Plausibility is enhanced since recirculation flow is used to control reactor power in Mode 1. However, recirculation flow does not add reactivity in a sub-critical reactor.

D. Incorrect. Plausible since (2) and (3) are used to control reactor power in Mode 1. However neither of these methods can be use to add reactivity in a sub-critical reactor.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 3.1.2, Reactor Plant Startup Flowcharts	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 103630 - List reactivity control mechanisms that exist for plant conditions during the approach to criticality

Question Source: Bank #: NRC GFES Bank B1065
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

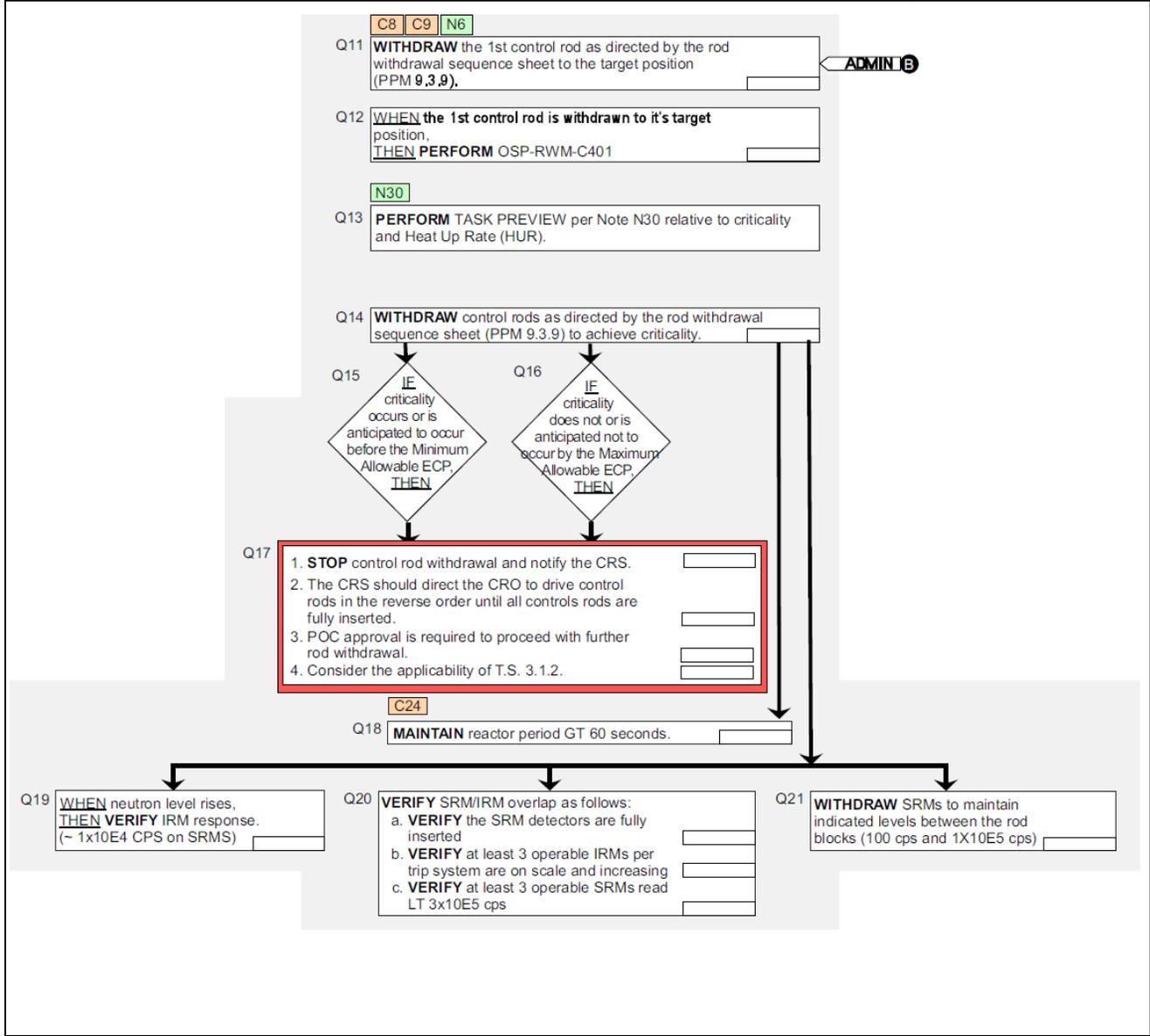
Justification for Cognitive Level

Requires knowledge of reactivity addition methods used during reactor startup

10 CFR Part 55 Content: 55.41 1

Question: RO-72

Comments / Reference: PPM 3.1.2 Reactor Plant Startup Flowcharts	Rev:	Major: 093	Minor: 000
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Question: RO-73

Examination Outline Cross-reference: 73

Revision: 0

Date: 6/30/22

Tier: 4

Group: N/A

K/A Number: 293007.K1.03

Level of Difficulty: 3

RO Importance Rating: 2.8

K/A Description: Heat Transfer - Explain the manner in which fluid films affects heat transfer

As fluid flow rate increases through the tubes of a shell-and-tube heat exchanger, the laminar film thickness (1), which causes the heat transfer rate to (2).

- A. (1) increases
(2) decrease
- B. (1) increases
(2) increase
- C. (1) decreases
(2) decrease
- D. (1) decreases
(2) increase

Answer: D

K/A Match:

Requires knowledge of how changes in fluid flow affect the laminar flow layer and how this change affects heat transfer rate.

Question: RO-73

Explanation:

A. Incorrect. Plausible since an increase in laminar film thickness will cause heat transfer rate to decrease. However, an increase in fluid flowrate will cause the laminar film thickness

B. Incorrect. Plausible since (2) is correct. However, when the rate of fluid flow increases through a heat exchanger, the laminar boundary decreases.

C. Incorrect. Plausible since (1) is correct. However, when the laminar layer decreases, better mixing occurs and the heat transfer rate increases

D. Correct. When flow rate through a heat exchanger increases, the laminar layer will decrease. This allows better mixing and heat is removed into the fluid channel at a higher rate. Therefore, heat transfer rate will increase.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
General Physics, chapter 7, BWR Thermodynamics – Heat Transfer	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 103749 - Explain the manner in which fluid films affects heat transfer

Question Source: Bank #: Bank #
 Modified Bank #: NRC GFE Bank B286 (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize an understanding of how changes in fluid flowrate affect the laminar flow layer with the knowledge of how the size of the laminar flow layer affects heat transfer rate.

10 CFR Part 55 Content: 55.41 1

Question: RO-73

Comments / Reference: General Dynamics, Chapter 7	Rev:	Major: 3	Minor: N/A
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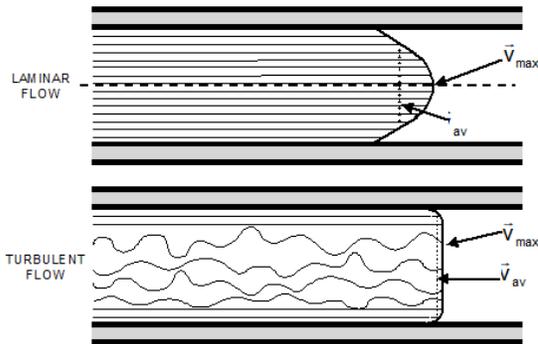


Figure 7-7 Laminar and Turbulent Velocity Profiles

The mode of flow, whether laminar or turbulent, is important both in the design and in the operation of any fluid flow system. The amount of fluid friction, which determines the amount of energy required to create the desired flow, depends on the mode of flow. The amount of mixing within the fluid also depends on the mode of flow. This is an important consideration in certain applications involving heat transfer to the fluid. This is because the temperature profile within the fluid is significant. Operationally, it is important to visualize the two different velocity profiles associated with the two different modes of flow. It is also essential to understand that the mode of flow depends not only on the average velocity of the fluid, but on the diameter of the pipe and the viscosity of the fluid. The Reynolds number provides the basic relationship. Thus, for low Reynolds numbers, the flow is laminar and has the characteristic laminar flow velocity profile. As the Reynolds number increases, the flow becomes turbulent and the turbulent flow velocity profile develops.

BOUNDARY LAYER FUNDAMENTALS

When a fluid flows along a surface, irrespective of whether the flow is laminar or turbulent, the particles in the vicinity of the surface are slowed by virtue of viscous forces. The fluid particles adjacent to the surface stick to it and have zero velocity relative to the boundary.

The boundary layer can be viewed as being comprised of three parts. The particles adjacent to the wall have a velocity of zero, the laminar sublayer where flow is not stagnant yet remains laminar, and the buffer layer which is the region between the laminar sub-layer and the completely turbulent portion of flow.

When the fluid velocity and the turbulence are small, the transport of energy across the fluid is not aided materially by mixing currents on a macroscopic scale. On the other hand, when the velocity is large and the mixing between warmer and colder fluids contributes substantially to the energy transfer, the conduction becomes less important. Consequently, to transfer heat through a fluid at a given rate, a larger temperature gradient is required in a region of low velocity than in a region of high velocity.

Applying these observations to heat transfer from a solid wall to a fluid in turbulent flow, we can roughly sketch the temperature profile. In the immediate vicinity of the wall, heat can only flow by conduction, because the fluid particles are stationary relative to the boundary. A large temperature drop occurs across this layer. Further from the wall, the movement of the fluid aids in the energy transport, and the temperature gradient will be less steep, eventually leveling out in the main stream.

Question: RO-73

The temperature distribution for a fluid flowing past a hot wall is shown in Figure 7-8. The temperature gradient in the fluid is confined to a relatively thin layer in the vicinity of the surface. The temperature profile represents the fluid temperature transition from surface temperature to fluid bulk temperature. We shall now simplify the true picture by replacing the actual temperature distribution profile (solid line) with the calculational temperature distribution profile (represented by the dashed line). The calculational profile (dashed line) is tangent to the actual temperature curve at the wall. It physically represents the temperature distribution in a hypothetical layer, which, if completely stagnant, offers the same thermal resistance to flow as the actual boundary layer.

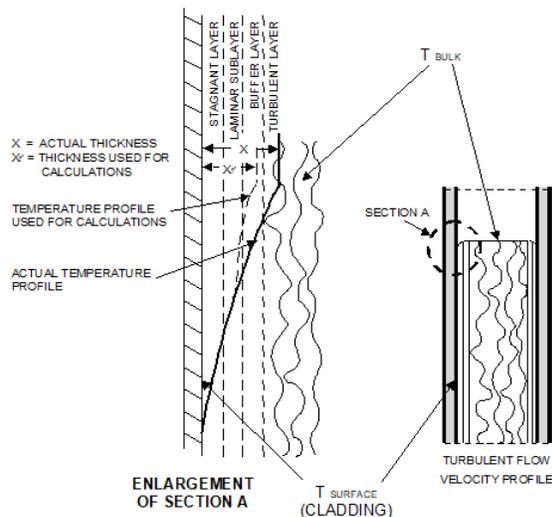


Figure 7-8 Temperature Distribution Profile

In the stagnant layer, heat flows only by conduction. Therefore, the heat flux (Q'') is expressed as:

$$Q'' = h(T_s - T_{\text{bulk}})$$

$$Q'' = \frac{k_f}{X_f}(T_s - T_{\text{bulk}})$$

Where:

- Q'' = heat flux (Btu/hr ft²)
- h = convection heat transfer coefficient (Btu/hr ft² °F)
- T_s = temperature at surface (°F)
- T_{bulk} = bulk fluid temperature (°F)
- k_f = stagnant film thermal conductivity (Btu/hr ft °F)
- X_f = thickness of the stagnant film layer (ft)

Equation 7-22

Although this is an oversimplified picture, it illustrates that as the boundary layer becomes thinner, the convection heat transfer coefficient becomes larger.

Several factors affect the heat transfer coefficient for a fluid film. Most of these affect the film thickness. Film thickness in conjunction with the fluid thermal conductivity determines the heat transfer coefficient.

- Fluid velocity - The greater the velocity of the fluid stream, the thinner the fluid film. This causes the heat transfer coefficient to increase.
- Fluid thermal conductivity - An increase in thermal conductivity results in an increase of the heat transfer coefficient.

Question: RO-73

Comments / Reference: Parent Question	Rev:	Major:	Maj	Minor:	Min
<p>TOPIC: 293007 KNOWLEDGE: K1.03 [2.7/2.8] QID: B286</p> <p>As fluid flow rate <u>decreases</u> through the tubes of a shell-and-tube heat exchanger, the laminar film thickness _____, which causes the heat transfer rate to _____.</p> <p>A. increases; decrease B. increases; increase C. decreases; decrease D. decreases; increase</p> <p>ANSWER: A.</p>					

Question: RO-74

Examination Outline Cross-reference: 74

Revision: 0

Date: 7/5/22

Tier: 4

Group: N/A

K/A Number: 293008.K1.28

Level of Difficulty: 3

RO Importance Rating: 2.5

K/A Description: Thermal-Hydraulics - Describe the factors affecting single- and two-phase flow resistance

Single-phase coolant flow resistance in a reactor core is directly proportional to the square of coolant (1); and inversely proportional to (2).

- A. (1) velocity
(2) fuel assembly length
- B. (1) velocity
(2) coolant channel cross-sectional area
- C. (1) temperature
(2) fuel assembly length
- D. (1) temperature
(2) coolant channel cross-sectional area

Answer: B

K/A Match:

Requires knowledge of factors that affected single-phase flow resistance.

Question: RO-74

Explanation:

A. Incorrect. Plausible since (1) is correct. Plausibility is enhanced since total flow resistance is proportional to the length of the fuel assembly. However, it is directly proportional to length, not inversely proportional.

B. Correct. The formula for flow resistance (head loss) is $H_f = f \left(\frac{L}{D} \right) \left(\frac{\bar{v}_{av}^2}{2g} \right)$, where L is pipe length, D is pipe diameter, and v_{av}^2 is the square of average fluid velocity. Therefore, flow resistance is proportional to the square of coolant velocity and inversely proportional to channel cross-sectional area.

C. Incorrect. Plausible since both temperature and fuel assembly length affect flow resistance. However, the relationship between temperature and resistance is not a square function, and pipe length is not inversely proportional to flow resistance, but rather directly proportional.

D. Incorrect. Plausible since (2) is correct. However, the relationship between temperature and resistance is not a square function.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
General Physics BWR Thermodynamics Chapter 6	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: Learning Objective.

Question Source: Bank #: NRC GFE bank #B1789
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level
 Requires recall of factors affecting flow resistance

10 CFR Part 55 Content: 55.41 14

Comments / Reference: GP Thermodyanmics	Rev:	Major:3	Minor:N/A
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KEY POINTS, AIDS, QUESTIONS/ANSWERS	INSTRUCTOR GUIDE
<p>Objective 25</p> <p>Equation 6-19 / TP 6-56</p> $H_f = f \left(\frac{L}{D} \right) \left(\frac{\vec{v}_{av}^2}{2g} \right)$	<ol style="list-style-type: none"> a. They are greatest as fluid flows through entrances, exits, pumps, valves, fittings, and any piping with rough inner surfaces b. Most techniques for evaluating head loss due to fluid friction are empirical and are based on proportionality constant, called friction factor (f) c. Head loss is conversion of fluid pressure and velocity to heat energy through friction 4. Experimental studies of flow of liquids in pipes showed that head loss due to fluid friction varies: <ol style="list-style-type: none"> a. Directly with length of pipe, since longer pipe has more surface area b. Inversely with diameter of pipe, since pipe with larger diameter has less surface area per unit of cross-sectional area than pipe with smaller diameter c. Directly with velocity head of fluid 5. The proportionality constant is friction factor (f) 6. Where: <ol style="list-style-type: none"> a. H_f = head loss due to fluid friction (ft) b. f = friction factor (no units) c. L = length of pipe (ft) d. D = diameter of pipe (ft)

Examination Outline Cross-reference: 75

Revision: 0

Date: 7/5/22

Tier: 4

Group: N/A

K/A Number: 293009.K1.24

Level of Difficulty: 4

RO Importance Rating: 3.2

K/A Description: Core Thermal Limits - Describe the effects of pressure on critical power

CGS is in Mode 1.

How does raising RPV pressure affect Critical Power?

Raising RPV pressure causes Critical Power to (1) because latent heat of vaporization for the reactor coolant (2).

- A. (1) increase
(2) decreases
- B. (1) increase
(2) increases
- C. (1) decrease
(2) decreases
- D. (1) decrease
(2) increases

Answer: C

K/A Match:

Requires knowledge of the effects of RPV pressure on critical power.

Explanation:

- A. Incorrect. Plausible since (2) is correct. However, when the latent heat of vaporization (LHV) decreases, less heat is required to be added to reach the Onset of Transition Boiling (OTB). Therefore, Critical Power decreases.
- B. Incorrect. Plausible since Critical Power is increase if the coolant LHV increases. However, for the conditions given LHV will decrease, causing Critical Power to decrease.
- C. Correct. As pressure increases the amount of enthalpy required to change coolant state (liquid to vapor) decreases. Therefore, the coolant is closer to OTB and Critical Power decreases.
- D. Incorrect. Plausible since (1) is correct. However, as pressure increases, the coolant latent heat of vaporization decreases.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
General Dynamics Thermodynamics Chapter 9 – Core Thermal Limits	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 103844 – Describe the effects of pressure on critical power.

Question Source: Bank #: NRC GFE bank #B995
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires knowledge of how RPV pressure affects critical power

10 CFR Part 55 Content: 55.41 14

Comments / Reference: GP Chapter 9, Core Thermal Limits	Rev:	Major:3	Minor:N/A
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KEY POINTS, AIDS, QUESTIONS/ANSWERS

INSTRUCTOR GUIDE

Objective [20.b](#)

Objective 20.c

Objective [20.d](#)

Objective [20.e](#)

d. Critical power increases with higher mass flow rates since at higher flow rates, greater power input is required to raise coolant enthalpy to saturation conditions and change water to steam

2. The effect of pressure on critical power is very small in nonboiling region of bundle

a. When coolant is saturated, however, lower enthalpy rise is required to change given mass of coolant from liquid state to vapor state at higher pressure than at lower pressure

b. This can be seen in steam tables, where enthalpy of evaporation (h_{fg}) decreases as pressure increases at high pressure

c. Thus, power required to reach OTB is less at higher pressure

3. For given power distribution and given plane in bundle, highest quality is expected to occur around hottest pin

a. As we have seen, ratio of hottest pin power, compared to average pin power, in plane is known as local peaking factor

b. As local peaking factor increases at given power level, it is expected that hottest pin is closer to OTB

c. Therefore, critical power is expected to decrease as local peaking increases

4. The critical power of bundle also depends on axial power distribution

a. In bundle with bottom power peak, most rapid enthalpy rise in coolant occurs in areas of low quality or even subcooled liquid, thus providing large margin to OTB

KEY POINTS, AIDS, QUESTIONS/ANSWERS

INSTRUCTOR GUIDE

Objective 20

Table 9-1 / TP 9-30

FACTOR	CRITICAL POWER	BUNDLE POWER	CPR
INLET SUBCOOLING:			
INCREASES	↑	↑	↓
DECREASES	↓	↓	↑
MASS FLOW RATE:			
INCREASES	↑	↑	↓
DECREASES	↓	↓	↑
PRESSURE:			
INCREASES	↓	↑	↓
DECREASES	↑	↓	↑
LOCAL PEAKING FACTOR			
INCREASES	↓	↔	↓
DECREASES	↑	↔	↑
AXIAL POWER DISTRIBUTION			
INCREASES	↓	↔	↓
DECREASES	↑	↔	↑

Objective 17

- b. In top-peaked bundle, largest coolant enthalpy rise occurs in high quality region that is closer to OTB
- c. Therefore, critical power is expected to be lower for top-peaked power distribution
- d. Table 9-1 provides summary on how each thermal limit is affected by specific parameter changes

K. Steady State And Transient

1. The primary design objective is to maintain nucleate boiling and avoid OTB
 - a. The CPR thermal limit is set to maintain adequate margin between nucleate boiling and OTB
 - b. The steady state and transient MCPR thermal limits are derived from this single design basis requirement
 - c. Transients caused by single operator error or equipment malfunction shall be limited so that, considering uncertainties in monitoring core operating state, more than 99.9% of fuel rods are expected to avoid OTB

Question: SRO- 76

Examination Outline Cross-reference: 76

Revision: 0

Date: 9/20/22

Tier: 1

Group: 1

K/A Number: 295001.2.4.45

Level of Difficulty: 2

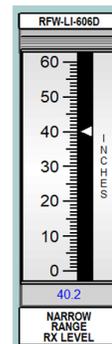
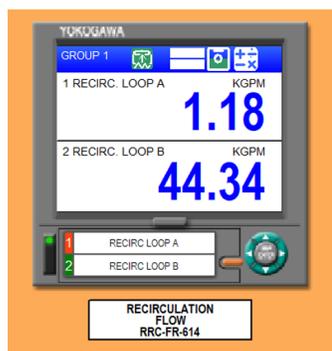
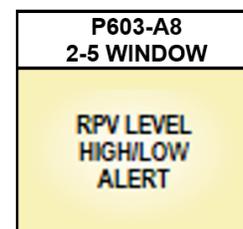
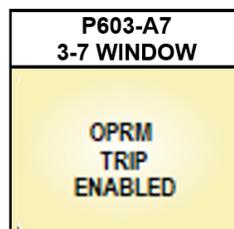
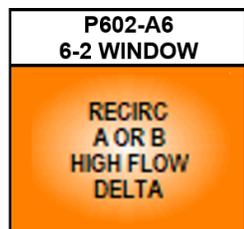
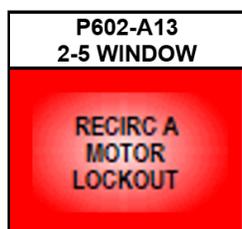
SRO Importance Rating: 4.3

K/A Description: Partial or Complete Loss of Forced Core Flow Circulation - Ability to prioritize and interpret the significance of each annunciator or alarm

CGS is operating at 100% power.

- OPRM-1 is inoperable.

A transient occurs resulting in the following annunciators and indications:



What actions should the CRS direct?

- Enter ABN-CORE and MANUALLY SCRAM the Reactor.
- Enter ABN-LEVEL and SCRAM the Reactor per PPM 3.3.1.
- Enter ABN-POWER and ADJUST RRC flow to balance loop flows.
- Enter ABN-RRC-LOSS and ADJUST RRC Loop B Drive Flow to LT 41,725 gpm.

Answer: D

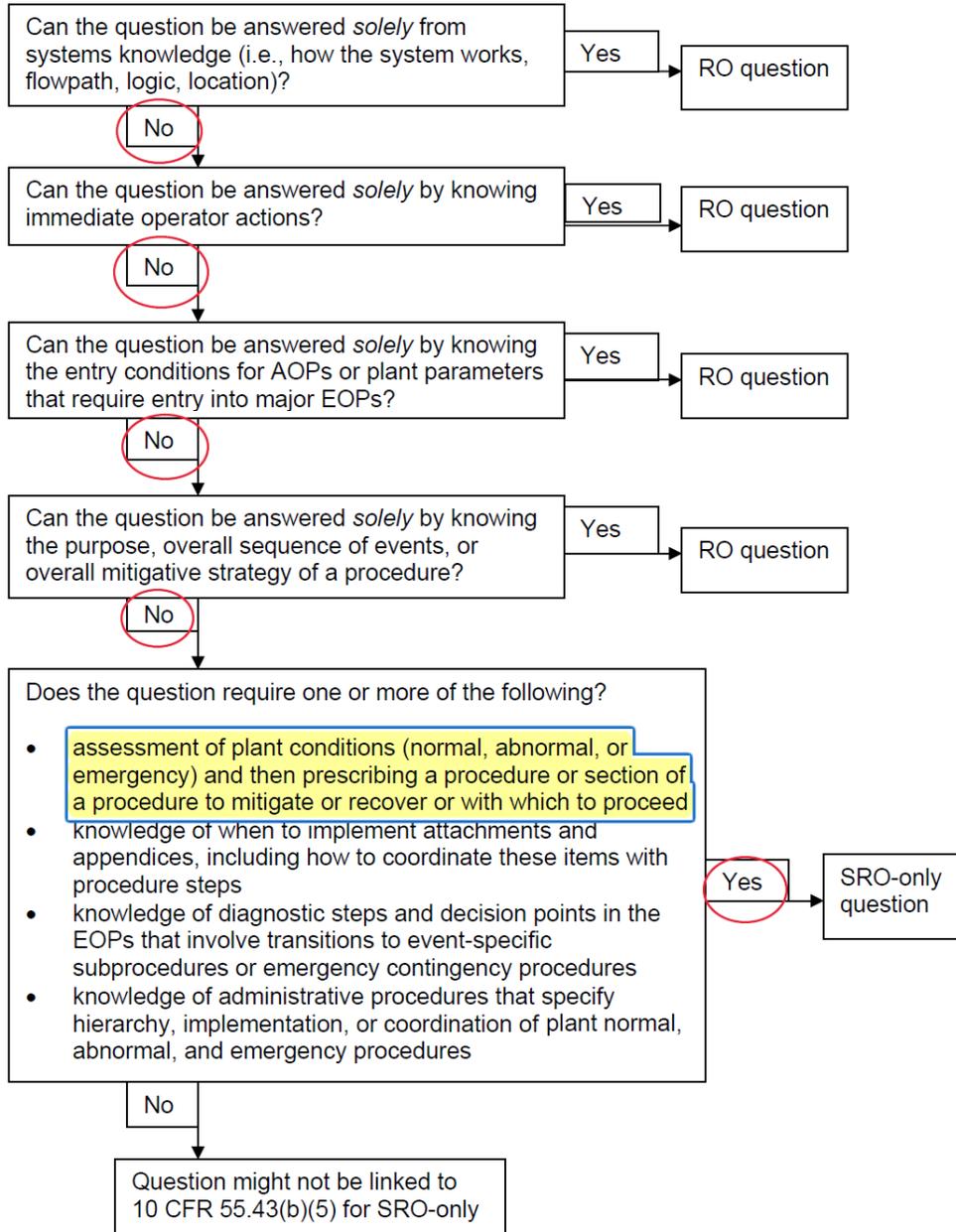
K/A Match:

Question: SRO- 76

The annunciators and indications signify a loss of RRC-P-1A. The candidate must interpret the annunciators and indications correctly to prioritize the necessary mitigative actions.

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Question: SRO- 76

Explanation: The annunciators and indications are indicative of a loss of RCC-P-1A from 100% power.

A. Incorrect: Plausible because ABN-CORE directs actions to “MANUALLY SCRAM the Reactor” in the event operation is in Region A of the Power to Flow map and the OPRM is inoperable. Although the stem specifies that OPRM-1 is inoperable, the OPRM is OPERABLE because 3 other channels are still available. Additionally, the plant is not in Region A. Per ABN-RRC-LOSS, if RRC Drive flow is 41,725 gpm or greater, the plant will NOT be in Region A. Per RRC-FR-614, RRC-B Drive flow is 44,340 gpm.

B. Incorrect: Plausible because ABN-LEVEL directs actions to “SCRAM the Reactor per PPM 3.3.1” in the event reactor power is GE 5% and RPV level GT +54.5” is imminent. RRC-A tripping from 100% power results in a high RPV level, however, the RPV level exceeding +54.5” is NOT imminent.

C. Incorrect: Plausible because ABN-POWER directs actions to balance loop flows, however, these actions can only be performed if in two RRC loop operations. RRC-P-1A has been lost and the plant is in single loop operation.

D. Correct: On a loss of RRC-P-1A and RRC Loop B Drive Flow GT 41,725 gpm, ABN-RRC-LOSS directs actions to “ADJUST RRC Loop B Drive Flow to LT 41,725 gpm.” Per RRC-FR-614, RRC Loop B Drive flow is 44,340 gpm.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-CORE, Unplanned Core Operating Conditions	
ABN-LEVEL, Unplanned RPV Water Level Change	
ABN-POWER, Unplanned Reactor Power Change	
ABN-RRC-LOSS, Loss of Reactor Recirculation Flow	

Proposed references to be provided during examination: Control Room indication pictures in question stem.

Learning Objective: SRO-0659 Direct response to a loss of RRC flow.

Question Source: Bank #: _____ Bank #
 Modified Bank #: _____ Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

The Candidate must synthesize a knowledge of multiple annunciators and indications to understand the significance of an event and determine which actions need to be prioritized.

Question: SRO- 76

10 CFR Part 55 Content: 55.43 5

Comments / Reference: ABN-CORE	Rev:	Major: 021	Minor: N/A
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Excerpt from ABN-CORE:

Number: ABN-CORE	Use Category: CONTINUOUS	Major Rev: 021
Title: Unplanned Core Operating Conditions		Minor Rev: N/A Page: 5 of 25

3.0 IMMEDIATE OPERATOR ACTIONS

3.1 IF reactor mode switch is in **RUN** or **STARTUP**,
AND both RRC pumps have tripped off (i.e., neither RRC pump is running),
THEN **MANUALLY SCRAM the Reactor.** {R-6.2} _____

3.2 IF operating in the OPRM Enabled Region,
AND any of the following conditions exist,
THEN **MANUALLY SCRAM the Reactor.** {R-6.12} _____

- Any unexplained significant and sustained periodic LPRM upscale or downscale alarms.
- Any unexplained significant and sustained oscillations in SRM period, LPRM or APRM levels (characteristic oscillation period is inside 1-3 seconds). The following are provided as examples and do not represent absolute values:
 - Sustained Peak to Peak APRM exceeds 10%. {C-6.1}
 - Any LPRM has oscillations GT 20 % Flux as determined by selecting an interior control rod (such as 30-43, 14-35, 42-27, 30-39), and monitoring the RBM ODA. {C-6.1}
- OPRM TRIP setpoints (H13-P603.A7-2.4) are verified on two or more channels.
- Evaluation of Power to Flow conditions indicate that operation is currently in Region A of the Power to Flow map and the OPRM is inoperable.

Comments / Reference: ABN-LEVEL	Rev:	Major: 007	Minor: N/A
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Excerpt from ABN-LEVEL:

Number: ABN-LEVEL	Use Category: CONTINUOUS	Major Rev: 007
Title: Unplanned RPV Water Level Change		Minor Rev: 003 Page: 8 of 17

4.2 HIGH RPV LEVEL

4.2.1 IF the reactor has scrammed,
THEN **PERFORM** PPM 3.3.1. _____

4.2.2 IF RPV level GT +54.5" is imminent,
AND reactor power is GE 5%,
THEN **SCRAM the Reactor** per PPM 3.3.1. _____

Question: SRO- 76

Comments / Reference: ABN-POWER	Rev:	Major: 018	Minor: N/A
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Excerpt from ABN-POWER:

Number: ABN-POWER	Use Category: CONTINUOUS	Major Rev: 018
Title: Unplanned Reactor Power Change		Minor Rev: N/A
		Page: 5 of 17

4.0 SUBSEQUENT OPERATOR ACTIONS

NOTE: Refer to Section 4.1 for RRC Flow change.
Refer to Section 4.2 for Jet Pump failure.
Refer to Section 4.3 for a unplanned feedwater temperature reduction.

4.1 RRC Flow Change

CAUTION

Reducing the core flow to LT ~64% may cause the OPRM to be Enabled. The OPRM trip functions will enable when the drive flow as measured by the APRM is reduced to LT 60% drive flow. If the OPRM enables then this is considered an entry into the OPRM Enabled Region and requires entry into ABN-CORE. Reducing core flow to LT 55% of rated flow may cause entry into the Area of Increased Awareness of the Power-to-Flow map.

4.1.1 IF RRC pump speed is lowering for one pump,
AND cannot be controlled,
THEN STOP the affected pump prior to exceeding the allowable mismatch in Attachment 7.2. _____

4.1.2 IF RRC flow is fluctuating,
THEN PLACE the RRC pump controllers in **MANUAL**,
AND VERIFY flow has stabilized,
OR STOP the uncontrolled pump. _____

4.1.3 IF RRC flow has risen,
AND RRC system flow control is restored,
THEN REDUCE RRC flow to the pre-transient value. _____

4.1.4 IF the OPRM Enables (APRM STP GE 24.6% and RRC Drive Flow LT 60% as specified in the COLR),
THEN REFER to ABN-CORE. _____

4.1.5 **REFER** to Technical Specification 3.4.1. _____

4.1.6 IF RRC flow must be changed to balance loop flows,
THEN REFER to the current power to flow map,
AND ADJUST RRC flow to match,
OR DECLARE the loop with the lower flow not in operation. _____

Question: SRO- 76

Comments / Reference: ABN-RRC-LOSS	Rev:	Major: 019	Minor: N/A
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Excerpt from ABN-RRC-LOSS:

Number: ABN-RRC-LOSS	Use Category: CONTINUOUS	Major Rev: 019
Title: Loss of Reactor Recirculation Flow		Minor Rev: N/A
		Page: 6 of 35

4.2 **Loss of RRC-P-1A**

CAUTION

If operating on a rod line GT 70%, it is possible to enter Region A following the removal of RRC-P-1A from service. Operation in Region A has an elevated risk of a core instability event. Operation in Region A is an accepted risk. Minimize operation in Region A by reducing the rod line as soon as practical to reduce the potential of a core instability event. {P-77714}

NOTE: This section provides proper actions to be taken in the event of an automatic tripping of RRC-P-1A or when RRC-P-1A is manually tripped per procedural direction.

NOTE: **If RRC Drive flow is 41,725 gpm or greater, the plant will NOT be in Region A.** If Jet Pump Loop Flow is 57 Mlb/hr, Region A will be exited at ~95% rod line.

NOTE: RRC Loop B Drive Flow indication in gpm can be obtained from any of the following: **RRC-FR-614**, RRC-FI-M/A/R676B; computer point B039GPM, B130GPM, B026GPM, or B040GPM.

NOTE: Jet Pump Loop B Flow indication in Mlb/hr can be obtained from either of the following: MS-FI-611B or TDAS pt. X049.

- 4.2.1 **CONCURRENTLY** enter ABN-CORE for AIA and OPRM Enabled entry. _____
- 4.2.2 **LOG** the time Single Loop was entered in the electronic logging system. _____
- 4.2.3 **VERIFY** RRC-M/A-676B is in **MANUAL** (Loop "B" Auto/Manual Controller) (H13-P602). _____
- 4.2.4 **IF** RRC Loop B Drive Flow is GT 41,725 gpm (RRC-FR-614), **THEN ADJUST** RRC Loop B Drive Flow to LT 41,725 gpm as follows: _____
{P-113021}

Examination Outline Cross-reference: 77

Revision: 0

Date: 9/20/22

Tier: 1

Group: 1

K/A Number: 295005.2.4.21

Level of Difficulty: 2

SRO Importance Rating: 4.6

K/A Description: Main Turbine Generator Trip - Knowledge of the parameters and logic used to assess the status of emergency operating procedures critical safety functions or shutdown critical safety functions

CGS is in Mode 1.

An event requires the crew to manually scram the reactor.

Current plant conditions:

- Reactor Mode switch is in SHUTDOWN.
- Reactor power: 30% and stable.
- MSIVs are open.
- The Main Turbine has not tripped.

NO additional operator actions have been taken.

How should the crew operate the main turbine, and what is the reason for this strategy.

The CRS should direct (1). This is accomplished to prevent (2).

- A. (1) tripping the Main Turbine
(2) localized fuel damage due to power oscillations
- B. (1) tripping the Main Turbine
(2) exceeding HCTL due to excessive energy released in Primary Containment
- C. (1) maintaining the Main Turbine operating
(2) localized fuel damage due to power oscillations
- D. (1) maintaining the Main Turbine operating
(2) exceeding HCTL due to excessive energy released in Primary Containment

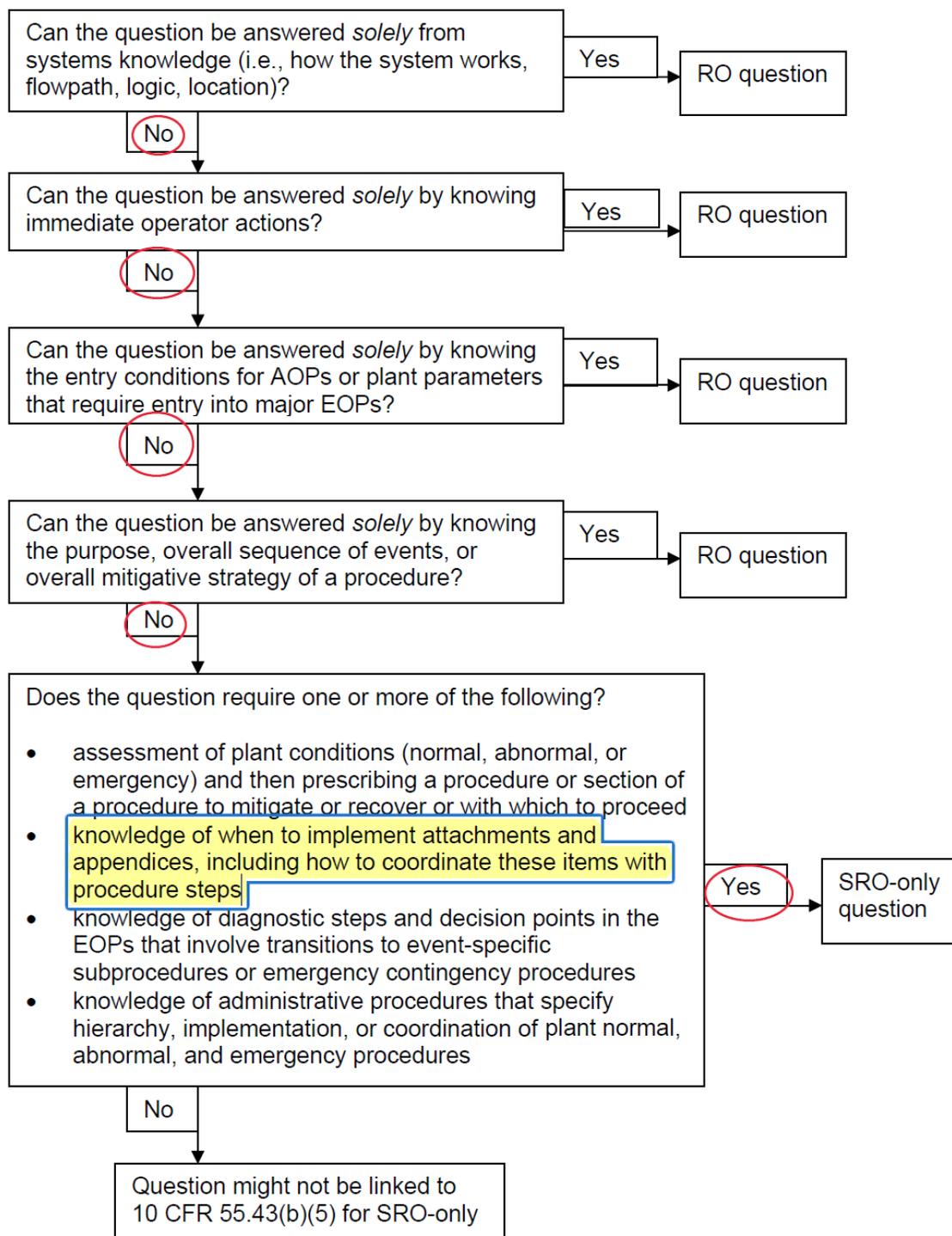
Answer: D

K/A Match:

Requires knowledge of actions in the EOPs to maintain the Containment Critical Safety Function by not initiating a Main Turbine trip while in ATWS conditions.

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Explanation:

- A. Incorrect. Plausible since tripping the Main Turbine is the correct action for non-ATWS conditions in accordance with PPM 3.3.1, Reactor Scram. Plausibility is enhanced since, during and ATWS, actions are taken to reduce power oscillations (lowering RPV level), and thereby reduce the chance of localized fuel damage. However, in ATWS conditions, actions are taken to prevent the Main Turbine from tripping to minimize heat added to primary containment from SRV actuation and prevent exceeding HCTL.
- B. Incorrect. Plausible since tripping the Main Turbine is the correct action for non-ATWS conditions in accordance with PPM 3.3.1, Reactor Scram. Plausibility is enhanced since the Main Turbine is operated during an ATWS to minimize the possibility of exceeding HCTL by reducing the amount of heat added to primary containment. However, during an ATWS, actions are taken to prevent the Main Turbine from tripping.
- C. Incorrect. Plausible since (1) is the correct action for the conditions given. However, this is performed to reduce the amount of heat added to the primary containment due to SRV operation.
- D. Correct. During ATWS conditions, the crew should enter PPM 5.1.1, RPV Control, based on ATWS conditions and RPV water level. Step RC-3 directs performance of table A1. Table A1 directs closing RCIC-V-1 to prevent tripping the Main Turbine. PPM 5.0.10A, Flowchart Training Manual Volume II, discusses the need to maintain the main condenser as a heat sink to minimize the pressure rise of primary containment due to additional heat added to the wetwell from SRV actuation. Maintaining the Main Turbine in operation ensures that SRV operation will be minimized.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 3.3.1, Reactor Scram	
PPM 5.1.1, RPV Control	
PPM 5.0.10A, Flowchart Training Manual Volume II	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 6301 - With procedures available identify under what conditions the Main Turbine should not be tripped after a reactor scram.

Question Source: Bank #: _____ Bank #
 Modified Bank #: _____ Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

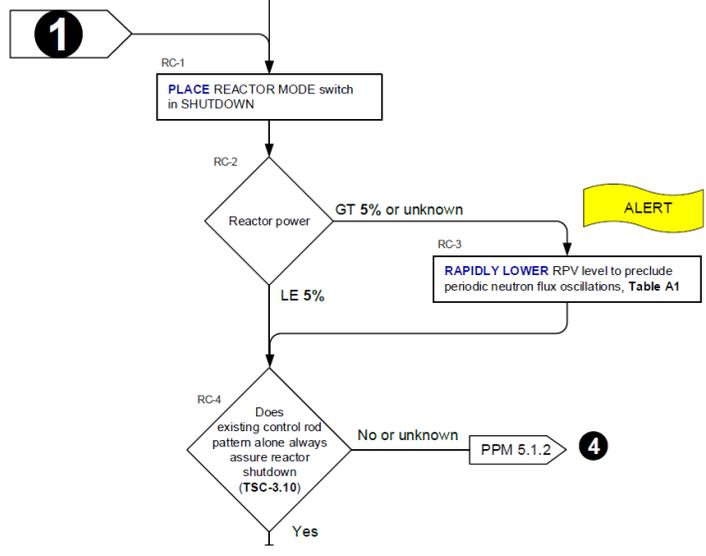
Justification for Cognitive Level

Requires candidate to synthesize a knowledge of required operation of the main turbine during ATWS conditions with a knowledge of the reason for these actions.

Comments / Reference: PPM 3.3.1	Rev:	Major:	Maj	Minor:	Min
<p>5.2.2 IF in an ATWS GT 5% Reactor power, AND directed by the CRS, THEN PERFORM the following:</p> <p>a. IF Main Turbine is online, THEN CLOSE RCIC-V-1 to prevent Main Turbine trip.</p>					_____
<p>5.3.2 WHEN the Main Generator output is LT 50 MWE, THEN PERFORM the following:</p> <p>a. VERIFY the Main Turbine trips.</p>					_____
<p>2H b. IF the Main Turbine fails to trip, THEN SIMULTANEOUSLY DEPRESS both Emergency Trip pushbuttons (H13-P820).</p> <p>c. IF the Main Generator has NOT tripped, THEN DEPRESS Unit Emergency Trip pushbutton (H13-P800) OR DEPRESS Unit Overall Trip pushbutton (H13-P800).</p>					_____ _____

Comments / Reference: PPM 5.1.1	Rev:	Major:	23	Minor:	N/A
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- RPV level below +13 in.
- RPV pressure above 1060 psig
- Drywell pressure above 1.68 psig
- Both:
 - a reactor scram is required
 - AND
 - reactor power is above 5% or cannot be determined



A1	High Power ATWS Condition Actions
<p>PERFORM the following:</p> <ul style="list-style-type: none"> • INHIBIT ADS • PREVENT main turbine trip by closing RCIC-V-1 if needed • BORON INJECT REQ'D, INJECT boron into RPV using SLC • IF any main steam line is open THEN START defeating low RPV level and high steam tunnel temp MSIV isolation interlocks to maintain the main condenser as a heat sink AND ALIGN firewater cooling to CAS air compressors as necessary • IF RPV level can be determined to be above -65 in. THEN STOP and PREVENT all injection into the RPV EXCEPT from boron injection, RCIC and CRD UNTIL RPV level drops below -65 in. <div style="border: 1px solid black; border-radius: 10px; padding: 5px; margin: 10px 0; text-align: center;"> Rapid injection may cause fuel damage </div> <p>MAINTAIN RPV level between -186 in. and -65 in. using one or more Preferred Injection Systems, Table L1, and Alternate Injection Subsystems, Table L2</p>	

Comments / Reference: PPM 5.0.10A	Rev:	Major: Maj	Minor: Min
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e. Table A1

1. First bullet: Under ATWS conditions, RPV water level may be deliberately lowered to below the ADS initiation setpoint to preclude thermal-hydraulic instabilities or reduce reactor power. ADS actuation could impose a severe thermal transient on the RPV and complicate efforts to maintain RPV water level within the preferred control band. Furthermore, rapid uncontrolled injection of large amounts of relatively cold, unborated water could occur as RPV pressure decreases below the shutoff head of low pressure injection system pumps. Such an occurrence could quickly dilute the in core boron concentration and reduce reactor coolant temperature. If the reactor is not shutdown or the shutdown margin is small, sufficient positive reactivity could be added to cause a reactor power excursion large enough to severely damage the core. Automatic ADS initiation is prevented as an initial step in ATWS mitigation to preclude these undesirable consequences. Explicit direction to depressurize the RPV when appropriate is provided in later steps of the EOPs-Hot, maintaining direct control of the transient and obviating the need for automatic depressurization.
- 1) Second bullet: For RCIC operation when the main turbine is in service, preventing RCIC operation may be necessary to avoid tripping the main turbine. Closing RCIC-V-1 will prevent RCIC injection.

Conditions:

- 8) An ATWS coupled with an MSIV isolation may result in rapid heatup of the Wetwell due to the steam discharged from the RPV via SRVs. The resulting containment challenge defines the second of the two conditions requiring boron injection in this step.

Examination Outline Cross-reference: 78

Revision: 0 Date: 11/15/22

Tier: 1

Group: 1

K/A Number: 295019.AA2.01

SRO Importance Rating: 3.9

Level of Difficulty: 2

K/A Description: Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of Instrument Air: Instrument air pressure

CGS is in Mode 1.

Containment Instrument Air (CIA) pressure is lowering.

The CRS enters ABN-CIA, Containment Instrument Air System Failures.

- CIA Main Header pressure (CIA-PI-20) is 78 psig, down slow.

What is the next action that the CRS should direct?

- A. Scram the reactor in accordance with PPM 3.3.1.
- B. Supply CIA with Control Air System (CAS) in accordance with SOP-CIA-OPS.
- C. Start an immediate controlled Reactor shutdown in accordance with PPM 3.2.1.
- D. Ensure proper Nitrogen valve alignment in accordance with SOP-CN-START.

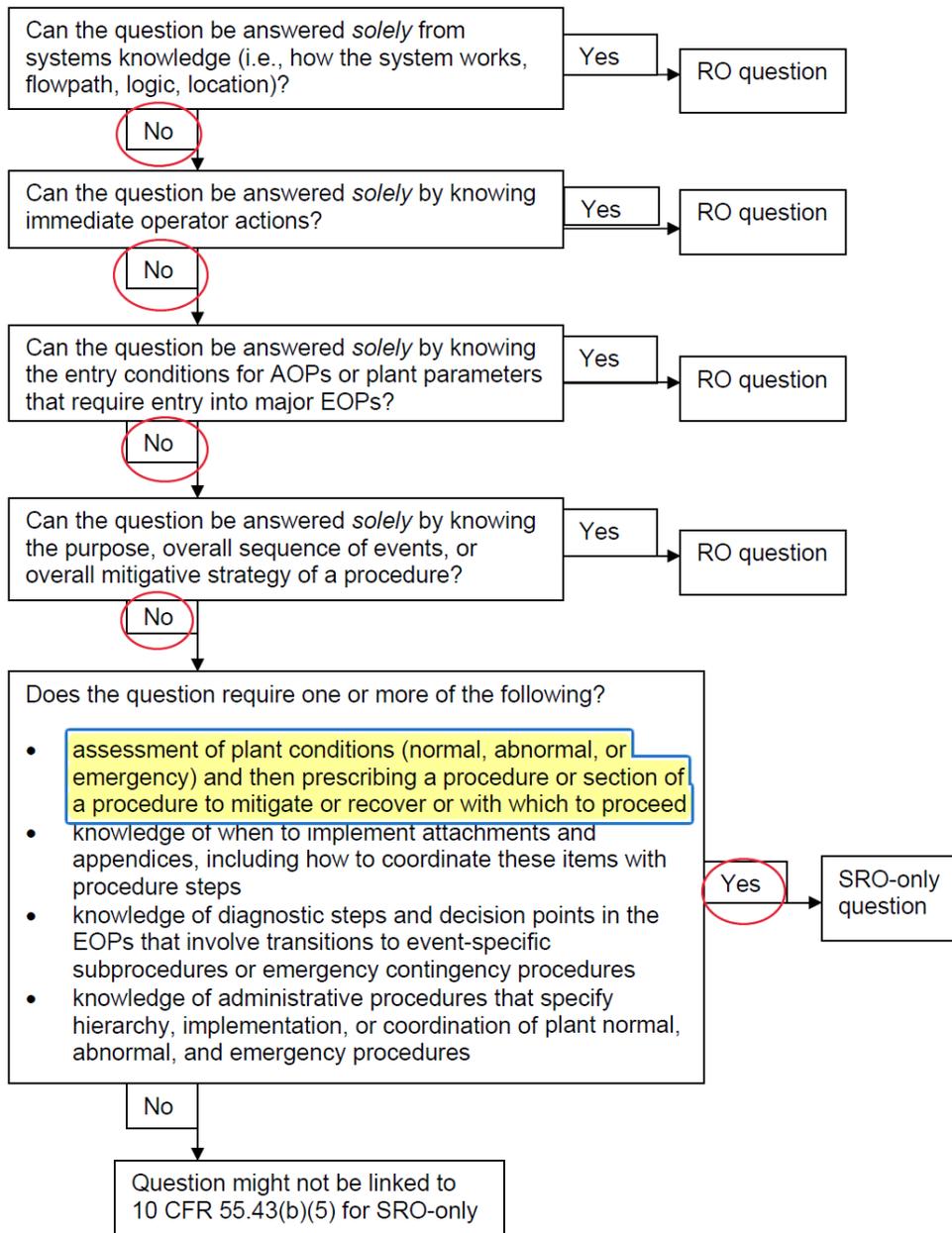
Answer: A

K/A Match:

Requires knowledge of actions required due to CIA header pressure.

SRO Only:

**Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)**



ES-4.2, Page 24 of 29

Explanation:

- A. Correct. In accordance with ABN-CIA, step 4.1.1, a reactor scram is initiated if CIA main Header pressure is LT 80 psig.

- B. Incorrect. Plausible since, in accordance with step 4.3 of ABN-CIA, If CIA header pressure approaches 135 psig, the CRS should direct supplying CIA with CAS in accordance with SOP-CIA OPS. However, CIA pressure is well below this and a reactor scram is required.
- C. Incorrect. Plausible since this is the required action if CIA pressure drops to 90 psig (ABN-CIA, step 4.4). However, CIA pressure is well below this and a reactor scram is required.
- D. Incorrect. Plausible since this is a listed action if the cause of the pressure loss is unknown (ABN-CIA, step 4.7.2) However, CIA pressure is well below this and a reactor scram is required.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-CIA, Containment Instrument Air System Failure	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 10340 - Given plant conditions associated with decreasing CIA pressure, determine if a reactor scram is required.

Question Source: Bank #: _____ Bank #
 Modified Bank #: _____ Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize knowledge of CIA system pressures where specific actions are required with the action that is required and the procedure used to accomplish the action.

10 CFR Part 55 Content: 55.43 5

Comments / Reference: ABN-CIA	Rev:	Major: 008	Minor: N/A
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Number: ABN-CIA	Use Category: CONTINUOUS	Major Rev: 008
Title: Containment Instrument Air System Failure		Minor Rev: N/A Page: 4 of 8

4.0 SUBSEQUENT OPERATOR ACTIONS

CAUTION

INBD MSIV closure will occur at approximately 50 – 80 psig.

CAUTION

If CIA Main Header pressure drops below 135 psig, SRV Solenoid C control from H13-P601 is impaired and will eventually be lost as pressure drops.

NOTE: Normal Containment Nitrogen Header pressure is 110 psig and CN-FIT-3 (RB 471) normal flow is approximately 1.1 SCFM. A sustained flowrate of GE 12 SCFM will drop Containment Nitrogen Header pressure to the alarm setpoint, and indicates a leak exists.

NOTE: If Containment Nitrogen Header pressure is normal on CIA-PI-20 (H13-P840, Bd A) and local Containment Nitrogen Header Pressure CIA-PI-29 (RB 522) is low, Drywell leakage is indicated.

NOTE: The following steps may be prioritized and performed out of order as determined by the CRS.

4.1 IF MSIV closure is imminent or occurring,
THEN **PERFORM** the following:

- 4.1.1 IF CIA Main Header pressure is LT 80 psig,
THEN **SCRAM the Reactor** per PPM 3.3.1. _____
- 4.1.2 IF any inboard MSIV has closed,
THEN **PLACE** the control switches for all MSIVs in the **CLOSED** position. _____
- 4.1.3 **INITIATE** RCIC for RPV level control. _____
- 4.1.4 **PLACE** one loop of RHR in Suppression Pool Cooling. _____

4.2 IF CIA pressure is unknown or continuing to decrease,
THEN **CHECK** CIA main header pressure at both of the following locations:

- CIA-PI-20 (H13-P840, Bd A) _____
- CIA-PI-4 (CIA-AR-1, RB 501) _____
- **PERFORM** a system walkdown to look for leaks or other contributing causes. _____

NOTE: The CIA header downstream of CIA-V-39A supplies three ADS SRVs; MS-RV-4A, 4B AND 5B. The CIA header downstream of CIA-V-39B supplies four ADS SRVs; MS-RV-3D, 4D, 4C AND 5C.

- 4.3 IF header pressure as indicated on CIA-PI-20 or CIA-PI-4 approaches 135 psig, THEN **PERFORM** the following:
- **INSTALL** ADS solenoid keys (H13-P628 or H13-P631) (keys 91-97/106-112). _____
 - **SUPPLY** CIA with CAS per SOP-CIA-OPS. _____
- 4.4 IF Containment Nitrogen Header pressure drops to 73 psig (CIA-PI-29), OR CIA-PI-20 (CIA Main Hdr. Press) (H13-P840, Bd A) drops to 90 psig, THEN **START** an immediate controlled Reactor shutdown per PPM 3.2.1. _____
- 4.5 IF header pressure is dropping rapidly, and it appears to be caused by a leak in the Nitrogen Supply System, THEN **CLOSE** the following as necessary: _____
- CN-V-112 (Containment N2 CIA Header Isolation) (DG Corridor) _____
 - CIA-V-728 (Nitrogen to CIA System Crosstie) (RB 515) _____
- 4.6 IF one of the ADS Headers has isolated, THEN **PERFORM** the following:
- 4.6.1 **ENSURE** CIA-V-39A(B) closes. _____
- 4.6.2 IF Div 1 CIA Programmer at IR-67 is NOT sequencing nitrogen bottles as required, THEN **CHECK** the following:
- Div 1- Power supply fuse (F11) at TB-IR-67-2 (RB 548) _____
 - Div 1- Power supply to IR-67 (E-DISC-PP7AE/37 and E-DISC-PP7AE/39) _____
- 4.6.3 IF Div 2 CIA Programmer at IR-68 is NOT sequencing nitrogen bottles as required, THEN **CHECK** the following:
- Div 2- Power supply fuse (F11) at TB-IR-68-2 (RB 548) _____
 - Div 2- Power supply to IR-68 (E-DISC-PP8AE/37 and E-DISC-PP8AE/39) _____

4.6.4 **ENSURE** Div 1(2) CIA Programmer, IR-67 (68), begins sequencing on backup nitrogen bottles if two of the following three conditions are satisfied:

- ADS Header A LT 160 psig. _____
- ADS Header A LT 156 psig. _____
- CIA-V-39A(B) closed. _____

4.7 IF the cause of the pressure loss is unknown,
THEN **PERFORM** the following:

4.7.1 **ENSURE** a N₂ flow path exists to the main header. _____

4.7.2 **ENSURE** proper N₂ valve alignment per SOP-CN-START or SOP-CN-OPS. _____

4.7.3 **ENSURE** proper N₂ cryogenic tank level per SOP-CN-START
or SOP-CN-OPS. _____

4.7.4 **ENSURE** proper N₂ evaporator operation per SOP-CN-START
or SOP-CN-OPS. _____

4.8 **REFER** to PPM 1.10.1 for reportability. _____

4.9 WHEN Plant conditions have been stabilized,
THEN **REFER** to Technical Specification 3.5.1 in Modes 1, 2, and 3. _____

NOTE: It may be necessary to contact I&C to use a nitrogen bottle in the local instrument rack when resetting CIA-PS-39A(B).

4.10 WHEN main header pressure is restored to normal (180 – 190 psig),
THEN **ENSURE** CIA-V-39A(B) opens. _____
AND **RESTORE** the system per SOP-CIA-OPS. _____

Question: SRO-79

Examination Outline Cross-reference: 79

Revision: 0

Date: 9/12/22

Tier: 1

Group: 2

K/A Number: 295021.AA2.03

Level of Difficulty: 3

SRO Importance Rating: 4.0

K/A Description: Ability to determine and/or interpret the following as they apply to Loss of Shutdown Cooling: Reactor water level

CGS is in Mode 4.

A loss of Shutdown Cooling (SDC) occurs.

The crew is taking actions to restore SDC.

Which of the following conditions require monitoring RPV metal temperatures every 30 minutes in accordance with OSP-RCS-C103 due to loss of forced core circulation flow?

- (1) 1 RRC pump in operation, no SDC loops in operation, RPV level 35 inches.
- (2) No RRC pumps in operation, 1 SDC loop in operation, RPV level 35 inches.
- (3) No RRC pumps in operation, no SDC loops in operation, RPV level 65 inches.

- A. (1) ONLY
- B. (2) ONLY
- C. (1) and (3) ONLY
- D. (2) and (3) ONLY

Answer: D

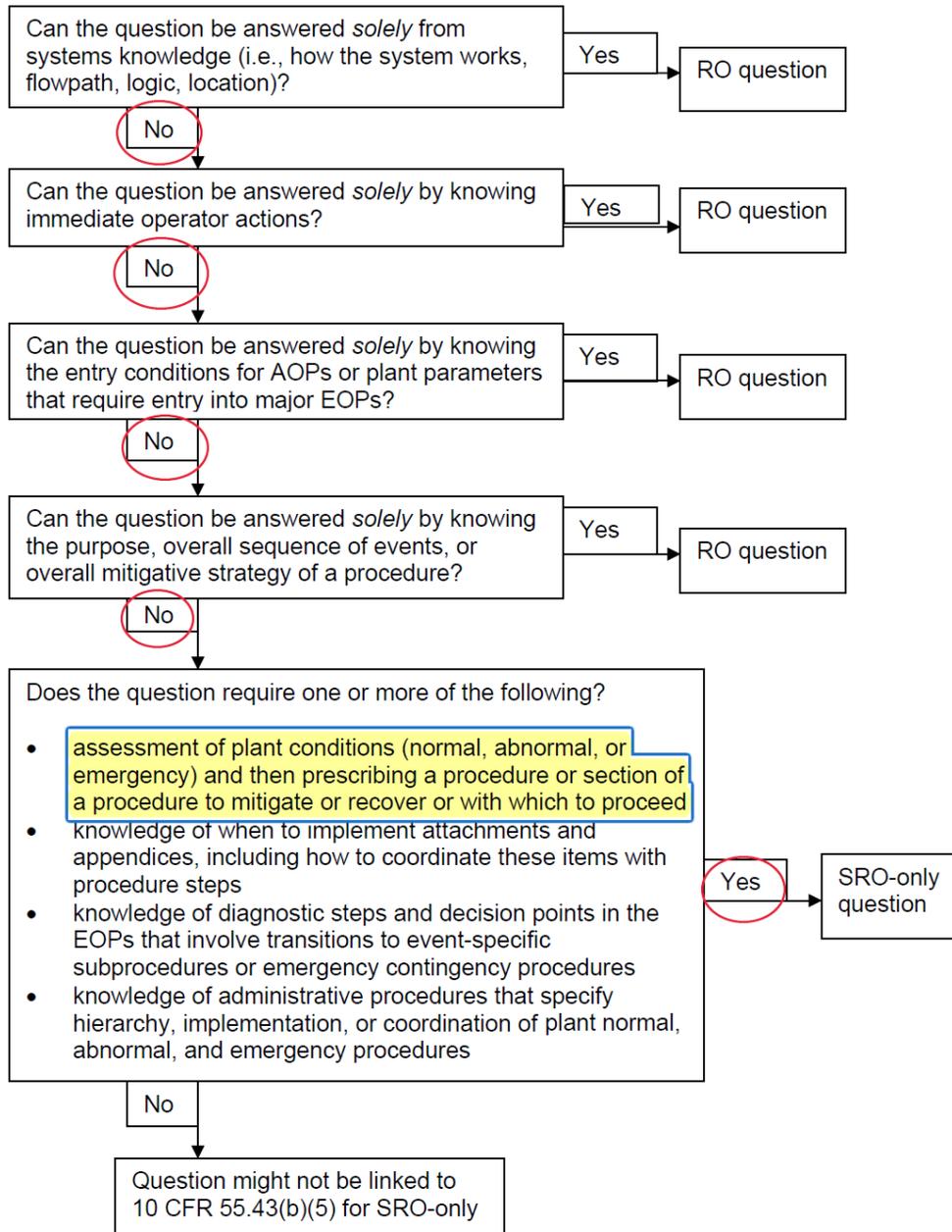
K/A Match:

Requires knowledge of required actions on a loss of SDC based on RPV level.

Question: SRO-79

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Examination Outline Cross-reference: RO-80

Revision: 0

Date: 5/16/2022

Tier: 1

Group: 1

K/A Number: 295030 EA2.03

SRO Importance Rating: 3.7

Level of Difficulty: 3

K/A Description: Ability to determine and/or interpret the following as they apply to Low Suppression Pool Water Level:
Reactor Pressure

CGS is operating at 100% power when a transient occurs.

The reactor is scrammed.

Current plant conditions:

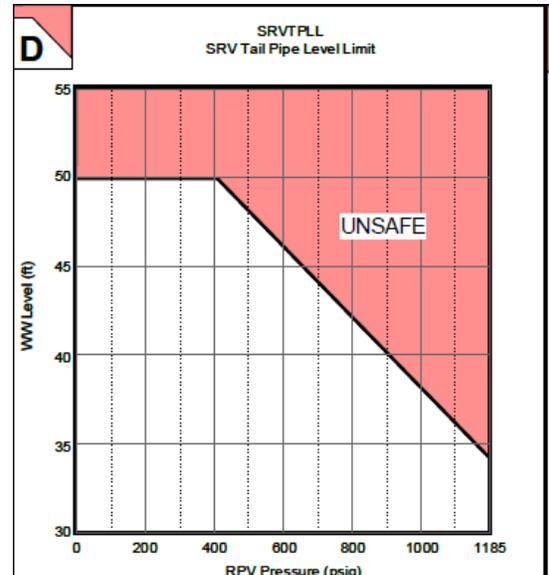
- Reactor Level: +36" and stable.
- Reactor Pressure: 900psig and stable.
- Drywell Pressure: 2.5psig and slowly rising.
- Wetwell Level: 42 feet and slowly rising.

Based on current plant conditions what action is required to be directed?

What is the bases for this action?

Direct the RO to (1).

This is done to (2).



- (1) Emergency Depressurize the reactor
(2) maintain pressure suppression capability, vent capability, and vacuum relief capability.
- (1) Emergency Depressurize the reactor
(2) ensure that opening the SRV will not exceed code allowable stresses in SRV tail pipe and supporting components.
- (1) Bypass interlocks and letdown the suppression pool to radwaste
(2) maintain pressure suppression capability, vent capability, and vacuum relief capability.
- (1) Bypass interlocks and letdown the suppression pool to radwaste
(2) ensure that opening the SRV will not exceed code allowable stresses ins SRV tail pipe and supporting components.

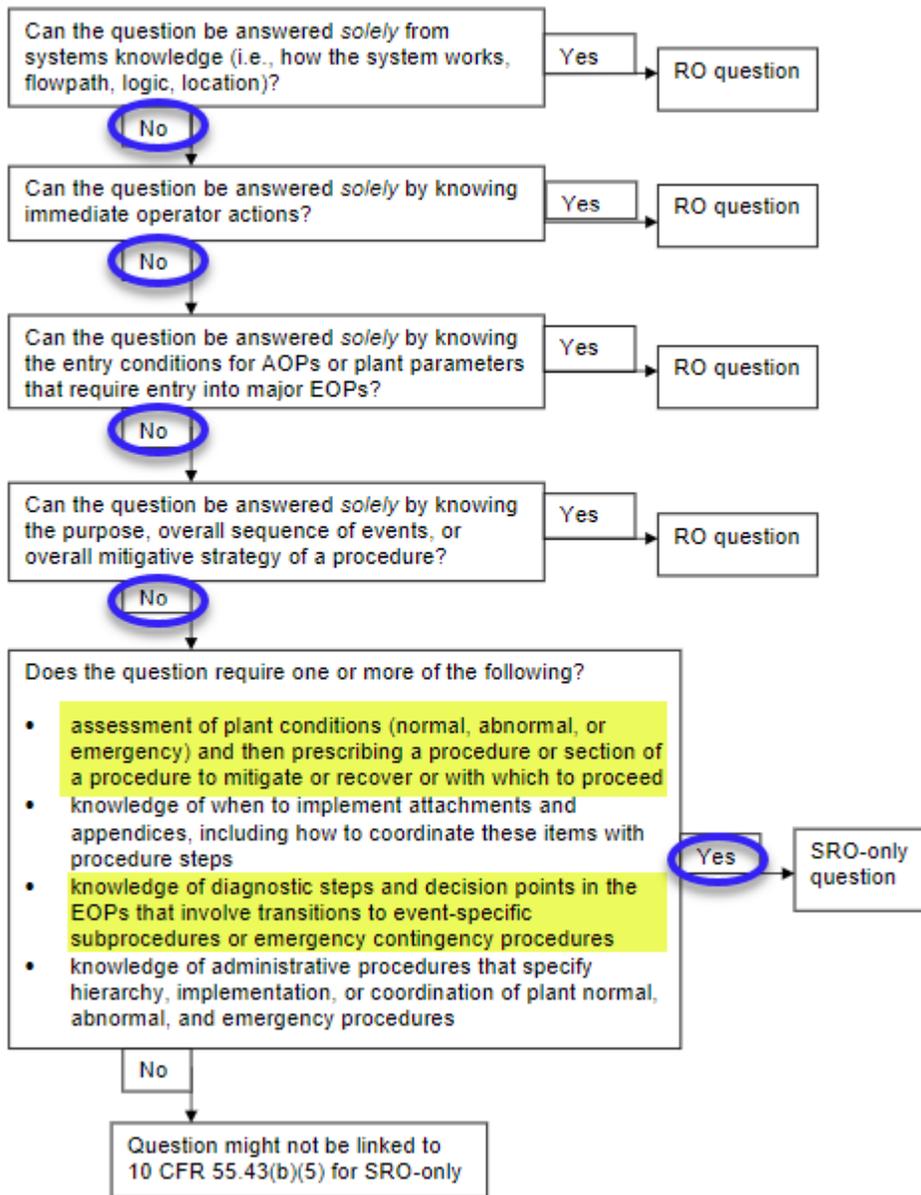
Answer: B

K/A Match:

Requires student to utilize plant conditions of reactor pressure and suppression pool level to determine the correct actions.

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Explanation:

- A. Incorrect. Plausible because part (1) is correct. Part (2) is plausible because this is the bases for letting down the wetwell to prevent wetewell level from exceeding 50 feet. Incorrect because the conditions presented in the stem have wetwell level exceeding SRVTPLL which requires ED. The bases for ED is to ensure that opening an SRV will not exceed allowable stress limits of SRV Tail Pipe Components.
- B. Correct. The stem of the question asks for the next action and provides conditions where WW level exceeds SRVTPLL which requires Emergency Depressurization per step L-12 of PPM 5.2.1. Part (2) is correct. SRVTPLL is is the highest Wetwell water level at which opening an SRV will not result in exceeding the code allowable stresses in the SRV tail pipe, tail pipe supports,

quencher, or quencher supports. This water level is a function of RPV pressure, and the Limit is utilized to preclude SRV system damage and containment failure.

- C. Incorrect. Part (1) is plausible because this is the action that is taken to prevent wetwell level from exceeding 50 feet. This is incorrect however because the stem of the question asks for the next action and provides conditions where WW level exceeds SRVTPLL which requires Emergency Depressurization. Part (2) is plausible because this is the bases for letting down the wetwell to prevent wetwell level from exceeding 50 feet. Incorrect because the conditions presented in the stem have wetwell level exceeding SRVTPLL which requires ED. The bases for ED is to ensure that opening an SRV will not exceed allowable stress limits of SRV Tail Pipe Components.
- D. Incorrect. Part (1) is plausible because this is the action that is taken to prevent wetwell level from exceeding 50 feet. This is incorrect however because the stem of the question asks for the next action and provides conditions where WW level exceeds SRVTPLL which requires Emergency Depressurization. Part (2) is correct.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.2.1	
PPM 5.0.10	
PPM 5.0.10a	
Tech Ref 4	

Proposed references to be provided during examination: Reference provided in stem (SRVTPLL graph).

Learning Objective: 8298 - Given plant conditions and the SRV Tail Pipe Level Limit Curve, determine the current operating point on the curve within 2.5 feet and 25 psig. (PPM 5.2.1)

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

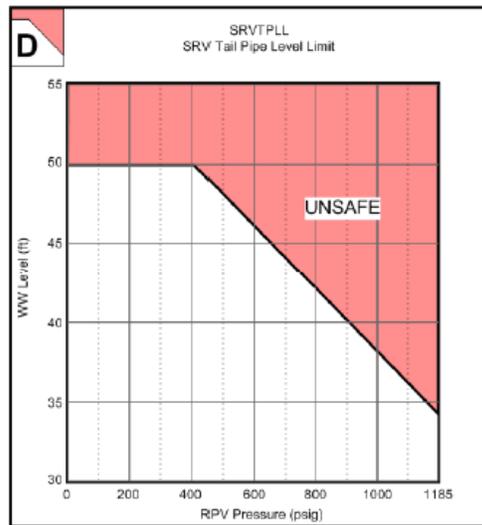
Justification for Cognitive Level

Requires analyzing graphs and utilizing this information to determine the correct course of action.

10 CFR Part 55 Content: 55.43 5

Comments / Reference: PPM 5.0.10	Rev:	Major:26	Minor:0
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SRV Tail Pipe Level Limit



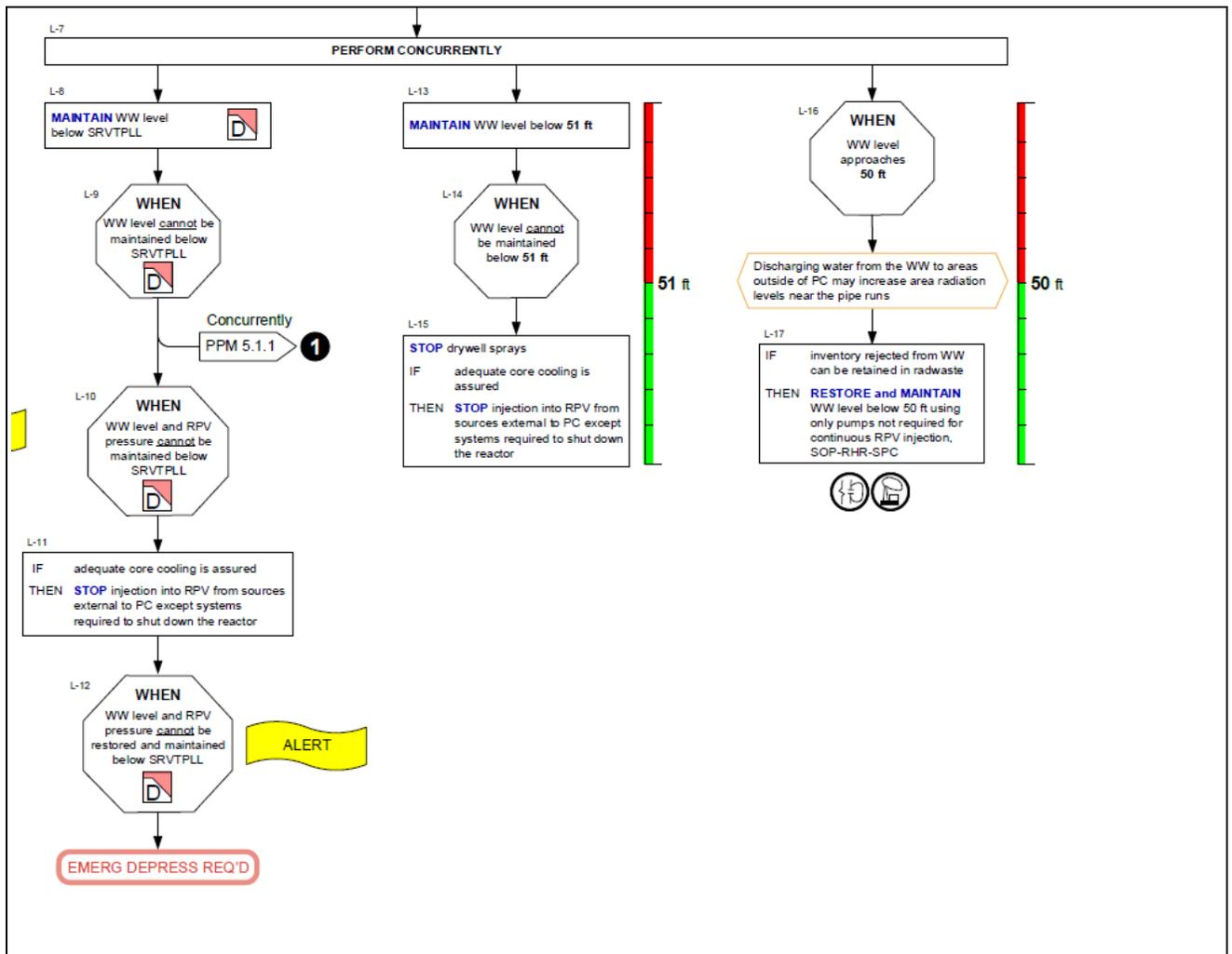
9.30.1 The SRV Tail Pipe Level Limit (SRVTPLL) is the highest Wetwell water level at which opening an SRV will not result in exceeding the code allowable stresses in the SRV tail pipe, tail pipe supports, quencher, or quencher supports. This water level is a function of RPV pressure, and the Limit is utilized to preclude SRV system damage and containment failure.

PPM 5.0.10a Major Rev. 1 – MR N/A

q. Steps L-16 and L-17:

- 1) Wetwell level is controlled to preserve pressure suppression capability, suppression chamber vent capability, and vacuum relief capability. The Maximum Pressure Suppression Primary Containment Water Level (MPSPCWL, 50 ft) is lower than the elevation of the Wetwell vent and Wetwell-to-drywell vacuum breaker penetrations, thus avoiding submergence of these components.

PPM 5.2.1 Major – 29, Minor – N/A



Examination Outline Cross-reference: RO-81

Revision: 0

Date: 5/16/2022

Tier: 1

Group: 1

K/A Number: 295031 EA2.01

SRO Importance Rating: 4.6

Level of Difficulty: 3

K/A Description: Ability to determine and/or interpret the following as they apply to Reactor Low Water Level:
Reactor water level

CGS is operating at 100% power.

A leak occurs in the drywell resulting in a reactor scram and lowering RPV level.

Following the reactor scram a loss of TR-S occurs.

Current plant conditions:

- Reactor is shutdown.
- RPV level: -100" and lowering at 10 inches/min.
- Reactor pressure: 850 psig and slowly lowering.
- Drywell pressure: 15 psig and slowly rising.

In accordance with PPM 5.1.1 which of the following actions must be performed?

- A. Reduce Reactor pressure to 500-600psig.
- B. Ensure HPCS is injecting at GT 6000GPM.
- C. Anticipate emergency depressurization and fully open all bypass valves.
- D. Emergency depressurize the reactor prior to RPV level lowering below -186".

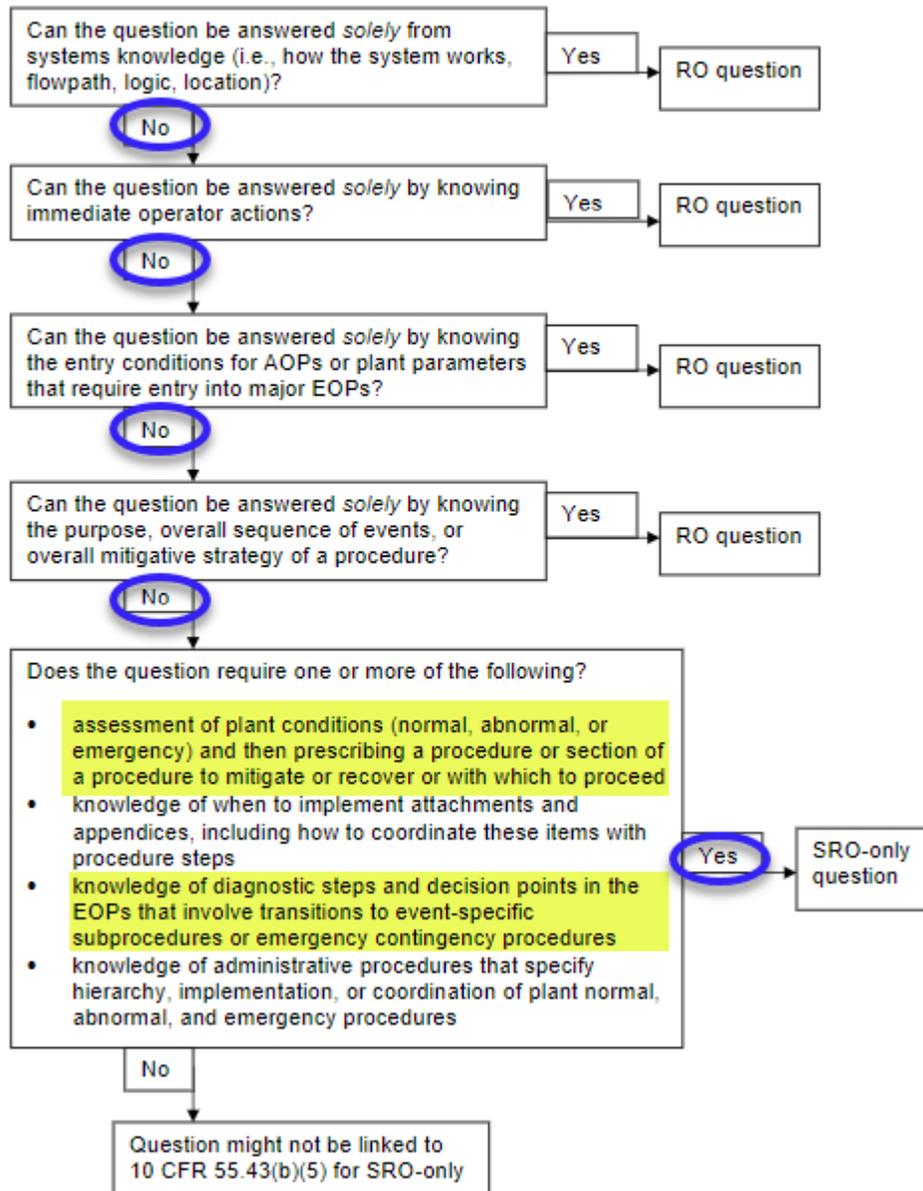
Answer: D

K/A Match:

Requires student to interpret RPV level trend and determine appropriate actions to be taken per EOPs.

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



Explanation:

- Incorrect. Plausible because reducing pressure to 500-600psig is a strategy utilized to minimize the impact of a leak inside containment. Incorrect because this strategy is only used if RPV level is under control and the stem of the questions states that level is lowering.
- Incorrect. Plausible because the conditions provided in the stem of the question would have HPCS injecting and if HPCS were injecting at GT 6000gpm at this RPV level adequate core cooling via spray cooling would be assured. Incorrect because spray cooling is no directed in PPM 5.1.1 until following emergency depressurization.
- Incorrect. Plausible because the plant is in a condition where emergency depressurization will be required. Incorrect because the anticipate ED step of PPM 5.1.1 pressure leg is not allowed to be

used when level challenged. The stem of the question places the plant in a level challenged condition.

- D. Correct. With RPV level at -100" and lowering, a loss of TR-S, and Containment pressure GT 1.68psig all available high capacity / high pressure injection sources are operating. With this information the student will determine that level cannot be restored and maintained above -161" and an emergency depressurization will be required prior to RPV level lowering below -186".

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.1.1	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 8032 - State the purpose of the Level Control Steps in PPM 5.1.1, "RPV Control." (PPM 5.1.1)

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

Question History: Last NRC Exam: N/A

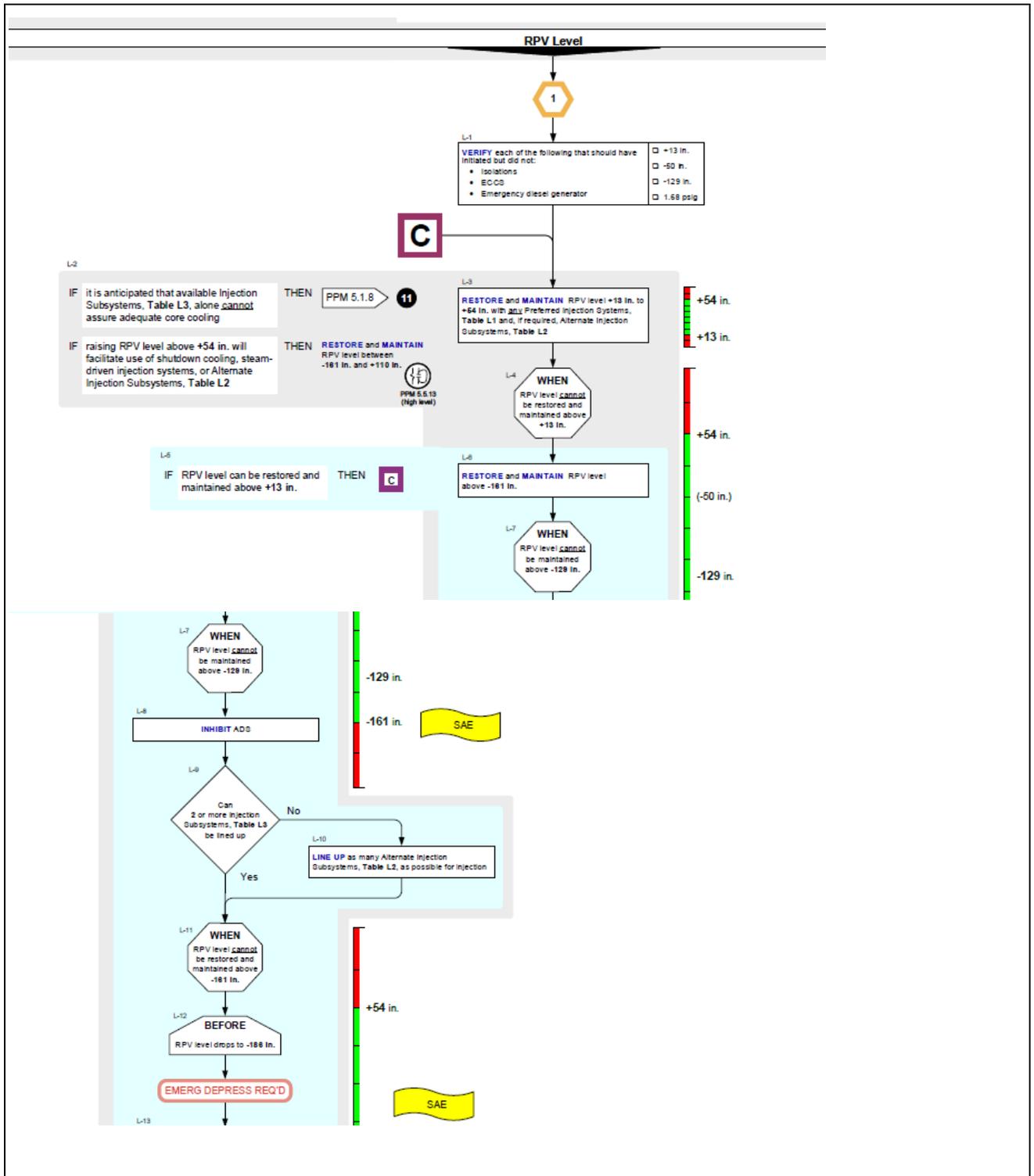
Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

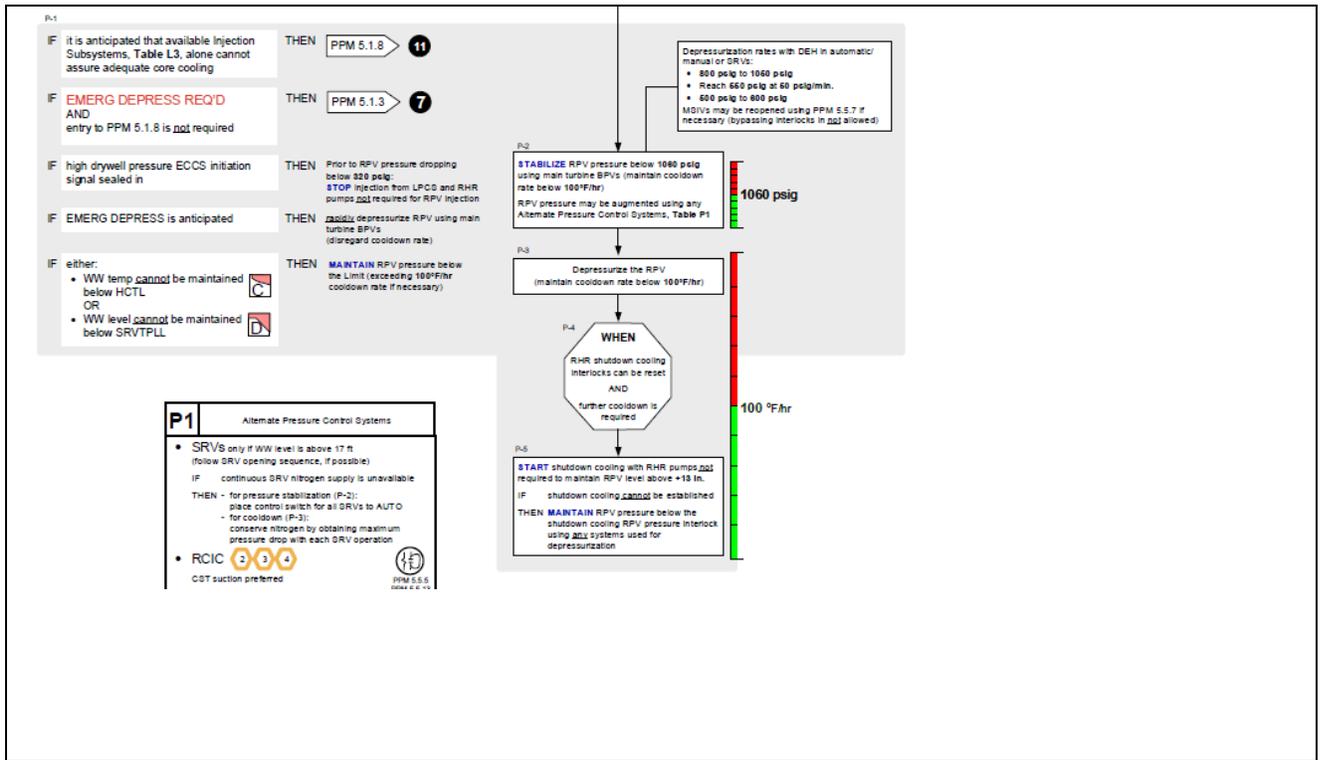
Justification for Cognitive Level

Requires student to evaluate RPV level and trend and determine the appropriate actions per EOPs.

10 CFR Part 55 Content: 55.43 5

Comments / Reference: PPM 5.1.1	Rev:	Major: 23	Minor: 0
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Question: SRO- 82

Examination Outline Cross-reference: 82

Revision: 0 Date: 10/18/22

Tier: 1

Group: 1

K/A Number: 700000.2.2.36

Level of Difficulty: 3

SRO Importance Rating: 4.4

K/A Description: Generator Voltage and Electric Grid Disturbances - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation

CGS is in Mode 1.

TR-S is inoperable for corrective maintenance.

BPA reports a grid disturbance that affects TR-B.

The following timeline is observed:

Date	Time	Occurrence
1 st	1000	TR-S declared INOPERABLE
3 rd	1000	TR-B declared INOPERABLE due to grid issues.
3 rd	2200	TR-S restored to OPERABLE

Using the reference provided, what is the latest date that TR-B must be restored to OPERABLE to meet Technical Specification LCO conditions?

TR-B must be declared OPERABLE by 1000 on the...

- A. 4th
- B. 5th
- C. 6th
- D. 7th

Answer: B

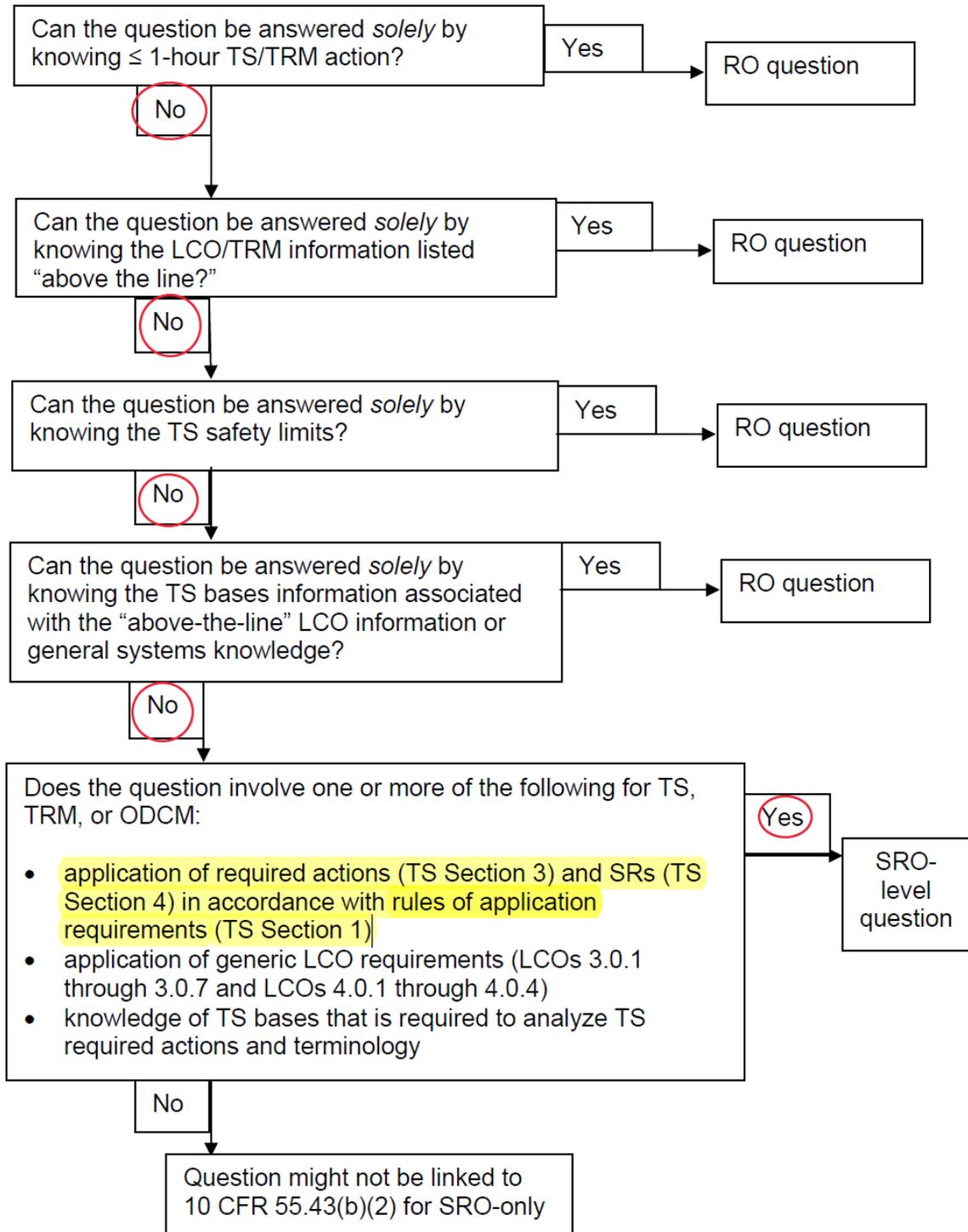
K/A Match:

Requires knowledge of the effects of maintenance on a TS during a grid disturbance.

Question: SRO- 82

SRO Only:

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



Question: SRO- 82

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize an understanding of the application of TS 3.8.1 along with a knowledge of action statement completion time allowances.

10 CFR Part 55 Content: 55.43 2

Comments / Reference: TS LCO 3.8.1 Required Actions		Amendment:	267
<p>ACTIONS</p> <p>-----NOTE-----</p> <p>LCO 3.0.4.b is not applicable to DGs.</p> <p>-----</p>			
CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for OPERABLE offsite circuit.	1 hour <u>AND</u> Once per 8 hours thereafter	
	<u>AND</u>		
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 division concurrent with inoperability of redundant required feature(s)	
	<u>AND</u> A.3 Restore offsite circuit to OPERABLE status.	72 hours	

Question: SRO- 82

<p>C. Two offsite circuits inoperable.</p>	<p>C.1 Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.</p> <p><u>AND</u></p> <p>C.2 Restore one offsite circuit to OPERABLE status.</p>	<p>12 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)</p> <p>24 hours</p>
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Question: SRO- 82

Comments / Reference: TS Section 1.3	Amendment:	267
<p>Once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition, unless otherwise specified.</p> <p>However, when a <u>subsequent</u> division, subsystem, component, or variable expressed in the Condition is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:</p> <ol style="list-style-type: none"> a. Must exist concurrent with the <u>first</u> inoperability; and b. Must remain inoperable or not within limits after the first inoperability is resolved. <p>The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:</p> <ol style="list-style-type: none"> a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or b. The stated Completion Time as measured from discovery of the subsequent inoperability. <p>The above Completion Time extension does not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each division, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.</p> <p>The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery. . ."</p>		

Examination Outline Cross-reference: 83

Revision: 0

Date: 9/19/22

Tier: 1

Group: 2

K/A Number: 295008.AA2.04

SRO Importance Rating: 3.6

Level of Difficulty: 4

K/A Description: Ability to determine and/or interpret the following as they apply to Loss of Main Condenser Vacuum:
Offgas system flow

CGS is in Mode 1.

Main Condenser backpressure is slowly rising.

Condenser Backpressure is 3.5 inHg, up slow.

The crew is taking actions to identify the cause of rising backpressure.

Offgas flow is higher than normal.

Which of the following should the CRS direct to be checked for the conditions given?

- (1) Excessive OG Air Purge Flow.
- (2) Step change in outlet temperatures indicating offgas system flooding.
- (3) Turbine Gland Seal Pressure low.

- A. (1) ONLY
- B. (2) ONLY
- C. (1) and (3) ONLY
- D. (2) and (3) ONLY

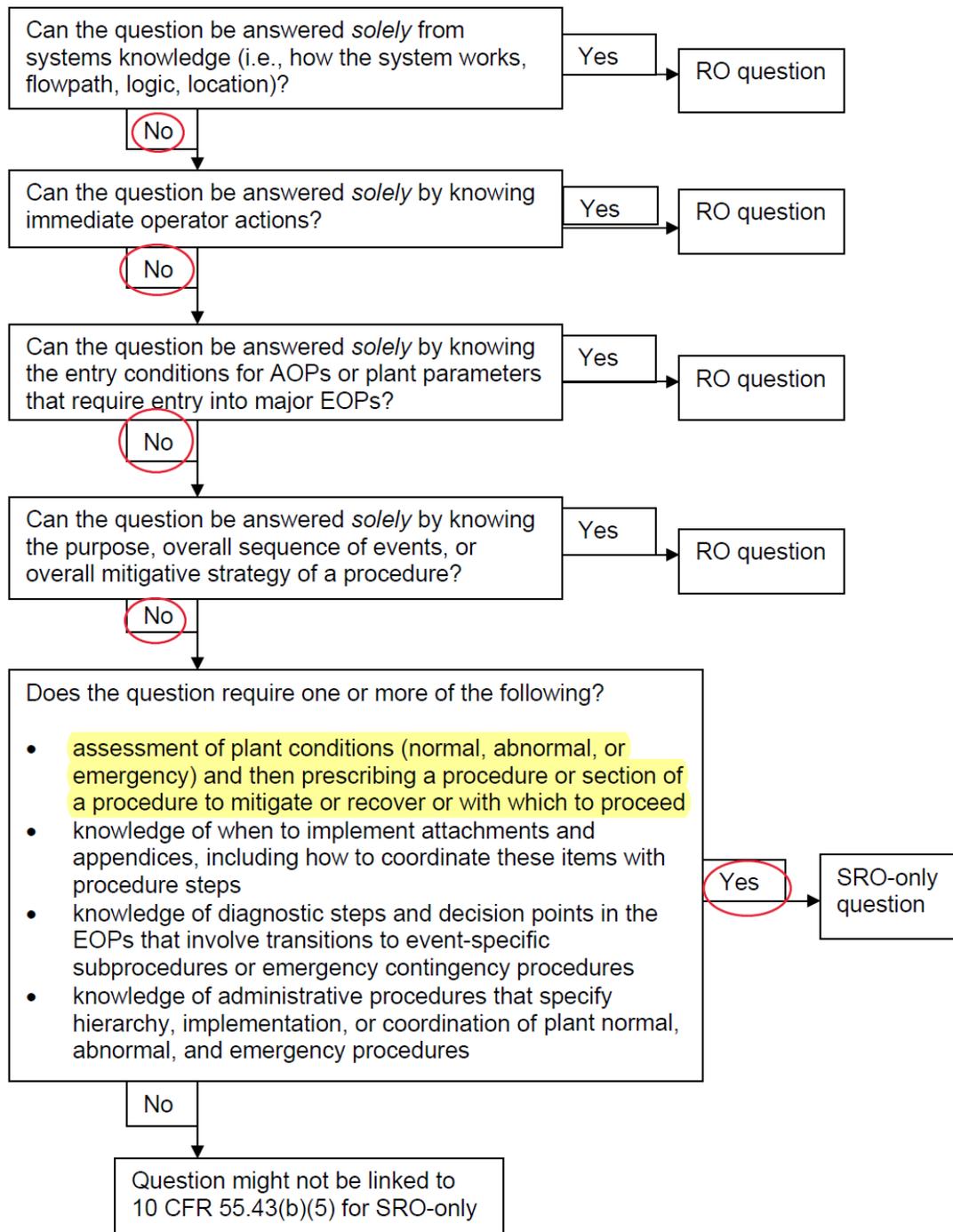
Answer: C

K/A Match:

Requires knowledge of actions to be taken on a loss of main condenser vacuum for abnormal Offgas system flow.

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Explanation:

A. Incorrect. Plausible since (1) is correct. However, (3) is also correct.

- B. Incorrect. Plausible since (2) is a condition that will cause abnormal Offgas system flow. However, this condition will cause lower than normal system flow.
- C. Correct. In accordance with ABN-Backpressure, step 4.21, if Offgas flow is higher than normal, several conditions are required to be checked to investigate the cause of high flow, including Excessive OG Air Purge Flow and Turbine Gland Seal Pressure low.
- D. Incorrect. Plausible since (3) is correct. Plausibility is enhanced since (2) will cause abnormal Offgas system flow. However, (2) will cause lower than normal flow.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-BACKPRESSURE	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 11650 - Predict the impact of the following on the Main Turbine System.
Loss of condenser vacuum

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Candidate must demonstrate knowledge of the procedure for mitigating lowering condenser vacuum due to changes in Offgas system flow.

10 CFR Part 55 Content: 55.43 5

Comments / Reference: ABN-BACKPRESSURE	Rev:	Major:009	Minor:N/A
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Number: ABN-BACKPRESSURE	Use Category: CONTINUOUS	Major Rev: 009
Title: Unexpected Rise in Main Condenser Backpressure		Minor Rev: N/A Page: 10 of 15

4.21 **IF** Offgas flow is **higher** than normal,
THEN CHECK the following items for excessive Main Condenser in-leakage:

- HP Turbine spillover pressure is properly regulated (1 – 3 psig). _____
- Turbine Steam Sealing system operation: Individual gland pressures may be checked on TDAS points T056 through T063, or P7 display, HP bearings (17 – 22 psid), LP bearings (1.8 – 3.1 psig). _____
- **Inadequate Main Condensate pump sealing** (no noticable seal water leakage out of the packing gland and/or seal water pressure **LT 30 psig** at COND-PI-3AG (BG, CG), IR-15). _____
- Leaking low pressure Turbine exhaust relief diaphragms or Vacuum Breaker valves. _____
- Failure of flexible components between the Main Condenser and other components. _____

NOTE: Excessive Purge Flow can cause a stall condition at the SJAEs resulting in a loss of Main Condenser vacuum.

- **Excessive OG Air Purge Flow** _____

4.22 **IF** Offgas system flow is **lower** than normal,
THEN CHECK for Offgas system blockage, including the following:

- Loop seal drain valves (OG-V-158, 159, 160) closed with seal not in alarm (H13-P672.A10-1.8, 1.9, 1.10). _____
- Water intrusion as indicated by high water separator outlet temperature on OG-TRS-610. _____
- **Step change in outlet temperatures indicating offgas system flooding** (outlet temperatures are normally within 5°F of each other). _____
- Proper Dryer lineup. _____

Examination Outline Cross-reference: 84

Revision: 0 Date: 10/18/22

Tier: 1

Group: 2

K/A Number: 295008.AA2.01

SRO Importance Rating: 4.4

Level of Difficulty: 3

K/A Description: Ability to determine and/or interpret the following as they apply to High Reactor Water Level: Reactor water level

CGS is in Mode 2.

A reactor startup is in progress.

An event causes RPV level to rise.

Current plant conditions:

- Reactor power is 10% and stable.
- RPV level is +40 inches, up slow.

If the current trends continue, when should the CRS direct a reactor scram and what is the basis for this?

The CRS should direct a reactor scram when an RPV level of GT (1) is imminent. This is done to minimize (2).

- A. (1) +54.5 inches
(2) the potential for a subsequent low RPV water level condition
- B. (1) +54.5 inches
(2) energy addition to the RPV which causes RPV water expansion
- C. (1) +108 inches
(2) the potential for a subsequent low RPV water level condition
- D. (1) +108 inches
(2) energy addition to the RPV which causes RPV water expansion

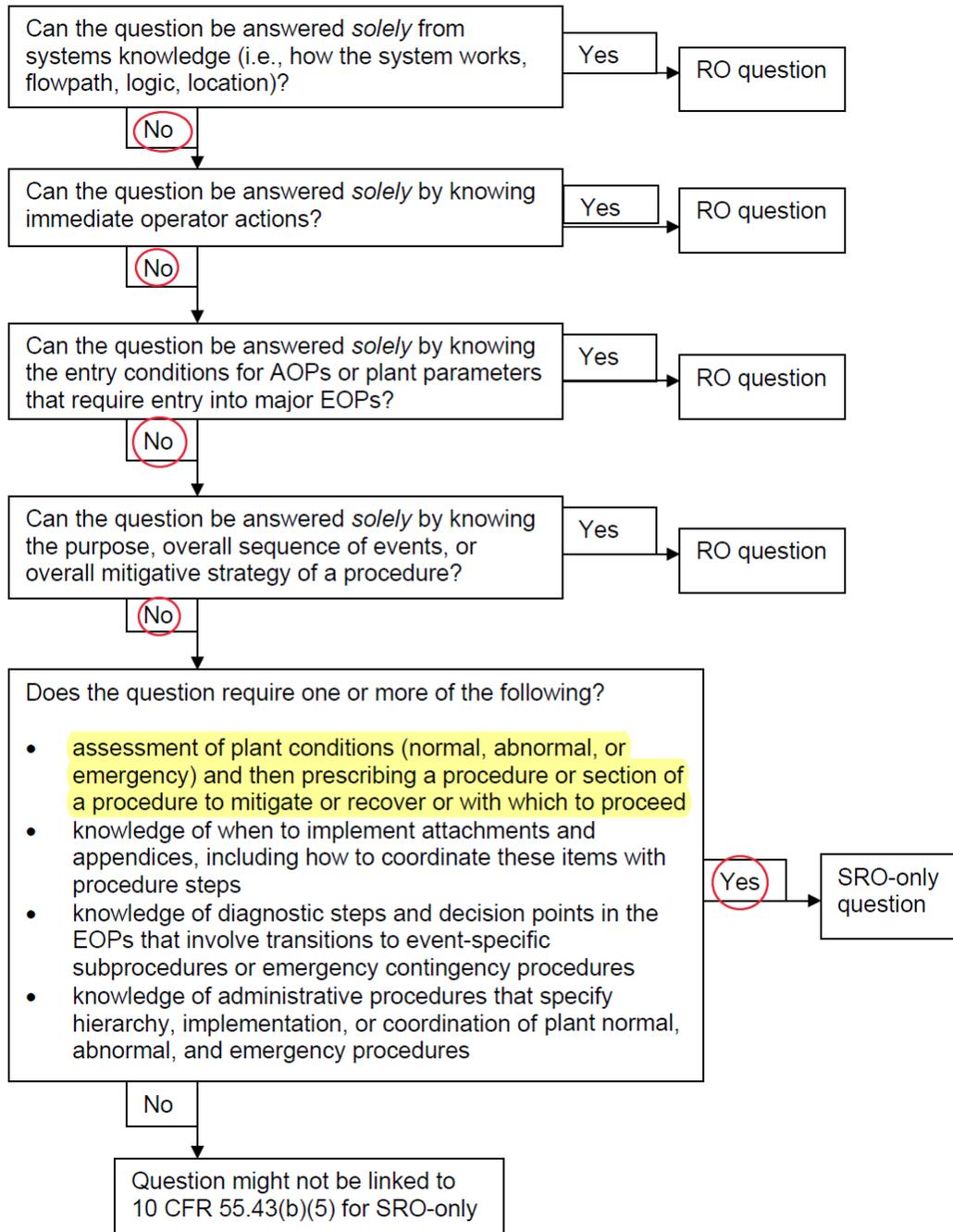
Answer: A

K/A Match:

Requires candidate to interpret RPV water level and determine when a reactor scram is required during a high reactor water level condition.

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Explanation:

- A. Correct. In accordance with ABN-LEVEL, step 4.2.2, if RPV level GT +54.5 inches is imminent, a reactor scram is initiated. According to the basis of this step, a scram is initiated since both feed pumps trip at this level, which will cause a low RPV level condition.

- B. Incorrect. Plausible since (1) is correct. Plausibility is enhanced since (2) is the basis for scrambling the reactor when power is LT 5%. However, this is not the basis for the conditions given.
- C. Incorrect. Plausible since (2) is correct. Plausibility is enhanced since (1) is the correct level for initiating a reactor scram with power LT 5%. However, for the conditions given, a reactor scram is initiated prior to +54.5 inches.
- D. Incorrect. Plausible since (1) is the level where a reactor scram is required if reactor power is LT 5%. Plausibility is enhanced since (2) is the basis for the scram with power LT 5%. However, power is GT 5%. Therefore these are not correct.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-LEVEL, Unplanned RPV Water Level Change	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 10310 - Given plant conditions associated with ABN-LEVEL, determine if a reactor scram is required.

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize the required actions of ABN-LEVEL with the basis of those actions

10 CFR Part 55 Content: 55.43 5

Comments / Reference: ABN-LEVEL		Rev:	Major:	Maj	Minor:	Min
4.2	<u>HIGH RPV LEVEL</u>					
4.2.1	<u>IF</u> the reactor has scrammed, <u>THEN</u> PERFORM PPM 3.3.1.					_____
4.2.2	<u>IF</u> RPV level GT +54.5" is imminent, <u>AND</u> reactor power is GE 5%, <u>THEN</u> SCRAM the Reactor per PPM 3.3.1.					_____
4.2.3	<u>IF</u> RPV level GT +108" is imminent, <u>THEN</u> PERFORM the following:					
	a. <u>IF</u> the Reactor Mode Switch is in STARTUP or RUN , <u>THEN</u> SCRAM the Reactor per PPM 3.3.1.					_____
	b. CLOSE the inboard MSIVs.					_____
	c. CLOSE RCIC-V-63 and RCIC-V-76.				{P-26365}	_____
4.2.13	<u>BEFORE</u> RPV level reaches +108", <u>THEN</u> PERFORM the following: N/A if already performed.					
	a. <u>IF</u> the Reactor Mode Switch is in STARTUP or RUN , <u>THEN</u> SCRAM the Reactor per PPM 3.3.1.					_____
	b. CLOSE the inboard MSIVs.					_____
	c. CLOSE RCIC-V-63 and RCIC-V-76.				{P-26365}	_____
4.2.13	If the reactor is at GT 5% power, then a reactor scram was inserted at +54.5". If the reactor is at low power and reactor water level is GT +80", then the reactor should be scrammed to stop energy addition that causes expansion of the water within the RPV, and in anticipation of closing the MSIVs.					

Question: SRO-85

Examination Outline Cross-reference: RO-85

Revision: 0

Date: 10/5/22

Tier: 1

Group: 1

K/A Number: 295036 2.4.41

Level of Difficulty: 2

SRO Importance Rating: 4.6

K/A Description: Secondary Containment High Sump/Area Water Level - Knowledge of the emergency action level thresholds and classifications

CGS is operating at 100% power with HPCS OOS for maintenance.

Using the reference provided, with no operator action taken the following sequence of alarms occurs:

- 10:00 – R1 Sump Hi Level Alarm and Hi-Hi level Levels Alarm.
- 10:03 – Ops 2 reports a leak from the suction line of RHR-P-2A.
- 10:05 – RHR-V-4A Control Power fuses clear, resulting in RHR A OOS alarms with an MOV Network Power Loss BISI.

(1) What level of emergency classification is required?

(2) When is the latest time this classification can be made?

- A. (1) Unusual Event
(2) 1018
- B. (1) Unusual Event
(2) 1020
- C. (1) Alert
(2) 1018
- D. (1) Alert
(2) 1020

Answer: B

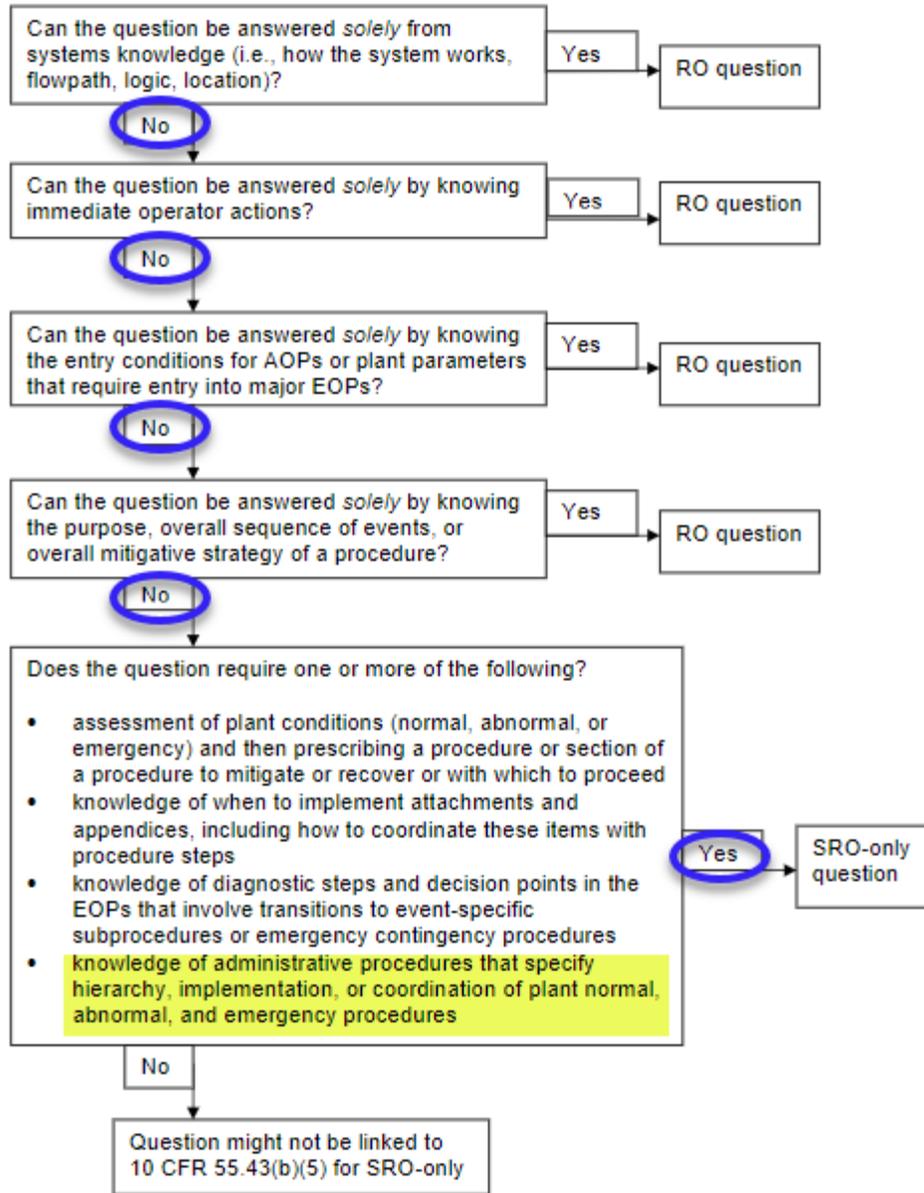
K/A Match:

Requires student to interpret the effects of room flooding on EAL declaration.

SRO Only:

Question: SRO-85

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Explanation:

- A. Incorrect. Plausible because (1) is correct. See explanation for B. (2) is plausible if the student believes the declaration is required 15 minutes after confirmation of the flooding condition. This is incorrect because the declaration is not required until 15 minutes after the condition is met, which is when the electrical isolation occurs.
- B. Correct. (1) The event classification is a UE because flooding occurs in the 'A' RHR room that caused an automatic electrical isolation of RHR-V-4A which is a safety component needed for the current operating mode.
(2) The declaration is required within 15 minutes of existence of a condition. The condition exists

Question: SRO-85

once the safety isolation occurs as a result of flooding, so declaration is required 15 minutes after RHR-V-4A loses power due to the flooding.

C. (1) incorrect. Plausible if the student believes that MA8.1 is required if there is a flooding condition and two trains of safety system out of service. This is incorrect because HPCS was OOS for maintenance, not due to the flooding condition, so only the UE would be required. (2) is plausible if the student believes the declaration is required 15 minutes after confirmation of the flooding condition. This is incorrect because the declaration is not required until 15 minutes after the condition is met, which is when the electrical isolation occurs.

D. (1) incorrect. Plausible if the student believes that MA8.1 is required if there is a flooding condition and two trains of safety system out of service. This is incorrect because HPCS was OOS for maintenance, not due to the flooding condition, so only the UE would be required. (2) is correct. See explanation for B.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
13.1.1	
13.1.1A	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: 13.1.1 Hard Cards

Learning Objective: 6131 - With the procedures available for reference and plant conditions such that an emergency classification should be declared, correctly classify the event. (SRO only) [PPM 13.1.1]

Question Source: Bank #: _____ Bank #
 Modified Bank #: _____ Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires student to evaluate plant conditions and determine the appropriate EAL.

10 CFR Part 55 Content: 55.43 5

Comments / Reference: 13.1.1	Rev:	Major: 49	Minor: 2
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Question: SRO-85

Hazardous event						
HU3.1	<input type="checkbox"/>					
(1) A tornado strike within the PROTECTED AREA OR (2) Volcanic ash fallout requiring plant shutdown						
HU3.2	<input type="checkbox"/>					
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode						
HU3.3	<input type="checkbox"/>					
(1) Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill, 618-11 event or toxic gas release) OR (2) A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)						

Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode						
MA8.1	<input type="checkbox"/>					
The occurrence of <u>any</u> Table 8 hazardous event AND Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode AND EITHER: Event damage has caused indications of degraded performance to a second train of a SAFETY SYSTEM needed for the current operating mode OR Event damage has resulted in VISIBLE DAMAGE to a second train of a SAFETY SYSTEM needed for the current operating mode (Notes 9, 10)						

Table 8 Hazardous Events
<ul style="list-style-type: none"> ● Seismic event ● Internal or external FLOODING event ● High winds ● Tornado strike ● FIRE ● EXPLOSION ● Volcanic ash fallout ● Other events with similar hazard characteristics as determined by the Shift Manager

Question: SRO-85

13.1.1A Major Rev 34 Minor Rev 2

EAL:

HU3.2 Unusual Event

Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode

Mode Applicability:

1	2	3	4	5	def
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Basis:

An uncontrolled flooding event may pose a direct threat to safety-related equipment. As such, the potential exists for substantial degradation of the level of safety of the plant. One indication of FLOODING is indicated by ECCS room level alarms on P601 (ref. 1, 2).

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses FLOODING of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, M or C.

EAL:

MA8.1 Alert

The occurrence of any Table 8 hazardous event

AND

Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode

AND EITHER:

Event damage has caused indications of degraded performance to a second train of a SAFETY SYSTEM needed for the current operating mode

OR

Event damage has resulted in **VISIBLE DAMAGE** to a second train of a SAFETY SYSTEM needed for the current operating mode

(Notes 9, 10)

Note 9: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then emergency classification is **not** warranted.

Note 10: If the hazardous event **only** resulted in **VISIBLE DAMAGE**, with **no** indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is **not** warranted

Examination Outline Cross-reference: 86

Revision: 0

Date: 9/14/22

Tier: 2

Group: 1

K/A Number: 215004.A2.03

Level of Difficulty: 3

SRO Importance Rating: 3.2

K/A Description: Ability to (a) predict the impacts of the following on the Source Range Monitor System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Stuck detector

CGS is in Mode 2.

A reactor startup is in progress in accordance with PPM 3.1.2, Reactor Plant Startup.

- SRM-A is BYPASSED.

The reactor is sub-critical. The RO is withdrawing control rods to achieve criticality.

SRMs are being withdrawn from the core to maintain on-scale indications.

SRM-D sticks and cannot be moved.

What actions should be taken?

- A. Control rod withdrawal should be suspended until at least 3 SRMs are operable.
- B. All control rods should be inserted to the Subcritical Hold Point using the shutdown pull sheet.
- C. Control rods should be withdrawn to raise reactor power until all IRM channels are Range 3 or higher.
- D. All control rods should be fully inserted by placing the Reactor Mode Switch in SHUTDOWN.

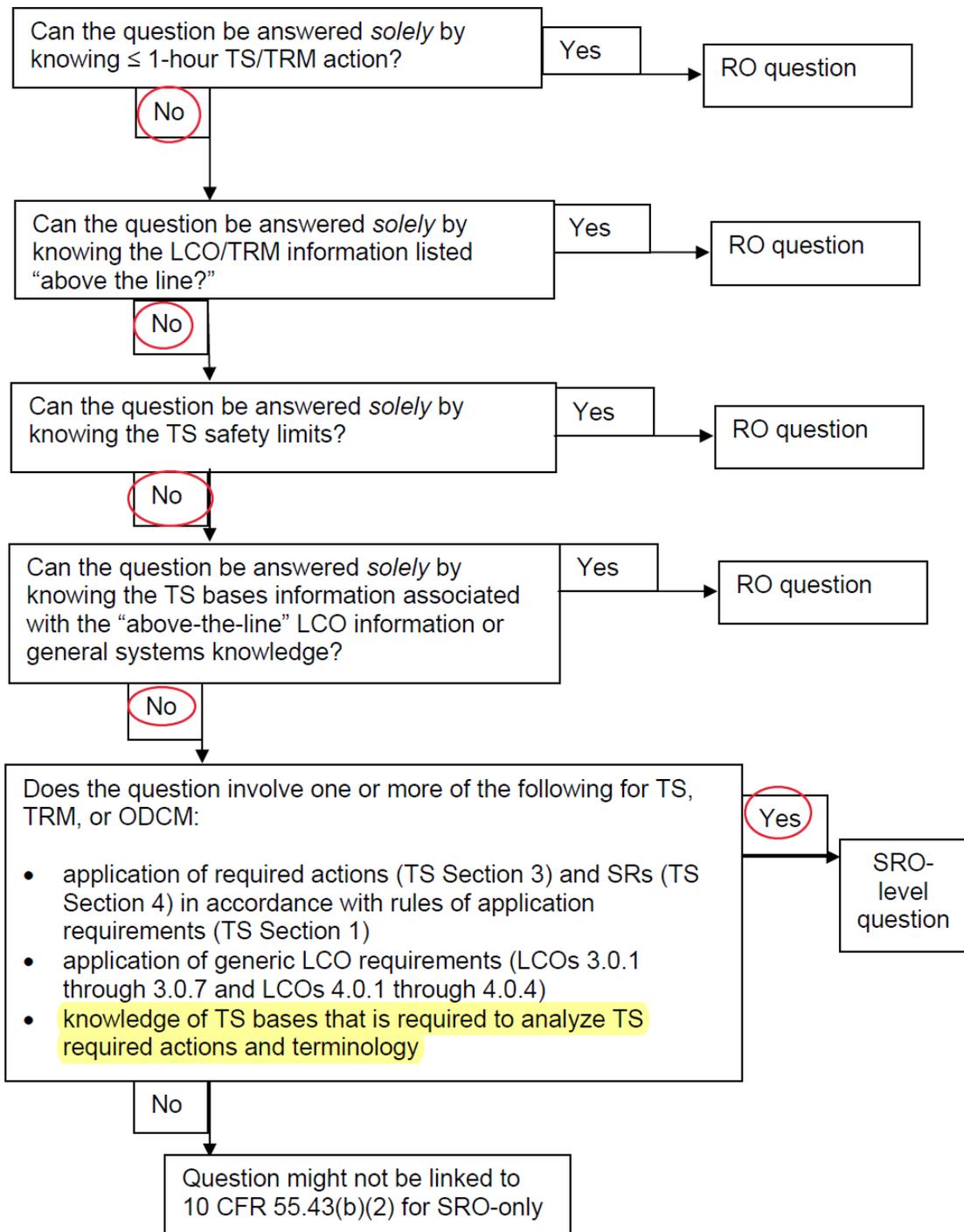
Answer: A

K/A Match:

Requires knowledge of actions to take when a SRM detector sticks.

SRO Only:

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



Explanation:

- A. Incorrect. Plausible since PPM 3.1.2, Precaution and Limitation 4.10 states that rod withdrawal should be suspended with less than 3 SRMs operable with the reactor subcritical. However, the initial conditions in the question stem states that the reactor is critical.

- B. Incorrect. Plausible since this is the action if the reactor returns to sustained subcriticality after it is critical with reactor coolant temperature LE 350F.
- C. Correct. In accordance with Technical Specification Bases for LCO 3.3.1.2, with at least one SRM operable, the crew should raise reactor power to establish all IRMs on Range 3 or greater. This will exit the applicability of the LCO.
- D. Incorrect. Plausible since this is the action if the reactor returns to sustained subcriticality after it is critical with reactor coolant temperature GT 350F.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
TS LCO 3.3.1.2, SRM Instrumentation Bases	
PPM 3.1.2, Reactor Plant Startup	
PPM 9.3.12, Plant Power Maneuvering	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: SRM-16 Given Source Range Monitor System operability status and/or a copy of Tech Specs, discuss the bases for the Source Range Monitoring System Tech Spec LCO, related safety limits and Limiting Safety System Settings.

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize a knowledge of Technical Specification LCO requirements with a knowledge of Technical Specification Bases.

10 CFR Part 55 Content: 55.43 2

Comments / Reference: LCO 3.3.1.2 Bases	Rev:	129			
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BASES

LCO (continued)

changes during fuel loading, since they can be positioned anywhere within the core during refueling. They must still meet the location requirements of SR 3.3.1.2.2, and all other required SRs for SRMs.

For an SRM channel to be considered OPERABLE, it must be providing neutron flux monitoring indication.

APPLICABILITY

The SRMs are required to be OPERABLE in MODE 2 prior to the IRMs being on scale on Range 3 and MODES 3, 4, and 5, to provide for neutron monitoring. In MODE 1, the APRMs provide adequate monitoring of reactivity changes in the core; therefore, the SRMs are not required. In MODE 2, with IRMs on Range 3 or above, the IRMs provide adequate monitoring and the SRMs are not required.

ACTIONS

A.1 and B.1

In MODE 2, with the IRMs on Range 2 or below, SRMs provide the means of monitoring core reactivity and criticality. With any number of the required SRMs inoperable, the ability to monitor is degraded. Therefore, a limited time is allowed to restore the inoperable channels to OPERABLE status.

Providing that at least one SRM remains OPERABLE, Required Action A.1 allows 4 hours to restore the required SRMs to OPERABLE status. This is a reasonable time since there is adequate capability remaining to monitor the core, limited risk of an event during this time, and sufficient time to take corrective actions to restore the required SRMs to OPERABLE status or to establish alternate IRM monitoring capability. During this time, control rod withdrawal and power increase are not precluded by this Required Action. Having the ability to monitor the core with at least one SRM, proceeding to IRM Range 3 or greater (with overlap required by SR 3.3.1.1.6) and thereby exiting the Applicability of this LCO, is acceptable for ensuring adequate core monitoring and allowing continued operation.

With three required SRMs inoperable, Required Action B.1 allows no positive changes in reactivity (control rod withdrawal must be immediately suspended) due to the inability to monitor the changes. Required Action A.1 still applies and allows 4 hours to restore monitoring capability prior to requiring control rod insertion. This allowance is based on the limited risk of an event during this time, provided that no control rod withdrawals are allowed, and the desire to concentrate efforts on repair, rather than to immediately shut down, with no SRMs OPERABLE.

Comments / Reference: PPM 3.1.2	Rev:	Major:093	Minor:N/A
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See damage.

4.10 With less than 3 SRMs operable in mode 2 and subcritical, suspend further control rod withdrawal until 3 SRMs are operable. {OE-2.42}

4.11 The accuracy of IDPM and ADPM instrumentation should be considered suspect until other

Comments / Reference: PPM 9.3.12	Rev:	Major:038	Minor:N/A
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NOTE: If the core has reached a subcritical condition and the reactor coolant is significantly elevated then there is a high risk that an uncontrolled re-criticality could occur as the core cools down. At coolant temperatures greater than 350 °F the risk is significantly higher therefore a manual reactor scram should be promptly inserted. Re-criticality event have occurred at other plants in as little as ten minutes after reaching the **Sustained Subcritical IRM Range Switch Position.** {7.1}

NOTE: Use the maximum reactor coolant temperature indicated by the operating RRC loops. Do not use the RRC loop temperature for an idle RRC loop.

NOTE: The basis for the limiting at 350 °F is the risk level of the core going re-critical. The higher the moderator temperature the greater the risk. For moderator temperatures less than 212 °F the coolant temperature will normally always continue to increase due to decay heat which normally adds negative reactivity. When the coolant temperature is at 212 °F the production of steam will begin and the moderator temperature will stabilize and remain relatively constant. At temperatures between 212 °F and 350 °F the change in core reactivity is normally small therefore the risk is low. For moderator temperatures significantly above 350 °F the available delta reactivity due to a decreasing moderator temperature can be large which increases the risk that the reactor will go re-critical during control rod insertion. Therefore above 350 °F a full core scram is appropriate.

- 3) IF the reactor coolant temperature is LE 350 °F, THEN recommend promptly inserting control rods per a shutdown pull sheet to the **Subcritical Hold Point**. If unable to insert control rods then insert a manual reactor scram.
- 4) IF the reactor coolant temperature is GT 350 °F, THEN recommend promptly inserting a **manual reactor scram**.

Question: SRO-87

Examination Outline Cross-reference: SRO-87

Revision: 0

Date: 10/5/22

Tier: 2

Group: 1

K/A Number: 223002 2.1.32

Level of Difficulty: 2

SRO Importance Rating: 4.0

K/A Description: PCIS/NSSSS - Ability to explain and apply system precautions, limitations, notes, or cautions

CGS is operating at 100% power.

An isolation of RCC to containment occurs.

Containment pressure is 1.70 psig and slowly rising.

To restore Primary Containment pressure, what action will be required?

Due to the F Isolation signal, ...

- A. drywell cooling interlocks will need to be overridden to maximize drywell cooling.
- B. the drywell will need to be vented using PPM 5.5.15, Emergency Drywell Venting.
- C. I&C support will be required to bypass containment isolation interlocks prior to venting.
- D. multiple operators will be required to hold open the containment vent valves per ABN-FAZ.

Answer: D

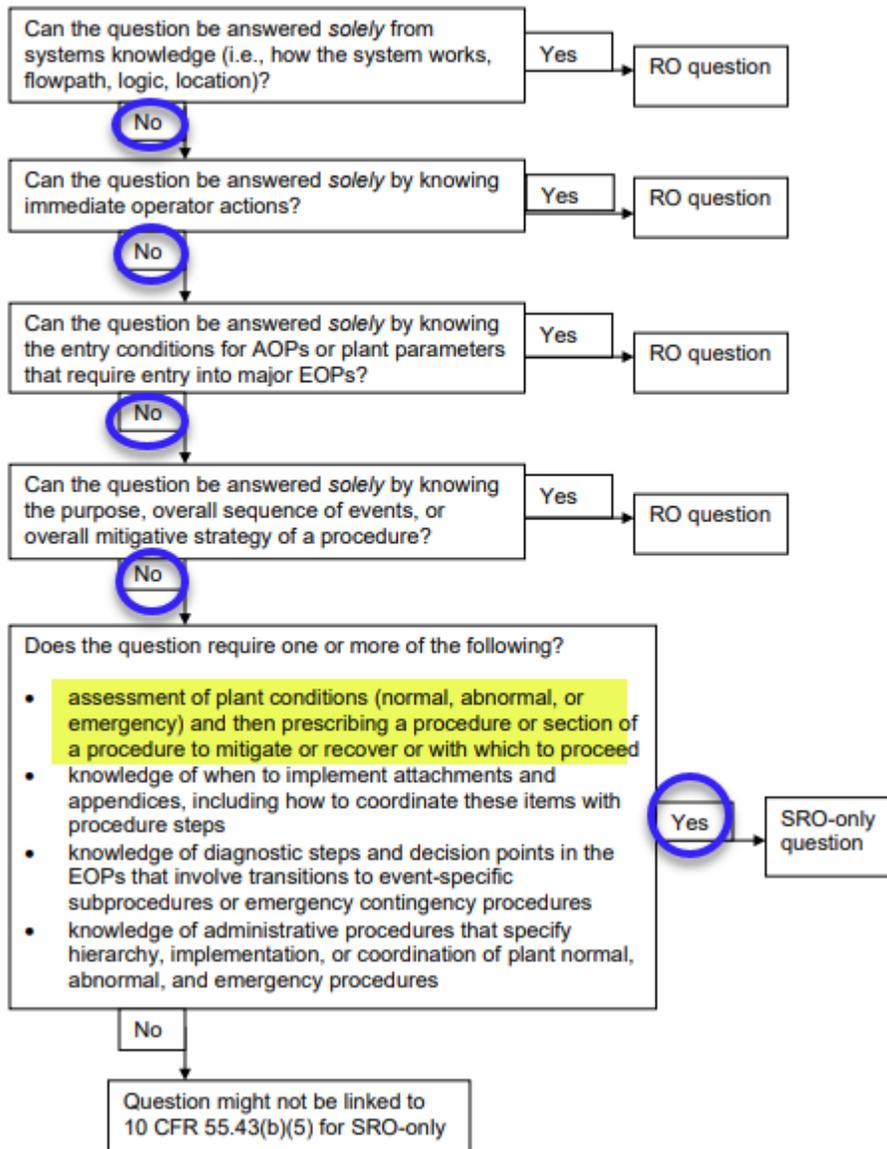
K/A Match:

Requires student to know and apply ABN-FAZ Note 4.2.2.b to determine the appropriate actions for opening PCIS valves.

Question: SRO-87

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Question: SRO-87

Explanation:

A. Incorrect. Plausible because the containment isolation was due to a loss of drywell cooling and restoring drywell cooling would lower containment pressure and allow resetting the containment isolation logic. Incorrect because this procedure is no longer active at CGS and the appropriate action is to vent containment per ABN-FAZ for this condition.

B. Incorrect. Plausible because the containment did isolate, and the student may believe this means an emergency condition does exist. Incorrect because PPM 5.5.15 is only to be used if containment is threatened and containment is not threatened at 1.7psig.

C. Incorrect. Plausible because with the F signal sealed in it is there is a signal being sent to maintain the containment vent valves closed. Incorrect because these valves will open if taken to open, and remain open as long as the switch is held in the open position. This is allowed iaw the note preceding 4.2.2.b of ABN-FAZ.

D. Correct. The stem of the question places the plant in a situation where containment pressure is above 1.68psig with no indication of a primary leak. This makes step 4.2.2.b of ABN-FAZ applicable for venting primary containment. The note for this step states that operators will need to hold SGT-V-1A, CEP-V-1B, and CEP-V-2B in the open position until logic can be reset.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-FAZ	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: NSSSS-02 – Describe the Nuclear Steam Supply Shutoff System (NS4) design feature(s) or interlocks that provide for the following: h. Manual defeating of selected isolations during specified emergency conditions

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires student to evaluate plant conditions and determine the appropriate response.

10 CFR Part 55 Content: 55.43 5

Question: SRO-87

Comments / Reference: ABN-FAZ	Rev:	Major: 21	Minor: N/A
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NOTE: If both divisions of RPS were lost, then power will need to be restored to both divisions to allow reset of the NS4 logic.

4.2.2 IF containment has lost cooling and cannot be rapidly restored, THEN PERFORM the following:

a. **VERIFY** a "Non-LOCA" condition exists by checking both of the following:

- No extreme local temperature transients have occurred in the Drywell. _____
- Drywell temperature and pressure are to the left of the prohibited region on Attachment 7.1, Drywell Venting Following an Isolation. _____

NOTE: In order to accomplish drywell venting with an F or A signal, the control switches for SGT-V-1A (1B), CEP-V-1B and CEP-V-2B have to be held in the open position until the isolation logic can be reset.

NOTE: CEP-V-11 loses position indication, position can be verified locally on RB 577 NE, K0/4.5.

b. WHEN a "Non-LOCA" condition has been verified, THEN PERFORM one of the following:

1) **VENT** the Drywell through SGT-A:

- a) **VERIFY** SGT train A is operating. _____
- b) **OPEN** SGT-V-1A (Inlet from Containment). _____
- c) **OPEN** CEP-V-1B (Drywell Exhaust Outbd Isol Bypass). _____
- d) **OPEN** CEP-V-2B (Drywell Exhaust Inbd Isol Bypass). _____

Question: SRO- 88

Examination Outline Cross-reference: 88

Revision: 0

Date: 9/20/22

Tier: 2

Group: 1

K/A Number: 259002 2.2.12

Level of Difficulty: 3

SRO Importance Rating: 4.0

K/A Description: Reactor Water Level Control: Knowledge of surveillance procedures

CGS is in Mode 1 and Reactor Power is being maintained at LE 99.5% by Heat Balance.

Surveillance ISP-RFW-Q401, Feedwater/Turbine Trip Actuation on Reactor High Level 8 – Channel Functional Test (CFT), is being performed to satisfy SR 3.3.2.2.2.

- CFT of Channel A completed satisfactory.
- CFT of Channel B was completed at 0600 on 1/13/23 and was **NOT** satisfactory. Parts are on order with an anticipated arrival date of 1/15/23.
- CFT of Channel C is in progress.

At 1200 on 1/13/23, the I&C Technician performing the surveillance presents the following information on Channel C to the CRS and states he anticipates completing the surveillance on Channel C by 1600:

Number: ISP-RFW-Q401	Use Category: CONTINUOUS	Major Rev: 014
Title: Feedwater/Turbine Trip Actuation on Reactor High Level 8 - CFT		Minor Rev: N/A
		Page: 16 of 18

7.4.8 **RECORD** trip point current in As-Found column of Table 7.3a. JS

Table 7.3a					
RFW-LS-624C					
Setpoint made	Calibration Limits made	As-Found		As-Left	
		Trip	Reset	Trip	Reset
18.52	18.36 to 18.52	<u>18.97</u>	N/A		N/A

7.4.9 On the DEH monitor screen that was selected in Section 7.1:
For DEH-CAB-03, CHASSIS: 1 SLOT: 3, **CONFIRM** point 30 (RFW-LS-624C) indicates that the STATUS **ON** light is green. JS

7.4.10 **CONFIRM** As-Found Trip in Table 7.3a occurred LE Allowable Value of 18.94 made.

a. Yes No JS

b. IF NO, THEN NOTIFY Operations. JS

Using the references provided, what Technical Specification (TS) Actions and Completion Times are required to avert a down-power?

Restore Channel C trip capability by (1). Place Channel B in trip by (2) if not restored.

- A. (1) 1400 on 1/13/23
 (2) 0600 on 1/20/23

Question: SRO- 88

- B. (1) 1800 on 1/13/23
(2) 0600 on 1/20/23
- C. (1) 1400 on 1/13/23
(2) 1200 on 1/20/23
- D. (1) 1800 on 1/13/23
(2) 1200 on 1/20/23

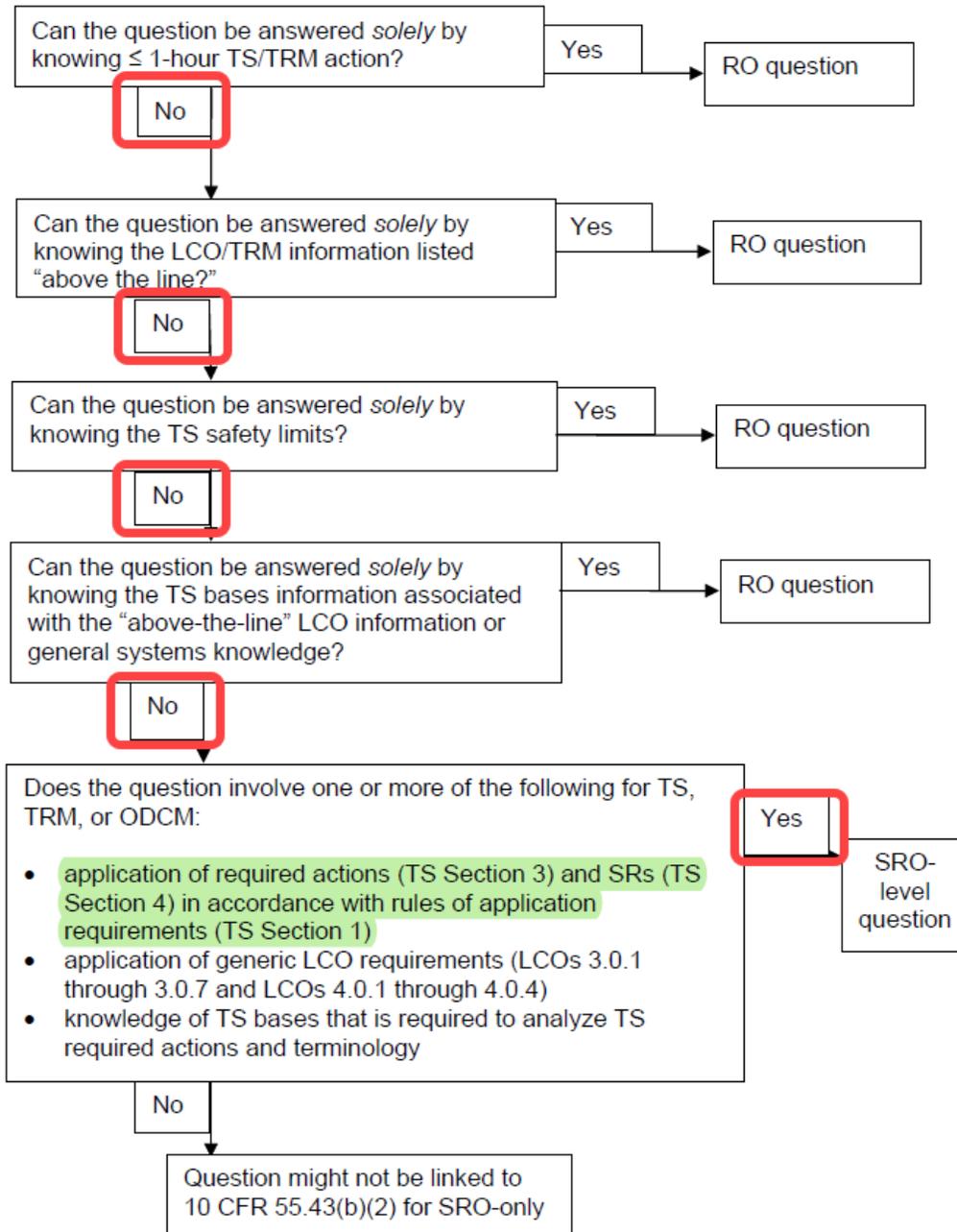
Answer: A

K/A Match:

The Candidate must recognize when a surveillance procedure for Reactor Water Level Control identifies a Channel out of tolerance and know that a surveillance failure constitutes a failure to meet the LCO.

Question: SRO- 88

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



c. Facility Licensee Procedures Required to Obtain Authority for Design and Operating Changes in the Facility [10 CFR 55.43(b)(3)]

Some examples of SRO-only examination items for this topic include the following:

Question: SRO- 88

Explanation: The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

- A. Correct: (1) Correct because the Note is not applicable if feedwater and main turbine high water level trip capability is not maintained. With both Channel B and Channel C out of tolerance, Condition B must be entered immediately, not after 6 hrs. Since Channel B is several days removed from being returned to service, Channel C must have trip capability restored within 2 hrs of the discovery time. Even though Channel C is still in surveillance, Condition B is entered at 1200 on 1/13/23 and Channel C trip capability must be restored by 1400 on 1/13/23.
 (2) Correct because the Note is not applicable due to the surveillance for Channel B being completed (vice being started) at 0600 on 1/13/23. As such, Condition A for Channel B is entered at 0600 on 1/13/23 and must be completed by 0600 on 1/20/23.
- B. Incorrect: (1) Incorrect – See A(1) above - Plausible because the surveillance modifying note could be misapplied.
 (2) Correct – See A(2) above.
- C. Incorrect: (1) Correct – See A(1) above.
 (2) Incorrect – See A(2) above - Plausible because the surveillance modifying note could be misapplied.
- D. Incorrect: (1) Incorrect – See A(1) above - Plausible because the surveillance modifying note could be misapplied.
 (2) Incorrect – See A(2) above - Plausible because the surveillance modifying note could be misapplied.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ISP-RFW-Q401, Feedwater/Turbine Trip Actuation on Reactor High Level 8 - CFT	
TS LCO 3.3.2.2, Feedwater and Main Turbine High Water Level Trip Instrumentation	
Instrument Master Data Sheet for RFW-DPT-4C	

- Proposed references to be provided during examination:
1. Excerpt of completed ISP-RFW-Q401 in question stem.
 2. LCO 3.3.2.2 including Surveillance Requirements. This does not include above the line information.
 3. Instrument Master Data Sheet for RFW-DPT-4C.

Question: SRO- 88

Learning Objective: FWLC-13 Given Feedwater Level Control System key parameter indications and plant conditions, determine if any Tech Spec Limiting Condition for Operations are applicable.

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

The Candidate must synthesize a knowledge of surveillance procedures with knowledge on the application of Technical Specifications to determine if any limiting conditions for operations are applicable and the actions and completion times required.

10 CFR Part 55 Content: 55.43 2

Comments / Reference: TS LCO 3.3.2.2	Rev:	Major: 225	Minor: N/A
Excerpt from LCO 3.3.2.2:			

Question: SRO- 88

Feedwater and Main Turbine High Water Level Trip Instrumentation 3.3.2.2			
3.3 INSTRUMENTATION			
3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation			
LCO 3.3.2.2 Three channels of feedwater and main turbine high water level trip instrumentation shall be OPERABLE.			
APPLICABILITY: THERMAL POWER ≥ 25% RTP.			
ACTIONS			
-----NOTE-----			
Separate Condition entry is allowed for each channel.			

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One feedwater and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days	
B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours	
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to < 25% RTP.	4 hours	

Feedwater and Main Turbine High Water Level Trip Instrumentation 3.3.2.2			
SURVEILLANCE REQUIREMENTS			
-----NOTE-----			
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided feedwater and main turbine high water level trip capability is maintained.			

SURVEILLANCE			FREQUENCY
SR 3.3.2.2.1	Perform CHANNEL CHECK.	In accordance with the	
Comments / Reference: LCO 3.3.2.2 Bases		Rev:	Major: 093 Minor: N/A

Question: SRO- 88

Excerpt from LCO 3.3.2.2 Bases:

LCO The LCO requires three channels of the Reactor Vessel Water Level - High, Level 8 instrumentation to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Level 8 signal. **Two of the three channels are needed to provide trip signals in order for the feedwater and main turbine trips to occur.** Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.3. The Allowable Value is set to

SURVEILLANCE REQUIREMENTS The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, **entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability.** Upon **completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.** This Note is based on the reliability analysis (Ref. 3) assumption that 6 hours is the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the feedwater pump turbines and main turbine will trip when necessary.

Comments / Reference: IMDS for RFW-DPT-4C Rev: Major: 013 Minor: N/A



WNP-2

DIC: 1801.1

INSTRUMENT MASTER DATA SHEET

Page 1 of 1

System Engineer _____

EPN: RFW-DPT-4C	SERVICE: Reactor Vessel Level 'C'
Loop: RFW-DPT-4C (± 1")	Rev. #13 Date: 10/22/2007
Work Order:	Input From: PI-V-X110/X112
Cal Procedure: ISP-RFW-Q401/ISP-RFW-B301	Output To: RFW-LI-606C, D, LS-624C

Input	Output (± tolerance)	Loop Check Data		
inwc (1) (2)	Desired mADC	Comp (B028) (3) TDAS x341 inwc	RFW-CRT-L003 RFW-LI-606B inwc	(Green Pen) RFW-LR-608 inwc
72.3	4 ± .08	0 ± 1"	0 ± 1"	0 ± 1"
61.7	8 ± .08	15 ± 1"	15 ± 1"	15 ± 1"
51.1	12 ± .08	30 ± 1"	30 ± 1"	30 ± 1"
40.6	16 ± .08	45 ± 1"	45 ± 1"	45 ± 1"
30.0	20 ± .08	60 ± 1"	60 ± 1"	60 ± 1"
RFW-ISOL-604C, RFW-LS-624C per MDS				

$$\frac{18.97 - 16}{20 - 16} = \frac{x - 45''}{60'' - 45''}$$

$$11.14'' + 45'' = 56.14'' \geq 56''$$

Therefore Channel C is out of tolerance and inoperable.

Examination Outline Cross-reference: 89

Revision: 0

Date: 9/12/22

Tier: 2

Group: 1

K/A Number: 263000

Level of Difficulty: 2

SRO Importance Rating: 4.2

K/A Description: Ability to (a) predict the impacts of the following on the DC Electrical Distribution and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Station blackout

CGS is in Mode 1.

A Station Blackout (SBO) occurs.

Operators are taking actions to mitigate the event.

Which of the following actions should the CRS direct to be completed within 2 hours?

- (1) RCIC Operations.
- (2) Compensatory Measures to Promote Control Room Cooling.
- (3) SBO DC Load Reduction.

- A. (1) ONLY
- B. (2) ONLY
- C. (1) and (3) ONLY
- D. (2) and (3) ONLY

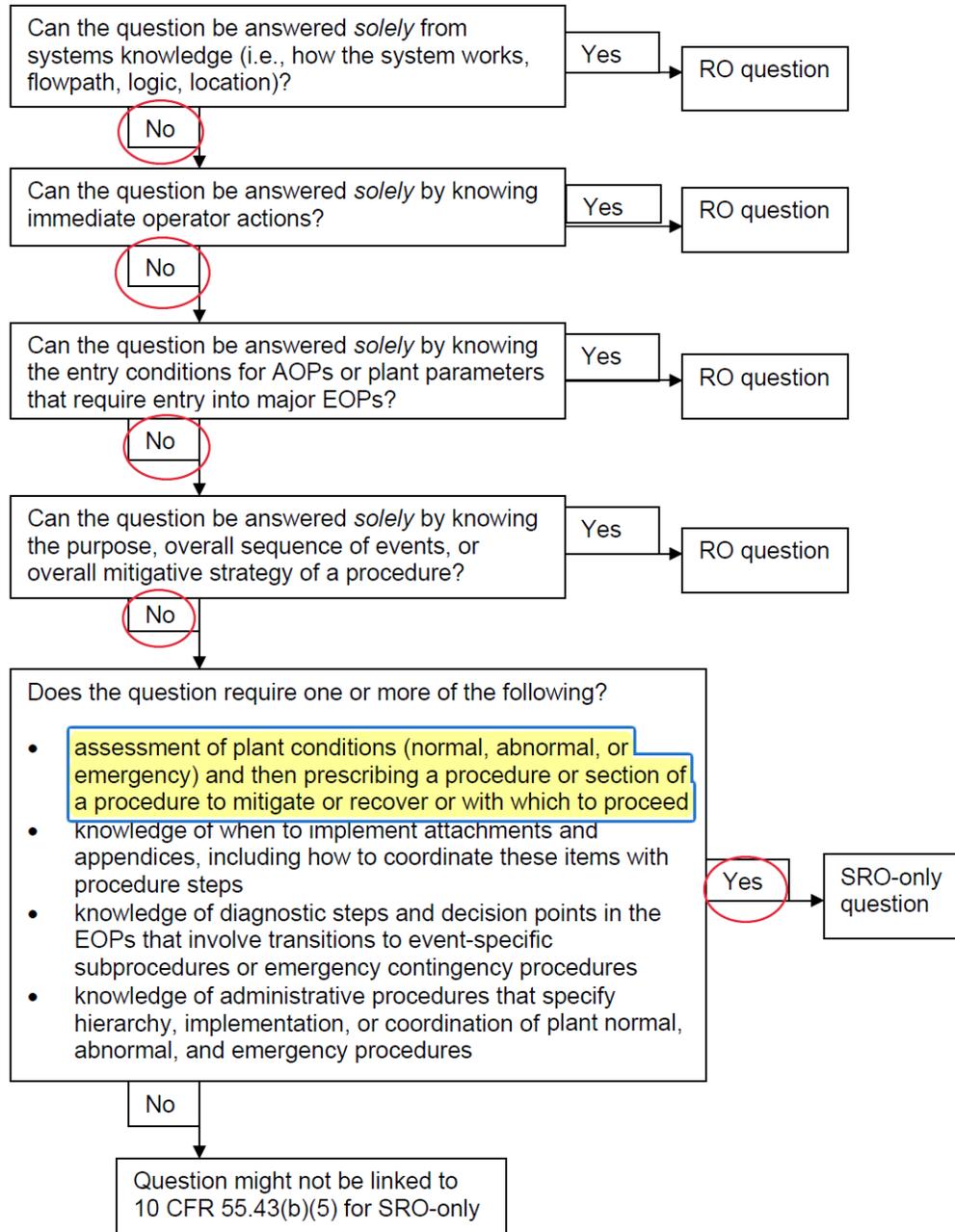
Answer: D

K/A Match:

Requires knowledge of the actions taken to mitigate effects on the DC distribution system during a Station Blackout.

SRO Only:

**Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)**



Explanation:

- A. Incorrect. Plausible since (3) is correct. However, compensatory actions to promote control room cooling must be complete within 30 minutes while DC load reduction must be completed within 60 minutes.

- B. Incorrect. Plausible if it is believed that reducing DC loads is the top priority and venting the generator is a higher priority than reducing heat loads in the control room. However, in accordance with PPM 5.6.1, SBO/ELAP, reducing control room heat loads must be completed in 30 minutes, reducing DC loads must be completed within 60 minutes and venting the generator must be completed within 120 minutes.
- C. Correct. In accordance with PPM 5.6.1, SBO/ELAP, the CRS should direct the completion of the following actions in this order: (2) Complete compensatory actions to promote CR cooling (PPM 5.6.2, Attachment 8.5) within 30 minutes; (1) Complete SBO DC Load reduction (PPM 5.6.2, Attachment 8.4) within 60 minutes; (3) vent the main generator (PPM 5.6.2, Attachment 8.9) within 120 minutes.
- D. Incorrect. Plausible if it is believed that venting the generator takes precedence over other actions due to safety implications. Plausibility is enhanced since (2) is the second action required to be taken. However, the correct sequence of completion is (2), (1), (3).

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 5.6.1, SBO/ELAP Flowchart	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 16261 - Given plant conditions and EOP flowcharts, evaluate plant conditions and determine the appropriate actions according to EOP 5.6.1

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Candidate must synthesize knowledge of the SBO/ELAP procedure, PPM 5.6.1, with an understanding of the time critical actions directed by the procedure.

10 CFR Part 55 Content: 55.43 5

Comments / Reference: PPM 5.6.1	Rev:	Major:031	Minor:000
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Key Time critical actions

30 min COMPLETE compensatory measures to promote Control Room cooling per Attachment 8.5

1 hour REDUCE station battery loads Attachment 8.4

2 hours VENT the main generator and stop SO-P-ASBU Attachment 8.9

Reference time line for required start times

1 PPM 5.6.2 SBO/ELAP Attachments		
No.	Title	Time Limit
8.1	Sound Powered Phones	3 hours
8.2	RCIC Operation	
8.3	Security Actions	
8.4	SBO DC Load Reduction Process	1 hour 30 min.
8.5	Compensatory Measure to Promote Control Room Cooling	
8.6	Power Restoration from Offsite Sources	2 hours
8.7	Power Restoration from 4160v DGs	
8.8	Power Restoration from 480v DG-4	
8.9	Main Generator Depressurization	
8.10	Balance of Plant	
8.11	Secondary Containment Supplementary Cooling	
8.12	Vital Island Temperature Control Actions	

Examination Outline Cross-reference: SRO-90

Revision: 0

Date: 10/5/22

Tier: 2

Group: 1

K/A Number: 400000 A2.01

Level of Difficulty: 2

SRO Importance Rating: 3.9

K/A Description: Ability to (a) predict the impacts of the following on the Component Cooling Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of CCW pump.

CGS is operating at 100% power with RCC-P-1B out of service for maintenance.

An overcurrent trip of RCC-P-1C occurs.

What plant response is expected?

What action will the CRS direct based on this response?

- A. (1) RCC-V-6 will close.
(2) Scram the reactor and secure RRC pumps.
- B. (1) RCC-V-6 will close.
(2) Stop RWCU Pumps, throttle open RWCU-V-104, and close RWCU-V-4.
- C. (1) RCC-V-6 will remain open.
(2) Scram the reactor and secure the RRC pumps.
- D. (1) RCC-V-6 will remain open.
(2) Stop RWCU Pumps, throttle open RWCU-V-104, and close RWCU-V-4.

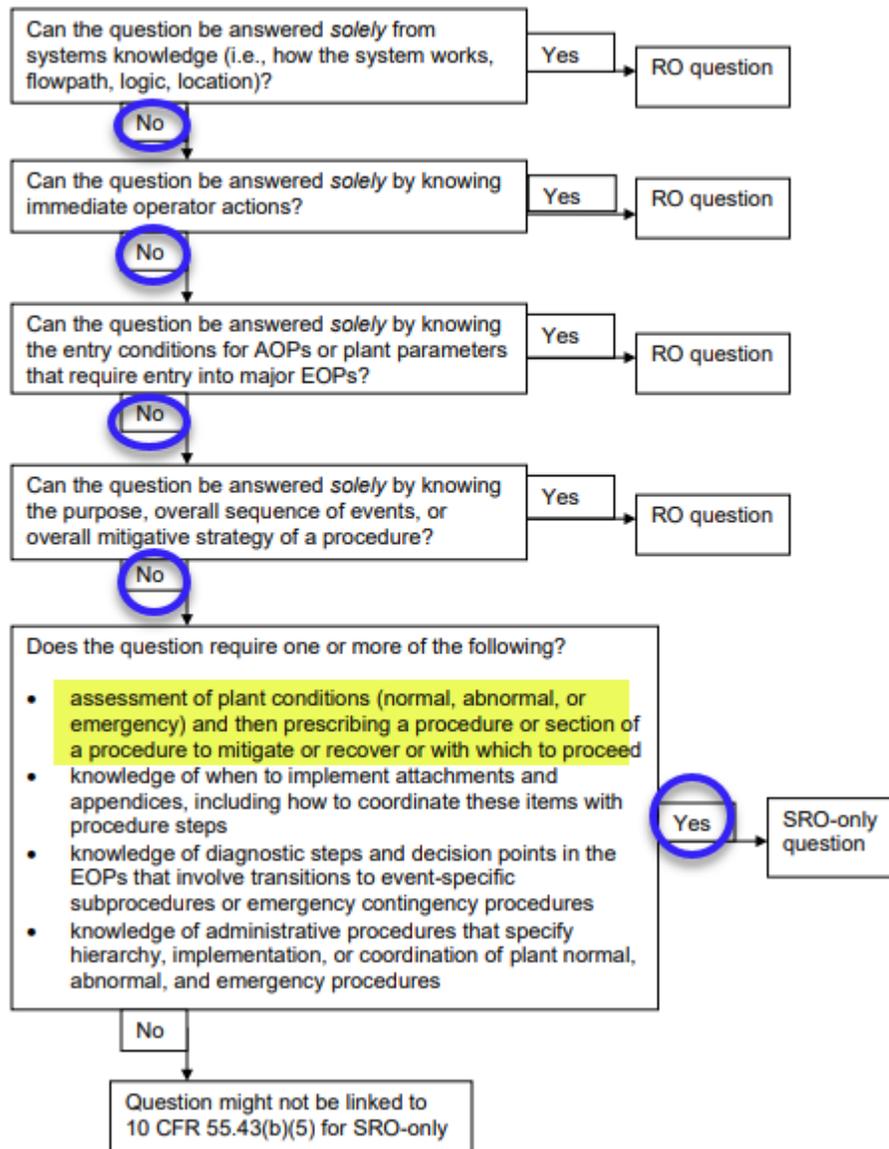
Answer: B

K/A Match:

Question requires understanding of the impact of a loss of an RCC pump on the RCC system and the actions required to mitigate adverse effects due to the loss of cooling.

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



Explanation:

- A. Incorrect. (1) is correct see explanation B.
(2) Incorrect. Plausible because if all RCC is lost the cooling to RCC pumps will be lost and ABN-RCC directs scrambling the reactor and securing RCC pumps. This is incorrect because RCC-P-1A is still running in this scenario so all cooling to the RCC pumps has not been lost.
- B. Correct. (1) In the stem of the question RCC-P-1B is OOS for maintenance and RCC-P-1C trips. This leaves only 1 RCC pump running. With only 1 RCC pump running RCC-V-6 will close.
(2) With RCC-V-6 closed there is no cooling water to the RWCU heat exchangers. This will cause the RWCU system to overheat, potentially damaging the resin. Due to this RWCU-V-104 is throttled opened and RWCU is isolate per ABN-RCC.
- C. Incorrect. (1) is plausible if the student believes that RCC-V-6 opens to provide additional cooling to components in the drywell when RCC flow is reduced. Incorrect because in the stem of the question RCC-P-1B is OOS and RCC-P-1C trips leaving only 1 RCC pump running, with only 1

RCC pump running RCC-V-6 will close.

(2) Incorrect. Plausible because if all RCC is lost the cooling to RCC pumps will be lost and ABN-RCC directs scrambling the reactor and securing RCC pumps. This is incorrect because RCC-P-1A is still running in this scenario so all cooling to the RCC pumps has not been lost.

D. Incorrect. (1) is plausible if the student believes that RCC-V-6 opens to provide additional cooling to components in the drywell when RCC flow is reduced. Incorrect because in the stem of the question RCC-P-1B is OOS and RCC-P-1C trips leaving only 1 RCC pump running, with only 1 RCC pump running RCC-V-6 will close.

(2) is correct see explanation B.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
ABN-RCC	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 15794 – Discuss all contingencies associated with the subsequent operator actions of ABN-RCC.

Question Source: Bank #: _____ Bank #
 Modified Bank #: _____ Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires student to evaluate plant status and determine the appropriate equipment and operator response.

10 CFR Part 55 Content: 55.43 5

Comments / Reference: ABN-RCC	Rev:	Major: 7	Minor: 1
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2.0 AUTOMATIC ACTIONS

2.1 The standby RCC pump starts when a running RCC pump trips.

2.2 RCC-V-6 **CLOSES** (RW/RB Supply) when LT two RCC pump motor breakers are **CLOSED**.

4.0 SUBSEQUENT OPERATOR ACTIONS

NOTE: Step 4.2 provides direction for a partial loss of RCC cooling capacity.

4.1 IF a complete loss of RCC flow occurs
OR a loss of RCC flow to the Drywell occurs,
THEN **PERFORM** the following:

4.1.1 **SCRAM the Reactor** per PPM 3.3.1. _____

4.1.2 IF NOT in ATWS GT 5% reactor power,
THEN **STOP** both Reactor Recirculation Pumps (RRC-P-1A and RRC-P-1B). _____

4.2.3 IF RCC-V-6 (RW/RB Supply) is closed,
THEN **PERFORM** the following.

a. **STOP** RWCU-P-1A(1B). _____

b. **THROTTLE OPEN** RWCU-V-104. _____

c. **CLOSE** RWCU-V-4 (RWCU Suction Outboard Isolation). _____

Examination Outline Cross-reference: SRO-91

Revision: 0

Date: 9/14/22

Tier: 2

Group: 2

K/A Number: 201002 A2.02

Level of Difficulty: 3

SRO Importance Rating: 4.0

K/A Description: Ability to (a) predict the impacts of the following on the Reactor Manual Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Rod drift

CGS is in Mode 1.

- Reactor Power is 100%.

Control rod exercise (Week 1) is in progress in accordance with OSP-CRD-M701.

The RO inserts rod 26-47 from position 48 to position 46 and releases the INSERT pushbutton.

Rod 26-47 continues to insert and stops at position 38.

The crew is unable to fully insert the control rod.

What is the next action should the CRS direct?

- A. Scram the reactor in accordance with PPM 3.3.1, Reactor Scram.
- B. Reduce core flow to 74 Mlbm/hr in accordance with ABN-ROD, Control Rod Faults.
- C. Hydraulically isolate rod 26-47 HCU in accordance with ABN-ROD, Control Rod Faults.
- D. Withdraw the control rod in one notch increments in accordance with OSP-CRD-M701, Control Rod Exercise (Week 1).

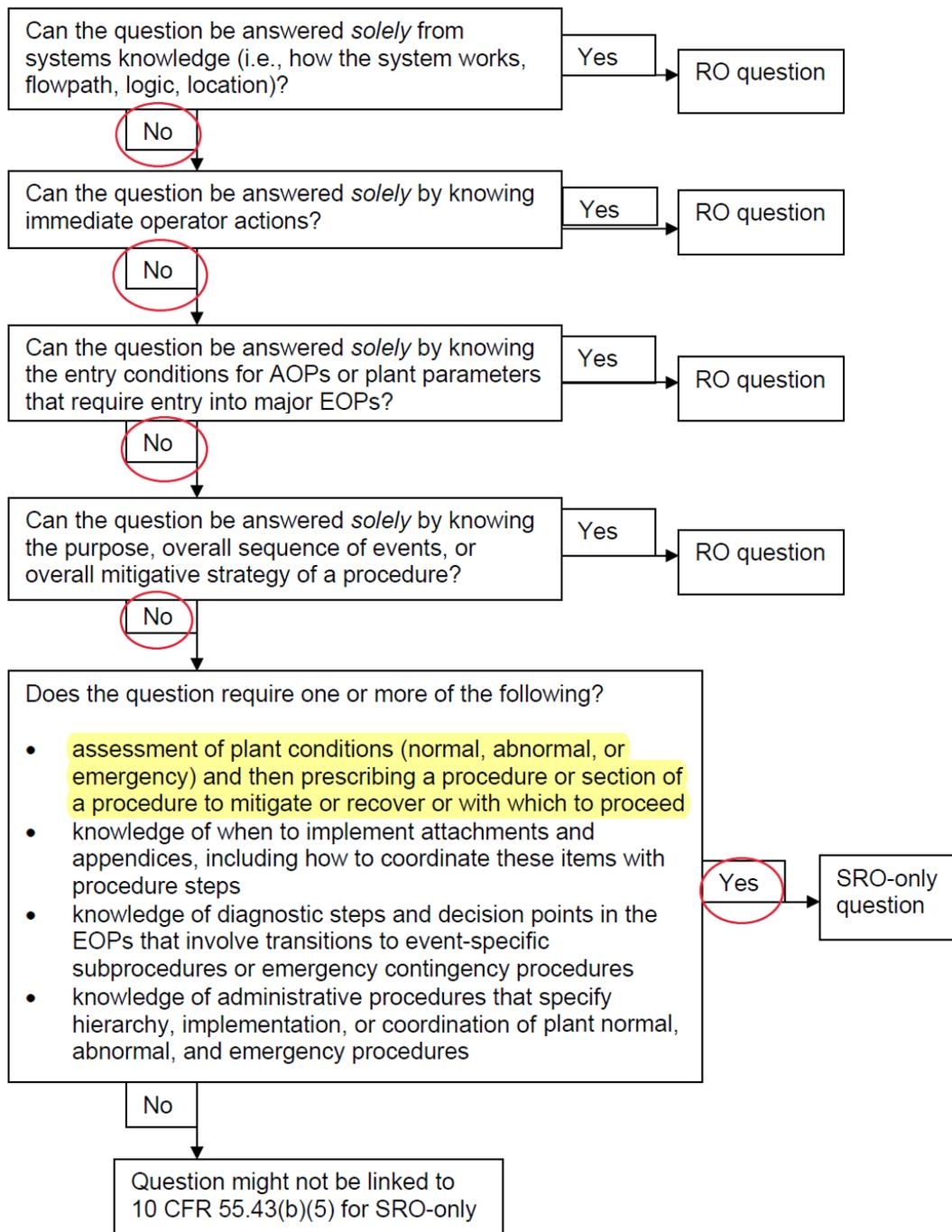
Answer: B

K/A Match:

Requires knowledge of actions required to mitigate the effects of a rod drift.

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Explanation:

- A. Incorrect. Plausible since this is the correct action for two or more simultaneous drifting rods. However, the reactor is not scrammed for a single drifting rod.

- B. Correct. For a drifting rod that cannot be fully inserted, core flow is reduced to 74 Mlbm/hr in accordance with ABN-ROD, step 4.1.1.
- C. Incorrect. Plausible since the HCU is isolated for drifting rods that must be maintained full in by holding the INSERT or CONTINUOUS INSERT pushbutton. However, for the conditions given, this action is not required.
- D. Incorrect. Plausible since this would be the action for an accidental insertion of more notches than desired (double notch). However, since the rod moved after the INSERT pushbutton was released, this is not an accidental insertion but a drifting control rod.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
OSP-CRD-M701, Control Rod Exercise (Week 1)	
ABN-ROD, Control Rod Faults	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: CRDH-07 Given a Control Rod Drive Hydraulics Annunciator, describe:
The condition causing the annunciator; Any automatic actions; Any operational implications

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize an understanding of the symptoms of a drifting control rod with a knowledge of the actions required for the conditions given.

10 CFR Part 55 Content: 55.43 5

Comments / Reference: OSP-CRD-M701	Rev:	Major: 031	Minor: N/A
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Number: OSP-CRD-M701	Use Category: CONTINUOUS	Major Rev: 031
Title: Control Rod Exercise (Week 1)		Minor Rev: N/A Page: 6 of 21

4.0 PRECAUTIONS AND LIMITATIONS

4.1 **ABN-ROD should be referred to whenever a control rod is placed in a position other than the position referenced on the pull sheet or as required by this procedure.** {P-2.10}

4.1.1 In the event of accidental withdrawal of any control rod more notches than desired (a double notch), insert the rod to its original position, inform the CRS and IA, and refer to ABN-ROD for additional actions.

4.1.2 If a flux shaping control rod is selected and moved inadvertently, notify the CRS and IA immediately and refer to ABN-ROD.

4.1.3 **In the event of accidental insertion of more notches than desired (a double notch), inform the CRS and IA and refer to ABN-ROD. Withdrawal should be by single notch to avoid any possible over notching on the withdrawal.**

Comments / Reference: ABN-ROD	Rev:	Major:035	Minor:N/A
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Number: ABN-ROD	Use Category: CONTINUOUS	Major Rev: 035
Title: Control Rod Faults		Minor Rev: N/A Page: 4 of 41

3.0 IMMEDIATE OPERATOR ACTIONS

3.1 **MANUALLY SCRAM the Reactor** if any of the following conditions exist: {P-214961} _____

- Three or more control rods have SCRAMMED and cause is NOT known.
- **Two or more control rods simultaneously drift.**

3.2 **VERIFY** the drifting or scrammed control rod(s) are full in. _____

3.3 IF the control rod starts to drift back out when the insert signal is removed, THEN **PERFORM** the following: _____

3.3.1 **APPLY** an insert signal. _____

3.3.2 **ISOLATE** the control rod per step 4.1.2. _____

4.1 Drifting or Scrammed Control Rod

1

- 4.1.1 **IF** the affected control rod is **NOT** full in,
AND core flow is GT 80 Mlbm/hr (on MS-FR-613)
THEN REDUCE core flow to 74 Mlbm/hr at 5% (Reactor Power) per minute.
{P-77715} _____

NOTE: A control rod that requires a CONTINUOUS INSERT command to remain inserted likely has a stuck open 122 DCV or a stuck collet mechanism (Reference GE SIL 292 and GE SIL 310).

- 4.1.2 **IF** the control rod is being maintained full in by holding the CONTINUOUS INSERT or INSERT pushbutton,
THEN PERFORM the following:
- a. **HYDRAULICALLY ISOLATE** the applicable HCU as follows:
- 1) **CLOSE** CRD-V-103/HCU (Drive Water Inlet Isolation). _____
 - 2) **CLOSE** CRD-V-105/HCU (Exhaust Water Isolation). _____

Examination Outline Cross-reference: 92

Revision: 0

Date: 9/13/22

Tier: 2

Group: 2

K/A Number: 201006.2.2.22

Level of Difficulty: 3

SRO Importance Rating: 4.2

K/A Description: Rod Worth Minimizer - Knowledge of limiting conditions for operation and safety limits.

CGS is in Mode 3.

The crew is preparing to enter Mode 2 for a reactor startup in accordance with PPM 3.1.2, Reactor Plant Startup.

The Rod Worth Minimizer (RWM) is inoperable.

- This is the first reactor startup to be performed in 24 months.

Using the reference provided, how is the mode change affected?

Mode 2 may be entered...

- A. only after returning the RWM to operable.
- B. but the RWM must be operable prior to entering Mode 1.
- C. with an additional licensed operator to verify control rod movement is in compliance with BPWS.
- D. only after completing a formal risk assessment and establishing risk management actions.

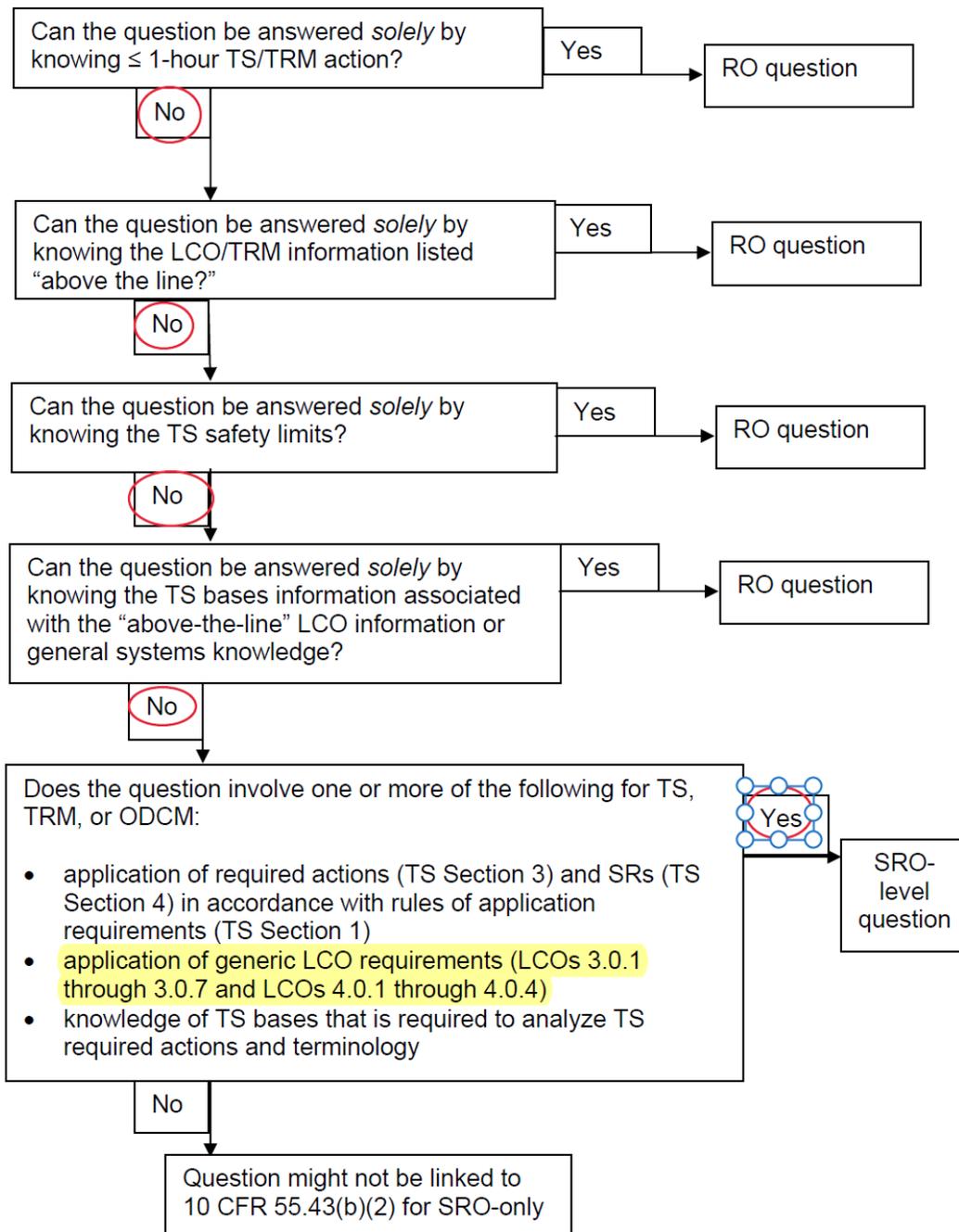
Answer: C

K/A Match:

Requires knowledge of general LCO 3.0.4.

SRO Only:

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



Explanation:

- A. Incorrect. Plausible since the RWM is required to be operable in Modes 2 and 3 in accordance with LCO 3.3.2.1. Additionally, a Mode change is typically not made if equipment required for that mode is not operable. However, LCO 3.0.4 allows entering a mode without all equipment operable is the associated actions allow the plant to remain in the new Mode indefinitely.
- B. Incorrect. Plausible since performing a risk assessment and establishing risk management actions is an allowed method to entering a mode where all required equipment is not operable in accordance with LCO 3.0.4b. However, since the actions required for the inoperable RWM allow the plant to remain in the new mode indefinitely, a formal risk assessment is not required.
- C. Correct. In accordance with TS LCO 3.0.4a: “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;”
- Since the actions listed in condition C of LCO 3.3.2.1 allow the plant to remain in Mode 2, Mode 2 entry is allowed. Additionally, LCO 3.3.2.1, Required Action C.2.2 requires an additional licensed operator to verify control rods remain in compliance with BPWS during rod motion.
- D. Incorrect. Plausible if it is believed that an additional Mode change is not allowed or that the mode of applicability for LCO 3.3.2.1 is only Mode 2. However, the mode of applicability is Modes 2 and 1. Additionally, there is no limit to the number of mode changes that can be made using LCO 3.0.4.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
Technical Specifications LCO 3.3.2.1	
Technical Specifications LCO 3.0.4	
Tech Ref 4	

Proposed references to be provided during examination: LCO 3.3.2.1, Actions (above the line blanked out).

Learning Objective: RWM-13 Given Rod Worth Minimizer key parameter indications and plant conditions, determine if any Tech Spec Limiting Condition for Operations are applicable.

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize a knowledge of the plant process for mode changes along with an understanding of the application of LCO 3.0.4.

10 CFR Part 55 Content: 55.43 2

Comments / Reference: LCO 3.3.2.1	Ammendment: 267
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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:

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 Suspend control rod movement except by scram. <u>OR</u> C.2.1.1 Verify ≥ 12 rods withdrawn. <u>OR</u>	Immediately Immediately

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2.1.2 Verify by administrative methods that startup with RWM inoperable has not been performed in the last calendar year.	Immediately
	<p style="text-align: center;"><u>AND</u></p> C.2.2 Verify movement of control rods is in compliance with banked position withdrawal sequence (BPWS) by a second licensed operator or other qualified member of the technical staff.	During control rod movement
D. RWM inoperable during reactor shutdown.	D.1 Verify movement of control rods is in compliance with BPWS by a second licensed operator or other qualified member of the technical staff.	During control rod movement
E. One or more Reactor Mode Switch - Shutdown Position channels inoperable.	E.1 Suspend control rod withdrawal.	Immediately
	<p style="text-align: center;"><u>AND</u></p> E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

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;
- b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate (exceptions to this Specification are stated in the individual Specifications); or

LCO 3.0.4 (continued)

- c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

Question: SRO-93

Examination Outline Cross-reference: SRO-93

Revision: 0

Date: 5/17/2022

Tier: 2

Group: 2

K/A Number: 234000 K6.02

Level of Difficulty: 4

SRO Importance Rating: 3.4

K/A Description: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Fuel Handling System: Reactor manual control system

CGS is in Mode 5 performing Refueling Operations.

The refuel bridge is currently over the core and inserting a fuel bundle when RO1 notes that there are no control rod blocks.

With regards to T.S. what actions are required? What is the bases for these actions?

Immediately (1).

To prevent criticality during refueling by ensuring (2).

- A. (1) Suspend in-vessel fuel movement
(2) that no more than one control rod can be withdrawn
- B. (1) Suspend in-vessel fuel movement
(2) fuel assemblies are not loaded into the core with a control rod withdrawn
- C. (1) Initiate action to insert all insertable control rods
(2) that no more than one control rod can be withdrawn
- D. (1) Initiate action to insert all insertable control rods
(2) fuel assemblies are not loaded into the core with a control rod withdrawn

Answer: B

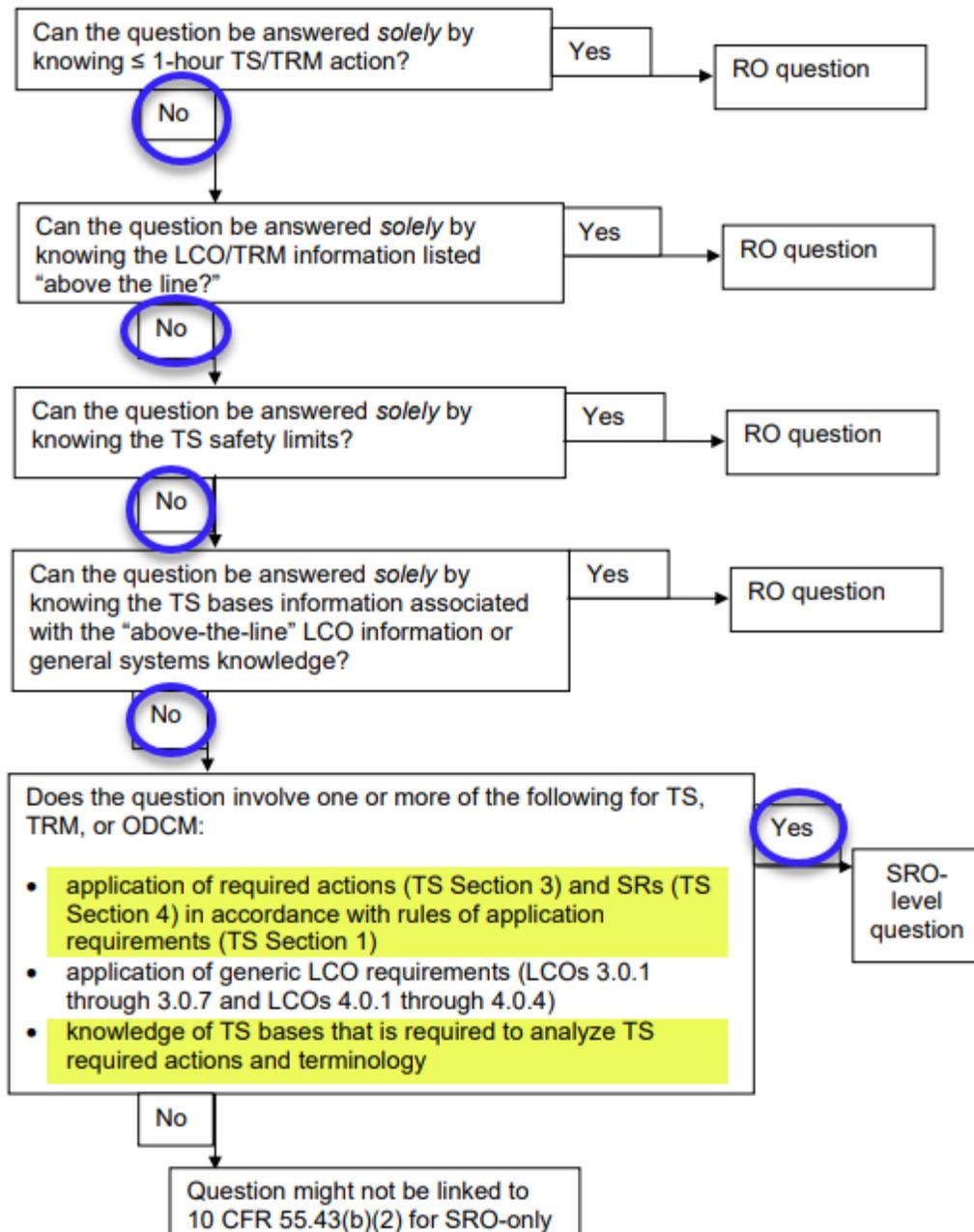
K/A Match:

Requires student to apply knowledge of a failure of RMCS control rod block circuitry to its effects on refuel floor interlocks.

Question: SRO-93

SRO Only:

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



Explanation:

- A. Incorrect. (1) is correct. See explanation B. (2) is incorrect. This is plausible since this is the bases for the One-Rod-Out Interlock T.S. (3.9.2) which requires immediately inserting all insertable control rods. This is incorrect because the failure was due to refuel bridge position which is covered under T.S. 3.9.1
- B. Correct. (1) with the refuel bridge over the core T.S. 3.9.1 requires that certain interlocks be operable. These interlocks require that a control rod block be enforced if the bridge is over the

Question: SRO-93

core while in refuel. The action for these interlocks being inoperable is to immediately suspend in-vessel fuel movement.
(2) is correct. Per T.S. 3.9.1 Bases this T.S. is to prevent criticality during refueling be ensuring fuel assemblies are not loaded into the core where there is a withdrawn control rod.

C. Incorrect. (1) Plausible because per T.S. 3.9.2 if the one-rod-out interlock is not working then action will be taken to immediately initiate action to insert all insertable control rods. This is incorrect because the failure in the stem is a refuel bridge over the core failure which requires suspending all fuel movement in the core per T.S. 3.9.1.
(2) is incorrect. This is plausible since this is the bases for the One-Rod-Out Interlock T.S. (3.9.2) which requires immediately inserting all insertable control rods. This is incorrect because the failure was due to refuel bridge position which is covered under T.S. 3.9.1

D. Incorrect. (1) Plausible because per T.S. 3.9.2 if the one-rod-out interlock is not working then action will be taken to immediately initiate action to insert all insertable control rods. This is incorrect because the failure in the stem is a refuel bridge over the core failure which requires suspending all fuel movement in the core per T.S. 3.9.1.
(2) is correct. See explanation B.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
Tech Specs 3.9.1 and 3.9.2	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 5362 - With a copy of the Technical Specifications provided, given a specific Plant condition determine the L.C.O.'s directly associated with the Fuel Handling system that are affected.

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

Question History: Last NRC Exam: Exam year or N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires student to interpret plant conditions and apply those conditions to a T.S. determination.

10 CFR Part 55 Content: 55.43 2

Question: SRO-93

Comments / Reference: T.S. 3.9.1 and 3.9.2	Rev:	Major:	Maj	Minor:	Min										
<p>3.9 REFUELING OPERATIONS</p> <p>3.9.1 Refueling Equipment Interlocks</p> <p>LCO 3.9.1 The refueling equipment interlocks associated with the refuel position shall be OPERABLE.</p> <p>APPLICABILITY: During in-vessel fuel movement with equipment associated with the interlocks when the reactor mode switch is in the refuel position.</p> <p>ACTIONS</p> <table border="1" style="width: 100%; border-collapse: collapse;"> <thead> <tr> <th style="width: 33%;">CONDITION</th> <th style="width: 33%;">REQUIRED ACTION</th> <th style="width: 33%;">COMPLETION TIME</th> </tr> </thead> <tbody> <tr> <td>A. One or more required refueling equipment interlocks inoperable.</td> <td>A.1 Suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s).</td> <td>Immediately</td> </tr> </tbody> </table> <p>SURVEILLANCE REQUIREMENTS</p> <table border="1" style="width: 100%; border-collapse: collapse;"> <thead> <tr> <th style="width: 60%;">SURVEILLANCE</th> <th style="width: 40%;">FREQUENCY</th> </tr> </thead> <tbody> <tr> <td> SR 3.9.1.1 Perform CHANNEL FUNCTIONAL TEST on each of the following required refueling equipment interlock inputs: <ul style="list-style-type: none"> a. All-rods-in, b. Refueling platform position, c. Refueling platform fuel grapple fuel-loaded, d. Refueling platform frame-mounted hoist fuel-loaded, and e. Refueling platform trolley-mounted hoist fuel-loaded. </td> <td>In accordance with the Surveillance Frequency Control Program</td> </tr> </tbody> </table>						CONDITION	REQUIRED ACTION	COMPLETION TIME	A. One or more required refueling equipment interlocks inoperable.	A.1 Suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s).	Immediately	SURVEILLANCE	FREQUENCY	SR 3.9.1.1 Perform CHANNEL FUNCTIONAL TEST on each of the following required refueling equipment interlock inputs: <ul style="list-style-type: none"> a. All-rods-in, b. Refueling platform position, c. Refueling platform fuel grapple fuel-loaded, d. Refueling platform frame-mounted hoist fuel-loaded, and e. Refueling platform trolley-mounted hoist fuel-loaded. 	In accordance with the Surveillance Frequency Control Program
CONDITION	REQUIRED ACTION	COMPLETION TIME													
A. One or more required refueling equipment interlocks inoperable.	A.1 Suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s).	Immediately													
SURVEILLANCE	FREQUENCY														
SR 3.9.1.1 Perform CHANNEL FUNCTIONAL TEST on each of the following required refueling equipment interlock inputs: <ul style="list-style-type: none"> a. All-rods-in, b. Refueling platform position, c. Refueling platform fuel grapple fuel-loaded, d. Refueling platform frame-mounted hoist fuel-loaded, and e. Refueling platform trolley-mounted hoist fuel-loaded. 	In accordance with the Surveillance Frequency Control Program														

Question: SRO-93

3.9 REFUELING OPERATIONS

3.9.2 Refuel Position One-Rod-Out Interlock

LCO 3.9.2 The refuel position one-rod-out interlock shall be OPERABLE.

APPLICABILITY: MODE 5 with the reactor mode switch in the refuel position and any control rod withdrawn.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Refuel position one-rod-out interlock inoperable.	A.1 Suspend control rod withdrawal.	Immediately
	<u>AND</u> A.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.2.1 Verify reactor mode switch locked in refuel position.	In accordance with the Surveillance Frequency Control Program

Question: SRO-93

B 3.9 REFUELING OPERATIONS

B 3.9.1 Refueling Equipment Interlocks

BASES

The refueling platform has two mechanical switches that open before the platform or any of its hoists are physically located over the reactor vessel. Each hoist load is sensed by an electronic load cell. The fuel grapple and frame-mounted hoist load signals are inputs to a programmable logic controller (PLC). The PLC performs the associated interlock and load functions. The trolley-mounted hoist load cell inputs to setpoint modules that perform their associated interlock and load functions. The PLC and setpoint modules open the associated fuel-loaded circuits at a load lighter

than the weight of a single fuel assembly in water. The refueling interlocks use these indications to prevent operation of the refueling equipment with fuel loaded over the core whenever any control rod is withdrawn, or to prevent control rod withdrawal whenever fuel loaded refueling equipment is over the core (Ref. 2).

LCO To prevent criticality during refueling, the refueling interlocks associated with the refuel position ensure that fuel assemblies are not loaded into the core with any control rod withdrawn.

To prevent these conditions from developing, the all-rods-in, the refueling platform position, the refueling platform fuel grapple fuel-loaded, the refueling platform frame-mounted hoist fuel-loaded, and the refueling platform trolley-mounted hoist fuel-loaded inputs are required to be OPERABLE. These inputs are combined in logic circuits that provide refueling equipment or control rod blocks to prevent operations that could result in criticality during refueling operations.

B 3.9 REFUELING OPERATIONS

B 3.9.2 Refuel Position One-Rod-Out Interlock

BASES

LCO To prevent criticality during MODE 5, the refuel position one-rod-out interlock ensures no more than one control rod may be withdrawn. Both channels of the refuel position one-rod-out interlock are required to be OPERABLE and the reactor mode switch must be locked in the refuel position to support the OPERABILITY of these channels.

Question: SRO-94

Examination Outline Cross-reference: 94

Revision: 0

Date: 4/28/22

Tier: 3

Group: N/A

K/A Number: 2.1.35

Level of Difficulty: 2

SRO Importance Rating: 3.9

K/A Description: Knowledge of the fuel handling responsibilities of SROs (SRO Only)

Which of the following evolutions must be directly supervised by a licensed SRO?

- (1) Fuel movement from the core to the Spent Fuel Pool.
- (2) Movement of LPRMs within the core.
- (3) Fuel movement between locations in the core.
- (4) Control Rod blade replacement in a defueled cell.

- A. (1) and (2) ONLY
- B. (1) and (3) ONLY
- C. (2) and (4) ONLY
- D. (3) and (4) ONLY

Answer: B

K/A Match:

Requires knowledge of SRO responsibilities during fuel handling evolutions.

SRO Only:

The question requires knowledge of refuel floor SRO responsibilities which is an SRO-only topic as delineated in NUREG-1021, ES-4.2, page 25 and 10 CFR 55.43(b)(7)

g. Fuel Handling Facilities and Procedures [10 CFR 55.43(b)(7)]

Some examples of SRO-only examination items for this topic include the following:

- refuel floor SRO responsibilities
- assessment of fuel handling equipment SR acceptance criteria
- prerequisites for vessel disassembly and reassembly
- decay heat assessment
- assessment of SRs for the refueling mode
- reporting requirements
- emergency classifications

Explanation:

- A. Incorrect. Plausible since (1) is considered a CORE ALTERATION in accordance with TS and must be supervised by a licensed SRO. However, (2) is specifically called out in TS as an evolution that is not considered a CORE ALTERATION, and therefore, licensed SRO supervision is not required.
- A. Correct. In accordance with TS, both (1) and (3) are considered a CORE ALTERATION, and must be supervised by a licensed SRO.
- B. Incorrect. Plausible since (4) must be supervised by a licensed SRO when a control rod blade is replaced in a fueled cell. However, (2) is specifically called out in TS as an evolution that is not considered a CORE ALTERATION and since the control rod blade is being replaced in a non-fueled cell, it is not considered a CORE ALTERATION. Therefore, licensed SRO supervision is not required for either evolution.
- C. Incorrect. Plausible since (3) must be supervised by a licensed SRO and 4) must be supervised by a licensed SRO when a control rod blade is replaced in a fueled cell. However, since the control rod blade is being replaced in a non-fueled cell, it is not considered a CORE ALTERATION. Therefore, licensed SRO supervision is not required for this evolution.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
Technical Specifications	
PPM 6.3.2, Fuel Shuffling and/or Offloading and Reloading	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 3051 - Define Core Alteration, including items that are specifically excluded from the definition.

Question Source: Bank #: 100456
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: 2021

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Candidate must synthesize a knowledge of evolutions that are considered Core Alterations with an understanding of the supervisory requirements for performing Core Alterations.

10 CFR Part 55 Content: 55.43 7

Comments / Reference: TS Section 1.1, Definitions	Rev:	Major:	265	Minor:	Min
CORE ALTERATION	<p>CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:</p> <ul style="list-style-type: none">a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); andb. Control rod movement, provided there are no fuel assemblies in the associated core cell.				

Comments / Reference: PPM 6.3.2		Rev:	Major:027	Minor:N/A
Number: 6.3.2		Use Category: REFERENCE		Major Rev: 027
Title: Fuel Movement with the Refueling Bridge				Minor Rev: N/A
				Page: 8 of 25
5.0	<u>PROCEDURE</u>			
5.1	Management Expectations			
5.1.1	When moving fuel involving CORE ALTERATIONS with the Refueling Bridge the personnel required include the following:			
	<u>Refueling Bridge</u>			
	<ul style="list-style-type: none"> • Core Alt Supervisor – an SRO with RFAE qualification • Operator – with RFAA or RFAD qualification • Spotter - with RFAA or RFAD qualification 			
	<u>Main Control Room</u>			
	<ul style="list-style-type: none"> • Reactor Operator – dedicated to refueling operations 			
	<p>With this configuration of bridge personnel, the Spotter is responsible for communicating NCTL steps to the Operator. Either the Operator or Spotter may sign the NCTL as the performer. The Core Alt Supervisor signs the NCTL as verifier. The Reactor Operator is responsible for monitoring SRM readings during bundle insertions to the core. Either the Spotter or Core Alt Supervisor will be in direct communications with the Reactor Operator during CORE ALTERATIONS.</p>			

Examination Outline Cross-reference: 95

Revision: 0

Date: 4/28/22

Tier: 3

Group: N/A

K/A Number: 2.1.37

Level of Difficulty: 3

SRO Importance Rating: 4.6

K/A Description: Knowledge of procedures, guidelines, or limitations associated with reactivity management

CGS is in Mode 1.

While performing a control rod pattern exchange, Rod 26-27, is found to be stuck at position 42.

The CRS enters LCO 3.1.3, Control Rod OPERABILITY, Condition A, One withdrawn control rod stuck.

The CRS is verifying that stuck control rod separation criteria are met in accordance with Required Action A.1.

Using the reference provided, which of the following combination of control rods, considered to be "slow" in accordance with LCO 3.1.4, will not violate control rod separation criteria?

- A. 26-15, 26-23 and 30-27
- B. 26-31, 22-39 and 30-39
- C. 26-23, 26-19 and 34-19
- D. 22-23, 34-19 and 30-15

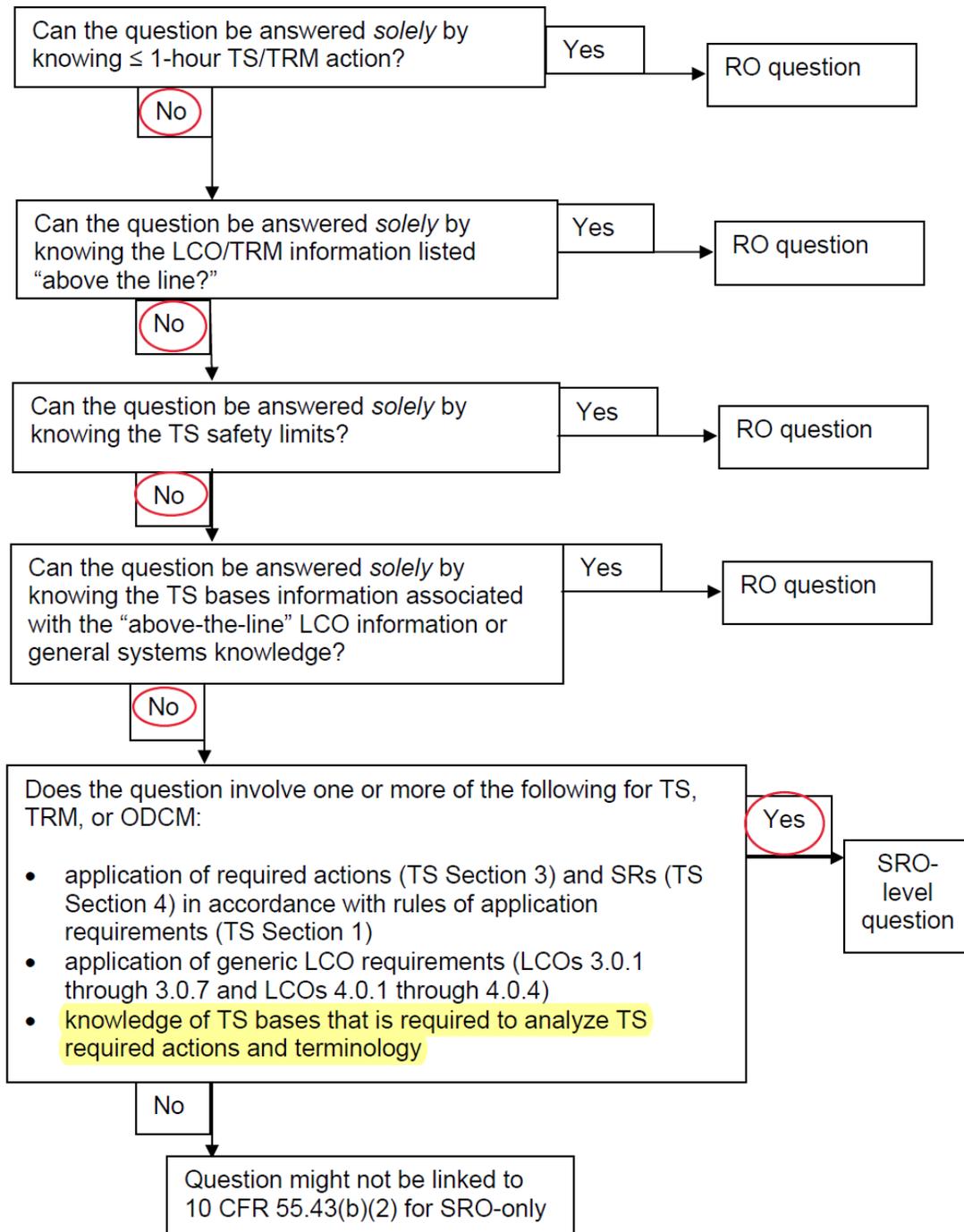
Answer: B

K/A Match:

Requires knowledge of procedure to determine stuck control rod criteria which is a reactivity management concern.

SRO Only:

Figure 4.2-2 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(2) (TS)



Explanation:

- A. Incorrect. Plausible since none of the “slow” control rods are adjacent to each other. However, since two “slow” control rods are adjacent to the stuck rod, stuck rod separation criteria is not met. See explanation for B below.
- B. Correct. TS Bases for LCO 3.1.4 describes conditions where the stuck control rod separation criteria are not met: *“The separation criteria are not met if: a) the stuck control rod occupies a location adjacent to two “slow” control rods, b) the stuck control rod occupies a location adjacent to one “slow” control rod, and the one “slow” control rod is also adjacent to another “slow” control rod, or c) if the stuck control rod occupies a location adjacent to one “slow” control rod when there is another pair of “slow” control rods adjacent to one another.”*
- Although the configuration of “slow” control rods does not meet the requirements of LCO 3.1.4, Control Rod Scram Times, with no “slow” control rods adjacent to the stuck control rod, stuck control rod separation criteria are met.
- C. Incorrect. Plausible since only one “slow” control rod is adjacent to the stuck control rod. However, since two “slow” control rods are adjacent to each other, and one of these control rod is adjacent to the stuck control rod, stuck control rod separation criteria are not met. See explanation for B above.
- D. Incorrect. Plausible since only one “slow” control rod is adjacent to the stuck control rod. However, since two “slow” control rods are adjacent to each other, and one “slow” control rod is adjacent to the stuck control rod, stuck control rod separation criteria are not met. See explanation for B above.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
TS LCO 3.1.3, Control Rod Operability	
TS Bases for 3.1.3, Control Rod Operability	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: Control rod placement in the core.

Learning Objective: 9512 - State the rules of interpretation for Technical Specifications.

Question Source: Bank #: 2019 NRC Exam SRO-93
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: 2019

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Question: SRO-95

Justification for Cognitive Level

Requires candidate to synthesize knowledge of the Technical Specification bases with an understanding of how to apply this knowledge to the physical arrangement of the control rods.

10 CFR Part 55 Content: 55.43 2 _____

Comments / Reference: TS LCO 3.1.3 Bases	Rev:	Major: 73	Minor: N/A
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Control Rod OPERABILITY
B 3.1.3

BASES

APPLICABILITY In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

ACTIONS The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

A.1, A.2, A.3, and A.4

A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note that allows the rod worth minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The separation criteria are not met if: a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) if the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods adjacent to one another. The description of "slow" control rods is provided in LCO 3.1.4, "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed within 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable amount of time to perform the Required Action in an orderly manner. The control rod must be isolated from both scram and normal insert and withdraw pressure. Isolating the control rod from scram and normal insert and withdraw pressure prevents damage to the CRDM. The control rod

Comments / Reference: Control rod placement in core										Rev:	Major:	N/A	Minor:	N/A
				18-59	22-59	26-59	30-59	34-59	38-59	42-59				
		14-55	18-55	22-55	26-55	30-55	34-55	38-55	42-55	46-55				
			A								B			
	10-51	14-51	18-51	22-51	26-51	30-51	34-51	38-51	42-51	46-51	50-51			
	06-47	10-47	14-47	18-47	22-47	26-47	30-47	34-47	38-47	42-47	46-47	50-47	54-47	
			A						B					
02-43	06-43	10-43	14-43	18-43	22-43	26-43	30-43	34-43	38-43	42-43	46-43	50-43	54-43	58-43
02-39	06-39	10-39	14-39	18-39	22-39	26-39	30-39	34-39	38-39	42-39	46-39	50-39	54-39	58-39
02-35	06-35	10-35	14-35	18-35	22-35	26-35	30-35	34-35	38-35	42-35	46-35	50-35	54-35	58-35
02-31	06-31	10-31	14-31	18-31	22-31	26-31	30-31	34-31	38-31	42-31	46-31	50-31	54-31	58-31
02-27	06-27	10-27	14-27	18-27	22-27	26-27	30-27	34-27	38-27	42-27	46-27	50-27	54-27	58-27
02-23	06-23	10-23	14-23	18-23	22-23	26-23	30-23	34-23	38-23	42-23	46-23	50-23	54-23	58-23
02-19	06-19	10-19	14-19	18-19	22-19	26-19	30-19	34-19	38-19	42-19	46-19	50-19	54-19	58-19
	06-15	10-15	14-15	18-15	22-15	26-15	30-15	34-15	38-15	42-15	46-15	50-15	54-15	
NO ROD SEL		10-11	14-11	18-11	22-11	26-11	30-11	34-11	38-11	42-11	46-11	50-11		
			14-07	18-07	22-07	26-07	30-07	34-07	38-07	42-07	46-07			
				18-03	22-03	26-03	30-03	34-03	38-03	42-03				

Question: SRO-96

Examination Outline Cross-reference: 96

Revision: 0

Date: 4/28/22

Tier: 3

Group: N/A

K/A Number: 2.2.5

Level of Difficulty: 2

SRO Importance Rating: 3.2

K/A Description: Knowledge of the process for making design or operating changes to the facility, such as 10 CFR 50.59, "Changes, Tests and Experiments," screening and evaluation processes, administrative processes for temporary modifications, disabling annunciators, or installation of temporary equipment

CGS is in Mode 1.

A control room annunciator is cycling at the setpoint.

The crew wants to disable the annunciator.

What action is required prior to disabling this nuisance annunciator?

- A. Perform a 50.59/72.48 evaluation.
- B. Place a Caution Tag on the annunciator.
- C. Notify the Crew Assistant Operations Manager (AOM-C).
- D. Write a work request to adjust the annunciator setpoint.

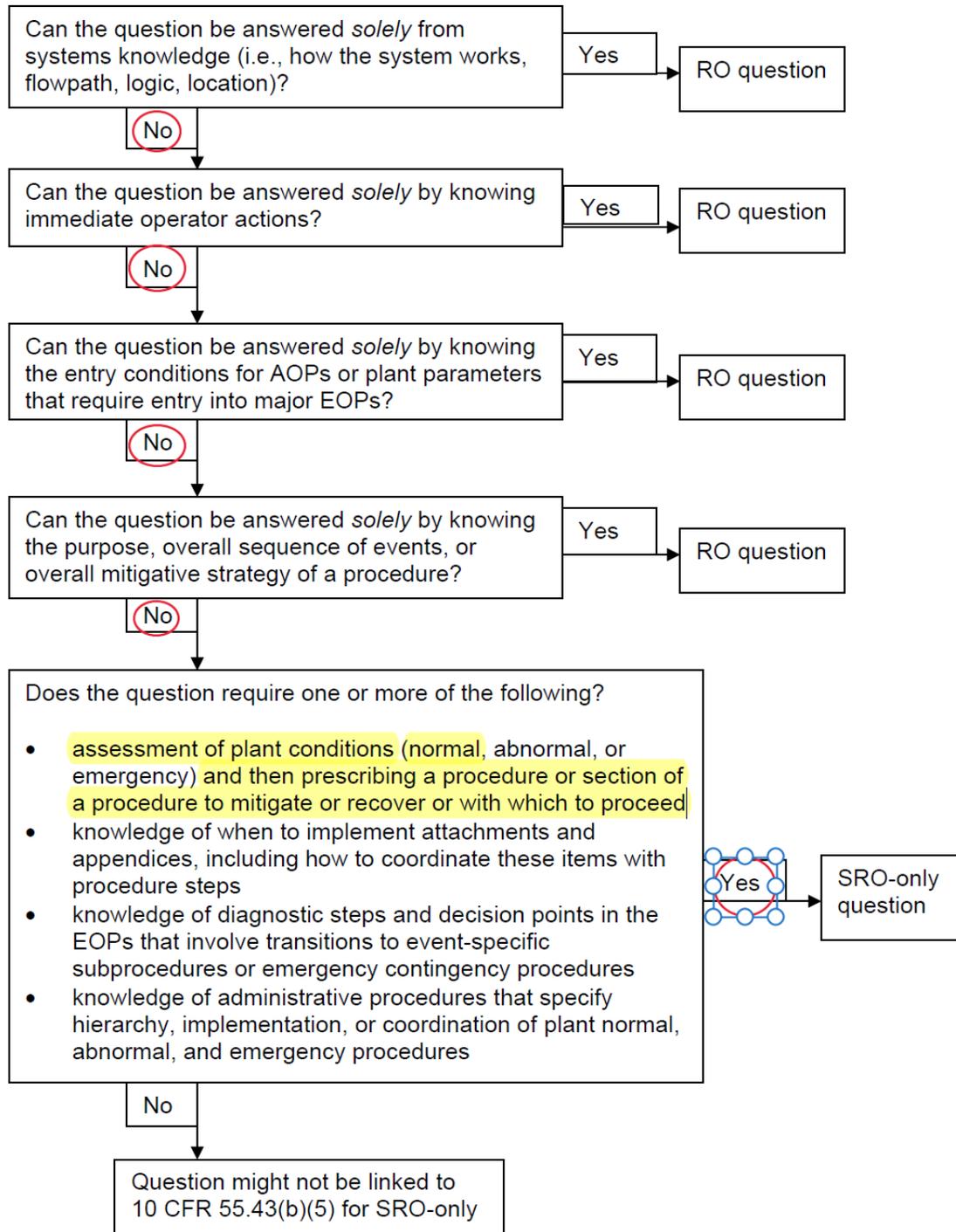
Answer: A

K/A Match:

Requires knowledge of the process for disabling annunciators.

SRO Only:

**Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)**



Explanation:

- A. Correct. In accordance with PPM 1.3.1, Operating Policies, Programs and Practices, section 4.13.7, a 50.59/72.48 evaluation is required prior to disabling the annunciator.
- B. Incorrect. Plausible since a Caution Tag should be placed on the annunciator if it will be disabled greater than one shift. However, this is not required prior to disabling the annunciator
- C. Incorrect. Plausible since AOM-C should be informed if the annunciator will be disabled greater than one shift. However, this is not required prior to disabling the annunciator
- D. Incorrect. Plausible since a work request should be written if an annunciator is disabled for greater than one shift due to inoperable source inputs. However, this is not required prior to disabling the annunciator due to setpoint shift.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 1.3.1, Operating Policies, Programs and Practices	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 7845 - Identify the actions required for 'nuisance alarms' in the control room.

Question Source: Bank #: 43940
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: 2017

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires knowledge of actions required prior to disabling an annunciator

10 CFR Part 55 Content: 55.43 3

Comments / Reference: Enter Ref # or comment.	Rev:	Major: Maj	Minor: Min
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Number: 1.3.1	Use Category: INFORMATION	Major Rev: 133
Title: Operating Policies, Programs and Practices		Minor Rev: N/A
		Page: 48 of 103

4.13.7 Disabling a valid annunciator, cycling at setpoint:

a. Check redundant indication.

CAUTION

If an annunciator is a valid alarm, cycling at or near setpoint, then a separate 50.59/72.48 is required to be performed prior to disabling the annunciator.

- b. IF the annunciator is being disabled for other than active maintenance, THEN perform a 50.59/72.48.
- c. IF the annunciator is Fire Protection related, THEN evaluate the need to initiate a fire impairment.
- d. IF applicable, THEN revise the applicable annunciator procedure to reference the 50.59 and specific conditions for disabling the annunciator.
- e. Obtain permission from the Shift Manager or above, prior to disabling the annunciator. {AR-2.82}
- f. Disable the annunciator.
- g. Log the disabled annunciator in the Control Room Log. {AR-2.82}
- h. Establish a temporary log to monitor the redundant indication at a frequency determined by the CRS.
- i. Initiate a Condition Report.
- j. IF the annunciator is disabled GT one shift, THEN perform the following:
- 1) Notify the Assistant Operations Manager of the disabled annunciator.
 - 2) Place a Caution Clearance containing the following information on the disabled annunciator: {R-2.22}, {AR-2.82}
 - The annunciator taken out of service.
 - Why the annunciator was taken out of service.
 - What actions or plant conditions are required to recover the annunciator and any compensatory actions being taken.
 - When the annunciator is expected to be returned to service.

Number: 1.3.1	Use Category: INFORMATION	Major Rev: 133 Minor Rev: N/A Page: 49 of 103
Title: Operating Policies, Programs and Practices		

4.13.8 **Annunciators disabled due to inoperable source inputs**

NOTE: An annunciator may be disabled if all the inputs to the annunciator have failed, causing the annunciator to alarm. If some of the inputs are still operable, then alternate methods should be used to disable the failed input(s).

- a. Obtain permission from the Shift Manager or above, prior to disabling the annunciator. {AR-2.82}
- b. Disable the annunciator.
- c. Log the disabled annunciator and reason disabled in the Control Room Log. {AR-2.82}
- d. Establish a temporary log to monitor the redundant indication at a frequency determined by the CRS.
- e. **IF the annunciator is disabled GT one shift, THEN perform the following:**
 - 1) Initiate a Condition Report.
 - 2) **Initiate a Work Request.**

Examination Outline Cross-reference: 97

Revision: 0

Date: 4/28/22

Tier: 3

Group: N/A

K/A Number: 2.2.19

SRO Importance Rating: 3.4

Level of Difficulty: 4

K/A Description: Knowledge of maintenance work order requirements

CGS is in Mode 1.

An event requires maintenance to be performed.

- The work will be performed in 7 days.

The Production SRO (PSRO) is reviewing the associated work package for risk in accordance with PPM 1.3.76, Integrated Risk Management.

The activity requires entry into a 12 hour Technical Specification LCO, which requires a plant shutdown if not exited.

How should the PSRO classify the activity for risk?

The activity should be classified as (1) Risk. A documented risk mitigation plan (2) required.

- A. (1) Medium
(2) is
- B. (1) Medium
(2) is NOT
- C. (1) High
(2) is
- D. (1) High
(2) is NOT

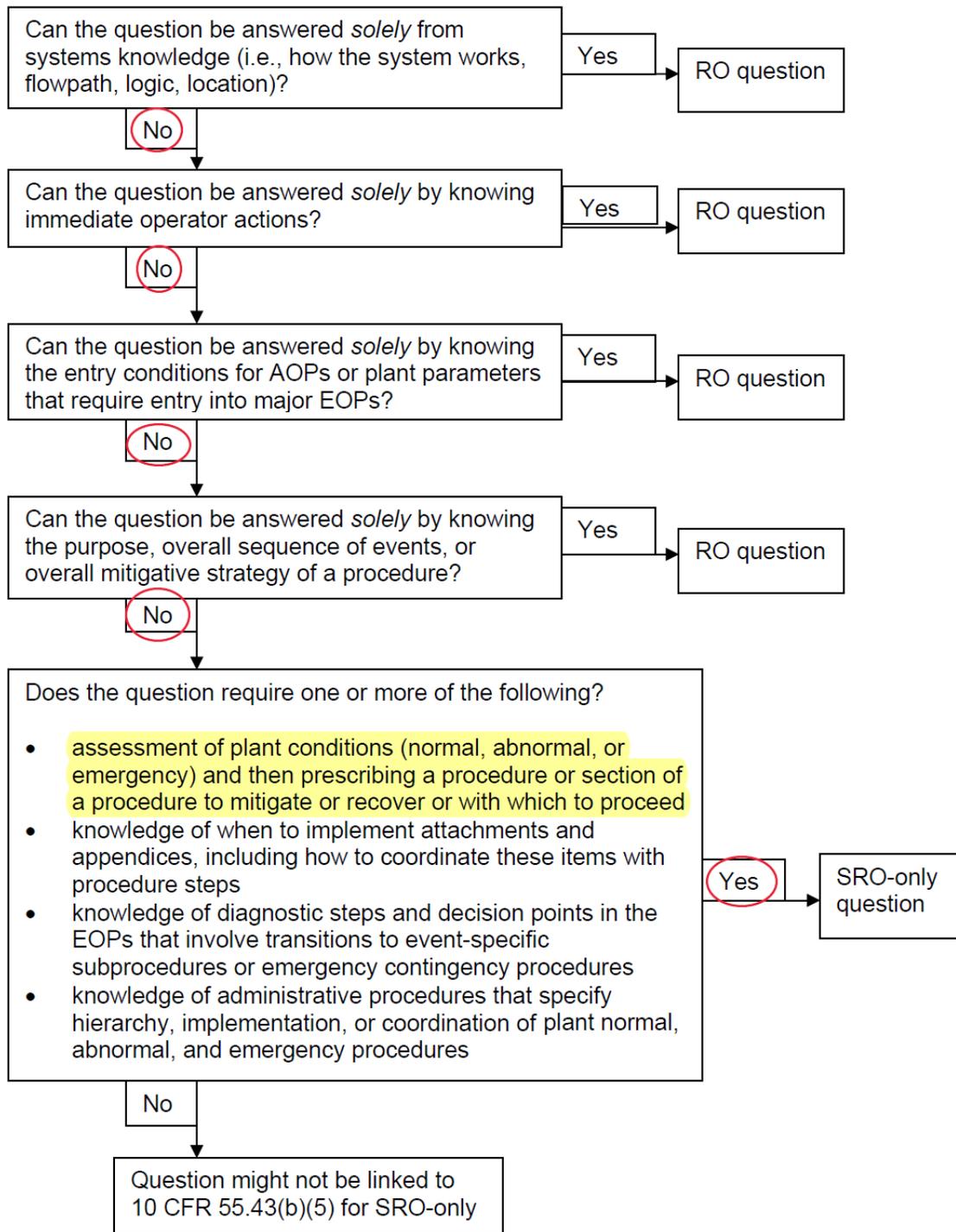
Answer: C

K/A Match:

Knowledge of process for evaluating a work order for risk.

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Explanation:

A. Incorrect. Plausible since entering a TS shutdown LCO ≤ 72 hours is a Medium Risk activity. Plausibility is enhanced since (2) is correct. However, in accordance with PPM 1.3.76, Integrated

Risk Management, Attachment 9.1, Risk Assessment Worksheet, Figure 2, question 15, entry into a 12 hour or less TS shutdown LCO, that is not associated with an activity with a scheduled frequency of annual or less, requires classification of “High Risk”.

- B. Incorrect. Plausible since entering a TS shutdown LCO ≤ 72 hours is a Medium Risk activity. Plausibility is enhanced since a documented plan is not required for a Medium Risk activity. However, in accordance with PPM 1.3.76, Integrated Risk Management, Attachment 9.1, Risk Assessment Worksheet, Figure 2, question 15, entry into a 12 hour or less TS shutdown LCO, that is not associated with an activity with a scheduled frequency of annual or less, requires classification of “High Risk”.
- C. Correct. in accordance with PPM 1.3.76, Integrated Risk Management, Attachment 9.1, Risk Assessment Worksheet, Figure 2, question 15, entry into a 12 hour or less TS shutdown LCO, that is not associated with an activity with a scheduled frequency of annual or less, requires classification of “High Risk”. Additionally, in accordance with PPM 1.3.76, section 5.3.3 requires generating a high risk work plan.
- D. Incorrect. Plausible since (1) is correct. However, PPM 1.3.76, section 5.3.3 requires generating a high risk work plan.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 1.3.76, Integrated Risk Management	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 11253 – Knowledge of maintenance work order requirements.

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize an understanding of activities that are considered “High Risk” with a knowledge of actions required for “High Risk” activities.

10 CFR Part 55 Content: 55.43 5

Comments / Reference: PPM 1.3.76	Rev:	Major: 061	Minor: N/A
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Number: 1.3.76	Use Category: INFORMATION	Major Rev: 061
Title: Integrated Risk Management		Minor Rev: N/A
		Page: 25 of 66

5.3 Managing and Approving Actions for HIGH Risk Work

NOTE: Section 5.3 may be waived as appropriate for Repetitive High Risk Work or for Emergent High Risk Work. For Repetitive High Risk Work, Section 5.4 may be used. For Emergent High Risk Work, Section 5.5 may be used.

NOTE: Actions other than those recommended may be supplemented to manage and minimize risk at the discretion of the responsible persons, to provide assurance that risk is adequately managed.

NOTE: When considering mitigating actions, one should ask the question, "What's the worst that can happen?" to ensure we are minimizing the risk associated with work.

- 5.3.1 **High Risk Work requires the use of a High Risk Work Plan (Form 26871).**
- 5.3.2 For High Risk Work, the Responsible Manager (RM) will ensure the following:
 - a. Assigns a Designated Point of Contact (DPC) or Critical Maintenance Project Manager (CMPM) The individual chosen should have technical expertise for the HIGH risk Work being managed.
 - b. If the work is an IPTE, contacts the PGM to designate a Senior Line Manager to provide oversight for the evolution. {OE-7.19}
 - c. Consideration should be given to scheduling or rescheduling work to a time that reduces the risk of the work activity.
- 5.3.3 The DPC/CMPM will perform the following:
 - a. **Generate the HIGH risk work plan per Form 26871.** Depending on the scope of the work, the DPC may need to engage additional departments and personnel.

5.2 Managing and Approving Actions for LOW and MEDIUM Risk Work

NOTE: Section 5.3 specifies approval requirements associated with Managing and Approving Actions for HIGH Risk Work.

NOTE: Actions other than those recommended may be supplemented to manage and minimize risk at the discretion of the responsible persons, to provide assurance that risk is adequately managed.

NOTE: When considering mitigating actions, one should ask the question, "What's the worst that can happen?" to ensure we are minimizing the risk associated with work.

5.2.1 Low Risk Work - LOW risk work activities do not require a documented action plan.

- a. Planner should incorporate any prudent risk mitigation actions in the work instructions or in the impact statement.
- b. PSRO should incorporate any prudent risk mitigation actions in the work instructions or in the impact statement.
- c. ALARA planner should incorporate any prudent risk mitigation actions in the work instructions or in the impact statement.

5.2.2 Medium Risk Work - Requires heightened awareness to prevent human error. MEDIUM risk Work activities do not require a documented action plan. MEDIUM risk mitigation actions are located in Attachment 9.2.

FIGURE 2 RISK ASSESSMENT WORKSHEET FOR HIGH RISK

<p>6. Plant maneuvering to remove or return the equipment to service before or after the Work activity, if performed incorrectly, would significantly increase the possibility of a plant transient and a peer check is not performed during the activity. Transient is defined as:</p> <ul style="list-style-type: none"> • Transfer of electrical buses • A change to reactor power. • An unplanned start/stop or actuation of trip sensitive, ESF, or major power train equipment. 	YES / NO
<p>7. Is the activity on redundant instrument trains or equipment (e.g. RPS, ESF) that cannot be safely bypassed? Answer NO if the activity is controlled by station procedure activities which have scheduled frequency of annual or less. (e.g., surveillance testing)</p>	YES / NO
<p>8. Does the activity remove equipment from service that upon a plant trip, with no additional failures, would cause an unplanned scram with complications?</p>	YES / NO
<p>9. Does the activity require extended low-power operations (activities that require maintaining LT 60% power for several hours)</p>	YES / NO
<p>10. Does the maintenance activity place an SSC in a configuration that would result in a unit trip or transient upon coincidental failure of a redundant system, train, or channel, e.g. RPS, ESF channel tripped and not bypassed, CRD or Stator cooling water pump removed from service? Reference SPV list for examples. Answer NO if the activity is controlled by station procedure activities which have scheduled frequency of annual or less. (e.g. surveillance testing) SOPs are exceptions to the annual requirement.</p>	YES / NO
<p>11. Does the activity energize or de-energize 125V DC or vital and instrument AC loads, such that additional unplanned loss of power from the next upstream electrical device (e.g., breaker/fuse) could result in a plant scram unless pre-planned actions are in place to mitigate a plant scram or transient?</p>	YES / NO
<p>12. Could the activity result in a loss of a, 4160 V, 6900 V, 480 V, MCC, or other load center and the loss of bus would result in either a reactor scram, entry into a 12 hour or less action statement or require a power reduction.</p>	YES / NO
<p>13. Will the activity result in a single main generator output breaker alignment?</p>	YES / NO
<p>14. Activity is a DG Extended Maintenance Outage.</p>	YES / NO
<p>15. Does the activity require entry into a 12 hour or less tech spec LCO, which requires a unit shutdown if not exited? Answer NO if the tech spec entry is due to station procedure activities which have scheduled frequency of annual or less. (e.g. surveillance testing and SOP directed activities that do not include maintenance)</p>	YES / NO

Question: SRO-98

Examination Outline Cross-reference: 98

Revision: 0

Date: 4/28/22

Tier: 3

Group: N/A

K/A Number: 2.3.6

Level of Difficulty: 3

SRO Importance Rating: 3.8

K/A Description: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures, or analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only)

An event occurs that requires staffing the Emergency Response Centers.

The emergency response staff is considering the use of Potassium Iodide (KI).

How is the use of KI by Energy Northwest emergency workers administered?

The use of KI should be authorized once projected thyroid CDE is GE (1). Once authorized, the use of KI by Energy Northwest workers within the Protected Area is (2).

- A. (1) 5 rem
(2) mandatory
- B. (1) 5 rem
(2) voluntary
- C. (1) 25 rem
(2) mandatory
- D. (1) 25 rem
(2) voluntary

Answer: D

K/A Match:

Requires analysis and interpretation of radiological conditions and determination of actions based on the analysis.

SRO Only:

Another area of SRO-level knowledge (with respect to selecting a procedure) is knowledge of decision points within the facility's emergency plan. These include emergency action level declarations and protective action recommendations since these would subsequently lead to the implementation of various subprocedures within the emergency plan.

Explanation:

- A. Incorrect. Plausible since 5 rem is the legal limit for exposure. Plausibility is enhanced since some actions are mandatory based on potential dose, such as evacuations, respiratory protection, etc. However, KI is recommended when CDE is GE 25 rem. The use of KI is voluntary.
- B. Incorrect. Plausible since (2) is correct. Plausibility is enhanced since (1) is the legal limit for exposure. However, KI is recommended when CDE is GE 25 rem.
- C. Incorrect. Plausible since (1) is correct. Plausibility is enhanced since some actions are mandatory based on potential dose, such as evacuations, respiratory protection, etc. However, the use of KI is voluntary.

Correct. IN accordance with PPM 13.2.1, Emergency Exposure Levels / Protective Action Guides, Attachment 7.3, Guidance for Administering Potassium Iodide (KI), the EOF Manager should recommend the use of KI if the projected thyroid CDE is 25 rem or more. Additionally the use of KI is strictly voluntary.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 13.2.1, Emergency Exposure Levels / Protective Action Guides	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: 19350 – State the conditions that require recommending that emergency workers in the protected area take KI (SRO Only).

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

Question History: Last NRC Exam: N/A

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

Justification for Cognitive Level

Requires candidate to synthesize a knowledge of the dose level where the use of KI should be recommended with an understanding that the use of KI is voluntary.

10 CFR Part 55 Content: 55.43 1

Comments / Reference: PPM 13.2.1	Rev:	Major:022	Minor:002
<p>2. The TSC Manager, or the Shift Manager acting as the Emergency Director, in consultation with the Radiation Protection Manager, should recommend that emergency workers in any affected Protected Area location take KI:</p> <p>If the projected thyroid CDE is 25 rem or more based on local air sampling and / or plant conditions. (See note.) 25 rem CDE is 1000 DAC hours and is equivalent to an intake of approximately 25 <u>microCuries</u> of Iodine-131.</p> <p></p> <div data-bbox="246 993 1367 1052" style="border: 1px solid black; padding: 5px;"><p>NOTE: The use of potassium iodide is strictly voluntary.</p></div> <p>KI Issuance Instructions</p>			

Question: SRO-99

Examination Outline Cross-reference: 99

Revision: 0

Date: 4/28/22

Tier: 3

Group: N/A

K/A Number: 2.4.16

Level of Difficulty: 2

SRO Importance Rating: 4.4

K/A Description: Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, or severe accident management guidelines

CGS is in Mode 1.

An event causes the crew to enter an Abnormal Procedure (ABN).

The event degrades to the point where the CRS directs a manual reactor scram and enters PPM 3.3.1, Reactor Scram and PPM 5.1.1, RPV Control, prior to completing ABN actions.

When may the crew continue execution of steps in the ABN?

The CRS may direct continuing ABN actions...

- A. concurrently with the EOP ONLY IF specifically called out by the EOP.
- B. concurrently with the EOP ONLY IF ABN actions do not conflict with EOP actions.
- C. ONLY AFTER the shift manager has determined that an emergency no longer exists.
- D. ONLY AFTER applicable steps of the reactor scram procedure have been completed.

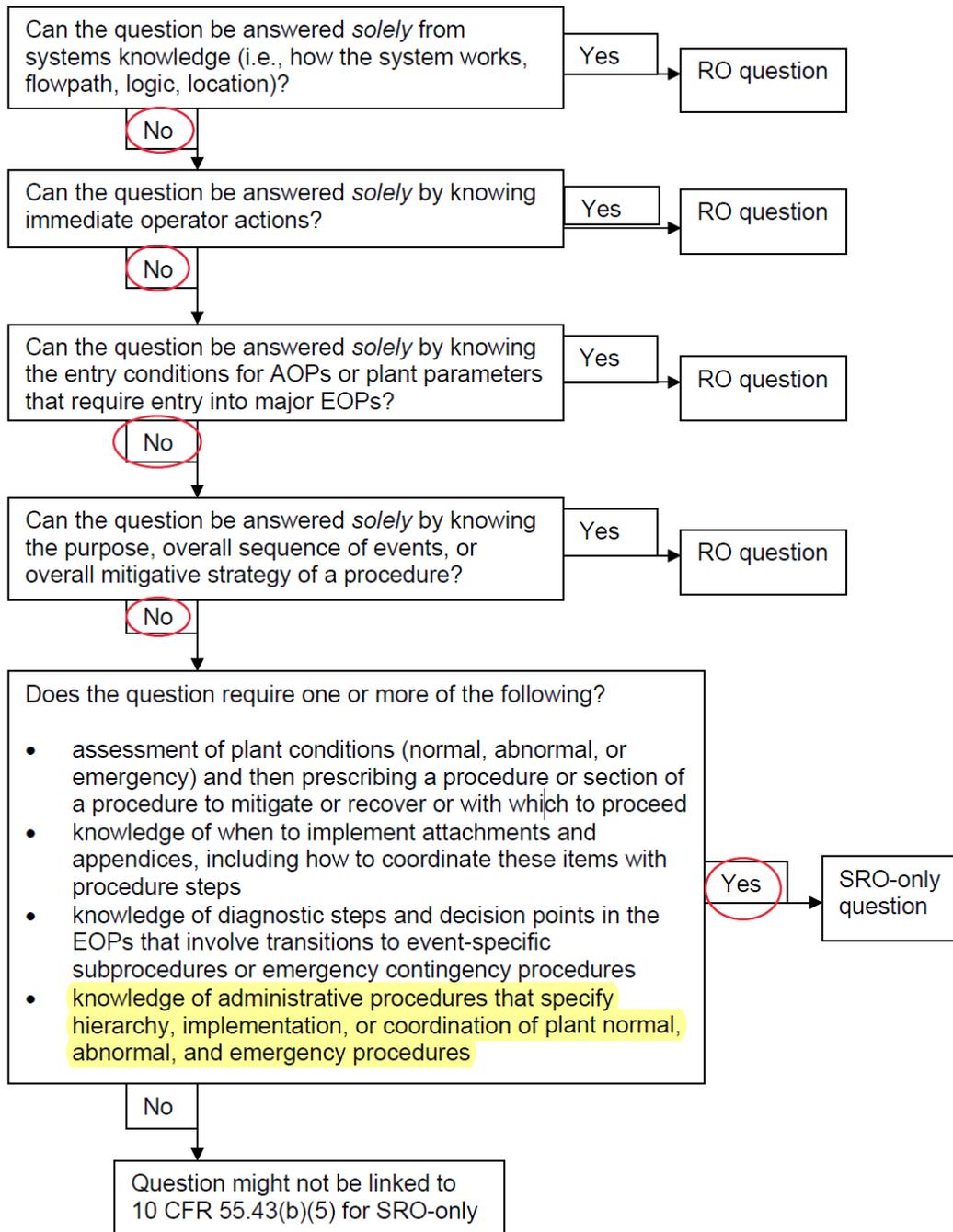
Answer: B

K/A Match:

Requires knowledge of requirements for coordination actions from abnormal procedures and emergency procedures.

SRO Only:

Figure 4.2-3 Screening for SRO-Only Questions Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Explanation:

- A. Incorrect. Plausible since Ref. A, section 4.8.3.a states that “All required concurrent execution of any Volume 2, 3, or 4 Procedure are specifically called out by the EOPs”. However, the section continues by stating “This does not mean that a Volume 4, Abnormal Procedure, cannot otherwise be concurrently executed with the EOPs so long as its specified actions do not conflict with the direction given by the EOPs”. Therefore, ABN actions may be performed concurrently with EOP actions without being specifically referenced in the EOP.
- B. Correct. As delineated in Ref. A, section 4.8.3.a, Abnormal procedure (ABN) actions may be executed concurrently with Emergency procedure (EOP) actions as long as the ABN actions “do not conflict with the direction given by the EOPs”.
- C. Incorrect. Plausible since Ref. A, section 4.8.1.a states “The Volume 5 Emergency Operating Procedures (EOPs) and the actions specified therein have priority/precedence over all Volume 2, 3, and 4 Procedures when an emergency exists (EOP entry condition(s) is/are met)”. This statement implies that once an EOP is entered, actions from lower tier procedures are stopped. Additionally, Ref. A, section 4.8.2.e states “EOPs are exited only if the Shift Manager determines that an emergency no longer exists and directs EOP exit or the EOPs direct exit to appropriate plant procedures”. This distractor infers that the EOPs must be exited prior to completing ABN actions. This is not the case, however. See explanation for answer ‘B’ above.
- D. Incorrect. Plausible since Ref. A, section 4.8.1.b states “During a transient, and as plant conditions continue to degrade, the flow path of procedure usage is from the Volume 2 and 3 Procedures to the Volume 4, Abnormal Procedures, and then to the EOPs”. This infers that actions from a Volume 3, General Operating Procedure, such as PPM 3.3.1, Reactor Scram, should be completed prior to completing actions from a Volume 4 procedure, such as ABNs. This is not the case. See the explanation for answers ‘B’ and ‘C’ above.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 1.3.1	
Tech Ref 2	
Tech Ref 3	
Tech Ref 4	

Proposed references to be provided during examination: N/A

Learning Objective: Learning Objective.

Question Source: Bank #: 43915
 Modified Bank #: Mod Bank #. (Note changes or attach parent)
 New

Question History: Last NRC Exam: 2017

Question: SRO-99

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to understand coordination between abnormal procedure actions and emergency procedure actions.

10 CFR Part 55 Content: 55.43 5

Comments / Reference: PPM 1.3.1	Rev:	Major: 133	Minor: N/A
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Number: 1.3.1	Use Category: INFORMATION	Major Rev: 133
Title: Operating Policies, Programs and Practices		Minor Rev: N/A Page: 23 of 103

4.8 Procedure Usage

NOTE: Use and adherence to approved Plant Procedures is controlled by SWP-PRO-01.

4.8.1 Procedure Hierarchy

- a. The Volume 5 Emergency Operating Procedures (EOPs) and the actions specified therein have priority/precedence over all Volume 2, 3, and 4 Procedures when an emergency exists (EOP entry condition(s) is/are met).
- b. During a transient, and as plant conditions continue to degrade, the flow path of procedure usage is from the Volume 2 and 3 Procedures to the Volume 4, Abnormal Procedures, and then to the EOPs.

4.8.2 Volume 5: Emergency Operating Procedures (EOPs)

NOTE: EOPs are symptom oriented rather than event based procedures. Operators need not understand the event in order to successfully respond using symptom based EOPs.

- a. Addresses other than normal conditions which, if not corrected, could result in core damage, loss of containment integrity or an uncontrolled radioactivity release to the environment.
- b. These procedures provide the necessary direction to shut down and cool down the reactor during emergencies and should be entered whenever any EOP entry condition exists.
- c. If following entry into an EOP, an additional entry condition is met, the EOP should be reentered at the beginning.
- d. Operator Actions are given in a logical sequence and should be performed in order, if possible.
 - Conditional WHEN/THEN statements act as stop signs and subsequent actions do not have to be performed until the condition has been met
 - Conditional IF/THEN statements act as stop signs and the condition should be determined before proceeding
 - Command steps may be performed in parallel, when appropriate
 - Overrides should be frequently checked since they become effective when read and remain effective in the flow chart area identified.
- e. EOPs are exited only if the Shift Manager determines that an emergency no longer exists and directs EOP exit or the EOPs direct exit to appropriate plant procedures

4.8.3 Procedure Usage during an Emergency

- a. The EOPs direct the operator to execute a particular section or an entire Volume 2, 3 or 4 Procedure. All required concurrent execution of any Volume 2, 3, or 4 Procedure are specifically called out by the EOPs.

This does not mean that a Volume 4, Abnormal Procedure, cannot otherwise be concurrently executed with the EOPs so long as its specified actions do not conflict with the direction given by the EOPs.

- b. Once EPIPs have been entered (Emergency Classification occurs), recovery actions not specifically authorized by plant procedures which have a potential for radioactive release to the environment require Emergency Director concurrence (this includes items listed in FAZ Procedures).
- c. If it becomes necessary to take actions other than those prescribed by the License Basis Documents or by plant procedures; (to protect the immediate health and safety of the public) those actions shall be implemented using 10 CFR 50.54(x) per Section 4.3.
- d. If a transient is in progress or an emergency condition dictates, take immediate corrective action up to and including power reduction or SCRAM to stabilize the situation. Then, as soon as possible, refer to appropriate procedures. Appropriate times to exercise this option include:
- To maintain the safety of the public
 - To protect the safety of individuals (e.g., deenergize equipment that is causing an electrocution or stop a leak that is endangering an individual)
 - To preclude unnecessary equipment damage (e.g., stop a piece of equipment that is quickly degrading or is in imminent danger)
 - SCRAM prior to exceeding an RPS limit

Examination Outline Cross-reference: 100

Revision: 0

Date: 4/25/22

Tier: 3

Group: N/A

K/A Number: 2.4.52

Level of Difficulty: 2

SRO Importance Rating: 4.0

K/A Description: Knowledge of the lines of authority during implementation of the emergency plan, emergency plan implementing procedures, emergency operating procedures, or severe accident guidelines

CGS is in Mode 1.

An "ALERT" Emergency Action Level (EAL) has been declared due to an event.

- The Emergency Response Centers required for this event have been activated and all turnovers are complete.

Who is responsible for completing the Emergency Classification for any subsequent event?

- A. Shift Manager
- B. TSC Manager
- C. EOF Manager
- D. TSC Operations Manager

Answer: C

K/A Match:

Requires knowledge of lines of authority for completing emergency classifications during heightened emergency postures.

SRO Only:

Question linked to a SRO-Only Learning Objective.

2. Each SRO-level question must be tied to one of the 10 CFR 55.43(b) items. If a facility licensee desires to evaluate a K/A that is not tied to one of the 10 CFR 55.43(b) items, then the licensee must classify the K/A as "unique to the SRO position" and provide documentation that ties the K/A to the facility licensee's SRO job position duties in accordance with the systematic approach to training, as in the following examples:
 - The question is linked to a learning objective that is specifically labeled in the lesson plan as being SRO-only (e.g., some facility licensee lesson plans have columns in the margin that differentiate auxiliary operator, RO, and SRO learning objectives).

Explanation:

- A. Incorrect. Plausible since the shift manager is responsible for emergency classifications during normal operation and at an EAL of Unusual Event. However, when the EAL is upgraded to "Alert" or higher, Emergency Centers are activated and the EOF Manager assumes Emergency Director (ED) duties and completes any subsequent EAL classifications.
- B. Incorrect. Plausible since the TSC Manager may assume ED duties prior to the EOF activating. However, the question stem states that all emergency centers are active, which includes the EOF. Therefore, the EOF manager assumes the duties of the ED, including the responsibility for completing emergency classifications.
- C. Correct. In accordance with Form 26531, EOF Manager Checklist, once the EOF is activated, the EOF assumes the duties of the ED, which includes the responsibility for emergency classification.
- D. Incorrect. Plausible since the TSC Operations Manager is responsible to assist in making emergency classifications. However, this position does not assume the duties of the ED and therefore, does not have the responsibility for completing emergency classifications.

Technical Reference(s)	Attached w/ Revision # See Comments / Reference
PPM 13.11.1, EOF Manager Duties	
Form 26531, EOF Manager Checklist	
Form 26512, TSC Operations Manager Checklist	
Form 26506, TSC Manager Checklist	

Proposed references to be provided during examination: N/A

Learning Objective: 6132 - Identify the persons who may assume the position of Emergency Director during plant emergencies. (SRO-only)

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

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge
 Comprehension or Analysis

Justification for Cognitive Level

Requires candidate to synthesize knowledge of the requirements for activating emergency centers at different EAL levels with an understanding of the person who assumes ED duties during an emergency.

10 CFR Part 55 Content: 55.43 5

Comments / Reference: PPM 13.11.1	Rev:	Major:044	Minor:005
<p>3.2 <u>Emergency Operations Facility (EOF) Manager</u></p> <p>3.2.1 Assume overall command and control of emergency response</p> <p>3.2.2 Approve requests for offsite assistance</p> <p>3.2.3 Integrate ERO activities with the ICP response activities</p> <p>3.2.4 Classify emergencies</p>			

Comments / Reference: Form 26531	Rev:	Major:9	Minor:3
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ENERGY NORTHWEST EOF MANAGER CHECKLIST

EPIP Ref: PPM 13.11.1

Position Functions/Responsibilities

- Assume overall command and control of emergency response
- Approve requests for offsite assistance
- Integrate ERO activities with the ICP response activities
- **Classify emergencies**
- Direct and approve offsite emergency notifications to state and local authorities
- Direct communications with the NRC
- Ensure flow of information within and between the Emergency Response Facilities
- Review and approve technical information for news releases
- Activate the EOF
- Ensure Implementation of extreme measures (SAGs, EDMGs, §50.54(x) or suspend security controls)
- Ensure appropriate evacuation actions for plant personnel
- Approve the issuance of KI
- Make Protective Action Recommendations to offsite authorities
- Approve emergency exposures
- Terminate the emergency event
- Establish a recovery plan and organization
- Conduct facility briefs and updates

Section 1, Initial Actions

- 1.1 Mobilization
- 1.2 EOF Activation
- 1.3 **Transfer of Command and Control (Emergency Director Duties)**

1.3	Transfer of Command and Control (Emergency Director Duties)	
1.3.1	CONTACT current ED and determine a time when conditions would permit turnover of ED duties and responsibilities.	<input type="checkbox"/>
1.3.2	<u>WHEN</u> conditions permit, <u>THEN</u> conduct a turnover using the latest completed Classification Notification Form (CNF) and form 25810, Emergency Director Turnover Sheet as a guide.	<input type="checkbox"/> Tab 17 Tab 3
1.3.3	RELIEVE the current ED of Emergency Director duties once current conditions and proposed actions are fully understood.	<input type="checkbox"/>
1.3.4	ANNOUNCE to the EOF staff that you have assumed the responsibilities of Emergency Director @ _____ hr.	<input type="checkbox"/>
1.3.5	DIRECT the EOF Information Coordinator to announce that the EOF Manager is the Emergency Director to other the emergency centers.	<input type="checkbox"/>

Comments / Reference: Form 26512	Rev:	Major:6	Minor:N/A
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ENERGY NORTHWEST
TSC OPERATIONS MANAGER CHECKLIST

EPIP Ref: PPM 13.10.5

Position Functions/Responsibilities

- **Assist with emergency classification**
- Perform offsite emergency notifications to state and local authorities
- Establish the priority of response actions
- Provide Operations support for accident detection and assessment to the TSC
- Coordinate between CR, OSC and TSC to set OSC Team task priorities
- Implements Severe Accident Guidelines (SAGs), analyzes issues and plans mitigative actions in accordance with engineering Technical Memorandum
- Provide the primary TSC communications path to the Shift Manager
- Provide input for facility briefs and updates

Comments / Reference: Form 26506	Rev:	Major:7	Minor:1
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1.4 Transfer of Command and Control (Emergency Director Duties)	
NOTE: You may assume the ED duties prior to TSC activation, but ensure you have sufficient personnel and communication links to assess accident conditions and communicate classification decisions and PARs to offsite authorities.	
1.4.1 IF the Shift Manager wishes to transfer ED duties AND the EOF Manager is NOT prepared to assume these duties THEN ASSUME ED duties from Shift Manager per Section 3.1 of this checklist.	<input type="checkbox"/>