Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Reactor Trip, Stabilization, Recovery: Ability to determine or interpret the following as they apply to a reactor trip: Reactor trip breaker position	**Tier**	**1**		
	Group	**1**		
	K/A	**007 EA2.03**		
	IR	**4.2**		

Question 1

Each EXTINGUISHED phase current light on Control Room Board 5 (B05) indicates a MINIMUM of ___(1)___ RTCB(s) is(are) open, and a MINIMUM of ___(2)___ phase current light(s) must be extinguished in order for the Reactor to trip.

A. (1) 1
   (2) 1

B. (1) 1
   (2) 2

C. (1) 2
   (2) 1

D. (1) 2
   (2) 2

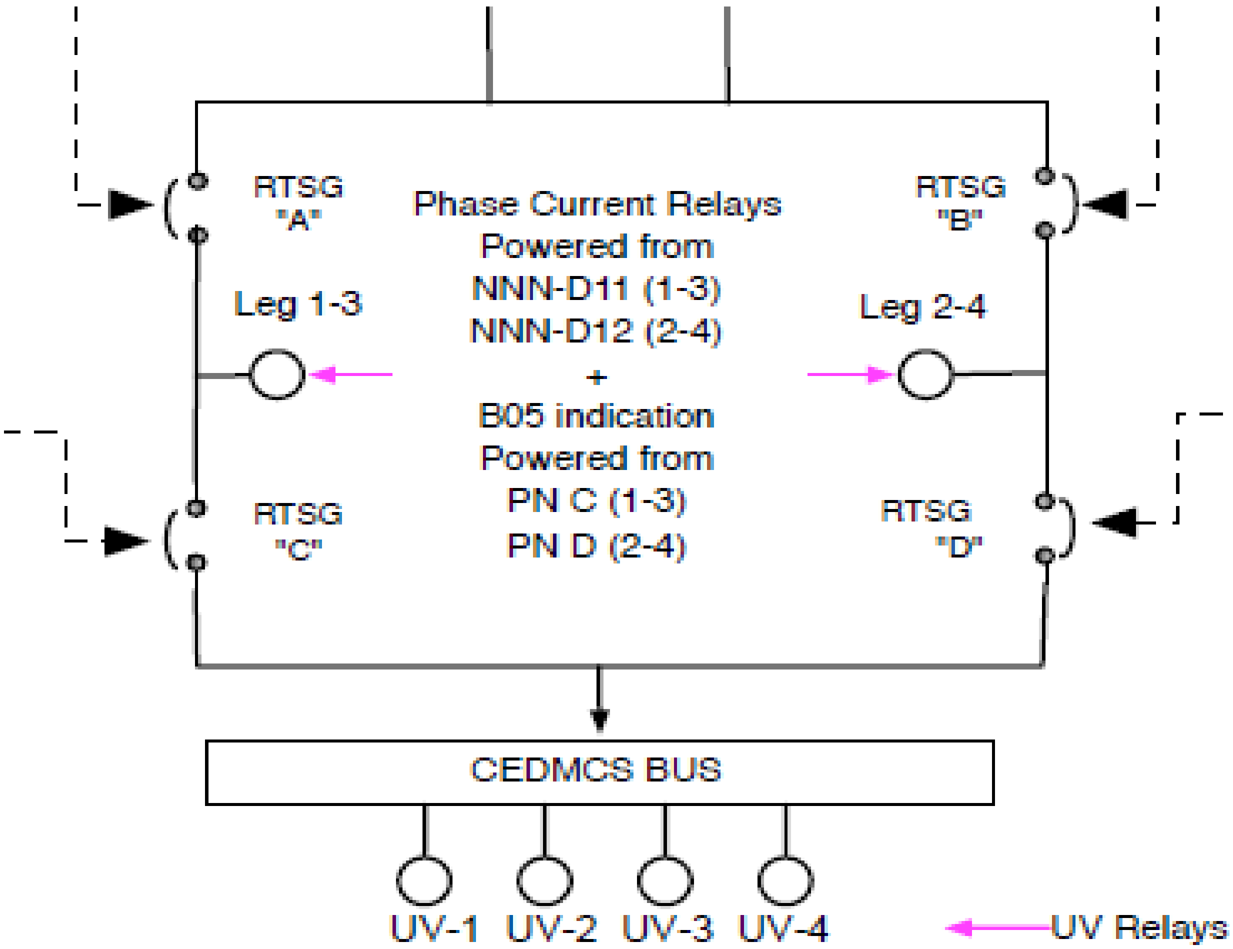
Proposed Answer:	B

Explanations:	
A.	First part is correct. The second part is plausible because to trip the Reactor a minimum of 2 breakers need to be opened. A combination of ‘A’ or ‘C’ AND ‘B’ or ‘D’ need to be opened to trip the Reactor. If ‘A’ and ‘C’ or ‘B’ and ‘D’ were the only 2 breakers open, only 1 phase current light would be extinguished and the Reactor wouldn’t trip.
B.	Correct
C.	First part is plausible if it is thought that there will still be a path for current to the phase current light and only by opening both breakers is power completely isolated to the light. The second part is plausible because to trip the Reactor a minimum of 2 breakers need to be opened. A combination of ‘A’ or ‘C’ AND ‘B’ or ‘D’ need to be opened to trip the Reactor. If ‘A’ and ‘C’ or ‘B’ and ‘D’ were the only 2 breakers open, only 1 phase current light would be extinguished and the Reactor wouldn’t trip.
D.	First part is plausible if it is thought that there will still be a path for current to the phase current light and only by opening both breakers is power completely isolated to the light. Second part is correct.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**Plant Protection System LP #NKASYC14907 – Describe how RTCBs are tripped and what indication or trip path status is available**	


RTSG
"A"
Phase Current Relays
Powered from
NNN-D11 (1-3)
NNN-D12 (2-4)
+
B05 indication
Powered from
PN C (1-3)
PN D (2-4)
RTSG
"B"
Leg 1-3
Leg 2-4
RTSG
"C"
RTSG
"D"
CEDMCS BUS
UV-1
UV-2
UV-3
UV-4
UV Relays

Technical Reference:	Plant Protection System Tech Manual

RTSG Phase Current Indicator

These white lamps at the bottom of the channel C and D ROMs are normally on to indicate current flow to the CEDMs. If power through one leg of the RTSG is interrupted by RTSG opening, the appropriate lamps will go out (RTSGs 1 and/or 3 for the phase current lamp on the channel C ROM: RTSGs 2 and/or 4 for that on channel D).

Examination Outline Cross-Reference:	**Level**	**RO**		**SRO**
K/A: Small Break LOCA: Ability to determine or interpret the following as they apply to a small break LOCA: Adequate core cooling	**Tier**	**1**		
	Group	**1**		
	K/A	**009 EA2.39**		
	IR	**4.3**		

Question 2

Given the following conditions:

- A LOCA is in progress
- The CRS has entered 40EP-9EO03, Loss of Coolant Accident
- Containment Temperature is 150°F

Adequate Core cooling is indicated by a MINIMUM subcooling or MAXIMUM superheat of …

A. 24°F subcooled

B. 0°F

C. 44°F superheat

D. 60°F superheat

Proposed Answer:	C

Explanations:	
A.	24°F subcooled is the minimum subcooled value for adequate core cooling during an uncomplicated Reactor trip
B.	0°F is the value in which water changes from a subcooled condition to a superheated condition
C.	Correct
D.	60°F superheat is the value of adequate core cooling when the Containment is in Harsh Condition

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**14**	
Reference Provided:	**N**	
Learning Objective:	**Given conditions of LOCA, analyze Core Heat Removal to determine if the SFSC acceptance criteria is satisfied per 40EP-9EO03**	

Technical Reference:	40EP-9EO03, Loss of Coolant Accident

5. Core Heat Removal

ACCEPTANCE CRITERIA:

a. CET Subcooling indicates less than 44°F [60°F] superheat and **NOT** rising.

b. RCS Subcooling indicates less than 44°F [60°F] superheat and **NOT** rising.

Technical Reference:	40EP-9EO02, Reactor Trip

5. Core Heat Removal

ACCEPTANCE CRITERIA:

a. RCS ΔT is less than 10°F.

b. The RCS is 24°F or more subcooled.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Large Break LOCA: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material	**Tier**	**1**		
	Group	**1**		
	K/A	**011 G 2.4.47**		
	IR	**4.2**		

Question 3

Given the following conditions:

- Unit 1 tripped due to a large break LOCA
- The CRS has entered 40EP-9EO03, Loss of Coolant Accident
- RWT level is 80% and lowering at a rate of 1%/min

As RWT level continues to lower, the crew will be procedurally REQUIRED to shift Charging Pump suction to an alternate source in MAXIMUM of…

A. 7 minutes

B. 30 minutes

C. 36 minutes

D. 46 minutes

Proposed Answer:	A

Explanations:	
A.	Correct
B.	RWT level lowering to 50% will require the crew to stop 1 charging pump
C.	RWT level lowering to 44% will require the crew to stop all charging pump
D.	This value correlates to the auto makeup to the VCT setpoint

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	29739 - Using the current copy of the Standard Appendices, perform Charging Pump Alternate Suction to the SFP / Restoration, per 40EP-9EO10, Appendix 11	

Technical Reference:	

INSTRUCTIONS	CONTINGENCY ACTIONS

✱ 6. **IF** SIAS has actuated,
THEN perform the following:

a. **IF** it is determined that RWT level may lower to less than 73% during the event,
OR it is desired to align Charging Pump suction through an alternate suction path,
THEN PERFORM **ONE** of the following:

- Appendix 10, Charging Pump Alternate Suction to the RWT / Restoration
- Appendix 11, Charging Pump Alternate Suction to the SFP / Restoration

b. **IF** RWT level is above 73%,
AND it is desired to align Charging Pump suction through CHE-HV-536 or CHN-UV-514,
THEN PERFORM Appendix 103, RCS Makeup / Emergency Boration.

✱ 7. **IF** pressurizer pressure remains below the SIAS setpoint,
THEN perform the following:

a. Ensure **ONE** RCP is stopped in each loop.

b. **IF** RCS subcooling is less than 24°F [44°F],
THEN ensure all RCPs are stopped.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Reactor Coolant Pump Malfunctions: Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on high stator temperatures	Tier	1		
	Group	1		
	K/A	015 AA2.09		
	IR	3.4		

Question 4

Given the following conditions:

- Unit 3 was tripped due to a high Motor Stator temperature on 1A RCP

Per 40AO-9ZZ04, Reactor Coolant Pump Emergencies, the crew should stop ___(1)___. Stopping the RCP(s) should be performed ___(2)___.

A. (1) ALL RCPs
   (2) BEFORE the Reactivity Control Safety function is addressed

B. (1) ALL RCPs
   (2) AFTER the Reactivity Control Safety function is addressed

C. (1) 1A RCP ONLY
   (2) BEFORE the Reactivity Control Safety function is addressed

D. (1) 1A RCP ONLY
   (2) AFTER the Reactivity Control Safety function is addressed

Proposed Answer:	D

Explanations:	
A.	First part is plausible because during malfunctions that cause a loss of cooling water to RCPs, all RCPs are stopped. Second part is plausible if it is thought that stopping the RCP takes precedence over verifying that the Reactor has tripped since there is a high temperature.
B.	First part is plausible because during malfunctions that cause a loss of cooling water to RCPs, all RCPs are stopped. Second part is correct.
C.	First part is correct. Second part is plausible if it is thought that stopping the RCP takes precedence over verifying that the Reactor has tripped since there is a high temperature.
D.	Correct

Question Source:		New	
	X	Bank	
		Modified	
		Previous NRC Exam	2016

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	18654 – Given that the ORP is being implemented, describe the use of an AOP or OP when the reactor trips or when performing an EOP, in accordance with 40DP-9AP16, EOP Users Guide	

Technical Reference:	Previous Question on 2016 NRC Exam

Question 99

While directing actions in an AOP, the CRS encounters the following set of steps:

- Step 4. Trip the Reactor.
- Step 5. Trip all 4 RCPs.
- Step 6. GO TO 40EP-9EO01, SPTAs.

Per 40DP-9AP18, Abnormal Operating Procedure Users Guide, the CRS should direct tripping the RCPs _____(1)_____ and the CRS should _____(2)_____ .

A. 1. prior to addressing the Reactivity Control Safety Function
2. exit the AOP and direct SPTAs

B. 1. prior to addressing the Reactivity Control Safety Function
2. continue in the AOP while directing SPTAs

C. 1. immediately after addressing the Reactivity Control Safety Function.
2. exit the AOP and direct SPTAs

D. 1. immediately after addressing the Reactivity Control Safety Function.
2. continue in the AOP while directing SPTAs

3.0 ABNORMAL RCP MOTOR OR BEARING PARAMETERS

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 5. IF the RCP parameters indicated on RMN-TJR-2 points 1-32 exceed any of the trip setpoints listed in Appendix A, RCP Motor Or Bearing Trip Setpoints, THEN perform the following: a. Ensure the Reactor is tripped. b. Stop the affected RCP. c. GO TO the appropriate procedure for current plant conditions.	
____ 6. IF any RCP motor or bearing parameter is trending to a trip setpoint (REFER TO Appendix D, Instrumentation and Setpoints), AND the CRS determines a plant shutdown or cooldown is needed, THEN perform BOTH of the following: • The appropriate procedure to shutdown or cooldown the plant • 40OP-9RC01, Reactor Coolant Pump Operation, to stop the affected RCP	

10. The CRS is responsible to ensure the Safety Function Status Check (SFSC) is completed. This is normally delegated to the STA. The STA will normally perform the SFSC at an interval of approximately every 15 minutes. If the STA is not in the control room when the CRS gets to this step, another member of the control room staff (normally the 3rd Reactor Operator) must be designated to perform the SFSC until the STA arrives.

11. When directing Chemistry to perform 74DP-9ZZ05, notify them of the event in progress.

12. When the plant is not at normal operating conditions, the operator must interpolate actual PZR level from the calibration graph. When the plant is in cold conditions, the operator can reference RCN-LI-103. Level interpolation is necessary if not at cold conditions. RCN-LI-103 is calibrated for 120°F and 700 psia.

13. The use of the term "is desired" is found throughout the Emergency Operating Procedures. The intent of the term "is desired" is to provide the latitude for making a decision as to whether a given step should be performed based upon the existing plant conditions under which an event is being mitigated and the mitigation strategy being employed. The CRS is responsible to determine whether a given "is desired" action step should be performed based upon the mitigation strategy and the plant conditions which exist at the time the decision is made.

14. The preferred instrumentation for determining containment temperature is ERFDADS.

    For post accident conditions, average containment temperature is calculated from the 5 ERFDADS temperature instruments: HCN-T-0042A1, HCN-T-0042B1, HCN-T-0042C1, HCN-T-0042D1, and HCN-T-0042E1.

    If ERFDADS is unavailable, determine average containment temperature using an average of ALL of the five readings from recorder RMN-TJR-1:

    - HCN-TE-42A
    - HCN-TE-42B
    - HCN-TE-42C
    - HCN-TE-42D
    - HCN-TE-42E

15. Performance of an AOP may be in progress when the reactor trips and EOPs are entered. Reactivity Control Safety Function shall be addressed immediately after a reactor trip, however, some operations in progress will require that additional steps in an AOP be performed prior to addressing additional safety functions (for example, stopping Reactor Coolant Pumps and isolating seal bleedoff).

16. If a Safety Function is not satisfied, the crew should take prompt corrective action to restore the safety function. While there is no time limit for how long a parameter is allowed to be outside the acceptance criteria, it is important that restoring the safety function becomes a high priority and that the crew actively uses all available resources to restore the safety functions as soon as possible.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Loss of Reactor Coolant Makeup: Knowledge of the operational implications of EOP warnings, cautions, and notes	Tier	1	[redacted]	
	Group	1	[redacted]	
	K/A	022 G 2.4.20		
	IR	3.8	[redacted]	

Question 5

Given the following conditions:

- Unit 2 is operating at 100% power
- ‘A’ & ‘B’ Charging Pumps are operating

Subsequently:

- The OATC recognizes that charging flow has lowered to 25 gpm
- An Auxiliary Operator reports to the Control Room that ‘A’ Charging Pump is partially gas bound and ‘B’ Charging Pump is completely gas bound

(1) A completely gas bound pump should be indicated by a…

(2) The crew should isolate letdown and stop…

A. (1) quieter than normal sound
 (2) both Charging Pumps

B. (1) quieter than normal sound
 (2) ONLY the ‘B’ Charging Pump and evaluate whether Charging flow restores to normal

C. (1) louder than normal sound
 (2) both Charging Pumps

D. (1) louder than normal sound
 (2) ONLY the ‘B’ Charging Pump and evaluate whether Charging flow restores to normal

Proposed Answer:	**A**

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because in a scenario, if there is a RAS actuation and there is indication of cavitation, the Containment Spray pump will be stopped and conditions evaluated for improvement
C.	First part is plausible since Charging Pumps are positive displacement pumps they try to maintain a higher discharge pressure. As the pump become more gas bound it will work harder and therefore make a louder than expected noise. Another possibility is that the more gas bound a pump becomes, the more cavitation is occurring because of air in the system. Therefore more cavitation will have a higher than normal noise. Second part is correct.
D.	First part is plausible since Charging Pumps are positive displacement pumps they try to maintain a higher discharge pressure. As the pump become more gas bound it will work harder and therefore make a louder than expected noise. Another possibility is that the more gas bound a pump becomes, the more cavitation is occurring because of air in the system. Therefore more cavitation will have a higher than normal noise Second part is plausible because in a scenario, if there is a RAS actuation and there is indication of cavitation, the Containment Spray pump will be stopped and conditions evaluated for improvement

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**3**	
Reference Provided:	**N**	
Learning Objective:	**311393 – Explain how gas binding of the charging pumps is mitigated in 40AO-9ZZ05, Loss of Charging or Letdown.**	

Technical Reference:	40AO-9ZZ05, Loss of Charging or Letdown

Appendix G, Responding to Gas Binding of Charging Pumps

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 6. **IF** gas intrusion was **NOT** due to VCT level lowering below 0%, **THEN** PERFORM Appendix I, Venting Charging Pumps and Header to the Recycle Drain Header.	
____ 7. GO TO Section 3.0, Step 5 OR Section 4.0, Step 4.	

------------------------------ **NOTE** ------------------------------

With two charging pumps operating while one of the pumps is gas bound, the primary indication of the gas bound pump will be the sound. A charging pump that is partly gas bound will initially have much louder cavitation noises than a filled pump. As the pump becomes fully gas bound, the plate valves will make much less noise than those in a pump that is filled with fluid.

Technical Reference:	40AO-9ZZ05, Loss of Charging or Letdown

Appendix G, Responding to Gas Binding of Charging Pumps

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 1. Enter Appendix Entry Time and Date: ________________	
____ 2. IF CHB-FI-212, Charging Pump to Regen HX, indicates greater than 40 gpm, **THEN** GO TO Step 8.	
____ 3. Close CHB-UV-515, Letdown To Regen HX Isolation Valve, to isolate letdown flow.	
____ 4. Place **ALL** of the following handswitches in "PULL TO LOCK": • CHA-HS-216, Charging Pump 1 P01 • CHB-HS-217, Charging Pump 2 P01 • CHA-HS-218A, Charging Pump 3 P01 • CHB-HS-218, Charging Pump 3 P01	

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Residual Heat Removal System: Ability to operate and/or monitor the following as they apply to the Loss of Residual Heat Removal System: RCS Inventory	Tier	1		
	Group	1		
	K/A	025 AA1.02		
	IR	3.8		

Question 6

Given the following conditions:

- Unit 3 is in MODE 4
- Train ‘A’ SDC is in-service using the ‘A’ LPSI pump
- RCS Pressure is 320 psia
- RCS Temperature is 250°F
- Pressurizer level is 40%

Subsequently:

- A leak in the SDC loop occurs
- RCS pressure is 310 psia and slowly lowering
- Pressurizer level is 35% and lowering

(1) With NO operator action, the Pressurizer Low Level alarm should annunciate AS SOON AS Pressurizer level lowers to…

(2) After the ‘A’ SDC Cooling Loop is isolated, the crew can shift to ‘B’ SDC Cooling Loop using…

A. (1) 10%
(2) ONLY ‘B’ LPSI Pump

B. (1) 10%
(2) ‘B’ LPSI OR ‘B’ CS Pump

C. (1) 25%
(2) ONLY ‘B’ LPSI Pump

D. (1) 25%
(2) ‘B’ LPSI OR ‘B’ CS Pump

Proposed Answer:	C

Explanations:	
A.	First part is plausible because 10% pressurizer level represents a value that ensures that there is subcooled liquid in the Pressurizer. During SPTAs, operators will maintain Pressurizer level greater than 10%. Second part is correct.
B.	First part is plausible because 10% pressurizer level represents a value that ensures that there is subcooled liquid in the Pressurizer. During SPTAs, operators will maintain Pressurizer level greater than 10%. Second part is plausible because a CS pump can be used if temperature and pressure requirements have been met (<210 psia and <185°F)
C.	Correct
D.	First part is correct. Second part is plausible because a CS pump can be used if temperature and pressure requirements have been met (<210 psia and <185°F)

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	19381 – Describe the purpose and conditions under which Shutdown Cooling System is designed to function	

Technical Reference:	40AL-9RK4A, Panel B04A Alarm Responses

Response Section

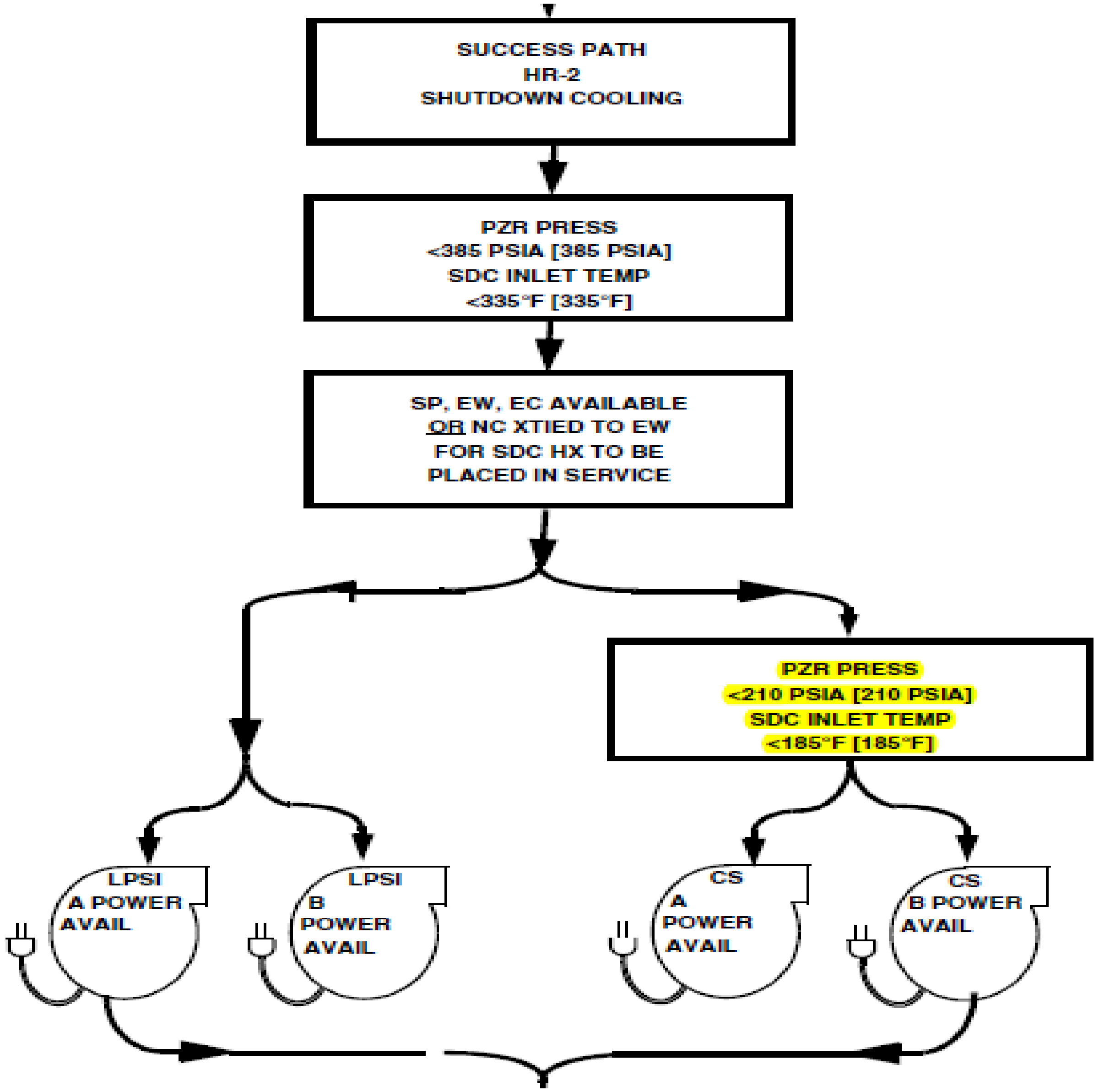
4A02B

PZR LVL HI-LO

Pressurizer Level High-Low

Point ID	Description	Setpoint
RCLS110X	Pressurizer Level Ch X Lo	25%
RCLS110Y	Pressurizer Level Ch Y Lo	25%

Technical Reference:	40EP-9EO11, Lower Mode Functional Recovery Procedure



Technical Reference:	40EP-9EO11, Lower Mode Functional Recovery

INSTRUCTIONS	CONTINGENCY ACTIONS
✱ 13. IF ANY of the following exist: • SDC is NOT running, • It is desired to start an idle train, AND ALL of the following are met: • Any CS Pump is available, • The 4.16 kV bus for the available CS pump is NOT energized from a single SBOG, • Pressurizer pressure is less than 210 psia [210 psia], • RCS temperature is less than 185°F [185°F], THEN PERFORM ONE of the following as appropriate: • Appendix 239, LM - Placing Train A CS on SDC • Appendix 240, LM - Placing Train B CS on SDC	

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Component Cooling Water: Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: Effect on the CCW flow header of a loss of CCW	Tier	1		
	Group	1		
	K/A	026 AK3.04		
	IR	3.5		

Question 7

Given the following conditions:

- Unit 1 is operating at 100% power
- A complete loss of Nuclear Cooling water occurred 5 minutes ago
- The CRS has directed the BOP to perform 40AO-9ZZ03 Loss of Cooling Water, Appendix A, Cross-connect EW to NC, using Train ‘A’ EW

Per Appendix A the BOP should ensure that a MAXIMUM of ___(1)___ Normal Chiller NC outlet valve(s) is(are) open in order to ___(2)___.

A. (1) one
(2) provide additional EW flow to NC priority loads

B. (1) one
(2) ensure sufficient flow to the ‘A’ SDCHX in the event of a design basis accident

C. (1) two
(2) provide additional EW flow to NC priority loads

D. (1) two
(2) ensure sufficient flow to the ‘A’ SDCHX in the event of a design basis accident

Proposed Answer:	**A**

Explanations: **This K/A is a match because there is a loss of NC (CCW) that results in flow lost to NC priority loads. After EW is cross-tied to NC, there is still insufficient flow to NC priority loads so ensuring that a maximum of 1 chiller outlet valve will increase flow.**	
A.	Correct
B.	First part is correct. Second part is plausible because losing the outlet valve initially will provide more flow to the SDCHX, however if there is an accident and a SIAS, the cross connect valves will close to ensure that there is enough flow to the SDCHX.
C.	First part is plausible because during summer months, two large chillers are needed to maintain adequate cooling supply. Second part is correct.
D.	First part is plausible because during summer months, two large chillers are needed to maintain adequate cooling supply. Second part is plausible because losing the outlet valve initially will provide more flow to the SDCHX, however if there is an accident and a SIAS, the cross connect valves will close to ensure that there is enough flow to the SDCHX.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**22359 – Given a loss of NC, describe how flow to the RCPs is increased after EW has been cross-tied**	

Technical Reference:	40AO-9ZZ03, Loss of Cooling Water, Appendix A, Cross-Connect EW to NC

Appendix A, Cross-connect EW to NC

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 4. Ensure that both Nuclear Cooling Water pump handswitches are in "PULL TO LOCK".	

---------- **NOTE** ----------

NCN-UV-99 must be fully closed before opening the EW cross-tie to NC valve in Step 7 or Step 8. If NCN-UV-99 is not fully closed, an uncontrolled transfer of water will occur between NC and EW.

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 5. Close NCN-UV-99, Nuclear Cooling Water Containment Header Return Valve.	
____ 6. Ensure that no more than one Normal Chiller NC outlet valve is open.	

EO: 1.11 Given a loss of NC, describe how flow to the RCPs is increased after EW has been cross tied in accordance with 40AO-9ZZ03.

Introduction

The 10 minute "clock" is still ticking until the RCP low flow alarms have cleared.

Main Idea

Before throttling the Train A or Train B SDHX Outlet Valve (EWX-HCV-53/54) the operator ensures that no more than one Normal Chiller outlet valve is open. This is done by taking the control room handswitch on the previously operating chiller(s) to stop. Normally only two chillers are operating, so only one chiller handswitch is taken to stop.

Taking the selected chiller's handswitch to stop, secures the chill water circ pump and closes the chiller NC outlet valve. Ensuring that no more than one Normal Chiller outlet valve is open provides for the following:

- Additional EW flow to NC priority loads by closing tripped chiller(s) NC outlet valve(s).
- Continued chill water flow (even though it's NOT being cooled by a chiller) to WC cooled components including the Main Generator collector housing. (The chill water circ pump continues to operate on the chiller that has NOT been taken to stop.)
- Prevents the RCP NC low flow alarm from recurring on a subsequent chiller start.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Pressurizer Pressure Control System Malfunction: Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions: Actions contained in EOP for PZR PCS malfunction	**Tier**	**1**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**027 AK3.03**		
	IR	**3.7**	[redacted]	

Question 8

Given the following conditions:

- Unit 3 is operating at 100% power

Subsequently:

- Pressurizer Spray Control Valve, RCE-PV-100E, failed open
- All ARP actions to close the failed valve were unsuccessful
- The CRS directed the BOP to trip the Reactor when RCS Pressure lowered to 1950 psia
- During SPTAs, the OATC operated RCPs as directed in the ARP

Which of the following describes the ARP directed action for RCP operation and the reason for this direction?

The OATC should trip ___(1)___ RCPs in order to ___(2)___ .

A. (1) ONLY 2
(2) protect RCPs due to insufficient NPSH for 4 RCPs to be in operation

B. (1) ONLY 2
(2) reduce DP across the Main Spray valves to allow heaters to restore pressure

C. (1) ALL 4
(2) protect RCPs due to insufficient NPSH for 4 RCPs to be in operation

D. (1) ALL 4
(2) reduce DP across the Main Spray valves to allow heaters to restore pressure

Proposed Answer:	D

Explanations:	
A.	First part is plausible since the Main Spray valves only tap off of two RCS loops and maintaining forced circulation is always preferred, however the ARP directs stopping all 4 RCPs. Second part is plausible as NPSH is degrading as RCS pressure lowers, and RCPs are required to be stopped if pressure lowers to less than minimum NPSH, however that pressure has not been reached and is not the basis for stopping RCPs following the reactor trip.
B.	First part is plausible since the Main Spray valves only tap off of two RCS loops and maintaining forced circulation is always preferred, however the ARP directs stopping all 4 RCPs. Second part is correct.
C.	First part is correct. Second part is plausible as NPSH is degrading as RCS pressure lowers, and RCPs are required to be stopped if pressure lowers to less than minimum NPSH, however that pressure has not been reached and is not the basis for stopping RCPs following the reactor trip.
D.	Correct.

Question Source:	X	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**14**	
Reference Provided:	**N**	
Learning Objective:	**24946 – Describe the response of the Pressurizer Pressure Control System to a failure of an input transmitter**	

Technical Reference:	40AL-9RK4A, Panel B04A Alarm Responses

_____ 5.3 **IF** ANY of the following will NOT close following attempts to close the open Pressurizer spray valve:

- ___ • RCE-PV-100E, Pressurizer Spray Control Valve from RCS Loop 1A
- ___ • RCE-PV-100F, Pressurizer Spray Control Valve from RCS Loop 1B

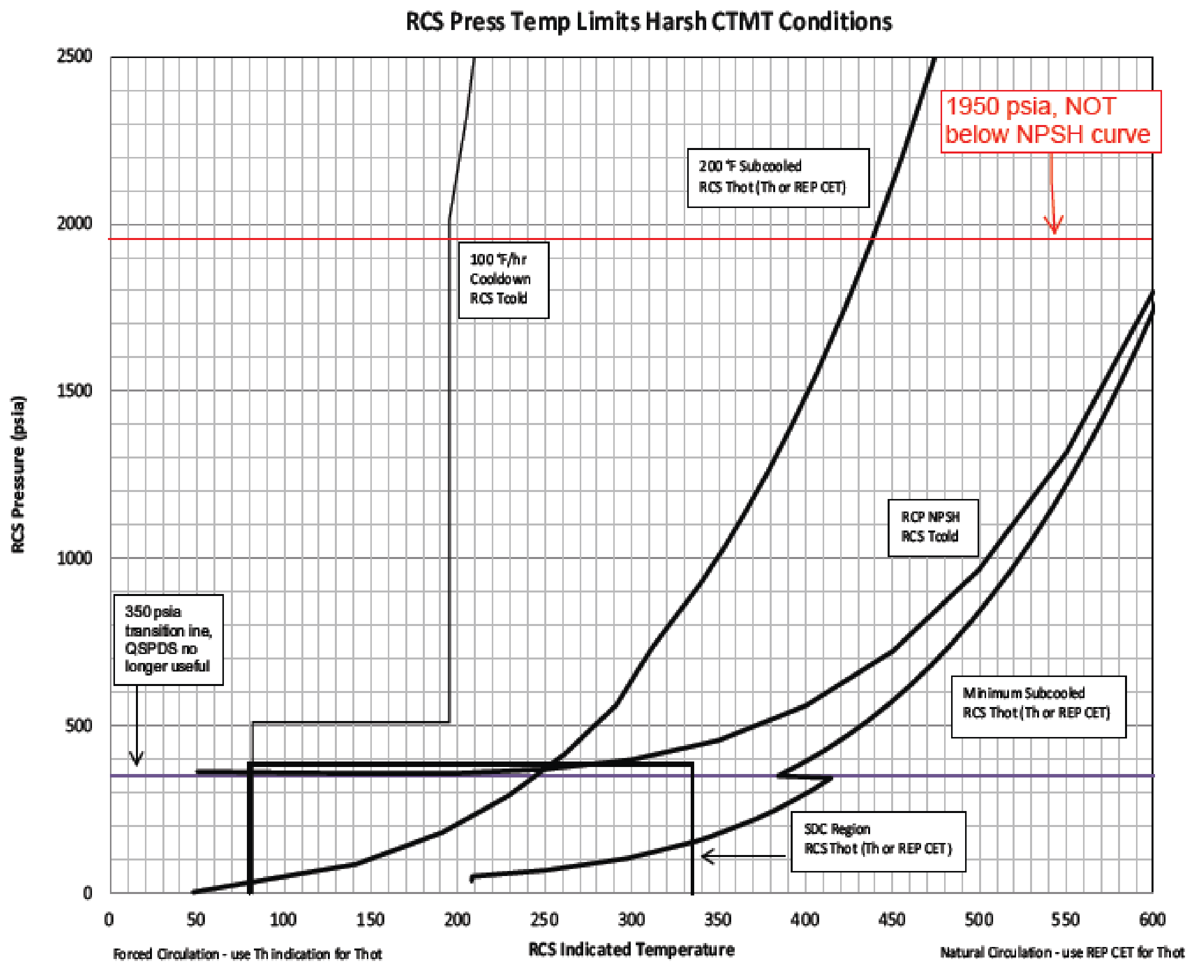
THEN perform the following:

___ 5.3.1 Trip the Reactor.

___ 5.3.2 Trip all four RCPs.

___ 5.3.3 GO TO 40EP-9EO01, Standard Post Trip Actions.

Technical Reference:	40EP-9EO10-002, Appendix 2: Figures



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Anticipated Transient Without Scram: Knowledge of the interrelations between the ATWS and the following an ATWS: Breakers, relays, and disconnects	**Tier**	**1**		
	Group	**1**		
	K/A	**029 EK2.06**		
	IR	**2.9**		

Question 9

Given the following conditions:

- A malfunction has caused Pressurizer pressure to rise

IF an ATWS occurred and NO OPERATOR ACTION is taken, the SPS should send a trip signal to ___(1)___ AS SOON AS RCS pressure reaches a MINIMUM of ___(2)___ psia.

A. (1) RTCBs ONLY
   (2) 2383

B. (1) RTCBs ONLY
   (2) 2409

C. (1) RTCBs and MG Set contactors
   (2) 2383

D. (1) RTCBs and MG Set contactors
   (2) 2409

Proposed Answer:	**D**

Explanations:	
A.	First part is plausible because if an RPS Reactor trip setpoint is exceeded, only the RTCBs will open. Second part is plausible because 2383 psia is the RPS Reactor trip setpoint.
B.	First part is plausible because if an RPS Reactor trip setpoint is exceeded, only the RTCBs will open. Second part is correct.
C.	First part is correct. Second part is plausible because 2383 psia is the RPS Reactor trip setpoint.
D.	Correct

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**24948 – Describe the Supplementary Protection System including its function, instrumentation, bases, and setpoint**	

Technical Reference:	40AL-9RK5A, Panel B05A Alarm Responses

Response Section

5A07A

SPS
CH
TRIP

Supplementary Protection System Channel Trip

Point ID	Description	Setpoint
SBYS45	Supplementary Protection System Channel A Trip	2409 psia
SBYS46	Supplementary Protection System Channel B Trip	2409 psia
SBYS47	Supplementary Protection System Channel C Trip	2409 psia
SBYS48	Supplementary Protection System Channel D Trip	2409 psia

Technical Reference:	Plant Protection System Tech Manual

1.2.3 Supplementary Protection System

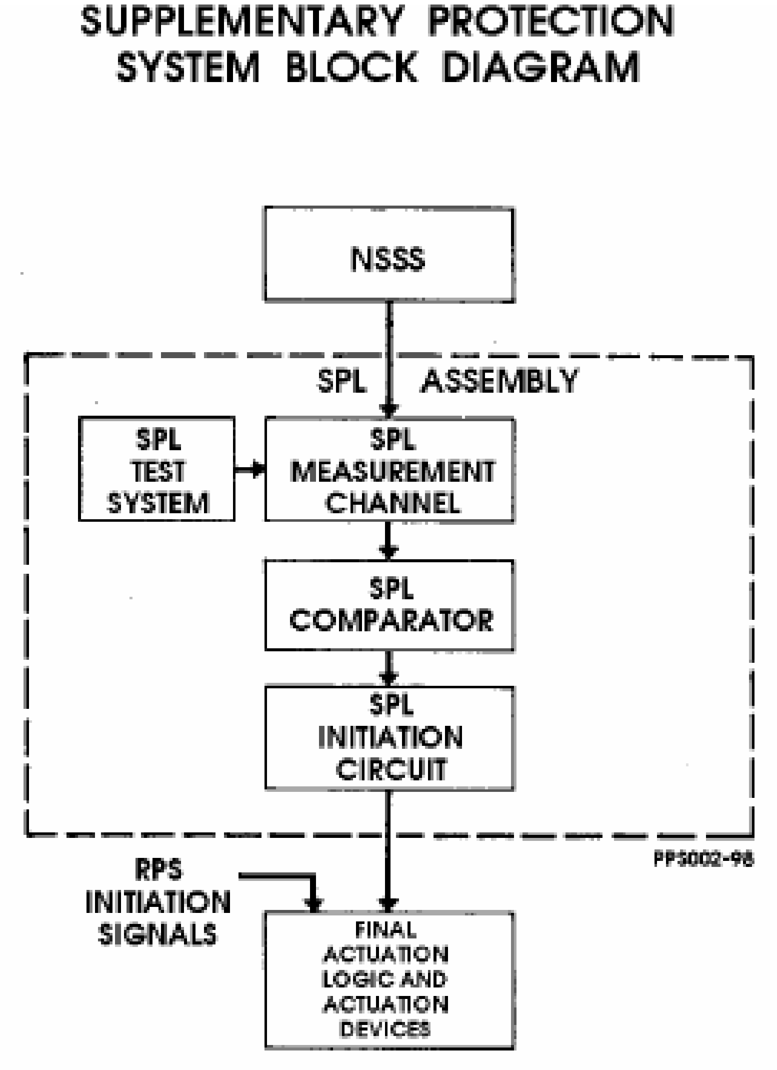


Figure 1 - 2 Supplementary Protection System Block Diagram

A supplementary protection system (SPS) augments the RPS by initiating a reactor trip using diverse sensing and trip logic when pressurizer pressure exceeds the SPS trip setpoint. The trip signal opens the reactor trip switchgear (RTSG) and CEDM MG set output contactors, either one of which will de-energize the CEDM coils allowing all CEAs to drop into the core.

Technical Reference:	40AL-9RK5A, Panel B05A Alarm Responses

Response Section

5A05A

HI PZR PRESS CH TRIP

High Pressurizer Pressure Channel Trip

Point ID	Description	Setpoint
SBTA05	Hi Pressurizer Pressure Channel A Trip	2383 psia
SBTB05	Hi Pressurizer Pressure Channel B Trip	2383 psia
SBTC05	Hi Pressurizer Pressure Channel C Trip	2383 psia
SBTD05	Hi Pressurizer Pressure Channel D Trip	2383 psia

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Steam Generator Tube Rupture: Knowledge of the operational implications of the following concepts as they apply to the SGTR: Use of steam tables	**Tier**	**1**		
	Group	**1**		
	K/A	**038 EK1.01**		
	IR	**3.1**		

Question 10

Given the following conditions:

- Unit 2 was tripped due to a SG tube rupture on SG #1
- The CRS entered 40EP-9EO04, SGTR

The BOP should lower Steam Generator pressures to a MAXIMUM of ___(1)___ to ensure that ___(2)___ is at the required temperature prior to isolating SG #1.

A. (1) 950 psia
   (2) T_{HOT}

B. (1) 950 psia
   (2) T_{COLD}

C. (1) 1135 psia
   (2) T_{HOT}

D. (1) 1135 psia
   (2) T_{COLD}

Proposed Answer:	A

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because when cooling down for all other events, T_{COLD} is used to track the cooldown.
C.	First part is plausible because 1135 psia is the pressure that the RCS will be lowered to prevent possibly lifting a Main Steam Safety Valve during a SGTR. Second part is correct.
D.	First part is plausible because 1135 psia is the pressure that the RCS will be lowered to prevent possibly lifting a Main Steam Safety Valve during a SGTR. Second part is plausible because when cooling down for all other events, T_{COLD} is used to track the cooldown.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**5**	
Reference Provided:	**Y**	**Steam Tables**
Learning Objective:	**29951 – Given that the SGTR ORP is being performed and the RCS is being cooled to allow SG isolation, state the associated parameter and value of the cooldown target and its basisin accordance with 40EP-9EO04 and the SGTR Technical Guideline**	

Technical Reference:	40EP-9EO04, Steam Generator Tube Rupture

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

KEY OPERATOR ACTION - Perfect performance of closely related steps 10. through 12. will significantly reduce plant risk.

INSTRUCTIONS	CONTINGENCY ACTIONS
10. Commence an RCS cooldown to a T_h of less than 540°F using SBCS.	10.1 **IF** SBCS to the condenser is **NOT** available, **THEN** cooldown using **ANY** of the following: • ADV operation from the Control Room • Appendix 116, Operation of SBCS Valves 1007 and 1008 • Appendix 18, Local ADV Operation

Technical Reference:	40EP-9EO04, Steam Generator Tube Rupture

INSTRUCTIONS	CONTINGENCY ACTIONS
12. Depressurize the RCS by performing the following: a. Maintain pressurizer pressure within **ALL** of the following criteria: • Less than 1135 psia • Approximately equal to the pressure of the Steam Generator with the tube rupture (± 50 psi) • Within the P/T Limits. REFER TO Appendix 2, Figures • Within RCP NPSH Limits. REFER TO Appendix 2, Figures	12.1 **IF** pressurizer pressure can **NOT** be lowered and maintained within the stated criteria, **THEN** PERFORM Appendix 102, RCS Depressurization using RCGVS.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Steam Line Rupture – Excessive Heat Transfer: Knowledge of the interrelations between the (Excess Steam Demand) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	**Tier**	**1**		
	Group	**1**		
	K/A	**040 EK2.2**		
	IR	**3.2**		

Question 11

Given the following conditions:

- An unisolable ESD event outside of Containment is in progress in Unit 1
- SG #1 pressure is 920 psia and lowering
- SG #2 pressure is 950 psia and stable
- The CRS has entered 40EP-9EO05, Excess Steam Demand

After dryout conditions have been met on the faulted Steam Generator, the crew should minimize the effects of Pressurized Thermal Shock by stabilizing T_{COLD} using ___(1)___ and ___(2)___.

A. (1) ADVs
(2) throttling closed HPSI Injection valves

B. (1) ADVs
(2) depressurizing the RCS with Auxiliary Spray Valves

C. (1) SBCVs
(2) throttling closed HPSI Injection valves

D. (1) SBCVs
(2) depressurizing the RCS with Auxiliary Spray Valves

Proposed Answer:	**A**

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because opening Auxiliary Spray Valves will stop (or limit) the repressurization of the RCS, however the continuing injection of SI flow will continue making PTS a possibility.
C.	First part is plausible because SBCVs would normally be used to control RCS temperature post Reactor trip. However since SG pressures have lowered below MSIS setpoints, therefore ADVs will be used. Second part is correct.
D.	First part is plausible because SBCVs would normally be used to control RCS temperature post Reactor trip. However since SG pressures have lowered below MSIS setpoints, therefore ADVs will be used. Second part is plausible because opening Auxiliary Spray Valves will stop (or limit) the repressurization of the RCS, however the continuing injection of SI flow will continue making PTS a possibility.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**5**	
Reference Provided:	**N**	
Learning Objective:	**25489 - Given a set of plant parameters determine when and how RCS temperature is stabilized during an ESD per 40EP-9EO05, ESD**	

Technical Reference:	40AL-9RK5A, Panel B05A Alarm Responses

Panel B05A Alarm Responses	40AL-9RK5A	Revision 2
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Response Section

5A07C

LO SG 1 PRESS CH TRIP

Low Steam Generator 1 Pressure Channel Trip

Point ID	Description	Setpoint
SBTA11	Lo Steam Generator 1 Pressure Ch A Trip	960 psia variable
SBTB11	Lo Steam Generator 1 Pressure Ch B Trip	960 psia variable
SBTC11	Lo Steam Generator 1 Pressure Ch C Trip	960 psia variable
SBTD11	Lo Steam Generator 1 Pressure Ch D Trip	960 psia variable

Technical Reference:	40EP-9EO05, Excess Steam Demand

INSTRUCTIONS	CONTINGENCY ACTIONS
14. Stabilize RCS temperature using the lowest T_c by performing the following:	
a. Maintain T_c within the P/T limits. REFER TO Appendix 2, Figures	
b. Steam the least affected Steam Generator using ANY of the following: • ADVs from the Control Room • Appendix 18, Local ADV Operation	
c. Control feedwater to the least affected Steam Generator.	
d. WHEN control is regained, THEN record the following: Time: ______ RCS T_C: ______ PZR Pressure: ______	

Technical Reference:	40EP-9EO05, Excess Steam Demand

INSTRUCTIONS | CONTINGENCY ACTIONS

CAUTION

Throttling HPSI injection valves will cause erosion damage to downstream piping.

✱ 16. IF at least one HPSI Pump is operating, AND ALL of the following conditions exist:

- RCS is 24°F [44°F] or more subcooled
- Pressurizer level is greater than 10% [15%] and NOT lowering
- The unisolated Steam Generator is available for RCS heat removal with level being maintained within or being restored to 45 - 60% [45 - 60%] NR
- RVLMS indicates RVUH level is 16% or more

THEN throttle HPSI flow or stop the HPSI Pumps one pump at a time.

4.5.14 Step 14 - Stabilize RCS Temperature

A. The main objective following an overcooling event is to minimize the stresses on the reactor vessel, return RCS temperature to within the Post Accident P/T limits and establish stable RCS pressure and temperature until a cooldown to SDC entry conditions can be started. In general, a controlled cooldown should be started as soon as possible.

RCS temperature control is achieved by steaming the least affected SG using the atmospheric dump valves as necessary, and by ensuring adequate SG inventory for heat removal.

A heat removal method via the least affected SG should be established before SG dry out occurs, if possible. PTS concerns regarding an ESD event and the factors or mitigating trends that tend to lessen PTS concerns include:

- Limiting RCS repressurization as much as possible while maintaining minimum RCS subcooling requirements.
- Restoring and maintaining control of RCS temperature within post accident P/T limits.

In case the atmospheric dump valves are not available, SG safeties on the unaffected SG will serve as a heat removal method. Steaming via the SG safeties is not the preferred method because it results in RCS pressure increasing as SG temperature/pressure rise to the SG safety valve lift setpoint. Therefore, every effort should be made to regain use of the atmospheric dump valves to eliminate the possibility of RCS repressurization.

RCS temperature will begin to increase after the affected steam generator dries out unless a means of controlling RCS heat removal is established. The increase in RCS temperature may result in a water-solid condition due to the inventory added from safety injection and charging operation during the blowdown phase of the event. The post-dryout heatup and repressurization also presents a PTS concern. Limiting RCS repressurization as much as possible while maintaining RCS subcooling requirements and maintaining control of RCS temperature within the limits of the post accident P/T curve will limit PTS concerns.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Main Feedwater: Knowledge of the operational implications of the following concepts as they apply to the (Loss of Feedwater): Components, capacity, and function of emergency systems	**Tier**	**1**		
	Group	**1**		
	K/A	**054 EK1.1**		
	IR	**3.2**		

Question 12

Given the following conditions:

- Unit 1 tripped due to a complete loss of Main Feedwater.
- AFB-P01 has been manually started and aligned to feed both SGs.

Subsequently:

- AFAS-1 actuates.

With NO operator action, how should the AFAS-1 affect the current feed lineup?

AFA-P01 should start and feed ___(1)___ and AFB-P01 should be feeding ___(2)___.

A. (1) SG #1 ONLY
(2) SG #1 ONLY

B. (1) SG #1 ONLY
(2) both SGs

C. (1) both SGs
(2) SG #1 ONLY

D. (1) both SGs
(2) both SGs

Proposed Answer:	A

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible since it will align to feed SG 1 and it was already aligned to feed SG 2, however on an AFAS-1, all feed will stop to SG 2 and both AFW Pumps will commence feeding SG 1.
C.	First part is plausible since AFA-P01 is drawing steam from both SGs on an AFAS-1 or AFAS-2, however it will only feed the SG with the active AFAS signal. Second part is correct.
D.	First part is plausible since AFA-P01 is drawing steam from both SGs on an AFAS-1 or AFAS-2, however it will only feed the SG with the active AFAS signal. Second part is plausible since it will align to feed SG 1 and it was already aligned to feed SG 2, however on an AFAS-1, all feed will stop to SG 2 and both AFW Pumps will commence feeding SG 1.

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2016

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	24499 - Describe the system response to an Auxiliary Feedwater Actuation Signal	

Technical Reference:	Auxiliary Feedwater System Tech Manual

3.1 Normal Operations

The essential components of the AFW system will be aligned for automatic operation during normal at power conditions of the unit. This allows for AFW system response to automatic actuation signals that would be generated by the plant protection system (PPS). When the S/G level decreases to 25.8 % WR, as sensed by two out of four class level instruments for that S/G, the PPS circuitry responds to initiate an auxiliary feedwater actuation signal (AFAS) for that S/G.

When an AFAS 1 signal is generated, at 25.8% WR on 2 of 4 class level instruments for the #1 S/G, the following automatic actions are initiated:

- Both essential AFW pumps are automatically started
- Both emergency diesel generators are started.
- The A and B trains of SP, EC, and EW are started.
- All S/G blowdown and sampling is isolated.
- The isolation and flow control valves to the #1 S/G from AFA-P01 are opened. (AFC-UV-36 and AFA-UV-32)
- The isolation and flow control valves to the #1 S/G from AFB-P01 are opened. (AFB-UV-34 and AFB-UV-30)

When these actions occur, the #1 S/G receives full AFW flow from the essential pumps. This flow continues until S/G level reaches 40.8% WR, as sensed by the class level instruments for that S/G. At 40.8%, all of the isolation and flow control valves (4) will automatically close to stop adding feedwater to the S/G. These valves will then open again, automatically, if level decreases to 25.8% to restore S/G level. All four valves will continue to open and close, as required, to maintain S/G level between 25.8% and 40.8% WR.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Station Blackout: Knowledge of the reasons for the following responses as they apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power	**Tier**	**1**		
	Group	**1**		
	K/A	**055 EK3.02**		
	IR	**4.3**		

Question 13

Per 40EP-9EO08, Blackout, the purpose of actuating a MSIS is to…

1. minimize cooldown of the RCS
2. minimize effects of loss of Instrument Air
3. prevent damage to the Main Condenser
4. prevent an inadvertent loss of steam pressure and inventory

A. 1 AND 4 ONLY

B. 1 AND 3 ONLY

C. 2 AND 3 ONLY

D. 2 AND 4 ONLY

Proposed Answer:	B

Explanations:	
A.	Minimize cooldown of the RCS is correct. Prevent an inadvertent loss of steam pressure and inventory is plausible because an MSIS will minimize pressure and inventory losses but this is not the reason for MSIS. During Blackout inventory will be maintained with AFA-P01 and will not be an issue.
B.	Correct
C.	Minimize the effects of loss of Instrument Air is plausible because all Instrument Air and Service Air Compressors lose power. As Instrument Air pressure lowers, there are valves that will fail open and could potentially affect the RCS. 40EP-9EO08, Blackout does have a step to monitor IA and Nitrogen air pressure, however it is not one of the purposes for actuating MSIS. It is also plausible because the second step in the Loss of Instrument Air AOP is to initiate MSIS if desired. Prevent damage to the Main Condenser is correct.
D.	Minimize the effects of loss of Instrument Air is plausible because all Instrument Air and Service Air Compressors lose power. As Instrument Air pressure lowers, there are valves that will fail open and could potentially affect the RCS. 40EP-9EO08, Blackout does have a step to monitor IA and Nitrogen air pressure, however it is not one of the purposes for actuating MSIS. It is also plausible because the second step in the Loss of Instrument Air AOP is to initiate MSIS if desired. Prevent an inadvertent loss of steam pressure and inventory is plausible because an MSIS will minimize pressure and inventory losses but this is not the reason for MSIS. During Blackout inventory will be maintained with AFA-P01 and will not be an issue.

Question Source:	X	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**4**	
Reference Provided:	**N**	
Learning Objective:	**26233 - Explain why an MSIS is initiated in accordance with 40EP-9EO08, Blackout**	

Technical Reference:	40DP-9AP13, Blackout Technical Guideline

4.5.3 **Step 3 - Open the Placekeeper and Enter the Time of the Event**

A. This step reminds the operator to open the placekeeper and record the time of the event.

The placekeeper provides a place for the CRS to keep track of progress through the procedure, and provides a broad overview of its implementation. The placekeeper is located in the back of the procedure.

4.5.4 **Step 4 - Actuate MSIS**

A. Various secondary valves may fail open on a Blackout causing a cooldown of the RCS and possibly causing Main Condenser damage due to the loss of vacuum. Actuating MSIS minimizes the cooldown and the potential for damage to the condenser.

Technical Reference:	40DP-9AP13, Blackout Technical Guideline

4.5.20 Step 20 - Monitor Instrument Air and N2 Supply

A. A loss of offsite power will result in a loss of Instrument Air Compressors. The low pressure nitrogen system automatically backs up the Instrument Air System. The appendix will monitor Instrument Air header pressure and N_2 tank level to ensure an adequate supply of nitrogen exists for air operated valve operation.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Vital AC Instrument Bus: Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus	Tier	1		
	Group	1		
	K/A	057 AK3.01		
	IR	4.1		

Question 14

Given the following conditions:

- Unit 2 is operating at 100% power
- 120 VAC Class Instrument Bus, PNA-D25, tripped on a fault

The crew is required to commence monitoring DNBR/LHR/AZTILT/ASI for ADVERSE trends within a MAXIMUM of ___(1)___ to ensure DNBR/LHR/AZTILT/ASI are within Technical Specification limits due to the loss of ___(2)___.

A. (1) 15 minutes
   (2) COLSS

B. (1) 15 minutes
   (2) CEAC 1

C. (1) 1 hour
   (2) COLSS

D. (1) 1 hour
   (2) CEAC 1

Proposed Answer:	A

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because CEAC 1 is INOPERABLE with a loss of PNA-D25
C.	First part is plausible because a loss of PNA-D25 makes COLSS out of service and it may be assumed that LHR and DNBR are exceeding the Technical Specification limits and is required to be restored within one hour. Second part is correct.
D.	First part is plausible because a loss of PNA-D25 makes COLSS out of service and it may be assumed that LHR and DNBR are exceeding the Technical Specification limits and is required to be restored within one hour. Second part is plausible because CEAC 1 is INOPERABLE with a loss of PNA-D25. CEAC can still be monitored from CEAC 2.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**4**	
Reference Provided:	**N**	
Learning Objective:	**17537 – Given conditions where COLSS is inoperable, monitor DNBR/LHR/ASI with COLSS out of service in accordance with 72ST-9RX03**	

Technical Reference:	40AO-9ZZ13, Loss of Class Instrument or Control Power

PALO VERDE NUCLEAR GENERATING STATION LOSS OF CLASS INSTRUMENT OR CONTROL POWER	40AO-9ZZ13 Revision 30 Page 21 of 184

4.0 LOSS OF PNA-D25

INSTRUCTIONS	CONTINGENCY ACTIONS
___ 6. **IF** reactor power is greater than 20% RTP **THEN** PERFORM 72ST-9RX03, DNBR/LHR/AZTILT/ASI With COLSS Out Of Service **within 15 minutes**	

Technical Reference:	

Appendix A, Effects of the Loss of Channel A

System	PKA M41	PKA D21	PNA D25	Response
SB	X			RTSG Breaker A trips open due to the UV relay de-energizing.
			X	RTSG Breaker A and C trip open due to loss of power to one leg of the RPS logic matrices AB, AC, AD. RTSG Breaker A trips on a SPLA trip and loss of power to RPS Initiation path #1. Lose power to all Channel A input parameter instruments resulting in 1-3 half leg trips on all parameters that have a trip setpoint. Parameters that fail high or low are inoperable. CEAC 1 in all CPC channels becomes inop due to loss of power to RSPTs and may generate penalty factors when reenergized.

Technical Reference:	Technical Specifications

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Linear Heat Rate (LHR)

LCO 3.2.1 LHR shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core Operating Limit Supervisory System (COLSS) calculated core power exceeds the COLSS calculated core power operating limit based on LHR.	A.1 Restore LHR to within limits.	1 hour

Technical Reference:	

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Departure From Nucleate Boiling Ratio (DNBR)

LCO 3.2.4 The DNBR shall be maintained by one of the following methods:

a. Core Operating Limit Supervisory System (COLSS) In Service:

1. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR when at least one Control Element Assembly Calculator (CEAC) is OPERABLE in each OPERABLE Core Protection Calculator (CPC) channel; or

2. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by the allowance specified in the COLR when the CEAC requirements of LCO 3.2.4.a.1 are not met.

b. COLSS Out of Service:

1. Operating within the region of acceptable operation specified in the COLR using any OPERABLE Core Protection Calculator (CPC) channel when at least one Control Element Assembly Calculator (CEAC) is OPERABLE in each OPERABLE CPC channel; or

2. Operating within the region of acceptable operation specified in the COLR using any OPERABLE CPC channel (with both CEACs inoperable) when the CEAC requirements of LCO 3.2.4.b.1 are not met.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. COLSS calculated core power not within limit.	A.1 Restore the DNBR to within limit.	1 hour

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of DC Power: Knowledge of limiting condition for operations and safety limits	Tier	1		
	Group	1		
	K/A	058 G 2.2.22		
	IR	4.0		

Question 15

Given the following conditions:

- Unit 1 is operating in MODE 4
- The crew is cooling down and depressurizing the RCS following a LOOP per 40EP-9EO07, Loss of Offsite Power/Loss of Forced Circulation

Subsequently:

- Class 125 VDC Bus, PKB-M42, tripped on overcurrent

Per Technical Specifications, the REQUIRED ACTION(s) of LCO ___(1)___ must be performed and ___(2)___ Auxiliary Spray Valve(s) is(are) available to continue the depressurization.

A. (1) 3.8.4, DC Sources - Operating
(2) one

B. (1) 3.8.4, DC Sources - Operating
(2) both

C. (1) 3.8.5, DC Sources - Shutdown
(2) one

D. (1) 3.8.5, DC Sources - Shutdown
(2) both

Proposed Answer:	A

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because only PKA-M41 and PKB-M42 cause a loss of an Auxiliary Spray Valve. PKC-M43 and PKD-M44 will not cause a loss of an Auxiliary Spray Valve.
C.	First part is plausible because the Lower Mode Functional Recovery procedure is used in Mode 4 so it could be assumed that if the LMFRP is being used then LCO 3.8.5 would be applicable. Second part is correct.
D.	First part is plausible because the Lower Mode Functional Recovery procedure is used in Mode 4 so it could be assumed that if the LMFRP is being used then LCO 3.8.5 would be applicable. Second part is plausible because only PKA-M41 and PKB-M42 cause a loss of an Auxiliary Spray Valve. PKC-M43 and PKD-M44 will not cause a loss of an Auxiliary Spray Valve.

Question Source:	X	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**8**	
Reference Provided:	**N**	
Learning Objective:	**21203 – Given a set of plant conditions determine whether or not the LCOs and TLCOs of 3.8 are satisfied in accordance with Tech Spec 3.8**	

Technical Reference:	Technical Specifications

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

Technical Reference:	40AO-9ZZ13, Loss of Class Instrument or Control Power

Appendix C, Effects of the Loss of Channel B

System	PKB M42	PKB D22	PNB D26	Response
AF	X	X		AFB-P01 has lost its control power.
CH			X	CHB-UV-515 closes due to loss of power to CHB-TT-221.
	X	X		CHB-UV-515/523, CVCS Letdown Isolation Valves, fail closed causing a loss of letdown flow. CHB-UV-505, RCP Controlled Bleedoff valve, fails closed. RCP bleedoff lifts CHN-PSV-199, RCP Seal Leak-Off To RDT Relief Valve. CHB-HV-203, Aux Spray Valve, fails closed. CHA-HV-205 must be used for pressure control if the Main Spray Valves are unavailable.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Nuclear Service Water: Ability to operate and/or monitor the following as they apply to the Loss of Nuclear Service Water (SWS): Loads on the SWS in the control room	**Tier**	**1**		
	Group	**1**		
	K/A	**062 AA1.02**		
	IR	**3.2**		

Question 16

Given the following conditions:

- Unit 2 is operating at 100% power
- ‘A’ PW pump is OOS

Subsequently:

- ‘B’ PW pump trips on overcurrent

(1) Per 40AO-9ZZ03, Loss of Cooling Water, the crew should trip…

(2) The crew should break vacuum on the Main Turbine…

A. (1) the Main Turbine ONLY
(2) IMMEDIATELY following the Main Turbine trip

B. (1) the Main Turbine ONLY
(2) as soon as the Main Turbine reaches 1200 RPM

C. (1) the Reactor
(2) IMMEDIATELY following the Main Turbine trip

D. (1) the Reactor
(2) as soon as the Main Turbine reaches 1200 RPM

Proposed Answer:	D

Explanations:	
A.	First part is plausible because Plant Cooling Water is the heat sink for the Turbine Cooling Water HX, therefore cools portions of the Main Turbine. However since it is also the heat sink for the Nuclear Cooling Water HXs, the Reactor must be tripped. Second part is plausible if it is thought that the quicker the Main Turbine is stopped, the less damage a high temperature condition will cause.
B.	First part is plausible because Plant Cooling Water is the heat sink for the Turbine Cooling Water HX, therefore cools portions of the Main Turbine. However since it is also the heat sink for the Nuclear Cooling Water HXs, the Reactor must be tripped. Second part is correct.
C.	First part is correct. Second part is plausible if it is thought that the quicker the Main Turbine is stopped, the less damage a high temperature condition will cause.
D.	Correct

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	27138 – Given the Loss of Cooling Water AOP is being performed determine the appropriate mitigating strategies for a loss of plant cooling water in accordance with 40AO-9ZZ03	

Technical Reference:	40AO-9ZZ03, Loss of Cooling Water

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3.0 PLANT COOLING WATER

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 5. **IF** the PW system is **NOT** restored, **AND** the Reactor is at power, **THEN** perform the following: a. Trip the Reactor. b. PERFORM 40EP-9EO01, Standard Post Trip Actions.	

Technical Reference:	40AO-9ZZ03, Loss of Cooling Water, Appendix B, Minimize Cooling Load on TC

Appendix B, Minimize Cooling Load on TC

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 15. **WHEN** Main Turbine speed is less than 1200 rpm, **THEN** perform the following: a. Ensure that Air Removal Pump D is in "PULL TO LOCK". b. Ensure that Air Removal Pumps A, B and C are stopped. c. Announce the following using the unit page system: "All personnel stay clear of the Turbine Building 140 ft elevation North side while breaking vacuum." d. Open the Main Condenser Vacuum Breakers.	

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Instrument Air: Ability to operate and/or monitor the following as they apply to the Loss of Instrument Air: RPS	Tier	1		
	Group	1		
	K/A	065 AA1.05		
	IR	3.3		

Question 17

Given the following conditions:

- Unit 3 is operating at 100% power
- A loss of Instrument Air and Service Air has caused system pressure to lower to 50 psig

With NO operator action, the Reactor should AUTOMATICALLY trip on…

A. Low DNBR

B. Variable Overpower

C. High Pressurizer Pressure

D. Low Steam Generator Water Level

Proposed Answer:	D

Explanations: **The Reactor will trip on Low Steam Generator Water Level because both MFPs trip.**	
A.	Since MFPs trip, steam generator water levels will lower causing the RCS to heat up. Also power will rise from HDP discharge valves closing causing DNBR to lower.
B.	HDP discharge valves close and will cause a minor rise in Reactor power but will not rise to an automatic Variable Overpower Reactor trip.
C.	SBCV valves fail closed and MFPs will trip sequentially therefore there Pressurizer pressure will rise. Pressurizer spray valves will fail closed once IA pressure lowers to 38-48 psig. Since they will still operate at 50 psig, Pressurizer pressure will not rise to the automatic Reactor trip setpoint
D.	Correct

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	25935 – Determine the major effects on plant operation as instrument air pressure degrades	

Technical Reference:	40AO-9ZZ06, Loss of Instrument Air

Appendix A, Expected Component Failure as System Pressure Drops

PRESS	COMPONENT	ACTION
80 psig SR		____ 3. **IF** transferring resin from the High or Low Activity Spent Resin Tank to **ANY** of the following: • Portable RW System, • Waste Feed Tank, **THEN** <u>perform</u> the following to prevent overfilling the Spent Resin Tank: a. <u>Stop</u> SRN-P01, Dewatering Pump. b. **WHEN** it is desired to stop line flushing, **THEN** <u>close</u> SRN-HV-402, Dewatering Pump SRN-P01 Seal Flush Valve.
75 - 65 psig ED	EDN-UV-40 / 41 / 42 / 43 / 44 / 45 / 46, Extraction Steam Air Operated Drain Valves EDN-UV-29 / 30 / 33 / 34 / 36, Feedwater Htr Stm Line Motor Operated Drn Valves	NOTE ARDV-1, Main Turbine Front Standard Turbine Trip Air Relay Dump Valve sends the same signal as during a turbine trip to MTNPSL900 & 901 to open the valves listed.
FT	FTN-PV-75, Auxiliary Steam Pressure Control Valve to Feed Pump Turbine (FC)	NOTE Valve normally closed during plant operation
FW	FWN-FV-1 / 2, Main Feedwater Pump Mini-flow Recirc Valves	NOTE FWN-FV-1 / 2, Main Feedwater Pump Mini-flow Recirc Valves will begin to open, possibly causing a reduction in Main Feedwater Pump Suction Pressure, lowering Steam Generator levels, and a Main Feedwater Pump trip.

Examination Outline Cross-Reference:	**Level**	**RO**	[redacted]	**SRO**
K/A: Generator Voltage and Electric Grid Disturbances: Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Sensors, detectors, indicators	**Tier**	**1**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**077 AK2.03**		
	IR	**3.0**	[redacted]	

Question 18

Given the following conditions:

- Unit 1 is operating at 100% power
- Main Generator MVARs are at UNITY

Subsequently:

- A transmission line relaying has caused grid voltage to rise

(1) Main Generator MVARs should initially be…

(2) Main Generator MVARs should be restored to UNITY…

A. (1) BUCKING
(2) by a manual voltage adjustment

B. (1) BUCKING
(2) by the Auto Voltage Regulator

C. (1) BOOSTING
(2) by a manual voltage adjustment

D. (1) BOOSTING
(2) by the Auto Voltage Regulator

Proposed Answer:	**A**

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because the AC Regulator will maintain Generator terminal voltage. As system load changes, the terminal voltage will need to be adjusted with a manual adjustment.
C.	First part is plausible because when grid voltage changes, MVARS will no longer be in Unity. Since grid voltage rises it may be assumed that the Main Generator will react in the same way and will be Boosting. Second part is correct.
D.	First part is plausible because when grid voltage changes, MVARS will no longer be in Unity. Since grid voltage rises it may be assumed that the Main Generator will react in the same way and will be Boosting. Second part is plausible because the AC Regulator will maintain Generator terminal voltage. As system load changes, the terminal voltage will need to be adjusted with a manual adjustment.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**5**	
Reference Provided:	**N**	
Learning Objective:	**367777 – Explain the operation of the EX2100e Voltage Regulators**	

EO: 1.19 State the meanings of VARs out, VARS in, +Vars, -Vars, boost VARs, buck VARs, lagging VARs, and leading VARS.

Introduction
Several different terms/designators are used to identify the type of and magnitude of reactive power that a generator is carrying. The terms presented here are used by personnel in the control rooms and ECC.

Main Idea
COMMON TERMS USED WHEN REFERRING TO REACTIVE POWER (VARS).

VARS when current is actually lagging the voltage.

- VARS out.
- Positive VARS (+VARS).
- Lagging VARS.
- Boost.

VARS when current is actually leading the voltage.

- VARS in.
- Negative VARS (-VARS).
- Leading VARS.
- Buck.

EO: 1.20 Discuss how ECC determines MVAR changes at Palo Verde

Main Idea
VARs flow whenever the circuit has inductance or capacitance. Capacitance creates VARs, inductance consumes VARs.

Electric motors need inductive VARs to set up magnetic fields. Transmission lines need inductive VARs to create magnetic fields around them. As load on a line goes up; current goes up; voltage gets lower; and more VARs are consumed by the line. This causes a need for greater VAR support.

As inductive loads are added on the system, a large inductive VAR demand is created. This causes current from the generator to go up due to its internal reactance and a drop in terminal voltage. The solution is to raise field excitation. This produces more VAR output and terminal voltage goes back up. VAR flow magnitude will be determined by the difference between voltage at the generator terminal and voltage at the load. Equal voltages would create zero VAR flow. "VARs flow downhill to voltage."

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Continuous Rod Withdrawal: Knowledge of the operational implications of the following concepts as they to Continuous Rod Withdrawal: Integral rod worth	Tier	1		
	Group	2		
	K/A	001 AK1.21		
	IR	2.9		

Question 19

Given the following conditions:

- Unit 3 is operating at 50% power
- The crew is recovering from a loss of the ‘A’ MFP
- The OATC is withdrawing CEAs to restore overlap per 40AO-9ZZ09, Reactor Power Cutback (Loss of Feedpump)
- The selected Group is 30 inches withdrawn
- After the RO lets go of the withdrawal switch, CEA 18 continues to withdraw

(1) As CEA 18 continues to withdraw, its integral rod worth available to insert should…

(2) If all actions to stop CEA 18 were unsuccessful, the crew should…

A. (1) increase
(2) trip the Reactor

B. (1) increase
(2) manually insert all other CEAs in the selected group

C. (1) decrease
(2) trip the Reactor

D. (1) decrease
(2) manually insert all other CEAs in the selected group

Proposed Answer:	**A**

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because SDM will still be met if the CEA completely withdraws. It may be assumed that because Reactor power is 50% a single CEA withdrawing will not cause power to exceed our thermal power limit so inserting the remaining CEAs in the group will allow the crew to maintain the Reactor on line and possible troubleshoot the malfunctioning CEA..
C.	First part is plausible because as a CEA is withdrawn differential rod worth will eventually decrease, it may be assumed that integral rod worth is the same. However as a CEA withdraws, the available integral rod worth will increase. Second part is correct.
D.	First part is plausible because as a CEA is withdrawn differential rod worth will eventually decrease, it may be assumed that integral rod worth is the same. However as a CEA withdraws, the available integral rod worth will increase. Second part is plausible because SDM will still be met if the CEA completely withdraws. It may be assumed that because Reactor power is 50% a single CEA withdrawing will not cause power to exceed our thermal power limit so inserting the remaining CEAs in the group will allow the crew to maintain the Reactor on line and possible troubleshoot the malfunctioning CEA..

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**6**	
Reference Provided:	**N**	
Learning Objective:	**25220 - Given conditions of a CEA Malfunction determine when a Reactor trip is required**	

Technical Reference:	40AO-9ZZ11, CEA Malfunctions

5.0 UNCONTROLLED CEA MOVEMENT MODE 1 OR 2

INSTRUCTIONS | CONTINGENCY ACTIONS

---------- **NOTE** ----------

If a twelve fingered CEA is misaligned from its group, a CEAC penalty factor will ramp up over a six hour period, reducing the margin to trip.

____ 4. **IF ANY** of the following conditions exist:

- CEA movement continues,
- Two or more CEAs are deviating by greater than 9.9 inches from their associated groups,
- CPC ASI exceeds $\pm$0.45 and trending to $\pm$0.5 (Pt ID 0187),
- A CPC DNBR or LPD trip will be received prior to restoring a misaligned 12 Finger CEA (REFER TO Appendix A, CEA Information),

THEN perform the following:

a. Trip the reactor.

b. GO TO 40EP-9EO01, Standard Post Trip Actions.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Inoperable/Stuck Rod: Ability to operate and/or monitor the following as they apply to the Inoperable/Stuck Control Rod: CRDS	**Tier**	**1**		
	Group	**2**		
	K/A	**005 AA1.01**		
	IR	**3.6**		

Question 20

Given the following conditions:

- Unit 1 is operating at 10% power
- A power ascension is in progress
- The OATC is withdrawing Group 5 CEAs in Manual Sequential
- Group 5 CEAs are currently 138 inches

Subsequently:

- The OATC withdraws Group 5 CEAs
- All Group 5 CEAs withdraw to 142.5 inches with the exception of CEA 14
- CEA 14 is stuck at 138 inches

CEA 14 can be verified stuck at 138 inches using ___(1)___. Once troubleshooting is complete and management concurrence is received, per 40AO-9ZZ11, CEA Malfunctions, the crew should re-align CEAs by ___(2)___.

A. (1) RSPTs
(2) withdrawing CEA 14 to 142.5 inches

B. (1) RSPTs
(2) inserting Group 5 CEAs to 138 inches

C. (1) pulse counters
(2) withdrawing CEA 14 to 142.5 inches

D. (1) pulse counters
(2) inserting Group 5 CEAs to 138 inches

Proposed Answer:	A

Explanations: **Pulse counters vs reed switches are commonly confused at PVNGS. It is very plausible for someone to think that the pulse counters are referring to a magnetic pulse that actuates as the rod passes each magnet and that each time the CEDMCS latches and unlatches from upper and lower grippers that the next reed switch in the system is actuated.**	
A.	Correct
B.	First part is correct. Second part is plausible because inserting CEAs is a possibility. However, 40AO-9ZZ11, CEA Malfunctions directs withdrawing a single CE to align with its group.
C.	First part is plausible because pulse counters can normally be used to determine CEA location. However since a withdraw demand was inputted and CEA position did not change, pulse counter indication will be 142.5 inches instead of 138 inches. RSPTs will indicate 140 inches. Second part is correct.
D.	First part is plausible because pulse counters can normally be used to determine CEA location. However since a withdraw demand was inputted and CEA position did not change, pulse counter indication will be 142.5 inches instead of 138 inches. RSPTs will indicate 140 inches. Second part is plausible because inserting CEAs is a possibility. However, 40AO-9ZZ11, CEA Malfunctions directs withdrawing a single CE to align with its group.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**6**	
Reference Provided:	**N**	
Learning Objective:	**22615 – Explain the operation of the RSPTs**	

Technical Reference:	Control Element Drive Mechanism Control System Tech Manual

MG, AS, or MS Operation

Group or sequential mode operation requires simultaneous motion of all subgroup CEAs. The individual CEA enable and pulse count logic performs this function by issuing an insert CEA (ICE) or withdraw CEA (WCE) signal to each of its interconnected CEA timer and coil driver actuating logic cards.

During MG, AS, or MS operation, a subgroup raise (SGR) or subgroup lower (SGL) command is issued to all individual CEA enable and pulse count logic cards within the selected group. It is received from the common logic housing slave subgroup sequencer of that group, where it originates as a control group raise or lower (CGR or CGL) signal.

Technical Reference:	40AO-9ZZ11, CEA Malfunctions

Appendix I, CEA Realignment

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 3. IF the affected CEA can be moved, AND BOTH of the following concur: • Operations Management, • Reactor Engineering,	

____ 3. (continued)

e. Align the affected CEA with the remainder of its group, taking at least the minimum withdrawal time.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Accidental Liquid Radwaste Release: Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following: Radioactive-gas monitors	**Tier**	**1**		
	Group	**2**		
	K/A	**059 AK2.02**		
	IR	**2.7**		

Question 21

Given the following indications:

- A leak on Liquid Radwaste Holdup Tank, LRN-T01C, caused a Lo-Lo Level Alarm at the Liquid Radwaste Annunciator Panel, LRN-E01.

(1) When LRN-T01C, Lo-Lo Level alarm annunciates a trip signal should be sent to…

(2) Airborne radioactivity vented from any Liquid Radwaste Holdup Tank should be detected INITIALLY by ___(2)___

A. (1) ALL Liquid Radwaste Holdup Tank Pumps
(2) RU-143, Plant Vent radiation monitor

B. (1) ALL Liquid Radwaste Holdup Tank Pumps
(2) RU-14, Radwaste Building Ventilation Exhaust Filter Inlet radiation monitor

C. (1) ONLY Liquid Radwaste Holdup Tank Pump, LRN-P01C
(2) RU-143, Plant Vent radiation monitor

D. (1) ONLY Liquid Radwaste Holdup Tank Pump LRN-P01C
(2) RU-14, Radwaste Building Ventilation Exhaust Filter Inlet radiation monitor

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible RU-143 is the Tech Spec radiation monitor that is used to monitor planned releases.
B.	Correct
C.	First part is plausible because if the pump suctions were not cross-tied to all of the tanks, it would make sense that only the pump that is pumping down a tank would actually trip. Second part is plausible RU-143 is the Tech Spec radiation monitor that is used to monitor planned releases.
D.	First part is plausible because if the pump suctions were not cross-tied to all of the tanks, it would make sense that only the pump that is pumping down a tank would actually trip. Second part is correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	11	
Reference Provided:	N	
Learning Objective:	31202 – Given a Radiation Monitor number and name describe the purposes and sample points of the monitor	

Technical Reference:	Liquid Radwaste System Tech Manual

LR Holdup Pump Controls (HS-10, 11, 12)

A three position (stop/start) spring return to unmarked neutral control switch (HS-10, 11, 12) is provided for each pump on panel ZRN-E04 in the radwaste control room. When momentarily placed in the start position, the associated breaker contactor closes, starting the pump. When momentarily placed in the stop position, the associated breaker contactor opens, stopping the pump. A lo-lo level in any one of the three tanks will trip all three pumps. Electrical protection will also trip a pump.

Red and green indicating lights are provided at the panel and at the MCC. Red indicates the breaker contactor is closed; green indicates the breaker contactor is open. Bright green indicates the associated thermal overloads have actuated. Following a loss and restoration of power, the pump(s) will require a manual restart.

Technical Reference:	Radiation Monitoring System Tech Manual

2.13 Radwaste Building Ventilation Exhaust Filter Inlet Monitor, (RBFI) SQN-RU-14

The purpose of the RBFI radiation monitor is to iso-kinetically sample and continuously monitor airborne radioactive particulates and noble gases exhausted for the radwaste building. Isokinetic sampling is conducted in accordance with ANSI N13.1-1969.

Taking into consideration the dilution factor for the contribution from each compartment, the monitor is capable of detecting the presence of the 10CFR20 maximum permissible concentration in any one compartment of the radwaste building within 1 hour for Cs-137 and within 8 hours for I-131. Iodine monitoring is not included in this monitor as there are no significant sources of radio-iodine in the radwaste building (see drawing 13-M-HRP-001).

Examination Outline Cross-Reference:	**Level**	**RO**		**SRO**
K/A: Accidental Gaseous Radwaste Release: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual	**Tier**	**1**		
	Group	**2**		
	K/A	**060 G 2.4.50**		
	IR	**4.2**		

Question 22

During a release of a Waste Gas Decay Tank, a high-high alarm on which of the following Radiation Monitors, INDIVIDUALLY, will require the crew to ensure that the Gaseous Discharge Header Isolation Valves, GR-UV-34A and GR-UV-34B are closed per 74AL-9SQ01, Radiation Monitoring System Alarm Validation and Response?

1. RU-8, Auxiliary Building Ventilation Exhaust Filter Monitor
2. RU-12, Waste Gas Decay Tank Monitor
3. RU-15, Waste Gas System Area Combined Ventilation Exhaust Monitor

A. 1 ONLY

B. 2 ONLY

C. 1 and 3 ONLY

D. 2 and 3 ONLY

Proposed Answer:	**B**

Explanations:	
A.	Plausible because the Auxiliary Building is adjacent to the Radwaste Building and if there was a leak from the Waste Gas header, it is possible that RU-8 will detect it. However the high radiation is in the enclosed Waste Gas system, therefore RU-8 will not detect it.
B.	Correct
C.	Plausible because the Auxiliary Building is adjacent to the Radwaste Building and if there was a leak in the Waste Gas header, it is possible that RU-8 will detect it. However the high radiation is in the enclosed Waste Gas system, therefore RU-8 will not detect it. RU-15 will detect a leak from the Waste Gas header however the high radiation is in the enclosed Waste Gas system, therefore RU-15 will not detect it.
D.	RU-12 is correct. RU-15 will detect a leak from the Waste Gas header however the high radiation is in the enclosed Waste Gas system, therefore RU-15 will not detect it.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**11**	
Reference Provided:	**N**	
Learning Objective:	**18733 – Describe the automatic functions / interlocks with Gaseous Discharge Header Isolation Valves (UV-34A & 34B)**	

Technical Reference:	Radiation Monitoring System Tech Manual

2.11 Waste Gas Decay Tank Monitor, (WGDT) SQN-RU-12

The function of the WGDT radiation monitor is to monitor the gross beta radioactivity level in the decay tank discharge header. A high activity alarm provides an indication of an abnormal operating condition, such as, an inadvertent discharge or incorrect valve lineup. The monitor high-high activity alarm initiates isolation of the decay tank discharge header (see drawing 13-M-GRP-001).

ODCM section 2.1 and table 2-1 apply. Required monitor features for operability are the gas channel, flow rate monitor, control room annunciation and indication, and the automatic isolation actuation function.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Control Room Evacuation: Knowledge of the interrelations between the Control Room Evacuation and the following: Auxiliary shutdown panel layout	**Tier**	**1**		
	Group	**2**		
	K/A	**068 AK2.01**		
	IR	**3.9**		

Question 23

Per 40AO-9ZZ19, Control Room Fire, the crew should actuate MSIS ___(1)___ exiting the Control Room and disconnect switches should be taken to LOCAL on the ___(2)___ Remote Shutdown Panel.

A. (1) prior to
(2) A

B. (1) prior to
(2) B

C. (1) after
(2) A

D. (1) after
(2) B

Proposed Answer:	**B**

Explanations:	
A.	First part is correct. The second part is plausible because ‘A’ Remote Shutdown Panel is similar to the ‘B’ panel with the exception of the disconnect switches.
B.	Correct
C.	First part is plausible because an MSIS can be actuated from the RSD panels and if there is a Control Room fire it is important to evacuate the CR as quickly as possible. The second part is plausible because ‘A’ Remote Shutdown Panel is similar to the ‘B’ panel with the exception of the disconnect switches.
D.	First part is plausible because an MSIS can be actuated from the RSD panels and if there is a Control Room fire it is important to evacuate the CR as quickly as possible. Second part is correct.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**8**	
Reference Provided:	**N**	
Learning Objective:	**26655 – State the indications available to the operator at the Remote Shutdown Panel (RSP)**	

Technical Reference:	40AO-9ZZ19, Control Room Fire

3.0 CONTROL ROOM FIRE

INSTRUCTIONS | CONTINGENCY ACTIONS

____ 1. Enter AOP Entry Time and Date:

____

-- NOTE --

40EP-9EO01, Standard Post Trip Actions, is NOT performed when the Control Room is evacuated.

-- NOTE --

Steps 2 through 5 are expected to be performed in the Control Room.

____ 2. Perform the following:

a. Trip the Reactor.

b. Check that power is lowering.

c. Check that all full strength CEAs are inserted.

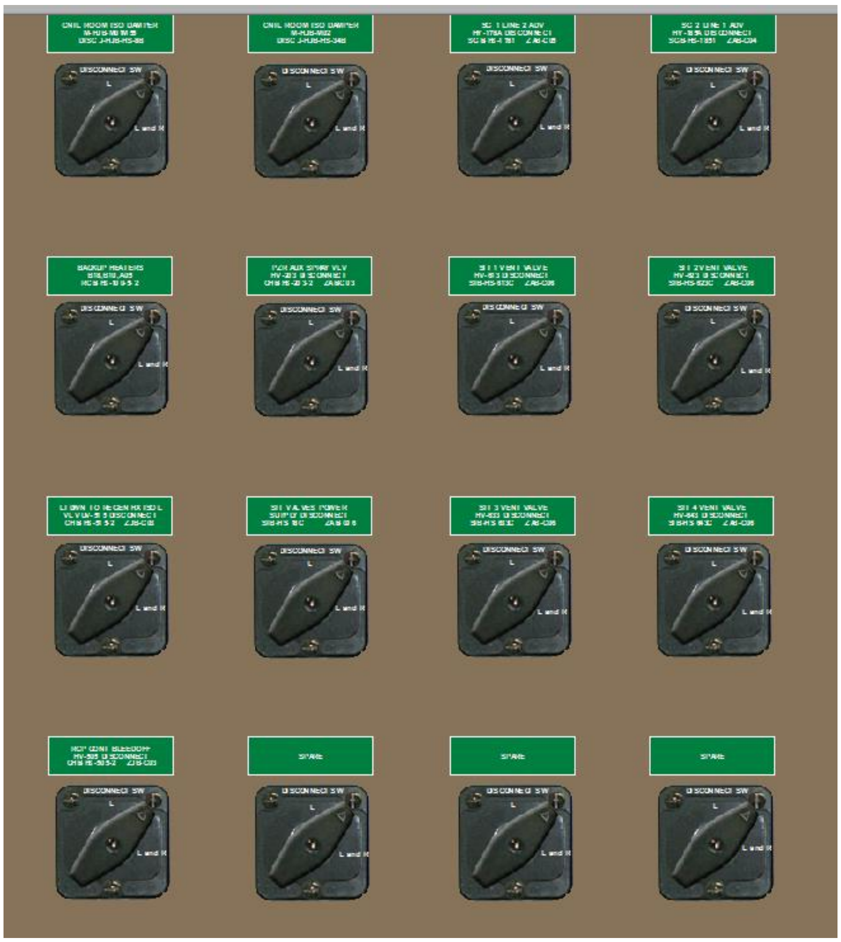
d. Enter the time of the trip:

____

____ 3. Initiate MSIS.

Technical Reference:	Simulator Computer Drawing

REMOTE SHUTDOWN PANEL
CHANNEL B
1-J-ZJB-E01



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Containment Integrity: Knowledge of the reasons for the following responses as they apply to the Loss of Containment Integrity: Guidance contained in EOP for loss of containment integrity	**Tier**	**1**		
	Group	**2**		
	K/A	**069 AK3.01**		
	IR	**3.8**		

Question 24

Given the following conditions:

- Unit 3 was manually tripped due to a LOCA on RCP 1B HP Seal Cooler
- All RCPs were stopped
- RCP Controlled Bleedoff was isolated
- The power supply to RCP 1B HP Seal Cooler isolation valves, NHN-M10 faulted
- When the crew attempted to close NCB-UV-403 NCWS Return Internal Isolation Valve, it failed to close

(1) To maintain Containment integrity the crew should close at a MINIMUM…

(2) Closing the correct NC valve should…

A. (1) NCB-UV-401 NCWS Supply External Isolation Valve
(2) isolate the RCS leak

B. (1) NCB-UV-401 NCWS Supply External Isolation Valve
(2) restrict the RCS leak to Containment ONLY

C. (1) NCA-UV-402 NCWS Return External Isolation Valve
(2) isolate the RCS leak

D. (1) NCA-UV-402 NCWS Return External Isolation Valve
(2) restrict the RCS leak to Containment ONLY

Proposed Answer:	D

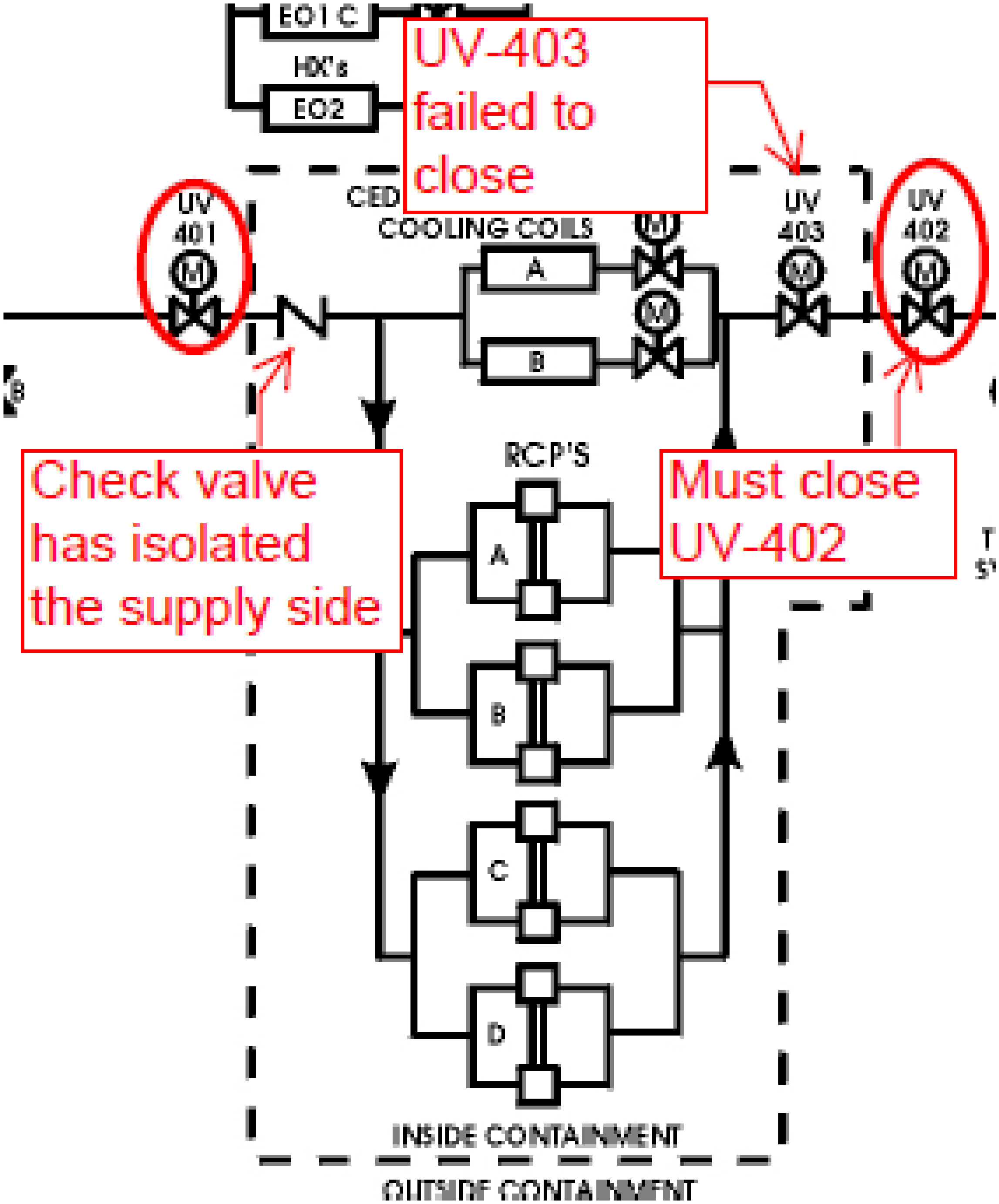
Explanations:	
A.	First part is plausible because NCB-UV-401 is a Containment Isolation Valve that the crew will attempt to close. However there is a check valve in line with NCB-UV-401 that will isolate the NC Supply. Second part is plausible because the RCS leak will be isolated to Containment. However, there is a relief valve that will lift inside of Containment, therefore the leak will continue inside of Containment.
B.	First part is plausible because NCB-UV-401 is a Containment Isolation Valve that the crew will attempt to close. However there is a check valve in line with NCB-UV-401 that will isolate the NC Supply. Second part is correct.
C.	First part is correct. Second part is plausible because the RCS leak will be isolated to Containment. However, there is a relief valve that will lift inside of Containment, therefore the leak will continue inside of Containment.
D.	Correct

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	23790 – Explain the operation of the NC Containment Isolation Valves under normal operating conditions	

Technical Reference:	Nuclear Cooling Water System Technical Manual



Technical Reference:	40EP-9EO03, Loss of Coolant Accident

INSTRUCTIONS	CONTINGENCY ACTIONS

* 10. **IF ANY** of the following conditions exist:

   - RU-6, Nuclear Cooling Water Radiation Monitor alarming
   - An abnormal rise in Nuclear Cooling Water surge tank level

   THEN perform the following:

   a. Stop all RCPs.

   b. Close the Nuclear Cooling Water Containment Isolation Valves.

   c. Isolate controlled bleedoff from the RCPs.

   d. Energize the RCP HP Cooler Isolation Valves for **ANY** leaking RCP High Pressure Cooler(s). REFER TO Appendix 36, RCP HP Seal Cooler Breaker List.

   e. Close the RCP HP Cooler Isolation Valves for **ANY** leaking High Pressure Cooler(s).

   f. Direct Chemistry to sample the Nuclear Cooling Water System for activity.

   g. **IF** the LOCA has been isolated by the isolation of any RCP HP Cooler,
   AND restoration of Nuclear Cooling Water to CTMT is desired,
   THEN open the Nuclear Cooling Water Containment Isolation Valves.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: High Reactor Coolant Activity: Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Location or process point that is causing alarm	**Tier**	**1**		
	Group	**2**		
	K/A	**076 AA2.01**		
	IR	**2.7**		

Question 25

An RMS alarm on ___(1)___ , which monitor(s) radiation levels of ___(2)___, is(are) the primary RMS indication(s) of high reactor coolant activity and possible fuel failure.

A. (1) Primary Coolant Activity Monitors, RU-150/151
   (2) each RCS hot leg

B. (1) Primary Coolant Activity Monitors, RU-150/151
   (2) one RCS cold leg of each Steam Generator

C. (1) Reactor Coolant Letdown Line Radiation Monitor, RU-155D
   (2) the letdown line at the inlet of the Letdown Heat Exchanger

D. (1) Reactor Coolant Letdown Line Radiation Monitor, RU-155D
   (2) the letdown line between the Letdown Heat Exchanger and the Ion Exchangers

Proposed Answer:	D

Explanations:	
A.	RU-150/151 is plausible since they are used to determine activity in the primary under post accident conditions, however the primary indicator for high RCS activity under non post accident conditions is RU-155D. Monitored location is plausible because that is where RU-150/151 monitors radiation levels.
B.	RU-150/151 is plausible since they are used to determine activity in the primary under post accident conditions, however the primary indicator for high RCS activity under non post accident conditions is RU-155D. Monitored location is plausible because the first location that high activity coolant will enter from the core is the hot leg.
C.	First part is correct. Plausible that RU-155D would detect radiation upstream of the letdown HX and downstream of the letdown containment isolation valve to provide earlier detection of high RCS activity than the actual monitoring point for RU-155D and while allowing for the isolation of letdown to determine if RU-155D was reading actual activity or the RM was providing false indications of high activity.
D.	Correct

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2016

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	11	
Reference Provided:	N	
Learning Objective:	31202 – Given a Radiation Monitor number and name describe the purposes and sample points of the monitor	

Technical Reference:	Radiation Monitoring System Tech Manual

2.44 Reactor Coolant Letdown Line Radiation Monitor SQN-RE-155D

This monitor is also referred to as an area monitor (ARM). The fourth available channel of penetration leakage monitor (SQN-RU-155) is used to trend the letdown primary coolant activity. This area radiation monitor provides a continuous recording in the control room of reactor coolant gross gamma activity thus providing a measure of fuel cladding integrity. A high alarm is provided in the control room. Local and remote samples in the CVCS provide the primary means for determining RCS activity. The reactor coolant letdown line monitor serves only as a trending device to warn the operator of possible fuel cladding failure. Verification of the ARM is done by grab sample measurement.

2.42 Primary Coolant Activity Monitors, (PCMA) SQA-RU-150 and (PCMB) SQB-RU-151

The primary coolant activity monitors consist of two independent ionization chamber channels. These monitors meet the monitoring requirements of NUREG-0737 and regulatory guide 1.97, Rev 2 for the circulating coolant activity monitors. The purpose of these monitors is to assess activity levels in the primary coolant under post accident conditions. The detectors are physically located next to a cold leg of each of the steam generators. These monitors are included as part of the SRMS, and meet IE qualification requirements as described in IEEE standard 323- 1974. These monitors are equipped with a RIC module to provide monitor control and indication in the event of DCU failure. The RIC is mounted in the SRMS panel in the control room with the balance of the safety related monitor RIC units. The micro-computers for these monitors are located in the control building to take advantage of the low radiation fields afforded by this category I structure (see drawings 13-J-ZCF-006, 13-J-ZCF-005 and 13-J-ZCF-009).

Although these monitors are part of the SRMS, they have no operability requirements per the technical specifications.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Natural Circulation: Ability to determine and interpret the following as they apply to the (Natural Circulation Operations): Adherence to appropriate procedures and operation within the limitations in the Facility's license and amendments	Tier	1		
	Group	2		
	K/A	CE A13 AA2.2		
	IR	2.9		

Question 26

Per 40EP-9EO07, Loss of Offsite Power / Loss of Forced Circulation, natural circulation flow is verified by checking that the RCS is a MINIMUM of ___(1)___ subcooled as indicated by ___(2)___ .

A. (1) 24°F
(2) CET Subcooling

B. (1) 24°F
(2) RCS Subcooling

C. (1) 30°F
(2) CET Subcooling

D. (1) 30°F
(2) RCS Subcooling

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since RCS Subcooling is the normal parameter used to verify subcooling, however when all RCPs are secured, CET Subcooling is the correct indication of subcooling.
C.	First part is plausible since 30°F is the maximum allowed delta-T between Th and max quadrant CET temp for verifying natural circulation, however the minimum allowable subcooling is 24°F. Second part is correct.
D.	First part is plausible since 30°F is the maximum allowed delta-T between Th and max quadrant CET temp for verifying natural circulation, however the minimum allowable subcooling is 24°F. Second part is plausible since RCS Subcooling is the normal parameter used to verify subcooling, however when all RCPs are secured, CET Subcooling is the correct indication of subcooling.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

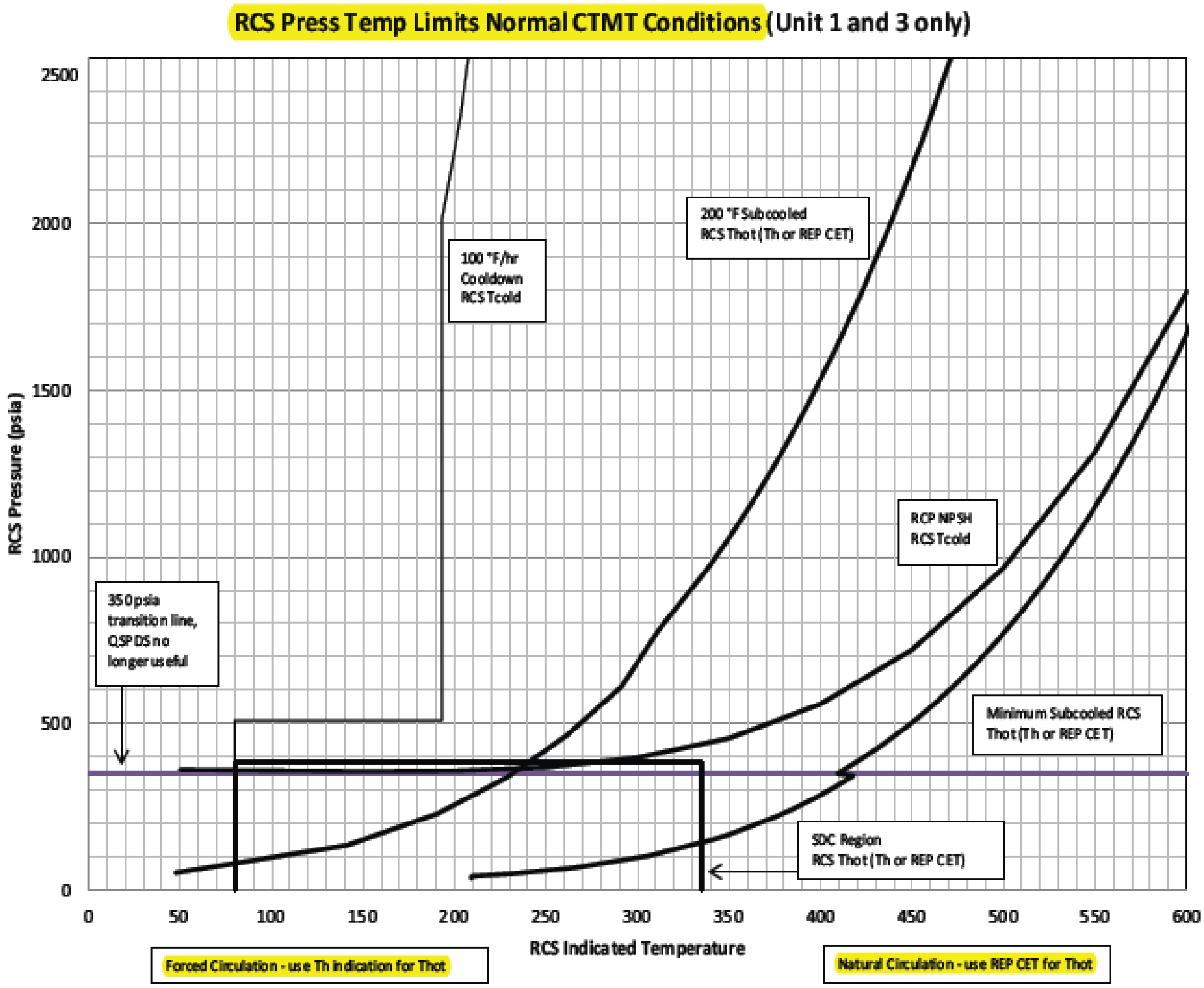
Level of Difficultly:	2	
10CFR55.41:	14	
Reference Provided:	N	
Learning Objective:	62760 – Given a loss of forced circulation, identify the parameters used to determine Natural Circulation flow per 40EP-9EO07.	

Technical Reference:	40EP-9EO07, LOOP/LOFC

INSTRUCTIONS	CONTINGENCY ACTIONS
✱ 15. **IF** RCPs are **NOT** operating, **THEN** check natural circulation flow in at least one loop by **ALL** of the following: • Loop ΔT is less than 65°F • Hot and cold leg temperatures are constant or lowering • RCS is 24°F or more subcooled using CET Subcooling • Less than a 30°F ΔT between T_h RTDs and the maximum quadrant CET temperature (QSPDS, pages 211 and 213)	15.1 Ensure proper control of Steam Generator feeding and steaming.

Technical Reference:	Appendix 2, Figures

RCS Subcooling is calculated using Th indication, CET Subcooling is calculated using REPCET for Th.



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: RCS Overcooling – Pressurized Thermal Shock: Ability to operate and/or monitor the following as they apply to the (RCS Overcooling) Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features	Tier	1		
	Group	2		
	K/A	CE A11 AA1.1		
	IR	3.3		

Question 27

Given the following conditions:

- Unit 1 tripped from 100% power due to an ESD outside of containment upstream of the MSIVs for SG #1
- SG #1 has been isolated per Appendix 113 - Steam Generator 1 Isolation

Using the provided Appendix 2, Figures, on the next page:

- The red dot represents the current RCS temperature and pressure after SG #1 has been isolated

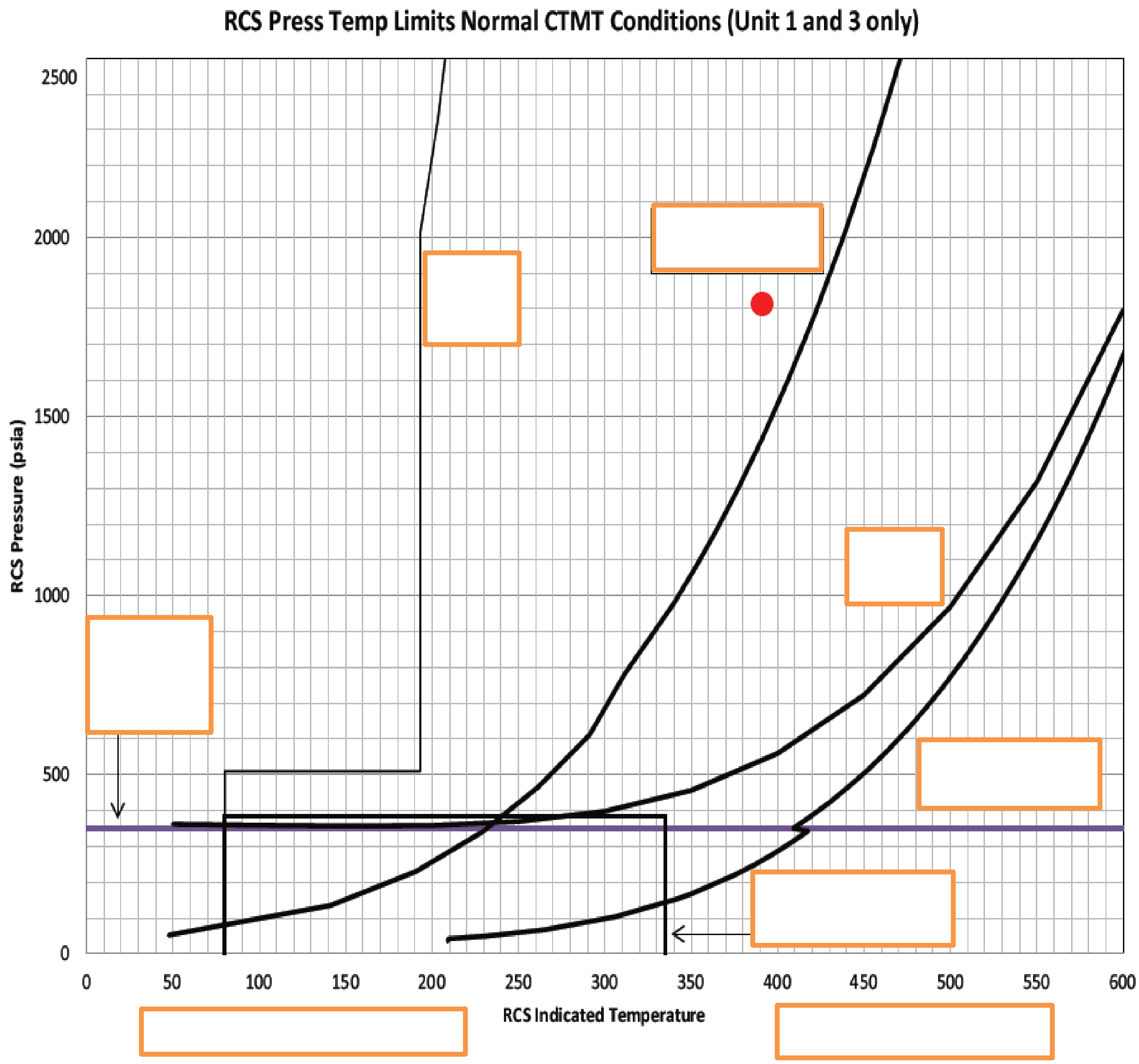
Which of the following is the FIRST action the crew is required to take?

A. Heatup the RCS ONLY

B. Depressurize the RCS ONLY

C. Heatup and depressurize the RCS

D. Perform an RCS soak for 2 hours


RCS Press Temp Limits Normal CTMT Conditions (Unit 1 and 3 only)
RCS Pressure (psia)
2500
2000
1500
1000
500
0
0
50
100
150
200
250
300
350
400
450
500
550
600
RCS Indicated Temperature

Proposed Answer:	B

Explanations:	
A.	Plausible because heating up the RCS will eventually put RCS temperature and pressure to the allowable side of the oversubcooled line. However once the RCS is oversubcooled, there should not be any additional thermal stresses placed on the RCS or Reactor internals.
B.	Correct
C.	Plausible because heating up the RCS will eventually put RCS temperature and pressure to the allowable side of the oversubcooled line. However once the RCS is oversubcooled, there should not be any additional thermal stresses placed on the RCS or Reactor internals. Depresurizing is correct, but it will be the only action taken.
D.	Plausible because since the RCS cooldown rate was violated, it is an action that will need to be taken. However, the RCS temperature and pressure must be on the allowable side of the oversubcooled curve, prior to performing a soak.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	Y	Attached Appendix 2, Figures
Learning Objective:	25501 - Given that the EOPs are being performed and specific plant conditions are given, determine whether or not the plant is oversubcooled, and if it is what actions must be taken per the appropriate procedure	

Technical Reference:	40EP-9EO05, Excess Steam Demand

PALO VERDE NUCLEAR GENERATING STATION EXCESS STEAM DEMAND	40EP-9EO05 Revision 33 Page 13 of 46

INSTRUCTIONS	CONTINGENCY ACTIONS
✱ 21. Maintain RCS pressure within the P/T limits by performing the following. REFER TO Appendix 2, Figures:	21.1 IF the RCS exceeds the P/T limits, THEN perform the following to restore pressure and temperature to within the P/T limits: REFER TO Appendix 2, Figures
a. Control the cooldown rate.	a. IF a cooldown is in progress, THEN stop the cooldown.
b. Control pressurizer heaters and main or auxiliary spray.	b. Depressurize the RCS using main or auxiliary pressurizer spray.
c. IF Main Spray is being used with less than four RCPs running, OR Auxiliary Spray is being used, THEN PERFORM Appendix 6, Spray Valve Actuation Data Sheet.	c. IF SI throttle criteria are met, THEN control charging, letdown, and HPSI flow.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Reactor Coolant Pump: Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP	Tier	2		
	Group	1		
	K/A	003 A2.02		
	IR	3.7		

Question 28

Given the following conditions:

- Unit 1 was tripped from 100% power due to a malfunction on RCP 1A
- RCP 1A was manually tripped during SPTAs per 40AO-9ZZ04, RCP Emergencies
- The CRS has entered 40EP-9EO02, Reactor Trip

(1) When securing a RCP per 40AO-9ZZ04, the RCP Oil Lift Pump should automatically start and then…

(2) If there was NO RCP malfunction and the crew entered 40OP-9ZZ10, Mode 3 to Mode 5 Operations, the FIRST RCP should be stopped once RCS temperature is lowered to a MAXIMUM of…

A. (1) automatically stop
(2) 350°F

B. (1) automatically stop
(2) 500°F

C. (1) will need to be manually stopped
(2) 350°F

D. (1) will need to be manually stopped
(2) 500°F

Proposed Answer:	**D**

Explanations:	
A.	First part is plausible because per 40OP-9RC01, Reactor Coolant Pump Operations, RCN-P02A, RC Pump 1A Oil Lift Pump will automatically stop after an RCP is started within 2 minutes. Second part is plausible because 350°F is MODE 3 entry and two RCPs are required to be stopped per 40OP-9ZZ10, Mode 3 to Mode 5 Operations.
B.	First part is plausible because per 40OP-9RC01, Reactor Coolant Pump Operations, RCN-P02A, RC Pump 1A Oil Lift Pump will automatically stop after an RCP is started within 2 minutes. Second part is correct.
C.	First part is correct. Second part is plausible because 350°F is MODE 3 entry and two RCPs are required to be stopped per 40OP-9ZZ10, Mode 3 to Mode 5 Operations.
D.	Correct

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**3**	
Reference Provided:	**N**	
Learning Objective:	**22613 – Explain the operation of the Reactor Coolant Pumps Lube Oil system under normal operating conditions**	

NOTE

___ • The Low Steam Generator Pressure Pre-Trip alarms may occur several times during the cooldown.

• Reset of the Low Steam Generator Pressure Trip setpoints is permissible to be performed prior to receiving the Low Steam Generator Pressure Pre-Trip alarms.

___ 6.2.38 **WHEN** ANY of the following alarms actuate:

___ • 5A07D, LO SG 1 PRESS CH PRE-TRIP

___ • 5A08D, LO SG 2 PRESS CH PRE-TRIP

THEN reset ALL of the following PPS Low Steam Generator Trip setpoints:

___ • Lo SG Press Setpoint Reset at SAA-UIC-37, PPS Remote Operator Module A

___ • Lo SG Press Setpoint Reset at SAB-UIC-38, PPS Remote Operator Module B

___ • Lo SG Press Setpoint Reset at SAC-UIC-39, PPS Remote Operator Module C

___ • Lo SG Press Setpoint Reset at SAD-UIC-40, PPS Remote Operator Module D

NOTE

___ RCPs 1A and 1B are the preferred pumps to leave running to maintain maximum availability of Main Spray capability.

___ 6.2.39 **WHEN** RCS temperature is less than or equal to 505°F, **THEN** ensure NO more than 3 RCPs are running.

Technical Reference:	40OP-9RC01, Reactor Coolant Pump Operations

CAUTION

___ Starting an RCP with any PK battery disconnected from its respective bus may result in a loss of the PN distribution panel supplied from that PK bus if it is being supplied from the inverter.

___ 6.1.37 **IF** any PK battery is disconnected from its respective bus,
THEN obtain Shift Manager approval signature prior to starting RCP 1A.

Signature ______________________________ Date ____________
Shift Manager

___ 6.1.38 Observe RCN-P02A, RCP 1A Oil Lift Pump, stops 2 minutes after RCP 1A starts.

3.0 ABNORMAL RCP MOTOR OR BEARING PARAMETERS

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 5. **IF** the RCP parameters indicated on RMN-TJR-2 points 1-32 exceed any of the trip setpoints listed in Appendix A, RCP Motor Or Bearing Trip Setpoints, **THEN** perform the following: a. Ensure the Reactor is tripped. b. Stop the affected RCP. c. GO TO the appropriate procedure for current plant conditions.	
____ 6. **IF** any RCP motor or bearing parameter is trending to a trip setpoint (REFER TO Appendix D, Instrumentation and Setpoints), **AND** the CRS determines a plant shutdown or cooldown is needed, **THEN** perform **BOTH** of the following: • The appropriate procedure to shutdown or cooldown the plant • 40OP-9RC01, Reactor Coolant Pump Operation, to stop the affected RCP	

Technical Reference:	

___ 6.2.46 WHEN RCS temperature is less than or equal to 350°F,
THEN perform the following:

> **NOTE**
>
> ___ RCPs 1A and 1B are the preferred pumps to leave running due to Pressurizer spray.

___ A. IF a forced cooldown of the Steam Generators will NOT be performed, THEN stop the third RCP per 40OP-9RC01, Reactor Coolant Pump Operation.

> **NOTE**
>
> ___ The combination of RCP 1A/2A or RCP 1B/2B are the preferred pumps for operation if performing a forced cooldown of the Steam Generators.

___ B. IF a forced cooldown of the Steam Generators will be performed per Appendix G - Forced Cooldown of the Steam Generators, THEN stop the second RCP per 40OP-9RC01, Reactor Coolant Pump Operation, ensuring ONE of the following RCP combinations:

- ___ RCPs 1A and 2A are running
- ___ RCPs 1B and 2B are running

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Chemical and Volume Control: Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: RCPs	**Tier**	**2**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**004 K3.04**		
	IR	**3.7**	[redacted]	

Question 29

Given the following conditions:

- Unit 1 is operating at 100% power
- CHB-UV-515, Letdown to Regenerative Heat Exchanger Isolation Valve fails closed
- The CRS enters 40AO-9ZZ05, Loss of Charging or Letdown

After the crew isolates Seal Injection, which of the following describes the effect (if any) on RCP temperatures?

A. HP Seal Cooler inlet temperature should remain relatively constant, while all other seal temperatures should rise by about 70°F

B. HP Seal Cooler inlet temperature should rise to between 200°F and 220°F, all other seal temperatures should rise by about 70°F

C. HP Seal Cooler inlet temperature should rise to between 200°F and 220°F while all other seal temperatures remain normal

D. Isolating Seal Injection should have NO impact on seal temperatures

Proposed Answer:	C

Explanations:	
A.	Plausible because once Seal Injection is lost, flow stagnates and temperatures will remain the same because heat from the RCP is only being transferred to the water around the seals and not around the temperature indicators. Second part is plausible because Seal Injection will isolate at 70°F. Therefore if it lost and Seal Injection temperature was at its minimum, temperature of seals will rise by approximately 70°F. Also, if NC flow is lost along with Seal Injection all seal temperatures will rise.
B.	First part is correct. Second part is plausible because Seal Injection will isolate at 70°F. Therefore if it lost and Seal Injection temperature was at its minimum, temperature of seals will rise by approximately 70°F. Also, if NC flow is lost along with Seal Injection all seal temperatures will rise
C.	Correct
D.	Plausible because once Seal Injection is lost, flow stagnates and temperatures will remain the same because heat from the RCP is only being transferred to the water around the seals and not around the temperature indicators.

Question Source:		New	
	X	Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	3	
Reference Provided:	N	
Learning Objective:	26284 – Given a RCP with seal injection removed, determine the temperature response when seal injection is secured to an RCP in accordance with 40AO-9ZZ04 or 40AO-9ZZ05	

Technical Reference:	40AO-9ZZ04, Reactor Coolant Pump Emergencies

4.0 ABNORMAL RCP SEAL PARAMETERS

INSTRUCTIONS | CONTINGENCY ACTIONS

____ 1. Enter AOP Entry Time and Date:

____

-------------------- **NOTE** --------------------

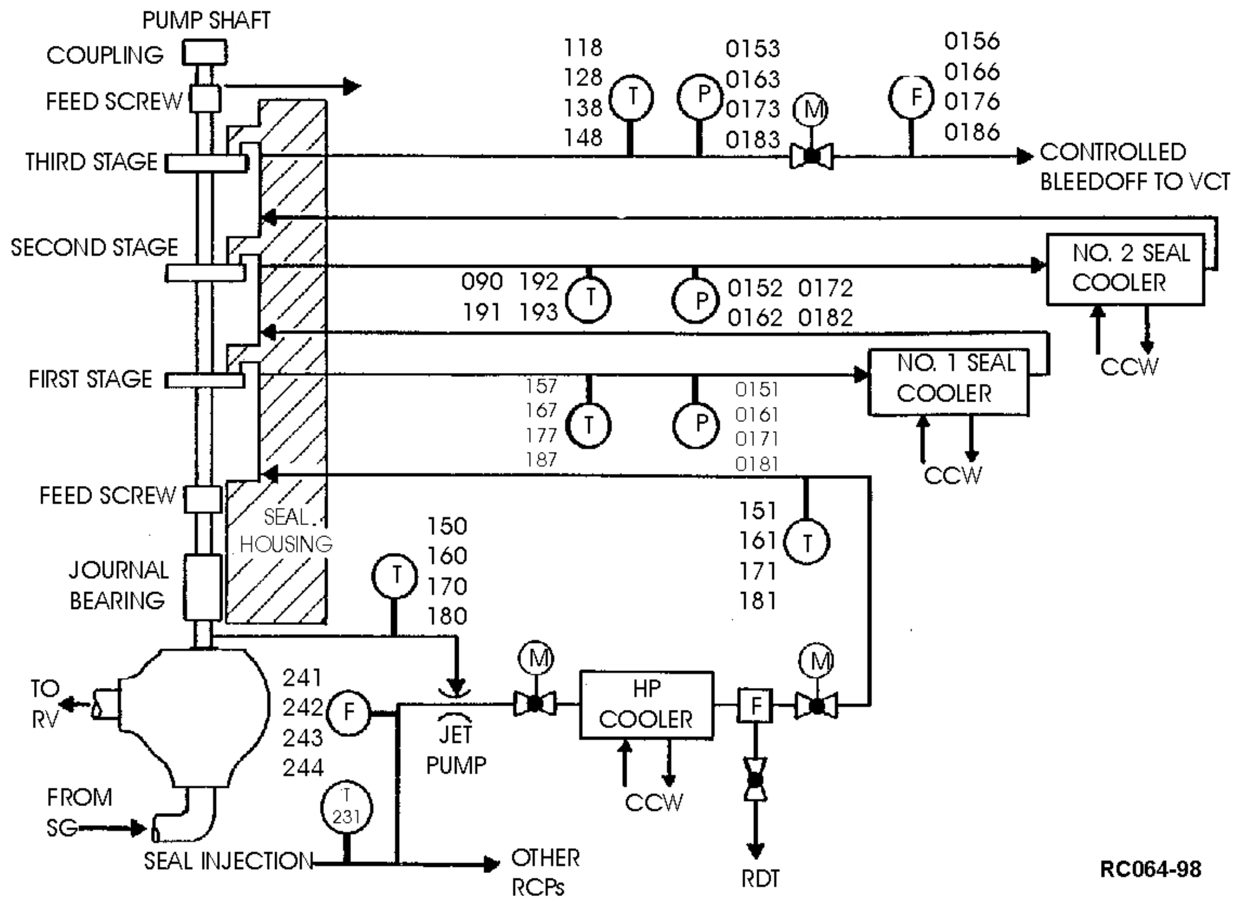
RCP HP Seal Cooler inlet temperature should rise to between 200°F and 220°F if seal injection is stopped. All other seal temperatures should remain normal.

Technical Reference:	Chemical Volume Control System Tech Manual

1.2.2 Reactor Coolant Pump Controlled Bleed-off, Seal Injection, and Chemical Addition Sub-System (Figure 1-4)

A portion of the charging flow is used to supply RCP seal injection. This flow passes through a temperature protection isolation valve which automatically isolates seal injection flow as the temperature downstream of the seal injection heat exchanger decreases to 70°F or increases above 150°F. This is

Technical Reference:	Reactor Coolant System Tech Manual



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Chemical and Volume Control: Knowledge of the effect of a loss or malfunction on the following CVCS components: Seal injection system and limits on flow range	Tier	2		
	Group	1		
	K/A	004 K6.31		
	IR	3.1		

Question 30

Given the following conditions:

- Unit 2 is operating at 100% power
- Seal Injection was isolated when CHB-HV-255, RCP Seal Injection Header Supply Valve, was inadvertently closed
- The CRS entered 40AO-9ZZ04, RCP Emergencies, and has directed restoring Seal Injection per Appendix H, Restoring RCP Seal Injection
- The individual Seal Injection Flow Controllers have been placed in MANUAL and the Seal Injection Flow Control Valves have been closed
- CHB-HV-255, RCP Seal Injection Header Supply Valve, has been reopened

In order to restore Seal Injection, the OATC should raise Seal Injection flow by ___(1)___ OUTPUT on the Seal Injection Flow Controllers to achieve a final target Seal Injection flow of ___(2)___ .

A. (1) raising
(2) 2.0 – 4.0 gpm

B. (1) raising
(2) 6.0 – 7.5 gpm

C. (1) lowering
(2) 2.0 – 4.0 gpm

D. (1) lowering
(2) 6.0 – 7.5 gpm

Proposed Answer:	**D**

Explanations:	
A.	First part is plausible since raising output on a controller usually results in raising flow, however Seal Injection Flow Controllers are reverse acting. Second part is plausible since 2-4 gpm is the normal flowrate for RCP Seal Bleedoff, however the normal flowrate for RCP Seal Injection is 6-75 gpm.
B.	First part is plausible since raising output on a controller usually results in raising flow, however Seal Injection Flow Controllers are reverse acting. Second part is correct.
C.	First part is correct. Second part is plausible since 2-4 gpm is the normal flowrate for RCP Seal Bleedoff, however the normal flowrate for RCP Seal Injection is 6-75 gpm.
D.	Correct.

Question Source:		**New**	
	X	**Bank**	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**3**	
Reference Provided:	**N**	
Learning Objective:	**311397 – Explain Restoration of Seal Injection**	

Technical Reference:	40AO-9ZZ04, RCP Emergencies
c. <u>Throttle open</u> the Seal Injection Flow Controllers by lowering the manual output on the individual flow controllers until RCP seal 1 outlet temperatures start dropping.	

Technical Reference:	**40AO-9ZZ04, RCP Emergencies**

Appendix H, Restoring RCP Seal Injection

INSTRUCTIONS	CONTINGENCY ACTIONS
____ **2.** **(continued)**	
d. **WHEN** temperatures have stabilized at approximately charging line temperature, **THEN** adjust the Seal Injection Flow Controllers to 6.0 to 7.5 GPM.	------------------------- **NOTE** ------------------------- Adequate Seal Injection flow exists when Seal Injection flow is greater than Controlled Bleedoff flow. --- d.1 **IF** RCP Seal Injection flow can **NOT** be adjusted to 6.0 to 7.5 gpm, **THEN** balance flows to ensure that Seal Injection flow is greater than Bleedoff flow to all RCPs.

Technical Reference:	40AO-9ZZ04, RCP Emergencies

Appendix D, Instrumentation and Setpoints

Parameter	Instrument Number	Normal	Alarm	Trip
No. 2 Seal Inlet Pressure	RCN-PT-152/162 (RCN-PI-152 on B04) RCN-PT-172/182 (RCN-PI-172 on B04)	See Appendix G	Lo 826 psig	-
			Hi 1766 psig	-
No. 2 Seal Outlet Pressure (Controlled Bleedoff)	RCN-PT-153/163 (RCN-PI-153 on B04) RCN-T-173/183) (RCN-PI-173 on B04)	See Appendix G	Lo 179 psig	-
			Hi 537 psig	-
Controlled Bleedoff Flow	RCN-FI-156/166/176/186 (B03)	2.0 - 4.0 gpm	Lo 1.6 gpm	-
			Hi 6.0 gpm	Hi ≥9.5 gpm

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Residual Heat Removal: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: Heatup/cooldown rates	Tier	2		
	Group	1		
	K/A	005 A1.01		
	IR	3.5		

Question 31

Given the following conditions:

- Unit 1 is in MODE 4
- The crew is placing SDC in service using the Train ‘A’ LPSI Pump
- The CRS directs warming up the Train ‘A’ SDCHX at the MAXIMUM heat up rate allowed by 40OP-9SI01, Shutdown Cooling Initiation
- SIA-HV-306, LPSI S/D Cooling HX A Bypass Valve, is 20% open
- The ‘A’ LPSI Pump has been started
- SIA-UV-635, LPSI Header A to RC Loop 1A, is 10% open

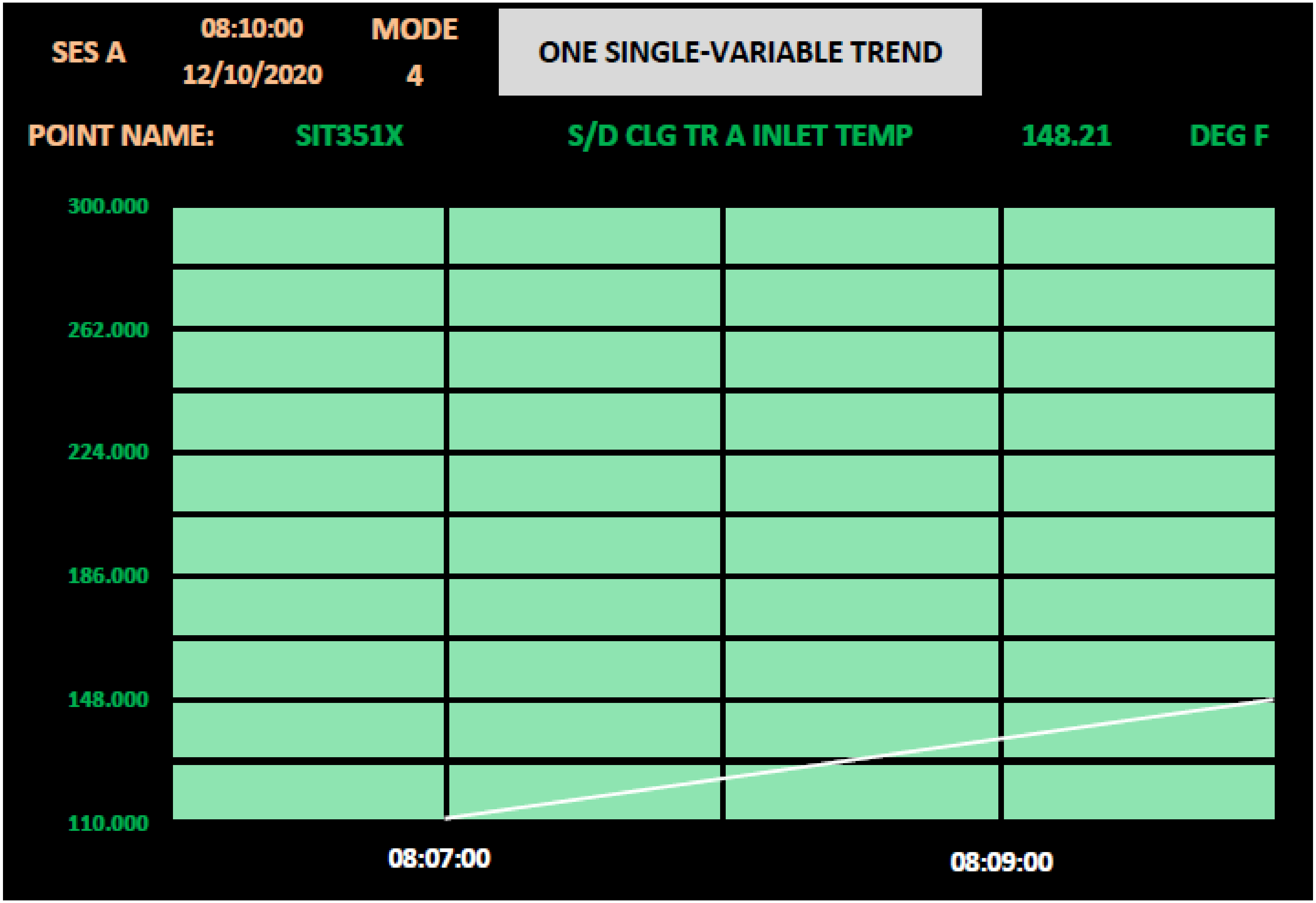
Based on the trend on the following page, in order to comply with the CRS direction, the ‘A’ SDCHX heat up rate should be ___(1)___ and the crew can accomplish this by throttling ___(2)___ on SIA-HV-306, LPSI S/D Cooling HX A Bypass Valve.

A. (1) raised
   (2) open

B. (1) raised
   (2) closed

C. (1) lowered
   (2) open

D. (1) lowered
   (2) closed


SES A
08:10:00
12/10/2020
MODE
4
ONE SINGLE-VARIABLE TREND
POINT NAME:
SIT351X
S/D CLG TR A INLET TEMP
148.21
DEG F
300.000
262.000
224.000
186.000
148.000
110.000
08:07:00
08:09:00

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible since HV-306 is throttled open to heatup (or slow the cooldown of) the RCS, however to raise the heatup rate of the SDCHX, HV-306 must be throttled closed.
B.	Correct.
C.	First part is plausible since the SDCHX is heating up at a rate of ~ 13°F/min, and the C/D rate limit for the RCS in MODE 4 is 100°F/hr (~ 1.6°F/min), which is could be assumed is the same temperature change limit for the SDCHX, however the heatup rate limit for the SDCHX is 19°F/min. Second part is plausible since opening HV-306 would lower the heatup rate, however in this case, HV-306 needs to be throttled closed to raise the heatup rate of the SDCHX.
D.	First part is plausible since the SDCHX is heating up at a rate of ~ 13°F/min, and the C/D rate limit for the RCS in MODE 4 is 100°F/hr (~ 1.6°F/min), which is could be assumed is the same temperature change limit for the SDCHX, however the heatup rate limit for the SDCHX is 19°F/min. Second part is correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	Y	Attached picture of the SDC Train A Inlet Temperature
Learning Objective:	21607 – Describe the temperature requirements and their their bases for initiating and securing SDC.	

Technical Reference:	40OP-9SI01, Shutdown Cooling Initiation

Step 6.15.23, Continued

___ G. Determine the SDC flow rate for the final SDC alignment using the table below:

Decision Table for Mode 4			
1 Pump Operating		2 Pumps Operating in Opposite Loops	
4,000 gpm to 5,000 gpm		4,000 gpm to 5,000 gpm	
Decision Table for Mode 5 or Mode 6			
1 Pump Operating		2 Pumps Operating in Opposite Loops	
RCS Level	Flow Rate (gpm)	RCS Level	Flow Rate (gpm)
101 ft 6 in to 102 ft	3780 to 4150	Greater than 104 ft	3780 to 5000
102 ft to 103 ft 1 in	3780 to 4600	Less than 104 ft	Not allowed except for loop swaps
Greater than 103 ft	3780 to 5000		

___ H. Monitor SIA-FI-306, LPSI-S/D Cooling A Header Flow to Loops.

___ I. Throttle SIA-HV-306 to achieve ALL of the following using handswitch SIA-HS-306, LPSI S/D Cooling HX A Bypass Vlv HV-306:

- ___ The flow rate determined in Step 6.15.23.G
- ___ RCS cooldown rate determined in Step 6.15.23.D.1
- ___ SIA-E01, Shutdown Cooling Heat Exchanger 1, heatup rate less than 19°F/ minute

Technical Reference:	Safety Injection System Tech Manual

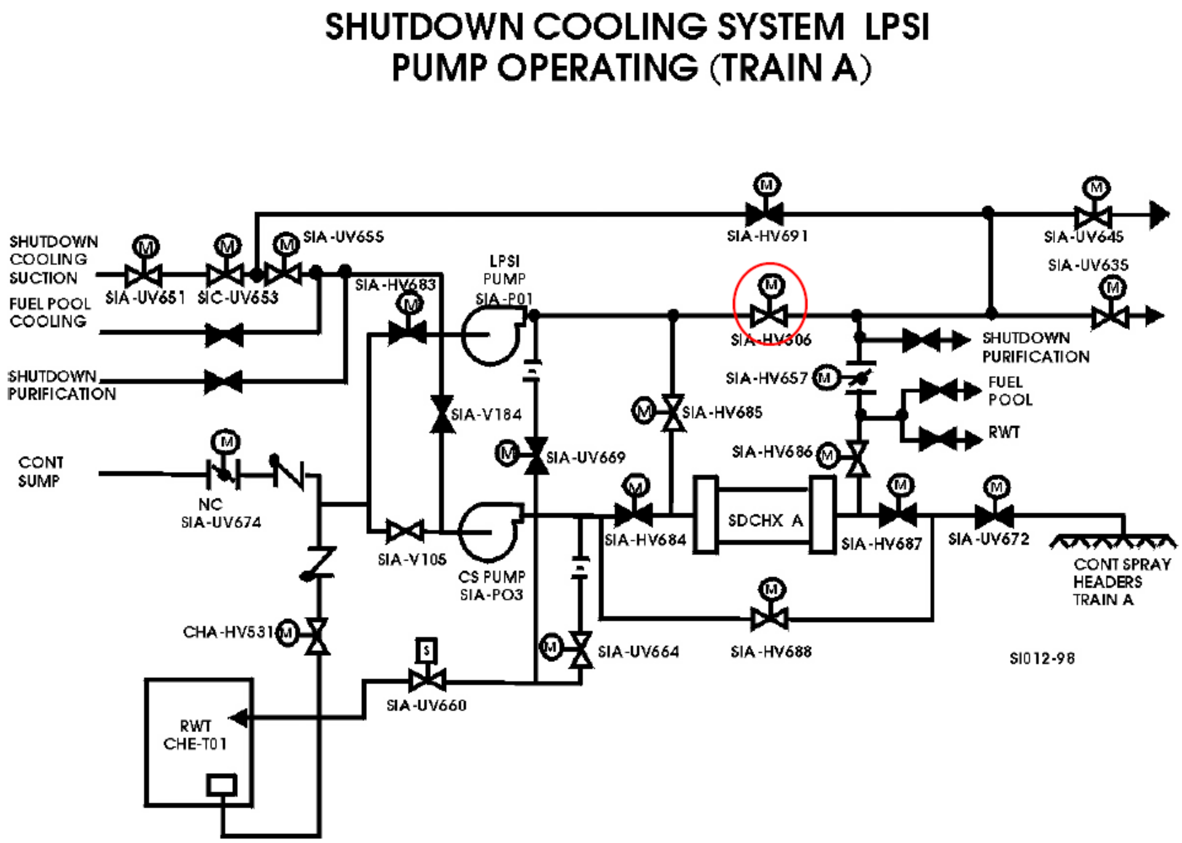


Figure 3 - 1 Shutdown Cooling System - LPSI Pump A Operating

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Residual Heat Removal: Ability to verify that the alarms are consistent with the plant conditions	Tier	2		
	Group	1		
	K/A	005 G 2.4.46		
	IR	4.2		

Question 32

Given the following conditions:

- Unit 2 is in MODE 5
- Train ‘A’ SDC is in service using ‘A’ LPSI Pump

Subsequently:

- The ‘A’ LPSI Pump tripped due to an 86 lockout

The crew should be alerted of the loss of the ‘A’ LPSI Pump by a ___(1)___ on the SESS Panel and annunciator 2B06A, SDC TRAIN A/B FLOW LO, ___(2)___ annunciate.

A. (1) white light AND a blue light
   (2) SHOULD

B. (1) white light AND a blue light
   (2) should NOT

C. (1) white light ONLY
   (2) SHOULD

D. (1) white light ONLY
   (2) should NOT

Proposed Answer:	**D**

Explanations:	
A.	First part is plausible since the blue SESS alarm indicates that a piece of equipment which should be running is not running, however this is only for ESF equipment which is running due to an ESF actuation. Second part is plausible since the trip of the LPSI pump will result in a loss of SDC flow, however in order for that alarm to annunciate, the SDC pump breaker must be closed, therefore on a loss of flow due to a pump trip, the SDC Train A/B Low Flow alarm does not come in.
B.	First part is plausible since the blue SESS alarm indicates that a piece of equipment which should be running is not running, however this is only for ESF equipment which is running due to an ESF actuation. Second part is correct.
C.	First part is correct. Second part is plausible since the trip of the LPSI pump will result in a loss of SDC flow, however in order for that alarm to annunciate, the SDC pump breaker must be closed, therefore on a loss of flow due to a pump trip, the SDC Train A/B Low Flow alarm does not come in.
D.	Correct

Question Source:		**New**	
	X	**Bank**	
		Modified	
	X	**Previous NRC Exam**	**2018**

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

Level of Difficultly:	**3**	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**19358 – Discuss the Shutdown Cooling Low Flow Alarms**	

Technical Reference:	LOIT Safety Injection Lesson Plan

SDC
TRAIN A/B
FLOW
LO

This alarm has two setpoints depending on the number of SDC pumps operating. The flow transmitter senses total SDC loop flow upstream of where the line branches to the individual loop injection valves. The setpoint for one pump is below the minimum Tech Spec SDC flow limit and thus requires immediate attention.

AMBER

One <u>VERY IMPORTANT</u> point about this particular alarm:

If the running LPSI pump trips, this alarm <u>won't</u> come in because breaker position is in the alarm logic circuitry!

The operator would get a SEIS alarm (white inoperable light) should this occur. But if another piece of equipment is already inoperable in that group, then the audible SEIS won't come in. The long and short of it is that this can be a fairly silent loss of SDC flow. That is the reason an audible ERFDADS alarm for SDC flow is established.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Emergency Core Cooling: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation	Tier	2		
	Group	1		
	K/A	006 G 2.1.7		
	IR	4.4		

Question 33

Given the following conditions:

- A LOCA is in progress on Unit 2
- The crew has entered 40EP-9EO03, Loss of Coolant Accident
- Containment pressure is 6.5 psig and rising at 1 psig/min

Per 40DP-9AP16, Emergency Operating Procedure Users Guide:

(1) When Containment pressure approaches the CSAS setpoint the crew should…

(2) When RWT level approaches the RAS setpoint the crew should…

A. (1) let CSAS actuate automatically
   (2) let RAS actuate automatically

B. (1) let CSAS actuate automatically
   (2) manually actuate RAS on trend

C. (1) manually actuate CSAS on trend
   (2) let RAS actuate automatically

D. (1) manually actuate CSAS on trend
   (2) manually actuate RAS on trend

Proposed Answer:	**C**

Explanations:	
A.	First part is plausible because a RAS is an ECCS actuation that needs to be automatically actuated. However, the reason for that is to make sure that there is enough inventory in Containment for the RAS. If a CSAS is imminent, it should be manually actuated on trend per EOP Operations Expectations. Second part is correct.
B.	First part is plausible because a RAS is an ECCS actuation that needs to be automatically actuated. However, the reason for that is to make sure that there is enough inventory in Containment for the RAS. If a CSAS is imminent, it should be manually actuated on trend per EOP Operations Expectations. Second part is plausible because every other ECCS actuation should be manually actuated prior to the auto setpoint.
C.	Correct
D.	First part is correct. Second part is plausible because every other ECCS actuation should be manually actuated prior to the auto setpoint.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**24932 – Given conditions of a LOCA, describe the problems associated with initiating a RAS early per 40EP-9EO03**	

Technical Reference:	Emergency Operating Procedure Users Guide

4.23.2 If plant parameters indicate that an ESFAS actuation is required and did not actuate, then an operator shall manually actuate all channels of the ESFAS signal while notifying other control room personnel of the condition.

4.23.3 If plant parameter trends indicate that an ESFAS actuation is imminent, then an operator should obtain CRS concurrence and actuate that signal manually. RAS shall not be actuated until the RWT level has reached the actuation setpoint.

4.5.57 Step 57 - Ensure RAS

A. For breaks inside containment, if the RWT level falls to the RAS setpoint the operator should ensure that RAS is actuated. Recirculation is actuated in order to maintain a continuous flow of SI fluid to the RCS and a continuous flow of CS.

B. Contingency Actions

1. If the signal did not automatically actuate the operator should manually actuate a RAS.

4.5.58 Step 58 - Take Actions for RAS

A. The operator should be cautioned against prematurely initiating RAS. A possible complication of a premature RAS is that pump suctions could become air bound, consequently leading to a loss of both heat removal loops. Pressurizer level or reactor vessel level may go lower when the LPSI pumps are stopped by a RAS actuation. Also, the RCS temperatures and containment pressure may rise for some time after the RAS because the injected water from the RWT is no longer removing heat. These are expected trends.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Pressurizer Relief/Quench Tank: Ability to manually operate and/or monitor in the control room: Relationships between PZR level and changing levels of the PRT and bleed holdup tank	Tier	2		
	Group	1		
	K/A	007 A4.09		
	IR	2.5		

Question 34

Given the following conditions:

- Unit 1 Reactor was tripped due to a Pressurizer relief valve stuck full open

One minute after the Reactor trip and with NO operator action, Pressurizer level should be ___(1)___ and ___(2)___ level should be rising.

A. (1) rising
   (2) EDT

B. (1) rising
   (2) RDT

C. (1) lowering
   (2) EDT

D. (1) lowering
   (2) RDT

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible because letdown relief valves discharge to the EDT. Most auxiliary systems will discharge to the EDT while any identified RCS leakage will collect in the RDT.
B.	Correct
C.	First part is plausible because RCS leakage that occurs anywhere but the Pressurizer will cause Pressurizer level to lower. If the leak is anywhere in the steam space, Pressurizer level will rise. Second part is plausible because letdown relief valves discharge to the EDT. Most auxiliary systems will discharge to the EDT while any identified RCS leakage will collect in the RDT.
D.	First part is plausible because RCS leakage that occurs anywhere but the Pressurizer will cause Pressurizer level to lower. If the leak is anywhere in the steam space, Pressurizer level will rise. Second part is correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	5	
Reference Provided:	N	
Learning Objective:	26636 – Given conditions of LOCA, describe how the plant would respond to various types of RCS leaks per 40EP-9EO03	

Once the RCS pressure and temperature are reduced, a determination is made as to whether long term cooling using Shutdown Cooling is appropriate (small break) or if simultaneous hot/cold leg injection in a recirculation mode will continue (large break).

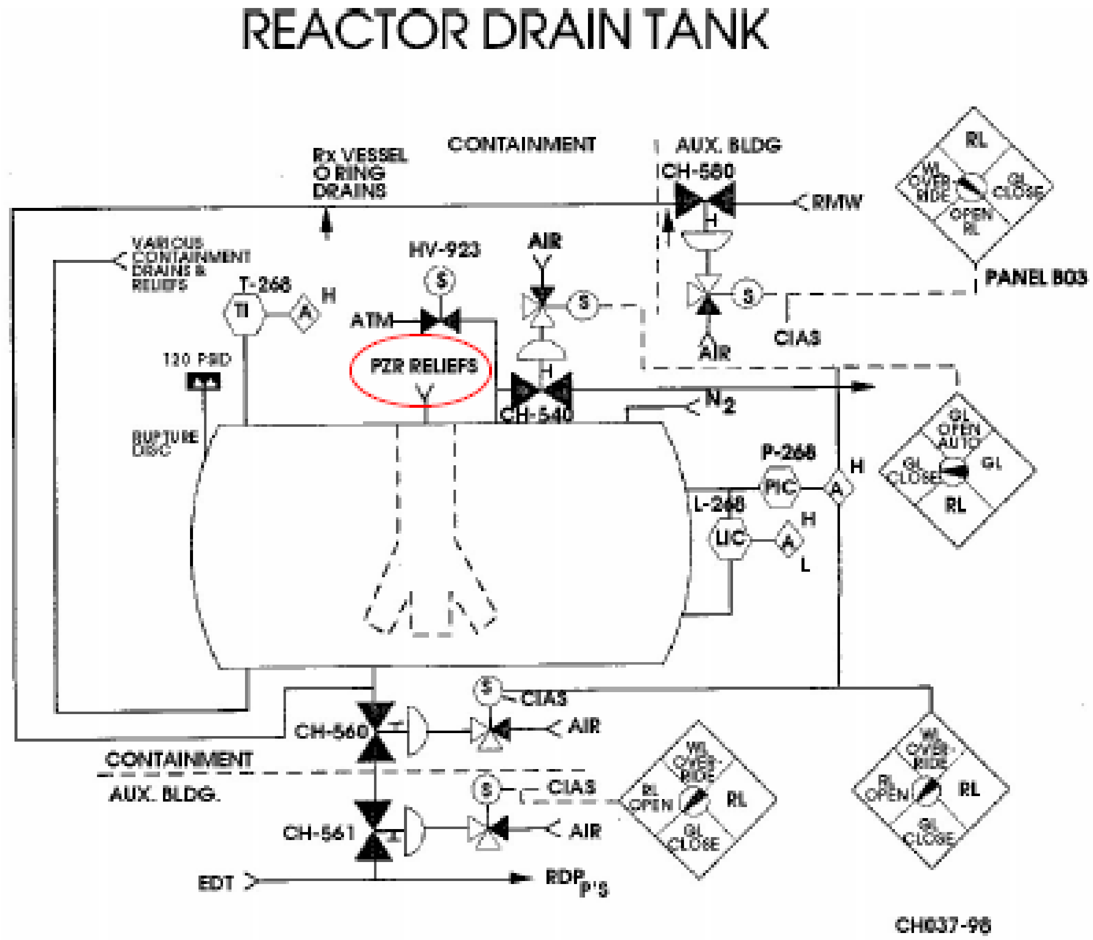
4.3 Entry Conditions

4.3.1 Optimal Recovery Procedure Entry Conditions describe those conditions that are expected to exist for the event mitigated by the procedure.

If the event initiated from Mode 1 or 2, the Standard Post Trip Actions must be completed, and the CRS has made a determination as to the appropriate procedure to enter using the Diagnostic Flowchart. If the event initiated from Mode 3 or 4 and LTOP is not in service, the recovery procedure is entered directly by the CRS making a determination of the event in progress based on available indication.

The plant conditions listed are those expected to exist if a Loss of Coolant Accident was to occur.

Small breaks located in the top of the pressurizer (e.g., stuck open PZR safety valve) will result in flashing and steam production in the reactor vessel and hot legs. This steam will flow towards the break through the pressurizer surge line and oppose the draining of the pressurizer liquid. Thus, the liquid level in the pressurizer may increase or exhibit erratic behavior due to the competing steam-water counter current flow condition. A similar behavior may be observed if the break is in the surge line.



Reactor Drain Tank

Figure 2 – 30

2.4.1 Reactor Drain Tank (CHN-X02) (Figure 2-37)

The reactor drain tank performs the following functions:

- Receives and condenses design discharges from the pressurizer safety valves to prevent the discharge from being released to the containment.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Component Cooling Water: Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: RCS, in order to determine source(s) of RCS leakage into the CCWS	Tier	2		
	Group	1		
	K/A	008 K1.04		
	IR	3.3		

Question 35

Given the following conditions:

- Unit 2 is operating at 100% power
- Train ‘A’ Essential Cooling Water is cross-tied to Nuclear Cooling Water

Subsequently:

- A small leak occurred in a RCP High Pressure Seal Cooler

In this condition, which of the following process radiation monitors should be able to detect the resultant activity?

1. RU-2, Train ‘A’ Essential Cooling Water
2. RU-3, Train ‘B’ Essential Cooling Water
3. RU-6, Nuclear Cooling Water

A. 1 ONLY

B. 1 AND 2 ONLY

C. 1 and 3 ONLY

D. 1, 2, and 3

Proposed Answer:	**A**

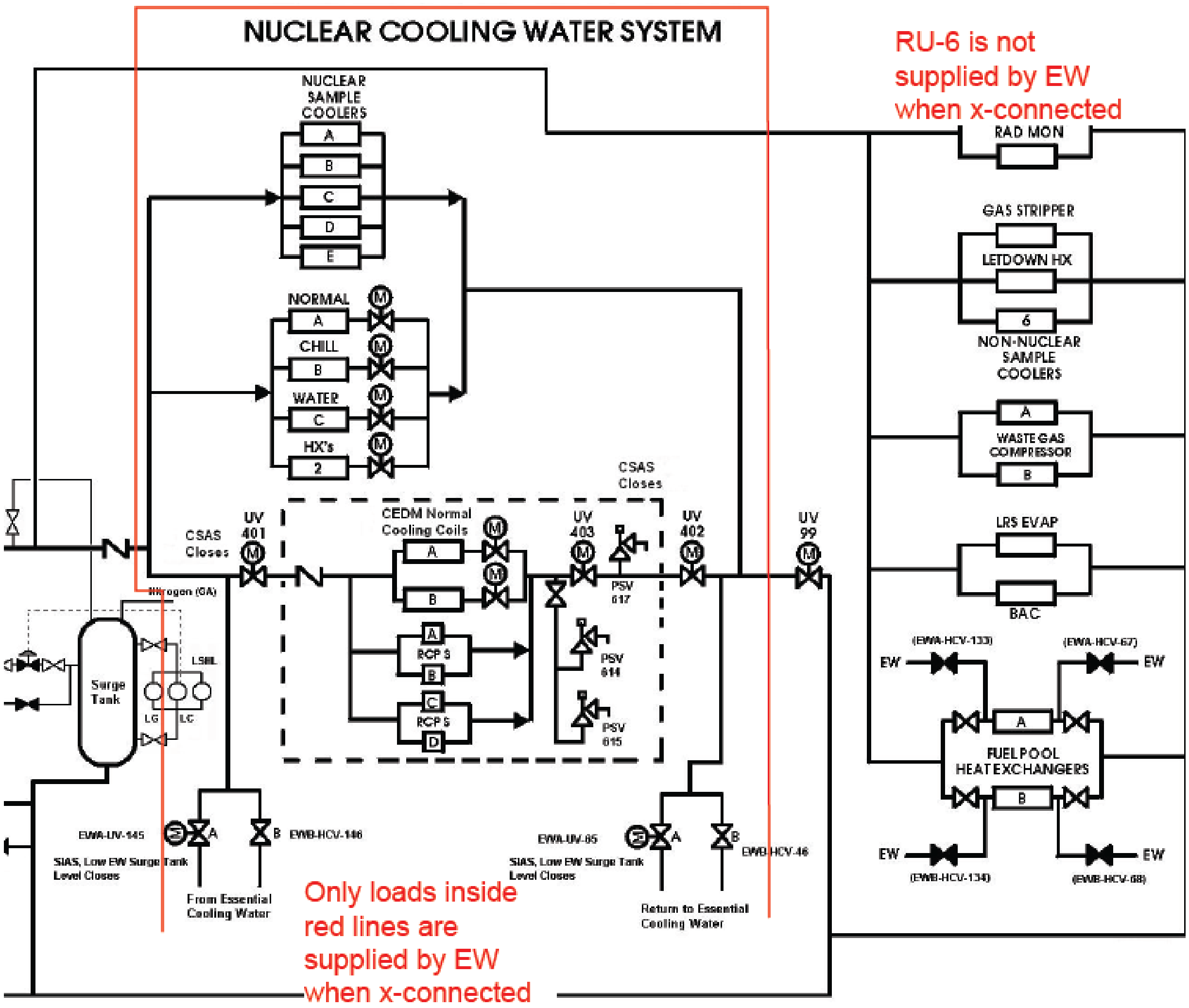
Explanations:	
A.	Correct
B.	RU-3 is plausible if thought that the trains of Essential Cooling Water shared a common header, however when one train of EW is supplying priority loads, only the associated train will detect activity due to RCS leakage.
C.	Plausible since RU-2 will be able to detect activity from an RCS leak, and plausible that RU-6 is located upstream of the NC-EW cross-tie valves, however RU-6 is isolated when the cross-tie is performed.
D.	RU-2 is correct. RU-3 is plausible if thought that the trains of Essential Cooling Water shared a common header, however when one train of EW is supplying priority loads, only the associated train will detect activity due to RCS leakage, and plausible that RU-6 is located upstream of the NC-EW cross-tie valves, however RU-6 is isolated when the cross-tie is performed.

Question Source:		**New**	
	X	**Bank**	
		Modified	
	X	**Previous NRC Exam**	**2019 Q42**

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

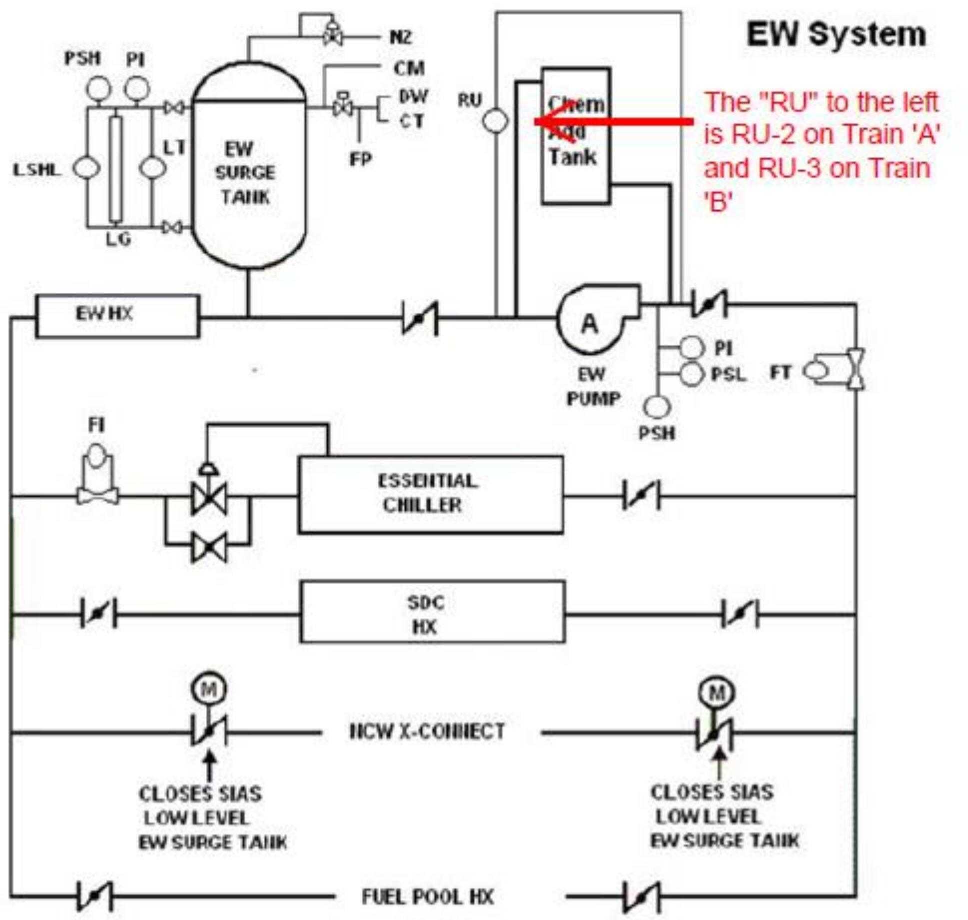
Level of Difficultly:	**3**	
10CFR55.41:	**11**	
Reference Provided:	**N**	
Learning Objective:	**66723 – Given a Radiation Monitor number and name, describe the purposes and sample points of the Radiation Monitors at PVNGS**	

Technical Reference:	LOIT Nuclear Cooling Water Lesson Plan



Technical Reference:	

This picture is for one train of EW – the 'A' Train uses RU-2, the 'B' Train uses RU-3…they do not share any common headers:



Examination Outline Cross-Reference:	**Level**	**RO**		**SRO**
K/A: Pressurizer Pressure Control: Knowledge of PZR PCS design feature(s) and/or interlock(s) which provide for the following: Over pressure control	**Tier**	**2**		
	Group	**1**		
	K/A		**010 K4.03**	
	IR	**3.8**		

Question 36

To aid in protecting the Pressurizer from an over pressure condition, the Main Spray Valves are designed to be FULLY OPEN if RCS pressure rises to a MINIMUM of ___(1)___ and Pressurizer Backup Heaters are designed to trip if RCS pressure rises to a MINIMUM of ___(2)___ .

A. (1) 2300 psia
   (2) 2285 psia

B. (1) 2300 psia
   (2) 35 psia above the setpoint of Pressure Master Controller, RCN-PIC-100

C. (1) 50 psia above the setpoint of Pressure Master Controller, RCN-PIC-100
   (2) 2285 psia

D. (1) 50 psia above the setpoint of Pressure Master Controller, RCN-PIC-100
   (2) 35 psia above the setpoint of Pressure Master Controller, RCN-PIC-100

Proposed Answer:	C

Explanations:	
A.	First part is plausible since the normal setpoint for PIC-100 is 2250 psia, which would result in the Main Spray Valves being full open as soon as RCS pressure reached 2300 psia, however the actual design of the system is for Main Spray Valves to be full open as soon as RCS pressure is 50 psia above the setpoint on PIC-100. Second part is correct.
B.	First part is plausible since the normal setpoint for PIC-100 is 2250 psia, which would result in the Main Spray Valves being full open as soon as RCS pressure reached 2300 psia, however the actual design of the system is for Main Spray Valves to be full open as soon as RCS pressure is 50 psia above the setpoint on PIC-100. Second part is plausible since 2285 psia is 35 psia above the normal setpoint of PIC-100, however all backup heaters trip at 2285 psia, regardless of PIC-100 setpoint.
C.	Correct.
D.	First part is correct. Second part is plausible since 2285 psia is 35 psia above the normal setpoint of PIC-100, however all backup heaters trip at 2285 psia, regardless of PIC-100 setpoint.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

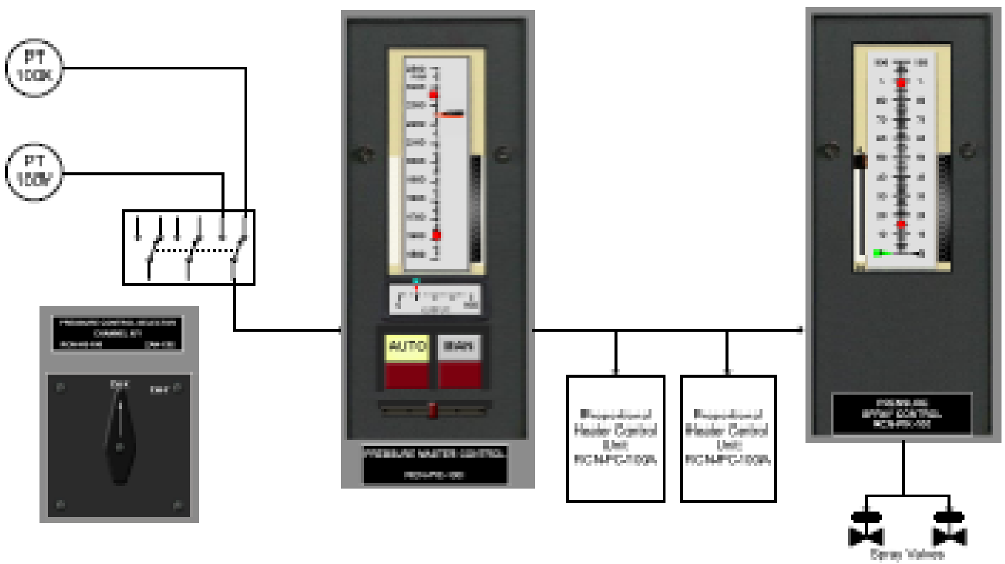
Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	22581 – Describe the automatic features associated with the Pressurizer Pressure Control System Bistables	

Pressurizer Pressure Control System (PPCS)

The pressurizer pressure controller normally functions to maintain system pressure at the desired setpoint by controlling the operation of the proportional heater banks and spray valves. When operating in the automatic mode, the pressure controller compares actual pressurizer pressure to the setpoint input from the operator. Controller output is adjusted in proportion to the deviation from setpoint value. This signal is sent to the spray valve electro-pneumatic controllers and the proportional heater control units. At zero error (measured pressure = setpoint pressure), controller output is at 16.5% (figure A-7). For this condition, the proportional heater banks would be 50% energized and the spray valves closed.

RCN-PIC-100 receives power from NNN-D12. A loss of power to this controller wil result in the controller failing to zero output (0 ma).



Automatic Control Overview

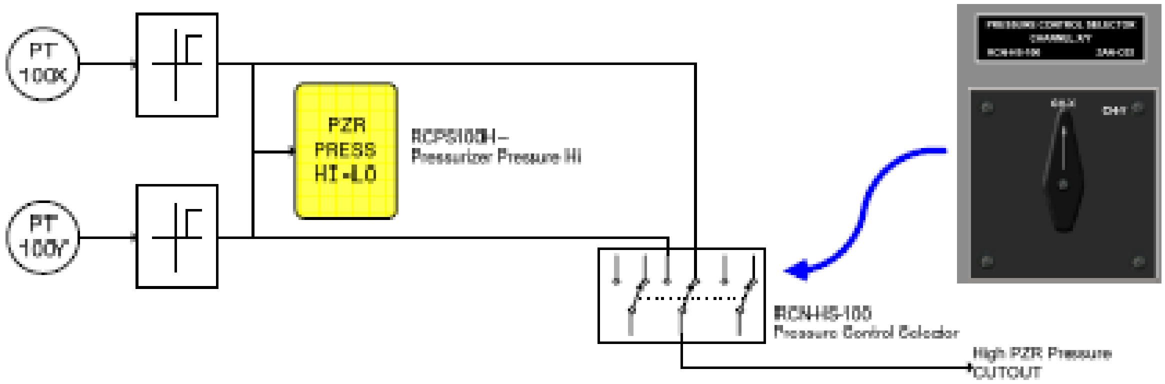
As measured pressure decreases below setpoint, proportional heater power is increased. The controller is at minimum output and the proportional heaters at maximum power when actual pressure is 25 psi below setpoint.

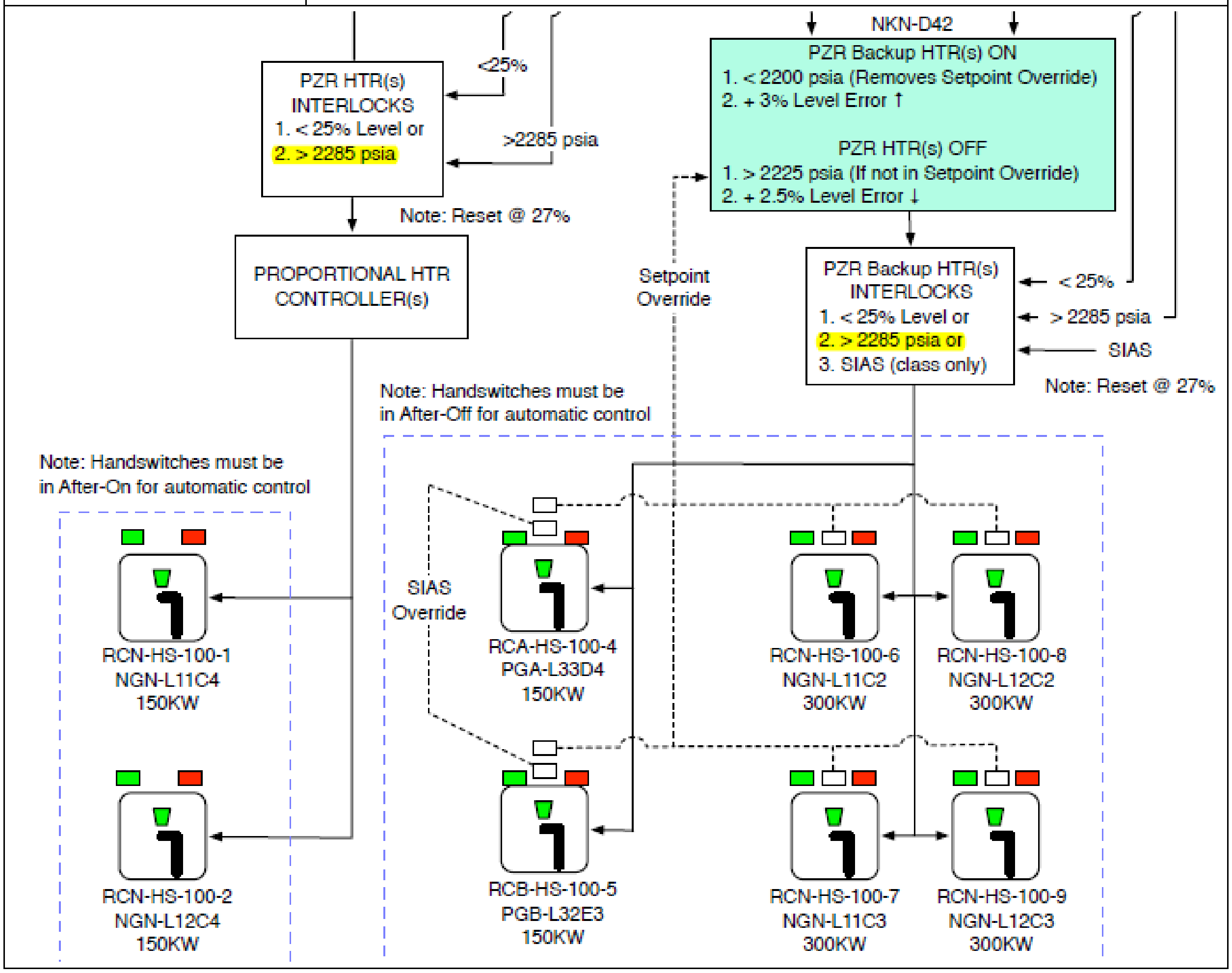
If measured pressure increases above setpoint, proportional heater power decreases. The heaters will be full off at 25 psi above setpoint (33.5% controller output). The main spray valves also begin to open at this point and open fully at +50 psi deviation (50% controller output). Deviation greater than +50 psi results in controller output increasing from 50% to 100% range with no further actions, maintaining the spray valves fully open. The full pressure control program is shown in figure A-8.

Technical Reference:	Pressurizer Pressure Control System Tech Manual

2.7.2 High Pressure Alarm/ Control Bistables (RCN-PSH-100X, RCN-PSH-100Y)

Two high pressure alarm/control bistables, which energize at 2285 psia increasing, provide dual functions. Either bistable, when energized, initiates a hi-lo pressure alarm in the main control room. Concurrently, the bistable selected for control initiates a high pressure interlock, which trips all pressurizer heaters. The bistables de-energize (reset) at 2275 psia decreasing (figure A-2).




NKN-D42
PZR Backup HTR(s) ON
1. < 2200 psia (Removes Setpoint Override)
2. + 3% Level Error ↑
PZR HTR(s) OFF
1. > 2225 psia (If not in Setpoint Override)
2. + 2.5% Level Error ↓
PZR HTR(s) INTERLOCKS
1. < 25% Level or
2. > 2285 psia
<25%
>2285 psia
Note: Reset @ 27%
PROPORTIONAL HTR CONTROLLER(s)
Setpoint Override
PZR Backup HTR(s) INTERLOCKS
1. < 25% Level or
2. > 2285 psia or
3. SIAS (class only)
< 25%
> 2285 psia
SIAS
Note: Reset @ 27%
Note: Handswitches must be in After-Off for automatic control
Note: Handswitches must be in After-On for automatic control
SIAS Override
RCN-HS-100-1
NGN-L11C4
150KW
RCA-HS-100-4
PGA-L33D4
150KW
RCN-HS-100-6
NGN-L11C2
300KW
RCN-HS-100-8
NGN-L12C2
300KW
RCN-HS-100-2
NGN-L12C4
150KW
RCB-HS-100-5
PGB-L32E3
150KW
RCN-HS-100-7
NGN-L11C3
300KW
RCN-HS-100-9
NGN-L12C3
300KW

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Pressurizer Pressure Control: Knowledge of the operational implications of the following concepts as the apply to the PZR PCS: Determination of condition of fluid in PZR, using steam tables	**Tier**	**2**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**010 K5.01**		
	IR	**3.5**	[redacted]	

Question 37

Given the following conditions:

- Unit 2 is recovering from an ESD on SG #2
- Inaction from the crew has caused the Pressurizer to go solid

Subsequently:

- The crew is drawing a bubble in the Pressurizer per 40EP-9EO05, Excess Steam Demand
- 1A and 2A RCPs are running
- Pressurizer pressure is 1800 psia
- Pressurizer temperature is 610°F

(1) The Pressurizer currently…

(2) Per 40DP-9AP10, Excess Steam Demand Technical Guideline, if RCP flow is not maintained, the most UNDESIRABLE place for bubble formation is in the…

A. (1) has a bubble
(2) Steam Generator U-Tubes

B. (1) has a bubble
(2) Reactor Vessel Upper Head

C. (1) is in a water solid condition
(2) Steam Generator U-Tubes

D. (1) is in a water solid condition
(2) Reactor Vessel Upper Head

Proposed Answer:	**C**

Explanations:	
A.	First part is plausible if a candidate thinks that a saturated system exists when pressure is greater than the pressure listed in the steam tables. Second part is correct.
B.	First part is plausible if a candidate thinks that a saturated system exists when pressure is greater than the pressure listed in the steam tables. Second part is plausible because it is not desirable to have voiding in the Reactor Vessel Head. However, it is not a problem if there is RCP flow or natural circulation.
C.	Correct
D.	First part is correct. Second part is plausible because it is not desirable to have voiding in the Reactor Vessel Head. However, it is not a problem if there is RCP flow or natural circulation.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**5**	
Reference Provided:	**N**	
Learning Objective:	**25498 – Given the EOPs are being performed, describe how the operator will diagnose water solid conditions per 40EP-9EO05**	

Technical Reference:	Steam Tables
1800.0	621.02
608.0	1637.3
612.0	1686.1

Technical Reference:	40EP-9EO05, Excess Steam Demand

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INSTRUCTIONS

CONTINGENCY ACTIONS

✱ 32. **IF** drawing a bubble in the Pressurizer is desired,
AND ANY of the following conditions exist:

- Both Steam Generators can be maintained less than RCS pressure
- At least one RCP is running

4.5.32 Step 32 - Establish a Bubble in the Pressurizer

A. This step provides directions for drawing a bubble in the RCS. Meeting either of the stated conditions will minimize the possibility of forming a void in the steam generator U-tubes. The Pressurizer is the preferred location for the bubble, but a bubble in the reactor vessel head is not a problem as long as it does not interfere with natural circulation flow, i.e., remains above the hot leg nozzles.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Reactor Protection: Knowledge of bus power supplies to the following: RPS channels, components, and interconnections	Tier	2	[redacted]	
	Group	1	[redacted]	
	K/A	012 K2.01		
	IR	3.3	[redacted]	

Question 38

Continuous power DIRECTLY to RPS Matrix Logic is supplied from ___(1)___ via ___(2)___.

A. (1) 120 VAC Class buses
   (2) auctioneering diodes

B. (1) 120 VAC Class buses
   (2) static transfer switches

C. (1) 125 VDC Class buses
   (2) auctioneering diodes

D. (1) 125 VDC Class buses
   (2) static transfer switches

Proposed Answer:	**A**

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because the power source is PN. PN busses use a static transfer switch to maintain power.
C.	First part is plausible because 120 VDC is used for control power for Reactor Trip Circuit Breakers which is also part of the Plant Protection System. Second part is correct.
D.	First part is plausible because 120 VDC is used for control power for Reactor Trip Circuit Breakers which is also part of the Plant Protection System. Second part is plausible because the power source is PN. PN busses use a static transfer switch to maintain power.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

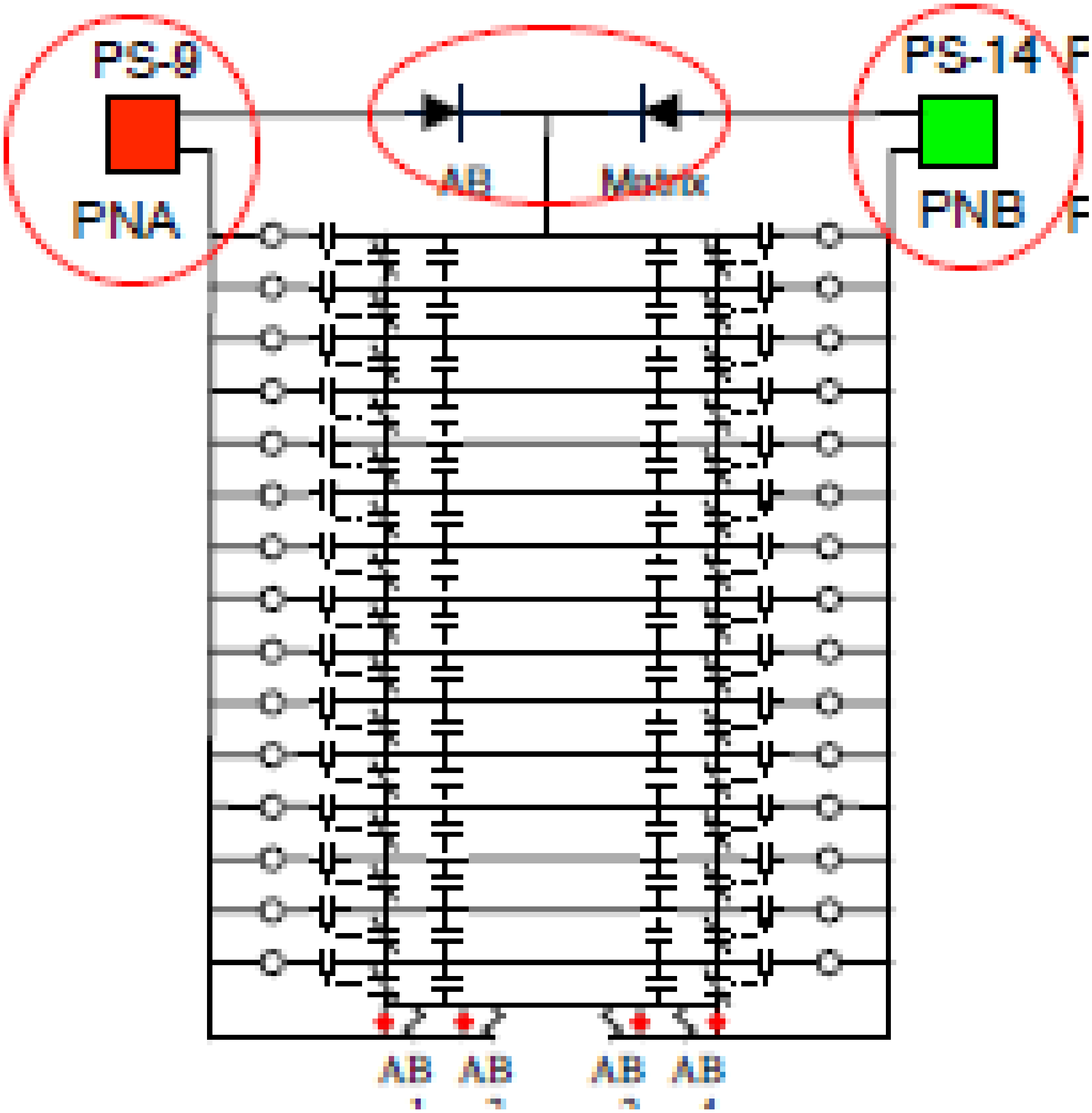
Level of Difficultly:	**2**	
10CFR55.41:	**8**	
Reference Provided:	**N**	
Learning Objective:	**18804 – Describe how matrix logic receives electrical power**	

Plant Protection System (PPS-SA/SB)

APPENDIX C: Electrical Power Supplies

Table C - 1
PPS System Power Supplies

PPS POWER SUPPLY #'S	A - PNA-D25 B - PNB-D26 C - PNC-D27 D - PND-D28 A	B	C	D	DESCRIPTION
1 3	X	 X			Auctioneered power supplies for Channel "A" Bistable Comparator Cards and Relays
2 4	X	 X			Auctioneered power supplies for Channel "B" Bistable Comparator Cards and Relays
5 7			X	 X	Auctioneered power supplies for Channel "C" Bistable Comparator Cards and Relays
6 8			X	 X	Auctioneered power supplies for Channel "D" Bistable Comparator Cards and Relays
9 14	X	 X			Auctioneered power supplies for the Reactor Protection System Logic Matrix "AB"
10 12	X	 X			Auctioneered power supplies for the Reactor Protection System Logic Matrix "AC"
11 20	X			 X	Auctioneered power supplies for the Reactor Protection System Logic Matrix "AD"
13 15		X	 X		Auctioneered power supplies for the Reactor Protection System Logic Matrix "BC"
16 19			X	 X	Auctioneered power supplies for the Reactor Protection System Logic Matrix "CD"
17 18			X	 X	Auctioneered power supplies for the Reactor Protection System Logic Matrix "BD"



PS-9
PNA
AB
Matrix
PS-14
PNB
AB
AB
AB
AB

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Reactor Protection: Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following: Logic matrix testing	Tier	2	[redacted]	
	Group	1	[redacted]	
	K/A	012 K4.08		
	IR	2.8	[redacted]	

Question 39

Given the following conditions:

- Unit 3 is performing logic matrix testing of the RPS system
- All Channel ‘C’ RPS parameters have been placed in BYPASS
- Testing on Channel ‘C’ is complete

If a Channel ‘B’ parameter is taken to BYPASS BEFORE the corresponding Channel ‘C’ is removed from BYPASS, the Channel ‘C’ parameter should ___(1)___ and the Channel ‘B’ parameter should ___(2)___ .

A. (1) remain in BYPASS
(2) be in BYPASS

B. (1) remain in BYPASS
(2) NOT go to BYPASS

C. (1) come out of BYPASS
(2) be in BYPASS

D. (1) come out of BYPASS
(2) NOT go to BYPASS

Proposed Answer:	**C**

Explanations:	
A.	First part is plausible because if Channel ‘A’ was in bypass it would remain in bypass because it is a higher priority channel. However, since Channel ‘C’ is a lower priority channel, it will come out of bypass. Second part is correct.
B.	First part is plausible because if Channel ‘A’ was in bypass it would remain in bypass because it is a higher priority channel. However, since Channel ‘C’ is a lower priority channel, it will come out of bypass. Second part is plausible because if Channel ‘A’ was in bypass, Channel ‘B’ would not go to bypass since it is higher priority channel. However, since Channel ‘C’ is a lower priority channel, Channel ‘B’ will go into bypass.
C.	Correct
D.	First part is correct. Second part is plausible because if Channel ‘A’ was in bypass, Channel ‘B’ would not go to bypass since it is higher priority channel. However, since Channel ‘C’ is a lower priority channel, Channel ‘B’ will go into bypass.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**18776 – Describe the Matrix Testing Interlock associated with the RPS**	

Technical Reference:	Plant Protection System Tech Manual

2.1.9 RPS Interlocks

Trip Channel Bypass

An electrical interlock prevents the operator from bypassing more than one trip channel at a time for any one type of trip. Different type trips may be bypassed simultaneously, either in one channel or in different channels. Attempting to insert a trip channel bypass in a second channel for the same type of trip will result in only the highest priority channel being in bypass, with A being the highest, and D the lowest priority.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Engineered Safety Features Actuation: Ability to predict and/or monitor changes in parameters (to Prevent exceeding design limits) associated with operating the ESFAS controls including: Containment pressure, temperature, and humidity	**Tier**	**2**		
	Group	**1**		
	K/A	**013 A1.02**		
	IR	**3.9**		

Question 40

Given the following conditions

- A LOCA inside containment is in progress
- Containment pressure is 5 psig and rising
- Containment temperature is 130°F and rising

Given the current conditions with NO OPERATOR ACTION, a CSAS ___(1)___ occurred and if parameters continue to rise will reach a harsh condition AS SOON AS Containment temperature reaches ___(2)___°F.

A. (1) HAS
(2) 170

B. (1) HAS
(2) 235

C. (1) has NOT
(2) 170

D. (1) has NOT
(2) 235

Proposed Answer:	C

Explanations:	
A.	First part is plausible because at 3 psig SIAS, CIAS, and MSIS all automatically actuate. Second part is correct.
B.	First part is plausible because at 3 psig SIAS, CIAS, and MSIS all automatically actuate. Second part is plausible because 235°F is the expected Containment temperature during a LOCA.
C.	Correct
D.	First part is correct. Second part is plausible because 235°F is the expected Containment temperature during a LOCA.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**22587 – Describe what automatically initiates the Containment Spray Actuation System (CSAS) and its function**	

Technical Reference:	40AL-9RK5B, Panel B05B Alarm Responses

Response Section

5B05C

HI-HI CNTMT PRESS CH TRIP

High High Containment Pressure Channel Trip

Point ID	Description	Setpoint
SATA17	Hi-Hi Containment Pressure Channel A Trip	8.06 psig
SATB17	Hi-Hi Containment Pressure Channel B Trip	8.06 psig
SATC17	Hi-Hi Containment Pressure Channel C Trip	8.06 psig
SATD17	Hi-Hi Containment Pressure Channel D Trip	8.06 psig

6.19.2 170F – Harsh Containment Temperature Limit

VALUE:

- > 170°F containment average air temperature

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Containment Cooling: Knowledge of the operational implications of EOP warnings, cautions, and notes	**Tier**	**2**		
	Group	**1**		
	K/A	**022 G 2.4.20**		
	IR	**3.8**		

Question 41

Given the following conditions:

- Unit 1 was tripped due to a loss of all Feedwater
- The CRS entered 40EP-9EO01, Standard Post Trip Actions
- Containment temperature is 120°F and slowly rising
- The BOP is performing step 9 of SPTAs and reports that NO Containment ACUs and NO Normal Chillers are running

Per EOP Operations Expectations, during SPTAs the BOP should start ___(1)___ of Containment ACUs and ___(2)___ Large Normal Chiller(s).

A. (1) one train
(2) one

B. (1) one train
(2) two

C. (1) both trains
(2) one

D. (1) both trains
(2) two

Proposed Answer:	A

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible if it is assumed that because Containment temperature is not meeting the required temperature in SPTAs and it is rising, more than one chiller should be started to aid in restoring temperature.
C.	First part is plausible because Containment temperature is not meeting the required temperature in SPTAs and it is rising, therefore more than one Containment ACU should be started to aid in restoring temperature. Second part is correct.
D.	First part is plausible because Containment temperature is not meeting the required temperature in SPTAs and it is rising, therefore more than one Containment ACU should be started to aid in restoring temperature. Second part is plausible to start two large Normal Chillers to support two trains of Containment ACUs

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	22504 – Given plant conditions following a Reactor trip, analyze whether the Containment Temperature, Pressure and Combustible Gas Control Safety Function is met and what contingency actions are required if it is not in accordance with 40EP-9EO01	

SPTA Step 8

1. Check CIAS initiated by ensuring the ESFAS signal has been initiated.
2. Use the available radiation monitors for the activity checks. The RMS checks should be limited to what is available in the Control Room. If Control Room indications are inadequate, then the CRS should make the diagnosis using available monitors and any associated pre-trip indications and rely on the SFSC or go to the FRP.
3. The RO should inform the CRS of which monitors were used to check the safety function.
4. "Unexplained" increase in activity is in reference to pre-trip or pre-event conditions.

SPTA Step 9

1. Opening CTMT Instrument Air isolation should be performed as needed to support RCP operation or to close RCP bleedoff valves.
2. If performing the contingency action for high containment temperature, only one train of Containment Normal ACUs, CEDM ACUs, Reactor Cavity fans, and Pressurizer Cooling fans should be started.
3. If Nuclear Cooling Water is in service, one Large Normal Chiller may be started if no Normal Chillers are running. Progress through the SPTAs should not be delayed to wait for the chiller to start. Other actions to restore containment cooling should wait until an ORP or the FRP is entered.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Containment Spray: Knowledge of bus power supplies to the following: MOVs	**Tier**	**2**		
	Group	**1**		
	K/A	**026 K2.02**		
	IR	**2.7**		

Question 42

The feeder breaker to Containment Spray Header Discharge Valve, SIA-UV-672, is located on which of the following panels?

A. PNA-D26

B. PKA-M41

C. PHA-M35

D. PGA-L35

Proposed Answer:	**C**

Explanations:	
A.	Plausible because the Containment Spray discharge valve is very important to safety during a LOCA or ESD, therefore should be powered from an inverter that has a backup ac power supply from a voltage transformer
B.	Plausible because the Containment Spray discharge valve is very important to safety during a LOCA or ESD, therefore should have a power supply that has a battery charger and a battery backup.
C.	Correct
D.	Plausible because the Containment Spray discharge valve is very important to safety during a LOCA or ESD and can be de-energized by a loss of PBA-S03 or PGA-L35. However, the feeder breaker for the valve is located on PHA-M35.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**8**	
Reference Provided:	**N**	
Learning Objective:	**23844 – Identify the power supplies to SI related equipment**	

Technical Reference:	40AO-9ZZ12, Degraded Electrical Power

PHA-M35 Loads

(Bold T.S. numbers indicate less than 2 hour action requirements)

Equipment	Alt Equip	Power	TS / TRM Reference
PKA-H11, Battery Charger A	PKA-H15	PHA-M3326	• **LCO 3.8.4 (Mode 1 - 4)** • **LCO 3.8.5 (Mode 5 & 6 During irradiated fuel movement)**
PNA-V25, Backup Voltage Reg For PNA-D25	PNA-N11	PKA-M4106	
HPA-E01, Hydrogen Recombiner	HPB-E01	PHB-M3426	• TLCO 3.6.300 (Mode 1 - 2)
SIA-UV-655, Shutdown Clg Ctmt Iso Loop 1 Valve SIA-UV-651, Shut Down Clg Iso Loop 1 Valve	None		• **LCO 3.4.6 (Mode 4)** • **LCO 3.4.7 (Mode 5 loops filled)** • **LCO 3.4.8 (Mode 5 loops not filled)** • LCO 3.6.3 (Mode 1 - 4) • **LCO 3.9.4 (Mode 6 ≥ 23 ft. above Rx vessel flange)** • **LCO 3.9.4 (Mode 6 < 23 ft. above Rx vessel flange)**
CTA-HV-4, Cond Tk to Aux FW Iso Vlv	None		• LCO 3.7.5 (Mode 1 - 3, Mode 4 when SG needed for heat removal) • TLCO 7.0.400
SIA-UV-664, Ctmt Spray Pump A To RWT Iso Vlv SIA-UV-672, Ctmt Spray Control Train A Vlv	None		• LCO 3.6.3 (Mode 1 - 4) • LCO 3.6.6 (Mode 1 - 3, Mode 4 RCS pressure ≥ 385 psia) • TLCO 7.0.400

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Containment Spray: Ability to manually operate and/or monitor in the control room: CSS controls	**Tier**	**2**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**026 A4.01**		
	IR	**4.5**	[redacted]	

Question 43

Given the following conditions:

- An inadvertent Train ‘B’ CSAS has occurred
- The CRS entered 40AO-9ZZ17, Inadvertent PPS-ESFAS Actuations
- The ‘B’ Containment Spray pump was stopped
- The ‘B’ Containment Spray header isolation valves were closed

Prior to the CSAS being reset:

- ESF Service Transformer NBN-X04 tripped
- ‘B’ EDG energized PBB-S04

‘B’ Containment Spray pump…

A. starts because the CSAS signal resets.

B. starts because the Train ‘B’ Load Sequencer goes through Mode 0.

C. does not start because the pump is overridden to the STOP position

D. does not start because the breaker remains in the anti-pump condition until control power is cycled

Proposed Answer:	B

Explanations:	
A.	Plausible if it is thought that the CSAS signal will be reset if there is a loss of power to class bus that powers all of the Containment Spray equipment
B.	Correct
C.	Plausible if this pump was overridden, however it is taken to ‘Stop’ and is anti-pumped
D.	Plausible if it is thought that because it is anti-pumped it cannot automatically be restarted, however the load sequencer will restart the pump

Question Source:		**New**	
	X	**Bank**	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	**23824 – Explain how the Load Sequencer changes between the different modes of operation**	

Technical Reference:	BOP ESFAS System Tech Manual

- SIAS/CSAS coincident with a loss of power. Sequencing is started on a diesel generator breaker closure signal (mode 2).
- Loss of power without an SIAS/CSAS. Sequencing is started on a diesel generator breaker closure signal (mode 3).
- Other signals without an SIAS/CSAS and without a loss of power (Mode 4). These signals are:
  CRVIAS or CREFAS
  FBEVAS
  AFAS-1 or AFAS-2
  Diesel generator running

Receipt of subsequent input signals, which require a change of operating mode cause, the load sequencer to reset, transfer to the required mode, and initiate sequencing of the required loads.

Devices which are sequentially actuated through the load sequencer receive a load shed signal on bus undervoltage to trip the device load, and a load sequencer start signal to start the device at the appropriate time. Reset of the load sequencer and its actuation relays does not stop or shed actuated devices. Devices are shed only on the load shed signal.

The following example will better illustrate the meaning of the previous statement.

Time	Condition	BOP ESFAS Response
Zero	SIAS	Sequencer receives signal from PPS ESFAS and shifts to Mode 1.
1 minute	SIAS/LOP	The sequencer receives the LOP Signal and resets to Mode 0 and then enters Mode 2. This action results in a Load Shed signal followed by a LOP signal that seals in for 60 seconds. Shortly after the Load Shed, the D/G will close in on the 4kV bus. Once this happens, the Load Sequencer starts sequencing on loads in Mode 2.
2 minutes	SIAS	At this time, the LOP clears and the sequencer resets through zero and back to Mode 1. However, no loads are cycled since the Load Shed relay did not actuate.

Example...

An inadvertent Train "A" CSAS occurs, resulting in the starting of the Train "A" CS pump. The operators "anti-pump" the CS pump, stopping spray flow.

Anti-pump refers to the circuit breakers anti-pump relay (52Y) located in the circuit breaker internal operating mechanism, being energized. When this relay is energized, it opens contacts in line with the closing coil, preventing the breaker from additional closing attempts. This relay initially energizes when the closing spring discharges, and is then maintained in an energized state for as long as the closing signal exists. When a breaker is closed with the handswitch, this closing signal would go away when the handswitch is taken out of the CLOSE position.

In the case of a start signal from the sequencer, the contact will stay closed for as long as the sequencer is in that mode.

A LOP then occurs on PBA-S03. BOP ESFAS now sees the LOP, and needs to shift out of Mode 1. The sequencer shifts to Mode 0, and then enters Mode 2 as soon as the DG Breaker closes (SIAS/CSAS with LOP and DG Breaker closed). This action clears (resets) the sequencer start signal that existed for the CS "A" pump. Because of this, the "A" CS pump will start after the DG closes in.

Examination Outline Cross-Reference:	**Level**	**RO**		**SRO**
K/A: Main and Reheat Steam: Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity	**Tier**	**2**		
	Group	**1**		
	K/A	**039 K5.03**		
	IR	**3.6**		

Question 44

Given the following conditions:

- Unit 2 is operating at 100% power

Subsequently:

- The 6A Feedwater Heater Normal Control Valve has failed closed
- The 6A Feedwater Heater High Level Control Valve is seized closed

With NO operator action, Reactor power should INITIALLY ___(1)___ due to ___(2)___.

A. (1) rise
(2) a decrease in feedwater heating

B. (1) rise
(2) an increase in steam being sent to the Main Turbine

C. (1) lower
(2) a decrease in feedwater heating

D. (1) lower
(2) an increase in steam being sent to the Main Turbine

Proposed Answer:	A

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible since the failures in the stem would result in the extraction steam valve to the 6A heater closing, thus diverting steam to the low pressure turbine, however this will not impact reactor power since the steam leaving the SGs will be unaffected.
C.	Plausible since hot water in the 6A heater can no longer be rejected to the condenser (due to the normal level control valve failing closed) which would potentially increase the amount of hot water available to be sent to the SG, however the 6A heater will have steam isolated to it resulting in a lower temperature and a net decrease in feedwater heating.
D.	Plausible since the isolation of extraction steam to the 6A heater will result in a lower feedwater temperature, which could cause more extraction steam to be aligned to other heaters to compensate for the reduction in feedwater heating (and thus taking steam which could have gone to the main turbine), however when the extraction steam to the 6A heater is stopped, the steam is diverted to the low pressure turbine.

Question Source:		**New**	
	X	**Bank**	
		Modified	
	X	**Previous NRC Exam**	**2018**

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

Level of Difficultly:	**3**	
10CFR55.41:	**14**	
Reference Provided:	**N**	
Learning Objective:	**18420 – Explain the operation of the High Pressure Feedwater Heaters under normal operating conditions**	

Technical Reference:	ICES Report #415578 (Palo Verde Unit 2 – May 2017)

Event Summary:
On 05/30/2017 at 1336, the Unit 2 Control Room received a HI-HI level alarm for high pressure feedwater heater 7B. The high level condition in the feedwater heater resulted in extraction steam and drains to the feedwater heater being automatically isolated. As a result, the temperature of the feedwater supplied to the steam generators decreased from 451 degrees to 430 degrees Fahrenheit. The lower feedwater temperature resulted in a reactor power increase to 101.13%. The Control Room operators took prompt action to reduce turbine load and at 1345 stabilized reactor power at 97.5%. Additionally, Group 5 Control Element Assemblies (CEAs) automatically inserted approximately six steps in response to this transient. Repairs were completed at approximately 2030 and Operations commenced increasing reactor power at approximately 2300. On 5/31/2017 at 12:09 AM reactor power was restored to 100%.

Upon investigation, the instrument air supply line to feedwater heater 7B normal level control valve was found disconnected from the valve actuator, causing the valve to fail closed. Also, the valve controller for the high level control valve failed to operate properly, which caused the valve to stay closed even though actual feedwater heater level was high. The normal and high level control valves maintain thermal efficiency and heater life and their proper operation are important for maintaining feed water temperature and reactivity control.

Technical Reference:	

Response Section

6B13C

HP HTRS TR A LVL HI-HI

High Pressure Heaters Train A Level High High

Point ID	Description	Setpoint
EDLS611	Heater 6A Level Hi-Hi	8 inches above zero level

MANUAL ACTIONS

___ 1. Check EDN-HS-29, HTR 6A LINE DRAIN VLV, to verify EDN-PV-29 open.

___ 2. Check EDN-UV-27, Extraction Steam into 6A HPFW Heater Block Valve, closed at EDN-HS-13A, HTR 6A VLVS.

___ 3. Direct Nuclear Operator to check EDN-BTV-13, 6A HPFW Htr Extract Steam Header Bleeder Trip Valve, closed.

___ 4. Check EDN-HS-615, 1ST STG SCAVENGING MODE SELECTOR, to verify BOTH of the following:

- ____ EDN-FV-611A, A & C 1st Stage RDT Scavenging Steam to 6A Htr Flow Control Valve, is closed.
- ____ EDN-FV-611B, A & C 1st Stage Reheater Drain Tank Vent Control Valve, is open.

___ 5. Observe reactor power for an increase due to a reduction in feedwater temperature.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Main Feedwater: Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of feedwater regulating valves	**Tier**	**2**		
	Group	**1**		
	K/A	**059 A2.12**		
	IR	**3.1**		

Question 45

Given the following conditions:

- Unit 3 Reactor was tripped for a Refueling Outage
- T_{COLD} is 571°F and rising
- RRS T_{AVE} has failed to 550°F

Based on the RRS failure, the DFWCS should automatically ___(1)___ and to mitigate the condition the BOP should ___(2)___.

A. (1) stop feeding
   (2) adjust MFP speed

B. (1) stop feeding
   (2) take MANUAL control of downcomer valves

C. (1) be feeding at the maximum rate
   (2) adjust MFP speed

D. (1) be feeding at the maximum rate
   (2) take MANUAL control of downcomer valves

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible because raising or lowering MFP speed would normally increase/decrease Feedflow, however with this malfunction the downcomer valves will be closed so changing MFP speed will not do anything.
B.	Correct
C.	First part is plausible if RTO is fed from T_{COLD} and not $T_{AVE.}$ Second part is plausible because raising or lowering MFP speed would normally increase/decrease Feedflow, however with this malfunction the downcomer valves will be closed so changing MFP speed will not do anything.
D.	First part is plausible if RTO is fed from T_{COLD} and not $T_{AVE.}$ Second part is correct

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**4**	
Reference Provided:	**N**	
Learning Objective:	**31226 – Describe the response of the Reactor Regulating System to a failure of a Temperature Transmitter input**	

Post Trip (RTO) Control Mode

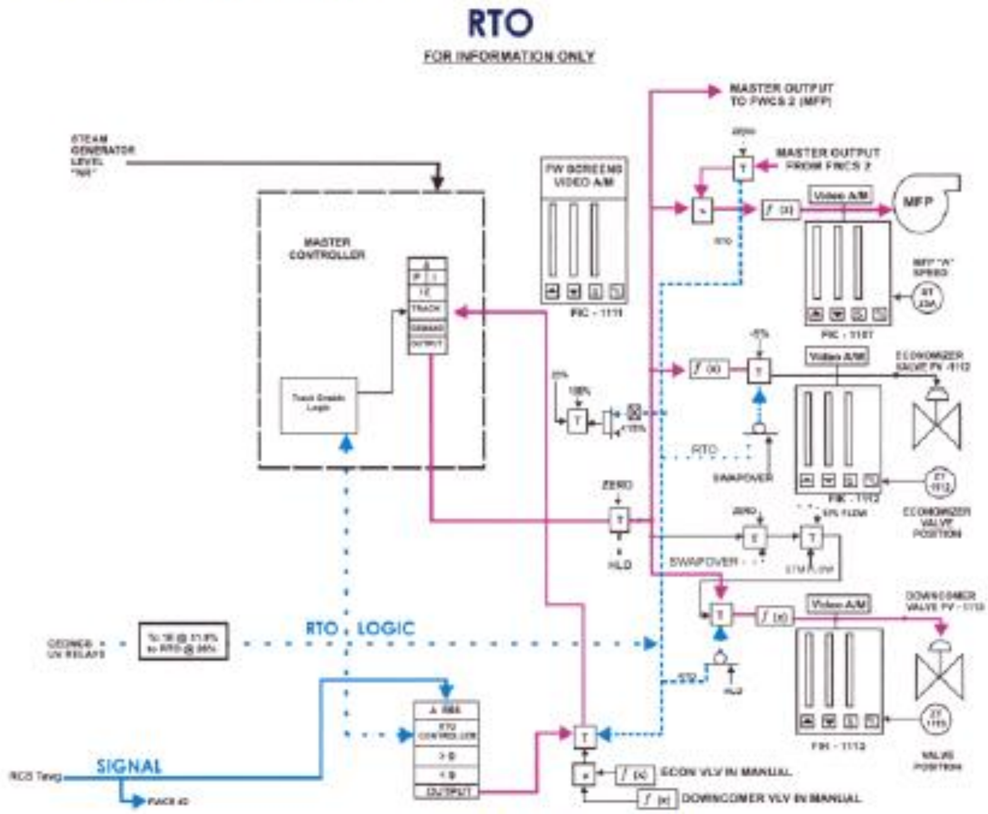


Figure 1 - 5, RTO Control

Why - A reactor trip places several challenges on the DFWCS. A large SG level shrink will normally occur following a Reactor Trip. For the sake of Nuclear Safety on will want to restore the SG levels to their normal band as quickly as possible. However, one does not want to overcool the RCS while attempting to refill the SGs. Once the reactor trip undervoltage relays decrease to 180 volts (typically a Reactor Trip) the RTO controller will refill the SGs as fast as possible while maintaining Tave at 564°F.

How – If the reactor trip undervoltage relays decrease to 180 volts (typically a Reactor Trip) the RTO controller will use one process input, Tave, to refill the Steam Generators. The RTO controller modulates the downcomer valve position based on the Tave variance from a 564°F setpoint. The maximum output of the RTO controller is 9% of the master controller output. This equates to a downcomer valve position of 40%. So following a reactor trip, while FW control is in RTO, an operator should not see the downcomer valve more than 40% open.

Response Section

6A06A

FWCS PROCESS TRBL

Feedwater Control System Process Trouble

Point ID	Description	Setpoint
(x)FWCS1:TAVG	Reactor Coolant Avg Temp (x)C0324-2 Bad	NA
(x)FWCS1:SBCS_MCD	SBCS MSTR CNTRL Output (x)C0334-2 Bad	NA
(x)FWCS1:TLI	Turbine Load Index (x)C0335-2 Bad	NA

AUTOMATIC ACTION

- None

MANUAL ACTIONS

NOTE

___ • Transmitters for this alarm group do NOT have redundant transmitters.

• On a Tavg signal bad to the DFWCS, Reactor Trip Override will NOT function as designed.

___ 1. IF BOTH of the following:

____ • (x)FWCS1:TAVG, Reactor Coolant Avg Temp (x)C0324-2 Bad, is in alarm.

____ • Reactor is tripped.

THEN place the affected DFWCS in Manual.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Auxiliary/Emergency Feedwater: Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Controllers and positioners	**Tier**	**2**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**061 K6.01**		
	IR	**2.5**	[redacted]	

Question 46

Given the following conditions:

- Unit 1 is in MODE 2 at 1% power during a startup
- AFN-P01 is feeding both Steam Generators via the Feedwater Isolation bypass valves SGN-HV-1143 and SGN-HV-1145

Subsequently:

- An inadvertent SIAS occurs

With NO operator action, SGN-HV-1143 and SGN-HV-1145 should fail ___(1)___ and AFN-P01 ___(2)___ running.

A. (1) closed
(2) is

B. (1) closed
(2) is not

C. (1) as-is
(2) is

D. (1) as-is
(2) is not

Proposed Answer:	D

Explanations:	
A.	First part is plausible because on a SIAS, NC cross-tie valves will fail closed. If valves fail as-is, the Feedrate will no longer be controlled by an operator in the Control Room. It is reasonable that at low power levels when this valve is used, the valve will fail closed until manual operation can be restored..Second part is plausible because if AFB-P01 was running, it would be stripped and then restarted during a SIAS. AFN-P01 is stripped but not restarted.
B.	First part is plausible because on a SIAS, NC cross-tie valves will fail closed. If valves fail as-is, the Feedrate will no longer be controlled by an operator in the Control Room. It is reasonable that at low power levels when this valve is used, the valve will fail closed until manual operation can be restored. Second part is correct.
C.	First part is correct. Second part is plausible because if AFB-P01 was running, it would be stripped and then restarted during a SIAS. AFN-P01 is stripped but not restarted.
D.	Correct

Question Source:	X	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**4**	
Reference Provided:	**X**	
Learning Objective:	**24524 – Describe the Control Room controls associated with the Non Essential Auxiliary Feedwater Pump AFN-P01 including its indications**	

Technical Reference:	40AO-9ZZ17, Inadvertent PPS-ESFAS Actuations

Attachment C-14 SIAS Train B Page 3 of 3

1-3		SI Line to RC Loop 1A Drain Valve	SIB-HS-638	Closed	Y / N	Open / Closed
1-3		SI Line to RC Loop 1B Drain Valve	SIB-HS-648	Closed	Y / N	Open / Closed
1-3		SI Line to RC Loop 2A Drain Valve	SIB-HS-618	Closed	Y / N	Open / Closed
1-3		SI Line to RC Loop 2B Drain Valve	SIB-HS-628	Closed	Y / N	Open / Closed
1-3		Letdown To Regen Hx Isolation Valve	CHB-HS-515	Closed	Y / N	Open / Closed
	2-4	Backup Heaters Bank	RCB-HS-100-5	Tripped	Y / N	Tripped / Closed
	2-4	Condensate Transfer Pump B	CTB-HS-16	Running	Y / N	Run / Stop
	2-4	Essential Electric Auxiliary Feed Pump	AFB-HS-10	Running	Y / N	Run / Stop

Technical Reference:	40AO-9ZZ17, Inadvertent PPS-ESFAS Actuations

PALO VERDE PROCEDURE INADVERTENT PPS-ESFAS ACTUATIONS	40AO-9ZZ17 Revision 26 Page 89 of 97 Appendix C Page 21 of 27

Attachment C-13 SIAS Train A Page 1 of 4

Actuation Leg		Component	Handswitch	Actuated Condition	In Actuated Condition (Circle one)	As Left Condition (Circle one)
	2-4	Diesel Generator A	DGA-HS-1	Running	Y / N	Run / Stop
	2-4	NHN-M71 (PGA-L33B3)	None	Tripped	Y / N	Tripped / Closed

NHN-M71 Loads

Equipment	Alt Equip	Power	TS / TRM Reference
• QFN-X04, In-Plant Communications • **Unit 3 Only -** AE-QFN-X06, Transformer for UPS/Charger Input Distribution Panel AE-QFN-D21	None	NHN-M7211	None
• HCN-M01A, Ctmt Norm ACU A Disch Damper • HCN-M01C, Ctmt Norm ACU C Disch Damper • SGN-HV-1143, Feedwater Bypass Valve • QMN-C04B3, Liquid Radwaste Ht Trace Panel • SGN-HV-1142, Feedwater Block Valve • SGN-HV-1145, Feedwater Bypass Valve	None	None	

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: AC Electrical Distribution: Knowledge of the effect that a loss or malfunction of the AC Distribution system will have on the following: DC system	**Tier**	**2**	[redacted]	
	Group	**1**	[redacted]	
	K/A	**062 K3.03**		
	IR	**3.7**	[redacted]	

Question 47

Assuming the battery room is maintained a minimum of 60°F during a Station Blackout, with NO operator action, the Class 1E batteries should supply DC system loads for a MINIMUM of…

A. 1 hour

B. 2 hours

C. 4 hours

D. 8 hours

Proposed Answer:	B

Explanations:	
A.	Plausible because in 40EP-9EO08, Blackout there is a 1 hour time requirement to start and place a Station Blackout Generator on a class bus within 1 hour if power is not available from offsite or an EDG.
B.	Correct
C.	Plausible because in 40EP-9EO08, Blackout there is a 4 hour time requirement to start a cooldown if offsite power or an EDG is not restored to a class bus.
D.	Plausible because in 40EP-9EO08, Blackout there is an 8 hour time requirement to align cooling to the Spent Fuel Pool if power has not been restored to a class bus with offsite power or an EDG.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	8	
Reference Provided:	X	
Learning Objective:	18175 – Explain the operation of the Class 1E 125 VDC Batteries under normal operating conditions	

Technical Reference:	Class 125 VDC Power System Tech Manual

System Description

The class 1E 125 VDC power system consists of four independent class 1E 125 VDC sub-systems divided into four channels. The A and B channels provide vital instrumentation and control power via inverters for channels A and B of the reactor protection and ESF systems and diesel generators A and B. The DC sub-systems C and D provide vital instrumentation and control power via inverters for channels C and D for the reactor protection and ESF systems and other safety related loads.

Each sub-system consists of a battery, control center, distribution panel and a battery charger supplied with three phase 480 VAC power from a different class 1E MCC. Each system has two backup chargers: backup charger "AC" for load group 1 and backup charger "BD" for load group 2. Backup charger AC is capable of providing 125 VDC power supply to either channel A or C of the load group 1 and the second backup charger "BD" is capable of providing 125 VDC power supply to either channel B or D of the load group 2. A mechanical interlock is provided between both of the output breakers of the back-up chargers which will prevent simultaneous closing of both of the breakers to both DC control panels, thus eliminating accidental paralleling of both of the DC control panels of two different load groups.

An equalizing charge is given to the battery at a higher than float voltage to correct any non-uniformity between the cell voltages or specific gravities when one or more cells fall below individual cell critical voltage of 2.14 volts corrected for temperature or whose corrected specific gravity has fallen below 1.197. Periodically or immediately after a battery discharge cycle (due to loss of power or failure of the battery charger) the battery is given an equalizing charge or a recharge at a higher voltage per cell than the float charge.

The class 1E 125 VDC Systems are designed for normal operation at a charger float voltage of 135 VDC. During equalizing mode of operation, the system voltage reaches a maximum operating voltage of 139.8 Volts DC.

The class 1E 125 VDC systems are designed for ungrounded (floating system) operation to reduce the possibilities of system degradation due to ground faults. An ungrounded system requires ground faults simultaneously in both the positive and the negative buses before losing the operability of the system.

Each class 1E battery has sufficient capacity to independently supply the required loads while maintaining the minimum required bus voltages for *2 hours* following the loss of battery charger connected to the 125 VDC control center at a minimum temperature of 60°F in the battery room.

Technical Reference:	40EP-9EO08, Blackout

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

In order to be successful in energizing a vital bus from a SBOG within one hour of the start of the event, Appendix 80, Align SBOG to PBA-S03 (BO), Attachment 80-A, Disable PBA-S03 Breakers, (or Appendix 81, Align SBOG to PBB-S04 (BO), Attachment 81-A, Disable PBB-S04 Breakers,) must be started as soon as possible, and performed concurrently with Standard Appendix 53.

✱ 13. IF at least one vital 4.16 kV AC bus is NOT expected to be energized within one hour of the start of the event from EITHER of the following:

- Offsite power
- Diesel Generator

THEN PERFORM Appendix 80, Align SBOG to PBA-S03 (BO).

13.1 IF PBA-S03 is NOT available, THEN PERFORM Appendix 81, Align SBOG to PBB-S04 (BO).

13.2 IF AC power will NOT be available from offsite power, an SBOG, or any Unit's EDG within one hour of the start of the event (ELAP), THEN perform the following:

a. Declare an ELAP is in progress.

b. PERFORM 40MG-9ZZ07, FLEX Support Guidelines.

c. GO TO step 14.

Technical Reference:	40EP-9EO08, Blackout

INSTRUCTIONS	CONTINGENCY ACTIONS
42. **IF** at least one vital 4.16 kV AC bus is **NOT** expected to be energized within four hours of the start of the event from **ANY** of the following: • Offsite Power • Affected Unit Diesel Generator • Other Unit Diesel Generator **THEN** continue in this procedure to commence a cooldown to SDC entry conditions.	

Technical Reference:	40EP-9EO08, Blackout

INSTRUCTIONS	CONTINGENCY ACTIONS
✱ 53. **IF** at least one vital 4.16 kV AC bus is **NOT** expected to be energized within eight hours of the start of the event from **ANY** of the following: • Offsite Power • Affected Unit Diesel Generator • Other Unit Diesel Generator **AND** PBA-S03 is energized, **THEN** perform the following to restore PC Cooling to the Spent Fuel Pool within eight hours of the start of the event: a. PERFORM Appendix 64, Align EW to SFP. b. Direct an operator to start PCA-P01, Fuel Pool Cooling Pump 1.	53.1 **IF** PBB-S04 is energized, **THEN** perform the following to restore PC cooling to the Spent Fuel Pool within eight hours of the start of the event: a. PERFORM Appendix 64, Align EW to SFP. b. Direct an operator to start PCB-P01, Fuel Pool Cooling Pump 2.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: DC Electrical Distribution: Knowledge of the physical connections and/or cause effect relationships between the DC electrical systems and the following systems: AC electrical system	**Tier**	**2**		
	Group	**1**		
	K/A	**063 K1.02**		
	IR	**2.7**		

Question 48

Given the following conditions:

- Unit 2 is operating at 100% power.
- Inverter PNC-N13 Manual Bypass Switch is in the Normal Operation position
- The supply breaker to inverter PNC-N13 was inadvertently opened at PKC-M43

Based on these conditions, PNC-D27 should…

A. NOT automatically align to its alternate power supply. Power can be restored by manually pressing the Bypass Source to Load pushbutton.

B. automatically align to its alternate power supply and should automatically transfer back to its normal source when the inverter is re-energized.

C. NOT automatically align to its alternate power supply. Power can be restored by manually placing the Manual Bypass Switch to the Bypass to Load position.

D. automatically align to its alternate power supply and can be manually realigned to its normal source when the inverter is re-energized by pressing the Inverter to Load pushbutton.

Proposed Answer:	D

Explanations:	
A.	Plausible that it will NOT auto align to the alternate source because unit 1 once did not have static switches with automatic switching capabilities. Also, the examinee may very well think that the Bypass Source to Load pushbutton reverses the last transfer, which would realign the bus to the normal source.
B.	Plausible since it will auto transfer to the alternate source, however it will not auto transfer back to the normal source.
C.	Plausible that it will NOT auto align to the alternate source because unit 1 once did not have static switches with automatic switching capabilities.
D.	Correct

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2016

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	378660 – Explain the operation of the Static Switch provided on Ametek Inverters (new)	

Technical Reference:	120 VAC Class 1E Instrument Power (PN) Lesson Plan

MANUAL BYPASS SWITCH (S1)

This three position switch provides the operator the ability to completely bypass the static switch. The switch positions perform the following functions:

BYPASS TO LOAD – For the normal inverters it connects the distribution panel directly to the swing inverter output through the Remote Swing Lineup switch.

For the swing inverters it connects the output going to the normal inverter, through the Remote Swing Lineup switch, to the voltage regulator.

NORMAL OPERATION - Aligns the distribution panel to the Static Switch.

Technical Reference:	120 VAC Class 1E Instrument Power (PN) Lesson Plan

Static Switch

This device is located in the right cabinet of the inverter. It is an electronic, solid state assembly which, on loss of normal power (inverter), automatically transfers the distribution panel to the voltage regulator without interruption. When power returns, the static transfer switch does NOT transfer back automatically. The distribution panel can be manually transferred by pushing one of the two pushbuttons provided:

INVERTER TO LOAD (Aligns to the inverter)

BYPASS SOURCE TO LOAD (Aligns to the voltage regulator)

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Emergency Diesel Generator: Ability to monitor automatic operation of the ED/G system, including: Load Sequencing	Tier	2		
	Group	1		
	K/A	064 A3.07		
	IR	3.6		

Question 49

Given the following conditions:

- Unit 1 has tripped due to a LOOP

Which of the following loads should automatically start after the EDGs start?

1. ‘A’ Auxiliary Feedwater Pump
2. ‘B’ Auxiliary Feedwater Pump
3. ‘N’ Auxiliary Feedwater Pump

A. 1 ONLY

B. 2 ONLY

C. 1 AND 3 ONLY

D. 2 AND 3 ONLY

Proposed Answer:	**B**

Explanations:	
A.	Plausible because AFA-P01 will start on an AFAS. However, the only pump that starts on a LOOP is AFB-P01.
B.	Correct
C.	First part is plausible because AFA-P01 will start on an AFAS. Second part is plausible if it is thought that both electrical pumps start because if there is a LOOP and a LOP on PBB-S04, there will be no auxiliary feed pumps that automatically start.
D.	First part is correct. Second part is plausible if it is thought that both electrical pumps start because if there is a LOOP and a LOP on PBB-S04, there will be no auxiliary feed pumps that automatically start.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**7**	
Reference Provided:	**N**	
Learning Objective:	**23823 – Explain the operation of the ESF Load Sequencer**	

MODE 3

LOSS OF OFFSITE POWER

A Loss of Power is sensed by 2 out of 4 undervoltage relays on either S03 ("A" train) and/or S04 ("B" train). A Load Shed on the affected train(s) takes place.

LOAD SHED first then Sequence starts when DG BRKR closes

Time Seconds Same as in Mode 2

LOP	Diesel Generators Start
0 - 10 sec	Diesel Generator Output Breaker Closes on S03 and/or S04

----

0)	Sequencer Starts
5	Control Room Essential Ventilation A & B
5	All Essential Batter Chargers and Voltage Regulators Re-energized.
5	Containment Normal Air Handling Units Restart (Previously running units will restart, units in Auto are enabled for Auto Start)
10	+ Auxiliary Feedwater Pump (Essential B)

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Process Radiation Monitoring: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels	**Tier**	**2**		
	Group	**1**		
	K/A	**073 A1.01**		
	IR	**3.2**		

Question 50

Given the following conditions:

- Unit 1 is operating at 100% power

A Steam Generator Tube Leak on SG #1 ___(1)___ cause rising radiation levels on SG #2 RU-142 N-16 Main Steam Line Radiation Monitor and once a downpower is started, INDICATED leak rates on RMS should ___(2)___.

A. (1) SHOULD
(2) lower

B. (1) SHOULD
(2) remain the same

C. (1) should NOT
(2) lower

D. (1) should NOT
(2) remain the same

Proposed Answer:	A

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible if it is thought that because the leak itself has not changed, than indicated leak rate should remain the same.
C.	First is plausible because the steam from SG #2 will not have any activity. However the radiation monitor will detect the radioactivity from SG #1. Second part is correct.
D.	First is plausible because the steam from SG #2 will not have any activity. However the radiation monitor will detect the radioactivity from SG #1. Second part is plausible if it is thought that because the leak itself has not changed, than indicated leak rate should remain the same.

Question Source:	X	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	3	
10CFR55.41:	11	
Reference Provided:	N	
Learning Objective:	**31204 – Explain the basic operation of Process Radiation Monitors**	

Technical Reference:	74AL-9SQ01, Radiation Monitoring System Alarm Validation and Response

OPERATOR ACTIONS

NOTE

- The steam generator tube leak may cause a response on all four channels for RU-142 because of the close proximity of the detectors to each other on the steam lines.
- The highest response will normally be observed on channels monitoring the steam lines of the affected Steam Generator.
- Monitor response to N-16 is proportional to reactor power level.
- Changing power level can result in corresponding changes in monitor readings without a change in leak rate.
- Changing power level can mask leak rate increases.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Service Water: Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Conditions initiating automatic closure of closed cooling water auxiliary building header supply and return valves	Tier	2	[redacted]	
	Group	1	[redacted]	
	K/A	076 K4.01		
	IR	2.5	[redacted]	

Question 51

Given the following conditions:

- Unit 2 is operating at 100% power
- Train ‘A’ EW-NC Cross-Tie Supply and Return Valves, EWA-UV-65 and EWA-UV-145, are open in support of Train ‘A’ EW cross-tied with NC

Which of the following conditions, INDIVIDUALLY, should result in the automatic closure of EWA-UV-65 and EWA-UV-145?

1. Inadvertent Train ‘A’ SIAS
2. Inadvertent Train ‘A’ CSAS
3. Low Level in the ‘A’ EW Surge Tank

A. 2 ONLY

B. 3 ONLY

C. 1 and 2 ONLY

D. 1 and 3 ONLY

Proposed Answer:	D

Explanations:	
A.	Plausible that an ‘A’ CSAS would be correct since an ‘A’ CSAS will stop EW flow through the aux building, however the CSAS stops flow through the aux building by closing the NC CIVs, not the EW-NC cross-tie valves.
B.	Plausible since a low level in the ‘A’ EW Surge Tank will close the EW-NC cross-tie valves, however an ‘A’ SIAS will also close the EW-NC cross-tie valves.
C.	Plausible since an ‘A’ SIAS will close the EW-NC cross-tie valves, and an ‘A’ CSAS will stop flow through the aux building, however the ‘A’ CSAS will not close the EW-NC cross-tie valves. Additionally, low level in the ‘A’ EW Surge Tank will close the EW-NC cross-tie valves.
D.	Correct.

Question Source:		New	
		Bank	
	X	Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	7	
Reference Provided:	N	
Learning Objective:	18454 – Describe the automatic functions associated with the Essential Cooling Water Cross-tie to Nuclear Cooling Water Valves EWA-UV-145 and EWA-UV-65	

Original Question:	2018 NRC Exam Q35 (correct answer was B)

Question 35

Given the following conditions:

- Unit 1 is operating at 100% power
- Train ‘B’ Essential Cooling Water is cross-tied with Nuclear Cooling Water supplying the essential NC loads
- Both NCW pumps are in Pull-to-Lock

Based on these conditions, which of the following conditions, individually, would isolate cooling water to the RCPs?

1. Train ‘B’ SIAS
2. Train ‘B’ CSAS
3. Low Level in the ‘B’ EW Surge Tank

A. 1 ONLY

B. 2 ONLY

C. 1 and 3 ONLY

D. 2 and 3 ONLY

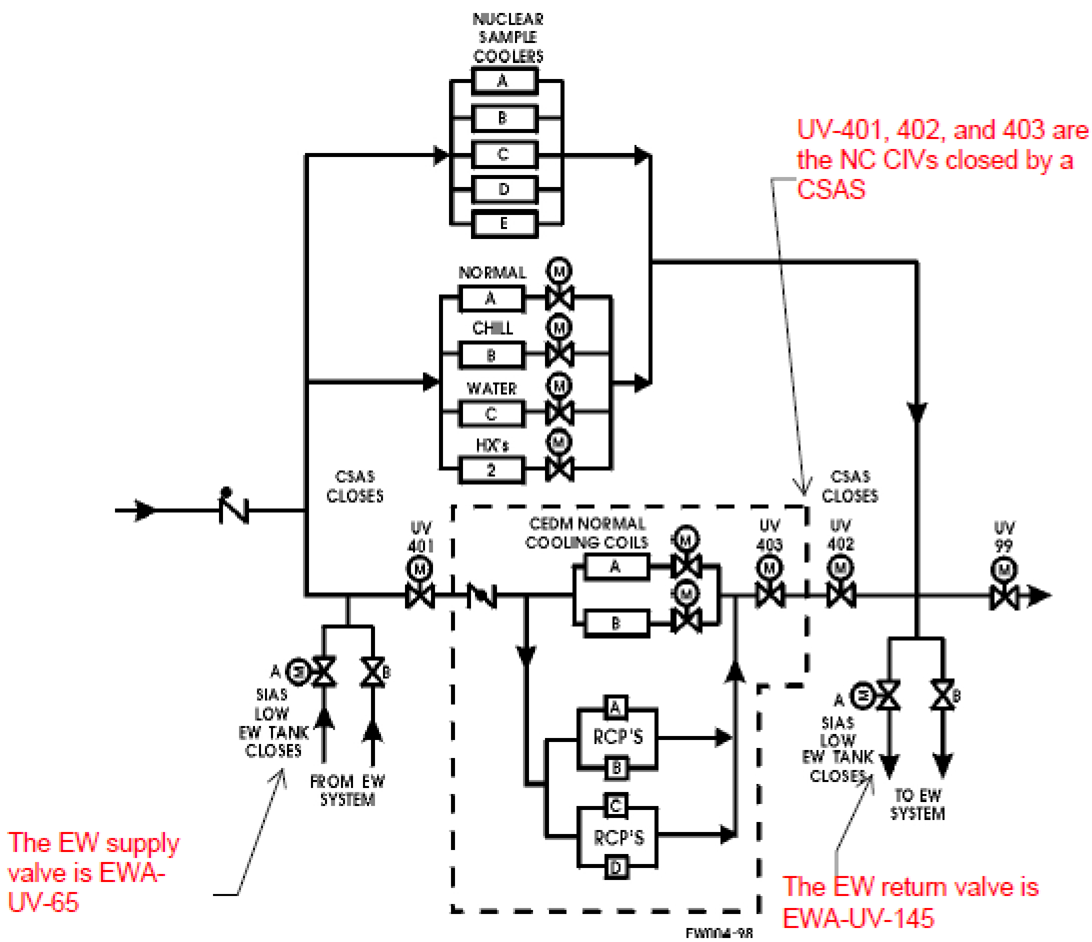
Technical Reference:	Essential Cooling Water Tech Manual

2.7 Piping and Valves

Piping to and from the essential cooling water heat exchangers is carbon steel (corrosion protection is provided by chemical addition). Supply and return piping to and from system components is physically separated from supply and return lines in the redundant flow train.

Two EW system valves can be operated from the control room. They are the isolation valves in the cross-tie lines between EW system train A and the NC system, UV-145 and UV-65. These valves close automatically upon receipt of either a low level signal from the train A surge tank or a SIAS. Manual valves HCV-66 and HCV-146 may be opened to supply the NC system from EW system train B.

Technical Reference:	Essential Cooling Water Tech Manual



Technical Reference:	40AO-9ZZ17, Inadvertent PPS-ESFAS Actuations

Attachment C-7 CSAS Train A

Actuation Leg		Component	Handswitch	Actuated Condition	In Actuated Condition (Circle one)	As Left Condition (Circle one)
1-3		Diesel Generator A	DGA-HS-1	Running	Y / N	Run / Stop
1-3		Control Room Essential AHU Fan A	HJA-HS-28	Running	Y / N	Run / Stop
1-3		Essential Chiller / Chilled Water Pump A	ECA-HS-1A	Running	Y / N	Run / Stop
1-3		Essential Cooling Water Pump A	EWA-HS-1	Running	Y / N	Run / Stop
1-3		Essential Spray Pond Pump A	SPA-HS-1	Running	Y / N	Run / Stop
	2-4	Containment Spray A Discharge to Spray Header 1 Valve	SIA-HS-672	Open	Y / N	Open / Closed
1-3		HPSI Pump A	SIA-HS-1	Running	Y / N	Run / Stop
1-3		Containment Spray Pump A	SIA-HS-5	Running	Y / N	Run / Stop
1-3		LPSI Pump A	SIA-HS-3	Running	Y / N	Run / Stop
1-3		RCP Control Bleed-Off Header to VCT Isolation Valve	CHA-HS-506	Closed	Y / N	Open / Closed
1-3		NCW Containment Downstream Return Isolation Valve	NCA-HS-402	Closed	Y / N	Open / Closed
1-3		Instrument Air Outside Containment Isolation Valve	IAA-HS-2	Closed	Y / N	Open / Closed

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Service Water: Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Service water header pressure	**Tier**	**2**		
	Group	**1**		
	K/A	**076 A2.02**		
	IR	**2.7**		

Question 52

Given the following conditions:

- Unit 1 is in MODE 5
- SDC is in service using Train 'A' auxiliaries and the 'A' LPSI Pump
- A tube leak in the 'A' Essential Cooling Water Heat Exchanger has just occurred

(1) The tube leak in the EW Heat Exchanger should send water from the…

(2) If the EW Pump is stopped in response to the tube leak, the in-service SDCHX can be cooled directly from the…

A. (1) Essential Cooling Water System to the Spray Pond Cooling Water System
(2) Nuclear Cooling Water System

B. (1) Essential Cooling Water System to the Spray Pond Cooling Water System
(2) Spray Pond Cooling Water System

C. (1) Spray Pond Cooling Water System to the Essential Cooling Water System
(2) Nuclear Cooling Water System

D. (1) Spray Pond Cooling Water System to the Essential Cooling Water System
(2) Spray Pond Cooling Water System

Proposed Answer:	C

Explanations: **Part (b) of the KA is met by knowing the procedure to use to restore cooling to the SDCHX. The reason the procedure to use was not included in the stem of the question is that the name of the procedure would give the correct answer away (Appendix 243, NC Cross-Tie to EW Train A)**

A.	First part is plausible since nominal system pressure of the SP system is ~ 50-55 psig compared to EW which has a nominal system pressure of ~ 95 psig, however the EW system is designed such that at the EW heat exchanger, EW pressure is lower than SP pressure to ensure that in the event of an EW HX tube leak, leakage goes from the SP system to the EW system to minimize the potential for environmental contamination. Second part is correct.
B.	First part is plausible since nominal system pressure of the SP system is ~ 50-55 psig compared to EW which has a nominal system pressure of ~ 95 psig, however the EW system is designed such that at the EW heat exchanger, EW pressure is lower than SP pressure to ensure that in the event of an EW HX tube leak, leakage goes from the SP system to the EW system to minimize the potential for environmental contamination. Second part is plausible since the Spray Pond system is the cooling medium for EW, and thus the ultimate heat sink for SDC, however the Spray Pond system cannot be directly lined up to the SDCHX.
C.	Correct.
D.	First part is correct. Second part is plausible since the Spray Pond system is the cooling medium for EW, and thus the ultimate heat sink for SDC, however the Spray Pond system cannot be directly lined up to the SDCHX.

Question Source:		**New**	
	x	**Bank – question was slightly modified but not to the point where the question can be classified as modified per NUREG 1021**	
		Modified	
	x	**Previous NRC Exam**	**2016**

Cognitive Level:		**Memory or Fundamental Knowledge**
	x	**Comprehension or Analysis**

Level of Difficultly:	**3**	
10CFR55.41:	**4**	
Reference Provided:	**N**	
Learning Objective:	**18538 – Describe the design characteristics of the Essential Cooling Water Heat Exchangers**	

Main Idea

The EW system has two separate and redundant trains each consisting of a pump, a heat exchanger, a surge tank, a radiation monitor, and a chemical addition tank.

The EW system is a closed loop system which is cooled by the Spray Pond system and removes heat from the Essential Chillers and the Shutdown Cooling Heat Exchanger.

The EW system can also provide cooling to the Nuclear Cooling Water priority loads (discussed later in the lesson plan) and the Spent Fuel Pool Heat Exchangers in situations when Nuclear Cooling is not available.

The EW system is maintained at a lower pressure than the Spray Pond system at the EW Heat Exchanger to prevent a radioactive release to the environment.

Technical Reference:	40EP-9EO11, Lower Mode Functional Recovery

INSTRUCTIONS	CONTINGENCY ACTIONS
✱ 10. IF the SDC train auxiliaries are NOT available, THEN perform the following: a. IF the SDC auxiliaries are NOT available to the appropriate train, THEN PERFORM ONE of the following: • Appendix 241, LM - SDC Train A using Train B Auxiliaries • Appendix 242, LM - SDC Train B using Train A Auxiliaries b. IF BOTH of the following: • Both SDC train auxiliaries are NOT available • The Nuclear Cooling Water System is available THEN PERFORM ONE of the following: • Appendix 243, LM - NC Cross Tie to EW Train A • Appendix 244, LM - NC Cross Tie to EW Train B	

Examination Outline Cross-Reference:	**Level**	**RO**		**SRO**
K/A: Instrument Air: Knowledge of the physical connections and/or cause-effect relationships between the IAS and following systems: MSIV air	**Tier**	**2**		
	Group	**1**		
	K/A	**078 K1.05**		
	IR	**3.4**		

Question 53

Given the following conditions:

- Unit 2 is operating at 100% power
- An Instrument Air rupture has occurred just downstream of the IA compressors
- IA pressure is at atmospheric pressure throughout the system
- The nitrogen backup supply valve has failed closed

Based on these conditions, the Main Steam Isolation Valves should…

A. slow close due to the loss of IA

B. fast close due to the loss of IA

C. remain open and can only be slow closed

D. remain open and can only be fast closed

Proposed Answer:	D

Explanations:	
A.	Plausible that the MSIVs would fail closed as this is the fail safe position, and the valves are stoked open in slow speed and can be closed in slow speed, however the MSIVs remain open on a loss of instrument air
B.	Plausible that the MSIVs would fail closed as this is the fail safe position, and the valves are normally closed in fast speed, however the MSIVs remain open on a loss of instrument air.
C.	Plausible since the MSIVs will remain open, however slow close is not available on a loss of instrument air.
D.	Correct.

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2019

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	25935 – Determine the major effects on plant operation as instrument air pressure degrade	

Technical Reference:	40AO-9ZZ06, Loss of Instrument Air

Appendix A, Expected Component Failure as System Pressure Drops

PRESS	COMPONENT	ACTION
60 - 50 psig ED	EDN-BTV-3 / 4 / 13 / 14 / 23 / 24 / 59 / 60 / 61 / 69 / 70 / 71 / 73 / 74 / 75, Bleeder Trip Valves (FC)	**NOTE** ARDV-1, Main Turbine Front Standard Turbine Trip Air Relay Dump Valve actuates EDNPSL76 closing the Bleeder Trip valves. Extraction steam flow will maintain the valves in the open position until pressure decays, allowing the valves to close preventing any backflow from the heaters to the turbine.
NC	NCN-LV-75, Nuclear Cooling Water Surge Tank Demin Water Makeup Valve (FC)	____ 1. **IF** makeup will be provided to the NC Surge Tank, **THEN** PERFORM 40OP-9NC01, Nuclear Cooling Water (NC), Alternate Makeup to NC System, to maintain normal level in the NC Surge Tank.
SG	SGE-UV-170 / 171 / 180 / 181, MSIV (FAIL AS IS)	**NOTE** Fast closure operation is available via the accumulator, slow mode valve operation will not be available.
		____ 1. **IF** the MSIVs will be closed, **THEN** fast close the MSIVs using **ANY** of the following: **SG #1** • SGA-HS-251 • SGB-HS-253 **SG #2** • SGA-HS-250 • SGB-HS-252

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Instrument Air: Ability to monitor automatic operation of the IAS, including: Air pressure	**Tier**	**2**		
	Group	**1**		
	K/A	**078 A3.01**		
	IR	**3.1**		

Question 54

During a leak on the Instrument Air header, the Nitrogen Backup Valve should automatically open AS SOON AS header pressure lowers to ___(1)___ psig and should re-close AS SOON AS header pressure rises to ___(2)___ psig.

A. (1) 85
(2) 105

B. (1) 85
(2) 115

C. (1) 95
(2) 105

D. (1) 95
(2) 115

Proposed Answer:	A

Explanations:	
A.	Correct.
B.	First part is correct. Second part is plausible since 115 psig is the middle of the control band for normal IA header pressure (109-119), however the backup N2 valve closes when pressure rises to 105 psig.
C.	First part is plausible since 95 psig is the setpoint for the IA header low pressure alarm, however the N2 backup valve doesn't open until 85 psig. Second part is correct.
D.	First part is plausible since 95 psig is the setpoint for the IA header low pressure alarm, however the N2 backup valve doesn't open until 85 psig. Second part is plausible since 115 psig is the middle of the control band for normal IA header pressure (109-119), however the backup N2 valve closes when pressure rises to 105 psig.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	4	
Reference Provided:	N	
Learning Objective:	31231 – Describe the automatic functions associated with the Instrument Air System	

Technical Reference:	40AO-9ZZ06, Loss of Instrument Air

3.0 LOSS OF INSTRUMENT AIR

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 15. IF it is desired to supply the Instrument Air Header with nitrogen, AND IA Header pressure is less than 85 psig, THEN perform the following:	
a. Check that IAN-PV-52, Nitrogen Backup Valve is open.	a.1 Direct an operator to throttle open IAN-V591, Nitrogen Backup Valve Bypass, to maintain the desired instrument air pressure.

Technical Reference:	Instrument Air System Tech Manual

3.2 Normal Operating Procedure Overview

3.2.1 Instrument Air System

The objectives of this procedure are to place the IA system in service with the compressors supplying the instrument air header through an air dryer, to allow routine transfer of operating compressors and dryers, and to shutdown the IA compressors and dryers.

Placing the Instrument Air System In Service

This section of the procedure will place the instrument air compressors in operating condition, place the air dryers in operating condition and pressurize the instrument air header. The air dryer air operated valves require at least 80 psig to be operable. Once placed in service, the air dryers are fully automatic. There is a nitrogen backup system that will supply the instrument air header if header pressure drops below 85 psig. The nitrogen backup system isolation valve shuts if pressure rises above 105 psig at the discharge of the instrument air dryers.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Containment: Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under shutdown conditions	**Tier**	**2**	[redacted]	
	Group	**1**	[redacted]	
	K/A		**103 K3.01**	
	IR	**3.3**	[redacted]	

Question 55

Given the following conditions:

- Unit 3 is in MODE 4
- A Containment vent is in progress

Subsequently:

- A malfunction causes Containment vent valves to be stuck open
- When the Containment vent valves are closed Containment pressure is -0.5 psig
- After the vent an AO reports that a Containment air lock inner door window has a crack causing it to be INOPERABLE

To maintain compliance with Technical Specifications the crew should raise Containment pressure to a MINIMUM of ___(1)___ psig, and the MINIMUM REQUIRED action(s) is(are) to ___(2)___.

A. (1) -0.3
(2) verify the OPERABLE door is closed in the affected air lock ONLY

B. (1) -0.3
(2) verify the OPERABLE door is closed in the affected air lock AND initiate action to evaluate overall containment leakage rate

C. (1) 0.25
(2) verify the OPERABLE door is closed in the affected air lock ONLY

D. (1) 0.25
(2) verify the OPERABLE door is closed in the affected air lock AND initiate action to evaluate overall containment leakage rate

Proposed Answer:	A

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because evaluating containment leakage rate is the action take if the air lock is INOPERABLE for reasons other than Condition A or B.
C.	First part is plausible because 0.25 is the pressure at which a Containment vent is stopped per 40OP-9CP01, Containment Purge System. Second part is correct.
D.	First part is plausible because 0.25 is the pressure at which a Containment vent is stopped per 40OP-9CP01, Containment Purge System. Second part is plausible because evaluating containment leakage rate is the action take if the air lock is INOPERABLE for reasons other than Condition A or B.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	9	
Reference Provided:	N	
Learning Objective:	Given a set of plant conditions, apply the one hour or less actions statements of T.S. 3.6 in accordance with Tech Spec 3.6	

Technical Reference:	Technical Specifications

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -0.3 psig and ≤ +2.5 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

Technical Reference:	40OP-9CP01, Containment Purge System

Step 6.2.11, Continued

___ B. Ensure CPB-V023, Isolation between Refueling Purge Duct/RU-34 Isolation Valve, is closed.

NOTE

___ Surveillance Requirement 3.3.8.1, Channel Check, is met by 74ST-9SQ07, Radiation Monitoring System Shiftly Surveillance Test.

___ 6.2.12 Direct the Radiation Monitoring Technician to ensure 74ST-9SQ07, Radiation Monitoring System Shiftly Surveillance Test, is current for BOTH of the following:

- ___ RU-37, CP PRE-ACCESS PURGE AREAA
- ___ RU-38, CP PRE-ACCESS PURGE AREAB

___ 6.2.13 IF directed by the SM/CRS,
THEN insert an Emergency Response Facilities Data Acquisition and Display System (ERFDADS) alarm at 0.3 psig to alert the operator NOT to go below 0.25 psig.

___ 6.2.14 IF venting the Containment only for pressure reduction,
THEN maintain Containment pressure greater than 0.25 psig.

Technical Reference:	Technical Specifications

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTES--

1. Entry and exit is permissible to perform repairs on the affected air lock components.
2. Separate Condition entry is allowed for each air lock.
3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	--------------NOTES------------- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. -------------------------------------	(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.1 Verify the OPERABLE door is closed in the affected air lock.	1 hour

Technical Reference:	Technical Specifications			
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	AND		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Control Rod Drive: Ability to monitor automatic operation of the CRDS, including: RCS temperature and pressure	**Tier**	**2**	[redacted]	
	Group	**2**	[redacted]	
	K/A	**001 A3.06**		
	IR	**3.9**	[redacted]	

Question 56

Given the following conditions:

- ‘A’ MFP has tripped causing a RPCB
- All required CEA subgroups have been verified to have fully inserted into the core
- T_{AVG} is 582°F
- T_{REF} is 576°F

Assuming T_{REF} remains constant, Group 3 CEAs are currently inserting at a ___(1)___ rate and should STOP inserting AS SOON AS T_{AVG} is less than ___(2)___ °F.

A. (1) low
(2) 579

B. (1) low
(2) 580.5

C. (1) high
(2) 579

D. (1) high
(2) 580.5

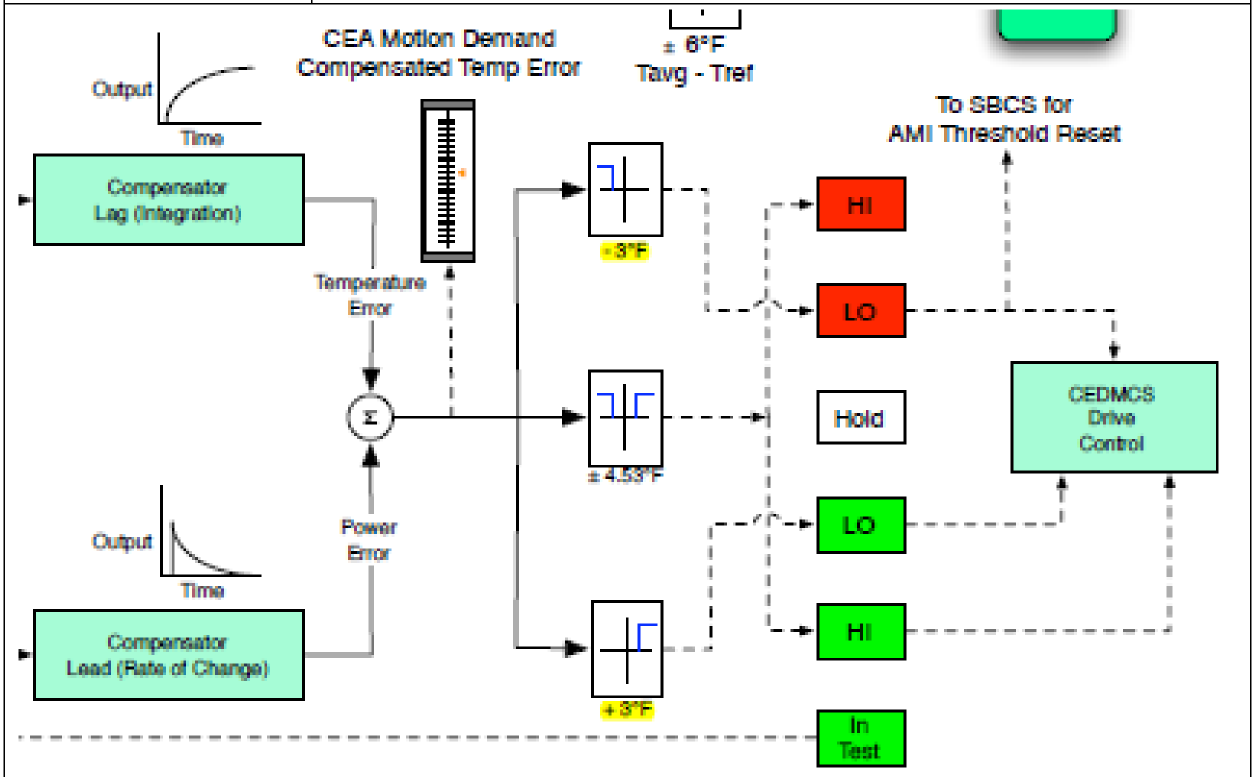
Proposed Answer:	C

Explanations:	
A.	First part is plausible because the CEAs will insert at a low rate once T_{AVG}-T_{REF} deviation is less than 4.5°F. The current deviation of 6°F will cause CEAs to insert at a high rate. Second part is correct.
B.	First part is plausible because the CEAs will insert at a low rate once T_{AVG}-T_{REF} deviation is less than 4.5°F. The current deviation of 6°F will cause CEAs to insert at a high rate. Second part is plausible because at 580.5°F the CEAs will stop inserting at a high rate, however they will still be inserting at a low rate.
C.	Correct
D.	First part is correct. Second part is plausible because at 580.5°F the CEAs will stop inserting at a high rate, however they will still be inserting at a low rate.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**6**	
Reference Provided:	**N**	
Learning Objective:	**19485 – Describe the automatic functions/interlocks associated with CEDMCS**	



CEA Motion Demand
Compensated Temp Error
± 6°F
Tavg - Tref
To SBCS for
AMI Threshold Reset
Output
Time
Compensator
Lag (Integration)
Temperature
Error
-3°F
HI
LO
Σ
± 4.53°F
Hold
CEDMCS
Drive
Control
LO
Output
Time
Power
Error
HI
Compensator
Lead (Rate of Change)
+3°F
In
Test

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Reactor Coolant: Ability to manually operate and/or monitor in the control room: Indications necessary to verify natural circulation from appropriate level, flow, and temperature indications and valve positions upon loss of forced circulation	Tier	2		
	Group	2		
	K/A	002 A4.02		
	IR	4.3		

Question 57

Given the following conditions:

- Unit 1 tripped from 100% power due to a loss of off-site power.
- The crew is verifying natural circulation has been established.

As natural circulation flow develops, the crew should expect to see loop ΔT indicating ___(1)___ 65°F and should expect a delay of approximately ___(2)___ before the RCS temperature response of feeding and steaming adjustments can be verified.

A. (1) less than
   (2) 1 to 2 minutes

B. (1) less than
   (2) 5 to 15 minutes

C. (1) greater than
   (2) 1 to 2 minutes

D. (1) greater than
   (2) 5 to 15 minutes

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible since frequent adjustments of steaming and feeding are needed when controlling in manual (as is the case in a LOOP/LOFC) in order to maintain parameters within post-trip control bands, however in natural circulation conditions, the plant response to these adjustments will not be seen for ~ 5 to 15 minutes.
B.	Correct
C.	First part is plausible since the driving head in natural circulation is developed by the difference in density between the hot and cold legs, therefore a higher delta-T than with forced circulation is plausible, however delta-T must be < 65°F (full power delta-T) in natural circulation conditions. Second part is plausible since frequent adjustments of steaming and feeding are needed when controlling in manual (as is the case in a LOOP/LOFC) in order to maintain parameters within post-trip control bands, however in natural circulation conditions, the plant response to these adjustments will not be seen for ~ 5 to 15 minutes.
D.	First part is plausible since the driving head in natural circulation is developed by the difference in density between the hot and cold legs, therefore a higher delta-T than with forced circulation is plausible, however delta-T must be < 65°F (full power delta-T) in natural circulation conditions. Second part is correct.

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2016

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	14	
Reference Provided:	N	
Learning Objective:	26275 – Explain the difference between single phase and two phase natural circulation	

Technical Reference:	40EP-9EO07, Loss of Off Site Power / Loss of Forced Circulation

PALO VERDE NUCLEAR GENERATING STATION LOSS OF OFF SITE POWER / LOSS OF FORCED CIRCULATION	40EP-9EO07 Revision 31 Page 12 of 54

INSTRUCTIONS	CONTINGENCY ACTIONS
✱ 15. **IF** RCPs are **NOT** operating, **THEN** check natural circulation flow in at least one loop by **ALL** of the following: • Loop ΔT is less than 65°F • Hot and cold leg temperatures are constant or lowering • RCS is 24°F or more subcooled using CET Subcooling • Less than a 30°F ΔT between T_h RTDs and the maximum quadrant CET temperature (QSPDS, pages 211 and 213)	15.1 Ensure proper control of Steam Generator feeding and steaming.

Technical Reference:	40DP-9AP12, Loss of Off Site Power / Loss of Forced Circulation Technical Guideline

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Loss of Offsite Power / Loss of Forced Circulation Technical Guideline	40DP-9AP12	Revision 27

4.5.15 Step 15 - Ensure Natural Circulation

A. The intent of this step is to check that natural circulation flow is established and is supporting RCS heat removal.

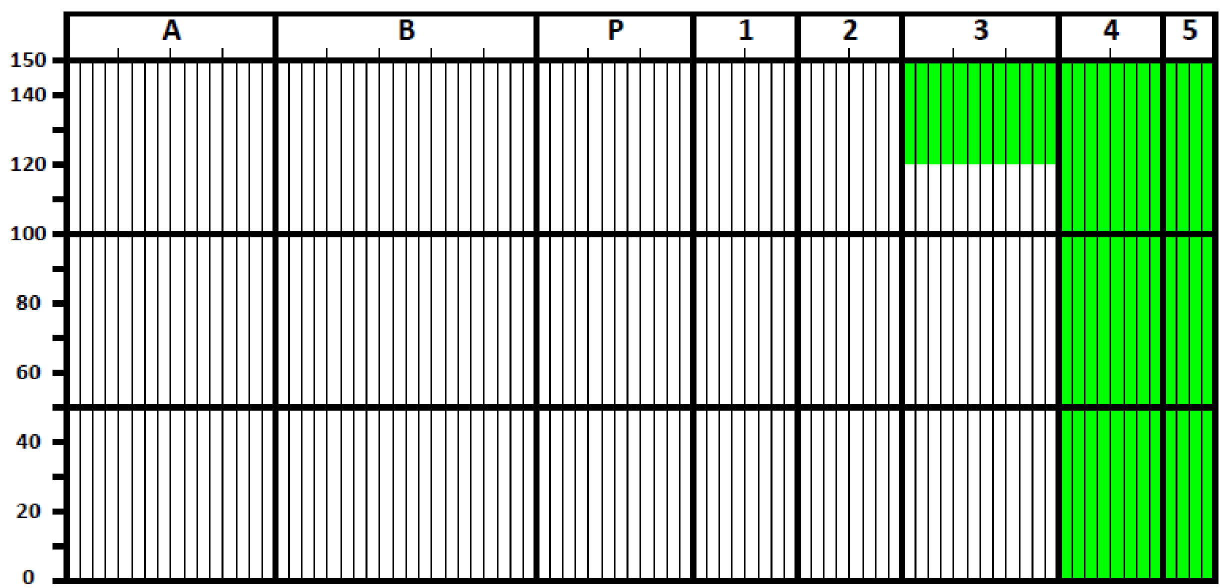
After RCPs are tripped, natural circulation flow should develop within 5 - 15 minutes (longer if the plant tripped from a low power). Natural circulation flow will continue as long as RCS pressure and inventory control are maintained and at least one steam generator is available for heat removal.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Rod Position Indication: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup	**Tier**	**2**		
	Group	**2**		
	K/A	**014 G 2.1.31**		
	IR	**4.6**		

Question 58

Given the following conditions:

- Unit 3 was operating at 100% power when a Reactor Power Cutback occurred
- The CRS has entered 40AO-9ZZ09, Reactor Power Cutback (Loss of Feedpump)
- All automatic CEA motion has stopped
- The CRS has just directed the OATC to restore CEA overlap
- Current CEA positions are as follows:



Per 40AO-9ZZ09, Reactor Power Cutback (Loss of Feedpump), prior to commencing the restoration of CEA overlap, the OATC should ensure that the CEDMCS Mode Selector Switch is selected to ___(1)___ and the FIRST CEA Reg Group to be withdrawn should be ___(2)___ .

A. (1) Manual Group
   (2) Reg Group 3

B. (1) Manual Group
   (2) Reg Group 4

C. (1) Manual Sequential
   (2) Reg Group 3

D. (1) Manual Sequential
   (2) Reg Group 4

Proposed Answer:	**B**

Explanations:	
A.	First part is correct. Second part is plausible since Group 3 CEAs were the last to insert so it would make sense that they would be first to withdraw, however prior to withdrawing Group 3 CEAs, Group 4 CEAs must be withdrawn until they are within 95 inches of Group 3 CEAs.
B.	Correct.
C.	First part is plausible since CEAs will be restored to an ARO condition using manual sequential control, however when re-establishing CEA group overlap, manual group is used. Second part is plausible since Group 3 CEAs were the last to insert so it would make sense that they would be first to withdraw, however prior to withdrawing Group 3 CEAs, Group 4 CEAs must be withdrawn until they are within 95 inches of Group 3 CEAs.
D.	First part is plausible since CEAs will be restored to an ARO condition using manual sequential control, however when re-establishing CEA group overlap, manual group is used. Second part is correct.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**6**	
Reference Provided:	**N**	
Learning Objective:	**19515 – Explain how electric control of the CEDMs is achieved**	

Technical Reference:	40AO-9ZZ09, Reactor Power Cutback (Loss of Feedpump)

____ 30. Determine the normal overlap position for RG-4.

RG-3 position ______ inches

- 95

RG-4 = _______ Inches

Technical Reference:	40AO-9ZZ09, Reactor Power Cutback (Loss of Feedpump)

____ 31. **IF** CEA Reg Group 3 is higher than 95 inches withdrawn, **THEN** perform the following to restore normal CEA group overlap:

a. PERFORM Appendix E, Reactivity Impact While Restoring CEA Overlap.

b. Monitor CEA alignment using the CEAC CRT when moving CEAs.

c. Maintain the Tave/Tref mismatch within ± 3°F.

d. Wait a minimum of 1 minute between CEA pulls.

e. Withdraw Reg Group 4 in Manual Group "MG" in 10 inch increments to 95 inches below the position of Reg Group 3 while closely monitoring the reactor response.

Examination Outline Cross-Reference:	**Level**	**RO**		**SRO**
K/A: In-Core Temperature Monitor: Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors	**Tier**	**2**		
	Group	**2**		
	K/A	**017 K6.01**		
	IR	**2.7**		

Question 59

Given the following conditions:

- Unit 3 is operating at 100%
- One Core Exit Thermocouple (CET) sensor has just failed out of range low

The failure of this CET should be indicated on QSPDS by ___(1)___ and the input from the failed CET into the overall CET calculation should be ___(2)___ .

A. (1) “NO DATA”
(2) ignored by QSPDS

B. (1) “NO DATA”
(2) replaced by a canned value

C. (1) question marks
(2) ignored by QSPDS

D. (1) question marks
(2) replaced by a canned value

Proposed Answer:	C

Explanations:	
A.	First part is plausible since invalid inputs on Plant PI are indicated by "NO DATA", however QSPDS uses a string of question marks to indicate a failed sensor. Second part is correct.
B.	First part is plausible since invalid inputs on Plant PI are indicated by "NO DATA", however QSPDS uses a string of question marks to indicate a failed sensor. Second part is plausible as a canned value is used in the DFWCS for inputs that are out of range, however CET data that is out of range is simply ignored by QSPDS.
C.	Correct.
D.	First part is correct. Second part is plausible as a canned value is used in the DFWCS for inputs that are out of range, however CET data that is out of range is simply ignored by QSPDS.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	2	
Reference Provided:	N	
Learning Objective:	19076 – Describe the Control Room indications associated with the QSPDS system	

Technical Reference:	QSPDS Lesson Plan

CET Failure

All CET inputs are examined for an out-of-range condition which would indicate a failed sensor. The out-of-range sensor is flagged by QSPDS and not used for further calculations unless accident

conditions exist. You could determine the existence of a CET out-of-range by a display of question marks instead of a temperature.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Containment Iodine Removal: Knowledge of the physical connections and/or cause effect relationships between the CIRS and the following systems: CSS	**Tier**	**2**		
	Group	**2**		
	K/A	**027 K1.01**		
	IR	**3.4**		

Question 60

The amount of gaseous iodine in the containment atmosphere is minimized during normal conditions by the use of ___(1)___ filters and is minimized during a LOCA by maintaining pH of the water in containment ___(2)___ 7.0.

A. (1) HEPA
(2) less than

B. (1) HEPA
(2) greater than

C. (1) charcoal
(2) less than

D. (1) charcoal
(2) greater than

Proposed Answer:	D

Explanations:	
A.	First part is plausible since HEPA filters are used in several air filtration units throughout the plant and filter our micro particles from the air, however the iodine is filtered by use of charcoal filters. Second part is plausible since the water injected into the core during a LOCA is a boric acid solution, and boric acid has a pH less than 7.0, however in order to maintain iodine in solution, trisodium phosphate is added to the water to raise the pH to greater than 7.0.
B.	First part is plausible since HEPA filters are used in several air filtration units throughout the plant and filter our micro particles from the air, however the iodine is filtered by use of charcoal filters. Second part is correct.
C.	First part is correct. Second part is plausible since the water injected into the core during a LOCA is a boric acid solution, and boric acid has a pH less than 7.0, however in order to maintain iodine in solution, trisodium phosphate is added to the water to raise the pH to greater than 7.0.
D.	Correct

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2016

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	13	
Reference Provided:	N	
Learning Objective:	Explain the operation of the Containment Building Pre-Access Filtration AFUs (HCNF01A and B) under normal operating conditions	

Technical Reference:	Containment HVAC System Tech Manual

Electric heaters are installed in the common distribution duct. These heaters can be energized as necessary to maintain Containment temperatures above 50°F during shutdowns. The Containment air cooling can be maintained by two of the four units. Temperature indicators for each level in the Containment are provided in the control room.

- Normal Cleanup (Figure A-1)

The Containment Normal Cleanup system consists of two 50% capacity, pre-access air filtration units (AFUs). Each AFU (HCN-F01A, B) consists of one high efficiency filter, two high efficiency particulate air (HEPA) filters, one charcoal filter, and a fan. During cleanup system operation, air is drawn through the filters by the associated fan and discharged directly to Containment atmosphere. The high efficiency filters remove particulate materials and the charcoal filters adsorb fission product gases (mainly radio-iodine) to minimize Containment atmospheric contaminants. This process cleans up the internal air without the need for dilution via outside air.

EO: 1.25 Describe the Recirculation Sumps and Trisodium Phospate baskets.

Main Idea

The recirculation sumps, one for each train, are large screened sumps in the basement of the containment building. Their purpose is to collect water released to the containment by either the safety injection system, through a break in the RCS, or the containment spray system, or a steam line rupture within the containment . When the normal source of SI and CS water (RWT) is depleted, the recirculation sump valves auto open to allow the HPSI pumps and CS pumps to take a suction on the water volume in containment. The signal that places the sumps in service is a recirculation actuation signal (RAS) and is generated by a low level in the RWT. The RAS in addition to opening the sump valves also trips the LPSI pumps to ensure adequate NPSH to the HPSI and CS pumps.

The recirculation sumps have metal screens located above the sump to prevent trash from entering the sump and the SI pumps after a LOCA.

Located in various locations on the floor of the containment are a number of baskets holding tri-sodium phosphate (TSP). When the containment is flooded, this chemical is dissolved in the water and raises the pH of the recirc water. The primary purpose of which is to maintain iodine in solution while minimizing the stress-corrosion cracking of stainless steel and reducing the generation of hydrogen by the corrosion of containment metals.

The TSP is required to adjust the pH of the recirculation water to $\geq$ 7.0 after a LOCA. A pH of greater than 7.0 is necessary to prevent significant amounts of Iodine dissolved in the coolant from becoming volatile and entering containment atmosphere. A pH of greater than 7.0 also helps to prevent stress corrosion cracking of stainless steel components in containment.

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Hydrogen Recombiner and Purge Control: Knowledge of bus power supplies to the following: Hydrogen recombiners	Tier	2	[redacted]	
	Group	2	[redacted]	
	K/A	028 K2.01		
	IR	2.5	[redacted]	

Question 61

The power supply to Hydrogen Recombiners is a ___(1)___ and ___(2)___.

A. (1) 480V Class Bus
   (2) is hardwired to the recombiners

B. (1) 480V Class Bus
   (2) must be manually connected to the recombiners

C. (1) 4.16 kV Class Bus
   (2) is hardwired to the recombiners

D. (1) 4.16 kV Class Bus
   (2) must be manually connected to the recombiners

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible because another component that is used for Containment Hydrogen during an accident, Hydrogen Analyzers, are hardwired into the electrical system.
B.	Correct
C.	First part is plausible because the power supply is Class power, however it is only 480 VAC. Second part is plausible because another component that is used for Containment Hydrogen during an accident, Hydrogen Analyzers, are hardwired into the electrical system.
D.	First part is plausible because the power supply is Class power, however it is only 480 VAC. Second part is correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	8	
Reference Provided:	N	
Learning Objective:	18259 – Describe the response of the Class AC Distribution System to an abnormal/emergency operating condition	

EO: 1.6 Explain the operation of the Hydrogen Recombiner under normal operating conditions.

Introduction

Per the Design Bases Manual the Hydrogen Recombiners are expected to be installed 72 hours after the start of a LOCA

Per the Design Bases Manual the Hydrogen Recombiners are expected to be place in service 100 hours after the start of a LOCA

Our LOCA EOP 40EP-9EO03 directs aligning the Hydrogen Recombiners per standard appendix 19 when CSAS has actuated

Our LOCA EOP 40EP-9EO03 directs placing the Hydrogen Recombiners in service per standard appendix 19 when hydrogen concentration reaches .7% & Containment pressure is < 8.5 psig (Narrow Range Containment pressure.)

40EP-9EO03 LOCA Safety Function – Containment Combustible Gas Control
Condition 1 – Hydrogen Concentration is less than .7%
Condition 2 All available Hydrogen Recombiners are Operating and hydrogen concentration is less than 4.5%

The Recombiner trip setpoints are optional and are not testable but are supplied for the student's information only.

Main Idea

The Recombiners are located in the 100' Aux Bldg in Unit 1. When required, the skid mounted unit is moved to the appropriate unit and connected to Class 1E electrical power. ("A" recombiner from PHA and "B" recombiner from PHB). When the recombiner is not in service, a low power trickle heater is turned on to maintain the recombiner free of moisture.

Examination Outline Cross-Reference:	**Level**	**RO**		**SRO**
K/A: Steam Dump/Turbine Bypass Control: Knowledge of the effect that a loss or malfunction of the SDS will have on the following: RCS	**Tier**	**2**		
	Group	**2**		
	K/A	**041 K3.02**		
	IR	**3.8**		

Question 62

Given the following conditions:

- Unit 2 is operating at 100% power
- NNN-D11 is de-energized

Subsequently:

- The Main Turbine trips

The Reactor should trip on ___(1)___ and the crew should control RCS temperature with ___(2)___.

A. (1) High Pressurizer pressure - RPS
(2) ADVs

B. (1) High Pressurizer pressure - RPS
(2) SBCV-1007 and 1008

C. (1) High Pressurizer pressure - SPS
(2) ADVs

D. (1) High Pressurizer pressure - SPS
(2) SBCV-1007 and 1008

Proposed Answer:	**A**

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because there are malfunction (loss of vacuum) that will allow for the use of SBCV-1007 and 1008. However the loss of NNN-D11 will cause a loss of power to SBCS and no valves can be operated automatically or manually.
C.	First part is plausible because if the Reactor failed to trip on High Pressure at 2383 psia, the SPS trip will trip the Reactor at 2409 psia. Second part is correct.
D.	First part is plausible because if the Reactor failed to trip on High Pressure at 2383 psia, the SPS trip will trip the Reactor at 2409 psia. Second part is plausible because there are malfunction (loss of vacuum) that will allow for the use of SBCV-1007 and 1008. However the loss of NNN-D11 will cause a loss of power to SBCS and no valves can be operated automatically or manually

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**4**	
Reference Provided:	**N**	
Learning Objective:	**25753 – Given a loss of non-class instrument power, describe how the loss impacts the operation of SBCS in accordance with 40AO-9ZZ14**	

Technical Reference:	Operator Information Manual

Loss of NNN-D11	NNN-D11 supplies power to instruments and Logic Power Assemblies within SBCS. It also supplies power to the SBCS Master controller.	No Power	Effect on SBCS 1. SBCS loses Logic Power, SBCS valves fail closed and can not be operated in Manual or Auto. When energized SBCS will come back in Manual with zero output and may also be in Emergency Off or Disconnected.

Examination Outline Cross-Reference:	**Level**	**RO**		**SRO**
K/A: Main Turbine Generator: Ability to (a) predict the impacts of the following malfunctions or operation on the MT/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Steam dumps are not cycling properly at low load, or stick open at higher load (isolate and use atmospheric reliefs when necessary)	**Tier**	**2**		
	Group	**2**		
	K/A	**045 A2.08**		
	IR	**2.8**		

Question 63

Given the following conditions:

- Unit 3 is operating at 100% power
- Core life is MOC

Subsequently:

- The Main Turbine tripped
- 10 minutes after the Main Turbine Trip Reactor Power stabilizes at 60%
- SBCV-1001 and SBCV-1004 are both FULL open

(1) If automatic control of SBCV-1001 is lost and SBCV-1001 Mode Selector Switch is taken to 'OFF', the FIRST set of valves to modulate to pick up steam load is…

(2) With NO operator action, over the next 4 hours, the SBCVs that modulated open after SBCV-1001 failed should modulate in the…

A. (1) SBCV-1002 & SBCV-1005
(2) open direction

B. (1) SBCV-1002 & SBCV-1005
(2) closed direction

C. (1) SBCV-1003 & SBCV-1006
(2) open direction

D. (1) SBCV-1003 & SBCV-1006
(2) closed direction

Proposed Answer:	D

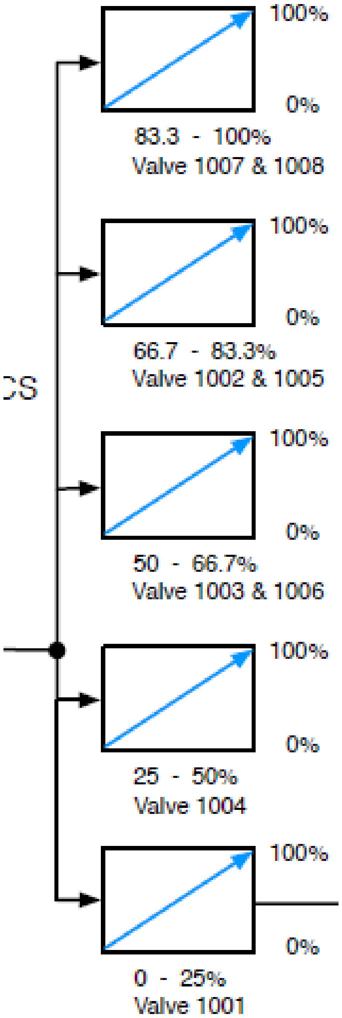
Explanations:	
A.	The first part is plausible because SBCV-1002 & SBCV-1005 are the next valves numerically. Second part is plausible because eventually xenon will decay away and the SBCVs will modulate open. However over the first 4 hours, xenon will be building in.
B.	The first part is plausible because SBCV-1002 & SBCV-1005 are the next valves numerically. Second part is correct.
C.	First Part is correct. Second part is plausible because eventually xenon will decay away and the SBCVs will modulate open. However over the first 4 hours, xenon will be building in.
D.	Correct

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	1	
Reference Provided:	N	
Learning Objective:	378910 – Describe the overall system operation of the Steam Bypass Control System	

Technical Reference:	Operator Information Manual



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Waste Gas Disposal: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Isolation of waste gas release tanks	Tier	2		
	Group	2		
	K/A		071 K4.04	
	IR	2.9		

Question 64

Given the following conditions:

- Unit 2 is venting the RDT to the Waste Gas system

Subsequently:

- An inadvertent CIAS occurs

The Waste Gas header should be isolated by ___(1)___ Containment Isolation valve(s) and if header pressure downstream of the Containment Isolation Valve(s) rises, there is a relief valve that should lift and relieve ___(2)___.

A. (1) one
   (2) DIRECTLY to the Radwaste Building Exhaust

B. (1) one
   (2) to the Radwaste Building Exhaust via the Gaseous Discharge Header Release path

C. (1) two
   (2) DIRECTLY to the Radwaste Building Exhaust

D. (1) two
   (2) to the Radwaste Building Exhaust via the Gaseous Discharge Header Release path

Proposed Answer:	C

Explanations:	
A.	First part is plausible because there are other systems that are isolated by only one Containment Isolation valves (e.g RDT Makeup Valve CHA-UV-580). Second part is correct.
B.	First part is plausible because there are other systems that are isolated by only one Containment Isolation valves (e.g RDT Makeup Valve CHA-UV-580). Second part is plausible because the Waste Gas Decay tanks are released through this path.
C.	Correct
D.	First part is correct. Second part is plausible because the Waste Gas Decay tanks are released through this path.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	9	
Reference Provided:	N	
Learning Objective:	20262 - Describe the Gaseous Release Flowpath	

Technical Reference:	Gaseous Radwaste System Tech Manual

Gas Surge Header Containment Isolation Valve Controls (HS-1, 2)

Two containment isolation valves (GR-UV-1 and GR-UV-2) provide isolation of the reactor drain tank, containment refueling failed fuel detector vent and the dry sipping auxiliary pump from the gas surge header. GR-UV-1 is a motor operated valve and is located inside containment. GR-UV-2 is a solenoid operated valve and is located outside containment. Both valves will close upon receipt of a containment isolation actuation signal (CIAS). Both valves are controlled from handswitches (HS-1, 2)

EO: 1.4 Explain the operation of the following components under normal operating conditions:

- **Gas Surge Header**
- **Gas Surge Header Containment Isolation Valve**
- **Gas Surge Tank (GRN-X01)**
- **Gas Compressor Pre-filters (GRN-F02A, F02B)**
- **Gas Compressors (GRN-C01A, C01B)**
- **Decay Tanks (GRN-X02A, X02B, X02C)**
- **Decay Tank Disch Header Isolation Valves (GRN-UV-34A, 34B)**
- **Gaseous Discharge Header Isolation Valves (HS-34A and 34B)**
- **Radiation Monitor (RU-12)**
- **Gaseous Discharge Flow Control Valve (GRN-FV33)**

Introduction

This objective requires that the license candidate understand the operation of the various components that make up this system.

Main Idea

Gas Surge Header

The 2 inch diameter gas surge header receives radioactive and potentially radioactive gases from the following sources:

- Reactor drain tank
- Gas stripper
- Volume control tank
- Volume control tank relief
- O2 analyzer return

The gas surge header is equipped with isolation valves located both inside and outside the containment. These valves close automatically upon receipt of a CIAS. A spring-loaded relief valve (GR-PSV-3) relieves to the radwaste building exhaust system to prevent over pressurization of the header.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Area Radiation Monitoring: Knowledge of the operational implications of the following concepts as they apply to the ARM system: Radiation theory, including sources, types, units, and effects	**Tier**	**2**		
	Group	**2**		
	K/A		**072 K5.01**	
	IR	**2.7**		

Question 65

Control Room Area Radiation Monitor, RU-18, measures ___(1)___ radiation and when it rises to the alarm setpoint ___(2)___ auto actuate CREFAS.

A. (1) neutron
   (2) SHOULD

B. (1) neutron
   (2) should NOT

C. (1) gamma
   (2) SHOULD

D. (1) gamma
   (2) should NOT

Proposed Answer:	D

Explanations:	
A.	First part is plausible because neutron radiation is highly hazardous, however RU-18 measures gamma radiation. Second part is plausible because RU-18 does monitor radiation levels around the Control Room, however only RU-29 and RU-30 directly will automatically actuate CREFAS.
B.	First part is plausible because neutron radiation is highly hazardous, however RU-18 measures gamma radiation. Second part is correct.
C.	First part is correct. Second part is plausible because RU-18 does monitor radiation levels around the Control Room, however only RU-29 and RU-30 directly will automatically actuate CREFAS.
D.	Correct

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	11	
Reference Provided:	N	
Learning Objective:	23911 - Explain the basic operation of Area Radiation Monitors	

Technical Reference:	LOIT Radiation Monitoring Lesson Plan

Title: Radiation Monitoring System Lesson Plan **Lesson Plan #:** NKASYC15507

EO: 1.4 Explain the basic operation of Area Radiation Monitors.

Introduction

Area radiation monitors are essential for the protection of our personnel.

Main Idea

As their name implies, area radiation monitors are used to monitor for general radiation fields in a physical area or space. Although not always the rule, area monitors are generally for protection of personnel and are measuring gamma radiation fields. Area monitors:

Technical Reference:	Radiation Monitoring System Tech Manual

2.27 Control Room Ventilation Intake Monitors, (CRVA) SQA-RU-29 and (CRVB) SQB-RU-30

These radiation monitors monitor the noble gas concentrations in the control room air intake. The primary function of these monitors is to provide engineered safety feature actuation on high-high alarm activating the control room essential filtration units (CREFAS). These monitors are redundant with their sample points as near the intake as practical (see drawing 13-M-HJP-001).

Technical specifications apply. Required monitor features for operability are the gas channel, sample pump, low sample flow detection, associated ESF actuation capability, and control room indication and alarm.

2.17 Control Room Area Monitor, (CRA) SQN-RU-18

The CRA radiation monitor continuously monitors radiation levels in the main control room. The primary function of this monitor is to provide warning to personnel of abnormal radiation levels in the control room, particularly under post accident conditions. The monitor has a local indication and alarm module. This function provides warning of abnormal radiation levels thus providing for protective actions (see drawing 13-J-ZJF-009).

This monitor has no operability requirements per the technical specifications. Backup monitoring using portable instrumentation can be employed in the event that this monitor fails.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Conduct of Operations: Knowledge of shift or short-term relief turnover practices	Tier	3		
	Group			
	K/A		G 2.1.3	
	IR	3.7		

Question 66

Given the following conditions:

- You are preparing to take the shift as the OATC
- The last shift you worked was 5 days ago

Per 40DP-9OP33, Shift Turnover:

(1) PRIOR to turnover, you must review the Unit Logs going back a MINIMUM of…

(2) AFTER turnover, how much more of the Unit Logs, if any, must be reviewed?

A. (1) 3 days
(2) No additional Unit Logs review is required

B. (1) 3 days
(2) 2 additional days of Unit Logs review is required

C. (1) 5 days
(2) No additional Unit Logs review is required

D. (1) 5 days
(2) 2 additional days of Unit Logs review is required

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible since no additional log review would be required if the last shift was within the last 3 days, however since the last shift was 5 days ago, an additional 2 days of log review is required.
B.	Correct.
C.	First part is plausible since 5 days of logs are required to be reviewed, however the minimum requirement for log review prior to turnover is 3 days or until the last shift, whichever is SHORTER. Second part is plausible as it would be correct if 5 days of log review was required prior to turnover, however since only 3 days of log review were required in part 1, the answer is incorrect.
D.	First part is plausible since 5 days of logs are required to be reviewed, however the minimum requirement for log review prior to turnover is 3 days or until the last shift, whichever is SHORTER. Second part is plausible since the requirement for log review after turnover is since the last shift worked or 7 days, but the requirement is the shorter of those two options, which in this case would be 5 days.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	25772 – Given the conditions associated with Control Room relief, describe the required review of operating logs prior to this relief in accordance with 40DP-9OP33	

Technical Reference:	40DP-9OP33, Shift Turnover

4.4.3 Prior to turnover (Oncoming OATC/BOP/TRO), perform the following:

A. Review Unit Logs back to the last shift worked or the previous three days, whichever is shorter.

Technical Reference:	40DP-9OP33, Shift Turnover

4.4.4 After turnover, perform the following:

A. Review LCO tracking log or TSCCR book.

B. Review Effluent Release Permits.

C. Review Chemistry Control Instructions.

D. Review remaining Unit Logs back to the last shift worked or seven days, whichever is shorter.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Conduct of Operations: Knowledge of the administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc	Tier	3		
	Group			
	K/A	G 2.1.15		
	IR	2.7		

Question 67

Given the following conditions:

- An Operational Decision Making Issue (ODMI) has been issued for a Pressurizer safety valve that is leaking by
- The crew is calculating RCS leakage from the Pressurizer Safety valve every hour to determine if 1 GPM is exceeded and additional action needs to be taken

The ODM Action Plan is approved by the ___(1)___ and if 1 GPM is exceeded the crew should refer to the ___(2)___ point section of the ODMI.

A. (1) Plant Manager
(2) hold

B. (1) Plant Manager
(2) trigger

C. (1) Operations Director
(2) hold

D. (1) Operations Director
(2) trigger

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible because a “hold” point is a term used for radiation exposure to maintain individual and collective doses ALARA and prevent exceeding dose limits.
B.	Correct
C.	First part is plausible because the Operations Director oversees Operations for all 3 units, however the Plant Manager approves the Action Plan. Second part is plausible because a “hold” point is a term used for radiation exposure to maintain individual and collective doses ALARA and prevent exceeding dose limits.
D.	First part is plausible because the Operations Director oversees Operations for all 3 units, however the Plant Manager approves the Action Plan. Send part is correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	203144 – Describe the Operational Decision Making process	

1.2.2 This procedure is intended to provide Shift Managers and Station Management with a tool to assist in decisions discussed in the first and second scenarios. In these scenarios, degraded conditions may involve reductions of safety margins and can occur over a period of hours, days, weeks, months, or entire operating cycles. Industry examples include primary system and containment leakage that remains below operational license limits, numerous long term pump and valve leaks, fuel defects, and the aggregate of equipment or material deficiencies. Such operational decisions often send a strong message to the station staff and have lasting effects on future decisions. Focusing management attention on the specific demands associated with effective operational decision-making can help stations avoid significant plant events.

1.2.3 This process does not address issues that affect the operability or functionality of equipment but may be entered after operability or functionality is determined to evaluate whether continued unit operation is prudent or address actions related to management of the issue.

1.2.4 The ODMI process is not intended to assign action(s) to fix the incident condition.

2.0 RESPONSIBILITIES

2.1 Plant Manager

2.1.1 The Plant Manager or Designee is responsible for the following:

- Oversight of the ODM process.
- Selection of appropriate personnel for Operational Decisions and Operational Decision Teams, including the ODM management sponsor.
- Approval of the ODM Action Plan decisions made and documented via this procedure.

Technical Reference:	01DP-0ZZ01, Operational Decision Making

Operational Decision Making	01DP-0ZZ01	Revision 6

3.5 **Trigger Points** — Explicit Operational Decision Making Issue Action Plan criteria that define those key worsening parameters and the specific actions required by site management and/or Unit operating personnel to address these degrading conditions; the subsequent tasks necessary to better control, lessen, or mitigate the worsening Operational Decision Making Issue conditions observed are also included within this listing.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Equipment Control: Knowledge of Surveillance procedures	Tier	3		
	Group			
	K/A	G 2.2.12		
	IR	3.7		

Question 68

Per 40ST-9ZZM1, Operations Mode 1 Surveillance Logs:

(1) Appendix B, Mode 1 SHIFTLY Surveillance Logs Data Sheets, must have the Acceptance Review completed NO LATER THAN…

(2) Appendix C, Mode 1 DAILY Surveillance Logs Data Sheets, is directed to be performed during…

A. (1) 0800 on day shift and 2000 on night shift
(2) day shift

B. (1) 0800 on day shift and 2000 on night shift
(2) night shift

C. (1) 1100 on day shift and 2300 on night shift
(2) day shift

D. (1) 1100 on day shift and 2300 on night shift
(2) night shift

Proposed Answer:	D

Explanations:	
A.	First part is plausible since 0800 is the earliest the shiftly logs can be completed and reviewed, however the latest is 1100. Second part is plausible since some of the Mode 1 daily surveillances are done on the day shift (i.e. ISFSI daily checks), however the Daily Surveillance Logs Data Sheets are done on the night shift.
B.	First part is plausible since 0800 is the earliest the shiftly logs can be completed and reviewed, however the latest is 1100. Second part is correct.
C.	First part is correct. Second part is plausible since some of the Mode 1 daily surveillances are done on the day shift (i.e. ISFSI daily checks), however the Daily Surveillance Logs Data Sheets are done on the night shift.
D.	Correct

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2016

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	27053 – Describe the responsibilities of the Reactor Operator with respect to logkeeping	

Appendix B - Mode 1 Shiftly Surveillance Logs Data Sheets

NOTE

___ • Appendix B - Mode 1 Shiftly Surveillance Logs Data Sheets, should be started prior to 0800 on the Day Shift and prior to 2000 on the Night Shift.

• Appendix B - Mode 1 Shiftly Surveillance Logs Data Sheets, should be completed and Acceptance Reviewed between 0800 to 1100 on the Day Shift and between 2000 and 2300 on the Night Shift.

Technical Reference:	40ST-9ZZM1, Operations Mode 1 Surveillance Logs

Appendix C - Mode 1 Daily Surveillance Logs Data Sheets

NOTE

___ • Appendix C is started prior to 2000 on the Night Shift.

• Appendix C is completed and Acceptance Reviewed between 2000 and 2300 on the Night Shift.

Technical Reference:	40ST-9ZZM1, Operations Mode 1 Surveillance Logs

Appendix D - Unit 1 ISFSI Temperature Monitoring Checks

NOTE

___ • Appendix D is started prior to 0800 on the Day Shift.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Equipment Control: Knowledge of tagging and clearance procedures	Tier	3		
	Group			
	K/A	G 2.2.13		
	IR	4.1		

Question 69

Per 40DP-9OP29, Power Block Clearance and Tagging, double valve isolation is REQUIRED for systems that are greater than a MINIMUM of ___(1)___ °F OR greater than a MINIMUM of ___(2)___ psig.

A. (1) 200
(2) 385

B. (1) 200
(2) 500

C. (1) 212
(2) 385

D. (1) 212
(2) 500

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible because greater than 385 psia double valve isolation of the RCS from the SDC system is required.
B.	Correct
C.	First part is plausible because 212°F is the temperature that water boils. Second part is plausible because greater than 385 psia double valve isolation of the RCS from the SDC system is required.
D.	First part is plausible because 212°F is the temperature that water boils. Second part is correct.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.41:	**10**	
Reference Provided:	**N**	
Learning Objective:	**27256 – Describe the special Clearance precautions used when establishing isolation boundaries on fluid or gas systems that operate at high temperatures or pressures**	

Technical Reference:	40DP-9OP29, Power Block Clearance and Tagging

Appendix H - Mechanical Tagging Practices

1.0 REQUIREMENTS FOR MECHANICAL ISOLATION BOUNDARIES

1.1 If using an isolation boundary to control process flow, then ensure a Danger tag is used.

1.2 Fluid or Gas Systems with temperature greater than 200° F or pressure greater than 500 psig should isolate the work area using double isolation boundary protection (two closed valves in series) and a tell-tale vent or drain valve between the two closed valves shall be opened.

Technical Reference:	40OP-9SI01, Shutdown Cooling Initiation

Shutdown Cooling Initiation	40OP-9SI01	57

3.2 Limitations

3.2.1 LCO 3.4.3, RCS Pressure and Temperature (P/T) Limits, Table 3.4.3-1, provides the heatup and cooldown rate limitations for the RCS.

3.2.2 The design allowable peak temperature for the Essential Cooling Water (EW) System is 135°F per calculation 13-MC-SP-307, is based on meeting the function of the Essential Chillers. The thermal analysis predicts that the allowable peak temperature for the Essential Cooling Water System may be exceeded if the RCS is cooled with SDC at the maximum possible rate. The RCS cooldown may be limited to maintain the EW System peak temperature less than 135°F.

3.2.3 Interlocks on the SDC containment isolation valves prevent them from being opened when pressurizer pressure is greater than 385 psia.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Equipment Control: Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator	Tier	3		
	Group			
	K/A	G 2.2.17		
	IR	2.6		

Question 70

Given the following conditions:

- All Units are operating at 100% power
- There is required maintenance on Westwing #1 transmission line
- An Auxiliary Operator needs to access the Switchyard to hang a clearance

Per 40DP-9OP34, Switchyard Administrative Controls:

(1) Auxiliary Operator access to the Switchyard may be granted by…

(2) In order to hang the clearance, a Switching Order is REQUIRED to be provided by…

A. (1) ANY of the Unit SMs
(2) the Energy Control Center (ECC) ONLY

B. (1) ANY of the Unit SMs
(2) the Energy Control Center (ECC) AND Salt River Project (SRP)

C. (1) the Unit 1 SM ONLY
(2) the Energy Control Center (ECC) ONLY

D. (1) the Unit 1 SM ONLY
(2) the Energy Control Center (ECC) AND Salt River Project (SRP)

Proposed Answer:	C

Explanations:	
A.	First part is plausible because each SM will oversee switching of their respective generator output breakers, including associated 525kV MOD and Start-up Transformer Secondary 13.8kV Disconnects. Second part is correct.
B.	First part is plausible because each SM will oversee switching of their respective generator output breakers, including associated 525kV MOD and Start-up Transformer Secondary 13.8kV Disconnects. Second part is plausible because SRP is involved in the coordination of switching orders, however, the ECC provides the procedures.
C.	Correct
D.	First part is correct. Second part is plausible because SRP is involved in the coordination of switching orders, however, the ECC provides the procedures

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	26287 – Identify the responsibilities of the Unit 1 Shift Manager concerning switchyard operations	

Technical Reference:	40DP-9OP34, Switchyard Administrative Controls

2.4 PVGS Unit 1 Shift Manager

2.4.1 Overall responsibility to act as the single point of contact. Primarily interfaces with ECC and PDO for emergency operations in the SRP 525kV Switchyard, Startup Transformer Yard, Main Transformers, Startup Transformers, and AEZYNX05, Power Expansion Plant West Transformer. The Switchyard Coordinator is the normal designee for non-emergency work planning and scheduling purposes.

2.4.2 Acts as coordinator when ECC requests major changes to total site megawatt/megavar loading.

2.4.3 Monitors access of personnel to the SRP 525kV Switchyard. This activity is normally administered through the Switchyard Coordinator who is responsible for notifying the Unit 1 Shift Manager or Control Room Supervisor of all personnel entering and leaving the Switchyard.

2.4.4 Controls access of personnel entering and leaving the Startup Transformer Yard.

Technical Reference:	40DP-9OP34, Switchyard Administrative Controls

2.6 APS Energy Control Center (ECC)

2.6.1 Functions as the primary interface between PVGS and PDO for operations in the SRP 525kV Switchyard. All communication, affecting the Switchyard, provided by PVGS shall be communicated to PDO.

2.6.2 Informs PVGS and PDO of all activities affecting the SRP 525kV Switchyard.

2.6.3 Initiates and coordinates System Blackstart Procedure for providing off site power to Palo Verde.

2.6.4 Notification of System Limits of Transmission and Generation on Path 49.

2.6.5 Provides coordination and clearance authorization between the following operations centers:

- PVGS Control Rooms Unit 1, 2, and 3
- ECC
- PDO

2.6.6 Provide written switching orders to PVGS for PVGS work.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Radiation Control: Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc	**Tier**	**3**		
	Group			
	K/A	**G 2.3.13**		
	IR	**3.4**		

Question 71

Per 40DP-9ZZ01, Containment Entry in Mode 1 Through Mode 4, prior to a Containment Entry in MODE 1, Containment must be purged if Containment Atmosphere…

(1) H_2 Concentration is GREATER than or equal to a MINIMUM of…

OR

(2) Containment Atmospheric O_2 Concentration is LESS than a MAXIMUM of…

A. (1) 0.04%
(2) 19.5%

B. (1) 0.04%
(2) 20.3%

C. (1) 0.15%
(2) 19.5%

D. (1) 0.15%
(2) 20.3%

Proposed Answer:	C

Explanations:	
A.	First part is plausible since if H2 concentration is > 0.15%, purge is required until H2 concentration is < 0.04%, however purge is not required if initial H2 concentration is < 0.15%. Second part is correct.
B.	First part is plausible since if H2 concentration is > 0.15%, purge is required until H2 concentration is < 0.04%, however purge is not required if initial H2 concentration is < 0.15%. Second part is plausible since if two consecutive O2 samples are less than 19.5%, purge is required until O2 concentration is > 20.3%, however purge is not required if initial O2 concentration is > 19.5%.
C.	Correct.
D.	First part is correct. Second part is plausible since if two consecutive O2 samples are less than 19.5%, purge is required until O2 concentration is > 20.3%, however purge is not required if initial O2 concentration is > 19.5%.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	9	
Reference Provided:	N	
Learning Objective:	27001 - Given that a confined space entry must be made, determine the necessary requirements per 01DP-01SI2	

Technical Reference:	40DP-9ZZ01, Containment Entry in Mode 1 Through Mode 4

6.2 Prior to Containment Entry

___ 6.2.1 Perform the following to check the Containment atmosphere conditions:

___ 6.2.1.1 Review the latest Containment Atmosphere H_2 Concentration data in Chemistry Laboratory Analysis Storage System (CLASS).

___ 6.2.1.2 IF CLASS is NOT available,
THEN request Chemistry to provide the latest Containment Atmosphere H_2 Concentration data.

___ 6.2.1.3 IF Containment Atmosphere H_2 Concentration is greater than or equal to 0.15%,
THEN perform the following:

___ a. Operate the Power Access Purge for a minimum of 12 hours per 40OP-9CP01, Containment Purge System.

___ b. Direct Chemistry to sample the Containment Atmosphere H_2 Concentration.

___ c. **WHEN** Containment Atmosphere H_2 Concentration is less than 0.04%,
THEN stop Power Access Purge per 40OP-9CP01, Containment Purge System.

___ 6.2.1.4 Request the Radiation Monitoring Technicians to check the most recent Containment Atmosphere O_2 Concentration is between 19.5% and 23.5%.

___ 6.2.1.5 IF two successive Containment Atmosphere O_2 Concentration samples are less than 19.5%,
THEN purge the Containment until the Containment Atmosphere O_2 Concentration is greater than or equal to 20.3% per 40OP-9CP01, Containment Purge System.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Radiation Control: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc	Tier	3		
	Group			
	K/A	G 2.3.12		
	IR	3.2		

Question 72

When working at heights in the RCA, fall protection is required for any work being performed above a MINIMUM height of ___(1)___ feet, and RP must be contacted to evaluate the need to perform a survey for any work being performed above a MINIMUM height of ___(2)___ feet.

A. (1) 4
(2) 6

B. (1) 4
(2) 7

C. (1) 6
(2) 6

D. (1) 6
(2) 7

Proposed Answer:	**B**

Explanations:	
A.	First part is correct. Second part is plausible because 6 feet was the old requirement and was recently changed to 7 feet.
B.	Correct
C.	First part is plausible since 6 feet was the minimum height which required fall protection until 2015, however the current minimum height requiring fall protection is 4 feet. Second part is plausible because 6 feet was the old requirement and was recently changed to 7 feet.
D.	First part is plausible since 6 feet was the minimum height which required fall protection until 2015, however the current minimum height requiring fall protection is 4 feet. Second part is correct.

Question Source:		**New**	
	X	**Bank**	
		Modified	
	X	**Previous NRC Exam**	**2016**

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

Level of Difficultly:	**2**	
10CFR55.41:	**12**	
Reference Provided:	**N**	
Learning Objective:	**228365 – Describe operations expectations when it comes to Safety in accordance with ODP-1, Operations Principles and Standards**	

Technical Reference:	75DP-0RP01, Radiological Posting and Labeling

4.23 Posting Overhead Areas Within an RCA

4.23.1 Overhead areas within a Radiologically Controlled Area may be posted with a white on blue background sign stating:

NOTIFY RP PRIOR TO
ANY WORK

ABOVE 7 FEET IN THE
OVERHEAD AREAS

Technical Reference:	01DP-01S20, Safety at Heights – Fall Protection

4.1 General

4.1.1 Any employee not protected by a Standard Rail System and subject to a fall greater than four feet shall be protected by either a Fall Restraint or a Personal Fall Arrest System (PFAS). Refer to Appendix A - Donning and Doffing a Full-Body Harness.

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions	**Tier**	**3**		
	Group			
	K/A	**G 2.3.4**		
	IR	**3.4**		

Question 73

Per 10CFR20.1201, Occupational Dose Limits, the annual limit for dose to the lens of the eye is ___(1)___ rem and to extremities is ___(2)___ rem.

A. (1) 12
(2) 40

B. (1) 12
(2) 50

C. (1) 15
(2) 40

D. (1) 15
(2) 50

Proposed Answer:	D

Explanations:	
A.	First part is plausible because 12 rem is an administrative dose limit. Second part is plausible because 40 rem is an administrative dose limit.
B.	First part is plausible because 12 rem is an administrative dose limit. Second part is correct.
C.	First part is correct. Second part is plausible because 40 rem is an administrative dose limit.
D.	Correct

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	2	
10CFR55.41:	12	
Reference Provided:	N	
Learning Objective:	State the Federal Dose Limits for lens of the eye and extremities	

Technical Reference:	75DP-9RP01, Radiation Exposure and Access Control

4.10 Radiation Exposure Limitations and Controls

4.10.1 10 CFR 20.1201 Occupational Dose Limits

A. Annual Occupational radiation dose to adults shall be limited to all of the following:

1. 5 rem total effective dose equivalent (TEDE) or 50 rem total organ dose equivalent (TODE), whichever is more limiting.

2. 15 rem lens dose equivalent (lens of the eye).

3. 50 rem shallow-dose equivalent (skin or any extremity).

Technical Reference:	75DP-9RP01, Radiation Exposure and Access Control

4.10.2 Common Industry Guidance for Occupational Dose Limits

A. Annual Occupational radiation dose to adults shall be limited to all of the following:

1. 2 rem total effective dose equivalent (TEDE) or 40 rem total organ dose equivalent (TODE), whichever is more limiting.

2. 12 rem lens dose equivalent (lens of the eye).

3. 40 rem shallow-dose equivalent (skin or any extremity).

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Emergency Procedures/Plan: Knowledge of the emergency plan	**Tier**	**3**		
	Group			
	K/A	**G 2.4.29**		
	IR	**3.1**		

Question 74

Given the following conditions:

- Unit 1 SM has just declared an ALERT for an event in progress

The STSC communicator duties are normally performed by a(n) ___(1)___ and offsite notifications are required to be made within a MAXIMUM of ___(2)___ minutes of the declaration.

A. (1) Reactor Operator
(2) 15

B. (1) Reactor Operator
(2) 30

C. (1) Auxiliary Operator
(2) 15

D. (1) Auxiliary Operator
(2) 30

Proposed Answer:	C

Explanations:	
A.	First part is plausible because the ENS Communicator is normally a Reactor Operator. Second part is coorect.
B.	First part is plausible because the ENS Communicator is normally a Reactor Operator. Second part is plausible because site Accountability is required within 30 minutes of a Site Area Emergency.
C.	Correct
D.	First part is correct. Second part is plausible because site Accountability is required within 30 minutes of a Site Area Emergency.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.41:	10	
Reference Provided:	N	
Learning Objective:	28129 – Identify actions to be taken as STSC Communicator	

Technical Reference:	40DP-9OP02, Conduct of Operations

2.9 Nuclear Auxiliary Operator

- Operates equipment outside the Control Room per approved procedures and as directed by Control Room personnel.
- Keeps Control Room personnel informed of activities outside the Control Room.
- Remains attentive to the operating condition of equipment within assigned area and initiates corrective action for deficiencies.
- Ensures proper turnover of information for assigned area when relieved.
- Serves as the Emergency Plan Satellite Technical Support Center (STSC) Communicator when directed.

Technical Reference:	

NOTE

- Notification duties and responsibilities transition from the Control Room to the EOF with transfer of Command and Control.
- If the EOF is not activated or is unable to accept Command and Control, the responsibility for required Notification of Offsite Agencies remain in the Satellite Technical Support Center (STSC).
- All of the required offsite state and local agencies must be notified within 15 minutes of determining any of the following criteria:
  - Initial Classification of the Emergency
  - Change in the Classification
  - Change in Protective Action Recommendations (PARs)
  - Change in Radiological Release Status
  - Event Termination

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Emergency Procedures/Plan: Knowledge of facility protection requirements, including fire brigade and portable firefighting equipment usage	**Tier**	**3**		
	Group			
	K/A	**G 2.4.26**		
	IR	**3.1**		

Question 75

Per 40DP-9OP02, Conduct of Operations, there will be one Fire Team Advisor (FTA) assigned to ___(1)___ and the LOWEST QUALIFICATION level he/she is required to be qualified is ___(2)___.

A. (1) each Unit
(2) Reactor Operator

B. (1) each Unit
(2) Auxiliary Operator

C. (1) the entire Site
(2) Reactor Operator

D. (1) the entire Site
(2) Auxiliary Operator

Proposed Answer:	**C**

Explanations:	
A.	First part is plausible because each unit will have their own STSC and ENS communicator during an emergency. Second part is correct.
B.	First part is plausible because each unit will have their own STSC and ENS communicator during an emergency. Second part is plausible because an Auxiliary Operator fills the role of STSC Communicator and it is reasonable to think that an Auxiliary Operator will be able to respond to fire faster.
C.	Correct
D.	First part is correct. Second part is plausible because an Auxiliary Operator fills the role of STSC Communicator and it is reasonable to think that an Auxiliary Operator will be able to respond to fire faster.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.41:	**10**	
Reference Provided:	**N**	
Learning Objective:	**445256 – Describe the Duties and Responsibilities of the Fire Team Advisor**	

Technical Reference:	40DP-9OP02, Conduct of Operations

4.8.1 Shift Composition

4.8.1.1 Normally during non-outage periods, operating crews will be manned on a five shift, self-relieving five crew basis. The minimum required shift crew manning is as follows:

- One Shift Manager per unit
- One Control Room Supervisor per unit
- Two licensed reactor operators per unit - In addition to the two reactor operators per unit, one unit shall have an additional reactor operator (who may be filling an Auxiliary Operator position) to fulfill the requirement of site Fire Team Advisor.

Technical Reference:	40DP-9OP02, Conduct of Operations

2.8 Reactor Operator

- Performs duties as the Fire Team Advisor as assigned.

qExamination Outline Cross-Reference:	Level	RO		SRO
K/A: Pressurizer Vapor Space Accident: Knowledge of the operational implications of EOP warnings, cautions, and notes	Tier		[redacted]	1
	Group			1
	K/A	008 G 2.4.20		
	IR		[redacted]	4.3

Question 76

(1) Per 40EP-9EO03, LOCA, one indication of voiding in the RCS occurs AS SOON AS RVLMS indicates a vessel level of less than…

(2) Per the EAL Hot Chart, a POTENTIAL LOSS of the Fuel Cladding Barrier occurs AS SOON AS RVLMS indicates a vessel level of less than…

A. (1) 16% in the RVUH
(2) 16% in the RVUH

B. (1) 16% in the RVUH
(2) 21% in the plenum

C. (1) 100% in the RVUH
(2) 16% in the RVUH

D. (1) 100% in the RVUH
(2) 21% in the plenum

Proposed Answer:	D

Explanations:	
A.	First part is plausible because greater than 16% in the RVUH means that the Inventory Control Safety Function is met and with the plenum full it could be assumed that there is no voiding. Second part is plausible because if there is not 16% in the RVUH then the Inventory Control Safety Function would net be met and there could potentially be voiding.
B.	First part is plausible because greater than 16% in the RVUH means that the Inventory Control Safety Function is met and with the plenum full it could be assumed that there is no voiding. Second part is correct.
C.	First part is correct. Second part is plausible because if there is not 16% in the RVUH then the Inventory Control Safety Function would net be met and there could potentially be voiding.
D.	Correct

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	4	
Reference Provided:	N	
Learning Objective:	297491 - Demonstrate RCS Void Control per Standard Appendices Appendix 15, RCS Void Control	

Technical Reference:	40EP-9EO03, Loss of Coolant Accident

------------------------------ **NOTE** ------------------------------

Voiding in the RCS may be indicated by **ANY** of the following:

- Letdown flow is greater than Charging flow
- Pressurizer level is rising significantly more than expected while operating pressurizer spray
- The RVLMS indicates less than 100% RVUH level
- HJTC unheated thermocouple temperature indicates saturated conditions in the RVUH

Technical Reference:	EAL Hot Chart

Fuel Clad (FC) Barrier	
Loss	**Potential Loss**
	1. RVLMS < 21% plenum (Detector #8)

Technical Reference:	SRO Level Question Criteria from NUREG-1021

D. Radiation Hazards That May Arise during Normal and Abnormal Situations, including Maintenance Activities and Various Contamination Conditions [10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include the following:

- process for gaseous/liquid release approvals (i.e., release permits)
- analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures
- analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Large Break LOCA: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits	**Tier**			**1**
	Group			**1**
	K/A	**011 G 2.2.25**		
	IR			**4.2**

Question 77

Per Technical Specifications, in order for Safety Injection Tanks to be OPERABLE, they must have a MINIMUM boron concentration of ___(1)___ ppm in order to ensure ___(2)___ in the event a LOCA.

A. (1) 2300
(2) the Reactor will remain subcritical following the injection of relatively colder SIT water volume into the RCS

B. (1) 2300
(2) back leakage from the RCS into the SITs during normal operations will not dilute the SITs to less than the minimum required boron concentration in the safety analysis

C. (1) 4000
(2) the Reactor will remain subcritical following the injection of relatively colder SIT water volume into the RCS

D. (1) 4000
(2) back leakage from the RCS into the SITs during normal operations will not dilute the SITs to less than the minimum required boron concentration in the safety analysis

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible as this is the basis for the minimum boron concentration in the RWT.
B.	Correct.
C.	First part is plausible since this is the minimum required boron concentration for the RWT. Second part is plausible as this is the basis for the minimum boron concentration in the RWT.
D.	First part is plausible since this is the minimum required boron concentration for the RWT. Second part is correct.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**2**	
Reference Provided:	**N**	
Learning Objective:	**21211 – Identify the basis of Technical Specifications LCOs and TLCOs for Section 3.5**	

Technical Reference:	Tech Specs LCO 3.5.1, SITs

SITs - Operating
3.5.1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.4	Verify boron concentration in each SIT is ≥ 2300 ppm and ≤ 4400 ppm.	In accordance with the Surveillance Frequency

Technical Reference:	Tech Specs LCO 3.5.5, RWT

RWT
3.5.5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	-----NOTE----- Only required to be performed when ambient air temperature is < 60°F or > 120°F. ----- Verify RWT borated water temperature is ≥ 60°F and ≤ 120°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.5.2	Verify RWT borated water volume is ≥ minimum required RWT volume in Figure 3.5.5-1.	In accordance with the Surveillance Frequency Control Program
SR 3.5.5.3	Verify RWT boron concentration is ≥ 4000 ppm and ≤ 4400 ppm.	In accordance with the Surveillance Frequency

Technical Reference:	Tech Spec Bases for LCO 3.5.1, SITs

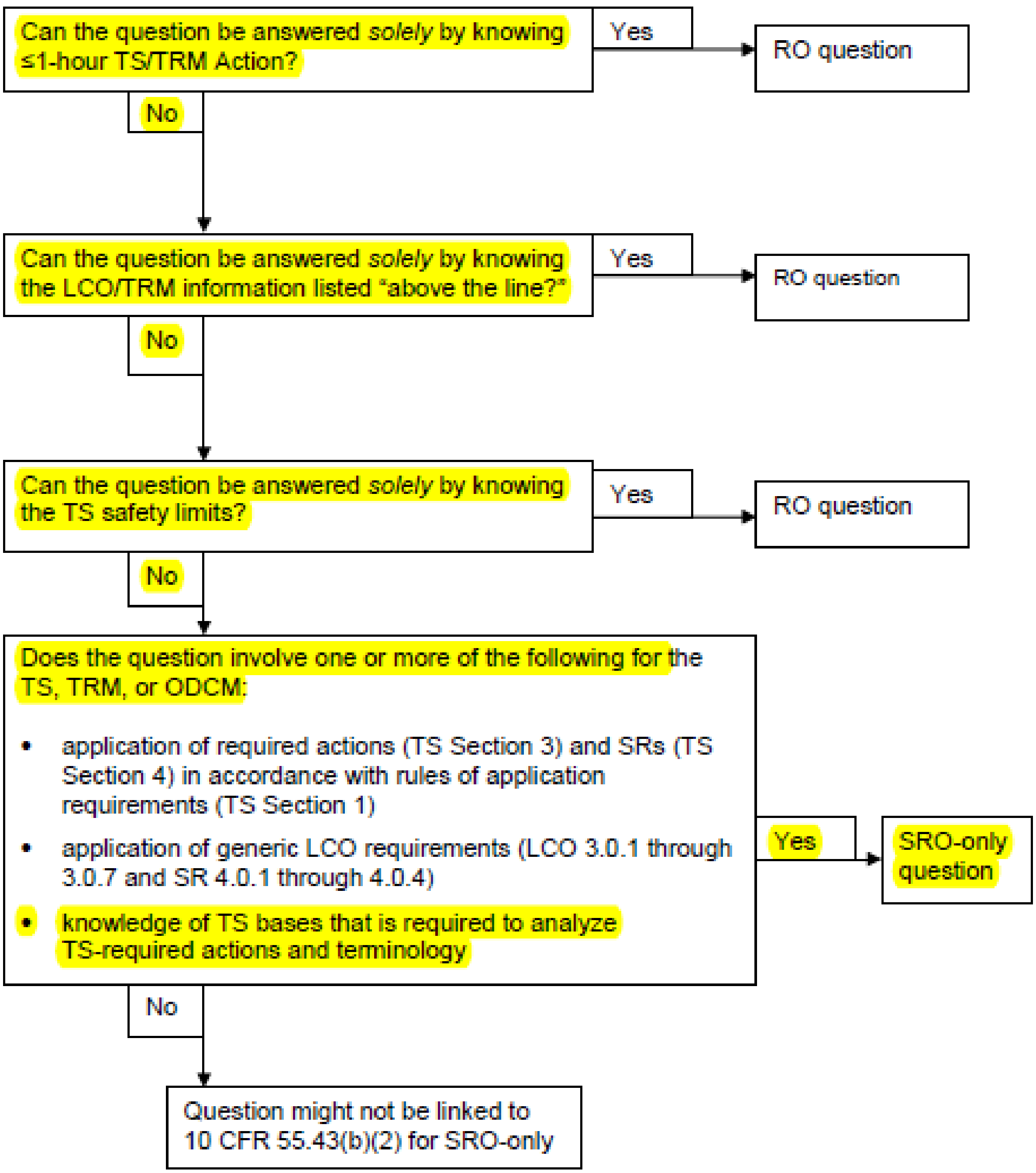
The 2300 ppm minimum boron concentration in the SITs assures that the back leakage from the RCS will not dilute the SITs below the minimum boron concentration in the safety analysis. The minimum safety analysis boron requirements of 2000 ppm are based on beginning of life reactivity values and are selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. During a large break LOCA, all Control Element Assemblies (CEAs) are assumed not to insert into the core, and the initial reactor shutdown is accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the SITs to prevent a return to criticality during reflood. Although this requirement is similar to the basis for the minimum boron concentration of the Refueling Water Tank (RWT), the minimum SIT concentration is lower than that of the RWT since the SITs need not account for dilution by the RCS during a large break LOCA.

Technical Reference:	Tech Spec Bases for LCO 3.5.5, RWT

The 4000 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum level in the RWT, the reactor will remain subcritical in the cold condition following mixing of the RWT and RCS water volumes. Small break LOCAs assume that all control rods are inserted, except for the Control Element Assembly (CEA) of highest worth, which is withdrawn from the core. Large break LOCAs assume that all CEAs remain withdrawn from the core. The most limiting case occurs at beginning of core life.

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Component Cooling Water: The normal values and upper limits for the temperatures of the components cooled by CCW	**Tier**			**1**
	Group			**1**
	K/A	**026 AA2.04**		
	IR			**2.9**

Question 78

Given the following conditions:

- Unit 2 is operating at 100% power

Subsequently:

- A loss of all Nuclear Cooling Water occurs
- The CRS enters 40AO-9ZZ03, Loss of Cooling Water

The crew should cross-tie Train ‘A’ EW to NC to prevent RCP HP Seal Cooler inlet temperature to prevent exceeding the procedural driven RCP trip setpoint of ___(1)___ °F. After the cross-tie is complete, Train ‘A’ EW is considered ___(2)___ per Technical Specifications.

A. (1) 250
(2) OPERABLE

B. (1) 250
(2) INOPERABLE

C. (1) 300
(2) OPERABLE

D. (1) 300
(2) INOPERABLE

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible because there are no components or valves that are out of service, however, since a manual valve has been throttled in the closed direction for the SDCHX and the cross connect valves are out of position, the EW System is INOPERABLE.
B.	Correct
C.	First part is plausible because to maintain LPSI seal life, SDC may not be placed in service until RCS temperature is less than 300°F. Second part is plausible because there are no components or valves that are out of service, however, since a manual valve has been throttled in the closed direction for the SDCHX and the cross connect valves are out of position, the EW System is INOPERABLE.
D.	First part is plausible because to maintain LPSI seal life, SDC may not be placed in service until RCS temperature is less than 300°F. Second part is correct.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**2**	
Reference Provided:	**N**	
Learning Objective:	**22357 - Given the status of NC and RCP seal injection, describe the limitations on RCP operation without NC in accordance with 40AO-9ZZ03**	

Technical Reference:	40AO-9ZZ04, Reactor Coolant Pump Emergencies

Appendix D, Instrumentation and Setpoints

Parameter	Instrument Number	Normal	Alarm	Trip
No. 2 Seal Inlet Pressure	RCN-PT-152/162 (RCN-PI-152 on B04) RCN-PT-172/182 (RCN-PI-172 on B04)	See Appendix G	Lo 826 psig	-
			Hi 1766 psig	-
No. 2 Seal Outlet Pressure (Controlled Bleedoff)	RCN-PT-153/163 (RCN-PI-153 on B04) RCN-T-173/183) (RCN-PI-173 on B04)	See Appendix G	Lo 179 psig	-
			Hi 537 psig	-
Controlled Bleedoff Flow	RCN-FI-156/166/176/186 (B03)	2.0 - 4.0 gpm	Lo 1.6 gpm	-
			Hi 6.0 gpm	Hi ≥9.5 gpm
NCW RCP Temperature	NCN-TI-471/470/473/472 (B04)	<120°F	130°F	-
H.P. Cooler Inlet Temperature	RCN-TT-150/160 (RCN-TI-150 on B04) RCN-TT-170/180) (RCN-TI-170 on B04)	170 - 212°F	221°F	≥250°F
H.P. Cooler Outlet Temperature	RCN-TT-151/161 (RCN-TI-151 on B04) RCN-TT-171/181 (RCN-TI-171 on B04)	80 - 160°F	175°F	≥200°F

Technical Reference:	40OP-9SI01, Shutdown Cooling Initiation

3.2.12 To extend LPSI Pump seal life, SDC may not be initiated when the RCS temperature is greater than or equal to 300°F.

Technical Reference:	40AO-9ZZ03, Loss of Cooling Water, Appendix A, Cross-connect EW to NC

PALO VERDE NUCLEAR GENERATING STATION LOSS OF COOLING WATER	40AO-9ZZ03 Revision 13 Page 26 of 44
	Appendix A Page 1 of 13

Appendix A, Cross-connect EW to NC

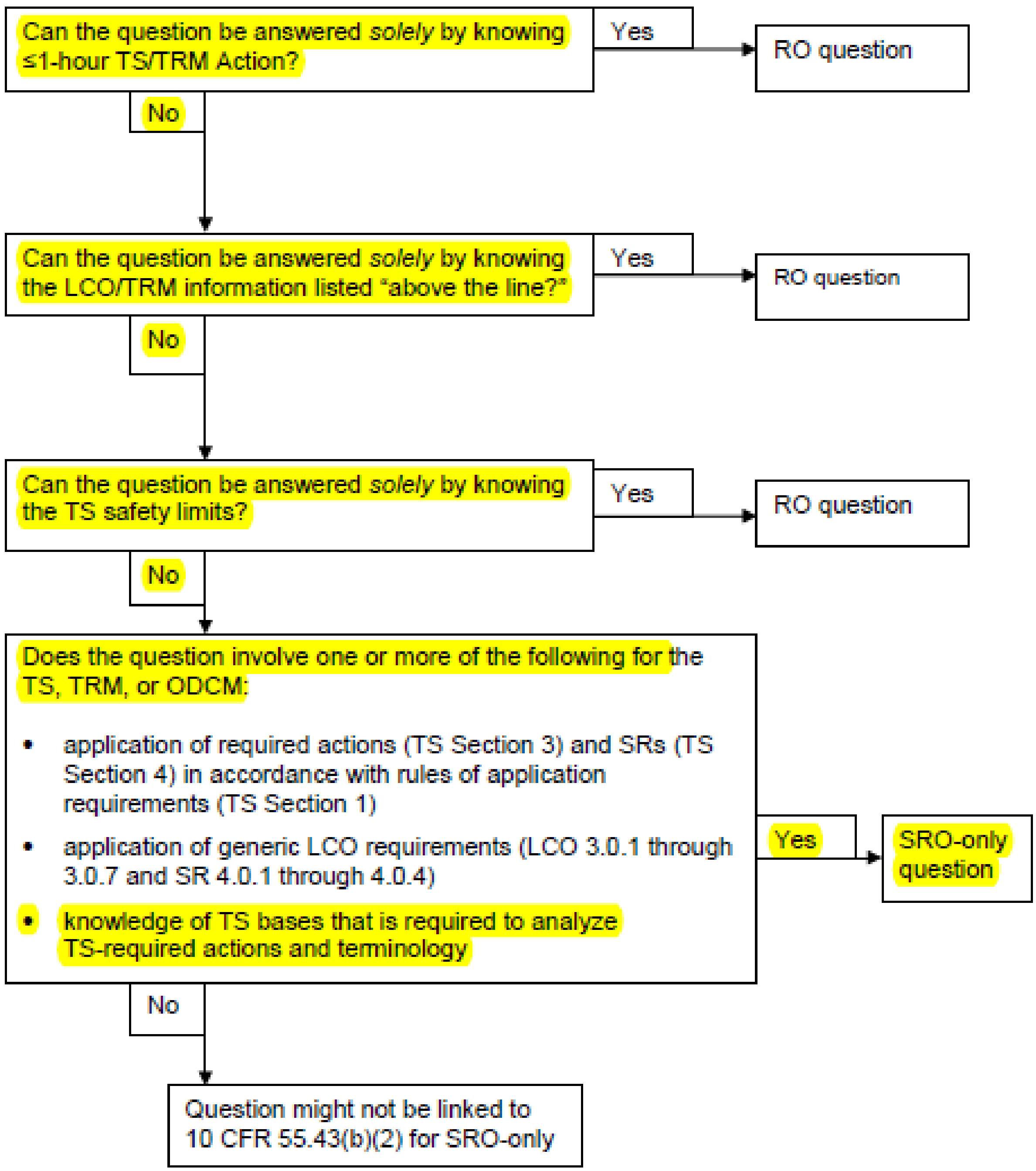
INSTRUCTIONS	CONTINGENCY ACTIONS
____ 1. Enter Appendix Entry Time and Date: ________________	

-- **NOTE** --

Cross-connecting EW and NC renders the EW Train inoperable per LCO 3.7.7, Essential Cooling Water (EW) System. Refer to 40ST-9EC03 and 40DP-9OP37 for impacts on supported system operability.

Technical Reference:	**SRO Level flowchart from NUREG-1021**

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Main Feedwater: Knowledge of limiting conditions for operations and safety limits	**Tier**			**1**
	Group			**1**
	K/A	**054 G 2.2.22**		
	IR			**4.7**

Question 79

Per Technical Specification Basis for LCO 3.3.1, RPS Instrumentation - Operating, which of the following RPS trips mitigates a Feedwater Line Break?

1. Departure from Nucleate Boiling Low
2. Containment Pressure High
3. Pressurizer Pressure High

A. 2 ONLY

B. 3 ONLY

C. 1 AND 2 ONLY

D. 1 AND 3 ONLY

Proposed Answer:	B

Explanations:	
A.	Plausible because a Feedwater line break is high energy release into containment. However only a LOCA and ESD are mitigated by a High Containment pressure trip per Technical Specification Bases
B.	Correct
C.	First part is plausible because a Feedwater line break is a heatup event (loss of Feedwater will cause a diminished heat sink and RCS temperature will rise), therefore DNBR will lower. Second part is plausible because a Feedwater line break is high energy release into containment. However only a LOCA and ESD are mitigated by a High Containment pressure trip per Technical Specification Bases.
D.	First part is plausible because a Feedwater line break is a heatup event (loss of Feedwater will cause a diminished heat sink and RCS temperature will rise), therefore DNBR will lower. Second part is correct

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**2**	
Reference Provided:	**N**	
Learning Objective:	**22620 – Identify the basis of Technical Specification LCOs and TLCOs for section 3.3 in accordance with Tech Spec 3.3 basis**	

Technical Reference:	Technical Specifications Basis

3. Pressurizer Pressure - High

The Pressurizer Pressure - High trip provides protection for the high RCS pressure SL. In conjunction with the pressurizer safety valves and the main steam safety valves (MSSVs), it provides protection against overpressurization of the RCPB during the following events:

- Loss of Condenser Vacuum (AOO);
- CEA Withdrawal From Low Power Conditions (AOO);
- Chemical and Volume Control System Malfunction (AOO); and
- Main Feedwater System Pipe Break (Accident).

Technical Reference:	Technical Specifications Basis

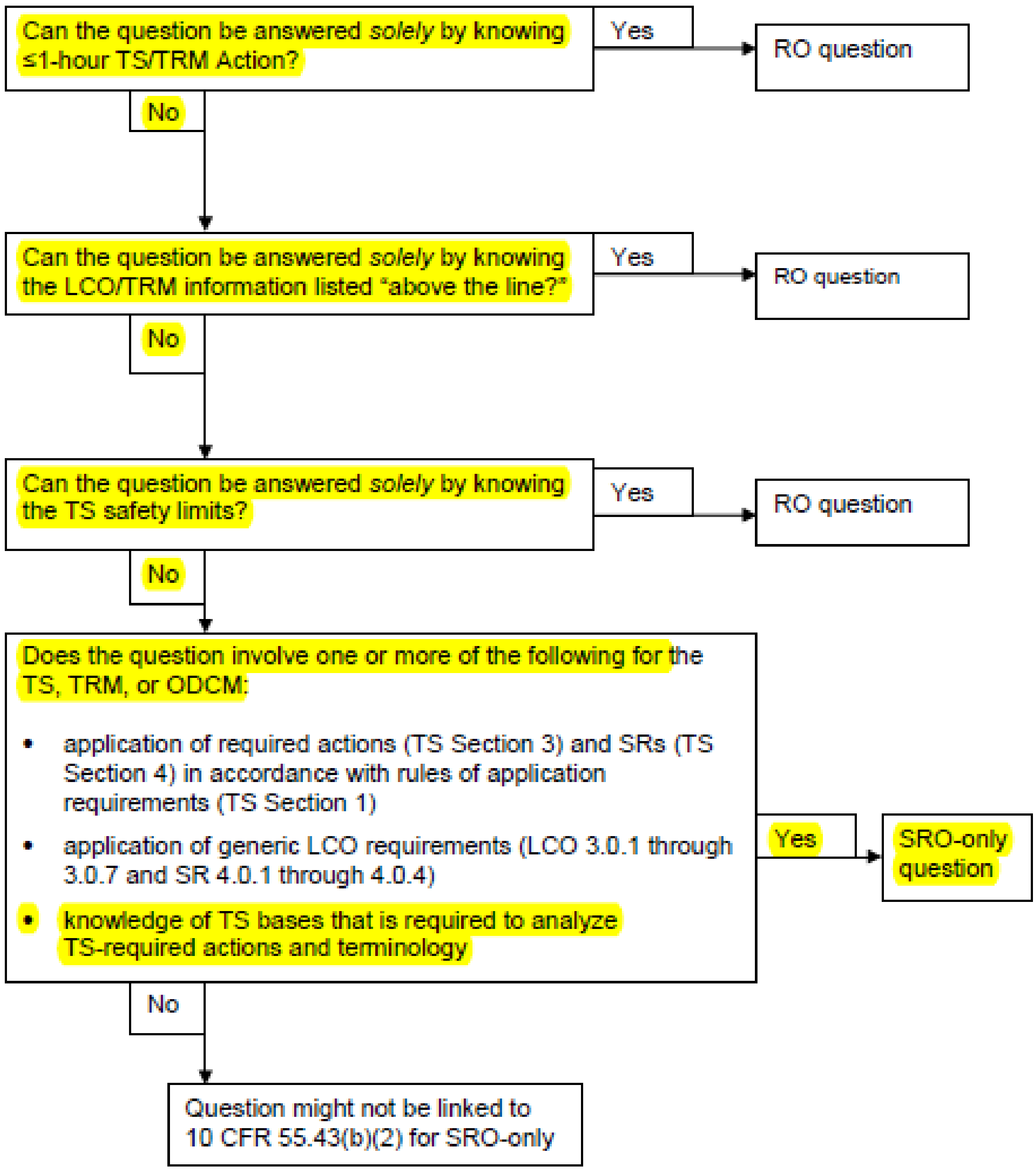
8, 9. Steam Generator Level - Low

The Steam Generator #1 Level - Low and Steam Generator #2 Level - Low trips ensure that a reactor trip signal is generated for the following events to help prevent exceeding the design pressure of the RCS due to the loss of the heat sink:

- Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (AOO);
- Loss of Condenser Vacuum (AOO);
- Loss of Normal Feedwater Event (AOO);
- Feedwater System Pipe Break (Accident); and
- Single RCP Rotor Seizure (AOO)

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Station Blackout: Ability to determine or interpret the following as they apply to a Station Blackout: Faults and lockouts that must be cleared prior to re-energizing buses	**Tier**		[redacted]	**1**
	Group		[redacted]	**1**
	K/A	**055 EA2.06**		
	IR		[redacted]	**4.1**

Question 80

Given the following conditions:

- Unit 1 is in a blackout condition
- The ‘B’ EDG is OOS and will not be available for the next 6 hours

Per Appendix 55, Restore DG A to PBA-S03, which of the following faults can the crew attempt to reset in order to restore power to PBA-S03?

1. Overspeed trip of the ‘A’ EDG
2. Generator Differential trip of the ‘A’ EDG
3. Overcurrent trip of the ‘A’ EDG Output Breaker

A. 2 ONLY

B. 3 ONLY

C. 1 AND 2 ONLY

D. 1 AND 3 ONLY

Proposed Answer:	C

Explanations:	
A.	Generator Differential is correct, however Overspeed trip is also correct.
B.	Plausible because overcurrent of the EDG output breaker can be reset to re-energize the bus, however it cannot be reset using Appendix 55. There is a separate Relay Resetting procedure that would be used.
C.	Correct
D.	Overspeed trip is correct. Overcurrent of the EDG output can be reset to re-energize the bus, however it cannot be reset using Appendix 55. There is a separate Relay Resetting procedure that would be used.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	5	
Reference Provided:	N	
Learning Objective:	26234 - Given conditions of a Blackout and plant status, determine the flowpath of power available to energize a vital bus in accordance with 40EP-9EO08	

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 2. **IF** the annunciator panel indicates "OVERSPEED ENGINE" (DGA-07A), **THEN** perform the following:	
a. Press "STOP" on DGA-HS-29, "DG "A" EMERGENCY STOP".	
b. Press the plunger on DGA-UV-237, "DG "A" OVERSPEED FUEL TRIP SOLENOID VALVE". (generator platform Southeast of the engine)	
c. Reset the intake air butterfly valve, ensuring that its handle is latched. REFER TO Attachment 55-A, Resetting Intake Air Butterfly Valve.	c.1 Request additional manpower to assist in resetting the air butterfly valve.
d. Inform the responsible operator that the Diesel Generator may start.	
e. Press "RESET" on DGA-HS-29, "DG "A" EMERGENCY STOP" to start the Diesel Generator.	
f. **IF** the Diesel Generator has started, **THEN** GO TO Step 19.	
____ 3. **IF** the annunciator panel indicates "GENERATOR DIFFERENTIAL" (DGA-05B), **THEN** inform the CRS of the alarm.	

Technical Reference:	40EP-9EO10-055, Appendix 55: Restore DG A to PBA-S03

INSTRUCTIONS	CONTINGENCY ACTIONS

____ 4. **IF** the CRS directs resetting the Generator Differential trip, **THEN** perform the following:

a. Press "STOP" on DGA-HS-29, "DG "A" EMERGENCY STOP".

b. Reset "GENERATOR DIFFERENTIAL LOCKOUT RELAY 86D".

c. Check that the "LOCKOUT RELAY RESET" white light is on.

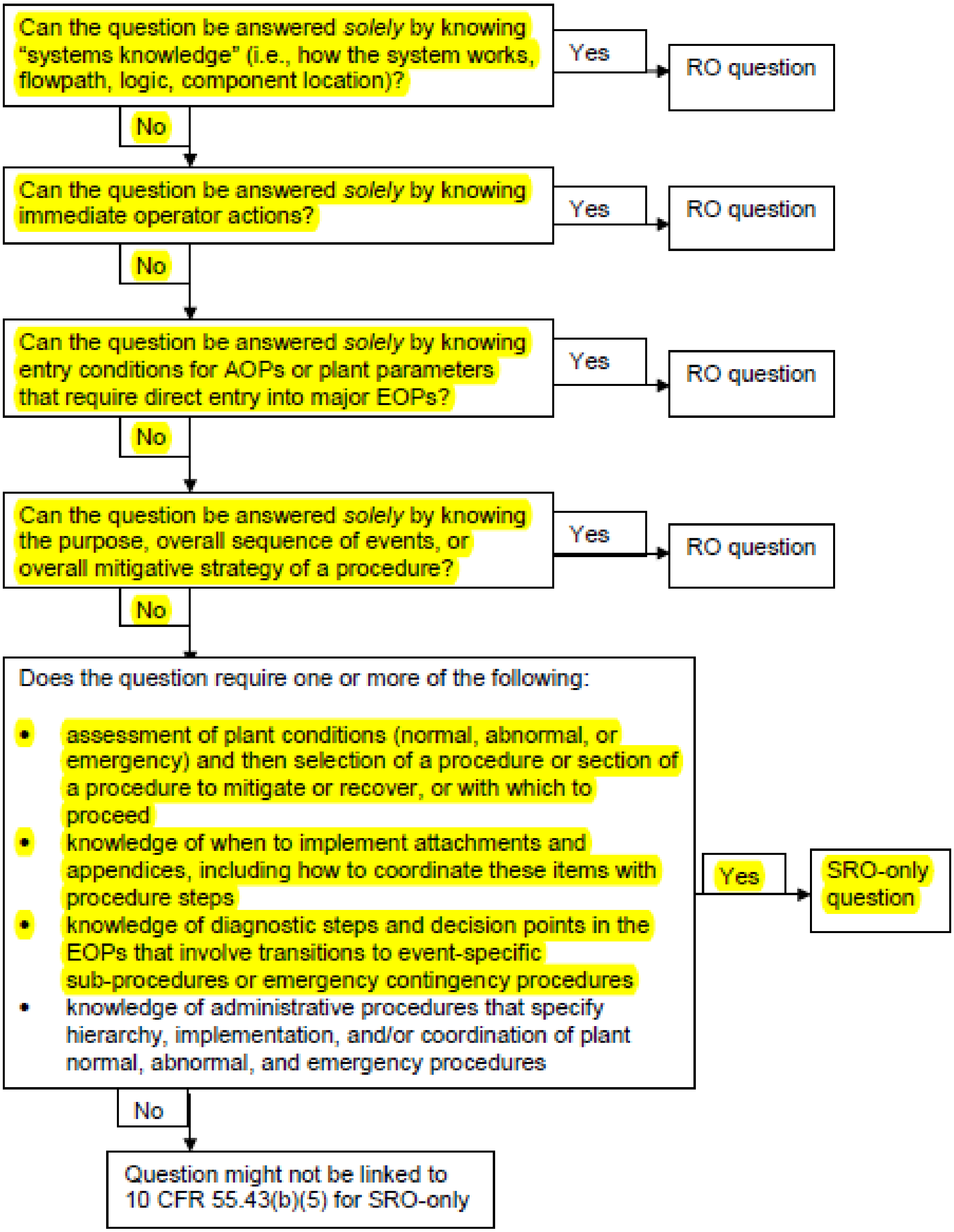
d. Inform the responsible operator that the Diesel Generator may start.

e. Press "RESET" on DGA-HS-29, "DG "A" EMERGENCY STOP" to start the diesel generator.

f. **IF** the Diesel Generator has started, **THEN** GO TO Step 19.

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Offsite Power: Ability to determine and interpret the following as they apply to the Loss of Offsite Power: EDG indicators for the following: voltage, frequency, load, load-status, and closure of the bus-tie breakers	Tier			1
	Group			1
	K/A	056 AA2.37		
	IR			3.8

Question 81

Given the following conditions:

- Unit 1 was manually tripped in preparation for a refueling outage
- SPTAs are complete and the CRS has transitioned to 40EP-9EO02, Reactor Trip

Subsequently:

- A LOOP occurred
- On the LOOP the following occurred:
    - ‘A’ EDG tripped on overspeed
    - NNN-D12 tripped on a fault
- The BOP reports that there is no frequency indication on ‘B’ EDG

‘B’ EDG frequency should be determined ___(1)___ and entry into 40EP-9EO09, Functional Recovery is ___(2)___.

A. (1) locally
(2) REQUIRED

B. (1) locally
(2) NOT required

C. (1) in the Control Room
(2) REQUIRED

D. (1) in the Control Room
(2) NOT required

Proposed Answer:	**D**

Explanations:	
A.	First part is plausible because speed is indicated locally and there is no direction in SPTAs to energize the synchroscope if frequency indication is lost. Second part is correct.
B.	First part is plausible because speed is indicated locally and there is no direction in SPTAs to energize the synchroscope if frequency indication is lost. Second part is plausible if it is thought that because there are multiple events with the LOOP and loss of NNN-D11, the Functional Recovery procedure is required to recover. However the MVAC Safety is met in the LOOP ORP.
C.	First part is correct. Second part is plausible if it is thought that because there are multiple events with the LOOP and loss of NNN-D11, the Functional Recovery procedure is required to recover. However the MVAC Safety is met in the LOOP ORP.
D.	Correct

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**5**	
Reference Provided:	**N**	
Learning Objective:	**23195 – Analyze MVA to determine if the SFSC acceptance citeria is satisfied**	

4.5.3 Step 3 - Maintenance of Vital Auxiliaries

A. The Maintenance of Vital Auxiliaries safety function ensures that vital electrical loads have power available. Electrical power is essential to the fulfillment of succeeding safety functions.

A critical task associated with this step is to energize at least one vital AC bus.

The acceptance criteria reflect automatic disconnect of the Main Generator and the transfer of power to offsite that should occur immediately upon a trip.

- Checks the automatic disconnect of the main generator from the offsite grid.
- Checks that the automatic transfer of electrical power to the offsite distribution system occurred.

B. Contingency Actions

Contingency actions are chosen to remedy the failure of automatic system responses and to ensure that the emergency diesel generators are available to supply AC power if necessary.

- Manually opening the Main Generator Output Breakers is a quick action taken from the Control Room which should give the same result as the automatic action.
- Diesel generators supplying the associated safety buses will help ensure continued fulfillment of succeeding safety functions. If a loss of power to either vital bus occurs without the associated Diesel Generator automatically starting and loading, operator action will be needed to start the DG and close its output breaker. The frequency meters for the diesel generator(s) on B01 are powered from NNN-D11 (Train A) and NNN-D12 (Train B). If NNN-D11 or NNN-D12 is de-energized, the respective frequency meter will not be available. Frequency indication for the diesel generator(s) can be obtained from either of the following:

  - Energizing the associated sync switch allowing MAN-SI-002I, Incoming (upper right hand side of B01) to be used to determine the diesel generator frequency

  - ERFDADS Points PES01 (A Train) or PES02 (B Train)

Technical Reference:	40EP-9EO07, Loss of Offsite Power / Loss of Forced Circulation

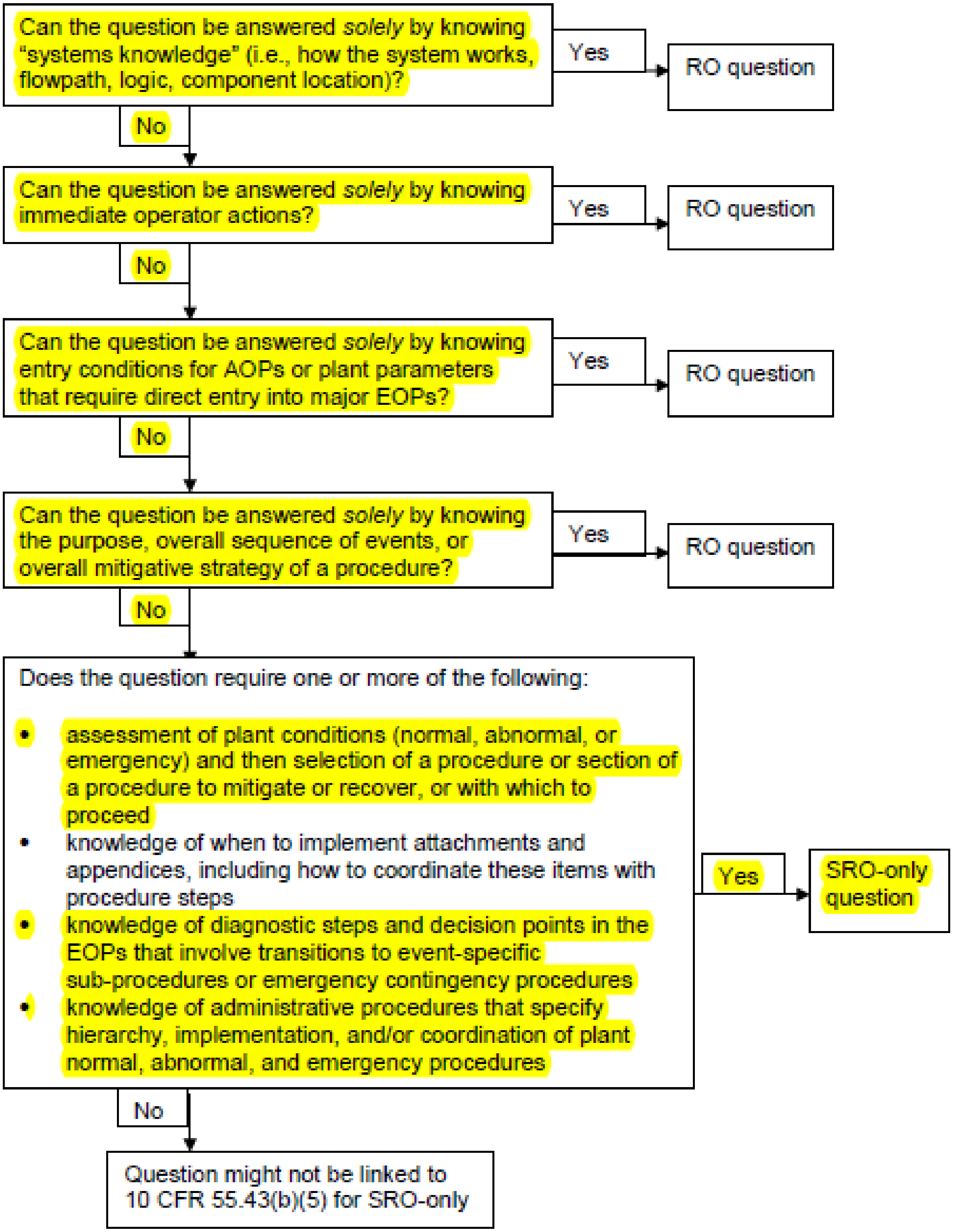
SAFETY FUNCTION:

2. Maintenance of Vital Auxiliaries

ACCEPTANCE CRITERIA:	CRITERIA SATISFIED
a. At least one vital 4.16 kV AC bus energized.	☐ ☐ ☐ ☐
b. At least one of the following trains of PK and PN is available and is on the same train as the powered vital 4.16 kV AC bus. • PKA-M41, PKC-M43, and PNA-D25 • PKB-M42, PKD-M44, and PNB-D26	☐ ☐ ☐ ☐
c. No jeopardized safety functions require restoration of electrical power to a vital AC or DC bus.	☐ ☐ ☐ ☐
d. Diesel Fuel Oil Transfer Pump is maintaining Day Tank level for at least one Diesel Generator supplying the powered vital 4.16 kV AC bus.	☐ ☐ ☐ ☐

Technical Reference:	**SRO Level flowchart from NUREG-1021**

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Dropped Control Rod: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures	**Tier**			**1**
	Group			**2**
	K/A	**003 G 2.4.4**		
	IR			**4.7**

Question 82

Given the following initial conditions:

- Unit 2 is operating at 100% power
- ‘B’ Boric Acid Makeup pump is OOS

Subsequently:

- A Seismic event occurs
- A 4-finger CEA dropped to the bottom of the core
- Reactor power stabilized at 97% following the dropped CEA
- The CRS entered 40AO-9ZZ11, CEA Malfunctions
- An Auxiliary Operator reports that the ‘A’ Boric Acid Makeup pump is severely damaged and CANNOT be used

The CRS should direct the crew to lower Reactor power to a MAXIMUM of ___(1)___ within the first hour from the CEA drop and direct the crew to borate the RCS using ___(2)___ .

A. (1) 77%
   (2) 40AO-9ZZ11, CEA Malfunctions, Appendix J, Boration for Power Reduction

B. (1) 77%
   (2) 40AO-9ZZ01, Emergency Boration

C. (1) 80%
   (2) 40AO-9ZZ11, CEA Malfunctions, Appendix J, Boration for Power Reduction

D. (1) 80%
   (2) 40AO-9ZZ01, Emergency Boration

Proposed Answer:	D

Explanations:	
A.	First part is plausible because it can be assumed that the downpower of 20% is from the power level after the CEA drops. Second part is plausible because if a BAMP is available then 40AO-9ZZ11, CEA Malfunctions, Appendix J can be used.
B.	First part is plausible because it can be assumed that the downpower of 20% is from the power level after the CEA drops. Second part is correct.
C.	First part is correct. Second part is plausible because if a BAMP is available then 40AO-9ZZ11, CEA Malfunctions, Appendix J can be used.
D.	Correct

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	5	
Reference Provided:	N	
Learning Objective:	25223 – Describe what action is taken to commence a Tech Spec required power reduction due to a dropped/slipped CEA	

Technical Reference:	40AO-9ZZ11, CEA Malfunctions

3.0 DROPPED OR SLIPPED CEA MODE 1 OR 2

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

The effects of a boration to the RCS may take 4 to 6 minutes to be seen, therefore initiating a boration (Step 16) should be done as soon as possible.

____ 13. Perform the following to start a power reduction **within 10 minutes** of the initial CEA deviation:

a. Log the start time for power reduction:

__________ (time)

b. Lower the turbine load to raise Tave 3°F greater than Tref.

____ 14. Determine the required power reduction based on initial power from ONE of the following:

- Greater than 80% - requires a 20% power reduction

Technical Reference:	40AO-9ZZ11, CEA Malfunctions, Appendix J, Boration for Power Reduction

Appendix J, Boration for Power Reduction

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 1. Set the boric acid makeup flow rate on CHN-FIC-210Y, Boric Acid Makeup to VCT Flow Control, to between 35 and 40 gpm.	
____ 2. Set the "Target" makeup volume (gallons) on CHN-FQIS-210Y, Boric Acid Makeup Totalized Flow Control, to the amount determined by the SM/CRS.	
____ 3. Place CHN-HS-210, Makeup Mode Select Switch, in "BORATE".	
____ 4. Check one Boric Acid Makeup Pump is running.	

Technical Reference:	40AO-9ZZ11, CEA Malfunctions

3.0 DROPPED OR SLIPPED CEA MODE 1 OR 2

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 15. Calculate the number of gallons of boric acid needed (STA reactivity worksheet) for the downpower: ____% X ____ gal/% = _____ gal (Step 14)	
____ 16. PERFORM Appendix J, Boration for Power Reduction to commence borating to the charging pump suction using BOTH of the following criteria: • Minimum rate of 35 gpm • Amount determined in Step 15	____16.1 PERFORM 40AO-9ZZ01, Emergency Boration, using ONE of the following boration flowpaths until the desired power reduction has been achieved: • CHN-UV-514 • CHE-HV-536

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing immediate operator actions?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?
- Yes → RO question
- No ↓

Does the question require one or more of the following:

- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

- Yes → SRO-only question
- No → Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Pressurizer Level Control Malfunction: Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunction: Letdown flow indicator	**Tier**		[redacted]	**1**
	Group		[redacted]	**2**
	K/A	**028 AA2.06**		
	IR		[redacted]	**2.8**

Question 83

Given the following conditions:

- Unit 3 is operating at 100% power
- A PLCS malfunction has caused a sudden increase in letdown flow
- The increase in letdown flow caused PSV-354, Low Pressure Letdown Relief Valve, to come off its closed seat
- ERFDADS indicates the leak rate through PSV-354 is 5 gpm and stable

PSV-354 should relieve to the ___(1)___ and per LCO 3.4.14 RCS Operational Leakage, this ___(2)___ considered RCS Leakage.

A. (1) EDT
(2) IS

B. (1) EDT
(2) is NOT

C. (1) RDT
(2) IS

D. (1) RDT
(2) is NOT

Proposed Answer:	**B**
Explanations:	
A.	First part is correct. Second part is plausible because the leakage does register as RCS leakage on ERFDADS and inventory is actually being lost from the RCS however intersystem leakage.
B.	Correct
C.	First part is plausible because the RDT is also connected to the CVCS system and collects various leakages. Second part is plausible because the leakage does register as RCS leakage on ERFDADS and inventory is actually being lost from the RCS.
D.	First part is plausible because the RDT is also connected to the CVCS system and collects various leakages. Second part is correct.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**2**	
Reference Provided:	**N**	
Learning Objective:	**22681 – Given conditions when an LCO is not met, apply Tech Spec Section 3.4.14 (RCS Operational Leakage) in accordance with Tech Spec 3.4.14**	

Technical Reference:	Chemical Volume Control System Tech Manual

2.1.12 Low Pressure Letdown Relief Valve (PSV-354)

The purpose of this relief valve is to protect the boronometer from over pressure.

The valve is located downstream of the BPCVs. It opens at 200 psig and relieves to the EDT.

Technical Reference:	LOIT Excessive RCS Leakrate Lesson Plan

EO: 1.2 Given a description of an RCS "leak", state whether or not this is considered RCS leakage in accordance with 40AO-9ZZ02 and Tech. Specs.

Introduction

Several years ago, a CRDR (9-5-0232) was written to define RCS leakage. This CRDR was generated after an event that occurred on February 17, 1995 when a Unit 3 Charging Pump (CHB-P01) discharge relief valve stuck open during the post maintenance run of the pump. The stuck open relief valve caused the transfer of VCT inventory to the EDT even after the pump was shutdown.

The relief valve failure raised the question of whether the flow from the VCT to the EDT should be considered RCS IDENTIFIED LEAKAGE. Considering that the flow rate was calculated to be greater than 25 gpm, an RCS IDENTIFIED LEAKAGE of this magnitude would require:

1. Entering LCO 3.4.14 Condition A for RCS leakage > 10 gpm IDENTIFIED LEAKAGE from the RCS.
2. Declaring an NUE per EP-0901 for RCS identified leakage > 25 gpm for ≥ 15 min OR Reactor coolant leakage to a location outside containment > 25 gpm for ≥ 15 min in Modes 1-4 [SU5.1].

Since neither of these two actions occurred, a meeting was held to discuss the issue. The outcome of the meeting was how PVNGS defines RCS LEAKAGE.

Main Idea

The following describe how PVNGS defines RCS LEAKAGE:

- There are four classifications of RCS leak.
- **Identified** – <u>Captured and conducted to a sump or tank</u> (anything that is routed to the Reactor Drain Tank is defined as identified). Any leak that has been <u>specifically located and then captured and conducted</u> to a monitored sump or tank (leak located, drip bag hung and hose routed to one of the containment sumps). <u>Steam Generator tube or tube sheet leaks</u> (very low threshold for required action per Tech Specs). Furthermore, if the leak interferes with leakage detection systems, it will be classified as Unidentified (more conservative).
- **Unidentified** – <u>Leakage into the containment atmosphere</u> that does not meet the definitions listed as **Identified**. Unidentified leaks are determined by containment atmosphere radiation monitors, containment radwaste sump level changes or water inventory balance calculations or combinations of those indications.

Technical Reference:	LOIT Excessive RCS Leakrate Lesson Plan

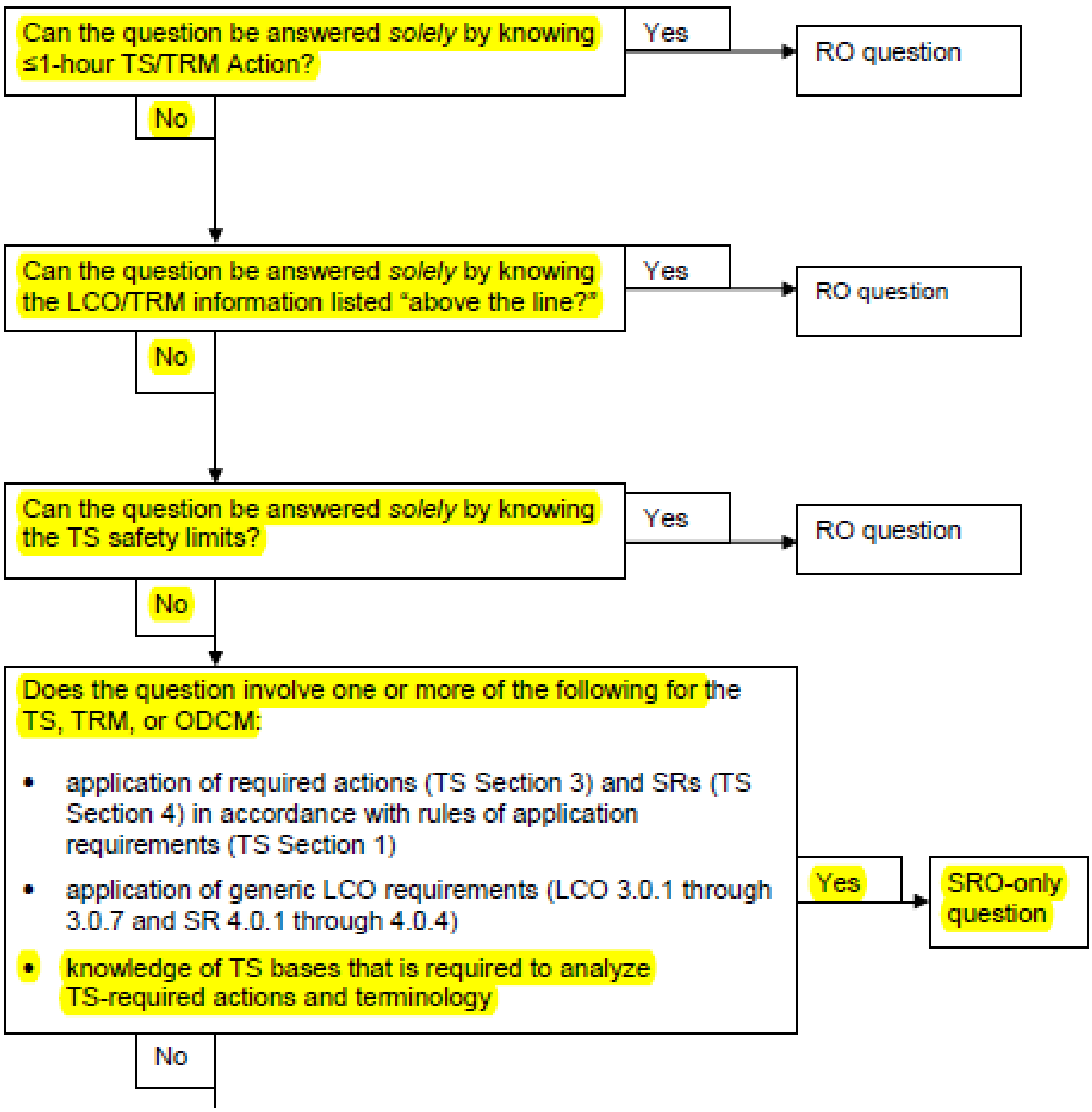
Title: Excessive RCS Leakrate **Lesson Plan #:** NKASMC030208

-
- **Reactor Coolant Pressure Boundary** – Leakage through a nonisolable fault in an RCS component body, pipe wall, vessel wall or weld with the exception of primary to secondary leakage. For the purpose of leak detection, Regulatory Guide 1.45 and PVNGS Technical Specifications limit the definition of the "reactor coolant pressure boundary" to those portions of the reactor coolant system which are constructed so that no leakage is expected to occur. UFSAR 5.2.5
  Example – A small RCS leak develops with the reactor at full power. The leak is sufficient quantity to be detected by RU-1, Containment Atmosphere Radiation Monitor. A Water Inventory Balance determines the leak to be 0.1 gpm. Initially, this leak will be classified as Unidentified. Visual inspection reveals that an RCS hot leg RTD thermowell is leaking through a flaw in the weld. The leak will subsequently be classified as Pressure Boundary leakage.

- **Intersystem Leakage** –RCS leaking into CVCS, SDC, SI, NC, sampling, etc. Although this type of leak may represent a loss of RCS inventory, it is not defined by Tech Specs. However, several procedures require operators to address and take action to mitigate this type of inventory loss.

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Loss of Source Range Nuclear Instrumentation: Ability to recognize system parameters that are entry-level conditions for Technical Specifications	**Tier**			**1**
	Group			**2**
	K/A	**032 G 2.2.42**		
	IR			**4.6**

Question 84

Given the following conditions:

- Unit 1 is in MODE 6
- Core reload is in progress

Subsequently:

- Audible indication of count rate for the Startup Range Monitors (SRMs) is lost inside Containment
- Audible and visual SRM indications remain available in the Control Room

Based on these indications, the core reload…

A. MAY continue provided audible AND visual source range indications are available in the control room

B. MUST be suspended in accordance with LCO 3.3.12 Boron Dilution Alarm System, Condition A, for two required SRMs inoperable

C. MAY continue provided audible source range indication is available in the control room AND the Refueling machine maintains constant communications with the control room

D. MUST be suspended and action must be taken to restore audible indication in Containment in accordance with LCO 3.9.2 Nuclear Instrumentation, Conditions A and B for two required SRMs inoperable

Proposed Answer:	D

Explanations:	
A.	Plausible since visual and audible indications will be maintained in the control room, and there is nothing to indicate that the SRM is not functioning (i.e. only the speaker in Containment is faulted), however in order for the SRM to be operable, audible indications are required in both the control room and containment.
B.	Plausible that LCO 3.3.12 would not be met in this situation as inoperability of an SRM normally makes BDAS inoperable, however if SRMs are inoperable SOLELY due to the loss of audible indication, BDAS remains operable.
C.	Plausible since audible indications will be maintained in the control room, and maintaining constant communication with the refueling machine could be interpreted as meeting the requirement for audible indication in containment, however communication from the control room to the refueling machine is not credited for meeting the operability requirement of LCO 3.9.2.
D.	Correct

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2016

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	2	
Reference Provided:	N	
Learning Objective:	25278 – Given a set of plant conditions identify whether or not LCO 3.9.2 is satisfied and any actions or surveillance requirements that would prevent core alterations per Tech Spec 3.9 and its basis	

Technical Reference:	Technical Specifications

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two startup range monitors (SRMs) shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

-----------------------------NOTE-----------------------------
Enter applicable Conditions and Required Actions of LCO 3.3.12, "Boron Dilution Alarm System (BDAS)" for BDAS made inoperable by SRMs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required SRM inoperable.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
B. Two required SRMs inoperable.	B.1 Initiate action to restore one SRM to OPERABLE status.	Immediately

Technical Reference:	Technical Specifications Basis

BASES

LCO (continued)

The SRMs include detectors, preamps, amplifiers, power supplies, indicators, recorders, speakers, alarms, switches and other components necessary to complete the SRM functions. Specifically, each SRM must provide continuous visual indication in the Control Room and each SRM must have the capability to provide audible indication in both the Control Room and Containment via use of the Control Room switch.

APPLICABILITY

In MODE 6, the SRMs must be OPERABLE to determine changes in core reactivity. There is no other direct means available to check core reactivity levels.

The requirements for the associated Boron Dilution Alarm System (BDAS) operability in MODE 6 are contained in LCO 3.3.12, "Boron Dilution Alarm System." LCO 3.3.12 also covers SRM and BDAS operability requirements for MODES 3, 4 and 5.

ACTIONS

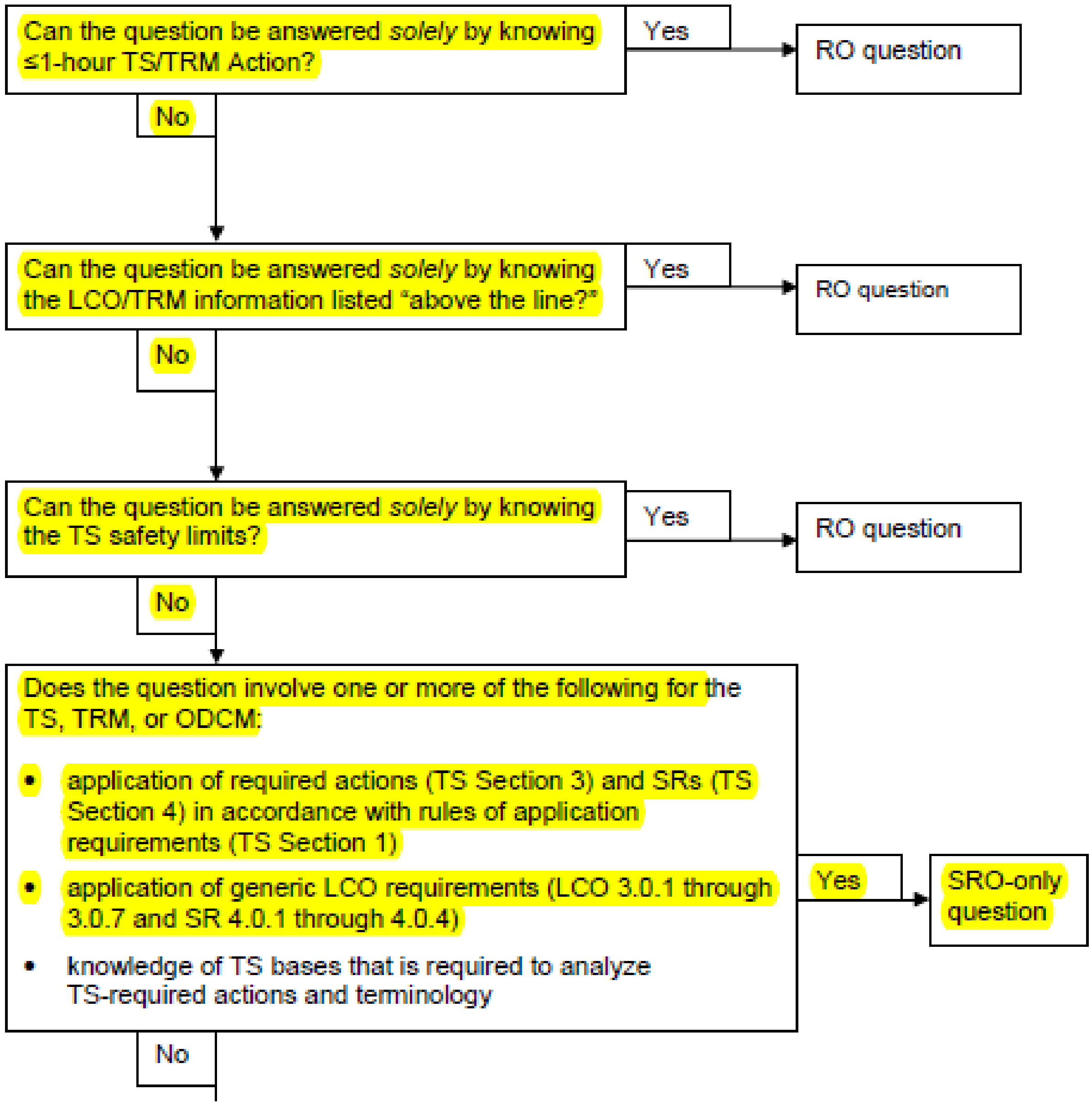
A.1 and A.2

With only one SRM OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

With one required SRM channel inoperable due to loss of its neutron flux indication function, the associated BDAS is also inoperable. If the SRM is inoperable strictly due to a loss of its audible indication function, and the SRM is able to provide neutron flux indication signal to the associated BDAS, the BDAS channel can be considered OPERABLE. With one required BDAS channel inoperable, Action A.1 of LCO 3.3.12 requires the RCS boron concentration to be determined immediately and at the applicable monitoring frequency specified in the COLR Section 3.3.12 in order to satisfy the requirements of the inadvertent deboration safety analysis. The monitoring frequency specified in the COLR ensures that a decrease in the boron concentration during a boron dilution event will be detected with sufficient time for termination of the event before the reactor achieves

Technical Reference:	SRO Level flowchart

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Steam Generator Tube Leak: Magnitude of atmospheric radioactive release if cooldown must be completed using steam dump or atmospheric reliefs	**Tier**			**1**
	Group			**2**
	K/A	**037 AA2.15**		
	IR			**4.2**

Question 85

Given the following conditions:

- Unit 1 was tripped due to a Design Basis Steam Generator Tube Rupture event on SG #1
- On the trip, offsite power was lost
- The crew is commencing a cooldown using ADVs to meet conditions required to isolate SG #1

The use of ADVs for the INITIAL cooldown ___(1)___ considered a loss of the Containment Barrier, and the release in progress ___(2)___ exceeding federally approved limits.

A. (1) IS
(2) IS

B. (1) IS
(2) is NOT

C. (1) is NOT
(2) IS

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

Proposed Answer:	D

Explanations:	
A.	First part is plausible because for the initial RCS cooldown, there will be a release to the environment. However, since it is not an unisolable fault (e.g stuck open MSSV), this is not a loss of the Containment Barrier. Second part is plausible because for the initial RCS cooldown, there will be a release to the environment. However, per 0903, Accident Assessment, is not a release that exceeds Federal limits.
B.	First part is plausible because for the initial RCS cooldown, there will be a release to the environment. However, since it is not an unisolable fault (e.g stuck open MSSV), this is not a loss of the Containment Barrier. Second part is correct.
C.	First part is correct. Second part is plausible because for the initial RCS cooldown, there will be a release to the environment. However, per 0903, Accident Assessment, is not a release that exceeds Federal limits.
D.	Correct

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	4	
Reference Provided:	N	
Learning Objective:	248390 – Determine whether a radioactive release is in progress	

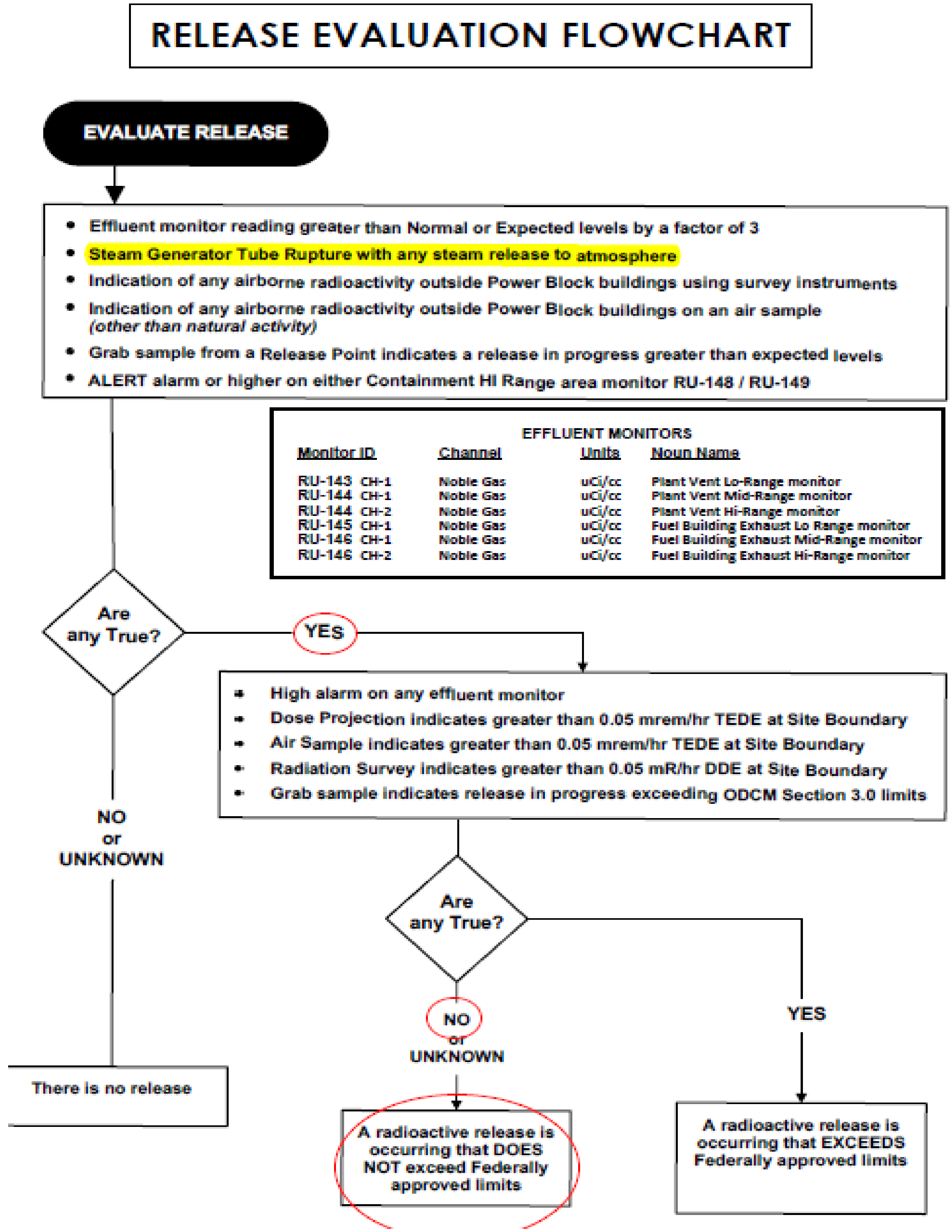
Technical Reference:	EP-0901, Classifications Basis

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant. These type of condition will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG Atmospheric Dump Valve(s) do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. This includes the initial cooldown to 540°F to isolate the ruptured SG using Atmospheric Dump Valves directed in the SGTR EOP. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Technical Reference:	EP-0903, Accident Assessment, Appendix A – Release Evaluation Flowchart



Technical Reference:	SRO Level Question Criteria from NUREG-1021

D. Radiation Hazards That May Arise during Normal and Abnormal Situations, including Maintenance Activities and Various Contamination Conditions [10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include the following:

- process for gaseous/liquid release approvals (i.e., release permits)
- analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures
- analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits

Examination Outline Cross-Reference:	Level	RO	[redacted]	SRO
K/A: Emergency Core Cooling: Knowledge of EOP mitigation strategies	Tier		[redacted]	2
	Group		[redacted]	1
	K/A	006 G 2.4.6		
	IR		[redacted]	4.7

Question 86

Given the following conditions:

- Unit 1 was tripped from 100% power due to a Pressurizer Safety lifting and sticking open
- SPTAs have been performed and the CRS has entered 40EP-9EO03, LOCA
- The RCS is 15°F subcooled and stable
- RCS T_{COLD} is 565°F and slowly lowering
- Indicated Pressurizer level is 95% and slowly rising
- Both SGs are 20% NR and slowly rising, being fed from AFB-P01
- QSPDS indicates 41% in the upper head
- Containment Temperature is 140°F and slowly rising
- Containment High Range Radiation Monitors RU-148 and RU-149 indicate 6.5 x 10^2 mR/hr and slowly rising

The RCS Heat Removal Safety Function is ___(1)___ and the CRS should implement ___(2)___ to lower pressurizer level.

A. (1) MET
(2) Appendix 15, RCS Void Control

B. (1) MET
(2) Appendix 2, Figures: HPSI Throttle Criteria

C. (1) NOT met
(2) Appendix 15, RCS Void Control

D. (1) NOT met
(2) Appendix 2, Figures: HPSI Throttle Criteria

Proposed Answer:	**A**

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because all the parameters meet HPSI Throttle Criteria with the exception of RCS subcooling.
C.	First part is plausible Steam Generator water levels are not in band. However to meet the Safety Function Feedwater only needs to be restoring SGWL back in band. Second part is correct.
D.	First part is plausible Steam Generator water levels are not in band. However to meet the Safety Function Feedwater only needs to be restoring SGWL back in band. Second part is plausible because all the parameters meet HPSI Throttle Criteria with the exception of RCS subcooling.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**5**	
Reference Provided:		
Learning Objective:	**24922 – Given conditions of LOCA, analyze RCS Heat Removal to determine if the SFSC acceptance criteria is satisfied**	

Technical Reference:	40EP-9EO03, Loss of Coolant Accident

* 30. IF at least one HPSI Pump is operating, AND ALL of the following conditions exist:

- RCS is 24°F [44°F] or more subcooled
- Pressurizer level is greater than 10% [15%] and NOT lowering
- At least one Steam Generator is available for RCS heat removal with level being maintained within or being restored to 45 - 60% NR [45 - 60% NR]
- RVLMS indicates RVUH level is 16% or more

THEN throttle HPSI flow or stop the HPSI Pumps one pump at a time.

Technical Reference:	40EP-9EO03, Loss of Coolant Accident

6. RCS Heat Removal

ACCEPTANCE CRITERIA: CRITERIA SATISFIED

a. At least one Steam Generator has level 45 - 60% [45 - 60%] NR. ☐ ☐ ☐ ☐

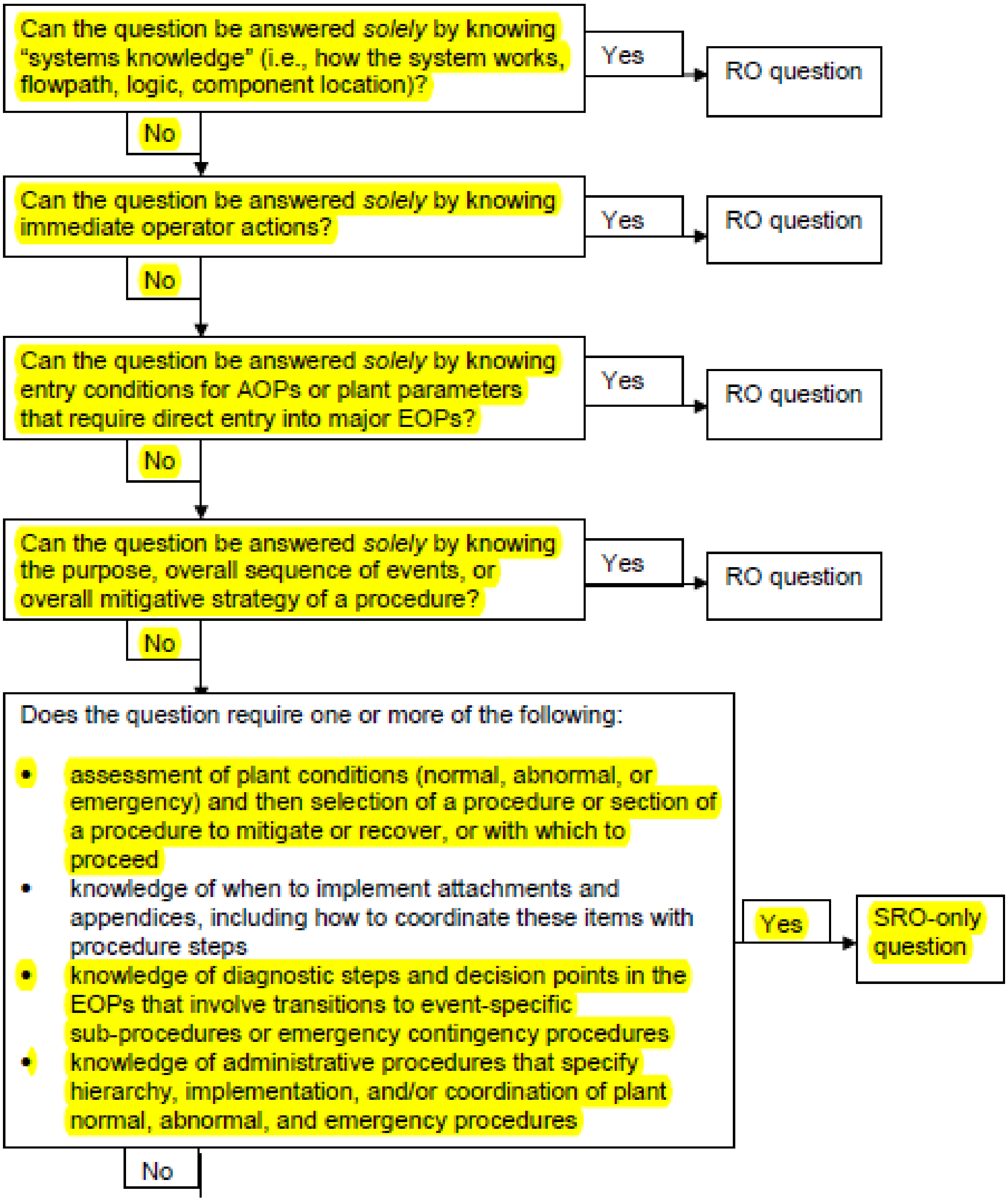
OR

Feedwater is restoring at least one Steam Generator level to 45 - 60% [45 - 60%] NR.

b. T_c is stable or lowering. ☐ ☐ ☐ ☐

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Pressurizer Relief / Quench Tank: Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Overpressurization of the waste gas vent header	Tier			2
	Group			1
	K/A	007 A2.04		
	IR			2.9

Question 87

Given the following conditions:

- Gaseous Radwaste Radiation Monitor RU-12 has failed off scale high and is alarming on RMS
- The Gaseous Radwaste header pressure is slowly rising
- A Waste Gas Decay Tank release is required

Which of the following describes the required action(s) in order to perform the release as planned?

In order for the release to be performed, ___(1)___ as required by ___(2)___.

A. (1) the valve galleries associated with the release path must be posted as a high radiation area
   (2) the Offsite Dose Calculation Manual

B. (1) the valve galleries associated with the release path must be posted as a high radiation area
   (2) 74RM-9EF41, Radiation Monitoring System Alarm Response

C. (1) at least two technically qualified personnel must independently verify the discharge valve lineup
   (2) the Offsite Dose Calculation Manual

D. (1) at least two technically qualified personnel must independently verify the discharge valve lineup
   (2) 74RM-9EF41, Radiation Monitoring System Alarm Response

Proposed Answer:	C

Explanations:	
A.	First part is plausible since there is guidance in the alarm response procedure to evaluate changing the radiation postings in the event of a radiation monitor alarm, and since RU-12 has failed high, it would be reasonable to raise the postings as a conservative approach to ALARA, however this is not required in order for the release to commence. Second part is correct.
B.	First part is plausible since there is guidance in the alarm response procedure to evaluate changing the radiation postings in the event of a radiation monitor alarm, and since RU-12 has failed high, it would be reasonable to raise the postings as a conservative approach to ALARA, however this is not required in order for the release to commence. Second part is plausible since the ARP provides contingency actions for alarming or failed RMs, however there are no requirements in the ARP related to gaseous releases.
C.	Correct
D.	First part is correct. Second part is plausible since the ARP provides contingency actions for alarming or failed RMs, however there are no requirements in the ARP related to gaseous releases.

Question Source:		New	
	X	Bank	
		Modified	
	X	Previous NRC Exam	2018

Cognitive Level:	X	Memory or Fundamental Knowledge
		Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	2	
Reference Provided:	N	
Learning Objective:	386147 – As an SRO describe what actions to take during an effluent release if RU-12, Waste Gas Decay Tank Monitor, goes inoperable	

Technical Reference:	

TABLE 2-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS FUNCTIONAL	APPLICABILITY	ACTIO
1. GASEOUS RADWASTE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release #RU-12	1	#	35

ACTION 35 - With the number of channels FUNCTIONAL less than required by the Minimum Channels FUNCTIONAL requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tanks contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

Technical Reference:	74RM-9EF41, Radiation Monitoring System Alarm Response

2.3 Radiation Protection (RP)

2.3.1 RP is responsible for ALL the following:

- evaluating the need to update/change radiological postings in response to radiation monitor alarms
- controlling of personnel exposure in response to radiation monitor alarms
- reviewing any on-going jobs and plant status with the control room and Chemistry in the evaluation of the cause for radiation monitor alarms
- Performing Section 6.3 actions

Technical Reference:	SRO Level Question Criteria from NUREG-1021

B. Facility Operating Limitations in the Technical Specifications and Their Bases [10 CFR 55.43(b)(2)]

Some examples of SRO exam items for this topic the following:

- application of required actions (TS Section 3) and surveillance requirements (SR) (TS Section 4) in accordance with rules of application requirements (TS, Section 1)
- application of generic limiting condition for operation (LCO) requirements (LCO 3.0.1 through 3.0.7; SR 4.0.1 through 4.0.4).
- knowledge of TS bases that are required to analyze TS-required actions and terminology
- same items listed above for the Technical Requirements Manual (TRM) and Offsite Dose Calculation Manual (ODCM)

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Engineered Safety Features Actuation: Knowledge of annunciator alarms, indications, or response procedures	**Tier**			**2**
	Group			**1**
	K/A	**013 G 2.4.31**		
	IR			**4.1**

Question 88

Given the following conditions:

- Unit 3 was tripped due to an unisolable small break LOCA into Containment

Given the SEIS panel drawing on the following page, the CRS should enter ___(1)___ and direct the crew to cooldown and depressurize the RCS to establish long term cooling via ___(2)___.

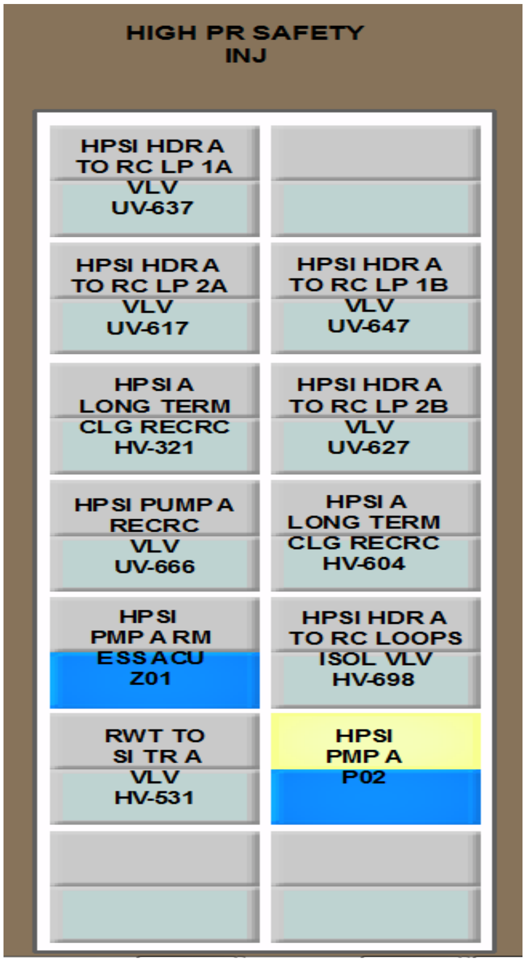
A. (1) 40EP-9EO09, Functional Recovery
   (2) LPSI injection

B. (1) 40EP-9EO09, Functional Recovery
   (2) Shutdown Cooling

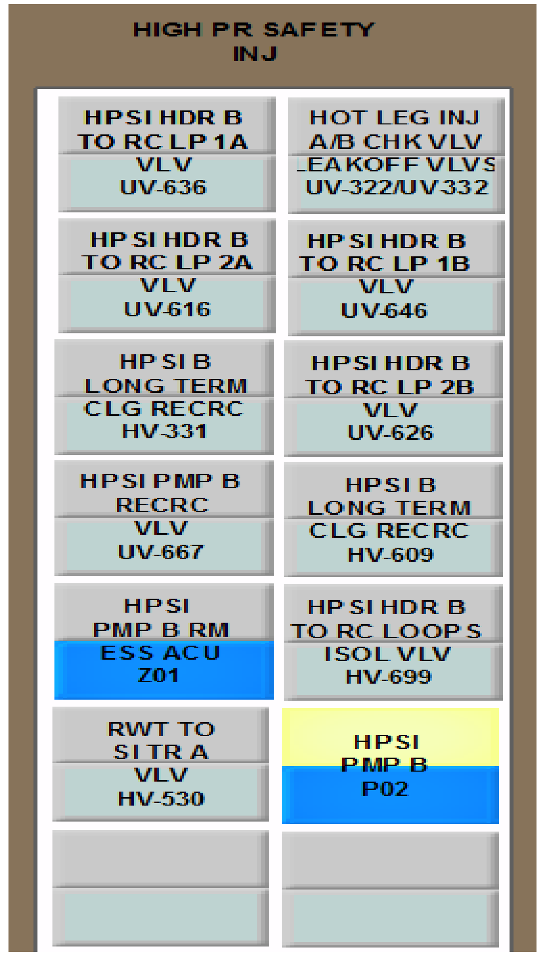
C. (1) 40EP-9EO03, Loss of Coolant Accident
   (2) LPSI injection

D. (1) 40EP-9EO03, Loss of Coolant Accident
   (2) Shutdown Cooling

Train A



Train B



Proposed Answer:	A

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because if the RCS leak was isolable, the crew would cooldown/depressurize to SDC entry conditions. Also, there is currently no injection flow due to the loss of both HPSI pumps. The crew will rapidly cooldown/depressurize to achieve LPSI injection.
C.	First part is plausible because there is a LOCA and the SPTA diagnostic flow chart says to consider LOCA. However, because the RCS Inventory safety function will not be met, the CRS will enter Functional Recovery. Second part is correct.
D.	First part is plausible because there is a LOCA and the SPTA diagnostic flow chart says to consider LOCA. However, because the RCS Inventory safety function will not be met, the CRS will enter Functional Recovery. Second part is plausible because if the RCS leak was isolable, the crew would cooldown/depressurize to SDC entry conditions. Also, there is currently no injection flow due to the loss of both HPSI pumps. The crew will rapidly cooldown/depressurize to achieve LPSI injection.

Question Source:	X	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	3	
10CFR55.43:	5	
Reference Provided:	Y	
Learning Objective:	**25046 – Given conditions of LOCA, analyze the RCS Inventory Control to determine if the SFSC acceptance criteria is satisfied per 40EP-9EO03**	

Technical Reference:	40EP-9EO03, Loss of Coolant Accident

SAFETY FUNCTION:

3. RCS Inventory Control

NOTE

Meeting the provisions of Condition 1 or Condition 2 will satisfy the RCS Inventory Control Safety Function.

ACCEPTANCE CRITERIA:	CRITERIA SATISFIED
Condition 1	
a. Pressurizer level greater than 10% [15%].	☐ ☐ ☐ ☐
b. RCS is 24°F [44°F] or more subcooled.	☐ ☐ ☐ ☐
c. RVLMS indicates that RVUH level is 16% or more.	☐ ☐ ☐ ☐
Condition 2	
a. Safety Injection flow is adequate. REFER TO Appendix 2, Figures.	☐ ☐ ☐ ☐
b. CET Subcooling indicates less than 44°F [60°F] superheat and NOT rising.	☐ ☐ ☐ ☐
c. RCS Subcooling indicates less than 44°F [60°F] superheat and NOT rising.	☐ ☐ ☐ ☐

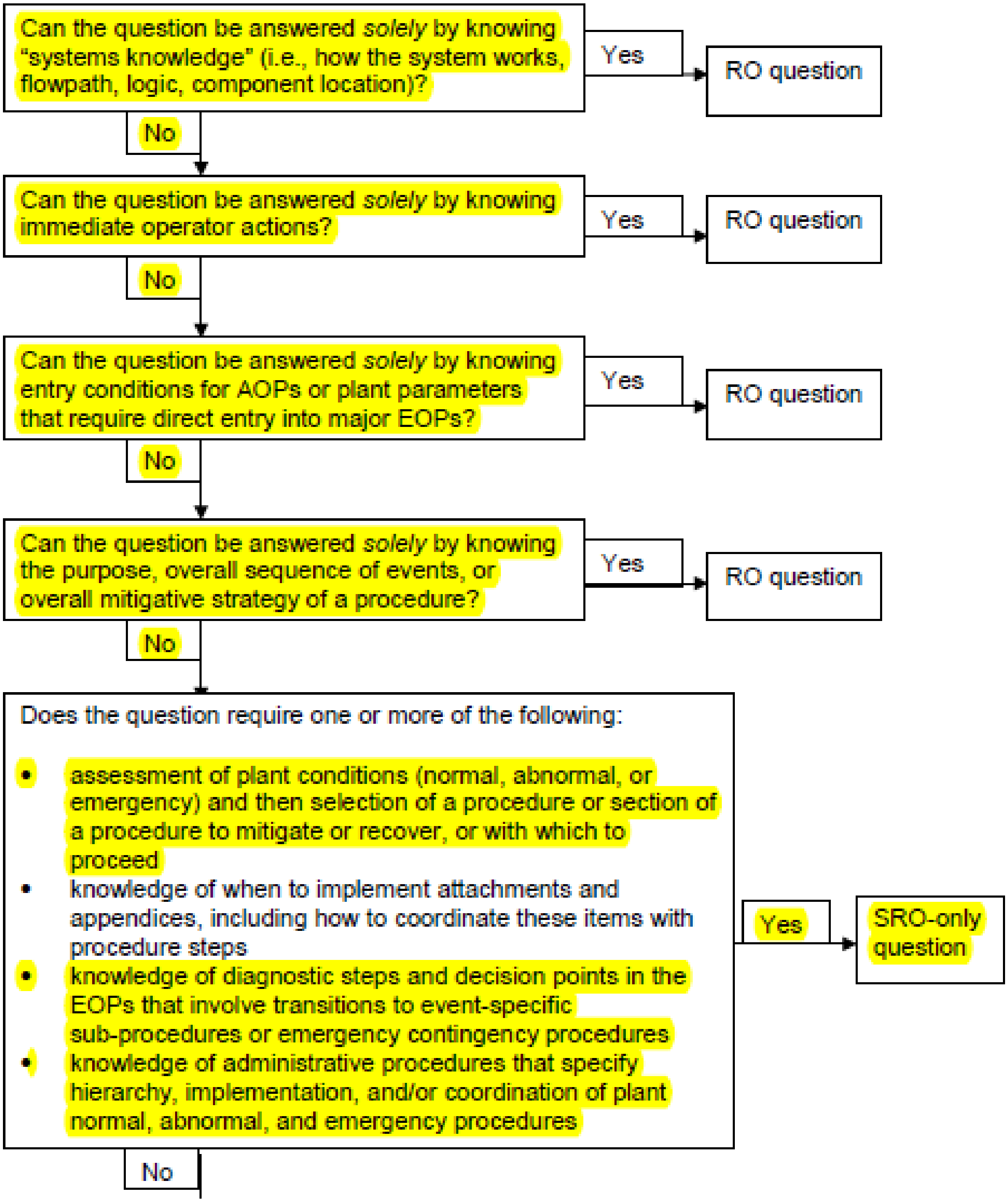
Technical Reference:	40EP-9EO09, Functional Recovery

PALO VERDE NUCLEAR GENERATING STATION FUNCTIONAL RECOVERY	40EP-9EO09 Revision 64 Page 125 of 246 IC-CA Page 2 of 6

INSTRUCTIONS	CONTINGENCY ACTIONS
* 5. **IF** RCS pressure is preventing adequate RCS injection flow, **THEN** depressurize the RCS by **ANY** of the following: • Maximizing RCS Heat Removal. REFER TO the HR Success Path currently in use. • Operation of RCGVS using Success Path PC-2, RCGVS.	

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Containment Cooling: Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of CCS Pump	**Tier**			**2**
	Group			**1**
	K/A	**022 A2.02**		
	IR			**2.6**

Question 89

Given the following conditions following a large break LOCA inside Containment:

- Containment Pressure is 20 psig and slowly rising
- Both CS Pumps have tripped
- The CRS has entered 40EP-9EO09, Functional Recovery

Per 40EP-9EO09, Functional Recovery, the crew should align a ___(1)___ pump to the CS Spray Header and should verify the CTPC Safety Function is met by ___(2)___.

A. (1) LPSI
(2) CS flow indication in the Control Room

B. (1) LPSI
(2) indicated Containment pressure either lowering or stabilizing

C. (1) HPSI
(2) CS flow indication in the Control Room

D. (1) HPSI
(2) indicated Containment pressure either lowering or stabilizing

Proposed Answer:	**B**

Explanations:	
A.	First part is correct. Second part is plausible because when CS is aligned to the spray header, there is indicated flow. However when LPSI is aligned to the spray header, there will be no indicated flow.
B.	Correct
C.	First part is plausible because HPSI will be used for Hot Leg Injection. However LPSI pumps are used for CS when CS pumps are not available. Second part is plausible because when CS is aligned to the spray header, there is indicated flow. However when LPSI is aligned to the spray header, there will be no indicated flow.
D.	First part is plausible because HPSI will be used for Hot Leg Injection. However LPSI pumps are used for CS when CS pumps are not available. Second part is correct.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**5**	
Reference Provided:	**N**	
Learning Objective:	**25485 – Given conditions of LOCA, analyze Containment Temperature and Pressure Control to determine if the SFSC acceptance criteria is satisfied per 40EP-9EO03**	

INSTRUCTIONS	CONTINGENCY ACTIONS
* 3. **IF** CSAS actuated, **THEN** check at least one CS header is delivering 4350 gpm or more.	----------------------- **NOTE** ---------------------- There is no indication of CS flow when using a LPSI pump for CS -- 3.1 **IF** it is desired to use LPSI Pump A to supply CS A train, **AND** LPSI Pump A is **NOT** needed to support any RC, IC, or HR success path, **THEN** perform the following: a. Ensure that LPSI Pump A is running. b. Ensure that SIA-HV-306, LPSI Shutdown Cooling Heat Exchanger A Bypass Valve, is closed. c. Ensure that SIA-HV-687, LPSI-Containment Spray From Shutdown Heat Exchanger A Cross-Tie Valve, is open. d. Ensure that SIA-UV-672, Containment Spray A Discharge To Spray Header 1 Valve, is open. e. Ensure SIA-HV-685, LPSI-Containment Spray To Shutdown Heat Exchanger A Cross-Tie Valve, is open. f. Check that the LPSI pump is running at less than 60 amps.
(continue)	

Technical Reference:	LOIT Functional Recovery Procedure Lesson Plan

EO: 1.33 Given the need to align a LPSI pump to spray the containment (per CTPC-2) , describe the requirements that must be satisfied in order to align a LPSI pump as a containment spray pump in accordance with 40EP-9EO09.

Main Idea

If CSAS actuated and adequate Containment Spray flow of at least one CS header delivering 4350 gpm or more is not present, then CTPC-2 contingency actions allow the use of a LPSI pump in place of a CS pump. The LPSI pumps and motors are similar in design to the CS pumps and will function to give reduced flow to the spray header as a last resort.

The requirement is that the LPSI Pump that is chosen is NOT needed to support any RC, IC, or HR success path as a LPSI pump. One of the LPSI pumps can be used for CS without degradation of safety concerning SI.

There is no indication of CS flow when using a LPSI pump for CS. The CTPC-2 Acceptance Criteria recognizes "LPSI is cross-connected to CS" as acceptable for this condition. Further verification is by plant response by observing properly trending containment parameters after the LPSI pump is aligned to CS.

INSTRUCTIONS	CONTINGENCY ACTIONS
____ 5. **IF BOTH** HPSI pumps are operating, **THEN** calculate the target hot leg flow as follows: a. Record the average indicated cold leg flow from Step 3: _____ gpm b. Multiply the average indicated cold leg flow by 1.5 to obtain the target hot leg flow: _____ gpm	
____ 6. Open the HPSI Long Term Recirc Isolation Valves on **ALL** running HPSI pumps: **HPSI A** • SIA-HV-604 **HPSI B** • SIB-HV-609	
____ 7. Throttle hot leg injection to the target hot leg flow for the running HPSI pump(s): **HPSI A** • SIC-HV-321, HPSI "A" Long Term Cooling Isolation valve **HPSI B** • SID-HV-331, HPSI "B" Long Term Cooling Isolation Valve	

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No ↓

Does the question require one or more of the following:

- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Emergency Diesel Generator: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects	**Tier**			**2**
	Group			**1**
	K/A	**064 G 2.4.35**		
	IR			**4.0**

Question 90

Given the following conditions:

- Unit 1 is operating at 100% power

Subsequently:

- A loss of offsite power occurs
- ‘B’ EDG trips on low lube oil pressure
- The crew performs SPTAs
- The CRS enters 40EP-9EO07, Loss of Offsite Power/Loss of Forced Circulation
- The ECC has informed the Control Room that estimated time for restoration of offsite power is 8 hours
- There is no thunderstorm activity in the area

Per 40EP-9EO07, Loss of Offsite Power/Loss of Forced Circulation, the crew should direct the ___(1)___ to energize NAN-S07 with an SBOG and the CRS should direct an RO to perform ___(2)___.

A. (1) Outside Area Operator
   (2) 40EP-9EO10-081, Appendix 81 - Align SBOG to PBB-S04 (BO)

B. (1) Outside Area Operator
   (2) 40EP-9EO10-054, Appendix 54 - Energize Switchyard Loads From the SBOGs

C. (1) Control Building Operator
   (2) 40EP-9EO10-081, Appendix 81 - Align SBOG to PBB-S04 (BO)

D. (1) Control Building Operator
   (2) 40EP-9EO10-054, Appendix 54 - Energize Switchyard Loads From the SBOGs

Proposed Answer:	B

Explanations:	
A.	First part is correct. Second part is plausible because PBB-S04 is de-energized, however since the MVAC safety function is met, there is no reason to align power to PBB-S04. Additionally, Appendix 81 is not directed in LOOP/LOFC. It only exists in the Blackout EOP.
B.	Correct
C.	First part is plausible because the Control Building Operator will check on a diesel, but it is the A and B EDGs. Second part is plausible because PBB-S04 is de-energized, however since the MVAC safety function is met, there is no reason to align power to PBB-S04. Additionally, Appendix 81 is not directed in LOOP/LOFC. It only exists in the Blackout EOP.
D.	First part is plausible because the Control Building Operator will check on a diesel, but it is the A and B EDGs. Second part is correct.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	5	
Reference Provided:	N	
Learning Objective:	Given conditions of a LOOP, analyze these conditions to determine if switchyard loads should be energized by a SBOG per 40EP-9EO07	

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

All operations involving the SBOGs will be coordinated through Unit 1.

✱ 19. IF power is lost to Unit 1 NAN-S06, THEN direct the Outside Area Operator to perform the following:

a. Start a SBOG using 40EP-9EO10, Appendix 111, Station Blackout Generator Operation.

b. Energize NAN-S07.

CAUTION

A lightning strike on the overhead lines between the switchyard and NAN-S03 may damage ESF Transformer NBN-X03 which will prevent energizing either vital 4.16 kV AC bus from the SBOGs.

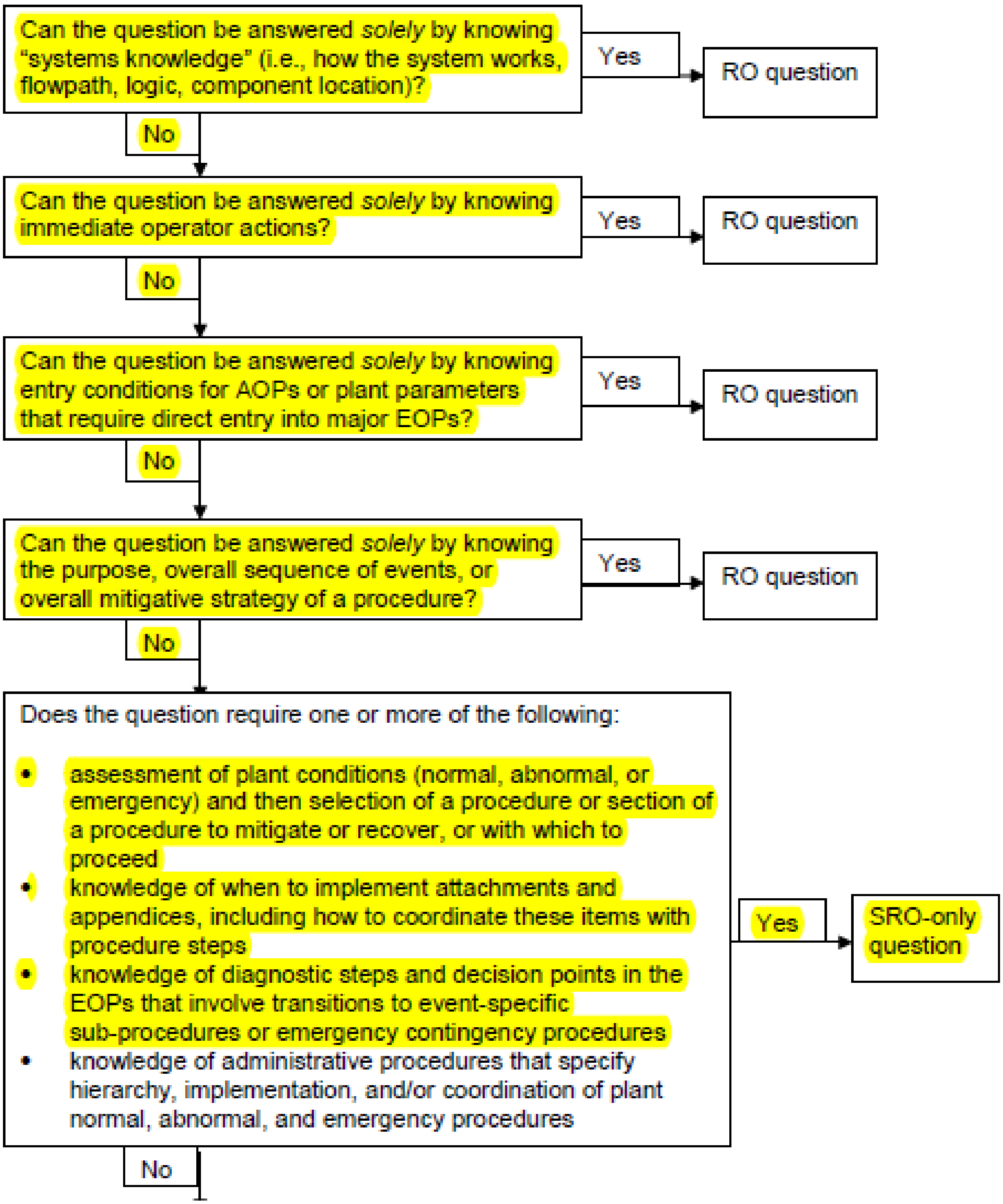
✱ 20. **Unit 1 only -**
IF the switchyard is de-energized,
AND BOTH of the following conditions exist:

- Power restoration is NOT expected within two hours
- Thunderstorm activity is NOT present in the vicinity of PVNGS

THEN PERFORM Appendix 54, Energize Switchyard Loads From the SBOGs.

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Pressurizer Level Control: Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Isolation of letdown	Tier			2
	Group			2
	K/A		011 A2.07	
	IR			3.3

Question 91

Given the following conditions:

- Unit 3 is operating at 12%
- The crew is preparing to sync the Main Generator to the Grid
- CHE-P01 is aligned to Train B
- Charging Pump Mode Selector switch, CHN-HS-4, is in the “2-3-1” position

Subsequently:

- PBB-S04 trips on overcurrent

With NO operator action, Pressurizer level should FIRST exceed the ___(1)___ level LCO 3.4.9 Tech Spec limit and the Unit will be required to be in MODE 3 within a MAXIMUM of ___(2)___ hours from the time that the LCO 3.4.9 limit is exceeded.

A. (1) low
(2) 6

B. (1) low
(2) 7

C. (1) high
(2) 6

D. (1) high
(2) 7

Proposed Answer:	A

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because if pressurizer level lowers to 25%, both class Pressurizer heater banks will be INOPERABLE resulting in LCO 3.0.3. However, per LCO 3.4.9 the unit will be in a 6 hour action once Pressurizer level lowers to 27%.
C.	First part is plausible because if the Charging Pump Mode Selector switch is in the 1-2-3 position, CHA-P01 will remain running, letdown will isolate and the LCO 3.4.9 Pressurizer level high Tech Spec value of 56% will be exceeded.
D.	First part is plausible because if the Charging Pump Mode Selector switch is in the 1-2-3 position, CHA-P01 will remain running, letdown will isolate and the LCO 3.4.9 Pressurizer level high Tech Spec value of 56% will be exceeded. Second part is plausible because as Pressurizer level rises, Pressurizer pressure will also rise. Once Pressurizer pressure rises to 2285 psia, both class Pressurizer heater banks will be INOPERABLE resulting in LCO 3.0.3.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**2**	
Reference Provided:	**N**	
Learning Objective:	**22676 – Given conditions when an LCO is not met, apply Tech Spec Section 3.4.9 (Pressurizer) in accordance with tech pec 3.4.9**	

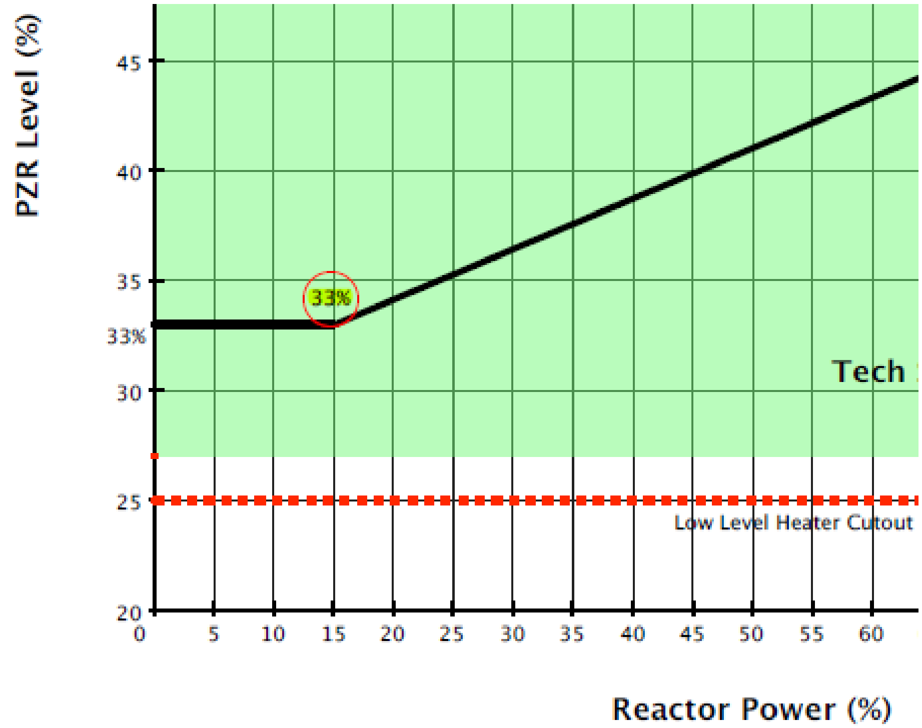
Level Deviation Program

Selected Level vs Selected Setpoint

+15% ↑ - Normally Running Pump Stops
+14% ↓ - Normally Running Pump Restarts
+ 3% ↑ – Hi Level Dev. Backup Heaters on
+2.5% ↓ – Hi Level Dev. resets, Backup Heaters off

- 14% ↑ - Standby Pump Stops
- 23% ↓ – Standby Pump Starts

Technical Reference:	Operator Information Manual



Technical Reference:	Technical Specifications

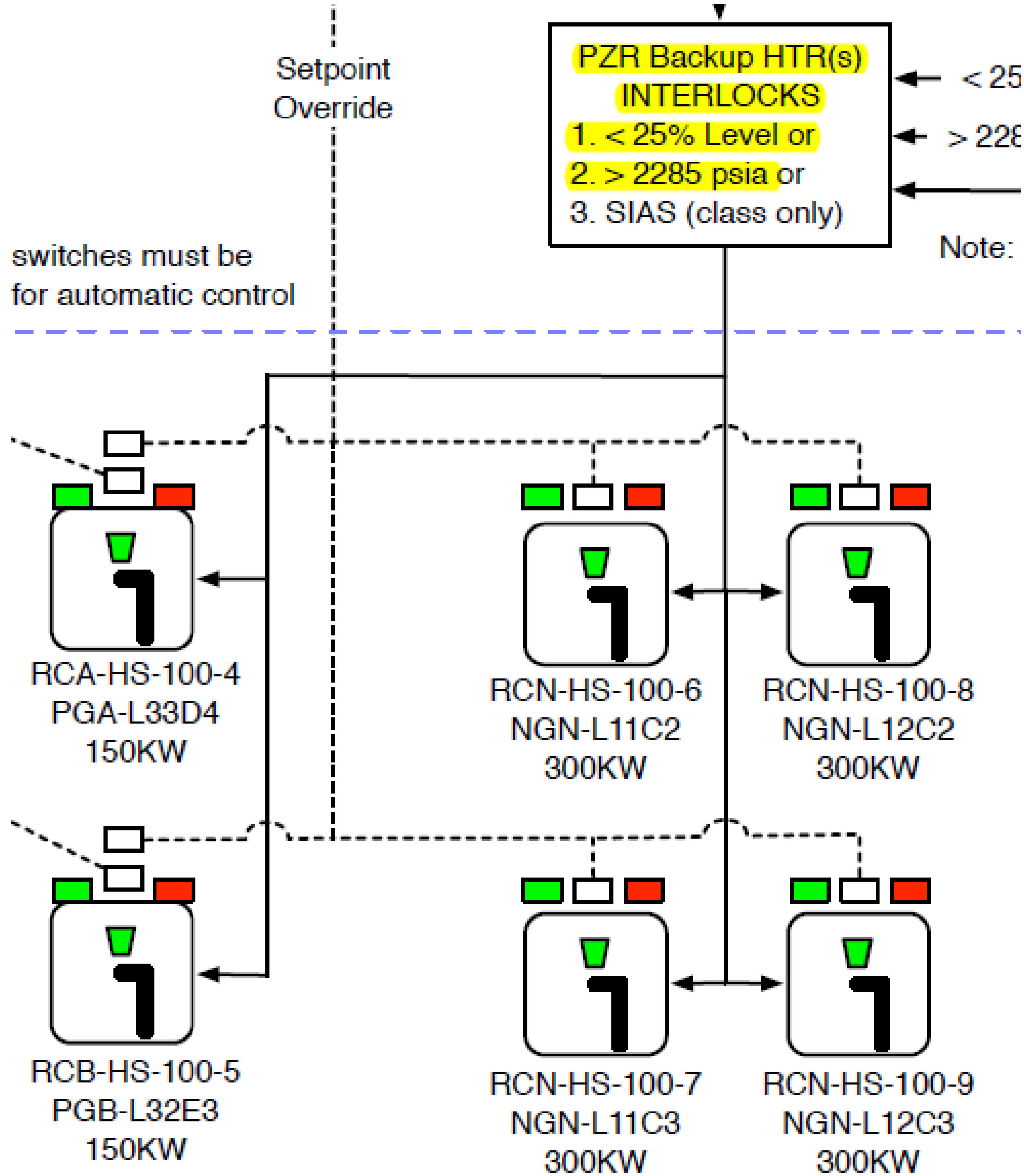
3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≥ 27% and ≤ 56%; and

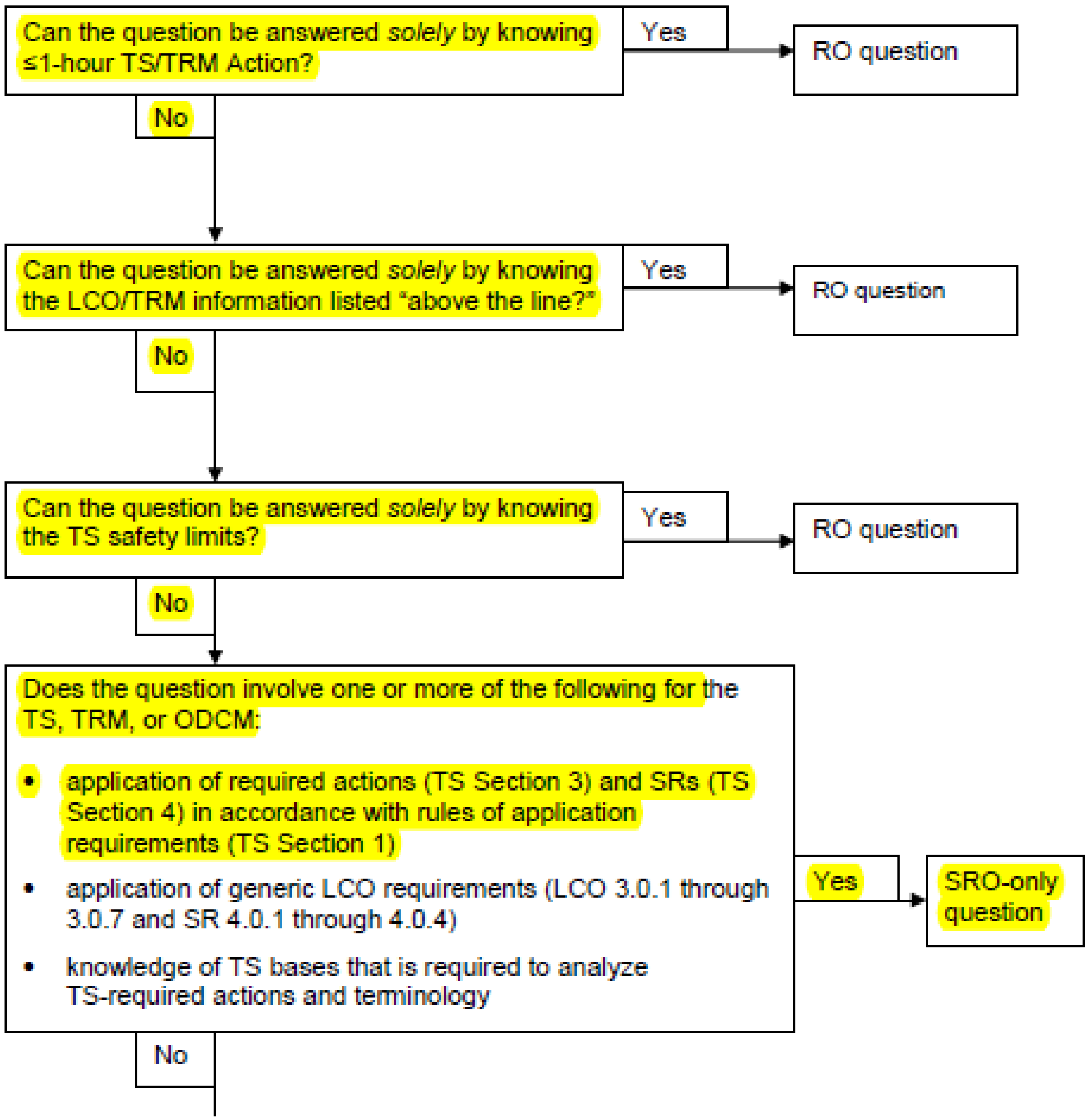
b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 125 kW.



Setpoint
Override
PZR Backup HTR(s)
INTERLOCKS
1. < 25% Level or
2. > 2285 psia or
3. SIAS (class only)
< 25
> 228
Note:
switches must be
for automatic control
RCA-HS-100-4
PGA-L33D4
150KW
RCN-HS-100-6
NGN-L11C2
300KW
RCN-HS-100-8
NGN-L12C2
300KW
RCB-HS-100-5
PGB-L32E3
150KW
RCN-HS-100-7
NGN-L11C3
300KW
RCN-HS-100-9
NGN-L12C3
300KW

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Fuel Handling Equipment: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fuel Handling System controls including: Water level in the refueling canal	**Tier**			**2**
	Group			**2**
	K/A	**034 A1.02**		
	IR			**3.7**

Question 92

Given the following conditions:

- A crack in the Spent Fuel Pool liner has caused the level to lower
- Spent Fuel Pool level is 23 ft 8 in above the irradiated fuel assemblies
- Level is lowering at a rate of 1 inch every 5 minutes

(1) With NO operator action, Spent Fuel Pool level should reach the Technical Specification MINIMUM level in…

(2) The basis for the Technical Specification MINIMUM level is to…

A. (1) 40 minutes
(2) shield and minimize the general area dose when the storage racks are filled to their maximum capacity

B. (1) 40 minutes
(2) maintain Spent Fuel Pool keff < 0.95 assuming the most limiting single fuel mishandling accident

C. (1) 60 minutes
(2) shield and minimize the general area dose when the storage racks are filled to their maximum capacity

D. (1) 60 minutes
(2) maintain Spent Fuel Pool keff < 0.95 assuming the most limiting single fuel mishandling accident

Proposed Answer:	**A**

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because it is the basis for the Spent Fuel Pool boron concentration. If the Spent Fuel Pool lowers and the temperature rises to the point that some Boron may come out of solution and lower the Boron Concentration.
C.	First part is plausible because Spent Fuel Pool level 22 feet 8 inches above irradiated fuel is the Spent Fuel Pool LO-LO alarm setpoint. Second part is correct.
D.	First part is plausible because Spent Fuel Pool level 22 feet 8 inches above irradiated fuel is the Spent Fuel Pool LO-LO alarm setpoint. Second part is plausible because it is the basis for the Spent Fuel Pool boron concentration. If the Spent Fuel Pool lowers and the temperature rises to the point that some Boron may come out of solution and lower the Boron Concentration.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**2**	
Reference Provided:	**N**	
Learning Objective:	**21218 – Given a set of plant conditions, determine whether or not the LCOs and TLCOs of 3.7 are satisfied in accordance with Tech Spec 3.7**	

Technical Reference:	Technical Specifications

3.7.14 Fuel Storage Pool Water Level

LCO 3.7.14 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks.

Technical Reference:	Technical Specifications Basis

B 3.7.14 Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

Technical Reference:	Technical Specifications Basis

B 3.7.15 Fuel Storage Pool Boron Concentration

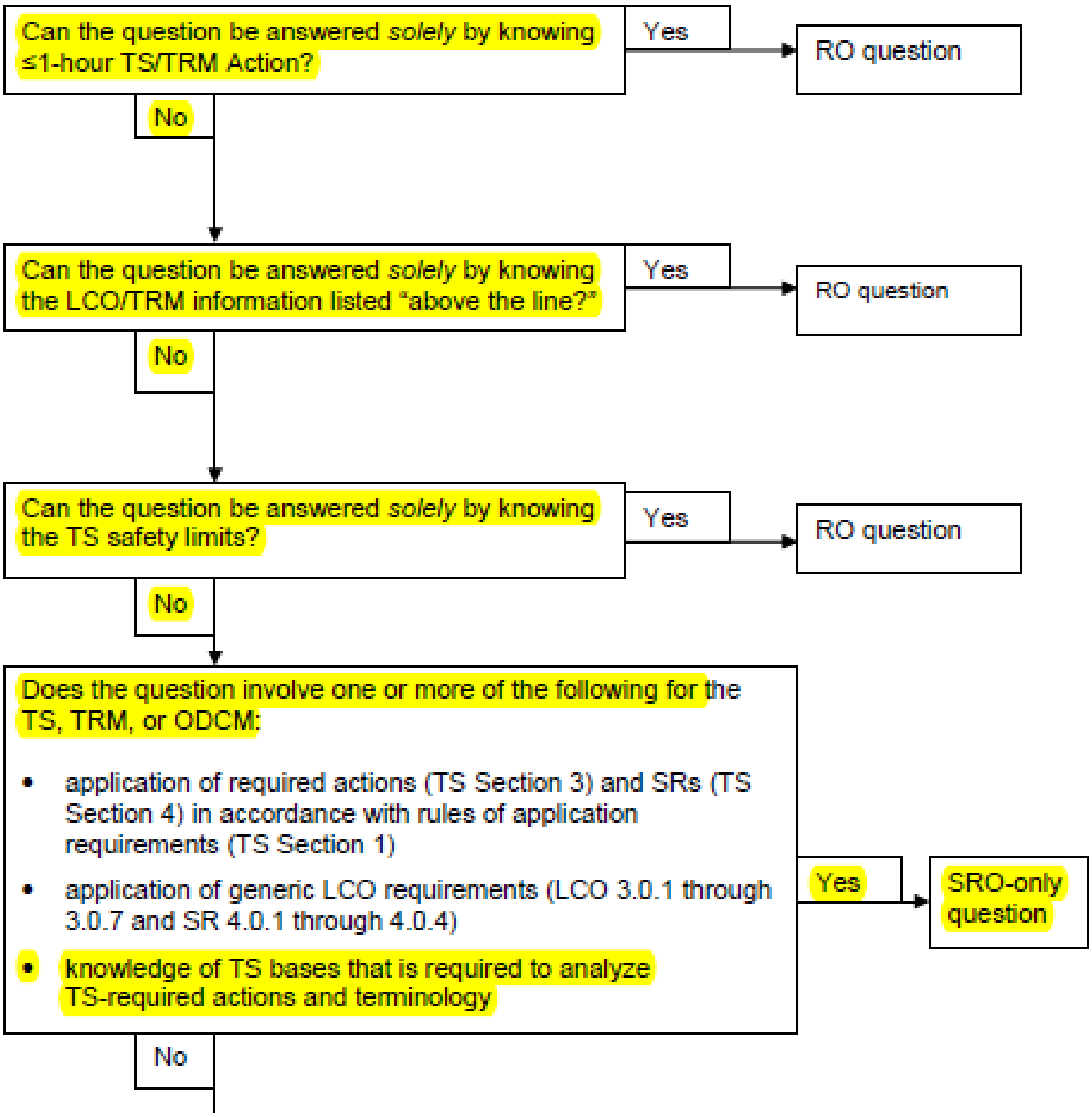
BASES

BACKGROUND

As described in LCO 3.7.17, "Spent Fuel Assembly Storage," fuel assemblies are stored in the spent fuel racks in accordance with criteria based on initial enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ 2150 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations is conservatively developed without taking credit for boron. In order to maintain the spent fuel pool $k_{eff} < 1.0$, a soluble boron concentration of 900 ppm is required to maintain the spent fuel pool $k_{eff} \leq 0.95$ assuming the most limiting single fuel mishandling accident.

Technical Reference:	**SRO Level flowchart from NUREG-1021**

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Condensate: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations	**Tier**			**2**
	Group			**2**
	K/A	**056 G 2.2.36**		
	IR			**4.2**

Question 93

Given the following conditions:

- Unit 1 is operating at 100% power
- The CST is being drained for emergent corrective maintenance
- CST has just dropped below 22 feet

Per LCO 3.7.6, Condensate Storage Tank, the crew must INITIALLY verify operability of the RMWT within a MAXIMUM of ___(1)___ hours and to restore the CST to OPERABLE the CST level will need to be raised to a MINIMUM of ___(2)___ feet.

A. (1) 4
(2) 29.5

B. (1) 4
(2) 31

C. (1) 5
(2) 29.5

D. (1) 5
(2) 31

Proposed Answer:	**A**

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because 31 feet is the alarm setpoint for ‘CST AT MINIMUM OPERATING LEVEL’.
C.	First part is plausible because 5 hours would be allowed if this was a surveillance per SR 3.0.2. However, there is no extension for an LCO. Second part is correct.
D.	First part is plausible because 5 hours would be allowed if this was a surveillance per SR 3.0.2. However, there is no extension for an LCO. Second part is plausible because 31 feet is the alarm setpoint for ‘CST AT MINIMUM OPERATING LEVEL’.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**2**	
Reference Provided:	**N**	
Learning Objective:	**17468 – Given conditions when an LCO is not met, apply Tech Spec 3.7.6 in accordance with Tech Specs**	

Technical Reference:	Technical Specifications

3.7.6 Condensate Storage Tank (CST)

LCO 3.7.6 The CST level shall be ≥ 29.5 ft.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CST level not within limit.	A.1 Verify OPERABILITY of backup water supply.	4 hours

Technical Reference:	40AL-9RK6A, Panel B06A Alarm Responses

Response Section

6A15B

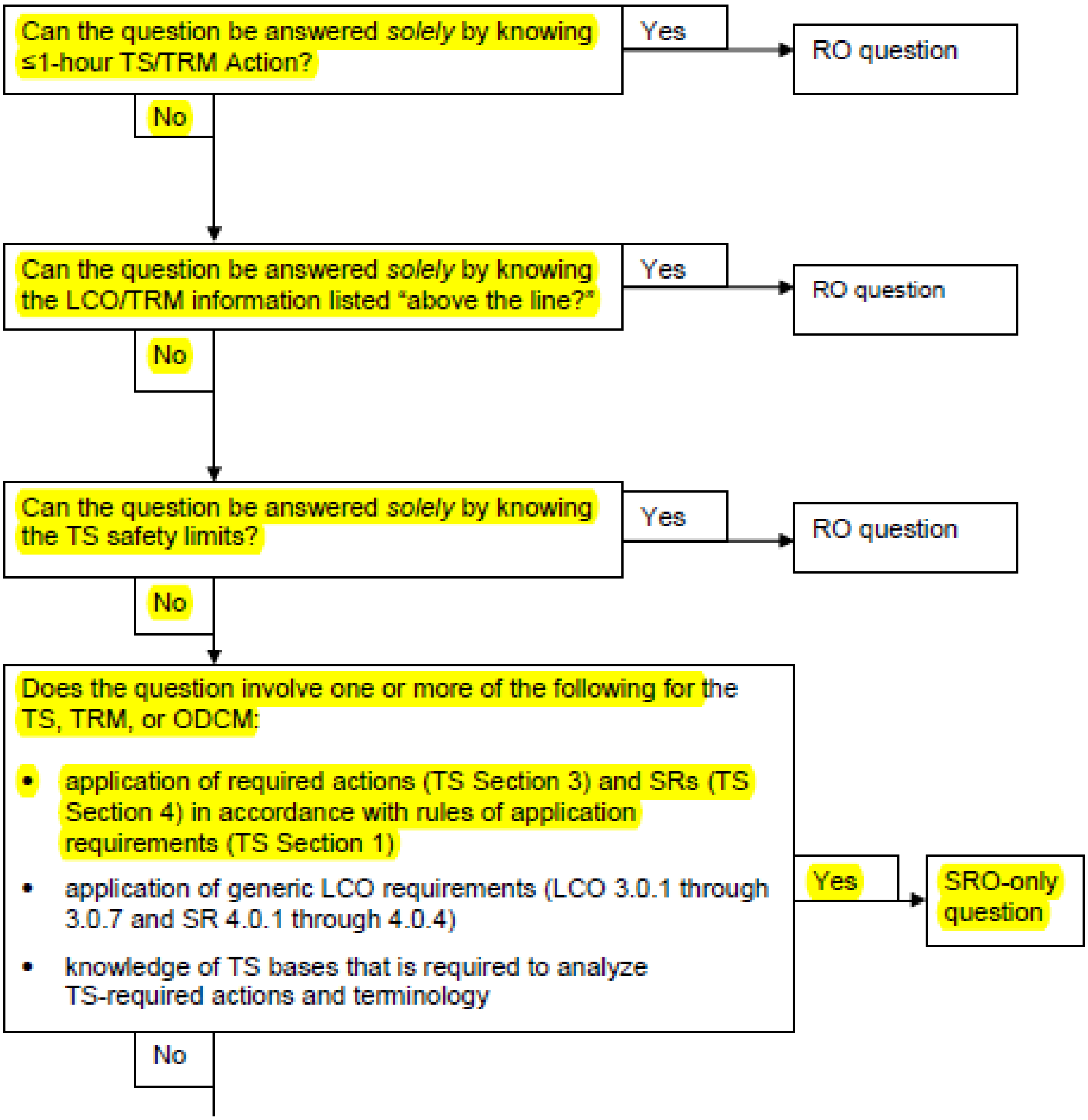
CST AT MINIMUM OPERATING LVL

Condensate Storage Tank at Minimum Operating Level

Point ID	Description	Setpoint
CTLSL11	CST at Minimum Operating Level	31 ft

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Ability to interpret reference materials, such as graphs, curves, tables, etc.	Tier			3
	Group			
	K/A	G 2.1.25		
	IR			4.2

Question 94

Using the Safety Function Tracking Sheet on the following page:

(1) The first Safety Function performed will be…

(2) After all Challenged and Jeopardized Safety Functions are performed, the next Success Path in use to be verified will be…

A. (1) Pressure Control
(2) MVDC

B. (1) Pressure Control
(2) Reactivity Control

C. (1) Heat Removal
(2) MVDC

D. (1) Heat Removal
(2) Reactivity Control

4.0 SAFETY FUNCTION TRACKING

EOP Entry Time ____________

Safety Function	Success Path	Path in use	Challenged	Jeopardized	Completed
RC	RC-1; CEA Insertion	X			
	RC-2; CVCS Boration				
	RC-3; HPSI Boration				
MVDC	MVDC-1; Batt Chargers/Station Batt	X			
MVAC	MVAC-1; Offsite Power	X			
	MVAC-2; DGs				
	MVAC-3; SBOGs				
	MVAC-4; Other Unit DGs				
IC	IC-1; CVCS	X			
	IC-2; SI				
PC	PC-1; Subcooled Pressure Control	X	X		
	PC-2; RCGVS				
	PC-3; Saturated Pressure Control				
HR	HR-1; SG with no SI	X		X	
	HR-2; SG with SI				
CI	CI-1; Auto/Man CTMT Isolation	X			
CTPC	CTPC-1; CTMT Fans	X			
	CTPC-2; CS				

Proposed Answer:	C

Explanations:	
A.	First part is plausible because the Pressure Control safety function is higher in the hierarchy than RCS Heat Removal. However, since it’s jeopardized it takes priority over a challenged safety function. Second part is correct.
B.	First part is plausible because the Pressure Control safety function is higher in the hierarchy than RCS Heat Removal. However, since it’s jeopardized it takes priority over a challenged safety function. Second part is plausible because Success Path performance is normally done in safety function hierarchy order. However, since the completed column is greyed out (all CEAs inserted), the first safety function success path verified will be MVDC.
C.	Correct
D.	First part is correct. Second part is plausible because Success Path performance is normally done in safety function hierarchy order. However, since the completed column is greyed out (all CEAs inserted), the first safety function success path verified will be MVDC.

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	3	
10CFR55.43:	5	
Reference Provided:	Y	Safety Function Tracking Sheet
Learning Objective:	27348 – Given the FRP is being performed and specific plant conditions, determine if a specific selected success path is jeopardized or challenged and how that information will be used in accordance with 40EP-9EO09	

4.0 INSTRUCTIONS

4.1 Characterization of Event

4.1.1 The Functional Recovery procedure is implemented following a reactor trip when any of the following conditions is met:

- The event cannot be diagnosed.
- An appropriate Optimal Recovery Procedure (ORP) is not available.
- The ORP in use is not satisfying safety functions.

4.2 Procedure Strategy

4.2.1 The basic strategy of the Functional Recovery procedure is to first determine the status of all of the safety functions and then build a procedure using the appropriate success paths that will recover or maintain the acceptance criteria of each safety function.

Once the Functional Recovery procedure has been entered, the operator will use the Safety Function Tracking page and the Resource Assessment Trees (RATs) to determine the acceptance criteria and the equipment needed to satisfy each safety function. The operator must determine whether each safety function is jeopardized (acceptance criteria not met), challenged (acceptance criteria met but action must be taken to ensure that the criteria continue to be met) or satisfied in order to set the priorities for performance of the procedure.

Jeopardized safety functions are addressed first. Challenged safety functions are addressed next, with appropriate actions for satisfied safety functions taken last. All safety functions are addressed in the established hierarchy.

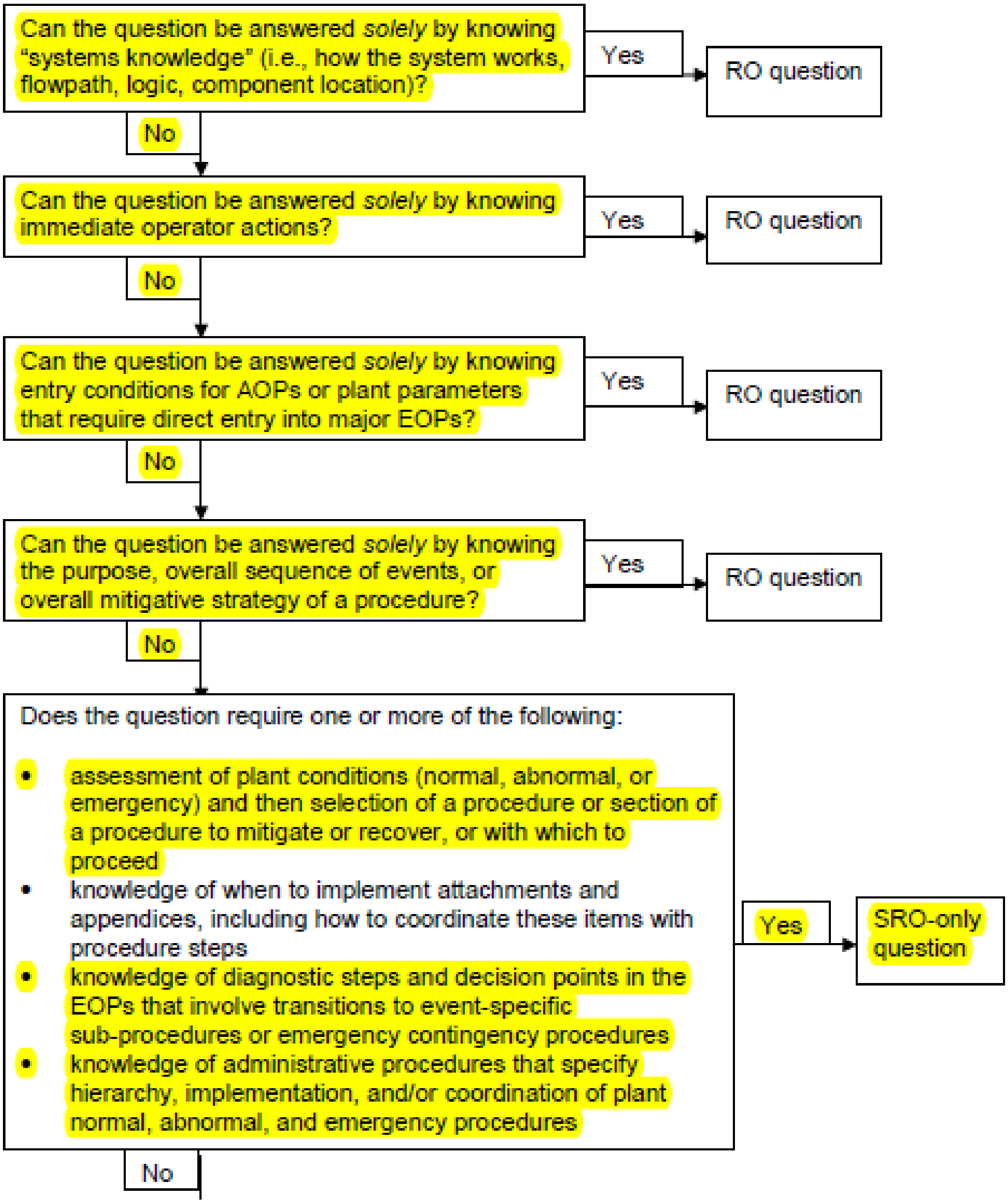
4.6 Safety Function Tracking

4.6.1 The Safety Function Tracking page organizes and condenses information about the success paths in use. It was created to give the CRS a convenient place to keep track of selected success paths in use and their status. The first two columns identify all the success paths as described in section 5.0, Safety Function Status Check. The third column provides the CRS a place to annotate the success paths in use. Also, the third column provides a place to annotate a new success path in use when conditions warrant selection of a new success path. The fourth column provides the CRS with a place to annotate whether the selected success path is challenged. The fifth column provides the CRS with a place to annotate whether the selected success path is in jeopardy or not. The sixth column provides the CRS with a place to annotate that appropriate instructions within the selected success path in use have been completed.

The grayed Completed blocks have a special meaning. Performing the instructions/contingencies for these paths is not required when the associated success path acceptance criteria are satisfied. Meeting the acceptance criteria for these paths implies that no further actions are needed.

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Knowledge of new and spent fuel movement procedures	Tier			3
	Group			
	K/A	G 2.1.42		
	IR			3.4

Question 95

The transportation of a dry cask from the Unit 2 Fuel Building to its designated storage location at the ISFSI is complete

(1) Ownership of the dry cask while being delivered to the ISFSI is the responsibility of the…

(2) Ownership of this dry cask concerning the performance of specific conditional surveillances and inspections is the responsibility of the…

A. (1) Unit 1 Shift Manager
(2) Unit 1 Shift Manager

B. (1) Unit 1 Shift Manager
(2) Unit 2 Shift Manager

C. (1) Unit 2 Shift Manager
(2) Unit 1 Shift Manager

D. (1) Unit 2 Shift Manager
(2) Unit 2 Shift Manager

Proposed Answer:	**C**

Explanations:	
A.	First part is plausible since Unit 1 has ownership of the dry cask conditional surveillances and inspections. Second part is correct.
B.	First part is plausible since Unit 1 has ownership of the dry cask conditional surveillances and inspections. Second part is plausible because Unit 2 has ownership of the dry cask while it being delivered to the ISFSI.
C.	Correct
D.	First part is correct. Second part is plausible because Unit 2 has ownership of the dry cask while it being delivered to the ISFSI.

Question Source:		**New**	
	X	**Bank**	
		Modified	
	X	**Previous NRC Exam**	**2016**

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

Level of Difficultly:	**2**	
10CFR55.43:	**7**	
Reference Provided:	**N**	
Learning Objective:	**90026 – Describe Operation's responsibilities for Dry Cask Storage Operations**	

Technical Reference:	**2016 NRC Exam Original Question**

The transportation of a dry cask from the Unit 2 Fuel Building to its designated storage location at the ISFSI is complete

Who has the ownership of this dry cask concerning the performance of specific conditional surveillances and inspections?

A. Unit 1 Shift Manager

B. Unit 2 Shift Manager

C. Unit 1 Control Room Supervisor

D. Unit 2 Control Room Supervisor

4.13.10 When notified of an emergency, then area operators shall immediately commence a walkdown of their respective areas looking for potential problems, paying particular attention to steam or water leaks and flags on electrical equipment.

4.13.11 There will be no turnovers conducted during the event without prior authorization from the SM/CRS.

4.14 Dry Cask Storage Operations

4.14.1 The SM maintains responsibility and ownership for all activities that take place in their respective Power Block.

4.14.1.1 Nuclear Safety and Reactivity Management are duties that reside with the SM and are not relinquished to any other Site Department.

4.14.2 All Dry Cask Storage Operations which take place in a Unit will be under the authority of the respective Unit SM.

4.14.2.1 The SM is responsible to ensure that all operations are per approved plant/site policies and programs.

4.14.3 The SM responsible for Dry Cask Storage Operations within a Unit is also responsible for Dry Cask Operations during the subsequent transport to the ISFSI and any E-Plan implementation associated with the activities.

4.14.3.1 Transport Operations start when the loaded Concrete Cask has left the respective Fuel Building and RCA and ends when the loaded Concrete Cask is in the designated storage location.

4.14.4 The Unit 1 SM has ownership of the ISFSI concerning the performance of specific conditional surveillance, inspections, and implementation of E-plan activities.

Technical Reference:	SRO Level Question Criteria from NUREG-1021

G. Fuel-Handling Facilities and Procedures [10 CFR 55.43(b)(7)]

Some examples of SRO exam 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

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Ability to perform pre-startup procedures for the facility including operating those controls associated with plant equipment that could affect reactivity	**Tier**			**3**
	Group			
	K/A	**G 2.2.33**		
	IR			**4.4**

Question 96

Given the following conditions:

- Unit 1 is conducting a Reactor startup

Per 40DP-9OP02 Conduct of Operations, during a Reactor Startup the Reactivity Manager can be ___(1)___ and the EARLIEST they are required to be stationed is when the ___(2)___.

A. (1) an off-watch SRO OR the Unit 1 licensed STA
   (2) Shutdown Group Bank CEAs are withdrawn

B. (1) an off-watch SRO OR the Unit 1 licensed STA
   (2) Regulating Group 1 CEAs are withdrawn

C. (1) an off-watch SRO ONLY
   (2) Shutdown Group Bank CEAs are withdrawn

D. (1) an off-watch SRO ONLY
   (2) Regulating Group 1 CEAs are withdrawn

Proposed Answer:	**C**

Explanations:	
A.	First part is plausible because the on watch STA has an SRO license however, he cannot perform duties as the STA and the Reactivity Manager at the same time. Second part is correct.
B.	First part is plausible because the on watch STA has an SRO license however, he cannot perform duties as the STA and the Reactivity Manager at the same time. Second part is plausible because when Regulating Group CEAs are withdrawn, it is the first time that a reactivity change is observable. However, the Reactivity Manager will be stationed during Shutdown bank withdrawal.
C.	Correct
D.	First part is correct. Second part is plausible because when Regulating Group CEAs are withdrawn, it is the first time that a reactivity change is observable. However, the Reactivity Manager will be stationed during Shutdown bank withdrawal.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**5**	
Reference Provided:	**N**	
Learning Objective:	**268823 – Determine the control room operator's responsibilities with respect to Reactivity Management**	

Technical Reference:	40DP-9OP02, Conduct of Operations

4.9.5 Reactor Startup and Shutdown

4.9.5.1 A Reactivity Manager (SRO, other than the CRS or STA, with cognizance of all reactivity manipulations) is assigned to the Control Room for all reactor startup and planned reactor shutdown activities, from the commencement of the first reactivity insertion for a shutdown or CEA withdrawal on a startup.

Technical Reference:	40OP-9ZZ23, Outage GOP

Step 6.5.34.C, Continued

___ 5. **IF** Chemistry reports the backup dip sample taken at Step 6.5.34.C.1. is less than 4000 ppm,
THEN perform the following:

___ a) Ensure PCN-V118, Fuel Transfer Tube Canal Isolation, remains closed.

___ b) Direct Chemistry to sample the Refueling Pool at level greater than 137.8 ft.

___ 6.5.35 **WHEN** the Refueling SRO reports ready to lift the CEA Support Plate,
THEN perform the following:

___ A. Check the SM has granted permission to raise the CEA Support Plate.

___ B. Record an entry in CORA Autolog Core Alterations have commenced.

NOTE
___ Step 6.5.35.C and Step 6.5.35.D are performed simultaneously.

___ C. Commence filling the Refueling Pool as the UGS Lift Rig working platform is raised per 40OP-9PC02, Filling and Draining the Refueling Pool.

___ D. Direct the Refueling SRO to perform the following to raise the UGS Lift Rig:

___ 1. Raise slowly the UGS Lift Rig working platform as the Refueling Water level increases.

___ 2. Check all CEAs are being withdrawn with the UGS Lift Rig working platform.

Technical Reference:	SRO Level Question Criteria from NUREG-1021

The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item. The following are examples:

- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the emergency operating procedures (EOPs) that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Ability to apply Technical Specifications for a system	Tier			3
	Group			
	K/A	G 2.2.40		
	IR			4.7

Question 97

Given the following conditions:

- Unit 1 is exiting an outage with T_{cold} 345°F
- Preparations are being made to enter MODE 3
- During the outage, maintenance on AFA-P01 was conducted and the governor was replaced
- All maintenance activities have been completed including all Surveillance Requirements, with the exception of Surveillances needed to be performed at NOP/NOT

Based on these conditions, AFA-P01 is considered…

A. OPERABLE, and SR 3.0.4 allows changing modes only after performing a risk assessment

B. OPERABLE, because SR 3.0.1 allows the completion of required surveillances when plant conditions support

C. INOPERABLE, however the mode change can be completed and the required surveillances must be completed within a MAXIMUM of 24 hours

D. INOPERABLE, however the mode change can be completed and the required surveillances must be completed within a MAXIMUM of 72 hours

Proposed Answer:	B

Explanations:	
A.	Plausible since AFA-P01 is operable, and its plausible since 3.0.4 addresses changing modes and when to perform a risk assessment.
B.	Correct
C.	Plausible since not all surveillances on AFA-P01 have been completed. 24 hours is plausible since when a surveillance is out of periodicity, the time requirement to complete the surveillance is that surveillance's completion time or 24 hours, whichever is longer.
D.	Plausible since not all surveillances on AFA-P01 have been completed. 72 hours is plausible since it is the time requirement to perform SR 3.7.5.3 once at NOT.

Question Source:		**New**	
	X	**Bank**	
		Modified	
	X	**Previous NRC Exam**	**2016**

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

Level of Difficultly:	**3**	
10CFR55.43:	**2**	
Reference Provided:	**N**	
Learning Objective:	**21081 – Concerning Technical Specification, describe the requirements of SR 3.0.1 in accordance with Tech Specs**	

Technical Reference:	Technical Specifications

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

Technical Reference:	Technical Specifications Bases

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.8 establish the general requirements applicable to all Specifications and apply at all times unless otherwise stated.
LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

Technical Reference:	Technical Specifications

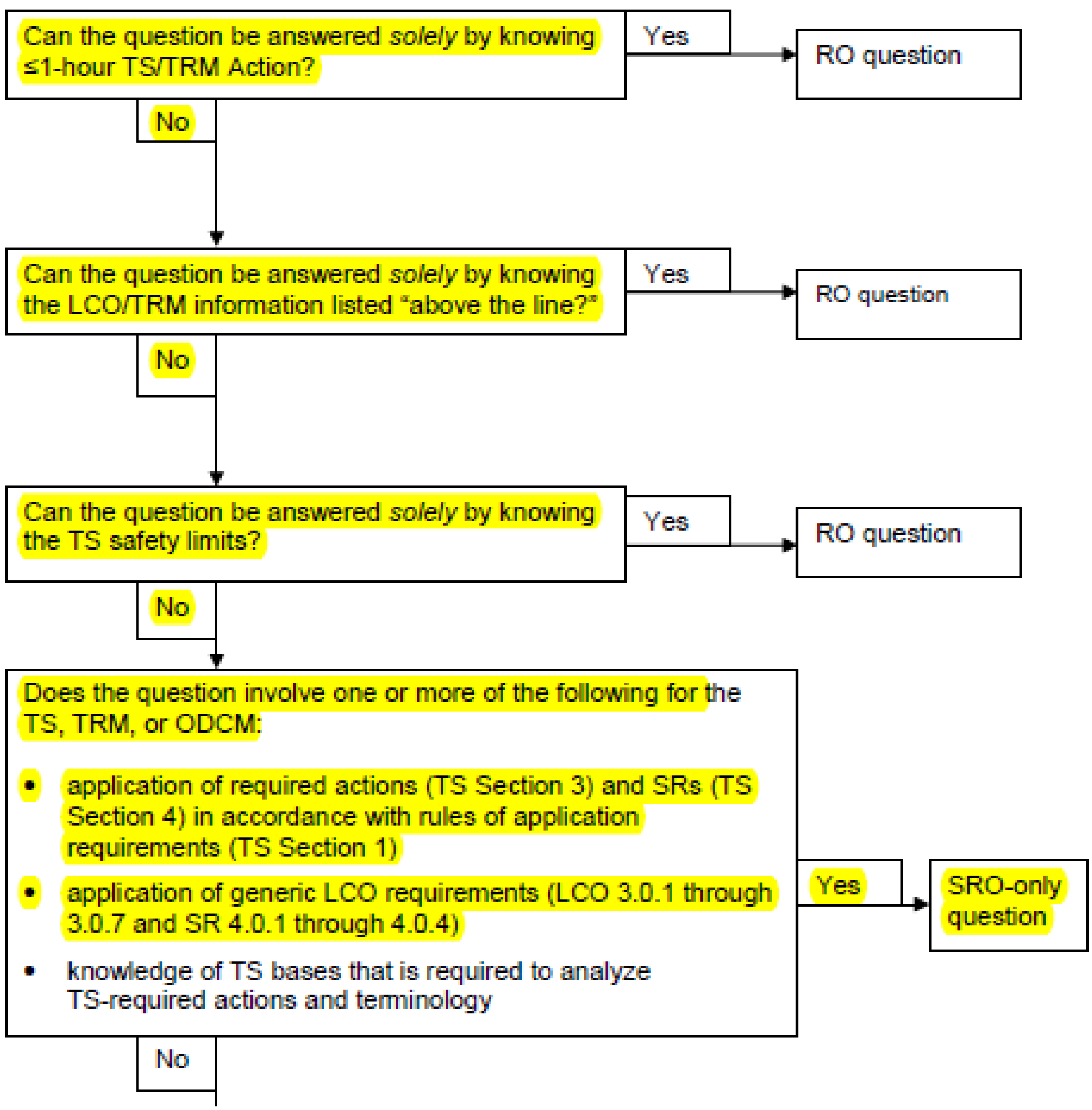
AFW System
3.7.5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each AFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	----------NOTE---------- Not required to be performed for the turbine driven AFW pump until 72 hours after reaching 532°F in the RCS. ---------- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.5.3	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until 72 hours after reaching 532°F in the RCS.	

Technical Reference:	SRO Level flowchart from NUREG-1021

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)



Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Ability to approve release permits	Tier			3
	Group			
	K/A	G 2.3.2		
	IR			3.8

Question 98

Given the following conditions:

- A large break LOCA has occurred.
- A Site Area Emergency has been declared
- Due to emergency conditions, a gaseous radioactive release from Containment must be performed to relieve pressure in the Containment and bring the plant to a safer condition.

(1) During a SAE, releases ___(1)___ exceed EPA Protective Action Guidelines (PAGs) at the site boundary

(2) The SM/CRS ___(2)___ the only personnel that may AUTHORIZE the release

A. (1) WILL
(2) ARE

B. (1) WILL
(2) are NOT

C. (1) will NOT
(2) ARE

D. (1) will NOT
(2) are NOT

Proposed Answer:	C

Explanations:	
A.	First part is plausible because during a SGTR when ADVs must be used for the initial cooldown, federal limits are not exceeded. That scenario would be an Alert declaration. It may be assumed that because an SAE is the next higher declaration, PAGs will be exceeded. Second part is correct.
B.	First part is plausible because during a SGTR when ADVs must be used for the initial cooldown, federal limits are not exceeded. That scenario would be an Alert declaration. It may be assumed that because an SAE is the next higher declaration, PAGs will be exceeded. Second part is plausible because the RP Manager and Sr VP Site Operations must acknowledge the release but they cannot authorize the release.
C.	Correct
D.	First part is correct. Second part is plausible because the RP Manager and Sr VP Site Operations must acknowledge the release but they cannot authorize the release.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**3**	
10CFR55.43:	**4**	
Reference Provided:	**N**	
Learning Objective:	**25949 - Describe whose authority is needed to exceed requirements and what reporting is necessary**	

Technical Reference:	74RM-9EF20, Gaseous Radioactive Release and Offsite Dose Assessment

Appendix B - Release Permit Review and Approval Matrix

Description of Release Action Levels	Release Level as % of any Dose/Dose Rate ODCM Requirement	Radiation Protection Supervision	Radiological Services Superintendent	Operations Department Leader	Radiation Protection Manager	Shift Manager/CRS	Sr Vice President Site Operations
Less than or Equal to 50% of the Admin. Dose/Dose Rate Limit (a)	Dose/Dose Rate < 40%	Review and Approval (e)	N/A	N/A	N/A	Authorize (c)	N/A
Greater than 50% but less than the Admin. Dose/Dose Rate Limit (a)	Dose/Dose Rate >40% and <80%	Review	Review and Approval	Acknowledge (b)	Acknowledge (b)	Authorize (c)	N/A
Greater than or equal to the Admin. Dose/Dose Rate Limit (a)(f)	Dose/Dose Rate ≥ 80%	Review	Review	Review and Approval	Acknowledge (b)	Authorize (d)	Acknowledge (b)

a. Applies to the quarterly and annual air and organ dose limits and instantaneous dose rate limits and not to the 31 day dose projection limits.

b. Acknowledgment requires that the appropriate individual be informed that the applicable dose/dose rate limit is being approached and that actions should be taken to reduce future releases. Acknowledgment should be obtained prior to release but can be obtained as soon as practical after the release.

c. Authorization of Permits for routine continuous releases are not required.

d. Under abnormal (emergency) conditions verbal approval for exceeding ODCM Requirement limits may be given by the CRS/Shift Manager when performing the release if it will bring the plant in to a safer condition. A notification to the NRC within one hour in accordance with 10CFR50.72 will be required after approval. If ODCM Requirement limits for dose are exceeded (ODCM sections 4.4a, 4.4b, 4.1a, 4.1b, 4.2a or 4.2b) comply with ODCM Requirement 5.1.

Technical Reference:	SRO Level Question Criteria from NUREG-1021

D. Radiation Hazards That May Arise during Normal and Abnormal Situations, including Maintenance Activities and Various Contamination Conditions [10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include the following:

- process for gaseous/liquid release approvals (i.e., release permits)
- analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures
- analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Knowledge of general guidelines for EOP usage	Tier			3
	Group			
	K/A	G 2.4.14		
	IR			4.5

Question 99

(1) Per EOP Operations Expectations, during performance of SPTAs, when a step requires going to contingency actions…

(2) Per EOP Operations Expectations, once an EOP is entered the CRS should ensure that Safety Function Status Checks are completed within a MAXIMUM of…

A. (1) CRS concurrence is REQUIRED
(2) 15 minutes

B. (1) CRS concurrence is REQUIRED
(2) 30 minutes

C. (1) CRS concurrence is NOT required
(2) 15 minutes

D. (1) CRS concurrence is NOT required
(2) 30 minutes

Proposed Answer:	**A**

Explanations:	
A.	Correct
B.	First part is correct. Second part is plausible because the STA (normally performs SFSCs) will perform Accountability and a Core Damage Assessment within 30 minutes of an EAL being exceeded.
C.	First part is plausible because when Reactor trip and ESFAS setpoints are exceeded, CRS concurrence is not required for manual actuation. Second part is correct.
D.	First part is plausible because when Reactor trip and ESFAS setpoints are exceeded, CRS concurrence is not required for manual actuation. Second part is plausible because the STA (normally performs SFSCs) will perform Accountability and a Core Damage Assessment within 30 minutes of an EAL being exceeded.

Question Source:	**X**	**New**	
		Bank	
		Modified	
		Previous NRC Exam	

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

Level of Difficultly:	**2**	
10CFR55.43:	**5**	
Reference Provided:	**N**	
Learning Objective:	**Given that an ORP, FRP, or LMFRP is in use, describe the performance of the Safety Function Status Checks in accordance with 40DP-9AP16**	

4.0 STANDARD POST TRIP ACTIONS (SPTAs)

SPTA Step: ALL

1. The CRS may use 1, 2, or 3 ROs to complete SPTAs. Normally 2 or 3 will be used. With 3 ROs, the third RO normally checks MVA, then checks the RMS. The ROs are to be flexible and gather information as requested by the CRS.
2. The STA and SM have to be careful not to disrupt the close teamwork of the CRS and the ROs during the SPTAs. Normally, the SM and STA should remain behind the communications console during SPTAs.
3. Unless otherwise noted, the SPTAs consist of quick actions from the control room. Any other actions directed by the CRS during SPTAs should conform to this rule.
4. The ROs should communicate directly to the CRS. Cross communication between the ROs should be limited to specific items such as changing feed flow, and initiating actions that will result in annunciators.
5. The CRS should communicate with the ROs and ensure the ROs acknowledge any prompted questions with a value, trend, and method of control, if applicable. The CRS should communicate with the STA as necessary to concur with the diagnosis. The CRS should communicate with the SM to discuss the diagnosis.
6. The CRS shall normally direct progress through the SPTA's by prompting the ROs for information relating to the SPTA steps. Expected performance is to detect and verify the condition, notify other control room personnel of the condition (using Update or focused communication).
7. CRS concurrence shall be obtained before taking contingency actions.
8. If Reactor Protection System trip setpoints have been exceeded and a trip has not occurred, then Trip the reactor while notifying the CRS and other control room personnel that PPS setpoints were exceeded.

Technical Reference:	EOP Operations Expectations

10. The CRS is responsible to ensure the Safety Function Status Check (SFSC) is completed. This is normally delegated to the STA. The STA will normally perform the SFSC at an interval of approximately every 15 minutes. If the STA is not in the control room when the CRS gets to this step, another member of the control room staff (normally the 3rd Reactor Operator) must be designated to perform the SFSC until the STA arrives.

Technical Reference:	SRO Level Question Criteria from NUREG-1021

The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item. The following are examples:

- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the emergency operating procedures (EOPs) that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Examination Outline Cross-Reference:	Level	RO		SRO
K/A: Knowledge of the lines of authority during implementation of the emergency plan	Tier			3
	Group			
	K/A	G 2.4.37		
	IR			4.1

Question 100

Given the following conditions:

- It is Thursday afternoon during a non-holiday workday
- At time = **1300** - An ALERT is declared for an event in progress
- At time = **1530** - A GENERAL EMERGENCY is declared

The Emergency Coordinator is located in the ___(1)___ and the PAR will be performed by the ___(2)___.

A. (1) Control Room
   (2) Emergency Coordinator

B. (1) Control Room
   (2) Emergency Operations Director (EOD)

C. (1) Technical Support Center (TSC)
   (2) Emergency Coordinator

D. (1) Technical Support Center (TSC)
   (2) Emergency Operations Director (EOD)

Proposed Answer:	D

Explanations:	
A.	Plausible because if the General Emergency was declared less than an hour after the Alert, then the Shift Manager will still have EC duties in the Control Room including PARs.
B.	First part is plausible because if the General Emergency was declared less than an hour after the Alert, then the Shift Manager will still have EC duties in the Control Room including PARs. Second part is correct.
C.	First part is correct. Second part is plausible if the General Emergency was declared less than an hour after the Alert, then the Shift Manager will still have EC duties in the Control Room including PARs.
D.	Correct

Question Source:	X	New	
		Bank	
		Modified	
		Previous NRC Exam	

Cognitive Level:		Memory or Fundamental Knowledge
	X	Comprehension or Analysis

Level of Difficultly:	**3**	
10CFR55.43:	**5**	
Reference Provided:	**N**	
Learning Objective:	**64480 – State the purpose and location of the onsite and offsite Emergency Response Facilities**	

Technical Reference:	EP-0904, ERO/ERF Activation and Operation

4.0 DEFINITIONS

4.1 **Command and Control** — The designated Emergency Response Facility (ERF) position that has overall responsibility for the Palo Verde emergency response efforts.

- Emergency Coordinator - Satellite Technical Support Center (STSC)
- Emergency Coordinator - Technical Support Center (TSC)
- Emergency Operations Director - Emergency Operations Facility (EOF)

4.2 **Emergency Personnel** — The organizational groups that perform a functional role during an emergency condition.

4.3 **Emergency Response Organization (ERO) Activation** — The process used with the intention of fully staffing the facility to assume the positional responsibilities described in the Emergency Plan. ERO Activation time is measured from the time when the event is classified until the responders have reported to the facility.

4.4 **Facility Activation** — A Facility is ready for activation when it is ready to assume its assigned functions in the Emergency Plan and relieve the on-shift staff of those functions. The goal is to be capable of activating the applicable ERF (TSC, OSC and EOF) within 60 minutes following an event classification during normal working hours and 120 minutes during off-normal working hours. The facility can be declared activated when the following conditions are met:

Technical Reference:	EP-0904, ERO/ERF Activation and Operation

6.0 INSTRUCTIONS

6.1 Non-Delegable Responsibilities

6.1.1 Transfer the following responsibilities from the Emergency Coordinator- Satellite Technical Support Center (EC-STSC) in the Control Room to the Emergency Operations Director (EOD) in the Emergency Operations Facility (EOF):

- Final decision to notify the offsite agencies
- Final decision to recommend protective actions to the offsite agencies

6.1.2 Transfer the following responsibilities from the EC-STSC to the Emergency Coordinator-Technical Support Center (EC-TSC):

- Final decision to declare the emergency classification
- Final decision for issuance of thyroid blocking agents (that is, Potassium Iodide or KI) to PVGS emergency workers and onsite personnel.
- Authorization of personnel exposure per EPA-400 (PAG Manual) limits.

6.1.3 Move non-delegable responsibilities with the transfer of Command and Control as indicated in the below table.

EC – STSC	EC - TSC	EOD - EOF
Classification	X	
PARs		X
Notifications		X
Emergency Exposure Controls	X	
Potassium Iodide (KI) Issuance	X	

Technical Reference:	SRO Level Question Criteria from NUREG-1021

The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item. The following are examples:

- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the emergency operating procedures (EOPs) that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Technical Reference:	SRO Level Question – SRO Master Task List

MASTER TASK LIST

Task list for OPTRNG at 2020/01/29: (189524) Senior Reactor Operator
All Tasks

Task#	Task	Selected for Training	Recurring	How Often	Training Setting
L392177	Transfer command and control of the Emergency Coordinator functions	Yes	No		Classroom

1	B	26	A	51	D	76	D
2	C	27	B	52	C	77	B
3	A	28	D	53	D	78	B
4	D	29	C	54	A	79	B
5	A	30	D	55	A	80	C
6	C	31	B	56	C	81	D
7	A	32	D	57	B	82	D
8	D	33	C	58	B	83	B
9	D	34	B	59	C	84	D
10	A	35	A	60	D	85	D
11	A	36	C	61	B	86	A
12	A	37	C	62	A	87	C
13	B	38	A	63	D	88	A
14	A	39	C	64	C	89	B
15	A	40	C	65	D	90	B
16	D	41	A	66	B	91	A
17	D	42	C	67	B	92	A
18	A	43	B	68	D	93	A
19	A	44	A	69	B	94	C
20	A	45	B	70	C	95	C
21	B	46	D	71	C	96	C
22	B	47	B	72	B	97	B
23	B	48	D	73	D	98	C
24	D	49	B	74	C	99	A
25	D	50	A	75	C	100	D