1 **ID: 979002** **Points: 1.00**

Plant Conditions:

- The reactor is at 75% power.
- Normal equipment lineups.

Event:

- RC-P-1B, "B" Reactor Coolant Pump, trips.
- Feedwater flow failed to re-ratio.

Given the above information and assuming no operator actions have been taken, a lower than expected level will occur in the ___(1)___ OTSG, which will result in (2) ΔTcold reading in the___(2)___ direction on the console indication.

A. (1) A
(2) positive

B. (1) A
(2) negative

C. (1) B
(2) positive

D. (1) B
(2) negative

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. Plausible if candidate believes that because RC-P-1B, "B" Reactor Coolant Pump, is in the "A" RCS Loop (correct) and the "A" RCS loop having excessive feedwater flow will result in a lower Tcold (correct), a lower Tcold requires less heat sink and therefore ICS will lower level in the "A" OTSG. Although this is true, it is also expected. The question asks for a "lower than expected level".

Part 2 is incorrect. Plausible if the candidate correctly understands that ΔTcold = A Tcold - B Tcold but believes that RC-P-1B is in the "B" RCS Loop.

B. **Incorrect.**
Part 1 is incorrect. Plausible if candidate believes that because RC-P-1B, "B" Reactor Coolant Pump, is in the "A" RCS Loop (correct) and the "A" RCS loop having excessive feedwater flow will result in a lower Tcold (correct), a lower Tcold requires less heat sink and therefore ICS will lower level in the "A" OTSG. Although this is true, it is also expected. The question asks for a "lower than expected level".

Part 2 is correct. If FW flow does not re-ratio, then feedwater flow in the A loop remains equal with the B loop. Therefore the A loop having an excessive flow results in a lower Tcold. Lower Tcold in the A loop will result in the B loop Tcold appearing hot and the ΔTcold indication will be in the negative direction on the console indication.

C. **Incorrect.**
Part1 is correct. If FW flow does not re-ratio, then feedwater flow in the A loop remains equal with the B loop. Therefore the A loop having an excessive flow results in a lower Tcold. A lower Tcold requires less heat sink and therefore, ICS will lower level in the "A" OTSG.

Part 2 is incorrect. Plausible if the candidate incorrectly believes that ΔTcold = B Tcold - A Tcold

D. **Correct.**

Part 1 is correct. If FW flow does not re-ratio, then feedwater flow in the A loop remains equal with the B loop. Therefore the A loop having an excessive flow results in a lower Tcold. A lower Tcold requires less heat sink and therefore, ICS will lower level in the "A" OTSG. Level will lower in the "B" OTSG due to more heat removal.

Part 2 is correct. If FW flow does not re-ratio, then feedwater flow in the A loop remains equal with the B loop. Therefore the A loop having an excessive flow results in a lower Tcold. Lower Tcold in the A loop will result in the B loop Tcold appearing hot and the ΔTcold indication will be in the negative direction on the console indication. IAW TQ-TM-104-621-C001, Integrated Control System:

- Feedwater Demand and ΔTc Control
  - The reactor receives coolant from each SG. Each SG receives the same reactor coolant temperature (Th). The amount of heat transferred in the steam generator will determine the temperature of the cold leg Tc returning to the reactor. Since we want to balance the load requirement to each SG, we try to maintain ΔTc equal to zero. (TCA – TCB)
  - Therefore, two important requirements for feedwater:
    - ΔTc near zero
    - Total feedwater flow equal to FW demand

- RC Flow Error, the second part of the ΔTc control is a feed forward circuit, which anticipates a ΔTc error upon the loss of a reactor coolant pump. If one of the reactor coolant pumps were to fail, the reactor coolant flow in that loop would drop. With a greater primary flow in one loop than the other and the same feedwater flow in both loops, the Tc of the loop with the low primary flow would be much less than that of the other loop. The only way to maintain ΔTc equal to setpoint is to re-ratio the feedwater flow in each loop to correspond to the new primary flow ratio.

- ΔTc Control and Example Operation Problems
    - Assume SG "A" has some fouling occur at time "1", T1. Assume before fouling incident station is operating at 50 percent full power with each SG handling 25 percent of total flow. Sequence of events starting at "T1": Since SG "A" has some fouling occur its heat transfer characteristics are lowered. This would create a condition where Tc for loop "A" would be higher than for loop "B" at the same flow. Therefore, ΔTc = (Tca – Tcb = error)

- For ICS automatic ΔTc control:
    - When SG "A" fouls, load SG "A" drops off, *Loop "A" Tc will rise causing a ΔTc positive error.

Although the description above is for a problem with the "A" loop, the reverse logic is made for the question. Since a pump in the "B" loop has been secured, Tc in the B loop will rise.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015	AK1.04
	Importance Rating	2.9	

K/A: Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions (Loss of RC Flow): Basic steady state thermodynamic relationship between RCS loops and S/Gs resulting from unbalanced RCS flow.

Proposed Question: RO Question # 1

Technical Reference(s): TQ-TM-104-621-C001, pg 41-43, Rev 006

Proposed References to be provided to applicants during examination: None

Learning Objective: 621-GLO-11

Question Source: Bank # IR-120214-PCO-4-Q01

Modified Bank #

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics .

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the thermodynamic relationship between RCS loops and OTSGs resulting from a RCP malfunction and unbalanced RCS flows.

The question is at the Comprehensive/Analysis cognitive level because the candidate must know the effects of FW to fail to re-ratio on FW flow, OTSG level, and fundamental heat exchanger properties when the A OTSG has lower RCS flow than than the B OTSG and how the combined effect changes overall core delta T

What MUST be known:
1. What RCP's are in the A and B RCS loops?
2. What is the change in OTSG levels for a RCP trip while at power?
3. What will be the resultant Delta Tc if feedwater fails to reratio?

2 ID: 978998 Points: 1.00

Plant Conditions (Time = 0 seconds):

- The plant is operating at 100% power.

Sequence of Events:

- Time = 10 seconds:
  - A transient occurred resulting in the following conditions:
    - Pzr level = 400 inches.
    - RCS pressure = 2285 psig.
    - Pzr temperature = 620F.
    - RC-V-1, Pressurizer Spray Valve is Open.
    - All pressurizer heater banks are ON.
  - The CRS announces entry into OP-TM-AOP-043, Loss of Pressurizer (Solid Ops Cooldown).

- Time = 40 seconds:
  - An automatic reactor trip occurs.

Based on the above information, which one of the following describes the reason for the reactor trip?

A. High RCS Thot.

B. Low RCS Pressure.

C. High RCS Pressure.

D. High Reactor Power.

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Plausible if candidate believes that Pzr heaters will raise RCS temperature faster than RCS Pressure.

B. **Incorrect.**
Plausible if candidate believes that the Pzr spray valve has an effect on the Pzr pressure while solid.

C. **Correct.**
The Pressurizer heaters are energized and the Pressurizer is solid. When the Pressurizer is solid, a 1F rise in Pzr temperature will cause an approximate rise of 100 psig in Pzr and RCS pressure and result in a High RCS pressure trip greater than 2355 psig. IAW OP-TM-AOP-0431, Loss of Pressurizer Basis Document:

1.0 DESIGN OR LICENSING BASIS REQUIREMENTS
1.1 UFSAR:
1. 4.2.2.3 Pressurizer
2. 4.3.7.b Low Temperature Over Pressurization [LTOP] Protection For events which cause the RCS pressure to increase, the pressure will increase significantly faster in a solid water system than it will in a system with steam or gas space. The RCS always operates with steam or gas space in pressurizer; no anticipated operations involve a solid water condition, other than system hydrotest.

Considering the modest rate of pressure rise (because of non-solid pressurizer) from the events and the high level alarms in the pressurizer that would normally alert the operator, it is reasonable to expect the operator to terminate the event prior to reaching an overpressurization condition.

D. **Incorrect.**
Plausible if candidate believes the higher Pzr temperature results in a higher RCS temperature and incorrectly believes the higher moderator temperature will cause reactor power to rise above RPS trip setpoint.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027	AK1.02
	Importance Rating	2.8	

K/A: Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Expansion of liquids as temperature increases.

Proposed Question: RO Question # 2

Technical Reference(s): OP-TM-AOP-0431, pg 1, Rev 003

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP043-PCO-4

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	5
	55.43	

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics .

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of expansion of liquids with a rise in temperature and its effects following a malfunction of Pressurizer pressure control.

The question is at the Comprehension/Analysis cognitive level because the candidate must be able to analyze the given conditions, determine that the plant is solid and then apply knowledge of what RPS setpoint will be violated in order to answer the question.

What MUST be known:
1. What are the requirements to be solid? 2. What is the pressure response in a solid plant? 3. What are the RPS setpoints that will be violated during solid ops?

3 **ID: 978987** **Points: 1.00**

Plant Conditions:

- The Reactor is shutdown.
- The "A" Station Battery is disconnected and out of service for maintenance.
- Battery Chargers 1C and 1E are in service.
- Battery Charger 1A is removed from service.
- A-2-7 Batt Charger 1A/1C/1E Trouble is NOT in alarm.

Event:

- A-2-7 Batt Charger 1A/1C/1E Trouble is in alarm.
- A Reactor Operator is sent to investigate locally at the Battery Chargers.

Given the above information, which one of the following:
(1) Identifies the correct battery charger indications, and
(2) Is the correct associated operational result to the "A" DC system?

A. (1) Battery Charger 1C Voltage - 0 VDC.
Battery Charger 1E Voltage - 0 VDC.
(2) "A" bank DC loads remain energized ONLY.

B. (1) Battery Charger 1C Voltage - 130 VDC.
Battery Charger 1E Voltage - 0 VDC.
(2) "C" bank DC loads remain energized ONLY.

C. (1) Battery Charger 1C Voltage - 0 VDC.
Battery Charger 1E Voltage - 130 VDC.
(2) Loss of all "A" and "C" bank DC loads.

D. (1) Battery Charger 1C Voltage - 130 VDC.
Battery Charger 1E Voltage - 130 VDC.
(2) Loss of all "A" and "C" bank DC loads.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is a correct answer since either battery charger reading 0 VDC will give the Battery Charger Trouble alarm. IAW OP-TM-MAP-A:

A-2-7, BATT CHGR 1A/1C/1E TROUBLE
- CAUSES:
  - High voltage shut down
  - Battery charger power failure
  - High voltage
  - Low voltage

Part 2 is plausible if candidate does not recognize that Battery Charger 1E is supplying "A" bank DC loads.

B. **Correct.**
Part 1 is a correct answer since either battery charger reading 0 VDC will give the Battery Charger Trouble alarm. IAW OP-TM-MAP-A:

A-2-7, BATT CHGR 1A/1C/1E TROUBLE
- CAUSES:
  - High voltage shut down
  - Battery charger power failure
  - High voltage
  - Low voltage

Part 2 is correct since battery charger 1E is supplying the "A" bank DC loads. IAW TQ-TM-104-734-C001, Vital ACDC Systems:

C. Explanation
1. Battery chargers (125 volts DC) (C & D Power Systems, Inc.)
The battery chargers take 480V AC input and convert it to 125V DC output.
a. 2 in service for each battery. Each charger charges only 1/2 of the battery, for that load group, through a direct connection to the 1A or 1B main DC distribution panels. The 1A, 1E, and 1C chargers feed the 1A station battery; the 1B, 1F, and 1D chargers feed the 1B station battery. The 1E and 1F chargers have the capability to charge either half of their associated battery through mechanically interlocked switches, and therefore are normally kept in standby as a spare for either side.

C. **Incorrect.**
Part 1 is a correct answer since either battery charger reading 0 VDC will give the Battery Charger Trouble alarm IAW OP-TM-MAP-A:

A-2-7, BATT CHGR 1A/1C/1E TROUBLE
- CAUSES:
  - High voltage shut down
  - Battery charger power failure
  - High voltage
  - Low voltage

Part 2 is plausible if candidate assumes all DC loads were lost when the battery disconnects were open for maintenance.

D. **Incorrect.**
Part 1 is an incorrect answer since either battery charger reading 0 VDC will give the Battery Charger Trouble alarm. IAW OP-TM-MAP-A:

A-2-7, BATT CHGR 1A/1C/1E TROUBLE
- CAUSES:
  - High voltage shut down
  - Battery charger power failure
  - High voltage
  - Low voltage

Part 2 is plausible if candidate assumes all DC loads were lost when the battery disconnects were open for maintenance.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	058	AK1.01	
	Importance Rating	2.8		

K/A: Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation.

Proposed Question: RO Question # 3

Technical Reference(s): MAP A, pg 17, Rev 13
TQ-TM-104-734-C001, pg 7, Rev 007

Proposed References to be provided to applicants during examination: None

Learning Objective: 734-GLO-6

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 8

55.43

(8) Components, capacity, and functions of emergency systems.

Comments:
The KA is matched because the question requires the candidates to demonstrate understanding of the battery chargers and instrumentation on a loss of DC power.

The question is at the Cognitive/Analysis cognitive level because the candidate must analyze the battery charger indications and associated loads for the chargers and predict the resultant loads lost.

What MUST be known:
1. What are the conditions that will cause MAP A-2-7 to alarm?
2. What are the loads lost when a battery charger is lost?

4 **ID: 978983** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- RPS Channel "A" is in MANUAL BYPASS for monthly surveillance testing.
- The following ICS stations are in HAND for CRD breaker testing:
  - SG/RX master.
  - Reactor Demand.
  - Both FW Loop Masters.
  - Diamond Rod Control Panel.
  - ΔTc.

Event:

- As part of the surveillance testing, CRD Breaker, CB-1A, is now open.

Given the above information, which one of the following additional events would result in an automatic reactor trip?

A. Inverter 1C failure.

B. Inverter 1D failure.

C. RPS channel "B" trip.

D. RPS channel "C" trip.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Plausible because loss of VBC will cause breaker CB-1C to trip, but all primary CRD power is still available through CB-1B and CB-1-D. IAW TQ-TM-104-641-C001, Reactor Protection System:

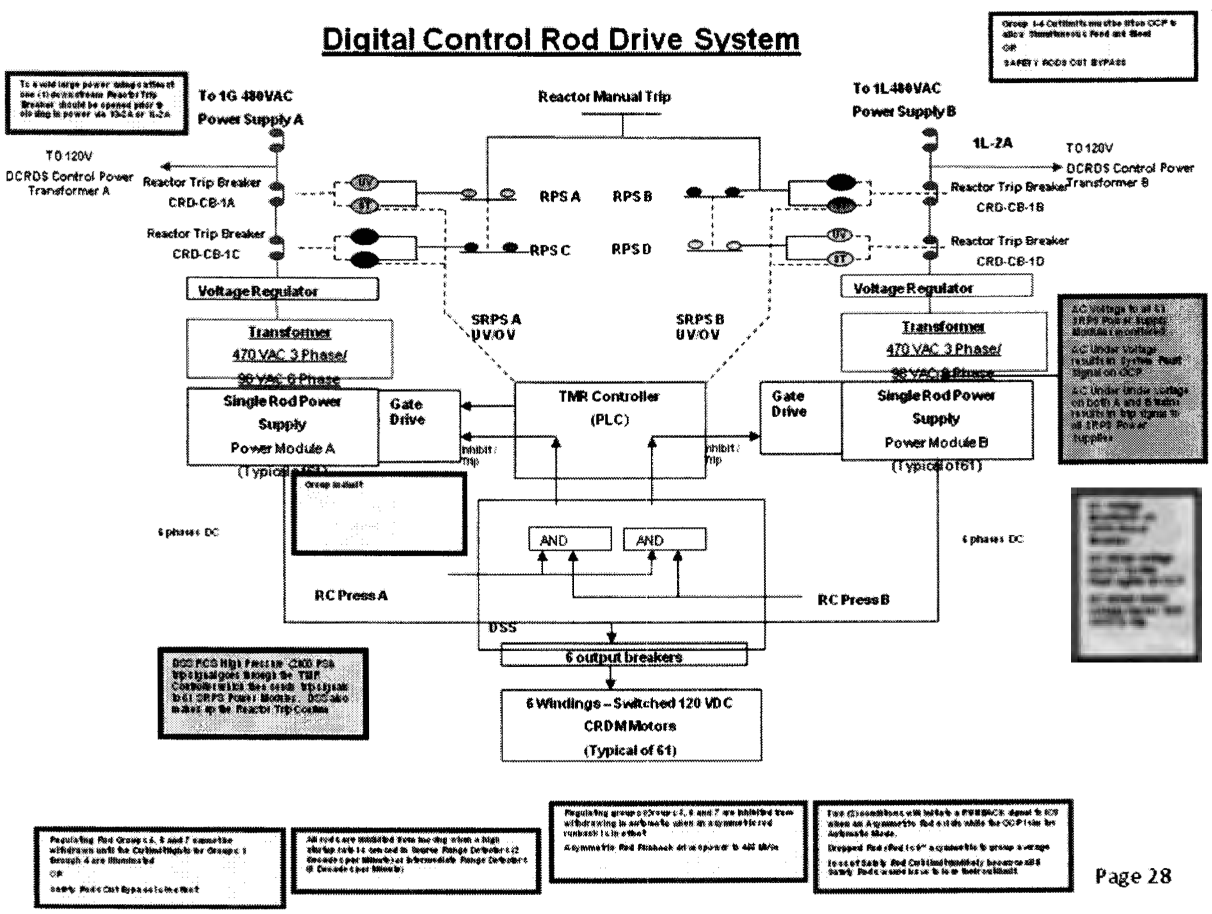
- CRD system review:
  - The CRD provides individual Single-Rod power supplies for each of the 61 control rods. There are two trains of power, fed from 1G 480 and 1L 480 volt busses.
  - The CRD consists of four circuit breakers (CRD-CB-1A, B, C and D). Power to the A train is through the “A” and “C” breakers. Power to the B train is through the “B” and “D” breakers.
  - When the power is interrupted (i.e. breakers open) the control rods will drop into the core by gravitational force, thus tripping the reactor.

B. **Correct.**
A loss of VBD will cause breaker CB-1D to trip and both power supplies to CRD will be de-energized since CB-1A is already open. IAW TQ-TM-104-641-C001, Reactor Protection System:

- CRD system review:
    - The CRD provides individual Single-Rod power supplies for each of the 61 control rods. There are two trains of power, fed from 1G 480 and 1L 480 volt busses.
    - The CRD consists of four circuit breakers (CRD-CB-1A, B, C and D). Power to the A train is through the "A" and "C" breakers. Power to the B train is through the "B" and "D" breakers.
    - When the power is interrupted (i.e. breakers open) the control rods will drop into the core by gravitational force, thus tripping the reactor.

C. **Incorrect.**
Plausible because of possible misconception that B RPS Channel trip will cause breaker CB-1B to trip.

D. **Incorrect.**
Plausible because of possible misconception that C RPS Channel trip will cause breakers CB-1C to trip.



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	007	EK2.02
	Importance Rating	2.6	

K/A: Knowledge of the interrelations between a reactor trip and the following: Breakers, relays and disconnects

Proposed Question: RO Question # 4

Technical Reference(s): TQ-TM-104-641-C001, pg 18, Rev 002

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO-8

Question Source: Bank #

Modified Bank # IR-641-GLO-8-Q02

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

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(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the interrelationship between a reactor trip and CRD breakers.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the combination of CRD breakers and then determine what type of event will cause the breakers to open.

What MUST be known:
1. What is the CRD breaker logic?
2. What combinations of CRD breakers, if open, would cause a reactor trip?

Original Question:

Plant conditions:

- Reactor power is 100%.
- B RPS Channel is in MANUAL BYPASS for monthly surveillance testing.
- The following ICS stations are in HAND for CRD breaker testing:
- SG/RX master.
- Reactor Demand.
- Both FW Loop masters.
- Diamond Rod Control Panel.
- ΔTc.
- CRD Breaker CB-1B is now open.

For the given plant conditions, which of the following events would result in an automatic reactor trip?

A. Inverter 1C failure.

B. Inverter 1D failure.

C. A-RPS channel trip.

D. C-RPS channel trip.

Answer: A

5 **ID: 978982** **Points: 1.00**

Plant Conditions:

- Due to issues with ES related equipment, the plant is in Cold Shutdown.
- "A" Train of Decay Heat Removal is in the Operating Mode.
- "B" Train of Decay Heat Removal is in the Standby Mode.
- Refueling Shutdown criteria have not been met since no fuel is scheduled to be manipulated.
- Incore temperatures are being maintained at 105°F.

Sequence of Events:

- "A" Decay Closed Cooling Water Pump, DC-P-1A, trips.
- "A" Decay Heat Removal Pump, DH-P-1A, motor stator temperature has reached Hi-1 (110°C PPC point A0737).
- Incore temperature has risen to 117°F, and continues to rise at 3°F / min.
- DH-P-1A motor current begins to oscillate erratically.

Given the above information, select the appropriate procedure and action(s) from the choices below .

A. IAW OP-TM-EOP-030, Loss of Decay Heat Removal: Place DH-P-1A in PTL, and then proceed to follow-up actions.

B. IAW OP-TM-EOP-030, Loss of Decay Heat Removal: Continue to monitor DH-P-1A operation and then proceed to follow-up actions.

C. IAW OP-TM-212-151, Shifting DH Train A From DHR Operating to DHR Standby Mode and OP-TM-212-112, Shifting DH Train B From DHR Standby to DHR Operating Mode: Shutdown "A" train of DH cooling and then place "B" train of DH cooling in service.

D. IAW OP-TM-MAP-B0105, 480v ES Motor Trip: Immediately start "B" Decay River Water Pump, DR-P-1B, "B" Decay Closed Cooling Water Pump, DC-P-1B, and "B" Decay Heat Removal Pump, DH-P-1B, and then throttle DC cooling to maintain RCS temperature.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
IAW OP-TM-EOP-030, Loss of Decay Heat Removal, the following entry conditions are met in the conditions given:

Entry Conditions: Reactor shutdown and BOTH of the following conditions:
- Decay Heat Removal is in service,
- Incore temperature increases > 10F due to an unplanned condition.

Once OP-TM-EOP-030 is entered, the conditions are met for the Immediate Manual Action to be performed. IAW OP-TM-EOP-030, Step 2.1:

If any of the following conditions exist:
- DH pump flow, discharge pressure, or motor current is varying excessively,
- DH pump bearing or stator temperatures are above HI-2 alarm limits,
- RCS water level below DH pump vortex limit,

then place DH-P-1A(B) in PTL.

Since this is the only Immediate Manual action, the Follow-up actions are then entered.

B. **Incorrect.**
IAW OP-TM-EOP-030, Loss of Decay Heat Removal, the following entry conditions are met in the conditions given:

Entry Conditions: Reactor shutdown and BOTH of the following conditions:
- Decay Heat Removal is in service,
- Incore temperature increases > 10F due to an unplanned condition.

Plausible if the candidate does not recognize erratic amps as a requirement to place the pump in PTL IAW the Immediate Manual Action step of 2.1.

C. **Incorrect.**
Although OP-TM-MAP-B0105 states to use OP-TM-212 series procedures, the entry conditions are met to enter OP-TM-EOP-030. Additionally, the proper order to swap DHR trains is to start the Standby Train and then secure the Operating Train. The order listed is reversed and therefore OP-TM-212-151, Shifting DH Train A From DHR Operating to DHR Standby Mode, conditions are not met:

Prerequisites
- DH (212)System Train A is in DHR Operating Mode IAW OP-TM-212-000, Decay Heat Removal System.
- One of the following:
  - All fuel is removed from the RV.
  - FTC Level > 344' 3" (i.e. green band).
  - RCS temperature is low and sufficient time available to allow start of standby train prior to reaching 140F (Refueling Mode) (Decay Heat low).
  - DH Train B in is DHR Operating Mode.
  - Plant Heatup is in progress and DHR is no longer required.

D. **Incorrect.**
Wrong procedure listed. OP-TM-MAP-B0105, 480v ES Motor Trip, does not give direction to restore Decay Heat Removal directly. OP-TM-MAP-B0105 directs entry into either OP-TM-212 series procedures or OP-TM-EOP-030. Plausible if the candidate believes that the actions are directly listed in the alarm response. IAW OP-TM-MAP-B0105:

> If a Decay Heat Closed Cooling Water pump or Decay Heat River Water pump has tripped, then PERFORM the following:
> a. If Decay Heat removal operations are in progress, then swap Decay Heat removal loops IAW OP-TM-212, 533 and 543 series procedures.
> b. If Decay Heat removal operations are in progress, then evaluate conditions for entry into OP-TM-EOP-030, Loss of Decay Heat Removal.
> c. If HPI (Makeup) pumps are required to be operable, then PERFORM OP-TM-543-439(440), Swapping MU-P-1A(C) Cooling to NS, for the affected train.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	025	AK2.02	
	Importance Rating	3.2		

K/A: Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: LPI or Decay Heat Removal/RHR pumps.

Proposed Question: RO Question # 5

Technical Reference(s): OP-TM-EOP-030,pg 1, Rev 004

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP030-PCO-4

Question Source: Bank #
Modified Bank # IR-EOP030-PCO-4-Q04
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 7

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(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the interrelationship between the loss decay heat removal pumps and loss of the Decay Heat Removal system.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must determine the correct procedure to implement along with the applicable action to take within the procedure.

Original Question: IR-EOP030-PCO-4-Q04

Initial conditions:
- The plant is in cold shutdown, with “B” loop of DH cooling in service.
- Incore temperatures are being maintained at 110°F.

Event:
- DC-P-1B trips.
- DH-P-1B motor stator temperature has reached Hi-1 (110 ^{0}C PPC point A0737)

Subsequent Conditions:
- Incore temperature have risen to 119°F, and continues to rise at 4°F / min.
- DH-P-1B motor current begins to oscillate erratically.

Select the appropriate procedure and action(s) from the choices below.

A. Enter OP-TM-EOP-030, "LOSS OF DECAY HEAT REMOVAL", place DH-P-1B in PTL, and proceed to follow-up actions.

B. Enter OP-TM-EOP-030, "LOSS OF DECAY HEAT REMOVAL", place BOTH DH-P-1B and DH-P-1A in PTL, and proceed to follow-up actions.

C. Immediately start DR-P-1A, DC-P-1A and DH-P-1A per "MAP B" alarm response for panel B-1-5, "480v ES motor trip", then throttle DC cooling to maintain RCS temperature.

D. Shutdown "B" train of DH cooling per OP-TM-212-152, "SHIFTING DH TRAIN B FROM DHR OPERATING TO DHR STANDBY MODE" and then place "A" train of DH cooling in service per OP-TM-212-111, "SHIFTING DH TRAIN A FROM DHR STANDBY TO DHR OPERATING MODE."

Answer: A

6 **ID: 978979** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- All "A" OTSG level indicators are indicating actual level.

Event:

- A steam rupture has occurred in the Reactor Building.
- Steam is impinging on the "A" OTSG full range reference leg.

Given the above information, the "A" OTSG **Full Range** level indication will be ___(1)___ than actual "A" OTSG level and the **Operating / Startup Range** level indication will be ___(2)___ than actual "A" OTSG level.

A. (1) lower
(2) lower

B. (1) lower
(2) higher

C. (1) higher
(2) lower

D. (1) higher
(2) higher

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
See explanation under the correct answer. Plausible if candidate believes all level indicators are temperature compensated.

B. **Incorrect.**
See explanation under the correct answer. Plausible if the candidate confuses which OTSG level indicators are temperature compensated.

C. **Correct.**
The examinee must know the following information to answer this question:

- The temperature compensation for reference leg density is located on the full range level reference leg.
- The full range instrument is NOT temperature compensated. Therefore, all three instruments are affected but for different reasons, two due to inaccurate calibration, and one due to uncompensated density changes.

IAW TQ-TM-104-644-C001, Heat Sink Protection System:

XIII. 5. "Reference Leg temperature instruments used for compensating Operating and Startup Range level instruments are located on the Full Range reference leg.

a. If localized heating of the reference leg or the reference leg RTDs were to occur due to a small steam leak or a small valve leak in the Full Range reference leg, then all OTSG level instruments on that OTSG will be invalid.
1) The Full Range instrument is not compensated for changes in reference leg temperature, therefore a hot reference leg will make it indicate higher than actual level
2) The Startup and Operating Range level DP transmitter output would be adjusted for a reference leg temperature much higher than actual temperature. This will make the indicated level lower than actual.
b. If this were to occur during power operation it will appear as if the Operating and Startup Range levels were decreasing
1) The rate will depend on the rate of heat up of the reference leg
2) The difference in magnitude could be approximately 30 to 40 inches on Startup Range and approximately 10% on the Operating Range"

D. **Incorrect.**
See explanation under the correct answer. Plausible if the candidate believes that none of the OTSG level indicators are temperature compensated.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	040	AK2.02
	Importance Rating	2.6	

K/A: Knowledge of the interrelations between the Steam Line Rupture and the following: Sensors and detectors.

Proposed Question: RO Question # 6

Technical Reference(s): TQ-TM-104-644-C001, pg 74 Rev 002

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-051-PCO-2

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	7
	55.43	

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to demonstrate understanding of the relationship between sensors and detectors and a steam line rupture.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the effects of reference leg heating on indicated level and must also know that the OTSG Full Range level indication is not temperature compensate whereas the Startup range indication is temperature compensated and thus have opposite effects based on the conditions.

What MUST be known:
1. Which OTSG level indicators are temperature compensated?
2. What will Startup Range OTSG Level indication be during a Steam Rupture?
3. What will Operating Range OTSG Level indication be during a Steam Rupture?

7 **ID: 978976** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- The reactor has tripped and safety injection has actuated.
- The RCS has a stuck open Pressurizer Safety valve.
- The RCS has rapidly depressurized to saturation conditions during the cooldown.
- OP-TM-EOP-002, Loss of 25 Degrees F Subcooling Margin, has been successfully completed in its entirety through Step 3.14 RNO, "GO TO OP-TM-EOP-006".
- Pressurizer level initially dropped below zero and now has risen to 46 inches and is rising rapidly.

Given the above information, which one of the following characterizes the relationship between indicated pressurizer level and RCS inventory, as well as the major reason for these conditions?

A. Level is NOT an accurate indication of inventory because reactor vessel voiding may result in a rapidly rising pressurizer level.

B. Level is an accurate indication of inventory because RCP flow would sweep any voids from the RCS to the pressurizer steam space and out the safety.

C. Level is NOT an accurate indication of inventory because the cold calibrated pressurizer level channels indicate high during high temperature, low pressure conditions.

D. Level is an accurate indication of inventory because voiding would occur first in the pressurizer steam space due to the higher temperature of the pressurizer walls.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
OPEX - Unit 2 Accident.

IAW TMI-1 UFSAR, CHAPTER 14:

c) Pressurizer Safety Valve Stuck Open: The small break resulting from the failure of a pressurizer code safety valve needs no special analysis because of three main points: a) challenges to these valves are infrequent events, b) the valves are designed to relieve and reseat without leakage, and c) the leak that would result if one (or both) of these valves stuck open is bounded by already existing analyses.

The general phenomena observed for the PORV stuck open cases would be observed for this break. However, since the area of the pressurizer safety valve is larger, the phenomena would occur earlier than that seen for the PORV breaks. Generally, the system behavior can be characterized by:

- A rapid system depressurization due to steam relief via the safety valve. This will result in reactor scram and ESAS actuation.
- The indicated pressurizer level will initially increase due to the break. After reactor scram, the pressurizer level decreases due to system contraction. Following saturation of the hot legs, there will be an insurge into the pressurizer resulting in the pressurizer going solid.
- After the pressurizer is filled by a two-phase mixture, a low quality mixture will be discharged through the valve. This will result in a large increase in the leak flow rate.

IAW TMI-1 UFSAR, CHAPTER 01

1.3.2.27 Inadequate Core Cooling Instrumentation: Additional and modified instrumentation for detection of inadequate core cooling includes connecting incore thermocouples to the plant computer, installing a diverse and separate system to the plant computer for monitoring incore coolant temperature, providing a wide range reactor outlet temperature measurement, and redundant Control Room indication of reactor coolant saturation margin, and installation of a Reactor Coolant Inventory Trending System (RCITS) to monitor coolant inventory in the reactor vessel head and hot legs.

IAW TMI-1 UFSAR, CHAPTER 07

d) Reactor Coolant Inventory Trending System (RCITS): The RCITS provides a means for the Control Room operator to monitor the void content of the reactor coolant system (RCS) when the reactor coolant pumps (RCP) are running, and the water inventory of the RCS when the RCPs are off. The RCS void fraction is calculated by the plant computer from RCP power using an empirical algorithm which was developed to yield the desired relationship between RCP power and RCS void fraction at the pump suction. The void fraction is calculated for each of the four RCPs displayed in the Control Room via the plant computer. The water level trending subsystem consists of two independent instrument loops to measure water level in the hot leg "candy canes" and two identical instrument loops to measure water level in the reactor vessel head above the RCS hot legs. The water level measurements are accomplished using differential pressure transmitters and are displayed via the plant computer.

B. **Incorrect.**
Incorrect because RCPs would be secured for the above saturated conditions.

C. **Incorrect.**
Although it is true that the cold calibrated pressurizer level channels will indicate higher, the the level instruments are density compensated to prevent this and therefore, the contribution to a false level is not significant in this event.

D. **Incorrect.**
Incorrect because voiding would occur in the top of the reactor vessel head first.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	008	AK3.01
Importance Rating	3.7	

K/A: Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: Why PZR level may come back on scale if RCS is saturated.

Proposed Question: RO Question # 7

Technical Reference(s):
UFSAR Chapter 01, pg 1.3-5, Rev 21
UFSAR Chapter 07, pg 7.3-11, Rev 21
UFSAR Chapter 14, pg 14.2-37,38, Rev 21

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO-13

Question Source:
Bank # ANO2 2009 NRC RO/SRO Exam
Modified Bank #
New

Question History: Last NRC Exam: 2009 ANO2

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

10 CFR Part 55 Content:
55.41 5
55.43

Comments:
The KA is matched because the question requires the candidates to demonstrate analysis of conditions and then apply knowledge of the reason pressurizer level comes back on scale following a pressurizer vapor space accident and the RCS becoming saturated.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze conditions involving a rapid rise in pressurizer level following the pressurizer emptying and the RCS becoming saturated is indication of voiding in the core.

What MUST be known:
1. What is the condition of the RCS when given certain parameters?
2. What is the FSAR analysis for a stuck open Pressurizer Safety Valve?
3. What does Pressurizer level change mean during a given scenario?
4. What were the causing factors involved with the TMI-2 accident?

8 **ID: 978966** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- A fuel movement is in progress within the fuel handling building.
- A liquid radwaste release is in progress with WDL-V-257, WDL-P-14A/B Discharge to MDCT/River, open.

Event:

- A loss of Inverter "C" occurs.

Given the above information, which one of the following actions would be correct?

A. Secure the fuel movement due to the loss of RM-A-8.

B. Stop all radwaste processing evolutions due to loss of RM-L-6.

C. Stop all radwaste processing evolutions due to loss of RM-A-7.

D. Secure the fuel movement due to the loss of RM-A-14 Remote Digital readout in the CR.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
RM-A-8 will deenergize but there is no requirement to secure the fuel movement as a result. Plausible if the operator confuses RM-A-8 with RM-A-14. Both are powered from VBC but loss of RM-A-8 does not require securing fuel movement. IAW OP-TM-AOP-017, Loss of VBC:

- ATTACHMENT 1 - EFFECTS OF LOSS OF VBC
    - RM-A-8 trips AH-E-10 and AH-E-11, closes WDG-V-47, and starts the MAP-5 Iodine Sampler.

IAW OP-TM-MAP-C0101, Radiation level Hi:

RM-A-8 - Aux. and Fuel Handling Building Exhaust Duct
4.0 MANUAL ACTIONS REQUIRED
4.1 If any of the following in alarm:
– RM-A-4
– RM-A-6
– RM-A-7,
then INITIATE manual action on appropriate alarm response.
4.2 NOTIFY Radiation Protection.
4.3 If possible, then ISOLATE the source of radioactive release.
4.4 IAAT high alarm is Lit, then perform the following:
1. ENSURE AH-E-10 and AH-E-11 are Shutdown.
2. ENSURE WDG-V-47 is Closed.
3. REQUEST SM to evaluate Emergency Action Levels (EALs).
4. NOTIFY Chemistry to sample RM-A-8 to determine source term.
5. NOTIFY Radiation Protection.
4.5 IAAT RM-A-8 indication is off-scale high, then DE-ENERGIZE the detector.

4.6 INITIATE an IR and RECORD the following:
– Duration of release
– Flow or volume data

B. **Incorrect.**
Plausible if the operator confuses loss of VBC with loss of VBD because a Loss of RM-L-6 will cause WDL-V-257 to Close. RM-L-6 is powered from VBD. IAW OP-TM-AOP-018, Loss of VBD:

- ATTACHMENT 1 - EFFECTS OF LOSS OF VBD
    - Loss of power to RM-L-6 will close WDL-V-257.

C. **Incorrect.**
Plausible if the operator confuses RM-L-7 with RM-A-7 but a Loss of RM-A-7 will secure waste gas release and is powered from VBD. IAW OP-TM-AOP-017, Loss of VBC:

- ATTACHMENT 1 - EFFECTS OF LOSS OF VBC
    - RM-L-7 closes WDL-V-257.

D. **Correct.**
IAW OP-TM-AOP-017, Loss of VBC:

- ATTACHMENT 3 - INOPERABLE INSTRUMENTATION ON LOSS OF VBC
  - RM-A-14 (Remote Digital Readout), ESF

3.3 If fuel movement is in progress in the Spent Fuel Pool, then STOP fuel movement.

IAW OP-TM-AOP-0171, Loss of VBC BAsis Document:

Step 3.3 This step provides guidance to stop fuel movement if VBC de-energizes. Loss of VBC de-energizes the RM-A-14 Remote Digital Readout in the Control Room.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	057	AK3.01
	Importance Rating	4.1	

K/A: 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.

Proposed Question: RO Question # 8

Technical Reference(s): OP-TM-AOP-017, pg 15, Rev 006
OP-TM-AOP-0171, pg 4, Rev 004

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-017-PCO-4

Question Source: Bank # IR-AOP-017-PCO-4-Q01

Modified Bank #

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	11
	55.43	

(11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:
The KA is matched because the question requires the candidate to demonstrate a knowledge of actions contained within the Abnormal Operating Procedure for a loss of a Vital AC bus.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the effected equipment for loss of VBC, the procedure to be entered and the specific component effected and the actions that must be taken to mitigate the loss.

What MUST be known:
1. What is the effect of a loss of Inverter "C"? 2. What loads are lost on a Loss of Vital Bus "C"? 3. What action must be taken IAW OP-TM-AOP-017 and why?

9 **ID: 978964** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- The plant has experienced a complete loss of feedwater.
- The Crew has implemented the OP-TM-EOP-004 Lack of Primary to Secondary Heat Transfer (LOHT) .
- Currently:
    - EF-P-2A & EF-P-2B, Motor Driven Emergency Feedwater Pumps "A" / "B", are NOT available.
    - EF-P-1, Steam Driven Emergency Feedwater Pump, has just been restored.
    - The crew has just reached step 3.15 which reads: "**REDUCE** OTSG Pressure so that secondary Tsat is 90 to 100°F lower than incore thermocouple temperature."

Given the above information, the basis for step 3.15 of OP-TM-EOP-004 is to:

A. Reduce OTSG tube stresses.

B. Reduce the RCS hot leg steam voids.

C. Establish a Delta - T to enhance the strength of the heat sink.

D. Establish a base temperature for the OTSG while maintaining enough steam for EF-P-1.

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**

OP-TM-EOP-004, Lack of Primary to Secondary Heat Transfer (LOHT) does not mention OTSG tube stress as a basis for any action. If OTSG tube stresses were to be reduced, the actions to take involve adjusting RCS cooldown rates/heatup rates and Subcooling Margin. IAW OP-TM-EOP-010, Emergency Procedure Rules Guides and Graphs, Guide 14, Tube-To-Shell Delta-T Limit/Control, states:

1. VERIFY OTSG tube-to-shell differential temperature (TSDT) (as indicated on PPC points C4015 and C4016) is above (less negative) the tensile limit, –70 °F.
   RNO: PERFORM actions in order listed until TSDT is controlled above limit:
   1. MINIMIZE SCM.
   2. REDUCE cooldown rate, as necessary, HOLD, or RAISE RCS temperature.

2. VERIFY OTSG tube-to-shell differential temperature (TSDT) (as indicated on PPC points C4015 and C4016) is below the compressive limit, +60 °F.
   RNO: PERFORM actions in order listed until TSDT is controlled below limit:
   1. REDUCE heatup rate, as necessary, HOLD, or LOWER RCS temperature.

Plausible since the OTSG Pressure is the item being manipulated and OTSG Tube stress is the distractor.

B. **Incorrect.**

This is basis for raising OTSG level to 75 to 85% Operate Range with EFW. IAW OP-TM-EOP-004, Lack of Primary to Secondary Heat Transfer (LOHT):

3.16 RAISE OTSG level to 75 to 85% Operate Range with EFW.
RNO: RAISE OTSG level to 75 to 85% Operate Range with MFW.

IAW OP-TM-EOP-0041, Lack of Primary to Secondary Heat Transfer Basis Document:

Step 3.16 Feedwater is available, primary to secondary heat transfer has not been established and SCM > 25F. The step increases the available area for primary to secondary heat transfer. If steam voids in the RCS hot legs have blocked natural circulation flow, establishing OTSG levels at the loss of subcooled margin setpoint will provide cooling and subsequent collapse of the steam void, allow refill of the hot leg and reestablish conditions favorable for natural circulation. The use of EFW as a preferred source is due to EFW injection directly on the OTSG tube bundle at sufficient height, however if unavailable, the use of main feedwater is a prudent alternative.

Plausible if the candidate confuses the basis for the different steps within OP-TM-EOP-004.

C. **Correct.**
IAW OP-TM-EOP-004, Lack of Primary to Secondary Heat Transfer (LOHT):

> 3.15 REDUCE OTSG Pressure so that secondary Tsat is 90 to 100°F lower than incore thermocouple temperature.

IAW OP-TM-EOP-0041, Lack of Primary to Secondary Heat Transfer Basis Document:

> Step 3.15 The step intent is to increase the relative strength of the OTSG heat sink. This step assumes that an adequate supply of feedwater exists, but heat transfer has not yet been established. The strength of the heat sink is enhanced by lowering the OTSG pressure to provide about a 100 °F temperature difference between incore thermocouple temperature and OTSG saturation temperature.

D. **Incorrect.**
IAW OP-TM-EOP-004, Lack of Primary to Secondary Heat Transfer (LOHT), Caution prior to step 3.13:

> CAUTION: If EF-P-2A and EF-P-2B and auxiliary steam to EF-P-1 are unavailable, then to prevent loss of EFW, do not lower steam pressure below 150 psig.

IAW OP-TM-EOP-0041, Lack of Primary to Secondary Heat Transfer Basis Document:

> The caution statement warns of potential loss of the steam driven emergency feedwater pump by reducing OTSG pressure. If the turbine driven pump is not being relied upon than OTSG pressure should be lowered as required. If EF-P-1 is the only feed source and MS is only steam supply, then OTSG pressure should not be reduced below 150 psig. A steam supply pressure above 150 psig ensures the full capability of EF-P-1 is maintained.

Plausible if the candidate confuses the basis for the different steps within OP-TM-EOP-004.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	E04	EK3.3	
	Importance Rating	4.2		

K/A: Knowledge of the reasons for the following responses as they apply to the (Inadequate Heat Transfer): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

Proposed Question: RO Question # 9

Technical Reference(s): OP-TM-EOP-004, pg 9,Rev 10
OP-TM-EOP-0041, pg 7, Rev 005

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP004-PCO-4

Question Source:	Bank #	IR-EOP004-PCO-4-Q03
	Modified Bank #	
	New	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	5
	55.43	

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics .

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of manipulation of controls required to obtain desired operating results during emergency situations with regards to inadequate heat transfer and the reason for the manipulations.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall the basis for the action taken.

What MUST be known:
1. What is the basis for actions within OP-TM-EOP-004?

10 **ID: 906765** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- The feeder breaker for 1S ES bus has opened due to an overcurrent fault on the 1S ES bus.

Sequence of Events:

- The plant has suffered a Loss of Nuclear Services Component Cooling.
- OP-TM-AOP-031, Loss of Nuclear Services Component Cooling, has been entered.
- Currently:
  - Reactor is tripped per OP-TM-EOP-001, Reactor Trip.
  - "A" Makeup Pump, MU-P-1A, is operating.
  - "A" Nuclear Services Closed Cooling Pump, NS-P-1A, is operating.
  - The breaker for "B" Nuclear Services Closed Cooling Pump, NS-P-1B, on the 1P ES bus has tripped and appears to be damaged.
    - Damage is limited to the NS-P-1B breaker ONLY.
  - "A" and "B" Secondary River Water Pumps, SR-P-1A and SR-P-1B, are operating.
  - NS surge tank, NS-T-1, level (NS-LI-800 / 801) indicates 1.8 ft. and steady.
  - NS cooler outlet temperature (A0330) is 101°F and rising slowly.
  - All RCPs are secured.
  - NS pump discharge pressure (NS-PI-169) is 70 psig and steady.

Given the above information, what action is required to be taken?

A. PLACE all Nuclear Service pumps in PTL.

B. GO TO AOP-005, River Water Systems Failures.

C. INITIATE OP-TM-541-444, Swap NS-P-1B To Alternate Power Supply.

D. INITIATE OP-TM-541-901, Secondary River to Nuclear River Cross-Tie.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
The conditions given in the stem indicate NS surge tank level to be 1.8 ft and steady and NS pump discharge pressure to be 70 psig and steady. Plausible if student thinks low surge tank level or low discharge pressure requires securing pumps, however level and pressure are not lowering and above 60# and 1.6 ft. IAW OP-TM-AOP-031, Loss of Nuclear Services Component Cooling, step 3.4:

3.4 IAAT either condition exists:
- NS surge tank level <1.6 ft and lowering
- NS pump discharge pressure <60 psig

then perform the following:
- Place all Nuclear Service pumps in PTL
- Close IA-V-49
- Open NS-V-100

B. **Incorrect.**
This would be the correct step if no Nuclear River or Secondary River Water pumps were operating/available. The conditions given in the stem indicate that Secondary River pumps 1A and 1B are operating. Plausible if student thinks high temps require AOP-005, however not all SR and NR pumps must be tripped. IAW OP-TM-AOP-031, Loss of Nuclear Services Component Cooling, step 3.13:

3.13 IAAT no Nuclear River or Secondary River pumps are operating or available to be started, then GO TO AOP-005 "River Water Systems Failures".

C. **Incorrect.**
The purpose of OP-TM-541-444, Swap NS-P-1B to Alternate Power Supply, is to power NS-P-1B from the 1S 480V bus. The conditions given in the stem indicate that the 1S 480V bus has failed open. Plausible if student thinks getting two NS pumps is possible per AOP-031 3.10 RNO, however the 1S ES bus is not powered according to the stem. IAW OP-TM-541-444:

1.0 Purpose
1.1 Provide guidance for transferring NS-P-1B power supply from the 1P 480V ES Bus to the alternate power supply located on the 1S 480V Bus.

D. **Correct.**
The conditions given in the stem indicate that Nuclear Services pump 1A is operating and that NS cooler outlet temperature is 101F. With Temp > 100F initiate OP-TM-541-901. IAW OP-TM-AOP-031, Loss of Nuclear Services Component Cooling, Step 3.9:

3.9 If a NS pump is operating, and NS cooler outlet temperature >100F, then initiate OP-TM-541-901 "Secondary River to Nuclear River Cross-Tie".

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	026	AA1.03
	Importance Rating	3.6	

K/A: Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: SWS as a backup to the CCWS.

Proposed Question: RO Question # 10

Technical Reference(s): OP-TM-AOP-031, pg 5,Rev 004

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-031-PCO-4

Question Source:	Bank #	IR-AOP-031-PCO-4-Q05
	Modified Bank #	
	New	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	10
	55.43	

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of operation of Secondary River Water as a backup to the Nuclear Closed Cooling Water System as a cooling medium in place of the Nuclear River Water System.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze and assess the conditions and determine the correct action and procedure to mitigate condition.

What MUST be known:
1. What are the power supplies of NSCCW pumps? 2. What are the plant conditions required for various routing steps within OP-TM-AOP-031? 3. What is the status of Nuclear Services Pump Discharge, Surge Tank Level, and Cooler Outlet Temperature?

11 **ID: 978952** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- A Loss of of Offsite Power occurred.
- EG-Y-1A, "A" Emergency Diesel Generator, failed to start.
- A fault has occurred on the 1E 4160V Bus.
- EG-Y-4, SBO Diesel Generator, failed to start.
- The CRS is in OP-TM-AOP-020, Loss of Offsite Power, Section 4.0, Station Blackout.
- The CRS has entered OP-TM-AOP-023, "A" DC System Failures and OP-TM-AOP-024, "B" DC System Failures.
  - "A" and "C" battery voltages are 127VDC and slowly lowering.
  - "B" and "D" battery voltages are 130VDC and slowly lowering.
- A report comes in to the Control Room that Offsite power has been restored via Middletown 1091.

After the report is received, the first bus to be energized will be the ___(1)___ bus IAW ___(2)___.

A. (1) 1D 4160V
(2) OP-TM-AOP-020, Attachment 2, Restoration of Offsite Power, Section 4.5, Energize 1D 4160V bus from offsite power

B. (1) 1A 6900V
(2) 1107-1, Normal Electrical System, Section 5.1, Operating a Bus Feeder Breaker (7KV) Locally at the Breaker.

C. (1) 1A 6900V
(2) OP-TM-AOP-020, Attachment 2, Restoration of Offsite Power, Section 4.3, Energize 1A & 1B BOP Buses From Offsite Power

D. (1) 1D 4160V
(2) 1107-2A, Emergency Electrical - 4KV and 480 Volt, Section 3.3, Local Operation for Closing of the 4KV Feeder Breakers.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
Parts 1 and 2 are correct. OP-TM-AOP-020, Section 4.0, Station Blackout, routing has a hold point to where nothing else may be procedurally performed unless one of the following Carry-Over Step conditions are met (of which the first two are the only ones that reference a power source being restored):

- Dispatcher reports that off site power is available, Step 3.10
- 1D or 1E 4160V bus is energized, Step 4.1
- SCM < 30°F, Step 4.8
- RCS pressure < 1750 psig, Step 4.9
- Condenser vacuum < 10 in Hg vac, Step 4.12.3
- EF-V-30 or MS-V-4 control is inadequate due to low air pressure Step 4.13

Since Offsite power has been reported to be available, Step 3.10 is applicable. The candidate will need to recognize that although Section 4.0 is the current section, a Carryover from section 3.0 is still applicable. Step 3.10 states:

> IAAT dispatcher reports that off site power is available, then perform Attachment 2, Restoration of Offsite Power.

Attachment 2, Step 4.2.8 (which is after restoration preparations have been performed and both Auxiliary Transformers have been restored), states:

> If neither 1D or 1E 4160V bus is energized, then perform Section 4.5 or 4.6, and then Section 4.3.

B. **Incorrect.**
Part 1 is incorrect. plausible if the candidate believes that the order for restoration of busses goes from the largest bus to the smallest bus.

Part 2 is incorrect. If the candidate believes that OP-TM-AOP-023, "A" DC System Failure, and OP-TM-AOP-024, "B" DC System Failure have been entered due to a loss of DC, then the candidate will believe that the switchgear breakers will need to be operated locally. IAW OP-TM-AOP-0201, Loss of Station Power Basis Document, Step 4.10:

> OP-TM-AOP-023 and OP-TM-AOP-024 are entered IAW OP-TM-AOP-020, Loss of Station Power, to drive the operators to shutdown large DC loads, prolonging the life of the station batteries.

Also plausible if the candidate does not recognize that OP-TM-AOP-020 fully addresses the bus restoration and believes that 1107-1 will be the controlling procedure.

C. **Incorrect.**
Part 1 is correct.
Part 2 is incorrect. If the candidate believes that OP-TM-AOP-023, "A" DC System Failure, and OP-TM-AOP-024, "B" DC System Failure have been entered due to a loss of DC, then the candidate will believe that the switchgear breakers will need to be operated locally. IAW OP-TM-AOP-0201, Loss of Station Power Basis Document, Step 4.10:

> OP-TM-AOP-023 and OP-TM-AOP-024 are entered IAW OP-TM-AOP-020, Loss of Station Power, to drive the operators to shutdown large DC loads, prolonging the life of the station batteries.

Also plausible if the candidate does not recognize that OP-TM-AOP-020 fully addresses the bus restoration and believes that 1107-2A will be the controlling procedure.

D. **Incorrect.**
Part 1 is incorrect. plausible if the candidate believes that the order for restoration of busses goes from the largest bus to the smallest bus.

Part 2 is the correct procedural guidance for the 1A 6900V bus and further makes the distractor plausible since it comes first in numerical order. Section 4.3 does come numerically before Section 4.5, but there is a procedure step at the end of Section 4.2 that states:

> If neither 1D or 1E 4160V bus is energized, then PERFORM Section 4.5 or 4.6, and then Section 4.3

Plausible if the candidate is not familiar with the flow and intent of the procedure and simply follows in numerical order.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	055	EA1.07
	Importance Rating	4.3	

K/A: Ability to operate and monitor the following as they apply to a Station Blackout: Restoration of power from offsite.

Proposed Question: RO Question # 11

Technical Reference(s): OP-TM-AOP-020, pg 5,20,32,35, Rev 017

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-020-PCO-4

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam:

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

10 CFR Part 55 Content:	55.41	10
	55.43	

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the ability to restore power from offsite following a Station Blackout.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must determine the correct actions based on knowledge of various procedures related to the event.

What MUST be known:
1. What is the first bus to be restored procedurally after a Loss of Offsite power?
2. What is the procedure to be implemented when restoring busses after a Loss of Offsite Power?

12 **ID: 978951** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- EG-Y-1B, "B" Emergency Diesel Generator, is in parallel with Offsite Power.

Sequence of Events:

- A Loss of Offsite Power (LOOP) has occurred.
- A fault on the 1P 480V bus has deenergized the bus.
- Pressurizer level is 100 inches and steady.
- The CRS has ordered that Pressurizer heaters be energized IAW OP-TM-220-901, Emergency Power Supply for Pressurizer Heaters.

Based on the above information, Group ___(1)___ heaters will be energized and, assuming no operator actions take place involving the Emergency Diesel Generators, the frequency of the associated Emergency Diesel Generator will ___(2)___.

A. (1) 8
(2) remain the same

B. (1) 8
(2) operate at a lower value

C. (1) 9
(2) remain the same

D. (1) 9
(2) operate at a lower value

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. IAW 1107-4, Electrical Distribution Panel Listing, Group 8 heaters are powered from the 1P 480V bus, Group 9 heaters are powered from the 1S 480V bus. Plausible if the candidate believes that Group 9 heaters are powered from 1P 480V bus.

Part 2 is incorrect. Plausible if candidate believes that Frequency droop switches to 0% (isynchronous) automatically on loss of off-site power.

B. **Incorrect.**
Part 1 is incorrect. IAW 1107-4, Electrical Distribution Panel Listing, Group 8 heaters are powered from the 1P 480V bus, Group 9 heaters are powered from the 1S 480V bus. Plausible if the candidate believes that Group 9 heaters are powered from 1P 480V bus.

Part 2 is correct. With the EDG initially in parallel with off-site power, both Frequency and Voltage droop are in effect. When Off-Site power is lost frequency and voltage will lower based on current running loads powered by the diesel. IAW OPM A-04, Diesel Generator and Auxiliary Equipment:

> If the diesel is operating in parallel with the system and off-site poer lost, the diesel generator breaker will remain closed; however a reduction in frequency and voltage will take place due to the speed droop and voltage droop.

C. **Incorrect.**
Part 1 is Correct. IAW 1107-4, Electrical Distribution Panel Listing, Group 8 heaters are powered from the 1P 480V bus, Group 9 heaters are powered from the 1S 480V bus.

Part 2 is incorrect. Plausible if candidate believes that Frequency droop switches to 0% (isynchronous) automatically on loss of off-site power.

D. **Correct.**
Part 1 is correct. IAW 1107-4, Electrical Distribution Panel Listing, Group 8 heaters are powered from the 1P 480V bus, Group 9 heaters are powered from the 1S 480V bus.

Part 2 is correct. With the EDG initially in parallel with off-site power, both Frequency and Voltage droop are in effect. When Off-Site power is lost frequency and voltage will lower based on current running loads powered by the diesel. IAW OPM A-04, Diesel Generator and Auxiliary Equipment:

> If the diesel is operating in parallel with the system and off-site power lost, the diesel generator breaker will remain closed; however a reduction in frequency and voltage will take place due to the speed droop and voltage droop.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	056	AA1.03
	Importance Rating	3.2	

K/A: Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power: Adjustment of ED/G load by selectively energizing PZR backup heaters.

Proposed Question: RO Question # 12

Technical Reference(s): 1107-4, pg 32,34, Rev 225
OPM A-04, pg 8, Rev 012

Proposed References to be provided to applicants during examination: None

Learning Objective: 861-GLO-8

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	8
	55.43	

(8) Components, capacity, and functions of emergency systems.

Comments:
The KA is matched because the question requires the candidates to demonstrate ability to monitor the emergency diesel generator load adjustment when selected pressurizer backup heaters are energized.

The question is at the Comprehension/Analysis cognitive level because the candidate must know the emergency power supplies for the Pzr heaters and analyze a sequence of events to determine the effect on the Emergency Diesel Generator speed and frequency when loads are started.

What MUST be known:
1. What is the status of the Emergency Diesel Generator when in parallel with offsite power?
2. What is the status of the Emergency Diesel Generator when in parallel and offsite power is lost?
3. What is the backup emergency power supplies to the Pressurizer heaters?

13 **ID: 951739** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- MU-V-17, Normal MU to RCS Control Valve, has failed closed.

Event:

- Time = 1715:
  - A 65 gpm small break LOCA occurs.
  - URO reports Pressurizer level is 210 inches and LOWERING.
  - Letdown flow is 85 gpm.
  - Seal injection flow is 37 gpm.
  - Seal Leakoff flow is 12 gpm.
  - MU-V-217, High Capacity Normal MU Valve, bypass flow indicates 20 gpm.

Given the above information and assuming no operator action, what is the earliest time that the Pressurizer will be empty (0 inches)?

A. 1745

B. 1815

C. 1818

D. 1920

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
37 gpm (Seal Injection) + 20 gpm (Makeup) - 12 gpm (Seal Leakoff) - 85 gpm (Letdown) - 65 gpm (RCS Leak) = -105 gpm

105 gpm (Total Leakrate) / 15 gallons per inch (Pzr Volume Calculation) = 7 inches per minute

210 inches (Current Pzr Level) / 7 inches per minute (Pzr Equivalent Leak Rate) = 30 minutes

1715 + 30 minutes = 1745

B. **Incorrect.**
Plausible if the candidate confuses the Pzr (15 gallons / inch) with the Makeup Tank (30 gallons / inch)

105 gpm / 30gallons per inch = 3.5 inches per minute
210 inches / 3.5 inches per minute = 60 minutes
1715 + 60 minutes = 1815

C. **Incorrect.**
Plausible if the candidate only takes into account the RCS leakrate and does not factor in the other methods that water is entering and exiting the system.

50 gpm / 15 gallons per inch = 3.33 inches per minute
210 inches / 3.33 inches per minute = 63 minutes
1715 + 63 minutes = 1818

D. **Incorrect.**
Plausible if the candidate only takes into account the RCS leakrate and does not factor in the other methods that water is entering and exiting the system and confuses the Pzr (15 gallons / inch) with the Makeup Tank (30 gallons / inch).

50 gpm / 30 gallons per inch = 1.67 inches per minute
210 inches / 1.67 inches per minute = 125 minutes
1715 + 125 minutes = 1920

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	009	EA2.05
	Importance Rating	3.4	

K/A: Ability to determine or interpret the following as they apply to a small break LOCA: The time available for action before PZR is empty, given the rate of decrease of PZR level.

Proposed Question: RO Question # 13

Technical Reference(s): OS-24, pg 35, Rev 024

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP024-PCO-5

Question Source:		
	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	5
	55.43	

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics .

Comments:
The KA is matched because the question requires the candidates to demonstrate ability to determine the time available for action before the pressurizer goes empty following a small break LOCA.

The question is at the Comprehension/Analysis cognitive level because the candidate must know all sources of water entering and leaving the RCS, the volume of the Pressurizer per inch, and calculate the amount of time based on the variables for the Pressurizer to go empty.

What MUST be known:
1. What are the sources of water entering the RCS? 2. What are the sources of water leaving the RCS? 3. What is the rule of thumb for gallons per inch in the Pressurizer? 4. What is the length of time until the Pressurizer is empty?

14 **ID: 978948** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Event:

- RCS pressure is 2150 psig and lowering at ≈ 10 psig per minute.
- Pressurizer level is 200 inches and lowering at ≈ 3.5 inches per minute.
- Makeup control valve MU-V-17 automatic demand is at 100%.
- MU Tank level is 80 inches and lowering at ≈ 6.0 inches per minute.
- Seal injection valve MU-V-32 automatic demand is at 100%.
- Total RCP seal injection flow indication has lowered to 26 gpm.

Given the above information, identify the one selection below that describes:
(1) The abnormal event occurring, and
(2) The procedure to be implemented in response to these conditions.

A. (1) RCP seal failure.
(2) OP-TM-AOP-041, Loss of Seal Injection.

B. (1) RCP seal failure.
(2) OP-TM-AOP-050, Reactor Coolant Leakage.

C. (1) RCS Makeup line leak downstream of MU-V-17.
(2) OP-TM-AOP-041, Loss of Seal Injection.

D. (1) RCS Makeup line leak downstream of MU-V-17.
(2) OP-TM-AOP-050, Reactor Coolant Leakage.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. Additional RCP seal failure symptoms would be required to support this distracter as the correct answer. RCP seal failure would not cause a reduction in seal injection flow.

Part 2 is incorrect. AOP-041 is for Loss of Seal injection not RCP Seal failure. Distracter is plausible because conditions indicate loss of mass from the system and the stem presents loss of inventory (possible for #2 or #3 seal failure).

B. **Incorrect.**
Part 1 is incorrect. Additional RCP seal failure symptoms would be required to support this distracter as the correct answer. RCP seal failure would not cause a reduction in seal injection flow.

Part 2 is correct. AOP-050 is for RCS Leakage not RCP Seal failure. Distracter is plausible because conditions indicate loss of mass from the system and the stem presents loss of inventory (possible for #2 or #3 seal failure).

C. **Incorrect.**
Part 1 is correct. With Makeup Tank level and Pressurizer level lowering, along with no concrete indications of a RCP seal failure, the indications lend to a leak downstream of MU-V-17. MU-V-17 is wide open based on Pressurizer level, but the flow is not reaching the Pressurizer.

Part 2 is incorrect. Makeup tank and Pressurizer levels are both lowering due to makeup line leak, however, AOP-041 is for Loss of Seal Injection and is not the correct procedure to use. AOP-041 does address MU-V-32 failure but MU-V-32 is responding correctly. Distracter is plausible because the Seal injection flow is reduced from normal when MU-V-17 is wide open, and the stem presents MU-V-32 control system demand at 100%. Low seal injection flow indication (sensed upstream of MU-V-32) and 100% demand for MU-V-32 position rule out this distracter as a correct answer. If a leak occurred downstream of MU-V-32 the automatic controller would reduce MU-V-32 demand to maintain normal seal injection flow rate.

D. **Correct.**
Part 1 is correct. With Makeup Tank level and Pressurizer level lowering, along with no concrete indications of a RCP seal failure, the indications lend to a leak downstream of MU-V-17. MU-V-17 is wide open based on Pressurizer level, but the flow is not reaching the Pressurizer.

Part 2 is correct. Makeup tank and Pressurizer levels are both lowering due to makeup line leak. Reduced seal injection flow is normal during high makeup flow conditions. Since the Makeup System is connected to the RCS, OP-TM-AOP-050, Reactor Coolant Leakage, is the correct procedure to enter.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022	AA2.01
	Importance Rating	3.2	

K/A: Ability to determine and interpret the following as they apply to | the Loss of Reactor Coolant Makeup: Whether charging line leak exists.

Proposed Question: RO Question # 14

Technical Reference(s): OP-TM-AOP-050, pg 1, Rev 002
Sheet 302661, Rev 061

Proposed References to be provided to applicants during examination: None

Learning Objective:	AOP-041-PCO-4	
Question Source:	Bank #	IR-AOP-041-PCO-4-Q03
	Modified Bank #	
	New	
Question History:		Last NRC Exam: N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	3
	55.43	

3) Mechanical components and design features of the reactor primary system.

Comments:
The KA is matched because the question requires the candidates to interpret conditions and determine that a Reactor Coolant Makeup charging line leak exists.

The question is at the Comprehensive/Analysis cognitive level because the candidate must interpret conditions to determine the location of a leak and then determine which procedure is to be entered as a result.

What MUST be known:
1. What is the location of a MAkeup leak, given conditions? 2. What procedure will be utilized to mitigate the event?

15 **ID: 978947** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- The Reactor has tripped due to a Loss of Offsite Power (LOOP).
    - The crew enters OP-TM-EOP-001, Reactor Trip, and OP-TM-AOP-020, Loss of Station Power.
- A Tube Rupture occurred in OTSG 1B with T_{ave} at 510°F:
    - The crew enters OP-TM-EOP-005, OTSG Tube Leakage.
    - URO reports RCS Pressure is 990 psig and lowering slowly.
    - ARO reports OTSG 1B level is 87% and rising.
- Current Conditions:
    - RCS T_{ave} is 477°F.
    - "A" loop T_{hot} is 495°F and lowering.
    - "A" loop T_{cold} is 459°F and lowering.
    - "B" loop T_{hot} is 477°F and stable.
    - "B" loop T_{cold} is 477°F and stable.
    - Incore thermocouples are 496°F and lowering.
    - RCS pressure is 890 psig.
    - OTSG 1A pressure is 425 psig.
    - OTSG 1B pressure is 550 psig.
    - Emergency Feedwater flow rate to OTSG 1A is 510 gpm.
    - Emergency Feedwater flow rate to OTSG 1B is 0 gpm.
    - OTSG 1A level is 30% OPERATE Range and rising at 1% every 3 minutes.

Given the above information and IAW OP-TM-EOP-010, Guide 10, Natural Circulation, which one of the following choices is correct for the current conditions?

Natural Circulation Core Cooling ___(1)___ occurring in the 1A OTSG and___(2)___ occurring in the 1B OTSG.

A. (1) is
(2) is

B. (1) is
(2) is not

C. (1) is not
(2) is

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

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is correct. IAW OP-TM-EOP-010 Guide 10, Natural Circulation:

If all of the following conditions exist, then adequate natural circulation is present:
- RCS THOT minus TCOLD stabilizes at less than 50 °F.
- THOT < 600 °F.
- Incore temperature stabilizes **and** tracks THOT.
- Cold leg temperatures approach saturation temperature for secondary side pressure.
- OTSG heat removal is indicated by feeding **or** steaming with stable OTSG pressure.
- SCM > 25 °F.

Part 2 is incorrect. See correct answer, part 2, explanation. Plausible if candidate does not understand that the 1B OTSG is isolated due to OTSG tube rupture and IAW OP-TM-EOP-005 Attachment 1B as directed by Step 3.34.

B. **Correct.**
Part 1 is correct. IAW OP-TM-EOP-010 Guide 10, Natural Circulation:

If all of the following conditions exist, then adequate natural circulation is present:
- RCS THOT minus TCOLD stabilizes at less than 50 °F.
- THOT < 600 °F.
- Incore temperature stabilizes **and** tracks THOT.
- Cold leg temperatures approach saturation temperature for secondary side pressure.
- OTSG heat removal is indicated by feeding **or** steaming with stable OTSG pressure.
- SCM > 25 °F.

Part 2 is correct. The "B" OTSG has been isolated IAW OP-TM-EOP-005 at 85% and there is no feeding to or steaming from it. OP-TM-EOP-005, Step 3.34:

IAAT OTSG level > 85% Operate Range, then perform the following:
1. When RCS pressure < 1000 psig, then INITIATE Attachment 1A or 1B to isolate the OTSG.
2. If both OTSGs are being isolated, then GO TO EOP-009.
3. When affected OTSG's TBVs and ADVs are closed, then PERFORM Guide 12 "RCS Stabilization".

Therefore the OP-TM-EOP-010, Guide 10, criteria for Natural Circulation to exist are not met. Plausible if the candidate is not familiar with the Natural Circulation criteria or fails to recognize that the "B" OTSG has been isolated.

C. **Incorrect.**
Part 1 is incorrect. With a Loss of Offsite Power, Reactor Coolant Pumps will trip and HSPS will automatically feed the "A" OTSG to 50% in the Operating Range. This is to promote Natural Circulation. This is not, however, a required level to achieve Natural Circulation. Natural Circulation can occur at the given conditions. Plausible if candidate believes 1A OTSG level at 30% is not sufficient to promote natural circulation.

Part 2 is incorrect. See correct answer, part 2, explanation. Plausible if candidate does not understand that the 1B OTSG is isolated due to OTSG tube rupture and IAW OP-TM-EOP-005 Attachment 1B as directed by Step 3.34.

D. **Incorrect.**
Part 1 is incorrect. With a Loss of Offsite Power, Reactor Coolant Pumps will trip and HSPS will automatically feed the "A" OTSG to 50% in the Operating Range. This is to promote Natural Circulation. This is not, however, a required level to achieve Natural Circulation. Natural Circulation can occur at the given conditions. Plausible if candidate believes 1A OTSG level at 30% is not sufficient to promote natural circulation.

Part 2 is correct. The "B" OTSG has been isolated IAW OP-TM-EOP-005 at 85% and there is no feeding to or steaming from it. OP-TM-EOP-005, Step 3.34:

IAAT OTSG level > 85% Operate Range, then perform the following:

1. When RCS pressure < 1000 psig, then INITIATE Attachment 1A or 1B to isolate the OTSG.
2. If both OTSGs are being isolated, then GO TO EOP-009.
3. When affected OTSG's TBVs and ADVs are closed, then PERFORM Guide 12 "RCS Stabilization".

Therefore the OP-TM-EOP-010, Guide 10, criteria for Natural Circulation to exist are not met. Plausible if the candidate is not familiar with the Natural Circulation criteria or fails to recognize that the "B" OTSG has been isolated.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	038	EA2.09
	Importance Rating	4.2	

K/A: Ability to determine or interpret the following as they apply to a SGTR: Existence of natural circulation, using plant parameters.

Proposed Question: RO Question # 15

Technical Reference(s): OP-TM-EOP-005, pg 17, Rev 009
OP-TM-EOP-010, pg 22, Rev 016

Proposed References to be provided to applicants during examination: None

Learning Objective: EOPG10-PCO-5

Question Source: Bank #

Modified Bank # QS-EOPG10-PCO-5-Q01

New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	2
	55.43	

(2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:
The KA is matched because the question requires the candidates to demonstrate ability to determine the existence of natural circulation following an OTSG tube rupture.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the data, including determining Subcooling Margin, to determine if conditions exist for Natural Circulation IAW OS-24.

What MUST be known:
1. What are the conditions required for Natural Circulation to exist? 2. What is the status of subcooling margin? 3. Does Natural Circulation exist in each OTSG?

Original Question: QS-EOPG10-PCO-5-Q01

The Plant was operating at 100% power when the following events occur:

- The Reactor has tripped due to a Loss of Offsite Power (LOOP).
- A Tube Rupture then occurred in OTSG 1B with T_{ave} at 510°F:
- OTSG 1B was isolated per OP-TM-EOP-005, Attachment 1B.
- OTSG 1B is solid.
- Emergency Feedwater Pump, EF-P-1 was secured.

The following conditions now exist:

- RCS T_{ave} is 477°F.
- "A" loop T_{hot} is 495°F and lowering.
- "A" loop T_{cold} is 459°F and lowering.
- "B" loop T_{hot} is 477°F and stable.
- "B" loop T_{cold} is 477°F and stable.
- Incore thermocouples are 496°F and lowering.
- RCS pressure is 890 psig.
- Emergency Feedwater flow rate to OTSG 1A is 510 gpm.
- Emergency Feedwater flow rate to OTSG 1B is 0 gpm.
- OTSG 1A level is 30% OPERATE Range and rising at 1% every three (3) minutes.
- RCITS suggests the existence of a small Reactor Vessel HEAD BUBBLE.

Based on these Plant conditions, Operators conclude that

"Natural Circulation Core Cooling is __(1)__ due to __(2)__ .

A. (1) NOT POSSIBLE in RCS Loop B,
(2) the Head Bubble.

B. (1) POSSIBLE in RCS Loop A,
(2) EFW spray effectiveness.

C. (1) NOT POSSIBLE in RCS Loop A,
(2) the low level in OTSG 1A.

D. (1) POSSIBLE in RCS Loop B,
(2) the large Heat Sink in OTSG 1B.

Answer: B

16 **ID: 906769** **Points: 1.00**

Plant Conditions:

- Reactor power is 50%.

Event:

- A steam leak was reported in the Turbine Building.
- FW flow is 2.4 E6 lbm/hr in each loop.
- RCS pressure is 1950 psig.
- RCS Thot is 617°F.

Based on the above information, an ATWS has:

A. NOT occurred and Main Feedwater flow should be raised.

B. NOT occurred and the reactor should be tripped if the steam leak cannot be isolated.

C. occurred, the Immediate Manual Actions of OP-TM-EOP-001, Reactor Trip, should be performed, and HPI initiation is required if reactor power remains at 50%.

D. occurred, the Immediate Manual Actions of OP-TM-EOP-001, Reactor Trip, should be performed, and a manual turbine trip is required if reactor power remains at 50%.

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
For most of Figure 2.3-1, 618.8F is the highest temperature allowed and 1900 psig is the highest pressure allowed. However, the Variable Low Pressure Trip (VLPT) is in effect on the bottom right hand corner of the figure. Plausible if the candidate answers the question based on a straight line for pressure and temperature and does not utilize the Figure.

Since the candidate does not believe an ATWS occurred, they may think that raising Feedwater will lower temperature away from near the limit. FW flow, however, is normal for this power level.

B. **Incorrect.**
For most of Figure 2.3-1, 618.8F is the highest temperature allowed and 1900 psig is the highest pressure allowed. However, the Variable Low Pressure Trip (VLPT) is in effect on the bottom right hand corner of the figure. Plausible if the candidate answers the question based on a straight line for pressure and temperature and does not utilize the Figure.

Since the candidate does not believe an ATWS occurred, they will continue in OP-TM-AOP-051, Secondary Side High Energy Leak. However, a leak in the Turbine Building is covered in Section 6.0 and never gives criteria for tripping the reactor. Plausible if the candidate believes that the steam leak is causing RCS pressure to decrease (but it would not cause Thot to increase).

C. **Correct.**
RCS pressure is too low for current RCS Thot. The variable low pressure trip setpoint is exceeded and the reactor should have automatically tripped, so an ATWS has occurred. The second immediate action of OP-TM-EOP-001, Reactor Trip, requires the operator to remain at that step and (Initiate HPI to) borate until the reactor is shut down. The next step (once the reactor is shutdown) requires the turbine trip actions.

D. **Incorrect.**
RCS pressure is too low for current RCS Thot. The variable low pressure trip setpoint is exceeded and the reactor should have automatically tripped, so an ATWS has occurred. The second immediate action of OP-TM-EOP-1 requires the operator to remain at that step and (Initiate HPI to) borate until the reactor is shut down. The next step (once the reactor is shutdown) requires the turbine trip actions. Plausible if the candidate believes that the RNO for Step 2 of OP-TM-EOP-001, Reactor Trip, is to trip the Main Turbine.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	029	2.4.21	
	Importance Rating	4.0		

K/A: Anticipated Transient Without Scram (ATWS): Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question: RO Question # 16

Technical Reference(s): OP-TM-EOP-001, pg 1, Rev 012
TS Figure 2.3-1, pg 2-11, Rev 262

Proposed References to be provided to applicants during examination: TS Figure 2.3-1

Learning Objective: 641-GLO-10

Question Source: Bank # IR-641-GLO-10-Q03
Modified Bank #
New

Question History: Last NRC Exam: N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	5
	55.43	

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the parameters and logic used to assess the status of safety functions to include reactivity control and core cooling and heat removal in relationship with an ATWS.

The question is at the Comprehension/Analysis cognitive level because candidate must know the operation curves of Tech Spec Figure 2.3-1 and understand that the variable high temperature and pressure should cause a reactor trip.

What MUST be known:
1. What are the parameters for VLPT IAW TS Figure 2.3-1?
2. What are the indications of an ATWS?
3. What actions must be taken in response to an ATWS?

17 **ID: 978943** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- T = 0 minutes:
  - An Instrument Air pipe rupture develops in the Turbine Building.
  - IA-P-4 instrument air compressor trips on overload.
  - PLB-1-6, IA-P-4 / IA-Q-2 Trouble, alarm actuates.
  - Instrument and service air compressors IA-P-1A/B and SA-P-1A/B start automatically.
  - Instrument Air Primary (PI-222) and Secondary (PI-1403) pressure indicators on PL are tracking together as they both lower.

- T = 1 minute:
  - Reactor trip is initiated IAW OP-TM-AOP-028, Loss of Instrument Air,
  - MAP G-1-1, Reactor Trip, alarm actuates.

- T = 2 minutes:
  - "Loss of Instrument Air and Reactor Trip" is announced over the plant page and radio.

Given the above information and after the plant announcement is made:
(1) What is the status of PI-222 and PI-1403, and
(2) With regards to Control Room annunciators, what is the NLO response priority IAW OS-24, Conduct of Operations During Abnormal and Emergency Events?

A. (1) PI-222 lowers and PI-1403 lowers.
(2) Reactor Trip.

B. (1) PI-222 lowers and PI-1403 lowers.
(2) Loss of Instrument Air.

C. (1) PI-222 rises and PI-1403 lowers.
(2) Reactor Trip.

D. (1) PI-222 rises and PI-1403 lowers.
(2) Loss of Instrument Air.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. Using OP-TM-AOP-028, Loss of Instrument Air, and prints 302-268 and 302-271: At 60psig instrument air pressure, IA-V-26 will close and isolate primary and secondary side instrument air. 60psig instrument air pressure is identified by the Reactor trip IAW OP-TM-AOP-028.

Part 2 is incorrect. Based on OS-24 Attachment E, the NLO priority for Loss of Instrument air higher than the Reactor trip response. Plausible if the candidate determines that a red annunciator (Reactor Trip) is a higher priority than a PLB alarm (Loss of Instrument Air). IAW OS-24 Attachment E:

> NOTE: If multiple conditions are present, then respond to the highest priority (e.g. LOOP is 2, RX TRIP is 4)
> 1A. REMOTE SHUTDOWN SEQUENCE
> 1B. FIRE
> 2. LOSS OF STATION POWER
> 3. LOSS OF INSTRUMENT AIR
> 4. REACTOR / TURBINE TRIP

B. **Incorrect.**
Part 1 is incorrect. Using OP-TM-AOP-028, Loss of Instrument Air, and prints 302-268 and 302-271: At 60psig instrument air pressure, IA-V-26 will close and isolate primary and secondary side instrument air. 60psig instrument air pressure is identified by the Reactor trip IAW OP-TM-AOP-028.

Part 2 is correct. Based on OS-24 Attachment E, the NLO priority for Loss of Instrument air higher than the Reactor trip response. IAW OS-24 Attachment E:

> NOTE: If multiple conditions are present, then respond to the highest priority (e.g. LOOP is 2, RX TRIP is 4)
> 1A. REMOTE SHUTDOWN SEQUENCE
> 1B. FIRE
> 2. LOSS OF STATION POWER
> 3. LOSS OF INSTRUMENT AIR
> 4. REACTOR / TURBINE TRIP

C. **Incorrect.**
Part 1 is correct. At 60psig instrument air pressure, IA-V-26 will close and isolate primary and secondary side instrument air. IAW TQ-TM-104-850-C001, Station Air Systems:

- Secondary Plant Instrument Air Supply IA-V-26
  - Located in the Intermediated Building basement room south of IA-P-1A
  - Automatically isolates secondary plant Instrument Air header from primary plant Instrument Air header if secondary plant header pressure lowers to 60 psig.

Part 2 is incorrect. Based on OS-24 Attachment E, the NLO priority for Loss of Instrument air higher than the Reactor trip response. Plausible if the candidate determines that a red annunciator (Reactor Trip) is a higher priority than a PLB alarm (Loss of Instrument Air). IAW OS-24 Attachment E:

NOTE: If multiple conditions are present, then respond to the highest priority (e.g. LOOP is 2, RX TRIP is 4)
1A. REMOTE SHUTDOWN SEQUENCE
1B. FIRE
2. LOSS OF STATION POWER
3. LOSS OF INSTRUMENT AIR
4. REACTOR / TURBINE TRIP

D. **Correct.**
Part 1 is correct. Using OP-TM-AOP-028, Loss of Instrument Air, and prints 302-268 and 302-271: At 60psig instrument air pressure, IA-V-26 will close and isolate primary and secondary side instrument air. 60psig instrument air pressure is identified by the Reactor trip IAW OP-TM-AOP-028.

Part 2 is correct. Based on OS-24 Attachment E, the NLO priority for Loss of Instrument air higher than the Reactor trip response. IAW OS-24 Attachment E:

NOTE: If multiple conditions are present, then respond to the highest priority (e.g. LOOP is 2, RX TRIP is 4)
1A. REMOTE SHUTDOWN SEQUENCE
1B. FIRE
2. LOSS OF STATION POWER
3. LOSS OF INSTRUMENT AIR
4. REACTOR / TURBINE TRIP

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	065	2.4.45	
	Importance Rating	4.1		

K/A: Loss of Instrument Air: Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question:	RO Question # 17		
Technical Reference(s):	TQ-TM-104-850-C001, pg 12, Rev 005 OS-24, pg 34, Rev 024		

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-028-PCO-2

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	7
	55.43	

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to prioritize between alarm conditions including a Loss of Instrument Air.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the location of the instrument air leak, automatic valve operations based on the instrument air pressure required to trip, indications based on isolation of various instrument air headers, and NLO priorities and assignments following a Reactor Trip when an instrument air leak exists

What MUST be known:
1. What are the various portions of the Instrument Air System? 2. How does an Instrument Air leak in the Turbine Building affect the Instrument Air System? 3. What is the order of priorities for Auxiliary Operator response IAW OS-24?

18 **ID: 978938** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- National Weather Service has issued a Severe Thunderstorm Warning for Middletown, PA.

Sequence of Events:

- Time = 1810:
  - The Reactor trips due to a Loss of coolant accident.
  - SCM is currently 27°F and lowering at a rate of 1°F per minute.
  - All Reactor Coolant Pumps (RCP's) are currently running.
- Time = 1814:
  - RC-P-1D, "D" Reactor Coolant Pump, is the only running RCP.
  - RCS T_{hot} is 612°F and rising slowly.
  - RCS Pressure is 1900 psig.
  - HPI has been initiated and is currently indicating a flow of 300 gpm.
- Time = 1815:
  - Electrical disturbances are causing severe fluctuations in grid voltage.

Given the above information, RC-P-1D should ___(1)___ at Time = 1815 because ___(2)___.

A. (1) continue to operate
(2) collapsed liquid level below the top of the core may occur

B. (1) continue to operate
(2) the load tap changer will minimize any effects on RC-P-1D

C. (1) be secured immediately
(2) collapsed liquid level below the top of the core will NOT occur

D. (1) be secured immediately
(2) the load tap changer will not minimize any effects on RC-P-1D

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**

Part 1 is correct. With a Loss of Subcooling Margin, the RCP's must be shutdown within one minute of Subcooling Margin going less than 25F. If a RCP is not secured within that first minute, it must remain running until certain criteria are met. None of those criteria have been met. IAW OP-TM-EOP-010, Rule 1, Loss of Subcooling Margin:

> Step 2 RNO: MAINTAIN RCP(s) On until one of the following conditions is satisfied:
> – SCM > 25F,
> – LPI flow > 1250 gpm in each line,
> – Tclad > 1800F.

Part 2 is correct. IAW OP-TM-EOP-0101, Emergency Procedure Rules, Guides, and Graphs Basis Document:

> Step 2 Basis: RCPs are tripped immediately to prevent possible core damage during specific size small break loss of coolant accidents (LOCA). Core damage could occur if the RCPs trip later in the event. Because RCP trip cannot be eliminated as a possibility, RCP trip is performed as soon as the loss of subcooling margin is recognized. This precludes the possibility of RCP trip when the RCS void fraction has progressed to the point (approx. 70% void fraction) where RCP trip would result in a collapsed liquid level below the top of the core.

B. **Incorrect.**

Part 1 is correct. With a Loss of Subcooling Margin, the RCP's must be shutdown within one minute of Subcooling Margin going less than 25F. If a RCP is not secured within that first minute, it must remain running until certain criteria are met. None of those criteria have been met. IAW OP-TM-EOP-010, Rule 1, Loss of Subcooling Margin:

> Step 2 RNO: MAINTAIN RCP(s) On until one of the following conditions is satisfied:
> – SCM > 25F,
> – LPI flow > 1250 gpm in each line,
> – Tclad > 1800F.

Part 2 is incorrect. The Load Tap Changer is associated with the 4kV busses and will adjust as necessary to assure that ES powered loads maintain the proper voltage. The 7kV busses, however, are not affected by the Load Tap Changer. Since the Reactor Coolant Pumps are powered from the 1A and 1B 7kV busses, the Load Tap Changer will not adjust the voltage of RC-P-1D. Plausible if the candidate believes that all busses associated with the Auxiliary Transformers have the ability to have voltage maintained by the Load Tap Changer since the Load Tap Changer is on the Auxiliary Transformer. IAW TQ-TM-104-701-C001, Main Electrical Distribution System:

> 3. Auxiliary Transformer Electrical Flowpaths
>   a. Auxiliary Transformer "A" and "B" take power from the 8 and 4 buses respectively to their primary windings and step down 230 kV to provide 6900 volts and 4160 volts from two separate secondary winding taps for plant use.
>   1) The 4160 volt tap is through a Load Tap Changer
>   2) The 6900 volt tap is a fixed tap.

C. **Incorrect.**

Part 1 is incorrect. With a Loss of Subcooling Margin, the RCP's must be shutdown within one minute of Subcooling Margin going less than 25F. If a RCP is not secured within that first minute, it must remain running until certain criteria are met. None of those criteria have been met. Plausible if the candidate believes that the fluctuation in grid voltage, which could cause damage to RC-P-1D, is justification to secure the RCP. IAW OP-TM-EOP-010, Rule 1, Loss of Subcooling Margin:

> Step 2 RNO: MAINTAIN RCP(s) On until one of the following conditions is satisfied:
> – SCM > 25F,
> – LPI flow > 1250 gpm in each line,
> – Tclad > 1800F.

Part 2 is incorrect. Plausible if the candidate is familiar with the basis but reverses the logic. The choice, although similar to Part 2 in choice "A" above, is tailored to match part 1. IAW OP-TM-EOP-0101, Emergency Procedure Rules, Guides, and Graphs Basis Document:

> Step 2 Basis: RCPs are tripped immediately to prevent possible core damage during specific size small break loss of coolant accidents (LOCA). Core damage could occur if the RCPs trip later in the event. Because RCP trip cannot be eliminated as a possibility, RCP trip is performed as soon as the loss of subcooling margin is recognized. This precludes the possibility of RCP trip when the RCS void fraction has progressed to the point (approx. 70% void fraction) where RCP trip would result in a collapsed liquid level below the top of the core.

D. **Incorrect.**

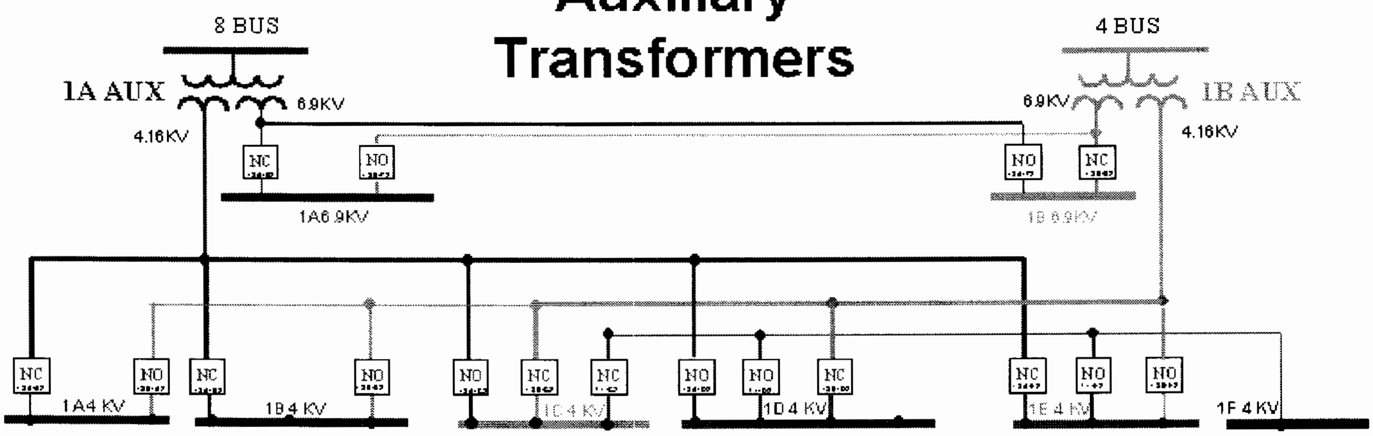
Part 1 is incorrect. With a Loss of Subcooling Margin, the RCP's must be shutdown within one minute of Subcooling Margin going less than 25F. If a RCP is not secured within that first minute, it must remain running until certain criteria are met. None of those criteria have been met. Plausible if the candidate believes that the fluctuation in grid voltage, which could cause damage to RC-P-1D, is justification to secure the RCP. IAW OP-TM-EOP-010, Rule 1, Loss of Subcooling Margin:

> Step 2 RNO: MAINTAIN RCP(s) On until one of the following conditions is satisfied:
> – SCM > 25F,
> – LPI flow > 1250 gpm in each line,
> – Tclad > 1800F.

Part 2 is a correct statement. The Load Tap Changer is associated with the 4kV busses and will adjust as necessary to assure that ES powered loads maintain the proper voltage. The 7kV busses, however, are not affected by the Load Tap Changer. Since the Reactor Coolant Pumps are powered from the 1A and 1B 7kV busses, the Load Tap Changer will not adjust the voltage of RC-P-1D. IAW TQ-TM-104-701-C001, Main Electrical Distribution System:

> 3. Auxiliary Transformer Electrical Flowpaths
>   a. Auxiliary Transformer "A" and "B" take power from the 8 and 4 buses respectively to their primary windings and step down 230 kV to provide 6900 volts and 4160 volts from two separate secondary winding taps for plant use.
>   1) The 4160 volt tap is through a Load Tap Changer
>   2) The 6900 volt tap is a fixed tap.

Auxiliary Transformers



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	077	2.4.18
	Importance Rating	3.3	

K/A: Generator Voltage and Electric Grid Disturbances: Knowledge of the specific bases for EOPs.

Proposed Question: RO Question # 18

Technical Reference(s): OP-TM-EOP-010, PG 3, Rev 016
OP-TM-EOP-0101, PG 6, Rev 008

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-12

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	2
	55.43	

(2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the specific EOP bases for events caused by electric grid disturbances.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze conditions in the stem and then response not obtained steps for Rule 1 and know the basis for not securing a RCP if SCM has been lost for over 1 minute.

What MUST be known:
1. What are the requirements for securing RCP's?
2. What is the status of SCM given plant parameters?
3. What is the reason for not securing a RCP under the conditions given?

19	ID: 978934	Points: 1.00

Plant Conditions:

- The plant is operating at 100% power.
- The discriminator voltage for the applicable Nuclear Instruments has been set too high.

Event:

- A reactor trip has occurred.
- All control rods have inserted fully.

Given the above information, the effect of the the discriminator voltage misadjustment is that ___(1)___ is reading too ___(2)___.

A. (1) NI-3
(2) low

B. (1) NI-3
(2) high

C. (1) NI-11
(2) low

D. (1) NI-11
(2) high

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. NI-3 is an Intermediate Range Nuclear Instrument. The INtermediate Range NI's have a compensating voltage to cancel out the signals created by gammas. They do not have a discriminating voltage. Plausible if the candidate confuses the style of Nuclear Instrumentation between the Source Range and Intermediate Range detectors.

Part 2 is correct. IAW TQ-TM-104-623-C001:

- If discriminator voltage is too high, some neutron pulses will be filtered and the count rate will read low.

B. **Incorrect.**
Part 1 is incorrect. NI-3 is an Intermediate Range Nuclear Instrument. The Intermediate Range NI's have a compensating voltage to cancel out the signals created by gammas. They do not have a discriminating voltage. Plausible if the candidate confuses the style of Nuclear Instrumentation between the Source Range and Intermediate Range detectors.

Part 2 is incorrect. Plausible if the candidate is not familiar with how a discriminating circuit works. IAW TQ-TM-104-623-C001:

- If discriminator voltage is too low, some gamma pulses will not be filtered and the count rate will read high.

C. **Correct.**
Part 1 is correct. NI-11 and NI-12, both Source Range Nuclear Instruments, utilize a Discrimination Circuit to distinguish between the gammas and neutrons that are being detected. IAW TQ-TM-104-623-C001:

- Discrimination circuit
  - Filters out signals caused by gamma interactions.
  - Gamma interactions produce a signal of lesser magnitude due to the fewer number of ion pairs produced, so a pulse height discriminator (PHD) is used to filter out pulses below a certain magnitude.
  - The discriminator voltage is set to eliminate the gamma generated pulses yet not filter out the neutron pulses.
    - If discriminator voltage is too high, some neutron pulses will be filtered and the count rate will read low.
    - If discriminator voltage is too low, some gamma pulses will not be filtered and the count rate will read high.
  - Useful for neutron events up through about 3X10-3 % reactor power, about the top of the Source Range indication (106 cps), where neutron event pulses are still countable as discrete events.

Part 2 is correct. IAW TQ-TM-104-623-C001:

- If discriminator voltage is too high, some neutron pulses will be filtered and the count rate will read low.

D. **Incorrect.**
Part 1 is correct. NI-11 and NI-12, both Source Range Nuclear Instruments, utilize a Discrimination Circuit to distinguish between the gammas and neutrons that are being detected. IAW TQ-TM-104-623-C001:

- Discrimination circuit
  - Filters out signals caused by gamma interactions.
  - Gamma interactions produce a signal of lesser magnitude due to the fewer number of ion pairs produced, so a pulse height discriminator (PHD) is used to filter out pulses below a certain magnitude.
  - The discriminator voltage is set to eliminate the gamma generated pulses yet not filter out the neutron pulses.
    - If discriminator voltage is too high, some neutron pulses will be filtered and the count rate will read low.
    - If discriminator voltage is too low, some gamma pulses will not be filtered and the count rate will read high.
  - Useful for neutron events up through about 3X10-3 % reactor power, about the top of the Source Range indication (106 cps), where neutron event pulses are still countable as discrete events.

Part 2 is incorrect. Plausible if the candidate is not familiar with how a discriminating circuit works. IAW TQ-TM-104-623-C001:

- If discriminator voltage is too low, some gamma pulses will not be filtered and the count rate will read high.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	032	AK1.01
	Importance Rating	2.5	

K/A: Knowledge of the operational implications of the following concepts as they apply to Loss of Source Range Nuclear Instrumentation: Effects of voltage changes on performance

Proposed Question: RO Question # 19

Technical Reference(s): TQ-TM-104-623-C001, pg 14, Rev 004

Proposed References to be provided to applicants during examination: None

Learning Objective: 623-GLO-11

Question Source:	Bank #	IR-623-GLO-11-Q02	
	Modified Bank #		
	New		

Question History:	Last NRC Exam:	N/A

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

10 CFR Part 55 Content:	55.41	6
	55.43	

(6) Design, components, and functions of reactivity control mechanisms and instrumentation.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of effects of voltages changes on the performance of source range nuclear instrumentation.

The question is at the Memory/Fundamental Knowledge cognitive level because candidate must know how discriminator voltage effects NI indication.

What MUST be known:
1. What Nuclear Instruments have Discriminating voltage applied?
2. What effect does high discriminator voltage have on an Nuclear Instrument?

20 **ID: 978930** **Points: 1.00**

Plant Conditions:

- Loss of ICS Auto power occurred 10 minutes ago.
- Plant stability has been verified.
- Reactor power 70% and steady.
- Seal Injection Flow is 38 gpm.
- Makeup flow is normal.
- OP-TM-AOP-027, Loss of ATA or ICS Auto Power has been initiated.

Event:

- Main Feed Pump, FW-P-1B, trips due to low oil pressure.

Given the above information, what condition will occur and what is the required action to stabilize the plant?

A. Turbine control will be lost and a manual reactor trip is required in accordance with OP-TM-AOP-027, Loss of ATA or ICS Auto Power.

B. FW-P-1B, "B" Main Feedwater Pump, Discharge Valve will not receive an automatic close signal and must be closed manually in accordance with OP-TM-MAP-M0107, FWP 1B TRIP.

C. Flow through FW-V-17A, "A" Main Feedwater Control Valve, will be excessive and demand must be reduced in accordance with OP-TM-421-451, Manual Control of Feed Flow to A OTSG.

D. Feedwater flow will be less and FW-P-1A, "A" Main Feedwater Pump, demand must be raised to maintain the required FW Valve ΔP in accordance with OP-TM-401-472, Manual Control of FW-P-1A.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
By stating that the plant is stable in the Plant Conditions given, and that occurs after the loss of ICS Auto power, then that indicates that the Main Turbine is being controlled using Local OWS control. Plausible if the candidate does not recognize that there is a method to control the Main Turbine during this event. The second part would be correct if there was no method to control the Main Turbine. IAW OP-TM-AOP-0271, Loss of ATA or ICS Auto Power Basis Document, Section 2.0, Mitigation Strategy:

- Verify plant parameters are stable using operable indications. Maintain plant control using:
  1) Diamond Control Panel to maintain reactor power,
  2) Hand control of Main FW valves to maintain Tavg & Dtc,
  3) LOCAL OWS control of turbine (or Hand control of TBVs) to maintain OTSG pressure.

B. **Incorrect.**
FW-V-1B, FW-P-1B DIscharge Isolation Valve, will close on a low oil pressure signal. Plausible since this action would be necessary for some FW Pump mechanical trips; however the oil pressure trip is an electronic signal. IAW OP-TM-MAP-M0107, FWP 1B Trip, Section 3.0 Automatic Actions:

- FW-V-1B, FW-P-1B Discharge Isolation Valve, will close on pump trip except if caused by loss of vacuum, overspeed, or local manual trip.

C. **Incorrect.**
When FW-P-1B trips, a runback signal is created that would run the plant back to approximately 68% NI power if ICS was in Auto control. Since the power level given (75%) is higher than 68%, this signal would be in place but ineffective since ICS is in Hand control. Plausible if the examinee believes that the Feedwater Pumps and valves are in Hand control but that the Diamond Control Panel will still run the plant back. In that case, there could be excessive FW flow. In the scenario given however, valve demand will have to be raised to provide more feedwater due to the tripped pump.

D. **Correct.**
When both Main Feedwater Pumps are operating and a loss of ICS Auto Power occurs, they will both switch to Hand Power. OP-TM-AOP-027, steps 3.8 and 3.9 state:

- INITIATE OP-TM-401-472 "Manual Control of FW-P-1A".
- INITIATE OP-TM-401-473 "Manual Control of FW-P-1B"

Those procedures state the following in their respective "Limitations" sections:

- If FW-P-1B is in Auto, then MAINTAIN FW-P-1A speed (in HAND) such that FW Pump flows are equal.
- If both FW Pumps are in HAND and Reactor power is stable above 75%, then MAINTAIN Main Feedwater Valve ΔP 30 to 50 psid.
- At all other times, MAINTAIN Main Feedwater Valve ΔP between 60 and 90 psid.

So initially, Main Feedwater Valve D/P is maintained between 30 psid and 50 psid. Once FW-P-1B trips, Feedwater Valve D/P should be between 60 psid and 90 psid. Since FW Valve D/P is in Hand Control, and being maintained between 30 psid and 50 psid, Valve D/P is too low for the given conditions. Reference directs maintaining DP at 60-90 psi.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	A03	AK1.2	
	Importance Rating	3.0		

K/A: Knowledge of the operational implications of the following concepts as they apply to the (Loss of NNI-Y): Normal, abnormal and emergency operating procedures associated with (Loss of NNI-Y).

Proposed Question: RO Question # 20

Technical Reference(s):
OP-TM-AOP-027, pg 5, Rev 007
OP-TM-401-472, pg 1, Rev 003
OP-TM-401-473, pg 1, Rev 003

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP027-PCO-4

Question Source:
Bank # IR-AOP-027-PCO-4-Q03
Modified Bank #
New

Question History: Last NRC Exam: 2007 (TMI ILT 05-01)

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

10 CFR Part 55 Content:
55.41 4
55.43

(4) Secondary coolant and auxiliary systems that affect the facility.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the operational implications of actions in abnormal operating procedures for a loss of NNI Auto power.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidates must analyze conditions from a scenario given and then choose the correct action with regards to feedwater control when ICS Auto power is lost and controls are in Hand powered.

What MUST be known:
1. What are the limitations for a single Main Feedwater Pump? 2. What are the results of a Loss of ICS Auto power on plant control? 3. What action must be performed to Main Feedwater Pumps upon a Loss of ICS Auto Power?

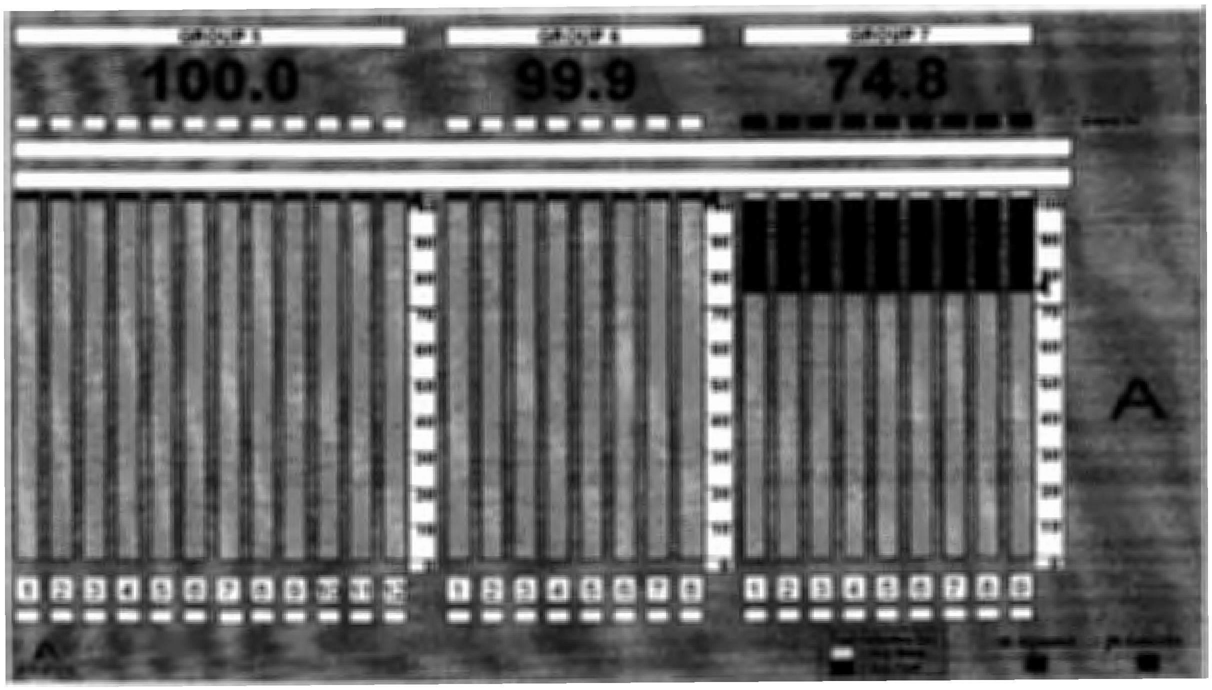
21 **ID: 978920** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- Due to an ICS fault, a rapid power reduction to 88% **Reactor Power** occurred.
- Group 7, Rod 3 indicated higher than the other rods in Group 7.
- Group 7, Rod 3 was inserted to match the other rods in Group 7.
- After trimming, the following alarms were in alarm.
  - PPC L3465, CRD API-RPI MISMATCH.
  - G-3-4, CRD SYSTEM FAULT.
- Control Rod Groups 1 - 4 are currently 100% withdrawn.
- Control Rod Groups 5 - 7 current indications are shown below:



Given the above information, the cause for the above alarms is ___(1)___, and depressing the ___(2)___ pushbutton followed by the Fault Reset pushbutton on the Diamond Control Panel will clear BOTH alarms.

A. (1) failed OPEN reed switches on a PI tube
   (2) RPI Reset

B. (1) failed OPEN reed switches on a PI tube
   (2) ASYM Fault Bypass

C. (1) the CRDM rotating excessively due to a sticking rod
   (2) RPI Reset

D. (1) the CRDM rotating excessively due to a sticking rod
   (2) ASYM Fault Bypass

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. Plausible if the student believes that RPI is developed from the reed switches on the PI tube. IAW TQ-TM-104-622-C001, Control Rod Drive System:

- Absolute Indication
  - There are 45 equally spaced reed switches mounted in a fiberglass housing which is strapped to the outside of the motor tube. These switches are used to build two API signals

Part 2 is correct. OP-TM-PPC-L3465 gives direction to depress RPI Reset and Fault Reset to correct the difference between RPI and API and clear alarms. IAW OP-TM-PPC-L3465, CRD API-RPI Mismatch:

4.0 MANUAL ACTIONS REQUIRED
4.3 API-RPI mismatch can be cleared by setting the RPI equal to the API.
4.3.1 SELECT the Control Rod Group on the Group Select Switch.
4.3.2 SELECT the Control Rod on the Single Select Switch.
4.3.3 PRESS RPI RESET.

B. **Incorrect.**
Part 1 is incorrect. Plausible if the student believes that RPI is developed from the reed switches on the PI tube. IAW TQ-TM-104-622-C001, Control Rod Drive System:

- Absolute Indication
  - There are 45 equally spaced reed switches mounted in a fiberglass housing which is strapped to the outside of the motor tube. These switches are used to build two API signals

Part 2 is incorrect but plausible if the student believes that ASYM Fault Bypass is associated with RPI. IAW OP-TM-MAP-G0201, CRD Pattern Asymmetric:

- 4.0 MANUAL ACTIONS REQUIRED
  - NOTE: A failed open reed switch on the PI tube for Group 1 – 7 rod causes API indication to fail to 62%. This will result in an asymmetric rod position if failure is greater than 7" from the group position. If failed open reed switch is determined to be the cause, OP-TM-622-416, Evaluating PI Problems will select ASYM FAULT BYPASS to enable rod out motion.

C. **Correct.**
Part 1 is correct. Based on drawing given, API is reading correctly and showing that the rod is aligned with the rest of Group 7 and therefore there is an apparent RPI problem since RPI comes off the lead screw and API signal is developed from reed switches on the PI tube. This can be the only correct choice. IAW TQ-TM-104-622-C001, Control Rod Drive System:

- Relative Indication
  - The PG/M Module maintains an RPI counter.
  - This counter registers the position of the rod as if it had been withdrawn from the fully inserted position to the present position in one smooth motion. It does this by counting up or down as the rod is removed or inserted, respectively. The maximum withdrawal for a control rod is slightly less than 12 feet. For the purposes of the RPI calculation, 139.25 inches has been used.
  - Each rotation of the motor moves a rod 0.75 inches; therefore,179 rotations move a rod from one end to the other.

Part 2 is correct. OP-TM-PPC-L3465 gives direction to depress RPI Reset and Fault Reset to correct the difference between RPI and API and clear alarms. IAW OP-TM-PPC-L3465, CRD API-RPI Mismatch:

4.0 MANUAL ACTIONS REQUIRED
4.3 API-RPI mismatch can be cleared by setting the RPI equal to the API.
4.3.1 SELECT the Control Rod Group on the Group Select Switch.
4.3.2 SELECT the Control Rod on the Single Select Switch.
4.3.3 PRESS RPI RESET.

D. **Incorrect.**
Part 1 is correct. Based on drawing given, API is reading correctly and showing that the rod is aligned with the rest of Group 7 and therefore there is an apparent RPI problem since RPI comes off the lead screw and API signal is developed from reed switches on the PI tube. This can be the only correct choice. IAW TQ-TM-104-622-C001, Control Rod Drive System:

- Relative Indication
  - The PG/M Module maintains an RPI counter.
  - This counter registers the position of the rod as if it had been withdrawn from the fully inserted position to the present position in one smooth motion. It does this by counting up or down as the rod is removed or inserted, respectively. The maximum withdrawal for a control rod is slightly less than 12 feet. For the purposes of the RPI calculation, 139.25 inches has been used.
  - Each rotation of the motor moves a rod 0.75 inches; therefore,179 rotations move a rod from one end to the other.

Part 2 is incorrect but plausible if the student believes that ASYM Fault Bypass is associated with RPI. IAW OP-TM-MAP-G0201, CRD Pattern Asymmetric:

- 4.0 MANUAL ACTIONS REQUIRED
  - NOTE: A failed open reed switch on the PI tube for Group 1 – 7 rod causes API indication to fail to 62%. This will result in an asymmetric rod position if failure is greater than 7" from the group position. If failed open reed switch is determined to be the cause, OP-TM-622-416, Evaluating PI Problems will select ASYM FAULT BYPASS to enable rod out motion.

Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	005	AK2.01
	Importance Rating	2.5	

K/A: Knowledge of the interrelations between the Inoperable / Stuck Control Rod and the following: Controllers and positioners.

Proposed Question: RO Question # 21

Technical Reference(s): TQ-TM-104-622-C001, pg 40, Rev 007
OP-TM-PPC-L3465, pg 1, Rev 000

Proposed References to be provided to applicants during examination: None

Learning Objective: 622-GLO-10

Question Source:	
Bank #	
Modified Bank #	
New	X

Question History: Last NRC Exam: N/A

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

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55.43

(2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the interrelationship between an inoperable / stuck control rod and positioners.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the conditions to determine the cause and know the correct actions to mitigate.

What MUST be known:
1. What is the fault associated with the Control Rod Drive System, as indicated in a diagram? 2. What actions must be taken to recover from the fault? 3. What is the difference between Relative Position Indication and Absolute Position Indication for the Control Rod Drive System?

22 **ID: 978917** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- t = 1 minute:
  - A 4# RB actuation occurs.
- t = 2 minutes:
  - A Loss of Offsite Power (LOOP) occurs.

Given the above information, which ONE of the following conditions will trip or shutdown EG-Y-1A, "A" Emergency Diesel Generator, at t = 3 minutes?

A. High Crankcase Pressure.

B. Low Lube Oil Pressure - Running.

C. Pressing the Stop Pushbutton at EG-Y-1A EMIP.

D. Pressing the Stop Pushbutton in the Control Room.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
High Crankcase Pressure is not in effect during an ES condition. See choice B. Plausible if the candidate is not familiar with what trips/interlocks are in effect to shutdown an Emergency Diesel Generator during an ESAS condition.

B. **Correct.**
IAW TQ-TM-104-861-C001:

- Engine Shutdown: Use DC to drive governor to 0 demand, allows restart
  - Low Lube Oil Pressure – idle speed 7 psi (Non-ES)
  - Start Failure (Non – ES)
    - <250 RPM after 9 seconds
    - OR
    - <6 psig lube press after 9 seconds
  - **2-3 low lube oil pressure Running 16psi**
  - 2-3 High Crankcase Pressure (Non – ES)
  - Stop Pushbutton at Engine and Control Room (Non – ES)
  - 86/G Diesel Generator Fault
    - Loss or Ground of Generator Field
    - Reset at EDG Breaker at 1D /1E 4160V Bus
- Engine Trip
  - Engine Over speed
    - Trips fuel rack to no fuel position
    - Requires resetting fuel rack

C. **Incorrect.**
Pressing the Stop Pushbutton in either the Control Room or at the Diesel Generator is not in effect during an ES condition. See choice B. Plausible if the candidate is not familiar with what trips/interlocks are in effect to shutdown an Emergency Diesel Generator during an ESAS condition.

D. **Incorrect.**
Pressing the Stop Pushbutton in either the Control Room or at the Diesel Generator is not in effect during an ES condition. See choice B. Plausible if the candidate is not familiar with what trips/interlocks are in effect to shutdown an Emergency Diesel Generator during an ESAS condition.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	A05	AK2.1	
	Importance Rating	4.0		

K/A: Knowledge of the interrelations between the (Emergency Diesel Actuation) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: RO Question # 22

Technical Reference(s): TQ-TM-104-861-C001, Slide 133, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective:	861-GLO-5		
Question Source:	Bank #		
	Modified Bank #		
	New	X	
Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to demonstrate the knowledge of the interrelationship between the emergency diesel generator and components and functions of safety systems including signals, interlocks, failure modes, and automatic and manual features.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must know the Emergency Diesel generator auto trips during an ES condition.

What MUST be known:
1. What trips/interlocks are in effect to automatically shutdown an Emergency Diesel Generator during an ESAS condition?

23 **ID: 978913** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Event:

- Fire in the Control Room.
- NO immediate actions of OP-TM-EOP-020, Cooldown From Outside of Control Room, were performed prior to the evacuation.

Given the above information and IAW OP-TM-EOP-020, identify:
(1) The actions required to trip the reactor, and
(2) The reason (basis) for the reactor trip.

A. (1) Open CRD breakers at 1G and 1L 480V switchgear.
(2) To prevent spurious operation of the Control Rod Drive System.

B. (1) Open CRD breakers at 1G and 1L 480V switchgear.
(2) To ensure heat production is within the capability of credited heat removal systems.

C. (1) Open CRD breakers CB-1A and CB-1B at the CRD Cabinet.
(2) To prevent spurious operation of the Control Rod Drive System.

D. (1) Open CRD breakers CB-1A and CB-1B at the CRD Cabinet.
(2) To ensure heat production is within the capability of credited heat removal systems.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is correct. IAW OP-TM-EOP-020 Step 2.2:

2.1 TRIP the reactor.
2.2 PERFORM EOP-001 Immediate Manual Actions.
RNO: 1. If reactor is not tripped, then OPEN CRD breakers 1G-2A and 1L-2A (CB 322: patio).

Part 2 is incorrect. OP-TM-EOP-020 has numerous steps taken based on potential system spurious operations of components such as MS-V-8A/B, the PORV, IC-V-2/3/4,and EF-V-2A/B. Choice is plausible if candidate confuses the spurious operations identified in OP-TM-EOP-020 with the reactor trip.

B. **Correct.**
Part 1 is correct. IAW OP-TM-EOP-020 Step 2.2:

2.1 TRIP the reactor.
2.2 PERFORM EOP-001 Immediate Manual Actions.
RNO: 1. If reactor is not tripped, then OPEN CRD breakers 1G-2A and 1L-2A (CB 322: patio).

Part 2 is correct. IAW OP-TM-EOP-0201, Cooldown From Outside of Control Room Basis Document:

Step 2.2 Performance of EOP-001 IMAs ensures that the reactor is shutdown and the turbine is tripped. Reactor shutdown ensures that the reactor is subcritical and that heat production is within the capability of credited heat removal systems. Turbine trip ensures that overcooling due to excessive steam load does not occur.

C. **Incorrect.**
Part 1 is incorrect. The correct action to take is listed in OP-TM-EOP-020 Step 2.2 RNO. Choice is plausible since the combination of CB-1A and CB-1B being opened will trip the reactor.

Part 2 is incorrect. OP-TM-EOP-020 has numerous steps taken based on potential system spurious operations of components such as MS-V-8A/B, the PORV, IC-V-2/3/4,and EF-V-2A/B. Choice is plausible if candidate confuses the spurious operations identified in OP-TM-EOP-020 with the reactor trip.

D. **Incorrect.**
Part 1 is incorrect. The correct action to take is listed in OP-TM-EOP-020 Step 2.2 RNO. Choice is plausible since the combination of CB-1A and CB-1B being opened will trip the reactor.

Part 2 is correct. IAW OP-TM-EOP-0201, Cooldown From Outside of Control Room Basis Document:

Step 2.2 Performance of EOP-001 IMAs ensures that the reactor is shutdown and the turbine is tripped. Reactor shutdown ensures that the reactor is subcritical and that heat production is within the capability of credited heat removal systems. Turbine trip ensures that overcooling due to excessive steam load does not occur.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	068	AK3.01
	Importance Rating	3.9	

K/A: Knowledge of the reasons for the following responses as they apply to the Control Room Evacuation: System response to reactor trip.

Proposed Question: RO Question # 23

Technical Reference(s): OP-TM-EOP-020, pg 1, Rev 015A
OP-TM-EOP-0201, pg 13, Rev 008

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP20-PCO-04

Question Source: Bank #

Modified Bank # IR-EOP20-PCO-04-Q62

New

Question History: Last NRC Exam: 2005 (TMI ILT 03-01)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 2

55.43

(2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:
The KA is matched because the question requires the candidates to demonstrate the knowledge of the reasons of performing the RNO (system response) for a reactor trip during Control Room evacuation.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate needs to have knowledge of procedure steps and basis.

What MUST be known:
1. What is the RNO step for tripping the reactor IAW OP-TM-EOP-020?
2. What is the basis for tripping the reactor in OP-TM-EOP-020?

Original Question:

Plant conditions:

- Reactor is operating at 100% power with ICS in full automatic.
- Fire conditions require Control Room to be evacuated.
- NO immediate actions of OP-TM-EOP-020, Cooldown From Outside of Control Room, were performed prior to the evacuation.

Based on these conditions, identify the ONE selection below that describes how OP-TM-EOP-020 directs you to TRIP THE REACTOR.

A. Open CRD Breakers 10 and 11.

B. Open CRD breakers at 1G and 1L 480V switchgear.

C. Pull the Main Turbine Trip handle at the Front Standard.

D. Pull the Trip Handles at FW-P-1A and FW-P-1B local control consoles.

Answer: B

24 **ID: 978910** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- A fire occurs in the Relay Room.
- Fire and Reactor Trip have been announced over the plant page and radio.
- OP-TM-EOP-020, Cooldown From Outside of Control Room, has been entered.
- Cardox has actuated.
- The Shift Manager determines that Control Room evacuation is required.

Given the above information and IAW plant procedures, which one of the following identifies all of the personnel that will don and operate a Self Contained Breathing Apparatus (SCBA)?

A. ARO,
STA, and
SM.

B. ARO,
STA, and
Primary Fire Brigade AO.

C. URO,
CRS, and
SM.

D. URO,
CRS, and
Primary Fire Brigade AO.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
The ARO and STA do don SCBA's, but the SM is not assigned to don. Plausible if the candidate believes that OS-24 Attachment E AO Reactor Trip actions take priority over OS-24 Attachment E AO Fire actions. Also plausible if the candidate believes that the SM is to don an SCBA.

B. **Correct.**
IAW OP-TM-EOP-020, Cooldown From Outside of Control Room (Note prior to Step 2.8):

- IMAs should be performed by the URO. When IMAs are complete, the URO, SM and CRS will go to the CB 2nd floor and then to the RSD Panels. The ARO and STA will don an SCBA (if required) and go to the CB 3rd floor and then to the RSD panels. The Secondary Safe Shutdown AO will go to TB 322' to isolate the RCPs from the Control Room then go to the EFW area, and the Primary Safe Shutdown AO will go to AB 305', 1A ES Valves MCC. If SCBAs are required for entry, the Fire Brigade leader should be notified to ensure that the fire fighting rescue team will also be a standby rescue team for Operators.

Additionally, IAW OP-TM-EOP-0201, Cooldown From Outside of Control Room Basis Document:

- This note describes operator roles if control room evacuation is required. Specific roles are provided so that RSD control can be established in as quickly and orderly manner as possible. SCBAs may be required for actions on the 3rd floor of the Control Building due potential CARDOX actuation and resultant spread of CO2 gas. These are available in the 3rd floor CB stairwell area.

Lastly, IAW OS-24, Conduct of Operations During Abnormal and Emergency Events, Attachment E, Auxiliary Operator Emergency Response Stations, the Primary Fire Brigade AO position will report to the Fire Brigade and don an SCBA. Although there is a response action for a reactor trip, the Fire is a higher priority according to Attachment E. IAW OS-24, Attachment E:

NOTE: If multiple conditions are present, then respond to the highest priority (e.g. LOOP is 2, RX TRIP is 4)
1A. REMOTE SHUTDOWN SEQUENCE
1B. FIRE
2. LOSS OF STATION POWER
3. LOSS OF INSTRUMENT AIR
4. REACTOR / TURBINE TRIP

C. **Incorrect.**
None of the listed positions (URO, CRS, and SM) don SCBA's. Plausible if the candidate believes that OS-24 Attachment E AO Reactor Trip actions take priority over OS-24 Attachment E AO Fire actions. Also plausible if the candidate is not familiar with OP-TM-EOP-020 actions for evacuating the Control Room.

D. **Incorrect.**
The Primary Fire Brigade AO does don SCBA's, but the CRS and URO are not assigned to don. Plausible if the candidate is not familiar with OP-TM-EOP-020 actions for evacuating the Control Room.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	067	AA1.01
	Importance Rating	3.6	

K/A: Ability to operate and / or monitor the following as they apply to the Plant Fire on Site: Respirator air pack.

Proposed Question:	RO Question # 24		
Technical Reference(s):	OP-TM-EOP-020, pg 3, Rev 15A OS-24, pg 34, Rev 024		

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP020-PCO-1

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	10
	55.43	

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidates to demonstrate the ability to monitor plant conditions and then decide who will operate a respirator air pack during a plant fire on site for a given scenario.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate needs to recall procedure information.

What MUST be known:
1. What Control Room personnel are required to don a respirator air pack during a fire in the Relay Room? 2. What Auxiliary Operator personnel are required to don a respirator air pack during a fire in the Relay Room?

25 **ID: 978907** **Points: 1.00**

Plant Conditions:

- A Large Break LOCA has occurred.
- The Reactor is Shutdown.
- Reactor Building Sump recirculation has been in progress for the last 12 hours IAW OP-TM-EOP-010 Guide 22, RB Sump Recirculation.
- HPI has been secured.
- Building Spray has been secured.
- DH-P-1A, and DH-P-1B, "A" and "B" Decay Heat Removal Pumps, are operating at 2975 gpm each.

Event:

- Both Decay Heat Pumps have the following indications:
  - Flows are oscillating.
  - Amps are oscillating.
  - Discharge Pressures are oscillating.

Given the above information, _________ is the cause of the oscillations.

A. sump screen blockage

B. air entrainment in the piggy back lines

C. insufficient water level in the RB basement

D. Decay Heat System flow being above the allowed limit

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
Since the stem says RB Sump recirc is in progress, Guide 22, RB Sump Recirculation, is being used. IAW OP-TM-010, Emergency Procedures Rules Guides and Graphs, Step 1 of Guide 22:

- Step 1- IAAT DH pump cavitation is evident (flow, amps and discharge pressure are degraded or oscillating and/or vibration has risen) then perform the following: Notify the TSC, Ensure the suction valve is open, Throttle both LPI trains to a min flow of $\geq$ 1250 gpm per train.

IAW OP-TM-0101, Emergency Procedures Rules Guides and Graphs Basis Document, Step 1 of Guide 22:

- Basis: The accumulation of debris on the ECCS sump strainer, and the effects of other post LOCA conditions on the DH pump NPSH have been analyzed. Decay heat pump cavitation should not occur. This step provides direction if it does.

B. **Incorrect.**
Although air entrainment could cause oscillations, it is on the wrong side of the Decay Heat Pumps. (relates to 2006 station OPEX). Additionally, HPI is secured, so being on piggyback is not possible. Plausible if the candidate believes that air entrapment in the location given would be the cause.

C. **Incorrect.**
Although insufficient water level in the RB basement could cause oscillations, however, this has not been a problem for the past 12 hours, and the RB basement inventory will not be significantly reduced by LPI with LB LOCA.

D. **Incorrect.**
Since the stem says RB Sump recirc is in progress, Guide 22, RB Sump Recirculation, is being used. That means that Guide 21, Transfer to RB Sump Recirculation, is complete. Step 4 of Guide 21 states:

> 4. THROTTLE both DH-V-4A and DH-V-4B to the maximum controllable flow < 3000 gpm in each line.

The Conditions given in the stem state that each Decay Heat Removal Pump is operating at 2975 gpm, which is less than the procedural maximum. Plausible if the candidate believes that the 2975 gpm is above the maximum limit (1250 is mentioned in step 1 of Guide 22 as an action to take, but not as a precursor).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	E13	EA1.2
	Importance Rating	2.8	

K/A: Ability to operate and / or monitor the following as they apply to the (EOP Rules): Operating behavior characteristics of the facility.

Proposed Question: RO Question # 25

Technical Reference(s): OP-TM-EOP-010, pg 31, Rev 016
OP-TM-EOP-0101, pg 67, Rev 008

Proposed References to be provided to applicants during examination: None

Learning Objective: EOPG22-PCO-5

Question Source:	Bank #	QR-EOPG22-PCO-5-Q01
	Modified Bank #	
	New	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	8
	55.43	

(8) Components, capacity, and functions of emergency systems.

Comments:
The KA is matched because the question requires the candidates to demonstrate the ability to monitor the operating characteristics of the facility during implementation of EOP Rules and Guides.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate needs to know procedural information.

What MUST be known:
1. What is the cause for Decay Heat Removal Pump oscillating flows/amps/discharge pressures while on RB Sump Recirculation?

26 **ID: 978905** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- Time = 1015:
  - The Reactor has tripped due a low RCS pressure.
  - An Automatic 1600 psig ES actuation has occurred.

- Time = 1030:
  - All four Reactor Coolant Pumps are tripped.
  - Natural Circulation has been verified.
  - RCS Pressure is 1570 psig and lowering at a steady rate.
  - Core exit thermocouple temperature is indicating 540°F and lowering slowly.
  - Tave indicates 500°F lowering slowly.
  - Tcold indicates 455°F lowering slowly.

Given the above information and IAW OP-TM-EOP-010 Guide 11, Cooldown Rate Limits, what is the MAXIMUM allowable RCS cooldown rate at Time = 1030?

A. 30° F/hr.

B. 50° F/hr.

C. 100° F/hr.

D. 240° F/hr.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Plausible because cooldown rate is listed on Guide 11 however this cooldown rate is for 1) RCS Tcold less than 255F or 2) Tc ≤ 400F. IAW OP-TM-EOP-010, Guide 11:

1. VERIFY RCS Tcold > 255 °F.
   RNO: RCS Cooldown rate limit is 30 °F/hr.

2. VERIFY an RCP is ON.
   RNO: If Tc ≤ 400°F, then RCS Cooldown rate limit is 30 °F/hr.

B. **Correct.**
The key to answering/understanding this question is the analysis of the given conditions show that SCM is greater than 25F and the fact that all the RCP's have been secured. With SCM greater than 25F the IAAT statement (entry condition) for Guide 11, this path will be following. Since NO RCP's are running the maximum cooldown rate specified in Step 2 applies.

With no RCP's running and Tcold >400F, Guide 11 states that RCS cooldown rate limit is 50F / hr. IAW OP-TM-EOP-010, Guide 11:

2. VERIFY an RCP is ON.
   RNO: If Tc > 400°F, then RCS Cooldown rate limit is 50 °F/hr.

C. **Incorrect.**
Plausible because cooldown rate is listed on Guide 11 however this cooldown rate is for RCS Tcold greater than 255F, a RCP is ON, and NO OTSG tube leakage exists. IAW OP-TM-EOP-010, Guide 11:

1. VERIFY RCS Tcold > 255 °F.
2. VERIFY an RCP is ON.
3. VERIFY OTSG TUBE LEAKAGE does not exist.
4. RCS Cooldown rate limit is 100 °F/hr.

D. **Incorrect.**
Plausible because cooldown rate is listed on Guide 11 however this cooldown rate is for RCS Tcold greater than 255F, a RCP is ON, and OTSG tube leakage exists. IAW OP-TM-EOP-010, Guide 11:

3. VERIFY OTSG TUBE LEAKAGE does not exist.
   RNO: If OTSG isolation is required, and RCS pressure > 1000 psig, and RCS Tcold > 500 °F, then RCS Cooldown rate limit is 240 °F/hr.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	E09	EA2.2	
	Importance Rating	3.5		

K/A: Ability to determine and interpret the following as they apply to the (Functional Recovery): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question: RO Question # 26

Technical Reference(s): OP-TM-EOP-010, pg 23, Rev 16

Proposed References to be provided to applicants during examination: None

Learning Objective:	EOPG11-PCO-4		
Question Source:	Bank #	QR-EOPG11-PCO-4-Q01	
	Modified Bank #		
	New		
Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	10	
	55.43		

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidates to demonstrate the ability to adhere to appropriate procedures and operation within the limitations in the facility license and amendments during natural circulation cooldown conditions.

The question is at the Comprehension/Analysis cognitive level because the candidate needs to analyze plant conditions, including referencing Steam Tables, and then decide the appropriate cooldown rate.

What MUST be known:
1. What is the status of Subcooling Margin, if given Pressure and Temperature? 2. What is the Cooldown rate for the RCS for a given set of conditions?

27 **ID: 978704** **Points: 1.00**

Plant Conditions:

- The Reactor is in Hot Shutdown.

Sequence of Events:

- A fire in the Relay Room has occurred.
- Immediate Manual Actions of OP-TM-EOP-020, Cooldown from Outside of Control Room, are complete.
- The pressurizer temperature compensation has failed low.
- Follow-up actions of OP-TM-EOP-020 have commenced at the Remote Shutdown Panels.
- The Control Room has been evacuated.
- Pressurizer level at the Remote Shutdown Panel indicates 90 inches and steady.

Given the above information, which one of the following is the correct response IAW OP-TM-EOP-020?

A. Control pressurizer level between 100 inches and 220 inches using MU-V-16C/D, "C" and "D" HPI Control Valves, ONLY.

B. Secure all pressurizer heaters until pressurizer level is restored to the normal Hot Shutdown level band.

C. Ensure MU-V-18, Normal Makeup Reactor Building Isolation Valve, is open and throttle MU-V-217, High Capacity Normal Makeup Valve, until pressurizer level is restored to the normal Hot Shutdown level band.

D. Start MU-P-1C, "C" Makeup Pump, DC-P-1B, "B" Decay Closed Cooling Water Pump, and DR-P-1B, "B" Decay River Water Pump, and then control pressurizer level between 100 inches and 220 inches using MU-V-16C/D, "C" and "D" HPI Control Valves.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Plausible if candidate believes actual pressurizer level is within band and improperly reads Figure 2 of 1105-6, swapping axis. If so, the candidate will believe Pressurizer to be 80 inches and pressurizer heater groups 8 or 9 will be powered from their ES emergency power supply and be required to be secured manually.

B. **Incorrect.**
MU-V-76A and MU-V-76B, MU-P-1B/1C Discharge Header Cross Connect Valves, are closed therefore any makeup from MU-P-1B can not flow through MU-V-16C / 16D. Plausible if the candidate believes that MU-P-1C cannot be started from the Remote Shutdown Panel.

C. **Incorrect.**
MU-V-217, unlike some other Makeup System valves, is not on the Remote Shutdown Panels. Plausible if candidate believes MU-V-217 can be operated from the Remote Shutdown panel.

D. **Correct.**
Since MU-V-76A and MU-V-76B, MU-P-1B/1C Discharge Header Cross Connect Valves, are closed, MU-P-1C is the only way to get Makeup into the Pressurizer. IAW OP-TM-EOP-020 Step 3.4:

IAAT Pzr level is less than 100 inches then perform the following:
1. Start MU-P-1C
2. Start DC-P-1B
3. Start DR-P-1B
4. Control pressurizer level between 100 - 220 inches using MU-V-16C / 16D.

Examination Outline Cross-reference:

Level	RO		SRO
Tier #	1		
Group #	2		
K/A #	028	2.4.6	
Importance Rating	3.7		

K/A: Pressurizer (PZR) Level Control Malfunction: Knowledge of EOP mitigation strategies.

Proposed Question: RO Question # 27

Technical Reference(s): 1105-6, Figure 2, Rev 038
OP-TM-EOP-020, pg 9, Rev 15A

Proposed References to be provided to applicants during examination: 1105-6, Figure 2

Learning Objective: EOP020-PCO-4

Question Source:
Bank #
Modified Bank #
New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.41 10
55.43

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidate to demonstrate the knowledge of EOP mitigation strategies for Pzr level control malfunctions.

The question is at the Comprehensive/Analysis cognitive level because the candidate must understand the effects of a loss of temperature compensation on Pzr level, the Pzr level indications located on the Remote Shutdown Panels, interpret a Pressurizer level graph for Compensated vs Uncompensated, and then determine the specific procedural actions to take based on the overall event.

What MUST be known:
1. What components can be controlled from the Remote Shutdown Panels? 2. What are the different axis for the Compensated vs Uncompensated Pressurizer Level graph? 3. What actions are to be taken IAW OP-TM-EOP-020 to regain Pressurizer level?

28 **ID: 978702** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Event:

- A Reactor trip has occurred due to LOCA.
- RCS pressure is 1400 psig and lowering.
- Reactor Building pressure is 22 psig and rising.
- All equipment is operating as designed.

Given the above information, which one of the following describes the status of:
(1) Reactor Building Spray Pumps, BS-P-1A/B, and
(2) Decay Closed Cooling Water Pumps, DC-P-1A/B and Decay River Water Pumps, DR-P-1A/1B?

A. (1) Running
(2) Started on ES Block 1

B. (1) Running
(2) Started on ES Block 3

C. (1) NOT Running
(2) Started on ES Block 1

D. (1) NOT Running
(2) Started on ES Block 3

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. Reactor Building Spray Pumps will start on a 30# RB signal with a Block 4 permissive signal already present. Since the maximum pressure given in the stem was 22#, they have not started as they have not received the actuation signal. Plausible if the candidate believes that the Building Spray Pumps will start on Block 4 of ESAS loading instead of simply getting a start permissive signal.

Part 2 is incorrect. Decay Closed Cooling Water Pumps and Decay River Water Pumps will start on Block 3 of ESAS loading. Plausible because the candidate may not know which ES Block the DC/DR Pumps started on.

B. **Incorrect.**

Part 1 is incorrect. Reactor Building Spray Pumps will start on a 30# RB signal with a Block 4 permissive signal already present. Since the maximum pressure given in the stem was 22#, they have not started as they have not received the actuation signal. Plausible if the candidate believes that the Building Spray Pumps will start on Block 4 of ESAS loading instead of simply getting a start permissive signal.

Part 2 is correct. DC/DR Pumps will have started on either the 4 # RB ES Actuation, or the 1600# RCS pressure ES actuation. They start on ES Block 3. IAW TQ-TM-104-533-C001:

- Decay Heat Closed Cooling System
  - The DC System will automatically start during an ES actuation from a 1600# or 500# low RCS pressure signal or a 4# high Reactor Building pressure signal.
  - DC-P-1A (1B) will start on Block 3 loading of ES components.
  - DR-P-1A (1B) will start on Block 3 loading of ES components.

C. **Incorrect.**

Part 1 is correct. Reactor Building Spray Pumps will start on a 30# RB signal with a Block 4 permissive signal already present. Since the maximum pressure given in the stem was 22#, they have not started as they have not received the actuation signal. IAW TQ-TM-104-214-C001, Reactor Building Spray:

- 1. Building Spray Normal Operating Conditions
  - a. Mode - Normal;(Auto)-Auto Start at 30 psig increasing Reactor Building pressure (2 out of 3 pressure switches).
- 4 Blocks of ESAS (IRCS)
  - I = Injection
  - R = RB Cooling
  - C = Component Cooling
  - S = Building Spray Permissive
  - Must have block 4 permissive.

Part 2 is incorrect. Decay Closed Cooling Water Pumps and Decay River Water Pumps will start on Block 3 of ESAS loading. Plausible because the candidate may not know which ES Block the DC/DR Pumps started on.

D. **Correct.**
Part 1 is correct. Reactor Building Spray Pumps will start on 1 30# RB signal with a Block 4 permissive signal already present. Since the maximum pressure given in the stem was 22#, they have not started as they have not received the actuation signal. IAW TQ-TM-104-214-C001, Reactor Building Spray:

- 1. Building Spray Normal Operating Conditions
  - a. Mode - Normal;(Auto)-Auto Start at 30 psig increasing Reactor Building pressure (2 out of 3 pressure switches).
- 4 Blocks of ESAS (IRCS)
  - I = Injection
  - R = RB Cooling
  - C = Component Cooling
  - S = Building Spray Permissive
  - Must have block 4 permissive.

Part 2 is correct. DC/DR Pumps will have started on either the 4 # RB ES Actuation, or the 1600# RCS pressure ES actuation. They start on ES Block 3. IAW TQ-TM-104-533-C001:

- Decay Heat Closed Cooling System
  - The DC System will automatically start during an ES actuation from a 1600# or 500# low RCS pressure signal or a 4# high Reactor Building pressure signal.
  - DC-P-1A (1B) will start on Block 3 loading of ES components.
  - DR-P-1A (1B) will start on Block 3 loading of ES components.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026 K1.02	
	Importance Rating	4.1	

K/A: Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: Cooling water.

Proposed Question: RO Question # 28

Technical Reference(s): TQ-TM-104-214-C001, pg 23, Rev 009
TQ-TM-104-533-C001, pg 26, Rev 007

Proposed References to be provided to applicants during examination: None

Learning Objective: 642-GLO-5

Question Source: Bank # IR-642-GLO-5-Q17

Modified Bank #

New

Question History:		Last NRC Exam:	2010 (08-1 Retest)
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	8	
	55.43		

(8) Components, capacity, and functions of emergency systems.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the cause-effect relationship between the containment spray system and its cooling water.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must know the start logic for RB Spray pumps and the block loads for ESAS.

What MUST be known:
1. What block of ESAS loading does DC-P-1A/B and DR-P-1A/B start on?
2. What are the requirements for BS-P-1A/B to start on high Reactor Building pressure?

29 **ID: 978700** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- A Reactor Trip has occurred.
- Currently:
  - A OTSG Level is 19" on the Startup range and rising.
  - B OTSG Level is 98% on the Operating Range due to an ICS Feedwater malfunction.
  - FW-P-1A is Operating.
  - FW-P1B is tripped.
  - RCS Pressure is 1800 psi and lowering.
  - RB Pressure is normal.

Given the above information, level in the A OTSG will be controlled using Startup Range ___(1)___ and level in the B OTSG will be controlled using Operating Range ___(2)___.

A. (1) Type 1
(2) Type 1

B. (1) Type 1
(2) Type 2

C. (1) Type 2
(2) Type 1

D. (1) Type 2
(2) Type 2.

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**

Part 1 is Incorrect. Startup Level is always type 2 and does not swap for any plant conditions. This is plausible if the candidate confuses the Startup and Operating Ranges.

Part 2 is Correct. Operating Range uses Type1 and switches to Type 2 for an EFW actuation signal. For the given plant conditions no EFW actuation signal is present, however HSPS OTSG MFW isolation has occurred on A OTSG but this does not meet the criteria for EFW actuation. IAW TQ-TM-104-644-C001:

- Level measurement will undergo either Type 1 or Type 2 compensation.
  - Type 1 used in Operating Range for normal operating functions
  - Type 1 compensation (operating range only – is used for indication and ICS control when EFW is NOT actuated)
  - This signal is also used for isolation on Hi-Hi

B. **Incorrect.**
Part 1 is Incorrect. Startup Level is always type 2 and does not swap for any plant conditions. Plausible if candidate confuses the Startup and Operating Ranges.

Part 2 is Incorrect. Operating Range uses Type1 and switches to Type 2 for an EFW actuation signal. For the given plant conditions no EFW actuation signal is present, however HSPS OTSG MFW isolation has occurred on A OTSG but this does not meet the criteria for EFW actuation. Plausible if candidate believes an EFW actuation signal exists.

C. **Correct.**
Part 1 is Correct. Startup Level is always type 2 and does not swap for any plant conditions. IAW TQ-TM-104-644-C001:

- Level measurement will undergo either Type 1 or Type 2 compensation.
  - Type 2 used in Startup Range for accident/ abnormal conditions and EF-V-30 control
  - Type 2 Compensation - used for EF-V-30 Indication and Control; selected for panel Indication and ICS Control when EFW is actuated.

Part 2 is Correct. Operating Range uses Type1 and switches to Type 2 for an EFW actuation signal. For the given plant conditions no EFW actuation signal is present, however HSPS OTSG MFW isolation has occurred on A OTSG but this does not meet the criteria for EFW actuation. IAW TQ-TM-104-644-C001:

- Level measurement will undergo either Type 1 or Type 2 compensation.
  - Type 1 used in Operating Range for normal operating functions
  - Type 1 compensation (operating range only – is used for indication and ICS control when EFW is NOT actuated)
  - This signal is also used for isolation on Hi-Hi

D. **Incorrect.**
Part 1 is Correct. Startup Level is always type 2 and does not swap for any plant conditions.
Part 1 is Correct. Startup Level is always type 2 and does not swap for any plant conditions.
IAW TQ-TM-104-644-C001:

- Level measurement will undergo either Type 1 or Type 2 compensation.
  - Type 2 used in Startup Range for accident/ abnormal conditions and EF-V-30 control
  - Type 2 Compensation - used for EF-V-30 Indication and Control; selected for panel Indication and ICS Control when EFW is actuated.

Part 2 is Incorrect. Operating Range uses Type1 and switches to Type 2 for an EFW actuation signal. For the given plant conditions no EFW actuation signal is present, however HSPS OTSG MFW isolation has occurred on A OTSG but this does not meet the criteria for EFW actuation. Plausible if candidate believes an EFW actuation signal exists.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059 K1.04	
	Importance Rating	3.4	

K/A: Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: S/GS water level control system.

Proposed Question: RO Question # 29

Technical Reference(s): TQ-TM-104-644-C001, pg 12,33, Rev 002

Proposed References to be provided to applicants during examination: None

Learning Objective: 644-GLO-6

Question Source: Bank #

Modified Bank # IR-644-GLO-6-Q01

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to demonstrate the knowledge of the cause and effect relationship between Main Feedwater and OTSG water level control.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must know the different level indications for OTSGs.

What MUST be known:
1. What type of control is used on an OTSG in the Startup Range?
2. What type of control is used on an OTSG in the Operating Range?

Original Question: IR-644-GLO-6-Q01

Initial Plant Conditions:

- Reactor Tripped.
- A OTSG Level 98% and rising.
- B OSTG Level 15" Startup range and rising.
- FW-P-1A Tripped.
- FW-P1B Operating.
- RCS Pressure 1700 psi and falling.
- RB Pressure Normal.

Given the above plant conditions, identify the type of level compensation being used for OTSG level.

A. Operating Range Type 1 and Startup Range Type 1.

B. Operating Range Type 1 and Startup Range Type 2.

C. Operating Range Type 2 and Startup Range Type 1.

D. Operating Range Type 2 and Startup Range Type 2.

Answer: B

30 **ID: 978699** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- Main Instrument Air Compressor, IA-P-4, has been secured for repairs.
- "A" and "B" Instrument Air Compressors, IA-P-1A and IA-P-1B, are running as required.

Event:

- A loss of all Secondary Closed Cooling Water Pumps has occurred.

Given the above information and two minutes after the event, the Instrument Air Compressors will have switched to ___(1)___ as the source of cooling due to ___(2)___.

A. (1) ambient air
(2) the tripping of all 3 SCCW pump breakers

B. (1) ambient air
(2) high Instrument Air compressor temperature

C. (1) Fire Service water
(2) the tripping of all 3 SCCW pump breakers

D. (1) Fire Service water
(2) high Instrument Air compressor temperature

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. All three SCCW pump breakers open will open SC-V-57A/B and SC-V-58A/B, fire service water to IA-P-1A/B. This will provide cooling to the Instrument Air compressors, not ambient air. Plausible if the candidate is not familiar with the cooling methods of the Instrument Air compressors and chooses ambient air as a more likely source than Fire Service.

Part 2 is correct. IAW OP-TM-AOP-0331, Loss of Secondary Component Cooling Basis Document, Section 4.10:

> Instrument air compressors (IA-P-1A/B) are controlled by pressure switches in a section of piping upstream of IA-V-2104A and IA-V-2104B, which isolate IA-P-1A and IA-P-1AB compressors from the IA header normally supplied by IA-P-4. When IA-P-4 is shutdown, IA-V-2104A and IA-V-2104B open on decreasing pressure, effectively tying the system together allowing IA-P-1A and IA-P-1B to supply the IA header. Decreasing pressure should auto start IA-P-1A and/or IA-P-1B depending on load and cycle automatically.
>
> Response Not Obtained guidance provides for failure of the auto circuits to maintain IA pressure as specified. The note reminds the procedure user, that Fire Service cooling is supplied to the IA compressors when all SC pumps are shutdown. (Logic for this circuit is satisfied when all SC pump breakers are either open or racked out.)

B. **Incorrect.**
Part 1 is incorrect. All three SCCW pump breakers open will open SC-V-57A/B and SC-V-58A/B, fire service water to IA-P-1A/B. This will provide cooling to the Instrument Air compressors, not ambient air. Plausible if the candidate is not familiar with the cooling methods of the Instrument Air compressors and chooses ambient air as a more likely source than Fire Service.

Part 2 is incorrect but plausible if the candidate is not familiar with the cause-effect relationship between the Cooling water supplies and the Instrument Air system and a high instrument air compressor temperature would indicate a need for supplemental cooling.

C. **Correct.**
Part 1 is correct. IAW 1104-12, Secondary Services Closed Cooling Water Systems, Section 3.7:

> The normal cooling water supply to the instrument air compressors is from the Secondary Services Closed Cooling Water system. If none of the closed cooling water pumps are operating the three way valves SC-V-57 (Compressor. A) and SC-V-58 (Compressor. B) will automatically position to allow Fire Service Water to cool its respective air compressor.

Part 2 is correct. IAW OP-TM-AOP-0331, Loss of Secondary Component Cooling Basis Document, Section 4.10:

> Instrument air compressors (IA-P-1A/B) are controlled by pressure switches in a section of piping upstream of IA-V-2104A and IA-V-2104B, which isolate IA-P-1A and IA-P-1AB compressors from the IA header normally supplied by IA-P-4. When IA-P-4 is shutdown, IA-V-2104A and IA-V-2104B open on decreasing pressure, effectively tying the system together allowing IA-P-1A and IA-P-1B to supply the IA header. Decreasing pressure should auto start IA-P-1A and/or IA-P-1B depending on load and cycle automatically.
>
> Response Not Obtained guidance provides for failure of the auto circuits to maintain IA pressure as specified. The note reminds the procedure user, that Fire Service cooling is supplied to the IA compressors when all SC pumps are shutdown. (Logic for this circuit is satisfied when all SC pump breakers are either open or racked out.)

D. **Incorrect.**
Part 1 is correct. IAW 1104-12, Secondary Services Closed Cooling Water Systems, Section 3.7:

> The normal cooling water supply to the instrument air compressors is from the Secondary Services Closed Cooling Water system. If none of the closed cooling water pumps are operating the three way valves SC-V-57 (Compressor. A) and SC-V-58 (Compressor. B) will automatically position to allow Fire Service Water to cool its respective air compressor.

Part 2 is incorrect but plausible if the candidate is not familiar with the cause-effect relationship between the Cooling water supplies and the Instrument Air system and a high instrument air compressor temperature would indicate a need for supplemental cooling.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078	K1.04
	Importance Rating	2.6	

K/A: Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Cooling water to compressor.

Proposed Question:	RO Question # 30
Technical Reference(s):	1104-12, pg 14, Rev 060 OP-TM-AOP-0331, pg 8, Rev 001

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-GLO-8

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	7
	55.43	

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to demonstrate the knowledge of the cause effect relationship between the instrument air system and cooling water to the compressors.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must know the cooling water supplies and interlocks for instrument air compressors.

What MUST be known:
1. What is the backup cooling supple for Instrument Air Compressors IA-P-1A/1B?
2. What condition initiates backup cooling to Instrument Air Compressors IA-P-1A/1B?

31 **ID: 978698** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- "A" and "C" Nuclear Services Closed Cooling Water Pumps, NS-P-1A and NS-P-1C, are the ES selected pumps and are running.
- "A" and "C" Nuclear Services River Water Pumps, NR-P-1A and NR-P-1C, are the ES selected pumps and are running.

Sequence of Events:

- A LOCA has occurred and the ESAS signals have been cleared.
  - No equipment has been manually secured from its Emergency Safeguards position.
- The 1E 4160 Volt bus lost power and the B Diesel has re-energized the bus.

Given the above information, which one of the following describes the correct component status?

A. Makeup Pump MU-P-1C is off.

B. Nuclear Services River Water Pump NR-P-1B is running.

C. Decay Heat Closed Cooling Water Pump DC-P-1B is running.

D. Nuclear Services Closed Cooling Water Pump NS-P-1B is off.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
When the ESAS signal comes in, MU-P-1A and MU-P-1C will start since they are normally ES selected. When the ESAS signal is cleared, the pumps are still running. When a Loss of the 1E 4kV bus occurs, MU-P-1B will trip but MU-P-1C will remain running. IAW TQ-TM-104-211-C001:

a) Two 43- selector switches are used to determine which pumps will start on Engineered Safeguards Actuation.
f) The other switch is on the MU-P-1B cubicle of the "E" bus.
g) Two positions
(1)MU-P-1B
(2)MU-P-1C
h) The selected pump will start on actuation from "B" side of Engineered Safeguards Actuation System

j) This same switch also prevents the diesel generator from being loaded with more than one Makeup Pump, when it is carrying the Engineered Safeguards bus without offsite power. It does this by causing the non- Engineered Safeguards pump to trip if both the normal (1SB-D2/1SA-E2) and alternate (1SA-D2/1SB-E2) bus feeder breakers are open and prevents starting the non- Engineered Safeguards selected pump at any time.

Plausible if the examinee does not know the MU-P-1C breaker remains closed when the 1E 4160 Volt bus de-energizes.

B. **Correct.**
When the ESAS signal comes in, NR-P-1A and NR-P-1C will continue to run. NR-P-1B remains secured in Normal-after-Stop. When the ESAS signal is cleared, no change in the pumps occur. When the 1E 4kV bus is deenergized, NR-P-1C trips and NR-P-1B loses power. Once the 1E 4kV bus is reenergized from the Emergency Diesel Generator, NR-P-1C breaker remains open and NR-P-1B will start on standby. IAW TQ-TM-104-531-C001:

- ESAS actuation
  a) ES selected NR pumps start on Block 3.
  (1) 43 SS - Determines which pump is selected for ES
  (2) A or B Pump selected on 1 R BUS
  (3) B or C Pump selected on 1T BUS
  b) Non-ES Selected NR pump will trip on an ES actuation.
  c) NR-V-4A/B Close on ES signal

- Standby pump auto-start

If a running pump trips, standby (non-running) pump auto-starts.

C. **Incorrect.**
When the ESAS signal comes in, DC-P-1A and DC-P-1B will start. When the ESAS signal is cleared, no change in the pumps occur. When the 1E 4kV bus is deenergized, DC-P-1B trips. Once the 1E 4kV bus is reenergized from the Emergency Diesel Generator, DC-P-1B breaker remains open.

Plausible if the examinee does not know the DC-P-1B breaker trips when the 1S bus loses power and does not reclose automatically.

D. **Incorrect.**
When the ESAS signal comes in, NS-P-1A and NS-P-1C will continue to run. NS-P-1B remains secured in Normal-after-Stop. When the ESAS signal is cleared, no change in the pumps occur. When the 1E 4kV bus is deenergized, NS-P-1C trips and NS-P-1B loses power. Once the 1E 4kV bus is reenergized from the Emergency Diesel Generator, NS-P-1C breaker remains open and NS-P-1B will start on standby. IAW TQ-TM-104-531-C001:

3) ESAS actuation
   a) ES selected NS pumps start on Block 3.
      (1) 43 SS - Determines which pump is selected for ES
      (2) A or B Pump selected on 1P BUS
      (3) B or C Pump selected on 1S BUS
   b) Non-ES Selected NS pump will trip on an ES actuation.

2) Standby pump auto-start
If a running pump trips, standby (non-running) pump auto-starts.

Plausible if the examinee does not know NS-P-1B will auto start on the NS-P-1C breaker disagreement when the 1S 480 Volt bus de-energizes.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	013	K2.01	
	Importance Rating	3.6		

K/A: Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control.

Proposed Question: RO Question # 31

Technical Reference(s): TQ-TM-104-531-C001, pg37, Rev 007

Proposed References to be provided to applicants during examination: None

Learning Objective: 541-GLO-6

Question Source: Bank # IR-541-GLO-5-Q06
Modified Bank #
New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to demonstrate the knowledge of the power supplies for ESAS equipment control.

The question is at the Comprehension/Analysis cognitive level because the candidate must understand what equipment will be running following a complex sequence of events including ESAS signals, a loss of power, and reinitation of power.

What MUST be known:
1. What happens to Nuclear River Water Pumps on an ES signal? 2. What happens to Nuclear River Water after an ES signal has been cleared? 3. What happens when power is lost to Nuclear River Water Pumps? 4. What happens when power is restored to Nuclear River Water Pumps?

32 **ID: 978697** **Points: 1.00**

Plant Conditions:

- The plant has tripped from 100% power due to a Total Loss of Offsite Power (LOOP).
- The crew has implemented OP-TM-AOP-020, Loss of Station Power.
- Emergency RB Cooling is **not** available.

Event:

- Conditions in the Reactor Building require the alignment of RB Cooling Systems.
- The crew has initiated OP-TM-823-901, RB Normal Cooling Without Offsite Power.

Given the above information and IAW OP-TM-823-901, ___(1)___ of the fans and pumps from each Normal RB Cooling Industrial Cooler Unit will be loaded onto Emergency Diesel Generator ___(2)___.

A. (1) all
(2) EG-Y-1A

B. (1) all
(2) EG-Y-1B

C. (1) half
(2) EG-Y-1A

D. (1) half
(2) EG-Y-1B

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. This is plausible because the operator may incorrectly believe that the Emergency Diesel Generator can load both Normal RB Cooling Industrial Units through the 480V Bus 1N. However, according to OP-TM-823-901 (p2; Rev 1), a Caution prior to Step 4.2 states that only one of 1A or 1B RB H&V MCC is to be energized. Consequently, when the procedure is completed, only one of the two Normal RB Cooling Industrial Cooler Units will be operating.

Part 2 is correct. 480V Bus 1N is powered from 4160V Bus 1D, which receives its power from EG-Y-1A under the LOOP conditions. According to OP-TM-823-901 (p2; Rev 1), a Caution prior to Step 4.2 states that only one of 1A or 1B RB H&V MCC is to be energized. Consequently, when the procedure is completed, only one of the two Normal RB Cooling Industrial Cooler Units will be operating.

B. **Incorrect.**
Part 1 is incorrect. This is plausible because the operator may incorrectly believe that the Emergency Diesel Generator can load both Normal RB Cooling Industrial Units through the 480V Bus 1N. However, according to OP-TM-823-901 (p2; Rev 1), a Caution prior to Step 4.2 states that only one of 1A or 1B RB H&V MCC is to be energized. Consequently, when the procedure is completed, only one of the two Normal RB Cooling Industrial Cooler Units will be operating.

Part 2 is incorrect. This is plausible because the operator may incorrectly believe that 480V Bus 1N is powered from the EG-Y-1B during a LOOP.

C. **Correct.**
Part 1 is correct. According to TQ-TM-104-824-C001, the RB Normal Cooling Industrial Cooler components, including the industrial cooler fans, spray pumps, and circulating pumps are powered from 1A & 1B Rx Building H&V 480V MCCs (half to each MCC). IAW TQ-TM-104-824-C001:

- All industrial cooler fans, spray pumps, and circulating pumps are powered from 1A & 1B Rx Building H&V 480V MCCs (half to each MCC).

According to TQ-TM-104-824-C001, if a total loss of Normal RB cooling (i.e. Loss of Offsite Power) AND loss of RB Emergency Cooling occurs, OP-TM-823-901, RB Normal Cooling Without Offsite Power, can be initiated. This procedure uses power from the Emergency Diesel Generator to energize 1A or 1B Reactor Bldg. H&V MCC through the 1N bus cross tie and a temporary bus tie.

Part 2 is correct. 480V Bus 1N is powered from 4160V Bus 1D, which receives its power from EG-Y-1A under the LOOP conditions. According to OP-TM-823-901 (p2; Rev 1), a Caution prior to Step 4.2 states that only one of 1A or 1B RB H&V MCC is to be energized. Consequently, when the procedure is completed, only one of the two Normal RB Cooling Industrial Cooler Units will be operating.

D. **Incorrect.**
Part 1 is correct. According to TQ-TM-104-824-C001, the RB Normal Cooling Industrial Cooler components, including the industrial cooler fans, spray pumps, and circulating pumps are powered from 1A & 1B Rx Building H&V 480V MCCs (half to each MCC). According to TQ-TM-104-824-C001, if a total loss of Normal RB cooling (i.e. Loss of Offsite Power) AND loss of RB Emergency Cooling occurs, OP-TM-823-901, RB Normal Cooling Without Offsite Power, can be initiated. This procedure uses power from the Emergency Diesel Generator to energize 1A or 1B Reactor Bldg. H&V MCC through the 1N bus cross tie and a temporary bus tie.

Part 2 is incorrect. This is plausible because the operator may incorrectly believe that 480V Bus 1N is powered from the EG-Y-1B during a LOOP.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	022	K2.02	
	Importance Rating	2.5		

K/A: Knowledge of power supplies to the following: Chillers.

Proposed Question: RO Question # 32

Technical Reference(s): TQ-TM-104-824-C001, pg 26,63,64, Rev 4
OP-TM-823-901, pg 3, Rev 1

Proposed References to be provided to applicants during examination: None

Learning Objective: 824-GLO-4

Question Source: Bank # IR-534-GLO-11-Q01

Modified Bank #

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

(9) Shielding, isolation, and containment design features, including access limitations.

Comments:
The KA is matched because the question requires the candidates to demonstrate the knowledge of the power supplies for the containment cooling system chillers.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must know procedural information.

What MUST be known:
1. What is the normal power supply to the Normal RB Cooling Industrial Coolers and support equipment? 2. How do these buses receive power during a LOOP? 3. How many of the Industrial Coolers can be re-powered during an LOOP?

33 **ID: 978696** **Points: 1.00**

Plant Conditions:

- A Reactor Start-up is in progress.
- The Reactor is **not** critical.
- RCS temperature is 530°F and steady.
- RCS pressure is 2160 psig and steady.
- Decay Heat Removal Pump, DH-P-1A, is out-of-service and is expected to be returned within 12 hours.
- All other systems are operable at this time.

Given the above information, which one of the following identifies the correct statement regarding the Reactor Startup?

A. DH-P-1A shall be returned to an operable condition prior to 2% power.

B. DH-P-1A shall be returned to an operable condition prior to criticality, ONLY.

C. The Reactor Startup may continue to 100% as long as DH-P-1A is returned to service within 72 hours.

D. The Reactor Startup may continue to 100% because only one of the DHR Loops is required to be operable.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Tech. Spec. 3.3.1 requires both DHR Loops operable prior to taking the Rx. critical:

> 3.3.1 The reactor shall not be made critical unless the following conditions are met:
> 3.3.1.1 Injection Systems
> c. Two decay heat removal pumps are OPERABLE. Specification 3.0.1 applies.

B. **Correct.**
Tech. Spec. 3.3.1 requires both DHR Loops operable prior to taking the Rx. critical.

> 3.3.1 The reactor shall not be made critical unless the following conditions are met:
> 3.3.1.1 Injection Systems
> c. Two decay heat removal pumps are OPERABLE. Specification 3.0.1 applies.

C. **Incorrect.**
This choice is given in Tech. Spec. 3.3.2 is only valid when the Reactor is operating at a power level.

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS (Contd.)

3.3.2 Maintenance or testing shall be allowed during reactor operation on any component(s) in the makeup and purification, decay heat, RB emergency cooling water, RB spray, BWST level instrumentation, or cooling water systems which will not remove more than one train of each system from service. Components shall not be removed from service so that the affected system train is inoperable for more than 72 consecutive hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.

D. **Incorrect.**
Tech. Spec. 3.4.2 allows this for Refueling operations only:

3.4.2 RCS temperature less than or equal to 250 degrees F.

3.4.2.1 At least two of the following means for maintaining DHR capability shall be OPERABLE and at least one shall be in operation except as allowed by Specifications 3.4.2.2, 3.4.2.3 and 3.4.2.4.

a. DHR String (Loop "A").
b. DHR String (Loop "B").
c. RCS Loop "A" and its associated OTSG with an EFW Pump and a flowpath.
d. RCS Loop "B" and its associated OTSG with an EFW Pump and a flowpath.

With less than the above required means for maintaining DHR capability OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

3.4.2.2 Operation of the means for DHR may be suspended provided the core outlet temperature is maintained below saturation temperature.

3.4.2.3 The number of means for DHR required to be OPERABLE per Specification 3.4.2.1 may be reduced to one provided that the Reactor is in a REFUELING SHUTDOWN condition with the Fuel Transfer Canal water level greater than or equal to 23 feet above the Reactor Vessel flange.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	005	K3.05	
	Importance Rating	3.7		

K/A: Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: ECCS.

Proposed Question: RO Question # 33

Technical Reference(s): T.S. 3.03, pg 3-21, Rev 278

Proposed References to be provided to applicants during examination: None

Learning Objective: 212-GLO-14

Question Source: Bank # QR-212-GLO-14-Q02

Modified Bank #

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

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55.43

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to describe the effect of a loss of a decay heat removal pump has on ECCS from a Tech Spec standpoint.

The question is at the Comprehension/Analysis cognitive level because the candidate must recall the applicable Tech Spec given the conditions.

What MUST be known:
1. What is the Decay Heat Removal System Tech Spec requirement to take the reactor critical?

34 **ID: 978695** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- The National Weather Service has issued a Heat Advisory for the Middletown area.

Event:

- RCS Pressure is 2045 psig and slowly lowering.
- Pressurizer Temperature is 640°F and slowly lowering.
- RCDT level, temperature, and pressure are steady.
- RB pressure is rising very slowly.

Given the above information, which one of the following describes:
(1) The cause of the condition described above, and
(2) The impact on departure from nucleate boiling ratio (DNBR)?

A. (1) PORV is leaking by.
(2) DNBR is approaching limits.

B. (1) PORV is leaking by.
(2) DNBR is moving further from limits.

C. (1) Pressurizer Spray Valve is leaking by.
(2) DNBR is approaching limits.

D. (1) Pressurizer Spray Valve is leaking by.
(2) DNBR is moving further from limits.

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**

Part 1 is incorrect. Plausible if candidate believes that the PORV leaking will have an effect on RB pressure. Additionally, the PORV would relieve to the Reactor Coolant Drain Tank, which is steady, as stated in the stem.

Part 2 is correct. DNBR is the ratio of CHF to Actual HF. Lowering RCS pressure reduces CHF value and decreases DNBR. IAW PWR Generic Fundamentals - Thermodynamics, Chapter 8, Thermal Hydraulics:

- Departure from Nucleate Boiling Ratio
  - DNBR is ratio of critical heat flux to actual heat flux
  - DNBR is maintained greater than 1.3 during all modes of operation, ensuring DNB is not reached
  - To keep DNBR above 1.3, operator monitors RCS temperature, pressure, flow rate, and reactor power level
  - If temperature is maintained and RCS pressure reduced, DNBR will decrease
    - A reduction in pressure shifts boiling curve to left
    - Thus, operating at lower pressures allows DNB to occur at lower temperatures
    - Pressure could be inadvertently reduced by pressurizer pressure controller failure, stuck open spray valve, or PORV (pressure operated relief valve)

B. **Incorrect.**

Part 1 is incorrect. Plausible if candidate believes that the PORV leaking will have an effect on RB pressure. Additionally, the PORV would relieve to the Reactor Coolant Drain Tank, which is steady, as stated in the stem.

Part 2 is incorrect. DNBR is the ratio of CHF to Actual HF. Lowering RCS pressure reduces CHF value and decreases DNBR. Plausible if definition of DNBR is inverted.

C. **Correct.**
Part 1 is correct. Indications are consistent with the spray valve failed open. RB pressure rise is do to the higher than normal outside air temperatures. IAW OP-TM-MAP-G0308, RC Pressure Narrow Range Hi/Lo:

Causes:
- RC-V-1, Pressurizer Spray Valve open
- Faulty pressurizer heater operation
- RCS temperature rising or lowering
- Primary coolant leak
- PORV open
- Failure of controlling RCS pressure Instruments
- Loss of ICS Auto or Hand power

Since there are only two choices (Spray valve or PORV), and RCDT level is not rising, it cannot be the PORV and must be the Spray Valve.

Part 2 is correct. DNBR is the ratio of CHF to Actual HF. Lowering RCS pressure reduces CHF value and decreases DNBR. IAW PWR Generic Fundamentals - Thermodynamics, Chapter 8, Thermal Hydraulics:

- Departure from Nucleate Boiling Ratio
    - DNBR is ratio of critical heat flux to actual heat flux
    - DNBR is maintained greater than 1.3 during all modes of operation, ensuring DNB is not reached
    - To keep DNBR above 1.3, operator monitors RCS temperature, pressure, flow rate, and reactor power level
    - If temperature is maintained and RCS pressure reduced, DNBR will decrease
        - A reduction in pressure shifts boiling curve to left
        - Thus, operating at lower pressures allows DNB to occur at lower temperatures
        - Pressure could be inadvertently reduced by pressurizer pressure controller failure, stuck open spray valve, or PORV (pressure operated relief valve)

D. **Incorrect.**
Part 1 is correct. Indications are consistent with the spray valve failed open. RB pressure rise is do to the higher than normal outside air temperatures. IAW OP-TM-MAP-G0308, RC Pressure Narrow Range Hi/Lo:

Causes:
- RC-V-1, Pressurizer Spray Valve open
- Faulty pressurizer heater operation
- RCS temperature rising or lowering
- Primary coolant leak
- PORV open
- Failure of controlling RCS pressure Instruments
- Loss of ICS Auto or Hand power

Since there are only two choices (Spray valve or PORV), and RCDT level is not rising, it cannot be the PORV and must be the Spray Valve.

Part 2 is incorrect. DNBR is the ratio of CHF to Actual HF. Lowering RCS pressure reduces CHF value and decreases DNBR. Plausible if definition of DNBR is inverted.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010 K3.01	
	Importance Rating	3.8	

K/A: Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: RCS.

Proposed Question: RO Question # 34

Technical Reference(s): OP-TM-MAP-G0308, pg 1, Rev 003
PWR Generic Fundamentals - Thermodynamics, Chapter 8 - Thermal Hydraulics, pg 23-25, Rev 4

Proposed References to be provided to applicants during examination: None

Learning Objective: 223-GLO-11

Question Source:
Bank #
Modified Bank #
New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 2
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(2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the effect of a malfunction of the pressurizer pressure control system on the reactor coolant system.

The question is at the Comprehension/Analysis cognitive level because the candidate must be able to analyze conditions to identify the cause of the pressure reduction and then determine how it will effect the departure from nucleate boiling ratio.

What MUST be known:
1. What are the indications of a leaking Pressurizer Spray Valve?
2. How does a leaking Pressurizer Spray Valve affect RCS Pressure?
3. How does a lowering RCS Pressure affect the Departure from Nucleate Boiling ratio?

35 **ID: 978694** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- Time = 0 minutes:
  - The Reactor has been tripped due to a Loss of Main Feedwater Pumps.
- Time = 2 minutes:
  - Due to stuck open Main Steam Safety Valves, a Loss of Subcooling Margin occurs.
- Time = 9 minutes:
  - EFW pumps are running and maintaining OTSG levels at the proper setpoint.
  - Boiler Condenser Cooling has commenced.
- Time = 10 minutes:
  - A Loss of all Emergency Feedwater Pumps has occurred.
  - It will be ten minutes to restore EF-P-1, Steam Driven Emergency Feedwater Pump, to service.

Given the above information and assuming no further operator actions are taken, RCS temperature and pressure will ___(1)___ until EF-P-1 is restored because Boiler Condenser Cooling will ___(2)___ exist.

A. (1) rise, ONLY
(2) no longer

B. (1) rise, ONLY
(2) continue to

C. (1) experience swings
(2) no longer

D. (1) experience swings
(2) continue to

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. Plausible if the candidate does not recognize that the minimum level setpoint is high enough to provide the required condensing surface during periods when EFW flow may not exist.

Part 2 is incorrect. Plausible since the candidate believes that Boiler Condenser Cooling has stopped, the candidate should also conclude that RCS temperatures and Pressures will rise due to heat generation from decay heat and an absence of a heat removal.

B. **Incorrect.**
Part 1 is incorrect. Plausible if the candidate does not recognize that the minimum level setpoint is high enough to provide the required condensing surface during periods when EFW flow may not exist.

Part 2 is correct. IAW OP-TM-EOP-0101, Emergency Procedure Rules Guides and Graphs Basis Document, pg 7:

> Boiler Condenser Cooling is a cyclic, but effective means of heat transfer. Swings in RCS temperature and pressure can be expected, while utilizing this mode of core cooling.

C. **Incorrect.**
Part 1 is correct. Rule 1, Loss of Subcooling Margin, step 4 states:

- INITIATE OP-TM-424-901, "Emergency Feedwater" and FEED IAW Rule 4.

The basis document, OP-TM-EOP-0101, Emergency Procedure Rules Guides and Graphs Basis Document, gives the following reason for this step:

- The step intent is to align Emergency Feedwater system to establish and maintain primary to secondary heat transfer. The specified OTSG level ensures adequate OTSG condensing surface area. Use of the emergency support procedure ensures the appropriate equipment is started, and helps to identify improper alignment or failed equipment for application of contingency guidance.
- EFW flow is supplied at a minimum rate or greater until the loss of subcooling margin setpoint is attained. The minimum EFW flowrate ensures sufficient energy removal from the RCS until the minimum level is established. EFW is used because the elevation of the EFW nozzles is high enough to provide the required condensing surface when the minimum level has not been established in the OTSGs. The minimum level setpoint is high enough to provide the required condensing surface during periods when EFW flow may not exist.
- The loss of subcooling margin OTSG level setpoint provides enough condensing surface area in the OTSG to allow "Boiler Condenser Cooling" (BCC) even if EFW flow is interrupted. During BCC, the condensed steam will collect in the OTSG tubes until the level is above the RCS spillover level, allowing condensate to be returned to the core.

Part 2 is incorrect but plausible if the candidate is not familiar with the characteristics of Boiler Condenser Cooling.

D. **Correct.**
Part 1 is correct. Rule 1, Loss of Subcooling Margin, step 4 states:

- INITIATE OP-TM-424-901, "Emergency Feedwater" and FEED IAW Rule 4.

The basis document, OP-TM-EOP-0101, Emergency Procedure Rules Guides and Graphs Basis Document, gives the following reason for this step:

- The step intent is to align Emergency Feedwater system to establish and maintain primary to secondary heat transfer. The specified OTSG level ensures adequate OTSG condensing surface area. Use of the emergency support procedure ensures the appropriate equipment is started, and helps to identify improper alignment or failed equipment for application of contingency guidance.
- EFW flow is supplied at a minimum rate or greater until the loss of subcooling margin setpoint is attained. The minimum EFW flowrate ensures sufficient energy removal from the RCS until the minimum level is established. EFW is used because the elevation of the EFW nozzles is high enough to provide the required condensing surface when the minimum level has not been established in the OTSGs. The minimum level setpoint is high enough to provide the required condensing surface during periods when EFW flow may not exist.
- The loss of subcooling margin OTSG level setpoint provides enough condensing surface area in the OTSG to allow "Boiler Condenser Cooling" (BCC) even if EFW flow is interrupted. During BCC, the condensed steam will collect in the OTSG tubes until the level is above the RCS spillover level, allowing condensate to be returned to the core.

Part 2 is correct. IAW OP-TM-EOP-0101, Emergency Procedure Rules Guides and Graphs Basis Document, pg 7:

- Boiler Condenser Cooling is a cyclic, but effective means of heat transfer. Swings in RCS temperature and pressure can be expected, while utilizing this mode of core cooling.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	061	K3.01	
	Importance Rating	4.4		

K/A: Knowledge of the effect that a loss or malfunction of the AFW will have on the following: RCS.

Proposed Question: RO Question # 35

Technical Reference(s): OP-TM-EOP-0101, pg 7, Rev 008

Proposed References to be provided to applicants during examination: None

Learning Objective:	EOPR4-PCO-5		
Question Source:	Bank #		
	Modified Bank #		(
	New	X	
Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	14	
	55.43		

(14) Principles of heat transfer thermodynamics and fluid mechanics.

Comments:
The KA is matched because the question requires the candidates to demonstrate the knowledge of the effect of a loss of the emergency feedwater system on the reactor coolant system.

The question is at the Comprehension/Analysis cognitive level because the candidate must understand the effects of a loss of EFW on RCS pressure and temperature and understand the Boiler Condenser fundamentals and how they apply to an event given a frame of time.

What MUST be known:
1. What is the time frame and conditions required for Boiler Condenser Cooling to occur? 2. What will happen to RCS temperature and pressure based on a loss of all feedwater to the OTSG's after Boiler Condenser Cooling has been established?

36 **ID: 978693** **Points: 1.00**

Plant Conditions:

- RC-P-1A, "A" Reactor Coolant Pump, is being prepared for start following a refueling outage.
  - This will be the first RCP started.
- The following indications are available:
  - RC-P-1A lower oil level is -1.5"
  - RC-P-1A #1 seal dP is 225 psid.
  - Total seal injection flow is 20 gpm.
  - Total ICCW flow is 600 gpm.

Given the above information, _________ must be corrected in order to start RC-P-1A.

A. total ICCW flow

B. RC-P-1A #1 seal dP

C. total seal injection flow

D. RC-P-1A lower oil level

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
IAW OP-TM-226-000, Reactor Coolant Pumps, Section 6.14, Reactor Coolant Pump and Motor Interlocks, Trips, and Alarms, the total ICCW flow interlock to start the RCP is 550 gpm. Plausible because the given value is lower than normal ICCW flow.

B. **Incorrect.**
IAW OP-TM-226-000, Reactor Coolant Pumps, Section 6.14, Reactor Coolant Pump and Motor Interlocks, Trips, and Alarms, the #1 seal dP interlock to start the RCP is 210 psid. Plausible because the given value is much lower than the normal value at full RCS pressure.

C. **Correct.**
IAW OP-TM-226-000, Reactor Coolant Pumps, Section 6.14, Reactor Coolant Pump and Motor Interlocks, Trips, and Alarms, the total Seal Injection flow interlock to start the RCP is 22 gpm. OP-TM-226-000, Reactor Coolant Pumps, Section 6.14:

D. **Incorrect.**
IAW OP-TM-226-000, Reactor Coolant Pumps, Section 6.14, Reactor Coolant Pump and Motor Interlocks, Trips, and Alarms, the lower oil level interlock to start the RCP is -2". Plausible because the given value is lower than the normal control band for RC-P-1A. Also plausible if the candidate confuses RC-P-1A setpoints with that of RC-P-1C (0.0").

Starting Interlock No. 1	**Setpoint**
Temperature Switch	407°F RCS Temperature
Starting Interlock No. 2	**Setpoint**
Reactor Power	Less than 30 percent
Mtr. Backstop Oil System Flow (RC-P-1A, 1B, and 1D only)	> 0.4 GPM
Mtr. Oil Lift System Pressure	Greater than 1000 PSIG Greater than 610 PSIG for RC-P-1C
Mtr. Upper Bearing Oil Level	> -1"
Mtr. Lower Bearing Oil Level	> -2" for RC-P-1A, B, D > 0.0" for RC-P-1C
Mtr. Heat Exchanger Cooling Water Flow	N.S. Pumps Bkr. Contacts
Pump Seal Injection Water Flow	**22 GPM**
Pump Thermal Barrier Cooling Water Flow Total Int. Cooling Water System Flow)	550 GPM
Mtr. Starting Under Voltage	6.62 KV
Pump Seal No. 1 Delta Pressure	210 psid

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 K4.04	
	Importance Rating	2.8	

K/A: Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Adequate cooling of RCP motor and seals.

Proposed Question: RO Question # 36

Technical Reference(s): OP-TM-226-000, pg 12, Rev 008

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-10

Question Source: Bank # IR-226-GLO-10-Q04

Modified Bank #

New

Question History: Last NRC Exam: TMI 2010 (08-01)

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

10 CFR Part 55 Content:	55.41	3
	55.43	

(3) Mechanical components and design features of the reactor primary system.

Comments:
The KA is matched because the question requires the candidates to demonstrate the knowledge of the reactor coolant pump system interlocks which provide adequate RCP motor and seal cooling prior to starting a RCP.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must know the starting pre-requisites along with starting interlocks for the RCPs.

What MUST be known:
1. What are the interlocks associated with starting a Reactor Coolant Pump?

37 **ID: 978692** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- All feed and bleed evolutions are secured.

Event:

- MU-V-8, Letdown Split Valve to MU-T-1 or RCBT, has failed to the "Bleed" position.

Given the above information, which one of the following is the correct system response?

A. RCBT relief valve will lift.

B. Letdown line relief valves will lift.

C. Makeup Filters will exceed Maximum design ΔP.

D. Makeup Demineralizers will exceed Maximum design ΔP.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
The RCBT is not listed as a susceptible component in OP-TM-211-000. Plausible if the candidate believes that the RCBT is connected straight through MU-V-8 without any RCBT-related WDL valves between them. OP-TM-211-000, Makeup and Purification System, Section 2.1.4.1:

> To avoid lifting Letdown relief valves or damaging WDL valve diaphragms, do not position MU-V-8 to the Bleed position prior to ensuring a proper bleed path.

B. **Correct.**
With MU-V-8 failing to the Bleed position, Letdown flow has been diverted from it's normal path of through to filters to the path of Bleeding to the RCBT, which has closed valves in the line. Therefore, Letdown is now through a deadheaded pipe, which will cause the Letdown line relief valves to lift, as is warned against in OP-TM-211-000, Makeup and Purification System, Section 2.1.4.1:

> To avoid lifting Letdown relief valves or damaging WDL valve diaphragms, do not position MU-V-8 to the Bleed position prior to ensuring a proper bleed path.

C. **Incorrect.**
Plausible if the candidate is unfamiliar with the positions of MU-V-8 and believes that the Bleed position makes a path through the Makeup Filters instead of to the RCBT. If this is the line of thinking, then the candidate might believe that more flow is now going through the filters and therefore would cause a higher ΔP. The "Bleed" position diverts Letdown to the Reactor Coolant Bleed Tanks and away from the Makeup Tank IAW TQ-TM-104-211, Makeup System:

4. Precautions for Feed and Bleed Processes
   a. Do NOT position MU-V-8 (Letdown Split Valve to MU-T-1 or Reactor Coolant Bleed Tank) to the Bleed position prior to ensuring a proper bleed path.
      1) Avoid lifting Letdown relief valves or damaging Waste Disposal Liquid System valve diaphragms

D. **Incorrect.**
Plausible if the candidate is unfamiliar with the location and/or operation of the Makeup Demineralizers. Since the demineralizers are in line prior to MU-V-8, and the line is dead-headed, there is no flow. Since there is no flow, the ΔP will actually lower to zero. If the candidate believes either that the demineralizers are downstream of the Bleed position path or that deadheading will cause ΔP to rise, then the candidate will select this choice.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 K4.05	
	Importance Rating	3.3	

K/A: Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: Interrelationships and design basis, including fluid flow splits in branching networks (e.g., charging and seal injection flow).

Proposed Question: RO Question # 37

Technical Reference(s): OP-TM-211-000, pg 3, Rev 026

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO-9

Question Source: Bank #

Modified Bank #

New X

Question History:		Last NRC Exam:	N/A

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

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(3) Mechanical components and design features of the reactor primary system.

Comments:
The KA is matched because the question requires the candidates to demonstrate the understanding of the makeup system flow paths, design features, interrelationships, and bases to include fluid flow splits in branching networks.

The question is at the Comprehension/Analysis cognitive level because the candidate must know the flowpath of the Makeup system and location of MU-V-8 in the system, the operation of the system when taken to the Bleed position, and system response when only MU-V-8 travels.

What MUST be known:
1. What is the physical layout of the Makeup System? 2. What is the flow path when MU-V-8 is taken to the "Bleed" position? 3. What are the precautions associated with placing MU-V-8 in a "Bleed" position without the Liquid Waste Disposal System being lined up to receive the bleed?

38 **ID: 978690** **Points: 1.00**

From the list below, identify the one signal that will initiate an automatic open signal for IC-V-20, RC Drain Tank Cooler cooling outlet valve.

A. RC Drain Tank pump running.

B. RC Drain Tank level greater than the high level alarm setpoint.

C. RC Drain Tank pressure greater than the high pressure alarm setpoint.

D. Both RC Drain Tank containment isolation valves open (WDL-V-304 and WDL-V-305).

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
IAW OP-TM-541-000, Primary Component Cooling, Section 6.0, System Information:

> 6.1 IC-V-20 opens when the RC Drain Tank Pump is running and closes when the pump is shutdown.

B. **Incorrect.**
Incorrect since the open signal is only from the drain tank pump running status and therefore is not associated with the RC Drain Tank Level directly. Plausible since high level could indicate a need for the RC Drain Tank Pump to run, which would open IC-V-20.

C. **Incorrect.**
Incorrect since the open signal is only from the drain tank pump running status and therefore is not associated with RC Drain Tank Pressure. Plausible since high pressure could indicate a need for drain tank cooling.

D. **Incorrect.**
Incorrect since the open signal is only from the drain tank pump running status and therefore is not associated with the containment isolation valves. Plausible since the most common evolution the CROs perform that opens IC-V-20 is pumping the drain tank, and these valves will be opened as part of pumping the drain tank.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	007	K4.01
Importance Rating	2.6	

K/A: Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: Quench tank cooling.

Proposed Question: RO Question # 38

Technical Reference(s): OP-TM-541-000, pg 11, Rev 18

Proposed References to be provided to applicants during examination: None

Learning Objective: 542-GLO-5

Question Source: Bank # IR-542-GLO-5-Q02

Modified Bank #

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 4

55.43

(4) Secondary coolant and auxiliary systems that affect the facility.

Comments:
The KA is matched because the question requires the candidates to demonstrate the knowledge of quench tank interlocks associated with the Pressurizer Relief Tank/Quench Tank system.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must know open interlocks for plant valves.

What MUST be known:
1. What are the interlocks associated with IC-V-20?

39 **ID: 978689** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Event:

- A Large Break LOCA has occurred.
- No HPI or LPI pumps are operating.
- PI 949A RCS Press indicates 10 psig.
- PI 981A RB Press indicates 15 psig.
- C4006 Avg of the 5 highest incore Thermocouples indicates 285°F.

GIven the above information, what is the status of the primary plant?

A. Approximately 18°F Subcooled.

B. Approximately 45°F Subcooled.

C. Approximately 18°F Superheated.

D. Approximately 45°F Superheated.

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
See calculation for "C". Plausible if candidate mixes signs (misunderstands that negative values indicate superheat).

B. **Incorrect.**
See calculation for "C". Plausible if candidate does not add RB pressure to RCS pressure to determine actual RCS saturation temperature and mixes signs (misunderstands that negative values indicate superheat).

C. **Correct.**
IAW OS-24
MANUAL RCS SCM / SUPERHEAT CALC
DETERMINE INCORE subcooled or superheat conditions as follows:

1. Hot leg pressure indication (PI-949A or RC3-PR Ch. 3) **10** psig

2. If indicated RCS Pressure is less than 500 psig and RB pressure is greater than 4 psig, then
a) record RB Pressure (PI-981A or PI-982A) **15** psig
b) add Hot Leg Press (step 1) + RB press (step 2a) **25** psig

3. Determine saturation temperature for RCS pressure (step 2b or if step 2b is NA, then use step 1 value) using Attachment G **267.3** F

4. If available, average temperature of 5 highest incore thermocouples (C4006) or if PPC is not available, highest BIRO thermocouple indication **285** F

5. Saturation Temperature (Step 3) MINUS Incore Temperature (Step 4) **-17.7 F**
Negative values indicate degrees of superheat
Positive values indicate degrees of subcooling

D. **Incorrect.**
See calculation for "C". Plausible if candidate does not add RB pressure to RCS pressure to determine actual RCS saturation temperature.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	006	K5.09	
	Importance Rating	3.7		

K/A: Knowledge of the operational implications of the following concepts as they apply to ECCS: Thermodynamics of water and steam, including subcooled margin, superheat,and saturation.

Proposed Question: RO Question # 39

Technical Reference(s): OS-24, pg 40, Rev 024

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-024-PCO-1

Question Source: Bank #

Modified Bank # ST-EOP-024-PCO-1-Q01

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments:
The KA is matched because the question requires the candidate to calculate a superheat condition associated with the ECCS.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze plant conditions and be able to calculate Superheat conditions.

What MUST be known:
1. What conditions require that RB pressure be added to RCS pressure for Superheat calculations?
2. How to calculate Superheat following an LOCA/ESAS situation?
3. Whether conditions are Superheated or Saturated.

Original Question: ST-EOP-024-PCO-1-Q01

Plant Conditions:
- A Large Break LOCA has occurred.
- PI 949A RCS Press indicates 10 psig.
- PI 981A RB Press indicates 15 psig.
- C4006 Avg of the 5 highest incore Thermocouples 280 °F.

Based on the above conditions, RCS Superheat would be _________ °F.

A. 0

B. 13

C. 28

D. 68

Answer: B

40 **ID: 978688** **Points: 1.00**

Plant Conditions:

- The Reactor is at 70% power.
- Due to a #1 Seal failure on RC-P-1C, "C" Reactor Coolant Pump, the crew is currently in OP-TM-AOP-040, RCP #1 Seal Failure.
- RC-P-1C is secured.

Event:

- A failure within 1A Inverter occurs.

Given the above information, which one of the following describes:
(1) The response of the RPS System, and
(2) The operational implications with the associated RPS response?

A. (1) All RPS Channels trip.
(2) Prevents excessive core coolant temperatures from occuring.

B. (1) All RPS Channels trip.
(2) Prevents the minimum core Departure from Nucleate Boiling Ratio (DNBR) from decreasing.

C. (1) RPS Channel A trips, ONLY.
(2) Prevents excessive core coolant temperatures from occuring.

D. (1) RPS Channel A trips, ONLY.
(2) Prevents the minimum core Departure from Nucleate Boiling Ratio (DNBR) from decreasing.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**

Part 1 is correct. A loss of the 1A inverter will cause a loss and trip of RPS Channel A but will also de-energize the RC-P-1A pump power monitor. IAW TQ-TM-104-641-C001:

Four Vital Busses (VBA,VBB,VBC,VBD) power:
Eight RC Pump power monitor channels

Two-Out-Of-Four Logic -
The tripping of two or more RPS channels actually causes all four RPS channels (reactor trip modules) to trip. Contacts opening on a channel trip string will deenergize the associated trip relay (KA, KB, KC, KD).

With that monitor de-energized, ALL RPS cabinets will see RC-P-1A and 1C de-energized and with reactor power greater than 55% (70% was given in the stem) and only one RCP sensed by RPS to be running in each loop, all RPS channels trip and a reactor trip occurs. IAW TQ-TM-104-641-C001:

Flux/Pumps
The RC pump contact monitor signal and a signal proportional to the total reactor power are received by a power/pumps trip bistable.
The reactor is tripped when the number of operating pumps in each loop does not correspond to the number of reactor coolant pumps required to be in operation for the existing reactor power.

Part 2 is incorrect. Plausible if candidate mistakes the bases for the RCS High Temperature RPS trip as the bases for the Power to Pumps RPS trip.

High RCS Temperature
The high reactor coolant outlet temperature trip setting limit has been established to prevent excessive core coolant temperature.

B. **Correct.**

Part 1 is correct. A loss of the 1A inverter will cause a loss and trip of RPS Channel A but will also de-energize the RC-P-1A pump power monitor. IAW TQ-TM-104-641-C001:

Four Vital Busses (VBA,VBB,VBC,VBD) power:
Eight RC Pump power monitor channels

Two-Out-Of-Four Logic -
The tripping of two or more RPS channels actually causes all four RPS channels (reactor trip modules) to trip. Contacts opening on a channel trip string will deenergize the associated trip relay (KA, KB, KC, KD).

With that monitor de-energized, ALL RPS cabinets will see RC-P-1A and 1C de-energized and with reactor power greater than 55% (70% was given in the stem) and only one RCP sensed by RPS to be running in each loop, all RPS channels trip and a reactor trip occurs. IAW TQ-TM-104-641-C001:

Flux/Pumps
The RC pump contact monitor signal and a signal proportional to the total reactor power are received by a power/pumps trip bistable.
The reactor is tripped when the number of operating pumps in each loop does not correspond to the number of reactor coolant pumps required to be in operation for the existing reactor power.

Part 2 is correct. IAW TQ-TM-104-641-C001:
Flux/Pumps
The purpose of this trip function is to prevent the minimum core DNBR from decreasing by tripping the reactor due to loss of one or more reactor coolant pumps that restricts the maximum power as a function of the number of RC pumps running in a loop. This is the DNBR protection trip for conditions of multiple RC pump coast-down or a single RC pump coast-down from partial pump operation.

C. **Incorrect.**

Part 1 is incorrect. Plausible if candidate does not recognize that a loss of the 1A inverter will cause a loss and trip of RPS Channel A but will also de-energize the RC-P-1A pump power monitor. With that monitor de-energized, ALL RPS cabinets will see RC-P-1A and 1C de-energized and with reactor power greater than 55% and only one RCP sensed by RPS to be running in each loop, all RPS channels trip and a reactor trip occurs. IAW TQ-TM-104-641-C001:

Four Vital Busses (VBA,VBB,VBC,VBD) power:
Eight RC Pump power monitor channels

Two-Out-Of-Four Logic -
The tripping of two or more RPS channels actually causes all four RPS channels (reactor trip modules) to trip. Contacts opening on a channel trip string will deenergize the associated trip relay (KA, KB, KC, KD).

Part 2 is incorrect. Plausible if candidate mistakes the bases for the RCS High Temperature RPS trip as the bases for the Power to Pumps RPS trip.

7. High RCS Temperature
The high reactor coolant outlet temperature trip setting limit has been established to prevent excessive core coolant temperature.

D. **Incorrect.**

Part 1 is incorrect. Plausible if candidate does not recognize that a loss of the 1A inverter will cause a loss and trip of RPS Channel A but will also de-energize the RC-P-1A pump power monitor. With that monitor de-energized, ALL RPS cabinets will see RC-P-1A and 1C de-energized and with reactor power greater than 55% and only one RCP sensed by RPS to be running in each loop, all RPS channels trip and a reactor trip occurs. IAW TQ-TM-104-641-C001:

Four Vital Busses (VBA,VBB,VBC,VBD) power:
Eight RC Pump power monitor channels

Two-Out-Of-Four Logic -
The tripping of two or more RPS channels actually causes all four RPS channels (reactor trip modules) to trip. Contacts opening on a channel trip string will deenergize the associated trip relay (KA, KB, KC, KD).

Part 2 is correct. IAW TQ-TM-104-641-C001:

3. Flux/Pumps
The purpose of this trip function is to prevent the minimum core DNBR from decreasing by tripping the reactor due to loss of one or more reactor coolant pumps that restricts the maximum power as a function of the number of RC pumps running in a loop. This is the DNBR protection trip for conditions of multiple RC pump coast-down or a single RC pump coast-down from partial pump operation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012	K5.01
	Importance Rating	3.3	

K/A: Knowledge of the operational implications of the following concepts as the apply to the RPS: DNB.

Proposed Question: RO Question # 40

Technical Reference(s): TQ-TM-104-641-C001, pg 18,19,43,44,71, Rev 002

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO-8

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to demonstrate the knowledge of the operational implications of departure from nucleate boiling as they apply to the Reactor Protection System.

The question is at the Comprehension/Analysis cognitive level because the candidate must be able to analyze plant conditions and determine response of the plant, response of the RPS system along with trip setpoints, and know the safety analysis bases for tripping the plant under the conditions listed.

What MUST be known:
1. What is the relationship between a Loss of an Inverter and the Reactor Protection System?
2. What is the Power to Pumps RPS Trip Setpoint for a 1/1 combination?
3. What is the reason for providing a Power to Pumps RPS trip?

41 **ID: 978684** **Points: 1.00**

Plant Conditions:

- A 5-day Late-Cycle outage has just completed.
- An approach to criticality is in progress IAW 1103-8, Approach to Criticality.

While withdrawing Control Rod Groups, which one of the following will result in the critical rod height being LOWER then the predicted value in the Estimated Critical Rod Position calculation?

A. Turbine header pressure setpoint fails to 600 psig.

B. Make up to the RCS with 20 gallons from the BAMT.

C. Feedwater temperature is 220°F vs. the expected 200°F.

D. Tave at 534°F due to reduction in Turbine Chest Warming.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
The Turbine Bypass valves are in AUTO IAW IAW 1103-8, Approach to Criticality:

> 3.2.6 Verify Turbine Bypass Valves are in AUTO and RCS average temperature is between 530F and 534F.

Turbine header pressure failing to 600 psig will cause the Turbine Bypass Valves to fail open, reducing RCS temperature, which will cause the actual rod height at criticality to be lower than predicted. OTSG Temperature is controlled by OTSG pressure, which is controlled by TBVs. OPEX related. IAW 11201360, Reactivity Management:

- SER 24-91, Inadequate Control of Reactivity Changes
  - Monticello (BWR) June 1991
    - Inadequate control of a reactor shutdown and inattention to reactor power resulted in a challenge to the Reactor Protection System and a plant transient.
    - Operators did not expect that a significant reactor coolant temperature reduction would have a positive reactivity affect, and thus an overpower condition. The significant temperature reduction was caused by inadequate control during the reactor shutdown, and operators did not expect a recriticality to occur during a reactor shutdown.
    - Root Cause: Inattention to plant conditions during operations that significantly or potentially affect reactivity.

B. **Incorrect.**
Plausible since if the examinee misinterprets the affect of adding boron to the RCS. The critical rod height will be higher than calculated due to the negative reactivity addition. IAW 11201360, Reactivity Management:

- Boron
  - The presence of Boron produces a negative reactivity contribution.

C. **Incorrect.**
Plausible if the examinee does not know feedwater temperature does not affect the ECP unless it results in higher or lower RCS temperature. In this situation FW temperature will not have an effect on the ECP because OTSG water levels are being maintained at Low Level Limits. OTSG Temperature is controlled by OTSG pressure, which is controlled by TBVs. IAW 1103-8, Approach to Criticality:

3.2.6 Verify Turbine Bypass Valves are in AUTO and RCS average temperature is between 530F and 534F.

D. **Incorrect.**
Plausible if the examinee does not know reference value for Tave at criticality is 532°F. A higher temperature would cause critical rot height to be higher than calculated. OPEX related distractor. IAW 11201360, Reactivity Management:

- SER 24-91, Inadequate Control of Reactivity Changes
  - Monticello (BWR) June 1991
    - Inadequate control of a reactor shutdown and inattention to reactor power resulted in a challenge to the Reactor Protection System and a plant transient.
    - Operators did not expect that a significant reactor coolant temperature reduction would have a positive reactivity affect, and thus an overpower condition. The significant temperature reduction was caused by inadequate control during the reactor shutdown, and operators did not expect a recriticality to occur during a reactor shutdown.
    - Root Cause: Inattention to plant conditions during operations that significantly or potentially affect reactivity.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	039	K5.08	
	Importance Rating	3.6		

K/A: Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity.

Proposed Question: RO Question # 41

Technical Reference(s): 1103-8, pg 5, Rev 054
11201360, pg 22,23, Rev 14

Proposed References to be provided to applicants during examination: None

Learning Objective: GOP-003-PCO-5

Question Source:	Bank #	IR-GOP-003-PCO-5-Q1
	Modified Bank #	
	New	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	1
	55.43	

(1) Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Comments: The KA is matched because the question requires the candidates to demonstrate the knowledge and implications of the effect of a change in steam removal on reactivity.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze how a lowering turbine header pressure will affect RCS temperature and, therefore, how the estimated critical position will be affected.

What MUST be known:
1. How does a lower Turbine Header Pressure affect RCS temperature while the OTSG's are on Low Level Limits? 2. How does a change in RCS temperature affect critical rod height during an Approach to Criticality?

42 **ID: 978683** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Event:

- The following data is observed on the "A" Reactor Coolant Pump, RC-P-1A:
  - Number One Seal Leak-off flow is 1.8 gpm and steady.
  - Periodic motor stand vibration alarms are occurring however the alarm can be reset without immediately alarming again.
  - Bentley-Nevada vibration is reading in the range between 14 and 18 mils.
  - Number one Seal leak-off temperature is 134°F and steady.
  - Radial Bearing temperature is 124°F and steady.
  - PPC Alarm Point L2755, RC-P-1A Standpipe Level -HI, is in alarm.
  - Reactor Coolant Drain Tank, RCDT, level is rising at 2GPM.

Given the above information, the Number ___(1)___ Seal leak-off flow is excessive, and ___(2)___ is required to be performed.

A. (1) Two
(2) shutting down RC-P-1A

B. (1) Two
(2) opening MU-V-38, Seal #1 Bypass Flow Isolation Valve

C. (1) Three
(2) shutting down RC-P-1A

D. (1) Three
(2) opening MU-V-38, Seal #1 Bypass Flow Isolation Valve

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
Part 1 is correct. The indications given are indicative of excessive Number 2 Seal leak-off flow. IAW OP-TM-226-000, Reactor Coolant Pumps, Section 2.2, Limitations:

> 2.2.8: Number 2 Seal Leak-Off flow is excessive at normal operating pressure, as evidenced by RCDT level rise is > 1 gpm attributable to a RCP and one of the following conditions exist on that RCP:
> - High Standpipe Level alarm
> - RCP Pump or Motor vibration rising

Part 2 is correct. A RCP must be shutdown if the Number Two Seal is failing. IAW OP-TM-226-000, Reactor Coolant Pumps, Section 2.2, Limitations:

> 2.2.8 To avoid or limit component damage, shutdown the affected RC Pump for any of the following:
> - Total Loss of Seal Injection and Intermediate Cooling.
> - Loss of NS cooling flow (All RCPs for a total loss of NS to RB)
> - Motor bearing Upper or Lower Guide temperatures exceed 185°F. (195°F for RC-P-1C)
> - Motor Thrust bearing (Up or Down) temperatures exceed 200°F. (195°F for RC-P-1C)
> - Motor winding temperature exceeds 302°F (150°C).
> - Pump bearing temperature exceeds 225°F.
> - Number 1 Seal Inlet temperature exceeds 235°F.
> - Number 1 Seal Leak-Off flow is > 6 gpm at normal operating pressure
> - Number 1 Seal Leak-Off flow is < 0.8 gpm at normal operating pressure
> - Number 2 Seal Leak-Off flow is excessive at normal operating pressure, as evidenced by RCDT level rise is > 1 gpm attributable to a RCP and one of the following conditions exist on that RCP:
>   - High Standpipe Level alarm
>   - RCP Pump or Motor vibration rising
> - Pump Vibration: exceeds 20 mils with 4-pump operation, or 30 mils with single pump operation.
> - Motor vibration exceeds 7 mils.

B. **Incorrect.**

Part 1 is correct. The indications given are indicative of excessive Number 2 Seal leak-off flow. IAW OP-TM-226-000, Reactor Coolant Pumps, Section 2.2, Limitations:

> 2.2.8: Number 2 Seal Leak-Off flow is excessive at normal operating pressure, as evidenced by RCDT level rise is > 1 gpm attributable to a RCP and one of the following conditions exist on that RCP:
> - High Standpipe Level alarm
> - RCP Pump or Motor vibration rising

Part 2 is incorrect. Opening MU-V-38 while at power used to be a common practice, but the industry and TMI have learned from OPEX to never open the Number 1 Seal Bypass Valve while at power. If MU-V-38 were opened during these conditions, the symptoms could very well get worse. Plausible if the candidate is unsure of the purpose of MU-V-38 and believes that Bypassing the Number One Seal will lower how much Seal Injection enters the Number Two Seal. IAW TQ-TM-104-226-C001, Reactor Coolant Pumps and Motors:

> Number 1 Seal – Bypass
>
> 1) During normal operation, this bypass valve should remain closed. The practice of periodically opening this valve during normal (Hot) operation increases risk of thermal shocking pump shaft, bearing, and seal during a loss of injection water. This is because the thermal barrier heat exchanger has a limited capacity, which could be exceeded by the sum of normal seal leakage plus bypass flow at high pressure. Therefore, the seal bypass valve should never be opened during a loss of injection water.
>
> 2) Opening the bypass valve during normal (hot) operation may also cause a high leakrate condition for the #2 seal due to pressure transients in the #1 seal leakoff line.
>
> 3) Opening the #1 seal bypass valve has occasionally resulted in lifting and jamming either the #1 or #2 seal ring. The purpose of the bypass line is to allow for additional seal injection water flow through the pump bearing for cooling purposes. The #1 seal bypass valve should be opened only if additional cooling of pump bearing is needed
>
> 4) The #1 seal bypass valve, MU-V-38, should not be opened unless all of the following conditions exist:
>
> a) The #1 radial pump bearing temperature approaches its alarm setpoint (225 °F) or #1 seal inlet temperature approaches its alarm setpoint (235 °F).
>
> b) Reactor coolant system is >100 PSIG but <1000 PSIG.
>
> c) The #1 seal leakoff flowrate is <1.0 GPM.

C. **Incorrect.**
Part 1 is incorrect. Plausible because the Number 3 Seal drains directly to the RDCT and bypasses the Standpipe. Plausible if the candidate does not recognize the other indications associated with the Number 2 Seal. Incorrect since the Standpipe level is greater than the low level alarm setpoint (given in the stem). IAW OP-TM-226-00, Reactor Coolant Pumps, Section 2.2, Limitations:

> 2.7: To ensure proper RCP Number 3 seal operation, maintain RCP Standpipe level ≥ 24 inches. When local Standpipe levels are inaccessible, this can be assured by maintaining Low RCP Standpipe level PPC alarms clear (setpoint 36"). Operation with a low standpipe level alarm may continue provided other indications of RCP seal health are normal . The primary indicators of RCP seal health are the #1 seal leakoff flowrate indicator/recorder (MU-FR-43 on PCL) and the #1 seal inlet temperature indication (computer points A0525 through A0528).

Part 2 is correct. A RCP must be shutdown if the Number Two Seal is failing. IAW OP-TM-226-000, Reactor Coolant Pumps, Section 2.2, Limitations:

> 2.2.8 To avoid or limit component damage, shutdown the affected RC Pump for any of the following:
> - Total Loss of Seal Injection and Intermediate Cooling.
> - Loss of NS cooling flow (All RCPs for a total loss of NS to RB)
> - Motor bearing Upper or Lower Guide temperatures exceed 185°F. (195°F for RC-P-1C)
> - Motor Thrust bearing (Up or Down) temperatures exceed 200°F. (195°F for RC-P-1C)
> - Motor winding temperature exceeds 302°F (150°C).
> - Pump bearing temperature exceeds 225°F.
> - Number 1 Seal Inlet temperature exceeds 235°F.
> - Number 1 Seal Leak-Off flow is > 6 gpm at normal operating pressure
> - Number 1 Seal Leak-Off flow is < 0.8 gpm at normal operating pressure
> - Number 2 Seal Leak-Off flow is excessive at normal operating pressure, as evidenced by RCDT level rise is > 1 gpm attributable to a RCP and one of the following conditions exist on that RCP:
>   - High Standpipe Level alarm
>   - RCP Pump or Motor vibration rising
> - Pump Vibration: exceeds 20 mils with 4-pump operation, or 30 mils with single pump operation.
> - Motor vibration exceeds 7 mils.

D. **Incorrect.**
Part 1 is incorrect. Plausible because the Number 3 Seal drains directly to the RDCT and bypasses the Standpipe. Plausible if the candidate does not recognize the other indications associated with the Number 2 Seal. Incorrect since the Standpipe level is greater than the low level alarm setpoint (given in the stem). IAW OP-TM-226-00, Reactor Coolant Pumps, Section 2.2, Limitations:

> 2.7: To ensure proper RCP Number 3 seal operation, maintain RCP Standpipe level ≥ 24 inches. When local Standpipe levels are inaccessible, this can be assured by maintaining Low RCP Standpipe level PPC alarms clear (setpoint 36"). Operation with a low standpipe level alarm may continue provided other indications of RCP seal health are normal . The primary indicators of RCP seal health are the #1 seal leakoff flowrate indicator/recorder (MU-FR-43 on PCL) and the #1 seal inlet temperature indication (computer points A0525 through A0528).

Part 2 is incorrect. Opening MU-V-38 while at power used to be a common practice, but the industry and TMI have learned from OPEX to never open the Number 1 Seal Bypass Valve while at power. If MU-V-38 were opened during these conditions, the symptoms could very well get worse. Plausible if the candidate is unsure of the purpose of MU-V-38 and believes that Bypassing the Number One Seal will lower how much Seal Injection enters the Number Two Seal. IAW TQ-TM-104-226-C001, Reactor Coolant Pumps and Motors:

Number 1 Seal – Bypass

1) During normal operation, this bypass valve should remain closed. The practice of periodically opening this valve during normal (Hot) operation increases risk of thermal shocking pump shaft, bearing, and seal during a loss of injection water. This is because the thermal barrier heat exchanger has a limited capacity, which could be exceeded by the sum of normal seal leakage plus bypass flow at high pressure. Therefore, the seal bypass valve should never be opened during a loss of injection water.

2) Opening the bypass valve during normal (hot) operation may also cause a high leakrate condition for the #2 seal due to pressure transients in the #1 seal leakoff line.

3) Opening the #1 seal bypass valve has occasionally resulted in lifting and jamming either the #1 or #2 seal ring. The purpose of the bypass line is to allow for additional seal injection water flow through the pump bearing for cooling purposes. The #1 seal bypass valve should be opened only if additional cooling of pump bearing is needed

4) The #1 seal bypass valve, MU-V-38, should not be opened unless all of the following conditions exist:

   a) The #1 radial pump bearing temperature approaches its alarm setpoint (225 °F) or #1 seal inlet temperature approaches its alarm setpoint (235 °F).
   b) Reactor coolant system is >100 PSIG but <1000 PSIG.
   c) The #1 seal leakoff flowrate is <1.0 GPM.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003	K6.02
	Importance Rating	2.7	

K/A: Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: RCP seals and seal water supply.

Proposed Question: RO Question # 42

Technical Reference(s): OP-TM-226-000, pg 4, Rev 008

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-6

Question Source: Bank #

Modified Bank # QR5C06-01-Q01

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 3

55.43

(3) Mechanical components and design features of the reactor primary system.

Comments:
The KA is matched because the question requires the candidates to demonstrate the knowledge of the loss or malfunction of the #2 RCP seal and seal water supply on the RCPs.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze plant conditions, have knowledge of the RCP seal design and flowpaths, and actions to take if a RCP seal failure occurs.

What MUST be known:
1. What is the normal value of RCP #1 Seal Leakoff flow?
2. What causes a RCP Standpipe High Level alarm?
3. What causes the RCDT to rise?
4. What action is procedurally required if a RCP experiences a failure of the #2 Seal?

Original Question:

The following data is observed on Reactor Coolant Pump 1A (RC-P-1A):

- Number One Seal Leak-off flow is 1.8 gpm.
- Periodic motor stand vibration alarms are occurring however the alarm can be reset without immediately alarming again.
- Bentley-Nevada vibration is reading in the range between 14 and 18 mils.
- Number one Seal leak-off temperature is 134°F.
- Radial Bearing temperature is 124°F.
- The high standpipe level alarm is in.

Which of the following conditions could be causing the above indications?

A. Number One Seal abnormally open.

B. Number Two Seal abnormally open.

C. Number Three Seal abnormally open.

D. Low Seal injection flow.

Answer: B

43 **ID: 978666** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- Diesel Generator Fuel Oil Storage Tank, DF-T-1, level is 24,000 gallons.
- BWST volume is 375,000 gallons (57.0 ft).
- "A" Core Flood Tank, CF-T-1A, level is 11.5 ft (940 ft3).
- EG-V-15A, Air Start Header Isolation Valve, is failed closed.

Given the above information, which of the following components is/are INOPERABLE?

A. "A" Core Flood Tank, CF-T-1A.

B. Borated Water Storage Tank, DH-T-1.

C. Emergency Diesel Generator, EG-Y-1A, ONLY.

D. Both Emergency Diesel Generators, EG-Y-1A/EG-Y-1B.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
IAW Tech Spec 3.3.1.2, the minimum level for Core Flood Tanks is 940 +/- 30 ft3. Plausible if the candidate is not familiar with the Tech Spec limit.

3.3.1 The reactor shall not be made critical unless the following conditions are met:
3.3.1.2 Core Flooding System
a. Two core flooding tanks (CFTs) each containing 940 ± 30 ft3 of borated water at 600 ± 25 psig shall be available. Specification 3.0.1 applies.

B. **Incorrect.**
IAW Tech Spec 3.3.1.1.a, the minimum volume for the BWST is 350,000 gallons. Plausible if the candidate is not familiar with the Tech Spec limit.

3.3.1 The reactor shall not be made critical unless the following conditions are met:
3.3.1.1 Injection Systems
a. The borated water storage tank (BWST) shall contain a minimum of 350,000 gallons of water having a minimum concentration of 2,500 ppm boron at a temperature not less than 40°F. If the boron concentration or water temperature is not within limits, restore the BWST to OPERABLE within 8 hrs. If the BWST volume is not within limits, restore the BWST to OPERABLE within one hour. Specification 3.0.1 applies.

C. **Incorrect.**
With EG-V-15A closed, the "A" Emergency Diesel Generator will not start and therefore is rendered inoperable by definition IAW Tech Spec 1.3. However, the "B" Emergency Diesel Generator is also inoperable due to the Diesel Generator Fuel Oil Storage Tank level being below the Tech Spec limit of 25000 gallons. Plausible if the candidate recognizes the inoperability concern associated with EG-Y-15A but does not recognize the inoperability concern associated with the Diesel Generator Fuel Oil Storage Tank.

> 1.3 OPERABLE: A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

D. **Correct.**
Tech Spec 3.7.1.e states:

> Engineered safeguards diesel generators are operable and at least 25,000 gallons of fuel oil are available in the storage tank.

With the conditions given in the stem being 24, 000 gallons in the Diesel Generator Fuel Oil Storage Tank, neither Emergency Diesel Generator may be considered Operable.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064 K6.08	
	Importance Rating	3.2	

K/A: Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Fuel oil storage tanks.

Proposed Question: RO Question # 43

Technical Reference(s): T.S. 3.07, pg 3-42, Rev 224
T.S. 1.0, pg 1-2, Rev 175

Proposed References to be provided to applicants during examination: None

Learning Objective: 861-GLO-14

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History: Last NRC Exam: N/A

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

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	55.43	

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidate to demonstrate the knowledge of the loss or malfunction of the fuel oil storage tanks on the emergency diesel generator system.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must know Tech Spec LCO's.

What MUST be known:
1. What is the Tech Spec requirements for Diesel Generator Fuel Oil Storage Tank, DF-T-1?

44 **ID: 978664** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- The Secondary Services River Water Coolers are currently in the back-wash mode.
- River water temperature is 85°F.

Event:

- The ARO places the "NORM-OFF-BACKWASH" switch on CL to the "NORM" position.
  - SR-V-3, Secondary Closed Cooler Inlet Isolation Valve, remains CLOSED.
  - SR-V-6, Secondary Closed Cooler Outlet Isolation Valve, remains CLOSED.
  - SR-V-7, Secondary Closed Cooler Backwash Outlet Valve, travels full CLOSED.
  - SR-V-8, Secondary Closed Cooler Backwash Inlet Valve, travels full CLOSED.
- Secondary Services Closed Cooling Water system temperatures are rising.

Given the above information, which one of the following is an AUTOMATIC action that will occur?

A. Main Turbine runback due to high turbine bearing lube oil temperature.

B. Main Turbine-Generator runback due to high stator cooling temperature.

C. Main Feedwater Pumps trip due to high pump bearing lube oil temperature.

D. Condensate Booster Pumps trip due to high pump bearing lube oil temperature.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Although there are Automatic trips associated with the Main Turbine, this will not a cause an automatic trip of the turbine. Plausible if the candidate believes that there is an automatic action associated with the Main Turbine because the high lube oil temperature condition will occur.
IAW OP-TM-AOP-0331, Loss of Secondary Component Cooling Basis Document, Attachment 1, Effects of Loss of Secondary Closed Cooling:

Component: Main Turbine Turbine Oil Coolers
Effects of Loss of Cooling: Heat up of lubricating oil results in reduced viscosity and possible bearing damage.

B. **Correct.**
Rising temperatures are a result of the ability of Secondary River Water to cool Secondary Closed Cooling water through the Secondary Closed coolers. Rising temperatures will lead to entry into the Abnormal Operating Procedure. A loss of Secondary Closed Cooling Water occurs if the following criteria are met IAW OP-TM-AOP-033, Loss of Secondary Component Cooling, Section 1, Entry Conditions:

At least one Nuclear River or Secondary River pump is available and any of the following:

- SC cooler outlet temperature (A0322) approaching 100F
- SC pump discharge pressure <50 psig (SC-PI-149) and either of the following:
  - Exciter cooling air from cooler(A0148) approaching 55C
  - High temperature alarm on a component cooled by SC
- SC surge tank level (SC-LI-84) <1 ft and lowering

IAW OP-TM-AOP-0331, Loss of Secondary Component Cooling Basis Document, Attachment 1, Effects of Loss of Secondary Closed Cooling:

Component: Main Generator Stator Coolers
Effects of Loss of Cooling: Automatic runback if SC temperature is high. Potential main generator damage if stator temperature exceeds 82C.

C. **Incorrect.**
Although there are Automatic trips associated with Main Feedwater Pumps, this will not a cause an automatic trip of the pumps. Plausible if the candidate believes that there is an automatic action associated with Main Feedwater Pumps because the high lube oil temperature condition will occur. IAW OP-TM-AOP-0331, Loss of Secondary Component Cooling Basis Document, Attachment 1, Effects of Loss of Secondary Closed Cooling:

Component: Main FW Pump Turbine Oil Coolers
Effects of Loss of Cooling: Heat up of lubricating oil results in reduced viscosity and possible bearing damage.

D. **Incorrect.**
Although there are Automatic trips associated with Condensate Booster Pumps, this will not a cause an automatic trip of the pumps. Plausible if the candidate believes that there is an automatic action associated with Condensate Booster Pumps because the high lube oil temperature condition will occur. IAW OP-TM-AOP-0331, Loss of Secondary Component Cooling Basis Document, Attachment 1, Effects of Loss of Secondary Closed Cooling:

Component: Condensate Booster Pump and Motor bearing oil coolers
Effects of Loss of Cooling: Heat up of lubricating oil results in reduced viscosity and possible bearing damage.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	076	A1.02	
	Importance Rating	2.6		

K/A: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: Reactor and turbine building closed cooling water temperatures.

Proposed Question: RO Question # 44

Technical Reference(s): OP-TM-AOP-0331, pg 5,16, Rev 1

Proposed References to be provided to applicants during examination: None

Learning Objective: 313-GLO-11

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	4
	55.43	

(4) Secondary coolant and auxiliary systems that affect the facility.

Comments:
The KA is matched because the the question requires the candidate to demonstrate the ability to predict changes in reactor building and turbine building closed cooling water temperatures associated with operating the river water system.

The question is at the Comprehensive/Analysis cognitive level because the candidates to understand the results of closing certain valves within the Secondary Closed Cooling Water System and then must understand what automatic actions will occur based on those results.

What MUST be known:
1. What effect does closing the valves associated with Secondary Closed Coolers have on the Secondary Closed Cooling System? 2. What effect do rising Secondary Closed Cooling Water temperatures have on components cooled by the Secondary Closed Cooling Water System? 3. What automatic actions occur as a result of high cooling temperatures on secondary equipment?

45 **ID: 978663** **Points: 1.00**

Plant Conditions:

- The plant is operating at 5% power.
- Reactor Building Pressure is 0.5 PSIG and steady.
- Radiation Monitor RM-A-9 is out-of-service for repairs.
- 1C ES Valves MCC is selected to the 1P 480 volt bus.

Given the above information and assuming no change in outside air temperature, which ONE of the following will cause Reactor Building pressure to RISE?

A. Inadvertant actuation of "B" Train ESAS occurs.

B. Starting two additional Industrial Cooler Spray Pumps.

C. Commencing a cooldown from hot standby to hot shutdown conditions.

D. The running Reactor Compartment Cooling Fan, AH-E-2A, trips on overload.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**

When 1C ES valves is selected to 1P 480V Bus and only B ESAS actuates, Reactor Building Cooling Fan, AH-E-1C, will trip and not restart. IAW OP-TM-823-000, Reactor Building Heating and Ventilation System, Section 6.0 SYSTEM INFORMATION

> 6.2 When 1C ES valves is selected to 1P 480V Bus and only B ESAS actuates, AH-E-1C will trip if running in fast speed and will **not** automatically restart in slow speed in response to the ESAS signal. Manual start capability in slow speed is maintained.

With Reactor Building Cooling lowered, Reactor Building temperature will rise and therefore Reactor Building pressure will rise as well. IAW TQ-TM-104-240-C001, Containment System, Section XIV.C.1:

> 1. RB atmospheric effects: Anything that can cause a change in temperature inside the Reactor Building will cause a parallel (in the same direction) change in pressure. RB pressure critical operation band is very small (-1 to +2 PSI per Tech Specs) and therefore should be monitored closely. Some potential RB pressure changing events include, but are not limited to, the following: (note: there are numerous failures that can cause many of these events)
>
> a. RB Ventilation Fan shifts
> b. Weather conditions (RB Cooling capability)
> c. Reactor power level changes
> d. Plant heatup/cooldowns
> e. RB Purge shifts
> f. Time of Day – night time tends to cool off and lower RB pressure.
> g. Seasonal changes (i.e. air/river temperature)
> h. Leakage from RCS/steam/feed lines
> i. RCP starts/stops/trips

B. **Incorrect.**

When two additional Industrial Cooler Spray Pumps are started, more heat is removed from the Reactor Building through the normal RB cooling system. Plausible if the candidate is not familiar with the purpose of the Industrial Cooler Spray Pumps and 1) believes that starting the pumps will cause heat generation in the Reactor Building, and 2) correlates the relationship of temperature and pressure in the Reactor Building and therefore, believes that starting more equipment in the Reactor Building may cause pressure to rise. IAW OP-TM-823-440, Controlling RB Air Temperature, Note prior to step 4.1.3:

> Note: If all available fans for an individual cooler are inservice and RB average air temperature approaches or exceeds the upper limit it is permissible to start additional spray pumps as required to maintain temperatures within the desired band.

C. **Incorrect.**

Changing from Hot Standby to Hot Shutdown is performed IAW 1102-10, Plant Shutdown, which states in Enclosure 1, Actions Prior to a Planned Reactor Shutdown:

> f. RB Purge (OP-TM-823-406) and Kidney filter operation (OP-TM-823-404) should be initiated to improve the RB environment for personnel entry.

However, a purge cannot be commenced due to RM-A-9 being out of service (given in the stem). IAW OP-TM-823-406, RB Purge - Containment Closed, Section 3.3, Prerequisites:

> 3.3.10 **VERIFY** RM-A-9 noble gas channel is operable with interlock defeat switch in NORMAL.

D. **Incorrect.**
When AH-E-2A trips, AH-E-2B will auto start on interlock (even if AH-E-2B is in Pull-To-Lock). This would be an even swap of running equipment in the Reactor Building and therefore temperature should not change. Plausible if the candidate is not aware of the interlocked auto-start feature. IAW OP-TM-823-000, Reactor Building Heating and Ventilation System, Section 2.1, Precautions:

> 2.1.9 Reactor Compartment Cooling Fans (AH-E-2A/B) have an auto start feature. This feature starts the idle fan (even if in pull to lock) if the running fan stops on motor overload.

Examination Outline Cross-reference:

Level	RO		SRO
Tier #	2		
Group #	1		
K/A #	103	A1.01	
Importance Rating	3.7		

K/A: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: Containment pressure, temperature, and humidity.

Proposed Question: RO Question # 45

Technical Reference(s): TQ-TM-104-240-C001, pg 67,68, Rev 005
OP-TM-823-000, pg 7, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: 240-GLO-8

Question Source: Bank # IR-240-GLO-8-Q02
Modified Bank #
New

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 5

55.43

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:
The KA is matched because the question requires the candidate to predict the changes in Containment Pressure and Temperature based on equipment/condition changes in the Reactor Building. Additionally, the change referenced is in the increasing direction, which would drive towards exceeding design limits.

The question is at the Comprehension/Analysis cognitive level because the candidate must recall the loads associated with 1P 480V bus, understand which of those loads is affected by a "B" Train ESAS actuation, and the analyze what effect the loss of the equipment will have on Pressure in the Reactor Building.

What MUST be known:
1. What are the loads powered from 1C ES Valves? 2. What effect does a "B" ESAS actuation have on equipment withing the Reactor Building? 3. What is the effect on Reactor Building Temperature and Temperature upon a loss of AH-E-1C?

46 **ID: 978662** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- "A" Intermediate Service Cooler, IC-C-1A, is in service.
- "A" Intermediate Closed Cooling Pump, IC-P-1A, is operating.
  - IC-C-1A outlet water temperature is 90°F and steady.
  - IC-C-1A Cooling Outlet Valve, NR-V-15A, is throttled partially open.
- Susquehanna River Water temperature is 32°F and steady.
- "A" and "C" Nuclear Service River Water Pumps, NR-P-1A and NR-P-1C, are operating.

Event:

- Both Letdown Cooler Isolation Valves, MU-V-1A and MU-V-1B, have closed due to a false actuation of an automatic closure interlock.
- ICCW outlet water temperature has rapidly dropped to 80°F, and is continuing to lower at 1°F per minute.

Given the above information, identify the one selection below that describes the operational impact of the false actuation and the action taken IAW OP-TM-541-461, IC & NS Temperature Control, to mitigate the consequences of the event.

A. A severe reduction in RCP radial bearing temperatures will require NR-P-1A or NR-P-1C to be secured.

B. Formation of condensate on CRD stator water jackets will require NR-V-15A to be throttled closed.

C. Thermal shock to Letdown Coolers, MU-C-1A and MU-C-1B, will require Letdown Cooler Isolation Valves, IC-V-1A and IC-V-1B, to be closed.

D. IC-T-1, ICCW Surge Tank, level will reduce to below the lower sight glass connection requiring IC-V-5, Demin Water to ICCW Surge Tank Valve, to be opened.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
This would be true if RCP Seal Injection was lost. However, nothing in the stem lends to believe that RCP seal injection flow is not available. Water being cooled by the thermal barrier heat exchangers (cooled by ICCW) is flowing into the RCS, rather than past the pump radial bearings. Distracter is plausible because this would occur if RCP seal injection flow is lost. Examinee is required to know physical relationships between seal injection point, labyrinth seals, thermal barrier heat exchangers, pump radial bearings, etc. associated with RCP seal packages. Additional plausibility is merited since the action stated would reduce heat sink cooling flow through IC-C-1A.

B. **Correct.**
IAW OP-TM-541-000, Primary Component Cooling, Section 2, Precautions and Limitations:

> 2.2.10 To minimize the possibility of forming condensate in the CRD stator water jacket and provide adequate CRD stator cooling, IC outlet temperature should be mainatined between 90F and 100F on IC-6TI (CR).

Since the ICCW outlet temperature in the stem's Event given is less than 90F, the possibility exists to form condensate in the CRD stator water jacket.

The action to recover from the low outlet temperature is found within OP-TM-541-461, IC & NS Temperature Control which provides guidance for, among other things, IC & NS cooling water system temperature control. OP-TM-541-461, Section 4.0, Main Body:

> 4.1.4 If IC-C-1A is available, then throttle NR-V-15A to maintain IC cooler outlet temperature IC-6TI (CR) between 90F and 100F.

C. **Incorrect.**
the Correct concern is described in ICCW System Limits and precautions. These valves are interlocked with MU-V-1A/1B and are required to be open in order to re-open MU-V-1A/1B. Distracter is plausible because ICCW temperature will reduce significantly if left unchecked, but the procedure guidance for temperatures being out of band is to throttle NR-V-15A.

D. **Incorrect.**
Although the water density will change, it will result in much less level reduction. Distracter is plausible because reduction in ICCS temperature will result in shrinkage and subsequent surge tank level reduction but the procedure guidance for temperatures being out of band is to throttle NR-V-15A.. Additional plausibility is merited since the action stated is appropriate for low surge tank level.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	008	A2.08	
	Importance Rating	2.5		

K/A: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effects of shutting (automatically or otherwise) the isolation valves of the letdown cooler.

Proposed Question: RO Question # 46

Technical Reference(s): OP-TM-541-000, pg 4, Rev 18
OP-TM-541-461, pg 3, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective:	531-GLO-9		
Question Source:	Bank #	IR-531-GLO-9-Q01	
	Modified Bank #		
	New		
Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	5	
	55.43		

(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:
The KA is matched because the question requires the candidates to (a) predict the impact to components cooled by the Intermediate Closed Cooling Water System upon a closure of the Letdown Coolers, and then to (b) identify the procedure action to recover from the impact.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall a Precaution and Limitation associated with a sudden lowering of cooling water temperature and then recall the procedure action taken to mitigate the event.

What MUST be known:
1. What is the impact/concern associated with Intermediate Closed Cooling Water temperature being too low? 2. What is the procedure guidance to take upon a low Intermediate Closed Cooling Water outlet temperature?

47 **ID: 978634** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- EE-INV-1A-SW1, Manual Bypass Switch on Inverter 1A, is in the "Bypass To Load" position.
- The following conditions exist at Inverter 1A:
  - The "TRA Supplying ATA" light is lit.
  - The "IN SYNC" light is NOT lit.

Event:

- EE-INV-1A-SW1, Manual Bypass Switch on Inverter 1A, is placed in the "Normal Operation" position.

Given the above information and assuming no bulbs are burned out, what will be the result of the above event IAW 1107-2B, 120 Volt Vital Electrical System?

A. Damage to equipment will occur due to TRA and Inverter 1A being parallelled out of phase.

B. ATA will continue to be powered from TRA due to an interlock requiring the "IN SYNC" light to be lit.

C. ATA will be powered from Inverter 1A due to EE-INV-1A-SW1 being a "Break-Before-Make" style switch.

D. ATA will be powered by both Inverter 1A and TRA in parallel until the "IN SYNC" light is lit due to EE-INV-1A-SW1 being a "Make-Before-Break" style switch.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
IAW 1107-2B, 120 Volt Vital Electrical System, Section 3.19.2:

3.19.2 Operating the Inverter 1A Manual Bypass Switch
NOTE: EE-INV-1A-SW1, Manual Bypass Switch on Inverter 1A, is a Make-Before-Break manual switch that is normally used to isolate the Static Switch without loss of power to ATA.

3.19.2.1 Prerequisites

CAUTION: To avoid paralleling sources out of sync, TRA must be supplying ATA via the static switch, and the IN SYNC light must be energized.
1. ENSURE the TRA Supplying ATA Light on 1A Inverter is Lit.
2. ENSURE the IN SYNC Light on 1A Inverter is Lit.

3.19.2.2 Procedure
1. If the Manual Bypass Switch will be transferred from the Static Switch output to TRA, then ROTATE EE-INV-1A-SW1, Manual Bypass Switch on Inverter 1A, to Bypass to Load.
2. If the Manual Bypass Switch will be transferred from TRA to the Static Switch output, then ROTATE EE-INV-1A-SW1, Manual Bypass Switch on Inverter 1A, to Normal Operation.

B. **Incorrect.**
There is no interlock associated with the IN SYNC light and EE-INV-1A-SW1, Manual Bypass Switch on Inverter 1A that will prevent TRA and Inverter 1A from being parallelled out of phase. Plausible because there are breakers that the candidates operate in the Control Room that will not close unless both sources being parallelled are in sync.

C. **Incorrect.**
The Inverter 1A Manual Bypass Switch is a Make-Before Break" style switch on Inverter !A and is designed to swap power supplies between Inverter 1A and TRA without a loss of power to ATA. The ATA Manual Transfer Switch is a "Break-Before-Make" style switch and will also select a power supply of either Inverter 1A or TRA but will cause a loss of ATA during the transfer. Plausible if the candidate is not familiar with the function of each switch.

D. **Incorrect.**
Although EE-INV-1A-SW1 is a "Make-Before-Break" style switch, the "IN SYNC" light should be lit before the switch is manipulated to avoid parallelling out of phase. Plausible if the candidate is not familiar with the operation of the switch or does not understand the prerequisites to operate the switch.

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	062	A2.03
Importance Rating	2.9	

K/A: Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of improper sequencing when transferring to or from an inverter.

Proposed Question: RO Question # 47

Technical Reference(s): 1107-2B, pg 127, Rev 033A

Proposed References to be provided to applicants during examination: None

Learning Objective: 740-GLO-10

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	7
	55.43	

7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidate to predict the impact of improper sequncing when transferring to an inverter. Part 2 of the K/A is not directly met because there is no procedural guidance to recover from damaging equipment by paralleling an inverter out of phase. However, the High-Cog portion of the K/A has been met (part 1) and the operators must have an understanding of the procedure to avoid the scenario altogether (the procedure warns against the damage and therefore is used to answer Part 1 of the K/A).

The question is at the Comprehension/Analysis cognitive level because the candidate must have knowledge of various switches and indications on the AC inverters, the operation of each switch and indication, and analyze the specific event to determine the system response.

What MUST be known:
1. What are the interlocks associated with Inverter 1A?
2. What are the interlocks associated with TRA?
3. What will be the outcome of parallelling an Inverter out of phase?

48 **ID: 978633** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- The crew has entered OP-TM-AOP-023, "A" DC System Failure, due to lowering battery voltage.
- The Reactor Operator at the battery ground detector reports that "A" and "C" battery voltages are both 115vdc and lowering slowly.
- A battery ground is suspected.

Sequence of Events:

- Time = 0 minutes:
  - The Reactor Operator places the 1A Battery Ground Detector to “HAND”.
  - The following indications are observed when the selector switch is moved from the "OFF" position:
    - When the selector switch is taken to the “P” position, the analog meter reads Infinity.
    - When the selector switch is taken to the “PN” position, the analog meter reads zero (0) to the right.
- Time = 1 minute:
  - The Reactor Operator at the battery ground detector reports A and C battery voltages are both 109vdc and lowering slowly

Given the above information, which one of the following describes:
(1) The location of the battery ground, and
(2) The crew’s next required action?

A. (1) P Bus.
(2) Trip the Reactor.

B. (1) P Bus.
(2) Deenergize the A DC System.

C. (1) N or PN Bus.
(2) Trip the Reactor.

D. (1) N or PN Bus.
(2) Deenergize the A DC System.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
Part 1 is Correct. During Hand Operation of the Battery Ground Detection System any current flow and meter deflection shows a ground on one of the buses NOT SELECTED. IAW TQ-TM-104-734-C001, Vital ACDC Systems:

2) Manual ground reading
a) Manual battery ground detector intentionally puts a ground on the system and looks for current flow. No current flow means that the ground detector is the only ground.
b) Obtain Manual Ground readings:
(1) Place selector in hand - removes auto ground function, enables hand
(2) Select bus - places meter between a ground and selected bus
(3) Take reading, any current flow shows a ground on one of the buses NOT SELECTED

Part 2 is Correct. IAW OP-TM-AOP-023, A DC System Failure, when battery voltage is < 110Vdc the Reactor is tripped.

B. **Incorrect.**
Part 1 is Correct. During Hand Operation of the Battery Ground Detection System any current flow and meter deflection shows a ground on one of the buses NOT SELECTED. IAW TQ-TM-104-734-C001, Vital ACDC Systems:

2) Manual ground reading
a) Manual battery ground detector intentionally puts a ground on the system and looks for current flow. No current flow means that the ground detector is the only ground.
b) Obtain Manual Ground readings:
(1) Place selector in hand - removes auto ground function, enables hand
(2) Select bus - places meter between a ground and selected bus
(3) Take reading, any current flow shows a ground on one of the buses NOT SELECTED

Part 2 is Incorrect. IAW OP-TM-AOP-023, A DC System Failure, when battery voltage is < 105Vdc the A DC System is deenergized. Plausible if candidate confuses the actions for the different battery voltages.

C. **Incorrect.**
Part 1 is Incorrect. Plausible if candidate does not understand the operation of the battery ground detector and that deflection of battery meter for the selected bus indicates a ground on one of the other busses.

Part 2 is Correct. IAW OP-TM-AOP-023, A DC System Failure, when battery voltage is < 110Vdc the Reactor is tripped.

D. **Incorrect.**
Part 1 is Incorrect. Plausible if candidate does not understand the operation of the battery ground detector and that deflection of battery meter for the selected bus indicates a ground on one of the other busses.

Part 2 is Incorrect. IAW OP-TM-AOP-023, A DC System Failure, when battery voltage is < 105Vdc the A DC System is deenergized. Plausible if candidate confuses the actions for the different battery voltages.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063	A2.01
	Importance Rating	2.5	

K/A: Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds.

Proposed Question: RO Question # 48

Technical Reference(s): OP-TM-AOP-023, pg 5, Rev 004
TQ-TM-104-734-C001, pg 9, Rev 007

Proposed References to be provided to applicants during examination: None

Learning Objective: 734-GLO-2

Question Source:		
	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:		
	55.41	8
	55.43	

(8) Components, capacity, and functions of emergency systems.

Comments:
The KA is matched because the question requires the candidates to predict the location of a ground on the DC electrical system and the required action to take based on the battery discharging excessively through the ground.

The question is at the Comprehension/Analysis cognitive level because the candidate must understand the operation of the DC ground system, be able to determine location of DC grounds based on ground detection indications, understand the DC voltage limits requiring crew actions and what actions to take when those limits are met.

What MUST be known:
1. How is the ground detector operated and interpreted? 2. Where is the location of the ground in the DC electrical system? 3. What is the appropriate action to take IAW Abnormal Operating Procedures for lowering battery voltage?

49 **ID: 978632** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- RB Cooling Fan AH-E-1A is SHUTDOWN.
    - The AH-E-1A Control Switch is in the Normal-After-Stop position.

Event:

- A manual reactor trip is performed due to a Small Break LOCA inside Containment.
- ESAS High RB Pressure Bistable operation is as follows:

Time	Action
T = +0 seconds	Channel 1 Bistable trips
T = +5 seconds	Channel 2 Bistable trips
T = +10 seconds	Channel 3 Bistable trips

Given the above information, AH-E-1A will be running in ___(1)___ speed, with ___(2)___ flow through the Emergency coolers at T = 11 seconds.

A. (1) fast
(2) Reactor River Water

B. (1) fast
(2) Nuclear Services Closed Cooling Water

C. (1) slow
(2) Reactor River Water

D. (1) slow
(2) Nuclear Services Closed Cooling Water

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. Plausible if candidate believes RB Cooling fans remain in Fast speed following an ES signal. IAW 1105-8, Safeguards Actuation System, Attachment 1, Section E:

- AH-E-1A, AH-E-1B, AH-E-1C start in Slow Speed.

IAW TQ-TM-104-642-C001, Engineered Safeguards Actuation System:

a. AH-E-1A/B/C fans:
3) Interlock: Shift from fast to slow speed on ESAS. Fan A from Channel "A", Fan B from Channel "B", Fan C from Channel "A and "B"

Part 2 is correct. RB Emergency cooling initiates (Reactor River) on the ES signal. IAW TQ-TM--104-824-C001, Reactor Building Ventilation:

f. System operation during an ESAS
1) Normal RB cooling and purge valves isolate (CIVs)
2) RB Emergency cooling initiates.
3) AH-E-1s go to slow speed.

B. **Incorrect.**
Part 1 is incorrect. Plausible if candidate believes RB Cooling fans remain in Fast speed following an ES signal.

IAW 1105-8, Safeguards Actuation System, Attachment 1, Section E:

- AH-E-1A, AH-E-1B, AH-E-1C start in Slow Speed.

IAW TQ-TM-104-642-C001, Engineered Safeguards Actuation System:

a. AH-E-1A/B/C fans:
3) Interlock: Shift from fast to slow speed on ESAS. Fan A from Channel "A", Fan B from Channel "B", Fan C from Channel "A and "B"

Part 2 is incorrect. Plausible if candidates believes NSCCW supplies flow through the coolers following an ES signal. NSCCW normally supplies the coolers but is dead headed in order to supply a pressure boundary in the cooler. When an ES signal is present, Reactor River Water will flow through the cooler to provide the cooling medium. IAW TQ-TM--104-824-C001, Reactor Building Ventilation:

c. NSCCW provides heat removal to the AH-E-1A/B/C motor coolers. The river is the ultimate heat sink via the NS River Water System.

C. **Correct.**
Part 1 is correct. After 2 channels actuate, RB Cooling fans start in slow speed on Block 2 (5 seconds after second ES channel trips).

IAW 1105-8, Safeguards Actuation System, Attachment 1, Section E:

- AH-E-1A, AH-E-1B, AH-E-1C start in Slow Speed.

IAW TQ-TM-104-642-C001, Engineered Safeguards Actuation System:

i. 4 psig bistable (six total)
  4) Used to actuate Engineered Safeguards
    a) High Pressure Injection
    b) Low Pressure Injection
    c) Reactor Building Isolation
    d) Reactor Building Cooling

a. AH-E-1A/B/C fans:
  3) Interlock: Shift from fast to slow speed on ESAS. Fan A from Channel "A", Fan B from Channel "B", Fan C from Channel "A and "B"

Part 2 is correct. RB Emergency cooling initiates (Reactor River) on the ES signal. IAW TQ-TM--104-824-C001, Reactor Building Ventilation:

f. System operation during an ESAS
  1) Normal RB cooling and purge valves isolate (CIVs)
  2) RB Emergency cooling initiates.
  3) AH-E-1s go to slow speed.

D. **Incorrect.**

Part 1 is correct. After 2 channels actuate, RB Cooling fans start in slow speed on Block 2 (5 seconds after second ES channel trips).

IAW 1105-8, Safeguards Actuation System, Attachment 1, Section E:

- AH-E-1A, AH-E-1B, AH-E-1C start in Slow Speed.

IAW TQ-TM-104-642-C001, Engineered Safeguards Actuation System:

i. 4 psig bistable (six total)
  4) Used to actuate Engineered Safeguards
    a) High Pressure Injection
    b) Low Pressure Injection
    c) Reactor Building Isolation
    d) Reactor Building Cooling

a. AH-E-1A/B/C fans:
  3) Interlock: Shift from fast to slow speed on ESAS. Fan A from Channel "A", Fan B from Channel "B", Fan C from Channel "A and "B"

Part 2 is incorrect. Plausible if candidates believes NSCCW supplies flow through the coolers following an ES signal. NSCCW normally supplies the coolers but is dead headed in order to supply a pressure boundary in the cooler. When an ES signal is present, Reactor River Water will flow through the cooler to provide the cooling medium. IAW TQ-TM--104-824-C001, Reactor Building Ventilation:

c. NSCCW provides heat removal to the AH-E-1A/B/C motor coolers. The river is the ultimate heat sink via the NS River Water System.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	022	A3.01	
	Importance Rating	4.1		

K/A: Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation.

Proposed Question: RO Question # 49

Technical Reference(s):
TQ-TM-104-824-C001, pg60 Rev 006
TQ-TM-104-642-C001, pg 18, Rev 006
1105-3, pg 19, Rev 51

Proposed References to be provided to applicants during examination: None

Learning Objective: 642-GLO-10

Question Source:	Bank #	IR-642-GLO-10-Q05
	Modified Bank #	
	New	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	9
	55.43	

(9) Shielding, isolation, and containment design features, including access limitations.

Comments:
The KA is matched because the question requires the candidates to determine what automatic operation will occur to a component of the Containment Cooling System upon a Safeguards initiation signal.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must know the operation of the RB normal ventilation during an ESAS actuation and the cooling medium used.

What MUST be known:
1. What is the required speed of AH-E-1A upon receiving an ESAS signal?
2. What is the cooling medium for the Emergency Coolers upon receiving an ESAS signal?

50 **ID: 978630** **Points: 1.00**

Plant Conditions (T = 0 seconds):

- The Reactor is at 70% power.
- Pressures are as follows:

Pressure Indication	Value (psig)
OTSG 1A Pressure	906
Turbine Header Pressure A	904
OTSG 1B Pressure	908
Turbine Header Pressure B	906

Sequence of Events:

- T = 1 second:
  - A steam leak occurs just upstream of MS-V-3A/B/C, Turbine Bypass Valves from "B" OTSG.
  - OP-TM-AOP-051, Secondary Side Steam Leak, is entered.

- T = 30 seconds:
  - The plant remains online.
  - Pressures are as follows:

Pressure Indication	Value (psig)
OTSG 1A Pressure	896
Turbine Header Pressure A	894
OTSG 1B Pressure	825
Turbine Header Pressure B	895

Given the above information, which ONE of the following describes the physical position of the Main Steam Isolation Valves (MS-V-1A/B/C/D) at T = 30 seconds?

A. MS-V-1A is open.
MS-V-1B is open.
MS-V-1C is open.
MS-V-1D is open.

B. MS-V-1A is open.
MS-V-1B is open.
MS-V-1C is seated closed.
MS-V-1D is seated closed.

C. MS-V-1A is open.
MS-V-1B is seated closed.
MS-V-1C is open.
MS-V-1D is seated closed.

D. MS-V-1A is seated closed.
MS-V-1B is seated closed.
MS-V-1C is open.
MS-V-1D is open.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
The candidate must recognize that the MSIV's are angled stop check valves designed to stay open as long as normal steam flow is from the OTSG to the HP turbine. When flow is going from the HP Turbine toward the OTSG, the stop check valves will close, causing an isolation of that portion of the Main Steam System. With a steam leak just upstream of the TBV's, the flow will reverse from the HP turbine (through the steam chest since the Turbine Stop valves do not close), directly to the condenser, thereby closing the affected side MSIV's. The affected OTSG MSIV's will stay closed until pressure in the OTSG is high enough to open the check valves to allow normal flow. Plausible if the candidate does not understand the design feature of the MSIV's.

B. **Correct.**
The MSIV's are angled stop check valves designed to stay open as long as normal steam flow is from the OTSG to the HP turbine. When flow is going from the HP Turbine toward the OTSG, the stop check valves will close, causing an isolation of that portion of the Main Steam System. With a steam leak just upstream of the TBV's, the flow will reverse from the HP turbine (through the steam chest since the Turbine Stop valves do not close), directly to the condenser, thereby closing the affected side MSIV's. The affected OTSG MSIV's will stay closed until pressure in the OTSG is high enough to open the check valves to allow normal flow.

C. **Incorrect.**
Plausible if the candidate correctly recognizes the "B" OTSG as the affected OTSG but does not know which of the four steam lines go to each OTSG and incorrectly chooses the "B" and "D" steam lines for the "B" OTSG.

D. **Incorrect.**
The candidate must recognize that the Turbine Bypass Valve nomenclature is backwards from normal convention. MS-V-3A/B/C are associated with the "B" OTSG and MS-V-3D/E/F are associated with the "A" OTSG. Plausible if the candidate incorrectly recognizes the "A" OTSG as the affected OTSG but does know which of the four steam lines go to each OTSG.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 A3.02	
	Importance Rating	3.1	

K/A: Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS.

Proposed Question: RO Question # 50

Technical Reference(s): TQ-TM-104-411-C001, pg 13,14, Rev 007

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-6

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	4
	55.43	

(4) Secondary coolant and auxiliary systems that affect the facility.

Comments:
The KA is matched because the question requires the candidates to monitor parameters associated with the Main Steam System and, using those parameters, identify which portion of the Main Steam System has been automatically isolated.

The question is at the Comprehension/Analysis cognitive level because the candidate must know the construction and location of the main steam isolation valves and how they operate during various OTSG and main steam header pressures.

What MUST be known:
1. What do the changes in secondary system parameters indicate has occurred to the Main Steam System components? 2. What is the physical layout of the Main Steam System? 3. Which Main Steam valves are associated with each OTSG?

51 **ID: 978629** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- Time = 0815:
  - A Reactor trip occurs due to a Large Break LOCA.
  - The following ESAS Actuations have initiated:
    - 1600 psig RCS pressure.
    - 4 psig RB pressure.
- Time = 0820:
  - Current RB pressure is 20 psig.
  - 1600 psig ESAS has been BYPASSED.
  - 4 psig ESAS actuation has been placed in DEFEAT.
- Time = 0825:
  - No ES component positions have been changed.
  - The leak in the RCS gets larger resulting in RB pressure rising to 35 psig.

Given the above information, the Reactor Building Spray Pumps, BS-P-1A/1B _________ at Time = 0825.

A. will automatically start.

B. must be secured since their suction and discharge valves are closed.

C. must be started by pushing the manual ES 30 psi actuation pushbuttons.

D. must be manually started using their associated BS pump extension control switches.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
IAW TQ-TM-104-214-C001, Reactor Building Spray:

C. Explanation
1. Building Spray Normal Operating Conditions
a. Mode - Normal;(Auto)-Auto Start at 30 psig increasing Reactor Building pressure (2 out of 3 pressure switches) with a block 4 permissive signal.

Plausible since RB pressure is at 30 psi, however ESAS Block 4 has been defeated therefore the BS pumps do not auto start.

B. **Incorrect.**
IAW 1105-3, Safeguards Actuation System, Attachment 1, Section B - 4 PSIG RB Pressure Actuation, BS-V-1A/B, RB Spray Header Isolation Valves (Building Spray Pump discharge valves) and BS-V-3A/B, BS-P-1A/B Suction Valves, all open upon a 4# RB ESAS signal. Since the 4# signal was in for greater than 20 seconds, the valves did open. Placing 1600# in Bypass and 4# in Defeat do not reposition these valves. Plausible misconception that the BS spray valves are closed based on ESAS signals being bypassed/defeated.

C. **Incorrect.**
Plausible if the candidate has a misconception on the affect of pushing the 30 psig ESAS manual pushbuttons. These pushbutton are for R.B. 30# isolation, not B.S. Pumps starting up. IAW OP-TM-642-901, 30 PSIG ESAS Actuation:

D. **Correct.**

IAW TQ-TM-104-214-C001, Reactor Building Spray:

C. Explanation
1. Building Spray Normal Operating Conditions
a. Mode - Normal;(Auto)-Auto Start at 30 psig increasing Reactor Building pressure (2 out of 3 pressure switches) with a block 4 permissive signal.

Since 1600# ESAS has been BYPASSED and 4# has been placed in DEFEAT, there is no ESAS signal anymore to give a Block 4 permissive. Therefore, when RB Pressure reaches 35 psig at time = 0825, a 30# signal has been received but there is no Block 4 permissive to start the Building Spray Pumps. OP-TM-642-903, 30 PSIG ESAS Actuation, is entered if RB pressure is greater than 30 psig, and directs Initiation of OP-TM-214-901, RB Spray Operation. OP-TM-214-901 provides direction for manually starting Building Spray Pumps in Section 4.2, Contingency Actions:

4.2.1.5: If BS-P-1A is not operating, then perform the following:
A. Verify DH-V-5A or DH-V-6A is open
B. Start BS-P-1A

4.2.2.5: If BS-P-1B is not operating, then perform the following:
A. Verify DH-V-5B or DH-V-6B is open
B. Start BS-P-1B

Examination Outline Cross-reference:

Level	RO		SRO
Tier #	2		
Group #	1		
K/A #	026	A4.05	
Importance Rating	3.5		

K/A: Ability to manually operate and/or monitor in the control room: Containment spray reset switches.

Proposed Question: RO Question # 51

Technical Reference(s):	OP-TM-214-901, pg 4,5, Rev 4 TQ-TM-104-214-C001, pg 29, Rev 009

Proposed References to be provided to applicants during examination: None

Learning Objective: 214-GLO-10

Question Source:	Bank #	QR-214-GLO-10-Q03
	Modified Bank #	
	New	

Question History: Last NRC Exam: 2000

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

10 CFR Part 55 Content:	55.41	9
	55.43	

(9) Shielding, isolation, and containment design features, including access limitations.

Comments:
The KA is matched because the question requires the candidate to understand what occurs when Containment Spray reset switches are defeated/bypassed by monitoring plant indications and manually operating Containment safety related equipment.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze the ESAS system and Reactor Building atmosphere conditions given throughout the timeline and then apply the appropriate action to verify that Containment is not violated.

What MUST be known:
1. What is the status of ESAS components given a timeline of conditions? 2. How does defeating/bypassing ESAS signals affect ESAS related equipment? 3. What is the start logic for Building Spray Pumps on an ES?

52 **ID: 978626** **Points: 1.00**

Plant Conditions (Time = 0 seconds):

- The plant is operating at 100% power.

Sequence of Events:

- Time = 30 seconds:
  - RM-L-1, Primary Coolant Letdown Monitor, instantly pegs full scale high.
- Time = 60 seconds:
  - The CRS requests that RM-L-1 be removed from service IAW 1105-8, Radiation Monitoring System.

Given the above information, which one of the following identifies:
(1) The alarm that will actuate at Time = 30 seconds, and
(2) The action required IAW 1105-8 that will allow for restoration of Letdown from the Control Room?

A. (1) MAP C-1-1, RADIATION LEVEL HI, will actuate.
(2) The RM-L-1 FAIL RESET Pushbutton must be depressed to clear the interlock.

B. (1) MAP C-1-1, RADIATION LEVEL HI, will actuate.
(2) The RM-L-1 Interlock Mode Switch must be placed in DEFEAT to clear the interlock.

C. (1) MAP C-3-1, RAD MON SYSTEM TROUBLE, will actuate.
(2) The RM-L-1 FAIL RESET Pushbutton must be depressed to clear the interlock.

D. (1) MAP C-3-1, RAD MON SYSTEM TROUBLE, will actuate.
(2) The RM-L-1 Interlock Mode Switch must be placed in DEFEAT to clear the interlock.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is correct. IAW 1101-2.1, Radiation Monitoring System Setpoints, RM-L-1 (Hi) Alert setpoint is 2E3 CPM and the High Alarm setpoint is 4E3 CPM. RM-L-1(Low) Alert setpoint is 1E5 CPM and the High Alarm setpoint is 4E5 CPM. When RM-L-1 pegs full scale high, it will indicate higher than the High Alarm setpoint and therefore, C-1-1, radiation Level Hi, will alarm (since C-1-1 setpoints are found in 1101-2.1).

Part 2 is incorrect. The RM-L-1Fail Reset pushbutton on the front of the ratemeter will not defeat the interlock to allow MU-V-2A and MU-V-2B to be opened. Plausible if the candidate believes that since their is a fault associated with RM-L-1, that the Fail/Reset pushbutton, which is located directly on the RM-L-1 ratemeter, must be pushed.

B. **Correct.**
Part 1 is correct. IAW 1101-2.1, Radiation Monitoring System Setpoints, RM-L-1 (Hi) Alert setpoint is 2E3 CPM and the High Alarm setpoint is 4E3 CPM. RM-L-1(Low) Alert setpoint is 1E5 CPM and the High Alarm setpoint is 4E5 CPM. When RM-L-1 pegs full scale high, it will indicate higher than the High Alarm setpoint and therefore, C-1-1, radiation Level Hi, will alarm (since C-1-1 setpoints are found in 1101-2.1).

Part 2 is correct. IAW OP-TM-MAP-C0101, Radiation Level Hi:

- 3.0 AUTOMATIC ACTIONS
  - RM-L-1: MU-V-2A and 2B close on Hi Alarm
  - RM-L-1 LO: None

Since the CRS directs that RM-L-1 be removed from service IAW 1105-8, Radiation Monitoring System, Section 3.6 is applicable:

- 3.6 Removing a Liquid Monitor from Service
- Note: If an instrument fails, perform this section. For a failed instrument, compensatory actions are not a prerequisite but must be completed.
  - 3.6.4 Place interlock mode switch in DEFEAT. Refer to Table 3.6 for switch location and interlock description.
  - 3.6.5 For each channel of monitor removed from service, Place monitor in OFF. NA any channels which will remain in service. Check off on Table 3.6

Placing the RM-L-1 Interlock Mode Switch to Defeat will remove the Interlock associated with MU-V-2A and MU-V-2B and will allow them to be opened from the Control Room to restore Letdown IAW OP-TM-211-950, Restoration of Letdown Flow.

C. **Incorrect.**
Part 1 is incorrect. Map C-3-2, Radiation Monitor System Trouble, would alarm if RM-L-1 had experienced a Loss of Voltage and pegged LOW. Plausible if the candidate is not familiar with the causes of MAP-C-3-1 or selects based on Alarm title alone.

Part 2 is incorrect. The RM-L-1Fail Reset pushbutton on the front of the ratemeter will not defeat the interlock to allow MU-V-2A and MU-V-2B to be opened. Plausible if the candidate believes that since their is a fault associated with RM-L-1, that the Fail/Reset pushbutton, which is located directly on the RM-L-1 ratemeter, must be pushed.

D. **Incorrect.**

Part 1 is incorrect. Map C-3-2, Radiation Monitor System Trouble, would alarm if RM-L-1 had experienced a Loss of Voltage and pegged LOW. Plausible if the candidate is not familiar with the causes of MAP-C-3-1 or selects based on Alarm title alone.

Part 2 is correct. IAW OP-TM-MAP-C0101, Radiation Level Hi:

- 3.0 AUTOMATIC ACTIONS
  - RM-L-1: MU-V-2A and 2B close on Hi Alarm
  - RM-L-1 LO: None

Since the CRS directs that RM-L-1 be removed from service IAW 1105-8, Radiation Monitoring System, Section 3.6 is applicable:

- 3.6 Removing a Liquid Monitor from Service
- Note: If an instrument fails, perform this section. For a failed instrument, compensatory actions are not a prerequisite but must be completed.
  - 3.6.4 Place interlock mode switch in DEFEAT. Refer to Table 3.6 for switch location and interlock description.
  - 3.6.5 For each channel of monitor removed from service, Place monitor in OFF. NA any channels which will remain in service. Check off on Table 3.6

Placing the RM-L-1 Interlock Mode Switch to Defeat will remove the Interlock associated with MU-V-2A and MU-V-2B and will allow them to be opened from the Control Room to restore Letdown IAW OP-TM-211-950, Restoration of Letdown Flow.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	073	A4.02	
	Importance Rating	3.7		

K/A: Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel.

Proposed Question: RO Question # 52

Technical Reference(s):
OP-TM-MAP-C0101, pg 42, Rev 2A
1101-2.1, pg 8, Rev 83
1105-8, pg 44, Rev 87

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-10

Question Source:	Bank #	IR-661-GLO-10-Q02	
	Modified Bank #		
	New		

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	11
	55.43	

(11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:
The KA is matched because the question requires the candidates to identify the expected alarm that they would monitor in the Control Room, and also what component on the Radiation Monitoring System Control Panel would be operated to compensate for a detector failure.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to know associated alarms and interlocks with radiation monitoring and controls to bypass and / or defeat to allow components to be repositioned.

What MUST be known:
1. What are the interlocks associated with Radiation Monitor RM-L-1? 2. What alarms are associated with a failure of RM-L-1? 3. What are the actions associated with a failed RM-L-1 detector to restore Letdown?

53 **ID: 978624** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power, 45 EFPD.

Sequence of Events:

- A severe fire develops in the Unit 1 Screen House which causes a loss of the 1R 480V and 1T 480V Busses.
- OP-TM-AOP-005, Loss of River Water Systems, is entered.
  - The reactor is tripped.
  - RCPs are secured.
- Emergency Boration is required to commence, as directed by OP-TM-AOP-005.

Given the above conditions and IAW OP-TM-AOP-005 to provide the required Boron concentration, Emergency Boration will occur IAW ___(1)___ and is required to establish ___(2)___ conditions within 6 hours.

A. (1) Rule 5
(2) Hot Shutdown

B. (1) Rule 5
(2) Cold Shutdown

C. (1) Guide 1
(2) Hot Shutdown

D. (1) Guide 1
(2) Cold Shutdown

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect but plausible if the candidate is not familiar with OP-TM-AOP-005 and chooses Rule 5 either because Rule 5 takes precedence over Guide 1.

Part 2 is correct. IAW OP-TM-AOP-005, Caution Note associated with step 4.3.13:

- CAUTION: Emergency Boration at 10 GPM from BAMT or RBAT will achieve HSD boron requirement within 6 hours. Emergency boration from BWST will not achieve HSD boron within 24 hours unless cycle EFPD is > 400.

B. **Incorrect.**
Part 1 is incorrect but plausible if the candidate is not familiar with OP-TM-AOP-005 and chooses Rule 5 either because Rule 5 takes precedence over Guide 1.

Part 2 is incorrect but plausible if the candidate believes that Cold Shutdown is the required condition per OP-TM-AOP-005. However, OP-TM-AOP-005, Caution Note associated with step 4.3.13 states:

- CAUTION: Emergency Boration at 10 GPM from BAMT or RBAT will achieve HSD boron requirement within 6 hours. Emergency boration from BWST will not achieve HSD boron within 24 hours unless cycle EFPD is > 400.

C. **Correct.**
This question requires the student to understand that RCS letdown may not be available for at least a couple of days. As a result, the only opportunity to change boron concentration is prior to the PZR going solid due to the sustained loss of letdown flow. HSD boron requirements are greater than normal operating boron concentrations. One goal of AOP-005 in this scenario is to establish a high boron concentration feed source early in the event to provide the greatest introduction of boron into the RCS.

Part 1: IAW OP-TM-AOP-005, Step 4.3.13:

- INITIATE Emergency boration from BAMT or RBAT IAW Guide 1.

Part 2: IAW OP-TM-AOP-005, Caution Note associated with step 4.3.13:

- CAUTION: Emergency Boration at 10 GPM from BAMT or RBAT will achieve HSD boron requirement within 6 hours. Emergency boration from BWST will not achieve HSD boron within 24 hours unless cycle EFPD is > 400.

D. **Incorrect.**
Part 1 is correct. IAW OP-TM-AOP-005, Step 4.3.13:

- INITIATE Emergency boration from BAMT or RBAT IAW Guide 1.

Part 2 is incorrect but plausible if the candidate believes that Cold Shutdown is the required condition per OP-TM-AOP-005. However, OP-TM-AOP-005, Caution Note associated with step 4.3.13 states:

- CAUTION: Emergency Boration at 10 GPM from BAMT or RBAT will achieve HSD boron requirement within 6 hours. Emergency boration from BWST will not achieve HSD boron within 24 hours unless cycle EFPD is > 400.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004	2.4.11
	Importance Rating	4.0	

K/A: Chemical and Volume Control System: Knowledge of abnormal condition procedures.

Proposed Question: RO Question # 53

Technical Reference(s): OP-TM-AOP-005, pg 15, Rev 009

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP005-PCO-1

Question Source: Bank #

Modified Bank # IR-AOP005-PCO-1-Q01

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

(6) Design, components, and functions of reactivity control mechanisms and instrumentation.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of an Abnormal Operating Procedure associated with the Chemical and Volume Control (Makeup) System. Guide 1 utilizes Makeup System valves and piping to accomplish the task.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to recall the method for emergency boration IAW procedural guidance, and the reason for the method.

What MUST be known:
1. What method of Emergency Boration is required IAW OP-TM-AOP-005? 2. What is the required plant condition that Emergency Boration is going to achieve?

Original Question: IR-AOP005-PCO-1-Q01

Initial Conditions:

- Plant is at 100% power and 45 EFPD.
- A severe fire develops in the U1 IPSH which causes a trip of "R" and "T" 480 v busses.
- AOP-005 is entered, the reactor is tripped and all RCPs are secured.
- Guide 1, "Emergency Boration Backup Methods" is entered as directed.

Why is emergency boration directed, and why is Guide 1 used instead of Rule 5?

Emergency boration is required because __(1)__ and guide 1 is used because __(2)__.

A. (1) Emergency boration is required to establish Hot shutdown conditions.
(2) BWST inventory must be maintained above T.S. minimum level.

B. (1) Emergency boration is required to establish Cold shutdown conditions.
(2) BWST inventory must be maintained above T.S. minimum level.

C. (1) Emergency boration is required to establish Hot shutdown conditions.
(2) The BWST lacks the required Boron concentration to reach Hot shutdown in the required time frame.

D. (1) Emergency boration is required to establish Cold shutdown conditions.
(2) The BWST lacks the required Boron concentration to reach Cold shutdown in the required time frame.

Answer: C

54 **ID: 978622** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Event:

- Vital Bus B (VBB) tripped 71 hours ago and has been re-energized from TRA.

Given the above information, ___(1)___ must be in a ___(2)___ state to continue plant operation.

A. (1) RPS Channel B
(2) tripped

B. (1) RPS Channel B
(2) bypassed

C. (1) ESAS Channel 2
(2) tripped

D. (1) ESAS Channel 2
(2) defeated

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. IAW Tech Spec Table 3.5-1, The Minimum number of Operable RPS Channels is 2 with a Minimum degree of Redundancy of 1.

Part 2 is incorrect. Since there are 4 channels, it is not required to do anything with RPS Channel B. Plausible since leaving the RPS Channel tripped will maintain RPS degree of redundancy; however it is maintained even if the Channel is reset due to the other three channels being operable and two needed to cause a trip.

B. **Incorrect.**
Part 1 is incorrect. IAW Tech Spec Table 3.5-1, The Minimum number of Operable RPS Channels is 2 with a Minimum degree of Redundancy of 1.

Part 2 is incorrect. Since there are 4 channels, it is not required to do anything with RPS Channel B. Plausible since bypassing the RPS Channel will maintain RPS degree of redundancy; however it is maintained even if the Channel is reset due to the other three channels being operable and two needed to cause a trip.

C. **Correct.**

Part 1 is correct. There are three channels of ESAS. IAW Tech Spec Table 3.5-1, Instruments Operating Conditions, the Minimum number of Operable Channels is 2 with a Minimum degree of Redundancy of 1.

Part 2 is correct. The B ESAS Channel is not operable when the Vital Bus is powered from a regulating Transformer. Therefore it should be left in the tripped state to maintain the necessary degree of redundancy to meet Tech Specs and continue operation. IAW 1107-2B, 120 Volt Vital Electrical System, Section 2.0 - Limits and Precautions:

- 2. When a Vital Bus is powered from a regulated transformer bus (TRA, TRB), the Vital Bus is inoperable".
  - a. The allowed outage time (AOT) for each vital bus is based on the limiting Tech Spec component on each Vital Bus. This may change with plant conditions and based on the availability of redundant components.
  - b. With the reactor critical the ESAS timeclock is limiting.
  - c. An ESAS channel powered from an inoperable vital bus is inoperable unless the channel is maintained in a tripped state. (Ref. CR 146271) The typical AOT for VBD is 7 days, while the typical AOT for all other vital buses is 72 hours.
  - d. When ESAS channel operability is required by T.S. 3.5.1 do **not** reset ESAS channels on a Vital Bus being powered from a regulated transformer supply.

D. **Incorrect.**

Part 1 is correct. There are three channels of ESAS. IAW Tech Spec Table 3.5-1, Instruments Operating Conditions, the Minimum number of Operable Channels is 2 with a Minimum degree of Redundancy of 1.

Part 2 is incorrect. Placing the ESAS Channel in defeat will leave the ESAS system with a degree of redundancy of zero, which would place the plant on a one hour timeclock IAW Action Statement (a) of Tech Spec Table 3.5-1. (Spec 3.0.1).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013	2.2.40
	Importance Rating	3.4	

K/A: Engineered Safety Features Actuation System (ESFAS): Ability to apply Technical Specifications for a system.

Proposed Question: RO Question # 54

Technical Reference(s): Tech Spec Table 3.5-1, pg 3-31, 3-32, Rev 247
1107-2B, pg 6, Rev 33

Proposed References to be provided to applicants during examination: None

Learning Objective:	735-GLO-14		
Question Source:	Bank #	IR-735-GLO-14-Q01	
	Modified Bank #		
	New		
Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	7	
	55.43		

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to apply a Tech Spec required action to the Engineered Safety Features Actuation System.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze the plant conditions to determine the limiting condition that applies to Tech Specs and know the actions required in order to maintain the plant operating with the current conditions.

What MUST be known:
1. What is the maximum time allowed for VBB to be inoperable IAW Tech Specs? 2. What is the status of VBB when powered from TRA? 3. What are the Minimum number of Operable Channels and Minimum degree of Redundancy requirements for the RPS System? 4. What are the Minimum number of Operable Channels and Minimum degree of Redundancy requirements for the ESAS System? 5. What are the effects of placing ESAS in a Defeated and/or Tripped state?

55 **ID: 978612** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- Due to unrelated conditions, one EFW flow meter is inoperable per OTSG.
- Due to corrosion, both Containment Sump level transmitters are inoperable.
- Due to a common mode failure, all Level Transmitters on Condensate Storage Tanks, CO-T-1A/B, are inoperable.
- Due to a faulted power cord, Containment Flood Level Transmitter, LT-806, is inoperable.
- All other components are operable.

Given the above information and IAW Technical Specification 3.5.5, which one of the following will require a Special Report submitted to the NRC within 30 days if not operable within 7 days?

A. One EFW flow meter is inoperable per OTSG.

B. All level transmitters on CO-T-1A/B are inoperable.

C. One Containment Flood level transmitter is inoperable.

D. Both Containment Sump level transmitters are inoperable.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Plausible if the candidate believes that more than one EFW flow meter per OTSG must be operable. According to Tech Spec table 3.5-2, a minimum of one is required to be operable.

TABLE 3.5-2

ACCIDENT MONITORING INSTRUMENTS

INSTRUMENTS	NUMBER OF CHANNELS	MINIMUM NUMBER OF CHANNELS
Saturation Margin Monitor	2	1
Safety Valve Differential Pressure Monitor	1 per discharge line	1 per discharge line
PORV Position Monitor	2	1*
Emergency Feedwater Flow	2 per OTSG	1 per OTSG
Pressurizer Level	2	1
Backup Incore Thermocouple Display Channel	4 thermocouples/core quadrant	2 thermocouples/core quadrant

B. **Correct.**
According to Tech Spec Table 3.5-3, the minimum number of operable level transmitters for CST's is 1/CST. With zero operable on either tank, The action statement applies.
Tech Spec 3.5 Action Statement A:

> A. With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirements:
>
> 1. either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or
> 2. prepare and submit a Special Report within 30 days following the event outlining action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 3.5-3

POST ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENTS	REQUIRED NUMBER OF CHANNELS	MINIMUM NUMBER OF CHANNELS	ACTION
High Range Noble Gas Effluent			
a. Condenser Vacuum Pump Exhaust (RM-A5-Hi)	1	1	A
b. Condenser Vacuum Pump Exhaust (RM-G25)	1	1	A
c. Auxiliary and Fuel Handling Building Exhaust (RM-A8-Hi)	1	1	A
d. Reactor Building Purge Exhaust (RM-A9-Hi)	1	1	A
e. Reactor Building Purge Exhaust (RM-G24)	1	1	A
f. Main Steam Lines Radiation (RM-G26/RM-G27)	1 each OTSG	1 each OTSG	A
Containment High Range Radiation (RM-G22/G-23)	2	2	A
Containment Pressure	2	1	B
Containment Water Level			
a. Containment Flood (LT-806/807)	2	1	B
b. Containment Sump (LT-804/805)	1	0	C
DELETED			
Wide Range Neutron Flux	2	1	A
Reactor Coolant System Cold Leg Water Temperature (TE-959, 961; TI-959A, 961A)	2	1	A
Reactor Coolant System Hot Leg Water Temperature (TE-958, 960; TI-958A, 960A)	2	1	A
Reactor Coolant System Pressure (PT-949, 963; PI-949A, 963)	2	1	A
Steam Generator Pressure (PT-950, 951, 1180, 1184; PI-950A, 951A, 1180, 1184)	2/OTSG	1/OTSG	A
Condensate Storage Tank Water Level (LT-1060, 1061, 1062, 1063; LI-1060, 1061, 1062, 1063)	2/Tank	1/Tank	A

C. **Incorrect.**
Plausible if the candidate believes that more than one Containment Flood Transmitter must be operable. According to Tech Spec table 3.5-3, a minimum of one is required to be operable. See above chart for requirement.

D. **Incorrect.**
Plausible if the candidate believes that at least one Containment Sump Level Transmitter must be operable. According to Tech Spec table 3.5-3, a minimum of zero is required to be operable. See above chart for requirement.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	061	2.4.30	
	Importance Rating	2.7		

K/A: Auxiliary / Emergency Feedwater System: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Proposed Question: RO Question # 55

Technical Reference(s): Tech Spec Table 3.5-3, pg 3-40d, Rev 240

Proposed References to be provided to applicants during examination: None

Learning Objective: 421-GLO-14

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of an event related to the Emergency Feedwater System that requires a Special Report to be submitted to the NRC (an external agency). Condensate Storage Tanks are a backup source of water for the Emergency Feedwater Pumps.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to know the post accident monitoring equipment and Special Reporting criteria if instruments are out of service.

What MUST be known:
1. What are the reportability requirements for not meeting the required/minimum number of channels of Condensate Storage Tank Level Indicators?

56 **ID: 978611** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- Spent Fuel Shuffle in progress in the FHB.

Sequence of Events:

- A spent fuel assembly was dropped.
- Currently the following conditions/Indications exist:
  - RM-A-4, Fuel Handling Building Exhaust Air Radiation Monitor, is in **alert** on all channels.
  - RM-G-9, Spent Fuel Bridge Radiation Monitor, is in **high alarm**.
  - The status of the following components are reported by the Non Licensed Operators:

Component	Name	Status
AH-D-4	Air Intake Tunnel Fire Damper	Open
AH-D-120	FH Bldg 348' Supply Damper	Closed
AH-D-121	FH Bldg 348' Supply Damper	Closed
AH-D-122	FH Bldg 348' Exhaust Damper	Open
AH-E-10	FH Bldg Supply Fan	Running

Given the above information, identify the one selection below that describes **ALL** required manual actions IAW OP-TM-MAP-C0101, Radiation Level HI.

A. Secure AH-E-10, ONLY.

B. Close AH-D-4, and
Close AH-D-122, ONLY.

C. Close AH-D-122, and
Secure AH-E-10, ONLY.

D. Close AH-D-4,
Close AH-D-122, and
Secure AH-E-10.

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Although securing AH-E-10 is an interlock for RM-G-9 and RM-A-4, it is not the only interlock for them. Plausible and discriminatory because AH-E-10 tripping is an automatic interlock for RM-G-9 and it is indicated as running in the stem.

B. **Incorrect.**
Although closing AH-D-122 is an interlock for RM-G-9 and RM-A-4, AH-D-4 is not interlocked with either. Plausible and discriminatory because AH-D-122 shutting is an automatic interlock for RM-G-9 and it is indicated as open in the stem.

C. **Correct.**
AH-D-120, 121, 122 are closed and AH-E-10 is tripped by RM-G-9 going into alarm. MAP C-1-1 requires manual closure/tripping of any interlocked component that does not actuate.

- RM-A-4 - Fuel Handling Bldg. Exhaust
    - 3.0 AUTOMATIC ACTIONS
        - AH-E-10 (FHB Supply Fan) trips on high alarm
        - AH-D-120, AH-D-121, AH-D-122 (FHB Dampers) close on high alarm

- RM-G-9 - Fuel Handling Area F.H.B
    - 3.0 AUTOMATIC ACTIONS
        - FHB Vent Supply Fan AH-E-10 Trips on a high alarm
        - Dampers AH-D-120, AH-D-121, AH-D-122 Close on a high alarm

    - 4.0 MANUAL ACTIONS REQUIRED
        - 4.3 IAAT high alarm is Lit, then perform the following:
            - 1. ENSURE AH-E-10 is Shutdown.
            - 2. ENSURE the following are Closed:
                - AH-D-120
                - AH-D-121
                - AH-D-122

D. **Incorrect.**
Although securing AH-E-10 and closing AH-D-122 are interlocks for RM-G-9 and RM-A-4, AH-D-4 is not interlocked with either. Plausible and discriminatory becauseAH-E-10 securing and AH-D-122 shutting is an automatic interlock for RM-G-9 and it is indicated as open in the stem.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	072	K1.03	
	Importance Rating	3.6		

K/A: Knowledge of the physical connections and/or cause-effect relationships between the ARM system and the following systems: Fuel building isolation.

Proposed Question: RO Question # 56

Technical Reference(s): OP-TM-MAP-C0101, pg 5,25,Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: 252-GLO-11

Question Source: Bank # IR-252-GLO-11-Q01

Modified Bank #

New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	11
	55.43	

(11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:
The KA is matched because the question requires the candidate to determine the Cause-Effect relationship between Fuel Building Isolation and the Area Radiation Monitoring System.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to know radiation monitoring interlocks and affected components.

What MUST be known:
1. What are the interlocks associated with RM-A-4? 2. What are the interlocks associated with RM-G-9? 3. What manual actions must be taken to compensate for failed automatic actions associated with the Area Radiation Monitoring System?

57 **ID: 978610** **Points: 1.00**

Plant Conditions:

- Preparations for a Reactor Startup are in progress.
- The reactor is subcritical greater than one percent delta k/k.
- Tave is 532°F.

Event:

- OP-TM-AOP-017, Loss of VBC, is entered due to a loss of 120V AC Vital Bus 1C.

Given the above information, identify the Nuclear Instrumentation System function(s) below that has(have) been rendered INOPERABLE, if any.

A. NI-11 source range count rate indication, ONLY.

B. NI-3 high startup rate CRD Out-Motion Inhibit, ONLY.

C. NI-11 source range count rate indication, and
NI-3 high startup rate CRD Out-Motion Inhibit.

D. None of the above listed Nuclear Instrumentation functions have been rendered INOPERABLE.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
NI-11 Instrument Loop (powered by VBA through signal conditioning cabinet) is located in RPS Channel A cabinet, powered by VBA. Since there are four vital busses and only two each of Source Range and Intermediate Range Nuclear Instruments, the distracter is plausible if the candidate is not familiar with the power supplies for the Nuclear Instruments (There are four Power Range Nuclear Instruments and four power supplies. The alphabetical order matches the numerical order for power supplies. This is not the case with Source Range and Intermediate Range NI's).

B. **Correct.**
Except for Control Console indicators, the NI-3 Instrument Loop is located in RPS Channel C cabinet, powered by VBC. IAW OP-TM-AOP-017 (Rev. 6, p.11) Attachment 1:

- One channel of CRD Trip Confirm will de-energize to its actuated state but the other channel remains operable. Turbine will not trip and FW-V-5A/B will not close.
- "C" RPS channel deenergizes. CRD-CB-1C opens. "C" RPS channel trip signal is sent to all other RPS cabinets. (TS Table 3.5-1)
- **NI-3 SUR Rod Withdrawal Inhibit is inoperable.**
- RC-P-1C tripped signal is sent to RPS and HSPS EFW Initiation.
- ES Bistable Cabinet 3, Relay Cabinet 3A, and Relay Cabinet 3B deenergize, resulting in one channel tripped in all ES actuations, A and B trains.
- HSPS Channel III deenergizes. HSPS Channel III tripped signal is sent to HSPS Trains A & B.
- Mark V: Engineering Workstation is deenergized, but Operator Interface (CL) remains powered from ATB. The "Y" protection channel fails, but the "X" and "Z" protection channels remain available. Turbine throttle and governor valve position meters and Main Steam Pressure digital indication at Mark V panel fail.
- Multiple SASS transfers occur due to loss of HSPS Channel III instruments.
- Annunciator SER A deenergizes. Annunciator power transfers to VBD.
- IC-V-4, MU-V-18, MU-V-20, and MU-P-1B can not be transferred to RSD.
- RM-L-7 closes WDL-V-257.
- RM-A-8 trips AH-E-10 and AH-E-11, closes WDG-V-47, starts the MAP-5 Iodine Sampler.
- RM-A-15 starts the MAP-5 Iodine Sampler.
- WDL-V-535 fails closed.

C. **Incorrect.**
Except for Control Console indicators, the NI-3 Instrument Loop is located in RPS Channel C cabinet, powered by VBC. IAW OP-TM-AOP-017 (Rev. 6, p.11) Attachment 1, and therefore there is a Nuclear Instrument listed that will be INOPERABLE.

D. **Incorrect.**
NI-11 Instrument Loop (powered by VBA through signal conditioning cabinet) is located in RPS Channel A cabinet, powered by VBA. Since there are four vital busses and only two each of Source Range and Intermediate Range Nuclear Instruments, the distracter is plausible if the candidate is not familiar with the power supplies for the Nuclear Instruments (There are four Power Range Nuclear Instruments and four power supplies. The alphabetical order matches the numerical order for power supplies. This is not the case with Source Range and Intermediate Range NI's).

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	015	K2.01	
	Importance Rating	3.3		

K/A: Knowledge of bus power supplies to the following: NIS channels, components, and interconnections.

Proposed Question:	RO Question # 57		
Technical Reference(s):	OP-TM-AOP-017, pg 11, Rev 6		

Proposed References to be provided to applicants during examination: None

Learning Objective: 623-GLO-5

Question Source:	Bank #	
	Modified Bank #	IR-623-GLO-5-Q04
	New	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	2
	55.43	

(2) General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:
The KA is matched because the question requires the candidate to know the power supplies to Nuclear Instrumentation.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to know the affected equipment with a loss of Vital Bus C.

What MUST be known:
1. What are the power supplies to the Nuclear Instrumentation? 2. What are the Nuclear Instrumentation associated interlocks?

Original Question: IR-623-GLO-5-Q04

Plant conditions:

- Reactor at Hot Shutdown conditions.

Event:

- Electrical malfunction results in loss of 120V AC Vital Bus 1C.

Based on these conditions, identify the ONE Nuclear Instrumentation System function below that has been rendered INOPERABLE.

A. NI-5 gamma signal compensation.

B. NI-11 source range count rate indication.

C. NI-3 high startup rate CRD Out-Motion Inhibit.

D. NI-12 Reactor Building Evacuation Alarm actuation.

Answer: C

58 **ID: 978603** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Event:

- A loss of both the 4 Bus and 8 Bus occurs.
- EG-Y-1A, "A" Emergency Diesel Generator, has responded properly.
- EG-Y-4, SBO Diesel Generator, has been placed on the 1E 4kV bus.

Given the above information, vacuum in the Main Condenser will:

A. Be lost due to a loss of all vacuum pumps.

B. Be lost due the Turbine Bypass Valves failing open.

C. Not be lost due to the Turbine Bypass Valves latching closed.

D. Not be lost due to the standby Main and Aux Vacuum Pumps starting.

Answer: A

Answer Explanation

Explanation (Optional):

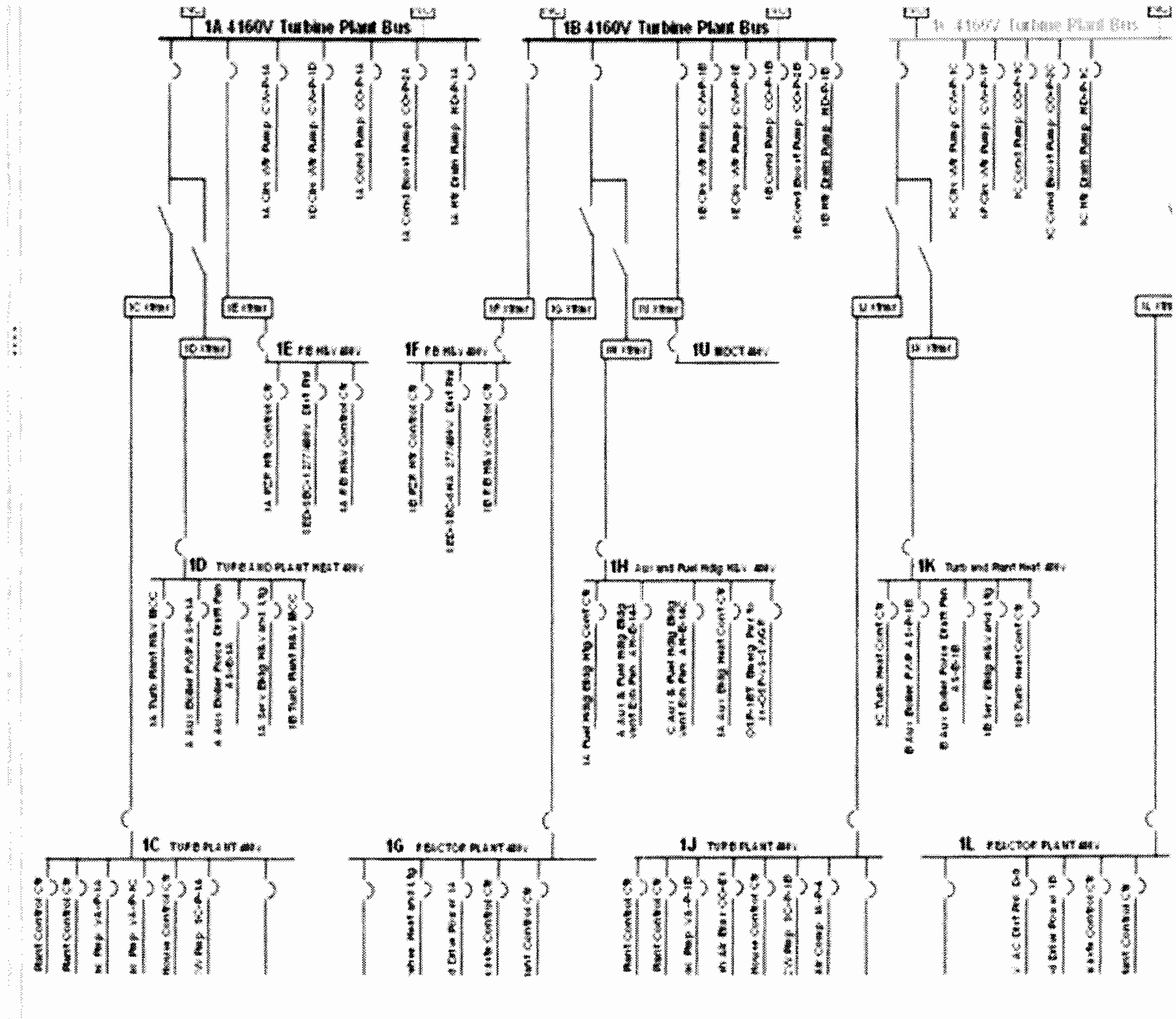
A. **Correct.**
The "A" and "C"Main Condenser Vacuum Pumps are powered from the 1C Turbine Plant 480V MCC. The "B" Main Condenser Vacuum Pump is powered from the 1J Turbine Plant MCC. These MCC's are both BOP powered (from 1A 4kV and 1C 4kV respectively). Even with both ES busses being energized, they cannot power the Main Condenser Vacuum Pumps. Additionally, the SBO Diesel Generator, EG-Y-4, could be placed on the 1C 4kV bus if both Emergency Diesel Generators were powering their associated bus. However, the SBO Diesel Generator in this case is on the 1E ES bus and so it cannot power the 1C 4kV BOP bus and the 1J 480V MCC to establish power to the "B" Main Condenser Vacuum Pump. With vacuum pumps lost, vacuum will degrade until the point where it is lost completely. IAW OP-TM-MAP-N0106, Main Condenser Vacuum Lo:

- 2.0 CAUSES
  - 2.1 High Circ Water Temperature
  - 2.2 Low Circ Water System flow
  - 2.3 Abnormal Condenser air in-leakage
  - 2.4 Loss of Condenser Vacuum Pumps
  - 2.5 Air binding in Water Boxes

B. **Incorrect.**
The Turbine Bypass Valves will latch closed due to a Loss of all Circulating Water Pumps. Control will be transferred to the Atmospheric Dump Valves. Therefore the Turbine Bypass Valves will not fail open. Plausible if the candidate is not familiar with the fact that a Loss of the 4 and 8 busses constitutes a Loss of Offsite Power and Circulating Water Pumps have been lost, therefore the Turbine Bypass Valves are closed.

C. **Incorrect.**
The Turbine Bypass Valves will latch closed due to a Loss of all Circulating Water Pumps. Control will be transferred to the Atmospheric Dump Valves. However, this is not enough to overcome the Loss of Main Condenser Vacuum Pumps. The purpose of the Turbine Bypass Valves latching closed is because of Main Condenser Vacuum deteriorating. Plausible if the candidate is not familiar with the purpose of why the Turbine Bypass Valves are closed.

D. **Incorrect.**
The "A" and "C"Main Condenser Vacuum Pumps are powered from the 1C Turbine Plant 480V MCC. The "B" Main Condenser Vacuum Pump is powered from the 1J Turbine Plant MCC. These MCC's are both BOP powered (from 1A 4kV and 1C 4kV respectively). Even with both ES busses being energized, they cannot power the Main Condenser Vacuum Pumps. Additionally, the SBO Diesel Generator, EG-Y-4, could be placed on the 1C 4kV bus if both Emergency Diesel Generators were powering their associated bus. However, the SBO Diesel Generator in this case is on the 1E ES bus and so it cannot power the 1C 4kV BOP bus and the 1J 480V MCC to establish power to the "B" Main Condenser Vacuum Pump. Plausible since loss of power to a vacuum pump will result in start of the standby pump; however on a loss of offsite power all vacuum pumps lose power.



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	055 K3.01	
	Importance Rating	2.5	

K/A: Knowledge of the effect that a loss or malfunction of the CARS will have on the following: Main Condenser.

Proposed Question: RO Question # 58

Technical Reference(s): 1107-5, pg 101, Rev 143
OP-TM-MAP-N0106, pg 1, Rev 8

Proposed References to be provided to applicants during examination:			None
Learning Objective:	331-GLO-11		
Question Source:	Bank #		
	Modified Bank #	IR-331-GLO-11-Q02	
	New		
Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	4	
	55.43		

(4) Secondary coolant and auxiliary systems that affect the facility.

Comments:
The KA is matched because the question requires the candidate to determine what effect a loss of the Condenser Air Removal System (Main Vacuum Pumps) will have on the Main Condenser.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to know the components that affect main condenser vacuum and the power supplies to those components.

What MUST be known:
1. What are the power supplies for the Main Condenser Vacuum Pumps? 2. How does a loss of Main Condenser Vacuum Pumps affect vacuum within the Main Condenser?

Original Question: IR-331-GLO-11-Q02

Plant Conditions:

The plant is at 100% power
Normal equipment lineups

Event:

A loss of offsite power has occurred
Both diesel generators operated as expected

With the above conditions __________.

A. condenser vacuum will be lost

B. the Atmospheric dump valves will fail open

C. the standby Main and Aux Vacuum Pumps will start

D. the Turbine Bypass Valves will have to be operated in hand

Answer: A

59 **ID: 978585** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- A Waste Gas release is in progress from a Waste Gas Decay Tank.

Event:

- Waste Gas Release Stop and Control Valve, WDG-V-47, closes and terminates the release .

Given the above conditions, which one of the following caused WDG-V-47 close?

A. Waste gas compressor trip.

B. Low pressure in the Waste Gas Decay Tank.

C. High Alarm on RM-A-6, Auxiliary Bldg Vent Exhaust.

D. High Alarm on RM-A-8, Aux and Fuel Handling Bldg Exhaust Duct.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
The Waste Gas Compressor is on the 'suction' side of Waste Gas Decay Tanks. Plausible if the candidate is not familiar with the physical layout of the Waste Gas system components. IAW RWA, Radwaste Panel A, RWA-1-6, Waste Gas to Atmosphere Terminated:

- Setpoint:
  - WDG-V-47 closed when demanded open.
- Causes:
  - WDG-FE-123 Waste Gas Decay Tank CFM release alarm
  - RM-A-7 Waste gas decay tank outlet radiation monitor
  - RM-A-8 Auxiliary Building ventilation radiation monitor
  - AH-E-14A-D trip

B. **Incorrect.**
Lowering pressure is the reason to perform the release. Plausible if the candidate believes that there is a Low Pressure trip associated with Waste Gas Decay Tanks that would close WDG-V-47. IAW 1104-27, Waste Disposal - Gaseous, Section 3.1.2:

> The Waste Gas System will be started and lined up to remove liberated gas that accumulates in the low pressure vent header from the tanks and vents served by the system. The Nitrogen System will be lined up to supply nitrogen to the Low Pressure Vent Header and blanket the tanks in the system.

C. **Incorrect.**
RM-A-6 secures AH-E-11 but does not close WDG-V-47. Plausible if the candidate is not familiar with the interlocks associated with RM-A-6. IAW OP-TM-MAP-C0101:

- RM-A-6 Auxiliary Building Vent Exhaust
    - 3.0 AUTOMATIC ACTIONS
        - Auxiliary Building Supply Fan AH-E-11 trips on gaseous Hi Alarm

D. **Correct.**
High Radiation on the Aux Building ventilation stack radiation monitor (RM-A-8) will trip the release (close WDG-V-47). IAW OP-TM-MAP-C0101:

- RM-A-8 Aux. and Fuel Handling Building Exhaust Duct
    - 3.0 AUTOMATIC ACTIONS
        - WDG-V-47 closes on a gaseous Hi Alarm
        - AH-E-10 and 11 will trip on a gaseous Hi Alarm
        - Remote sampler starts (MAP-5).

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	071	K4.04	
	Importance Rating	2.9		

K/A: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Isolation of waste gas release tanks.

Proposed Question: RO Question # 59

Technical Reference(s): OP-TM-MAP-C0101, pg 13, Rev 002A

Proposed References to be provided to applicants during examination: None

Learning Objective: 231-GLO-10

Question Source:
Bank #
Modified Bank #
New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 13

55.43

(13) Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Comments:
The KA is matched because the question requires the candidates to understand what equipment is designed to have an interlock that would isolate the Waste Gas Release Tanks for releasing to the environment.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must know the interlocks and associated components affected by radiation monitoring alarms.

What MUST be known:
1. What are interlocks associated with WDG-V-47?

60 **ID: 978573** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- The status of CRD-PGM-R17-18, Digital Control Rod Drive System PGM Module for Rods 17 and 18, is as follows:
  - Slices A and B indicate solid green health indicator lights.
  - Slice C indicates a solid red health indicator light.

Event:

- The Slice A health indicator for CRD-PGM-R17-18, Digital Control Rod Drive System PGM Module for Rods 17 and 18, changes from solid green to solid red.

Given the above information, what is the correct automatic response that will occur FIRST?

A. The Reactor will automatically trip.

B. Two Control Rods will deenergize and fully insert, ONLY.

C. Two Groups of Control Rods will deenergize and fully insert.

D. One Group of Control Rods will deenergize and fully insert, ONLY.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Once a second slice has a fault the module will go to a fail-safe condition and deenergize. Since PG/M module can only control Two control rods, those two rods will deenergize and fully insert into the core. Plausible if the candidate is not familiar with the results and believes that a fail-safe condition on a single PG/M will cause a Reactor Trip.

B. **Correct.**
IAW TQ-TM-104-622-C001, Control Rod Drive System:

- At the heart of the DCRDS is the Trusted Industrial Control System (ICS Trusted) located in the Relay Room. The system utilizes a Triple Modular Redundant (TMR) scheme within the hardware to achieve internal redundancy of all critical circuits. Triple Modular Redundancy is achieved by utilizing three identical slices in each module that perform identical functions simultaneously and independently. The output of each slice is voted in a majority-voting circuit before affecting the system's outputs, ensuring that a single hardware failure does not effect normal system operation
- All modules have front panel status LEDs that indicate the modules status (Healthy and Educated) and mode of operation (Active and Standby)
- Unhealthy modules will result in flashing or solid red indicators, which may be observed on operator rounds, and will result in a control room alarm.

Additionally, Once a second slice has a fault the module will go to a fail-safe condition and deenergize. Since PG/M module can only control Two control rods, those two rods will deenergize and fully insert into the core (their fail-safe condition).

C. **Incorrect.**
Once a second slice has a fault the module will go to a fail-safe condition and deenergize. Since PG/M module can only control Two control rods, those two rods will deenergize and fully insert into the core. Those two rods may or may not be in the same Group. Plausible if the candidate is not familiar with the failure mode of the PGM module and assumes that the rods are in separate Groups and that those two entire Groups of Control Rods will deenergize and fully insert.

D. **Incorrect.**
Once a second slice has a fault the module will go to a fail-safe condition and deenergize. Since PG/M module can only control Two control rods, those two rods will deenergize and fully insert into the core. Those two rods may or may not be in the same Group. Plausible if the candidate is not familiar with the failure mode of the PGM module and assumes that the rods are both in the same Group and that the entire Group of Control Rods will deenergize and fully insert.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	001	K5.65
	Importance Rating	3.2	

K/A: Knowledge of the following operational implications as they apply to the CRDS: CRDS circuitry, including effects of primary/secondary power mismatch on rod motion.

Proposed Question: RO Question # 60

Technical Reference(s): TQ-TM-104-622-C001, pg 55, Rev 007

Proposed References to be provided to applicants during examination:			None
Learning Objective:	622-GLO-5		
Question Source:	Bank #		
	Modified Bank #	IR-622-GLO-11-Q007	
	New		
Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	6	
	55.43		

(6) Design, components, and functions of reactivity control mechanisms and instrumentation.

Comments:
The KA is matched because the question requires the candidates to have a knowledge about CRDS circuitry, including faulted power slices within a Pulse Generator/Monitoring module.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to know indications and cause-effect parameters for the Digital Rod Control System.

What MUST be known:
1. What do the different indicating colors represent on a PG/M module? 2. What is the effect on the CRDS upon a failure of a PG/M module?

Original Question:

Plant Conditions:

- With regards to the Digital Control Rod Drive System, PG/M (Pulse Generator/Monitor) Module #4:

- Slices B and C indicate, solid green health indicator lights.
- Slice A indicate, solid red health indicator light.

Event:

- Slice B health indicator changes from solid green to solid red.

Based on the above event, what is the correct response?

A. The Reactor will automatically trip.

B. Two Control Rods will deenergize and fully insert **Only**.

C. MAP G-3-4, CRD System Fault will be received in the CR **Only.**

D. One Group of Control Rods will deenergize and fully insert **Only**.

Answer: B

61 **ID: 978572** **Points: 1.00**

Plant Conditions:

- The plant is operating at 40% power.
- OTSG Pressure Instrument is selected to "A OTSG", SP6A-PT1, for Control & Indication.

Event:

- MAP H-3-2, SASS Mismatch, is in alarm.
- The selected ""A OTSG #1", SP6A-PT1, pressure transmitter is failing slowly upscale
- Pressure signal rising at 6%/second as observed at CC.

Given the above information, SASS will ___(1)___ to the unaffected transmitter, and the ___(2)___.

A. 1) swap
2) TBV and ADV control will not be affected.

B. 1) swap
2) TBVs must be controlled in manual at the ICS station. ADVs must be controlled on the Backup Loader

C. 1) not swap
2) TBV and ADV control will not be affected.

D. 1) not swap
2) TBVs must be controlled in manual at the ICS station. ADVs must be controlled on the Backup Loader

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. With a slow instrument signal failure < 8%/second, the automatic signal transfer is blocked for the affected channel. Therefore, no SASS automatic swap to the alternate instrument will occur.

Part 2 is incorrect. The false high pressure signal will cause TBVs and ADVs to open. Manual control of TBVs must be taken at the ICS station on CC and ADVs on the Back Loader Station on CC. Then both sets of valves can be closed to mitigate the transient.

B. **Incorrect.**
Part 1 is incorrect. With a slow instrument signal failure < 8%/second, the automatic signal transfer is blocked for the affected channel. Therefore, no SASS automatic swap to the alternate instrument will occur.

Part 2 is correct. The false high pressure signal will cause TBVs and ADVs to open. Manual control of TBVs must be taken at the ICS station on CC and ADVs on the Back Loader Station on CC. Then both sets of valves can be closed to mitigate the transient.

C. **Incorrect.**
Part 1 is correct. With a slow instrument signal failure < 8%/second, the automatic signal transfer is blocked for the affected channel. Therefore, no SASS automatic swap to the alternate instrument will occur.

Part 2 is incorrect. The false high pressure signal will cause TBVs and ADVs to open. Manual control of TBVs must be taken at the ICS station on CC and ADVs on the Back Loader Station on CC. Then both sets of valves can be closed to mitigate the transient.

D. **Correct.**
Part 1 is correct. With a slow instrument signal failure < 8%/second, the automatic signal transfer is blocked for the affected channel. Therefore, no SASS automatic swap to the alternate instrument will occur.

Part 2 is correct. The false high pressure signal will cause TBVs and ADVs to open. Manual control of TBVs must be taken at the ICS station on CC and ADVs on the Back Loader Station on CC. Then both sets of valves can be closed to mitigate the transient.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	041	K6.03
	Importance Rating	2.7	

K/A: Knowledge of the effect of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS.

Proposed Question: RO Question # 61

Technical Reference(s): OP-TM-MAP-H0302, pg 1, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: 621-GLO-8

Question Source: Bank # IR-621-GLO-8-Q08
Modified Bank #
New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	

(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:
The KA is matched because the question requires the candidates to understand an ICS controller malfunction and the candidate must understand the effect on the Steam Dump System.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to know the operation of SASS, be able to analyze plant conditions to determine the status of SASS components, and have knowledge on the affect of a high OTSG pressure on the operation of the Turbine Bypass Valves and the Atmospheric Dump Valves.

What MUST be known:
1. What are the different types of failure recognized by SASS? 2. How will a failing OTSG Pressure Instrument affect the associated Turbine Bypass Valves and Atmospheric Dump Valve?

62 **ID: 978570** **Points: 1.00**

Plant Conditions:

- Preparations for refueling operations are complete.
- RCS and Fuel Transfer Canal (FTC) boron exceeds minimum required for refueling operations.
- One door is closed on both the Personnel and Emergency Access hatches.
- Fuel Transfer Canal level exceeds minimum (Tech Spec) requirements for refueling operations.
- Fuel Transfer valves FH-V-1A/B are open.
- IRRADIATED fuel assembly is being removed from the core for transport to the Spent Fuel Pool.

Event:

- The Control Room operator is ready to initiate a Reactor Building Purge.

Given the above information, identify the one statement below that describes operational implications of these operations during the movement of irradiated fuel.

A. Personnel passage into/out of the RB is NOT PERMISSIBLE at this time and BOTH doors at each of the personnel and emergency air locks are required to be closed.

B. FTC level will RISE from the initial level when Purge Exhaust Fans, AH-E-7A/B, are started and at least one foot of freeboard is required in the FTC prior to starting the purge.

C. FTC level will LOWER up to one foot from the initial level when Purge Supply Fans, AH-E-6A/B, are started andFTC level is required to be above the "black band" prior to starting the purge.

D. Both doors in at least one air lock (personnel or emergency) must be open before starting the purge exhaust fans to ensure that excessive differential pressure across the door does not result.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Requirement is for at least one door at each air lock to be capable of being closed. IAW 1505-1, Fuel and Control Component Shuffles, Data Sheet 2, Daily Reactor Building Checklist:

- RB personnel, emergency and equipment hatches capable of being closed or secured as defined in TS 3.8.6 (CM-1)

T.S. 3.8.6:

- During the handling of irradiated fuel in the Reactor Building at least one door in each of the personnel and emergency air locks shall be capable of being closed. The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

B. **Correct.**
IAW OP-TM-823-000, Reactor Building Heating and Ventilation System, Precautions:

2.1.8 Starting and stopping an RB purge can affect RB pressure. During fuel transfer operations (fuel transfer valves open) a decrease in RB pressure will cause a level increase in the fuel transfer canal. Prior to starting AH-E-7A/7B, fuel transfer canal freeboard must be verified greater than 1 foot.

C. **Incorrect.**
Refer to OP-TM-823-000 FTC level can rise up to one foot due to start of AH-E-7A/B (exhaust fans). AH-E-7A/B are started before AH-E-6A/B.

Distracter is plausible if examinee does not know RB Purge startup sequence (exhaust fans are started before supply fans).

D. **Incorrect.**
Both doors in at least one air lock (personnel or emergency) must be open before starting the purge exhaust fans to ensure that excessive differential pressure across the door does not result.

Distracter is plausible because both doors will be open after the purge is begun, and excessive high differential pressures have existed across those doors in the past.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	034	A1.02	
	Importance Rating	2.9		

K/A: 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.

Proposed Question: RO Question # 62

Technical Reference(s): OP-TM-823-000, pg 3, Rev 007

Proposed References to be provided to applicants during examination: None

Learning Objective: 824-GLO-8

Question Source: Bank # IR-824-GLO-8-Q01
Modified Bank #
New

Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	9	
	55.43		

(9) Shielding, isolation, and containment design features, including access limitations.

Comments:
The KA is matched because the question requires the candidates to predict what will happen to a Fuel Handling System parameter (water level in the refueling canal) when a Reactor Building Purge is started.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to understand the operation of the RB purge system and affects on the fuel transfer canal, containment door / hatch requirements prior to starting a reactor building purge.

What MUST be known:
1. What are the implications on the Fuel Handling System (water level in the refueling canal) when a Reactor Building purge is initiated?

63 **ID: 979107** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- High RCS activity exists due to fuel pin failure.
- There is High Containment Building activity.
- Kidney Filter Fan, AH-E-101, is operating to reduce activity in preparation for personnel entry.
- The Control Switches for RB Clean-up System Water Spray Pumps, FS-P-5A/B, are in the Normal positions.

Event:

- 2 of the 15 Kidney Filter Charcoal Filter temperature switches (AH-TS-928A-O) are 200 degrees F and rising due to adsorption of iodine.
- AH-E-101 continues to run.

Given the above information and assuming that temperatures will continue to rise, which one of the following describes:
(1) The Automatic or Manual response required, and
(2) The FIRST procedural action to take when the Shift Manager determines that fire system operation is no longer necessary.

A. (1) Manually trip AH-E-101 and ensure FS-P-5A/FS-P-5B are running IAW OP-TM-HVB-0609, Rx Bldg Kidney Filter Trouble (Hi Temp).
(2) Reset temperature switch actuation at RB Cleanup Filter Panel.

B. (1) Manually trip AH-E-101 and ensure FS-P-5A/FS-P-5B are running IAW OP-TM-HVB-0609, Rx Bldg Kidney Filter Trouble (Hi Temp).
(2) Secure FS-P-5A and FS-P-5B from the Remote Control Switches outside of the RB Personnel Hatch.

C. (1) AH-E-101 will trip automatically and then FS-P-5A/FS-P-5B will automatically start once a third Kidney Filter Charcoal Filter temperature switch reaches 200 degrees F because a 3 out of 15 logic is required.
(2) Reset temperature switch actuation at RB Cleanup Filter Panel.

D. (1) AH-E-101 will trip automatically and then FS-P-5A/FS-P-5B will automatically start once a third Kidney Filter Charcoal Filter temperature switch reaches 200 degrees F because a 3 out of 15 logic is required.
(2) Secure FS-P-5A and FS-P-5B from the Remote Control Switches outside of the RB Personnel Hatch.

Answer: A

Answer Explanation

For all answers, the controlling procedure is OP-TM-HVB-0609, and the actions applicable to the CRO is to ensure the automatic actions in response to this condition have occurred. The automatic actions are that AH-E-101 trips, and FS-P-5A/B auto start.

A. NOT CORRECT. The fan does not continue to run, and the pumps auto-start. Plausible since iodine removal can only occur if air flow is maintained. Wrong Procedure.

B. CORRECT. See above.

C. NOT CORRECT. The fan does not continue to run, and the pumps auto-start. Plausible since iodine removal can only occur if air flow is maintained. Wrong Procedure.

D. NOT CORRECT. The FS-P-5A/B pumps will auto-start. Plausible if the examinee does not recall the automatic start of the FS pumps. Correct Procedure.

Explanation (Optional):

A. **Correct.**
Part 1 is correct. AH-E-101 should have tripped and therefore manual actions are required to occur. IAW OP-TM-HVB-0609 Rx Bldg Kidney Filter Trouble (Hi Temp):

- 1.0 SETPOINTS
    - 200°F on any two of 15 temperature switches (AH-TS-928A-928O).
- 3.0 AUTOMATIC ACTIONS
    - AH-E-101 trips
    - FS-P-5A and FS-P-5B start
- 4.0 MANUAL ACTIONS REQUIRED
    - 4.1 ENSURE automatic actions have occurred.

Part 2 is correct. IAW OP-TM-HVB-0609 Rx Bldg Kidney Filter Trouble (Hi Temp):

4.5 When SM/CRS determines fire system operation is no longer necessary, then RESET temperature switch actuation at RB Clean-up Filter Panel.

B. **Incorrect.**
Part 1 is correct. AH-E-101 should have tripped and therefore manual actions are required to occur. IAW OP-TM-HVB-0609 Rx Bldg Kidney Filter Trouble (Hi Temp):

- 1.0 SETPOINTS
    - 200°F on any two of 15 temperature switches (AH-TS-928A-928O).
- 3.0 AUTOMATIC ACTIONS
    - AH-E-101 trips
    - FS-P-5A and FS-P-5B start
- 4.0 MANUAL ACTIONS REQUIRED
- 4.1 ENSURE automatic actions have occurred.

Part 2 is incorrect. IAW OP-TM-HVB-0609 Rx Bldg Kidney Filter Trouble (Hi Temp) gives no direction to secure FS-P-5A and FS-P-5B. Plausible since they are started IAW the procedure and would be the first step to return to normal if working backwards.

C. **Incorrect.**
Part 1 is incorrect. AH-E-101 should have tripped and therefore manual actions are required to occur because the interlock is a 2 out of 15 logic. Plausible if the candidate is not familiar with the interlock requirement for the Kidney Filter System. IAW OP-TM-HVB-0609 Rx Bldg Kidney Filter Trouble (Hi Temp):

- 1.0 SETPOINTS
  - 200°F on any two of 15 temperature switches (AH-TS-928A-928O).

Part 2 is correct. IAW OP-TM-HVB-0609 Rx Bldg Kidney Filter Trouble (Hi Temp):

- 4.5 When SM/CRS determines fire system operation is no longer necessary, then RESET temperature switch actuation at RB Clean-up Filter Panel.

D. **Incorrect.**
Part 1 is incorrect. AH-E-101 should have tripped and therefore manual actions are required to occur because the interlock is a 2 out of 15 logic. Plausible if the candidate is not familiar with the interlock requirement for the Kidney Filter System. IAW OP-TM-HVB-0609 Rx Bldg Kidney Filter Trouble (Hi Temp):

- 1.0 SETPOINTS
  - 200°F on any two of 15 temperature switches (AH-TS-928A-928O).

Part 2 is incorrect. IAW OP-TM-HVB-0609 Rx Bldg Kidney Filter Trouble (Hi Temp) gives no direction to secure FS-P-5A and FS-P-5B. Plausible since they are started IAW the procedure and would be the first step to return to normal if working backwards.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	027	A2.01	
	Importance Rating	3.0		

K/A: Ability to (a) predict the impacts of the following malfunctions or operations on the CIRS; and (b) based on those predictions, use Procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High temperature in the filter system.

Proposed Question: RO Question # 63

Technical Reference(s): OP-TM-HVB-0609, pg 1, Rev 1

Proposed References to be provided to applicants during examination: None

Learning Objective: 824-GLO-10

Question Source:	Bank #		
	Modified Bank #	IR-824-GLO-10-Q02	
	New		

Question History: Last NRC Exam: 2009 (TMI 08-01)

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

10 CFR Part 55 Content:	55.41	8
	55.43	

(8) Components, capacity, and functions of emergency systems.

Comments:
The KA is matched because, given high temperature in the Kidney Filter System, the candidate must recognize a malfunction of an automatic action to occur, and then take compensatory actions IAW the Alarm Response procedure.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to recall the interlock and associated components with the RB Kidney Filter system and also to recall the required actions to mitigate the event IAW procedures.

What MUST be known:
1. What is the logic used to trip AH-E-101? 2. What are the procedurally-driven Manual actions to take when Automatic actions have not occurred? 3. What are the procedurally-driven Manual actions to take when fir service operations are no longer required?

Original Question:

Plant Conditions:

- Reactor is operating at 100% power, with ICS in full automatic.
- High RCS activity exists due to fuel pin failure.
- High Containment Building activity.
- Kidney Filter Fan AH-E-101 is operating to reduce activity in preparation for personnel entry.
- RB Clean-up System Water Spray Pumps FS-P-5A/B control switches are both in Normal position.

Event:

- Kidney Filter charcoal RTD temperature indications at TR-858 are 200 degrees F, rising at 20 degrees per minute due to adsorption of iodine.

If these conditions continue, identify the ONE selection below that describes:
(1) The expected system operation, and
(2) The controlling procedure to be implemented.

A. (1) AH-E-101 trips, FS-P-5A and FS-P-5B automatically start.
(2) OP-TM-AOP-001, Fire.

B. (1) AH-E-101 trips, FS-P-5A and FS-P-5B automatically start.
(2) OP-TM-HVB-0609 Rx Bldg Kidney Filter Trouble (Hi Temp).

C. (1) Manually trip AH-E-101, and manually start FS-P-5A and FS-P-5B.
(2) OP-TM-AOP-001, Fire.

D. (1) Manually trip AH-E-101, and manually start FS-P-5A and FS-P-5B.
(2) OP-TM-HVB-0609 Rx Bldg Kidney Filter Trouble (Hi Temp).

Answer: B

64 **ID: 978042** **Points: 1.00**

Plant Conditions:

- A Liquid Radioactive Release is in progress from Waste Evaporator Condensate Storage Tank, WDL-T-11A.
- Liquid Radioactive Waste Release Monitor, RM-L-6, is fully operable.
- Plant effluent Radiation Monitor, RM-L-7, is energized with the PCR interlock defeat switch in the DEFEAT position due to erratic detector output.

Event:

- A Reactor Operator removes RM-L-7 from service by rotating the Control Switch on PCR to the "OFF" position.

Given the above information and after the event takes place, the liquid release will:

A. Continue because the RM-L-7 interlock is defeated.

B. Continue because the RM-L-7 detector is deenergized.

C. Terminate because of loss of power to the RM-L-7 interlock defeat circuit.

D. Terminate because RM-L-7 detector and WDL-V-257 solenoid power supplies are common.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
Placing the PCR interlock defeat switch for RM-L-7 will defeat the interlock.
IAW Drawing 209-708:
  - Defeating this interlock prevents any RM-L-7 signal from energizing Relay XRM-L7.

IAW Drawing 209-297:
  - Energizing the XRM-L7 Relay opens a contact to de-energize 20X-257 relay.
  - 20X-257 relay deenergizing will de-energize the WDL-V-257 solenoid at the valve, preventing it from closing WDL-V-257.

B. **Incorrect.**
If RM-L-7 was de-energized without being defeated, the interlock would terminate the release. Operation of the defeat switch disables RM-L-7 from terminating the liquid release, even if the detector is de-energized. See "B" explanation for the print logic.

Distracter is plausible if the Candidate is not familiar with interpreting the drawings and also because the interlock function is provided by an "energize to actuate" AC circuit. IAW TQ-TM-104-232-C001, Liquid Radwaste Disposal System, Section VI.C.3.b:

- WDL-V-257 ia air loaded open or closed by solenoid.
- WDL-V-257 will not OPEN unless the following interlocks are satisfied:
  - Unit II release valve (WDL-V-99) closed
  - MDCT flow greater than the setpoint.
  - RM-L-6 is less than the setpoint
  - RM-L-7 is less than the setpoint
  - FR-84 flow is less than the setpoint

C. **Incorrect.**
The defeat switch cuts out the interlock by opening a contact to prevent energizing XRM-L7, the high radiation interlock relay that starts the chain of events to de-energize WDL-V-257 to terminate the release if not defeated.

Distracter is plausible if the Candidate mistakingly believes that closing WDL-V-257 is via the normal fail-safe (de-energize to actuate) philosophy. In fact, if not defeated, de-energizing RM-L-7 will result is termination of the release.

D. **Incorrect.**
The defeat switch cuts out the interlock by opening a contact to prevent energizing XRM-L7, the high radiation interlock relay that starts the chain of events to de-energize WDL-V-257 to terminate the release if not defeated. Detector power supply is from 480V AC MCC, and WDL-V-247 solenoid power is from 1B RW MCC Unit 4AR (209-297 and 209-156). Distracter is plausible if the Candidate is unfamiliar with the power supplies of the Liquid Radwaste System, as they are both 120 VAC circuits.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	068	A3.02
	Importance Rating	3.6	

K/A: Ability to monitor automatic operation of the Liquid Radwaste System including: Automatic isolation.

Proposed Question: RO Question # 64

Technical Reference(s): 209-708, Rev 011
209-297, Rev 015

Proposed References to be provided to applicants during examination:		209-708, Rev 011 209-297, Rev 015

Learning Objective: 232-GLO-5

Question Source:	Bank #	IR-232-GLO-5-Q04
	Modified Bank #	
	New	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	11
	55.43	

(11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:
The KA is matched because the question requires the candidates to determine what happens to the Automatic Isolation ability of the Liquid Radwaste System given a set of conditions.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to interpret a set of prints to understand the status of interlocks and associated components with radiation monitoring.

What MUST be known:
1. What happens when RM-L-7 interlock is placed in defeat? 2. What happens electrically when the Control Switch for RM-L-7 is taken to the "OFF" position?

65 **ID: 978041** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.

Sequence of Events:

- MU-P-3B, Main Oil Pump for MU-P-1B, trips.
- Annunciator D-1-2, MU PMP 1B LUBE OIL PRESS LO, alarms and remains lit.
- Annunciator F-1-5, RCP SEAL TOT INJECT FLOW HI LO, alarms and remains lit.
- Annunciator G-2-5, PZR LEVEL HI LO, alarms and remains lit.

Given the above information and assuming no operator action, the "B" Makeup Pump, MU-P-1B is ___(1)___ and MU-P-2B, Aux Oil Pump for MU-P-1B, is ___(2)___.

A. (1) running
(2) running

B. (1) running
(2) not running

C. (1) not running
(2) running

D. (1) not running
(2) not running

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Plausible if candidate does not recognize that MU-P-1B will automatically trip on a loss of Main and Aux lube oil pumps (Less than 3 psig oil pressure). Candidate would then believe that a loss of lube oil to MU-P-1B would cause an increase in pump/motor bearing temperatures and therefore cause the motor to do more work to maintain the same amount of flow which causes motor current to increase, B-2-2 to alarm and motor would trip on overload.

D-1-2 - Candidate recognizes that alarm does not clear following the MU-P-1B trip

G-2-5 - would alarm within 3.5 minutes following a trip of MU-P-1B as tested in the simulator.

B. **Incorrect.**
Plausible if candidate does not recognize that MU-P-1B will automatically trip on a loss of Main and Aux lube oil pumps (Less than 3 psig oil pressure). Candidate would then believe that a loss of lube oil to MU-P-1B would cause an increase in pump/motor bearing temperatures and therefore cause the motor to do more work to maintain the same amount of flow which causes motor current to increase, B-2-2 to alarm and motor would trip on overload.

D-1-2 - Candidate recognizes that alarm does not clear following the MU-P-1B trip.

F-1-5 - Candidate recognizes that alarm will actuate on loss of MU-P-1B.

C. **Incorrect.**
Plausible if candidate does not recognize that MU-P-1B will automatically trip on a loss of Main and Aux lube oil pumps (Less than 3 psig oil pressure). Candidate would then believe that a loss of lube oil to MU-P-1B would cause an increase in pump/motor bearing temperatures and therefore cause the motor to do more work to maintain the same amount of flow which causes motor current to increase, B-2-2 to alarm and motor would trip on overload.

F-1-5 - Candidate recognizes that alarm will actuate on loss of MU-P-1B.

G-2-5 - would alarm within 3.5 minutes following a trip of MU-P-1B as tested in the simulator.

D. **Correct.**
Loss of both MU-P-2B and MU-P-3B, as evidence by solid green indicators on Control Room indication would actuate annunciator D-1-2, MU PMP 1B Lube Oil Press Lo. This annunciator actuates at 5psig lowering lube oil pressure IAW OP-TM-MAP-D-1-2. MU-P-2B, Aux oil pump for MU-P-1B should auto start on the 5 psig lube oil signal, however with the Control Room indicator a solid green for MU-P-2B, this indicates the pump failed to auto start and lube oil pressure would immediately drop below 3 psig and cause MU-P-1B to trip. With the only running makeup pump tripped, annunciator F-1-5 would immediately alarm due to low RCP seal injection flow and G-2-5 would alarm approximately 3.5 minutes later for Pressurizer low level if no further operator actions occurred.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	011	A4.01	
	Importance Rating	3.5		

K/A: Ability to manually operate and/or monitor in the control room: Charging pump and flow controls.

Proposed Question: RO Question # 65

Technical Reference(s):
OP-TM-MAP-D0102, pg 1, Rev 001
OP-TM-MAP-F0105, pg 1, Rev 002
OP-TM-MAP-G0205, pg 1, Rev 003

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO-11

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	3
	55.43	

3) Mechanical components and design features of the reactor primary system.

The KA is matched because the indications given in the question regarding the monitoring of Charging flow make it possible for the candidates to predict what they would observe (monitor) in relation to the Charging Pump (Makeup Pump).

The question is at the Comprehension/Analysis cognitive level because the candidate is required to interpret alarm indications for the Makeup system, determine the status of the system based on the conditions, and predict the system response based on system knowledge and response and plant conditions.

What MUST be known:
1. What is the implication of Annunciator D-1-2 remaining in alarm? 2. What are the causes for Annunciator F-1-5 to alarm? 3. What are the causes for Annunciator G-2-5 to alarm? 4. What is the impact on the Charging (Makeup) System if the combination of Annunciators D-1-2, F-1-5, and G-2-5 are all in alarm?

66 **ID: 978040** **Points: 1.00**

Plant Conditions:

- An operator is performing a procedure for the Powdex System.
- One of the steps in the procedure is a "WAAT" step.

Event:

- As the operator is performing the procedure, he reaches the last step.

With regards to the open "WAAT" step, the operator should:

A. Wait until the condition is met and then perform the step.

B. Mark the step as "N/A" (Not Applicable) and then exit the procedure.

C. Mark the step as "CNM" (Condition Not Met) and then exit the procedure.

D. Leave the step blank and then tell the Reactor Operator to log the procedure as complete with noted exceptions.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
IAW HU-TM-104-101-1001, Procedure Utilization, Step 4.1.10.4:
- "WAAT" (When at any time) is used for actions which are required to complete the procedure but the action is performed only when a specific condition occurs.
  - MAINTAIN cognizance of these conditions throughout procedure performance, until the condition occurs.
  - When the conditions(s) occurs, then PERFORM the required action and MARK the step (checkoff or initials) when the action is complete..

B. **Incorrect.**
Plausible if the candidate confuses the WAAT with an IAAT step (HU-TM-104-101-1001, Procedure Utilization, Section 4.1.10.2)

- "IAAT" (If at any time) is used for contingency actions where the action is continuously applicable until the end of the procedure:
  - MAINTAIN cognizance of the conditions when the step applies throughout procedure performance.
  - If the condition(s) occurred, then PERFORM the required action and MARK the step (checkoff or initials).
  - If the procedure is complete and the conditions(s) did not occur, then MARK the step as N/A.

C. **Incorrect.**
Term no longer used in HU-TM-104-101-1001, Procedure Utilization,. Plausible as it is a legacy term that the candidate may refer back to.

D. **Incorrect.**
Not discussed in HU-TM-104-101-1001, Procedure Utilization,. Plausible if the candidate is not familiar with both the requirements of HU-TM-104-101-1001 and the Document Rigor policy within Exelon Nuclear.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.20	
	Importance Rating	4.6	

K/A: Ability to interpret and execute procedure steps.

Proposed Question: RO Question # 66

Technical Reference(s): HU-TM-104-101-1001, pg 6, Rev 6

Proposed References to be provided to applicants during examination: None

Learning Objective: PREWATCH-DBIG-APCO-1

Question Source: Bank # IR-HUTM104-PCO-3-Q01
Modified Bank #
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 10
55.43

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidates to interpret the requirements to complete procedure steps prior to closing out the procedure.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to recall procedural requirements and direction.

What MUST be known:
1. What are the requirements with respect to an open When-At-Any-Time step when coming to the end of the procedure?

67 **ID: 978034** **Points: 1.00**

Plant Conditions:

- Refueling Operations are in progress.
- "A" Decay Heat Removal Loop is in operation.
- RCS temperature at the "A" Decay Heat Removal Pump, DH-P-1A, suction is 130°F and steady.

Event:

- RCS Temperature at the "A" Decay Heat Removal Pump suction has risen 6°F, and continues to rise at 1°F every 15 minutes.

Given the above information and assuming the RCS temperature continues to rise consistently, identify the one selection below that describes the maximum time allowed to continue core alterations.

A. 1 Hour.

B. 2 Hours.

C. 16 Hours.

D. 17 Hours.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
Tech Spec 1.2.6 REFUELING SHUTDOWN

> The reactor is in the refueling shutdown condition when, even with all rods removed, the reactor would be subcritical by at least one percent delta k/k and the coolant temperature at the decay heat removal pump suction is no more than 140°F. Pressure is defined by Specification 3.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.

Given an initial "A" Decay Heat Removal Pump, DH-P-1A, suction temperature of 130°F and a rise of 6°F, the difference between 140 and 136 is 4°F. With a rise of 1°F every 15 minutes times 4°F, the "A" Decay Heat Removal Pump, DH-P-1A, suction temperature will reach 140°F in 60 minutes, or 1 Hour.

B. **Incorrect.**
If the examinee recognizes the Tech Spec requirement of 140°F "A" Decay Heat Removal Pump, DH-P-1A, suction temperature, but fails to account for the rise of 6°F in temperature, the difference between 140 and 130 is 10°F. With a rise of 1°F every 15 minutes times 10°F, the "A" Decay Heat Removal Pump, DH-P-1A, suction temperature will reach 140°F in 2.5 Hours. Since this falls between 2 Hours and 3 Hours, and the question is asking maximum time, 2 Hours would be plausible.

C. **Incorrect.**
Incorrect but plausible if the examinee believes that RCS temperature is the highest concern.

Tech Spec 1.2.1 COLD SHUTDOWN

> The reactor is in the cold shutdown condition when it is subcritical by at least one percent delta k/k and Tavg is no more than 200°F. Pressure is defined by Specification 3.1.2.

Given an initial RCS temperature of 130°F and a rise of 6°F, the difference between 200 and 136 is 64°F. With a rise of 1°F every 15 minutes times 64°F, the RCS temperature will reach 200°F in 16 Hours.

D. **Incorrect.**
If the examinee recognizes the Tech Spec requirement of 200°F RCS temperature, but fails to account for the rise of 6°F in temperature, the difference between 200 and 130 is 70°F. With a rise of 1°F every 15 minutes times 0°F, the RCS temperature will reach 200°F in 17.5 Hours. Since this falls between 17 Hours and 18 Hours, and the question is asking maximum time, 17 Hours would be plausible.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	2.1.36	
Importance Rating	3.0	

K/A: Knowledge of procedures and limitations involved in core alterations.

Proposed Question: RO Question # 67

Technical Reference(s): TS 1.0, pg 1-1, Rev 278

Proposed References to be provided to applicants during examination: None

Learning Objective: 212-GLO-14

Question Source: Bank # ILT 10-02 NRC
Modified Bank #
New

Question History: Last NRC Exam: 2012 (TMI 10-02)

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

10 CFR Part 55 Content:	55.41	10
	55.43	

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of limitations involved in core alterations. The knowledge that must be demonstrated is the maximum temperature allowed for performing core alterations.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information and then apply this information to a set of plant conditions to correctly answer the questions.

What MUST be known:
1. What is the maximum temperature allowed per Tech Specs for performing core alterations? 2. The math to figure out time until a temperature limit is reached.

68 **ID: 977986** **Points: 1.00**

While performing fuel handling operations IAW 1505-1, Fuel and Control Component Shuffles, which one of the following requires permission from the CRO **and** the SRO (Licensed Fuel Handling Supervisor):

A. Inserting a control rod assembly into the core.

B. Withdrawing a spent fuel assembly from the core.

C. Disengaging the grapple from a fuel assembly in the Spent Fuel Pool.

D. Removing a control rod from fuel assembly in the Spent Fuel Pool upender basket.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.** Approval of both the CRO and LFHS (SRO) is not required per 1505-1, Fuel and Control Component Shuffles. Plausible since permission from both is required to insert a fuel assembly into the core; or withdraw control components from the core, but not to insert control components such as the Control Rod Assembly.

B. **Correct.** 1505-1, Fuel and Control Component Shuffles, Step 5.3.5 requires approval of both the CRO and LFHS (SRO) prior to removing a fuel assembly from the core. IAW 1505-1:

5.3.5 Approval must be given by CRO and LFHS prior to performing the following (communications are included in 1507-3):
- Remove or insert fuel assembly out of or into core.
- Disengage grapple from assembly being inserted into core.
- Withdraw control component from any fuel assembly in core.

C. **Incorrect.** Approval of both the CRO and LFHS (SRO) is not required per 1505-1, Fuel and Control Component Shuffles. Plausible since permission from both is required to disengage the grapple from a fuel assembly in the core; however it is not required in the Spent Fuel Pool.

D. **Incorrect.** Approval of both the CRO and LFHS (SRO) is not required per 1505-1, Fuel and Control Component Shuffles. Plausible since permission from both is required to remove a control component from a fuel assembly in the core; however it is not required in the Spent Fuel Pool.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.42	
	Importance Rating	2.5	

K/A: Knowledge of new and spent fuel movement procedures.

Proposed Question: RO Question # 68

Technical Reference(s): 1505-1, pg 9, Rev 57

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-DBIG-APCO-1

Question Source:	Bank #	IR-0340030101-Q01
	Modified Bank #	
	New	

Question History: Last NRC Exam: 2009 (TMI 08-01)

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

10 CFR Part 55 Content:	55.41	12
	55.43	

(12) Radiological safety principles and procedures.

Comments:
The KA is matched because the question requires the candidates to demonstrate a knowledge of a procedure associated with spent fuel movement at the RO level.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must have knowledge of procedural requirements associated with spent fuel movements.

What MUST be known:
1. What spent fuel activities require both CRO and SRO permissions?

69 **ID: 907833** **Points: 1.00**

Plant Conditions:

- A reactor startup is in progress IAW 1103-8, Approach to Criticality.

Sequence of Events:

- The following information was determined IAW OP-TM-300-403, Estimated Critical Rod Position, Attachment 7.1:
  - Initial count rate was 4 x 10^1 cps on Source Range NI-11 and NI-12.
  - Estimated Critical Rod Position is 50% WD on CRG 6.
  - Minimum Rod Withdrawal Limit is 95% Rod Index.
  - Maximum Rod Withdrawal Limit is 225% Rod Index.

- Currently:
  - Rod Position is 25% on CRG 7.
  - Count Rate is stable at 6 x 10^2 cps on Source Range NI-11 and NI-12.
  - A new ECB has been calculated but has **NOT** been approved.

Given the above information, which one of the following is the next action that will be taken IAW 1103-8?

A. Begin emergency boration to achieve 1% dk/k shutdown margin.

B. Begin a dilution to lower rod position and continue the reactor startup.

C. Continue to withdraw Control Rod Group 7 and continue the reactor startup.

D. Insert Rods in sequence until Group 7 Rods and Group 6 Rods are fully inserted.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.** Emergency Boration is performed IAW Rule 5, Emergency Boration. Entry criteria for Rule 5 is as follows:

- IAAT any of the following conditions exist:
  - Emergency boration is directed by procedure,
  - Reactor is shutdown and all control rods are not fully inserted,
  - Reactor is shutdown and Neutron flux is not lowering as expected,
- then Emergency Borate as follows:

The Controlling procedure given in the stem was 1103-8, Approach to Criticality. Rule 5 is not referenced from 1103-8 at any time. Plausible if the candidate believes that, since there was a reactivity miscalculation, then Emergency Boration must be performed.

B. **Incorrect.** A dilution of Reactor Coolant would lower Boron concentration. Lower Boron while maintaining power constant would cause a need for rods to be inserted. Once the rods have been inserted, then reactor power could be raised while staying below the upper limit of the ECP. Plausible since this would allow the startup to proceed. However, this is not a conservative choice and would invalidate the data input for the ECP.

C. **Incorrect.** 1103-8, Approach to Criticality, Step 3.2:

3.2.1 IAAT ECB becomes INVALID, then
1. If a new ECB has NOT already been approved, then GO TO Step 3.3 "Missed ECP".
2. If a new ECB has already been approved, then
(a) ENSURE rods are inserted greater than the MINIMUM ROD WITHDRAW LIMIT of the new ECP.
(b) GO TO Step 3.2.2.

The conditions given in the stem state that the new ECB has not been approved yet. Therefore, the reactor startup is not allowed to continue. Plausible if the candidate does not recognize the requirement to have the new ECB approved and not merely calculated.

D. **Correct**. IAW 1103-8, Approach to Criticality:

2.4 If criticality occurs outside the MIN and MAX ROD WITHDRAW LIMITS calculated IAW OP-TM-300-403, Estimated Critical Rod Position, then control rods shall be inserted to maintain at least a 1% ƒ´k/k subcritical condition.

3.3 If the ECP was missed or ECB/ECP becomes INVALID, then perform the following:

3.3.1 INSERT control rods in sequence, until the rod group which was being withdrawn is fully inserted and one additional group is fully inserted.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.1	
	Importance Rating	4.5	

K/A: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Proposed Question: RO Question # 69

Technical Reference(s): 1103-8, pg 3,9, Rev 54

Proposed References to be provided to applicants during examination: None

Learning Objective: GOP-003-PCO-4

Question Source:	Bank #	IR-GOP-003-PCO-4-Q04
	Modified Bank #	
	New	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	1
	55.43	

(1) Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Comments:
The KA is matched because the question requires the candidates to understand the pre-startup procedure for TMI-1, specifically, what is the correct action to take when an ECP is missed, which could affect reactivity.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze plant conditions and interpret CRD indications, have knowledge on procedural requirements for the approach to criticality, understand the indications associated with the approach to criticality, and determine correct actions if indications and plant conditions are outside procedural requirements.

What MUST be known:
1. What are the indications necessary to call the reactor critical? 2. What are the required actions to perform for a missed ECP?

70 **ID: 977950** **Points: 1.00**

Plant Conditions:

- As part of an approved Clearance, an operator has been assigned to apply a Danger Tag to a drain valve in the CLOSED position.
- The drain valve is a normally closed valve that has a failed controller that would automatically open the valve based on water level.

Event:

- Upon arrival at the valve the operator observes that a tag (associated with a separate Clearance) has already been applied to the valve.

Given the above information and IAW OP-MA-109-101, Clearance and Tagging, the operator can apply the Danger Tag if the original tag is a(n) ___(1)___ tag that ___(2)___.

A. (1) Information
(2) lists the valve as closed, ONLY

B. (1) Information
(2) lists the manual steps to take, compensating for failure of the automatic controller

C. (1) Special Condition
(2) lists the valve as closed, ONLY

D. (1) Special Condition
(2) lists the manual steps to take, compensating for failure of the automatic controller

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.** Part 1 is correct. Part 2 is correct. A Danger Tag can be placed on a component with an Information Tag already in place as long as the positions do not conflict. IAW OP-MA-109-101, Clearance and Tagging, Section 5.2: Danger Tags:

> 4. A danger tag can be applied to a component bearing another danger tag or an information tag provided the component positions do not conflict. The danger tag shall not be obstructed by the information tag.

B. **Incorrect.** Part 1 is correct., Part2 is incorrect. A Danger Tag can be placed on a component with an Information Tag already in place as long as the positions do not conflict. Plausible if the candidates believe that a Danger Tag can be applied to any Information Tag because the Danger Tag is more important and therefore trumps the Information Tag. IAW OP-MA-109-101, Clearance and Tagging, Section 5.2: Danger Tags:

> 4. A danger tag can be applied to a component bearing another danger tag or an information tag provided the component positions do not conflict. The danger tag shall not be obstructed by the information tag.

C. **Incorrect.** Part 1 is incorrect. Part 2 is incorrect. A Danger tag cannot be placed on a component with an SCT already in place. Plausible if the candidates believe that a Danger Tag can be applied to any Special Condition Tag because the Danger Tag is more important and therefore trumps the Special Condition Tag. IAW OP-MA-109-101, Clearance and Tagging, Section 5.2: Danger Tags:

> 5. A danger tag shall not be applied to a component bearing an SCT.

D. **Incorrect.** Part 1 is incorrect. Part 2 is incorrect. A Danger tag cannot be placed on a component with an SCT already in place. Plausible if the candidate is not familiar with the restrictions of placing a Danger Tag on a component that has a Special Condition Tag already applied. IAW OP-MA-109-101, Clearance and Tagging, Section 5.2: Danger Tags:

> 5. A danger tag shall not be applied to a component bearing an SCT.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.13	
	Importance Rating	4.1	

K/A: Knowledge of tagging and clearance procedures.

Proposed Question: RO Question # 70

Technical Reference(s): OP-MA-109-101, pg 12, Rev 19

Proposed References to be provided to applicants during examination: None

Learning Objective: NOP-DBIG-PCO-2

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	10
	55.43	

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidates to have a knowledge of the TMI-1 Clearance and Tagging Procedure.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to know procedural steps and requirements associated with Clearance and Tagging operations.

What MUST be known:
1. What are the procedural requirements associated with hanging a Danger Tag that has another tag applied already?

71 **ID: 977948** **Points: 1.00**

Plant Conditions:

- Plant is in a Hot Shut Down condition.
- 230kV line 1092 (Middletown sub) is out of service.
- 230kV line 1091 (Middletown sub) is out of service.
- TMI 230kV 8 bus is out of service.
- 1E 4kV bus is powered from the "A" Emergency Diesel generator, EG-Y-1B.

Event:

- Due to major problems with the Grid, the decision has been made to expedite returning the plant to online.

Given the above information, which one of the following statements is correct?

A. The reactor may be made critical, but operation is limited to 30 days.

B. The reactor may not be made critical until a third 230kV line is restored.

C. The reactor may not be made critical until a second 230kV bus is restored.

D. The reactor may be made critical, but both EDGs must be started and run continuously.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.** IAW T.S. 3.7.1:

- The reactor shall not be made critical unless all of the following requirements are satisfied:
    - All engineered safeguards buses, engineered safeguards switchgear, and engineered safeguards load shedding systems are operable.
    - One 7200 volt bus is energized.
    - Two 230 kV lines are in service.
    - One 230 kV bus is in service.
    - Engineered safeguards diesel generators are operable and at least 25,000 gallons of fuel oil are available in the storage tank.
    - Station batteries are charged and in service. Two battery chargers per battery are in service.

Furthermore, the reactor shall not remain critical unless: 3.7.2.b:

- Both 230/4.16 kV unit auxiliary transformers shall be in operation except that within a period not to exceed eight hours in duration from and after the time one Unit 1 auxiliary transformer is made or found inoperable, two diesel generators shall be operable, and one of the operable diesel generator will be started and run continuously until both unit auxiliary transformers are in operation. This mode of operation may continue for a period not exceeding 30 days.

B. **Incorrect.** A third 230kV line is not required per Tech Specs. Plausible if the examinee does not recognize the Tech Spec limit is 2 lines. T.S. 3.7.1:

- The reactor shall not be made critical unless all of the following requirements are satisfied:
    - Two 230 kV lines are in service.

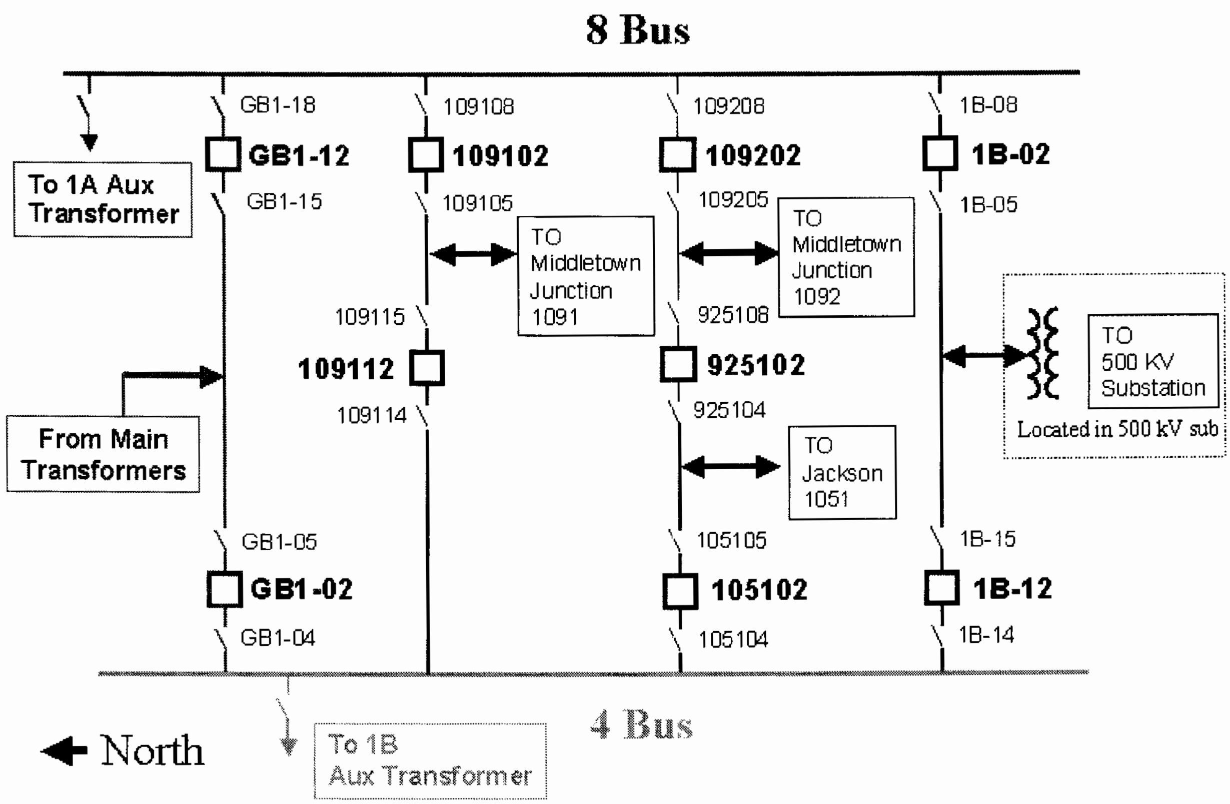
C. **Incorrect.** A second 230kV bus is not required per Tech Specs. Plausible if the examinee does not recognize the Tech Spec limit is 1 bus. T.S. 3.7.1:

- The reactor shall not be made critical unless all of the following requirements are satisfied:
    - One 230 kV bus is in service.

D. **Incorrect.** A second Emergency Diesel Generator is not required per Tech Specs. Plausible if the examinee does not recognize the Tech Spec limit is 1 EDG to be started and run continuously. T.S. 3.7.2:

Both 230/4.16 kV unit auxiliary transformers shall be in operation except that within a period not to exceed eight hours in duration from and after the time one Unit 1 auxiliary transformer is made or found inoperable, two diesel generators shall be operable, and one of the operable diesel generator will be started and run continuously until both unit auxiliary transformers are in operation. This mode of operation may continue for a period not exceeding 30 days.

230 KV Substation



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.22	
	Importance Rating	4.0	

K/A: Knowledge of limiting conditions for operations and safety limits.

Proposed Question: RO Question # 71

Technical Reference(s): Tech Spec 3.7, pg 3-42, Amendment 278

Proposed References to be provided to applicants during examination: None

Learning Objective: 701-GLO-14

Question Source:	Bank #	IR-701-GLO-14-Q01
	Modified Bank #	
	New	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	10
	55.43	

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidates to have a knowledge of limiting conditions for operations associated with the Unit 1 Electrical Power System.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze and interpret plant conditions and determine whether they meet Tech Spec requirements and actions that are required when conditions do not meet Tech Specs.

What MUST be known:
1. What are the available offsite power supplies and lines for TMI?
2. Under what circumstances may the reactor be made critical?
3. What Tech Specs are applicable once the reactor has been made critical?

72 **ID: 977946** **Points: 1.00**

Plant Conditions (T = 1600):

- The plant is operating at 100% power.
- A Reactor Building purge has commenced.

Sequence of Events:

- T = 1700:
  - The Reactor Building purge is secured due to the Chemistry Supervisor questioning the calculations used.
- T = 2030:
  - The Chemistry Supervisor agrees that all calculations are correct and the Reactor Building purge is ready to recommence.

GIven the above information and IAW 6610-ADM-4250.12, Releasing Radioactive Gaseous Effluents, which ONE of the following statements is correct with regards to the Reactor Building purge?

A. The same release permit may be used.

B. No release permit is needed for a Reactor Building purge.

C. A new release permit needs to be generated with a new release number assigned.

D. A new release permit needs to be generated but the same release number can be used.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.** IAW 6610-ADM-4250.12, Releasing Radioactive Gaseous Effluents - Reactor Building Purges, Section 4.1.C

- A Reactor Building purge may be stopped then restarted within 4 hours using the same release permit. If the purge is secured for more than 4 hours, or if conditions change that may warrant resampling as determined by Radiological Protection, a new release permit (with new air samples) will be required.

The conditions given in the stem indicate that the purge was secured for less than 4 hours. Additionally, nothing in the stem indicates that conditions have changed during the time frame.

B. **Incorrect.** Plausible if the candidate is unfamiliar with the requirements for conducting a Reactor Building purge. However, IAW 6610-ADM-4250.12, Releasing Radioactive Gaseous Effluents - Reactor Building Purges. Step 4.1.A:

- Prior to releasing gaseous effluents to the environment from the Reactor Building, a Gaseous Release Permit will be initiated by Operations.

C. **Incorrect.** Plausible if the candidate is either not familiar with the time requirement to be allowed to use the same permit or incorrectly performs the required math to recognize that the Purge was secured for less than 4 hours. IAW 6610-ADM-4250.12, Releasing Radioactive Gaseous Effluents - Reactor Building Purges, Section 4.1.C

- A Reactor Building purge may be stopped then restarted within 4 hours using the same release permit. If the purge is secured for more than 4 hours, or if conditions change that may warrant resampling as determined by Radiological Protection, a new release permit (with new air samples) will be required.

D. **Incorrect.** Plausible if the candidate is either not familiar with the time requirement to be allowed to use the same permit or incorrectly performs the required math to recognize that the Purge was secured for less than 4 hours, and believes that the same number may be used since it's the same purge.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.11	
	Importance Rating	3.8	

K/A: Ability to control radiation releases.

Proposed Question: RO Question # 72

Technical Reference(s): 6610-ADM-4250.12, pg 3, Rev 16

Proposed References to be provided to applicants during examination: None

Learning Objective: NOP-DBIG-APCO-1

Question Source: Bank # IR-RPT-APCO-1-Q12

Modified Bank #

New

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	13
	55.43	

(13) Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Comments:
The KA is matched because the question requires the candidates to recognize the requirements prior to reestablishing a radiation release in the form of a Reactor Building purge to ensure that the release is performed in a controlled manner.

The question is at the Comprehension/Analysis level because the candidate is required to perform a math problem and, based on the results of the math problem, the candidate must recall procedural steps and requirements associated with Reactor Building Purge.

What MUST be known:
1. What is the time that has elapsed between the suspension and recommencement of the RB purge?
2. Under what circumstances may the same permit be used to recommence an RB purge?

73 **ID: 977944** **Points: 1.00**

Plant Conditions:

- The reactor is operating at 60% power.

Event:

- A 4# ESAS has occurred.
- The CRS orders a determination of whether it is a steam leak or RCS leak.

Given the above information, Reactor Building Atmospheric Monitor, RM-A-2, ___(1)___ be used to determine whether the pressure increase is from the RCS because ___(2)___.

A. (1) can
(2) it is NOT isolated from Containment

B. (1) can
(2) wetting of charcoal and paper filters will NOT block flow

C. (1) can NOT
(2) it is isolated from Containment

D. (1) can NOT
(2) wetting of charcoal and paper filters will block flow

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 is incorrect. 4# ESAS isolates monitor via CM-V-1,2,3,4. This isolates RM-A-2. Plausible if the candidate believes that, although RM-A-2 may isolate on a 1600# or a 500# RCS pressure signal, it does not isolate on a 4# ES signal. IAW 1105-3, Safeguards Actuation System:

B. 4 PSIG R.B. Pressure Actuation:
- Containment Air Sample:
  - A Train: CM-V-1 and CM-V-3
  - B Train: CM-V-2 and CM-V-4

Part 2 is incorrect. Plausible if the candidate does not recognize that RM-A-2 will isolate on a 4# ES signal. IAW TQ-TM-104-661-C001, Radiation Monitoring System, Section III.C.2.f.7.i) and j):

- RM-A2's pump will automatically de-energize if either CM-V-1 or CM-V-2 close.
- Installed to prevent RM-A-2 pump discharge header pressure increase, which can lead to loss of filter gasket integrity and possible loss of Containment Atmosphere into the Intermediate Building.

B. **Incorrect.**
Part 1 is incorrect. 4# ESAS isolates monitor via CM-V-1,2,3,4. This isolates RM-A-2. Plausible if the candidate believes that, although RM-A-2 may isolate on a 1600# or a 500# RCS pressure signal, it does not isolate on a 4# ES signal. IAW 1105-3, Safeguards Actuation System:

B. 4 PSIG R.B. Pressure Actuation:
- Containment Air Sample:
  - A Train: CM-V-1 and CM-V-3
  - B Train: CM-V-2 and CM-V-4

Part 2 is incorrect. RM-A-2 does have charcoal and paper filters. Plausible if the candidate recognizes that they would saturate if the detector was not isolated.

C. **Correct.**
Part 1 is correct. 4# ESAS isolates monitor via CM-V-1,2,3,4. This isolates RM-A-2. IAW 1105-3, Safeguards Actuation System:

B. 4 PSIG R.B. Pressure Actuation:
- Containment Air Sample:
  - A Train: CM-V-1 and CM-V-3
  - B Train: CM-V-2 and CM-V-4

Part 2 is correct. IAW TQ-TM-104-661-C001, Radiation Monitoring System:

- There is an interlock which will automatically de-energize RM-A-2's pump when either CM-V-1 or CM-V-2 close or if both CM-V-1 and CM-V-2 close.
- In addition, RM-A-2 pump will not start if CMV- 1 and CM-V-2 are not both open.
- h) This will prevent a RM-A-2 pump discharge header pressure increase, which could lead to a loss of filter gasket integrity and possible containment atmosphere leakage into the Intermediate Building.

D. **Incorrect.**
Part 1 is correct. 4# ESAS isolates monitor via CM-V-1,2,3,4. This isolates RM-A-2. IAW 1105-3, Safeguards Actuation System:

B. 4 PSIG R.B. Pressure Actuation:
- Containment Air Sample:
  - A Train: CM-V-1 and CM-V-3
  - B Train: CM-V-2 and CM-V-4

Part 2 is incorrect. RM-A-2 does have charcoal and paper filters. Plausible if the candidate recognizes that they would saturate if the detector was not isolated.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.15	
	Importance Rating	2.9	

K/A: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: RO Question # 73

Technical Reference(s): 1105-3, pg 14,15, Rev 051
TQ-TM-104-661-C001, pg 27, Rev 007

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-2

Question Source: Bank # IR-661-GLO-2-Q05
Modified Bank #
New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 11
55.43

(11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of interlocks and operation associated with fixed radiation monitor, RM-A-2.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to have system knowledge on ESAS affected equipment and components.

What MUST be known:
1. What is the response of RM-A-2 upon a 4# ES signal?
2. Under what circumstances may RM-A-2 be used to determine whether a leak in the Reactor Building is from a steam leak or an RCS leak?

74 **ID: 977940** **Points: 1.00**

Which one of the following identifies a group of Post Accident Monitoring Instruments that are required to be OPERABLE by Technical Specification Table 3.5-3, Post Accident Monitoring Instrumentation?

A. Containment Pressure,
PORV Position Monitor, and
Steam Generator Pressure.

B. Containment Pressure,
PORV Position Monitor, and
RCS Cold Leg Temperature.

C. Containment Pressure,
Steam Generator Pressure, and
RCS Cold Leg Temperature.

D. PORV Position Monitor,
Steam Generator Pressure, and
RCS Cold Leg Temperature.

Answer: C

Answer Explanation

Explanation (Optional):

A. **Incorrect**. PORV Position Monitor is an Accident Monitoring Instrument required per Table 3.5.2 of Tech Spec 3.5.5 and not part of the Post Accident Monitoring Instruments in Table 3.5.3. Plausible if candidate confuses the tables of Tech Spec 3.5.5. IAW Tech Spec Table 3.5-2, Accident Monitoring Instruments:

TABLE 3.5-2: ACCIDENT MONITORING INSTRUMENTS
- Saturation Margin Monitor
- Safety Valve Differential Pressure Monitor
- PORV Position Monitor
- Emergency Feedwater Flow
- Pressurizer Level
- Backup Incore Thermocouple Display Channel

B. **Incorrect.** PORV Position Monitor is an Accident Monitoring Instrument required per Table 3.5.2 of Tech Spec 3.5.5 and not part of the Post Accident Monitoring Instruments in Table 3.5.3. Plausible if candidate confuses the tables of Tech Spec 3.5.5. IAW Tech Spec Table 3.5-2, Accident Monitoring Instruments:

TABLE 3.5-2: ACCIDENT MONITORING INSTRUMENTS
- Saturation Margin Monitor
- Safety Valve Differential Pressure Monitor
- PORV Position Monitor
- Emergency Feedwater Flow
- Pressurizer Level
- Backup Incore Thermocouple Display Channel

C. **Correct.** All instruments are listed in Table 3.5.3 of Tech Spec 3.5.5. IAW Tech Spec Table 3.5-3, Post Accident Monitoring Instrumentation:

TABLE 3.5-3: POST ACCIDENT MONITORING INSTRUMENTATION

- High Range Noble Gas Effluent
  - Condenser Vacuum Pump Exhaust (RM-A5-Hi)
  - Condenser Vacuum Pump Exhaust (RM-G25)
  - Auxiliary and Fuel Handling Building Exhaust (RM-A8-Hi)
  - Reactor Building Purge Exhaust (RM-A9-Hi)
  - Reactor Building Purge Exhaust (RM-G24)
  - Main Steam Lines Radiation (RM-G26/RM-G27)
- Containment High Range Radiation (RM-G22/G-23)
- Containment Pressure
- Containment Water Level
  - Containment Flood (LT-806/807)
  - Containment Sump (LT-804/805)
- Wide Range Neutron Flux
- Reactor Coolant System Cold Leg Water Temperature (TE-959, 961; TI-959A, 961A)
- Reactor Coolant System Hot Leg Water Temperature (TE-958, 960; TI-958A, 960A)
- Reactor Coolant System Pressure (PT-949, 963; PI-949A, 963)
- Steam Generator Pressure (PT-950, 951, 1180, 1184; PI-950A, 951A, 1180, 1184)
- Condensate Storage Tank Water Level (LT-1060, 1061, 1062, 1063; LI-1060, 1061, 1062, 1063)

D. **Incorrect.** PORV Position Monitor is an Accident Monitoring Instrument required per Table 3.5.2 of Tech Spec 3.5.5 and not part of the Post Accident Monitoring Instruments in Table 3.5.3. Plausible if candidate confuses the tables of Tech Spec 3.5.5. IAW Tech Spec Table 3.5-2, Accident Monitoring Instruments:

TABLE 3.5-2: ACCIDENT MONITORING INSTRUMENTS

- Saturation Margin Monitor
- Safety Valve Differential Pressure Monitor
- PORV Position Monitor
- Emergency Feedwater Flow
- Pressurizer Level
- Backup Incore Thermocouple Display Channel

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.3	
	Importance Rating	3.7	

K/A: Ability to identify post-accident instrumentation.

Proposed Question: RO Question # 74

Technical Reference(s): Tech Spec 3.5.5, pg 3-40d, Rev 278

Proposed References to be provided to applicants during examination: None

Learning Objective: 624-GLO-14

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidates to identify post-accident instrumentation from a list given.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall required instrumentation from a list.

What MUST be known:
1. Which Instruments are identified as Post Accident Monitoring Instruments IAW Tech Specs?

75 **ID: 977936** **Points: 1.00**

Plant Conditions:

- The Reactor is operating at 100% power.

Event:

- An event occurs that causes a Reactor Trip and a Loss of Subcooling Margin.
- All HPI flow transmitters are unavailable.
- The CRS enters OP-TM-EOP-002, Loss of 25 °F Subcooling Margin, and reaches Step 4.1, which reads:
  - "IAAT ADEQUATE HPI exists, then GO TO Step 3.10".

Given the above information, which one of the following meets the facility criteria for "HPI flow is ADEQUATE" IAW OS-24, Conduct of Operations During Abnormal and Emergency Events?

A. Pressurizer level is rising.

B. All components in HPI Train "B" are in their ES positions.

C. HPI flow ≥ 200 gpm is verified from a calculation of the rate of BWST level lowering.

D. Indicated HPI flow exceeds the flow on Attachment 7.4 of OP-TM-211-901, Emergency Injection.

Answer: B

Answer Explanation

Explanation (Optional):

A. **Incorrect.** Not IAW OS-24, Conduct of Operations During Abnormal and Emergency Events. Under most normal circumstances, this would indicate an increase in inventory. Therefore the choice is plausible if the candidate assumes that a rising Pressurizer level is indicative of adequate HPI.

B. **Correct**. Definition meets the OS-24, Conduct of Operations During Abnormal and Emergency Events definition IAW 3.1:

> 3.0: DEFINITIONS: When a defined term is used in EOP/AOP procedures, the term is capitalized to indicate that a specific definition is described below.
>
> > 3.1 ADEQUATE HPI: HPI flow is "adequate" when flow exceeds the flow assumed in the ECCS analysis. This condition may be confirmed by Indicated HPI flow exceeding flow on Attachment 7.4 in OP-TM-211-901. If HPI flow transmitters are unavailable, adequate flow can be assured if all components of one HPI train are in their ES position.

Step 4.1 of OP-TM-EOP-002 had the term "Adequate HPI" capitalized, the definition in OS-24 applies:

> 4.1 IAAT ADEQUATE HPI exists, then GO TO Step 3.10.

The candidate will have to recognize from the conditions given that all HPI flow transmitters are unavailable in the question stem, and therefore the last part of the definition applies.

C. **Incorrect.** Not IAW OS-24, Conduct of Operations During Abnormal and Emergency Events. Adequate HPI flow is greater than (but never equal to) 200 gpm according to OP-TM-211-901 Attachment 7.4. Adequate HPI. Plausible if the candidate believes that this Attachment is to be used and that 200 gpm is a included in the possible values for adequate HPI.

D. **Incorrect.** Although it is a definition of Adequate HPI IAW OS-24, Conduct of Operations During Abnormal and Emergency Events, it is not applicable due to all HPI flow transmitters being unavailable. Plausible if the candidate does not recognize that all HPI flow transmitters are unavailable, as stated in the question stem.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.17	
	Importance Rating	3.9	

K/A: Knowledge of EOP terms and definitions.

Proposed Question: RO Question # 75

Technical Reference(s): OS-24, pg 3, Rev 024

OP-TM-EOP-002, pg 7, Rev 009

Proposed References to be provided to applicants during examination: None

Learning Objective:	EOP-024-PCO-1		
Question Source:	Bank #		
	Modified Bank #		
	New	X	
Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
The KA is matched because the question requires the candidates to know the definition of a term found within an EOP.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall the definition of a term from one procedure that is applicable to an EOP.

What MUST be known:
1. What is the definition of "Adequate HPI" when HPI flow meters are not available?

1 **ID: 984302** **Points: 1.00**

Plant Conditions (Time = 0 seconds):

- A power reduction is in progress due to a small tube leak on the "B" OTSG.
- The Reactor is operating at 90% power and lowering at 1%/minute.
- The plant had been operating at 100% power for the previous 21 days.

Sequence of Events:

- Time = 20 seconds:
    - A reactor trip occurs.
    - During the Reactor Trip Immediate Manual Actions, the Subcooling Margin indicators on Panel Center lowered to 22°F and then started to recover, as witnessed by a shift trainee.
- Time = 90 seconds:
    - OP-TM-EOP-001, Reactor Trip, Immediate Manual Actions are the **ONLY** actions that have been completed.
    - The shift trainee informs the CRS of the Subcooling Margin data that was witnessed at Time = 20 seconds.
    - OP-TM-EOP-001 VSSV's are entered.
    - A symptom check is performed on the following Parametric Data:

Parameter	Value	Trend
OTSG 1A Startup Level	28 inches	Lowering
OTSG 1B Startup Level	55 inches	Rising
OTSG 1A Pressure	800 psig	Lowering
OTSG 1B Pressure	950 psig	Lowering
Loop A T-Cold	525°F	Lowering
Loop B T-Cold	536°F	Lowering
Loop A T-Ave	526°F	Lowering
Loop B T-Ave	537°F	Lowering
RCS Pressure	1690 psig	Lowering
Pressurizer Level	83 inches	Lowering
RB Pressure	+0.1 psig	Rising

Based on the above information, which one of the following is the **FIRST** action that the CRS will direct the crew to perform?

A. Perform Guide 8, RCS Pressure Control, IAW OP-TM-EOP-001, Reactor Trip.

B. Perform Guide 8, RCS Pressure Control, IAW OP-TM-EOP-005, OTSG Tube Leakage.

C. Perform Rule 1, Loss of SCM, IAW OP-TM-EOP-002, Loss of 25 Degree F Subcooling Margin.

D. Perform Rule 3, Excessive Heat Transfer, IAW OP-TM-EOP-003, Excessive Primary to Secondary Heat Transfer.

Answer: C

Answer Explanation

A. Incorrect. If no condition existed that would require routing to another procedure, then the candidate would continue in the Vital System Status Verification (VSSV) section of OP-TM-EOP-001. IAW OP-TM-EOP-001, Reactor Trip, Step 3.11:

- Initiate Guide 8, "RCS Pressure Control".

Plausible if the candidate does not recognize a symptom or if the candidate believes that initiating Guides comes before the routing steps in the VSSV's of OP-TM-EOP-001. See the correct answer explanation for the priority order of routing.

B. Incorrect. OP-TM-EOP-005 is a potential procedure that is entered from step 3.1 of the Vital System Status Verification (VSSV) section of OP-TM-EOP-001. However, OP-TM-EOP-005 is the lowest of the procedures in priority. IAW OP-TM-EOP-005, OTSG Tube Leakage, Step 3.33:

- Minimize SCM IAW Guide 8, "RCS Pressure Control".

Plausible if the candidate does not recognize a higher priority symptom. See the correct answer explanation for the priority order of routing.

C. Correct. A Loss of Subcooling Margin has occurred. Although Subcooling Margin has been restored, entry into OP-TM-EOP-002 is still required. Also, although it has been greater than a minute and the RCP's will not be secured, Rule 1 is still performed. Even though other symptoms exist, a Loss of Subcooling Margin is the highest priority.

IAW OS-24:

4.1.5 Performing Parallel Procedures

A. Any other procedure actions should be interrupted to perform Reactor Trip Immediate Manual Actions and the initial Symptom Check.

B. Once Immediate Manual Actions and the initial symptom check have been accomplished, the Control Room Supervisor determines the sequence of action between parallel procedures. The CRS selects the action most significant to overall event mitigation. The CRS bases this decision on the following mitigation priorities:

(1) Protect public health and safety
(2) Protect site personnel safety
(3) Protect plant equipment

IAW OP-TM-EOP-0011, Reactor Trip Basis Document, Step 3.1:

- Symptom checks are applicable whenever the reactor is shutdown. A formal symptom check should be performed whenever a significant change is plant conditions occurs. Successful mitigation of abnormal transients and the design of the EOP network is predicated on prompt recognition and response to the "symptoms" (i.e. Loss of SCM, Excessive Heat Transfer, Lack of Heat Transfer or OTSG Tube Leakage). If a symptom of a core cooling is identified then further actions in the EOP are directed by the associated EOP.

- The symptoms are reviewed in priority order. When a symptom is identified, action is initiated and the remainder of the symptom check is not immediately relevant, but is re-initiated following mitigation of the first symptom. This step is equivalent to GEOG III.A NOTE at beginning of VSSV.

IAW OP-TM-EOP-001, Step 3.1:

- IAAT a symptom exists, then GO TO the symptom response procedure using the following priority:
    1. EOP-002, "Loss of 25 °F Subcooling Margin",
    2. EOP-003, "Excessive Primary to Secondary Heat Transfer",
    3. EOP-004, "Lack of Primary to Secondary Heat Transfer",
    4. EOP-005, "OTSG Tube Leakage".

D. Incorrect. An Excessive Primary to Secondary Heat Transfer does exist, but it is not, however, the highest priority identified in the Symptom Check. IAW OP-TM-EOP-003, Excessive Primary to Secondary Heat Transfer, Step 2.1:

- Perform Rule 3, XHT

Plausible if the candidate does not recognize that the Loss of Subcooling Margin procedure must be entered even though the condition no longer exists. Furthermore, OP-TM-EOP-002, Loss of Subcooling Margin, will route to OP-TM-EOP-003, Excessive Primary to Secondary Heat Transfer, when the appropriate steps are performed to deal with the Loss of Subcooling Margin. See the correct answer explanation for the priority order of routing.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	E02	EA2.1	
	Importance Rating			4.0

K/A: Ability to determine and interpret the following as they apply to the (Vital System Status Verification): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: SRO Question # 76

Technical Reference(s): OP-TM-EOP-001, pg 5, Rev 12
OP-TM-EOP-0011, pg 9, Rev 5
OS-24, pg 8, Rev 24

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP003-PCO-2

Question Source: Bank #

Modified Bank # IS-EOP003-PCO-2-Q01

New

Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43	5	

Comments:
The KA is matched because the candidates must determine and interpret facility conditions and select the appropriate procedure via performance of VSSV's during an emergency operating situation

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze the plant conditions to determine priority actions, understand procedural requirements when symptom limits are met for only a brief time period, understand time limits for actions when sub-cooling margin is lost, and which actions and procedures are required to mitigate conditions.

The question is at the SRO level because the candidates must provide knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 7).

Original Question: IS-EOP003-PCO-2-Q01

SRO ONLY
Plant conditions:

- Reactor trip from full power due to low RCS pressure.
- All 4 RCPs are operating.
- Parametric Data:

Parameter	Value	Trend
OTSG 1A Startup Level	28 inches	Lowering
OTSG 1B Startup Level	55 inches	Lowering
OTSG 1A Pressure	800 psig	Lowering
OTSG 1B Pressure	950 psig	Lowering
Loop A T-Cold	525°F	Lowering
Loop B T-Cold	536°F	Lowering
Loop A T-Ave	526°F	Lowering
Loop B T-Ave	537°F	Lowering
RCS Pressure	1690 psig	Lowering
Pressurizer Level	73 inches	Lowering
RB Pressure	+0.1 psig	Steady

Based on these conditions, identify the OP-TM-EOP-010 procedural guidance required to be implemented to mitigate the transient.

A. OP-TM-424-901, Emergency Feedwater.

B. Rule 3, Excessive Heat Transfer.

C. Rule 2, HPI/LPI Throttling.

D. Rule 1, Loss of SCM.

Answer: B

2 **ID: 984304** **Points: 1.00**

Key:

- FW-V-5A = "A" Main Feedwater Block Valve.
- FW-V-16A = "A" Feedwater Startup Control Valve.
- FW-V-17A = "A" Main Feedwater Control Valve.
- FW-V-92A = "A" OTSG Startup FW Block Valve.

Plant Conditions:

- Entire "B" Train of HSPS in Defeat.

Event:

- Reactor Trip occurs due to a Loss of both Main Feedwater Pumps.
- Emergency Feedwater has **not** actuated.
- "A" OTSG pressure is 590 psig.
- "A" OTSG level is 15 inches in the Startup Range.

Given the above information, HSPS will close ___(1)___ in order to ___(2)___ IAW Tech Spec bases.

A. (1) FW-V-5A & 92A, ONLY
(2) assure that the EFW system will actuate and control at the appropriate OTSG level without operator action

B. (1) FW-V-5A & 92A, ONLY
(2) maintain appropriate RCS cooling following a loss of OTSG integrity and to minimize the energy released to the Reactor Building atmosphere

C. (1) FW-V-5A,16A, 17A, & 92A
(2) assure that the EFW system will actuate and control at the appropriate OTSG level without operator action

D. (1) FW-V-5A,16A, 17A, & 92A
(2) maintain appropriate RCS cooling following a loss of OTSG integrity and to minimize the energy released to the Reactor Building atmosphere

Answer: B

Answer Explanation

A. Incorrect.
Part 1 is correct. "A" OTSG is below the pressure setpoint for HSPS LO-LO pressure isolation. Each OTSG has an "A" train and a "B" train of HSPS associated with it. With the "B" train defeated, only the block valves will close. IAW TQ-TM-104-644-C001, Slide 65:

- OTSG Train Valves

	OTSG A	OTSG B
TRAIN A	BLOCK	REGULATING
TRAIN B	REGULATING	BLOCK

Part 2 is incorrect. None of the requirement to initiate EFW have been met and therefore is not the reason for the given response. PLausible since the basis is within the same section and is associated with the OTSG's and HSPS. IAW Tech Spec 3.5.1 Basis document:

- Automatic initiation of EFW is provided on Loss of all Reactor Coolant Pumps, Loss of both Main Feedwater Pumps, low OTSG level, and high Reactor Building Pressure. High Reactor Building pressure would be indicative of a Loss of Coolant Accident, Main Steam Line, or Feedwater Line Break inside the Reactor Building. Operability of these instruments is required in order to assure that the EFW system will actuate and control at the appropriate OTSG level without operator action for those events where timely initiation of EFW is required.

B. Correct.
Part 1 is correct. "A" OTSG is below the pressure setpoint for HSPS LO-LO pressure isolation. Each OTSG has an "A" train and a "B" train of HSPS associated with it. With the "B" train defeated, only the block valves will close. IAW TQ-TM-104-644-C001, Slide 65:

- OTSG Train Valves

	OTSG A	OTSG B
TRAIN A	BLOCK	REGULATING
TRAIN B	REGULATING	BLOCK

Part 2 is correct. IAW Tech Spec 3.5.1 Basis document:

- Automatic isolation of Main Feedwater is provided on low OTSG pressure in order to maintain appropriate RCS cooling (minimize cooling) following a loss of OTSG integrity and minimize the energy released to the Reactor Building atmosphere.

C. Incorrect.
Part 1 is incorrect. "A" OTSG is below the pressure setpoint for HSPS LO-LO pressure isolation. Each OTSG has an "A" train and a "B" train of HSPS associated with it. With the "B" train defeated, only the block valves will close. Plausible if the candidate does not recognize that "B" train HSPS affects the "A" OTSG.

Part 2 is incorrect. None of the requirement to initiate EFW have been met and therefore is not the reason for the given response. PLausible since the basis is within the same section and is associated with the OTSG's and HSPS. IAW Tech Spec 3.5.1 Basis document:

- Automatic initiation of EFW is provided on Loss of all Reactor Coolant Pumps, Loss of both Main Feedwater Pumps, low OTSG level, and high Reactor Building Pressure. High Reactor Building pressure would be indicative of a Loss of Coolant Accident, Main Steam Line, or Feedwater Line Break inside the Reactor Building. Operability of these instruments is required in order to assure that the EFW system will actuate and control at the appropriate OTSG level without operator action for those events where timely initiation of EFW is required.

D. Incorrect.
Part 1 is incorrect. "A" OTSG is below the pressure setpoint for HSPS LO-LO pressure isolation. Each OTSG has an "A" train and a "B" train of HSPS associated with it. With the "B" train defeated, only the block valves will close. Plausible if the candidate does not recognize that "B" train HSPS affects the "A" OTSG.

Part 2 is correct. IAW Tech Spec 3.5.1 Basis document:

- Automatic isolation of Main Feedwater is provided on low OTSG pressure in order to maintain appropriate RCS cooling (minimize cooling) following a loss of OTSG integrity and minimize the energy released to the Reactor Building atmosphere.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	054	AA2.05
	Importance Rating		3.7

K/A: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Status of MFW pumps, regulating and stop valves.

Proposed Question: SRO Question # 77

Technical Reference(s): TQ-TM-104-644-C001, Slide 65, Rev 001
Tech Spec 3.5.1, Pg 3-28, Rev 273

Proposed References to be provided to applicants during examination: None

Learning Objective: 644-GLO-14

Question Source:		
	Bank #	
	Modified Bank #	
	New	X

Question History:	Last NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:
The KA is matched because the candidates must determine and interpret the Status of Main Feedwater regulating and stop valves when given a Loss of Main Feedwater (MFW).

The question is at the Comprehension/Analysis cognitive level because the candidate is required to understand the setpoints and affected components of HSPS, interpret plant conditions to determine affected components, and knowledge of automatic HSPS Tech Spec bases.

The question is at the SRO level because the candidates must provide knowledge of Tech Spec bases that are required to analyze TS required actions and terminology. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

3 ID: 984387 Points: 1.00

Plant Conditions:

- The Reactor is operating at 100% power.
- "B" and "C" Makeup Pumps, MU-P-1B and MU-P-1C, are ES selected.
- "A" Makeup Pump is Out of Service (OOS).

Event:

- NR-V-18, Nuclear River to MDCT Valve, has gone full CLOSED.

Given the above information, Nuclear Service River Water flow will be ___(1)___ than normal to the Intermediate Closed Cooling Water Coolers, and the basis that a T.S 3.01 reactor shutdown is required is due to the ___(2)___ following a Loss-of-Coolant Accident (LOCA).

A. (1) lower
(2) Makeup System not being able to supply any emergency coolant

B. (1) lower
(2) Nuclear Services Closed Cooling Water System not being able to provide sufficient cooling

C. (1) higher
(2) Makeup System not being able to supply any emergency coolant

D. (1) higher
(2) Nuclear Services Closed Cooling Water System not being able to provide sufficient cooling

Answer: B

Answer Explanation

A. Incorrect.
Part 1 is correct. Since NR-V-18 is the discharge valve for the NSRW System, it will stop most NSRW flow when closed. Therefore, NSRW flow will lower to the ICCW and the NSCCW heat exchangers. IAW TQ-TM-104-531-C001, Primary Cooling Systems:

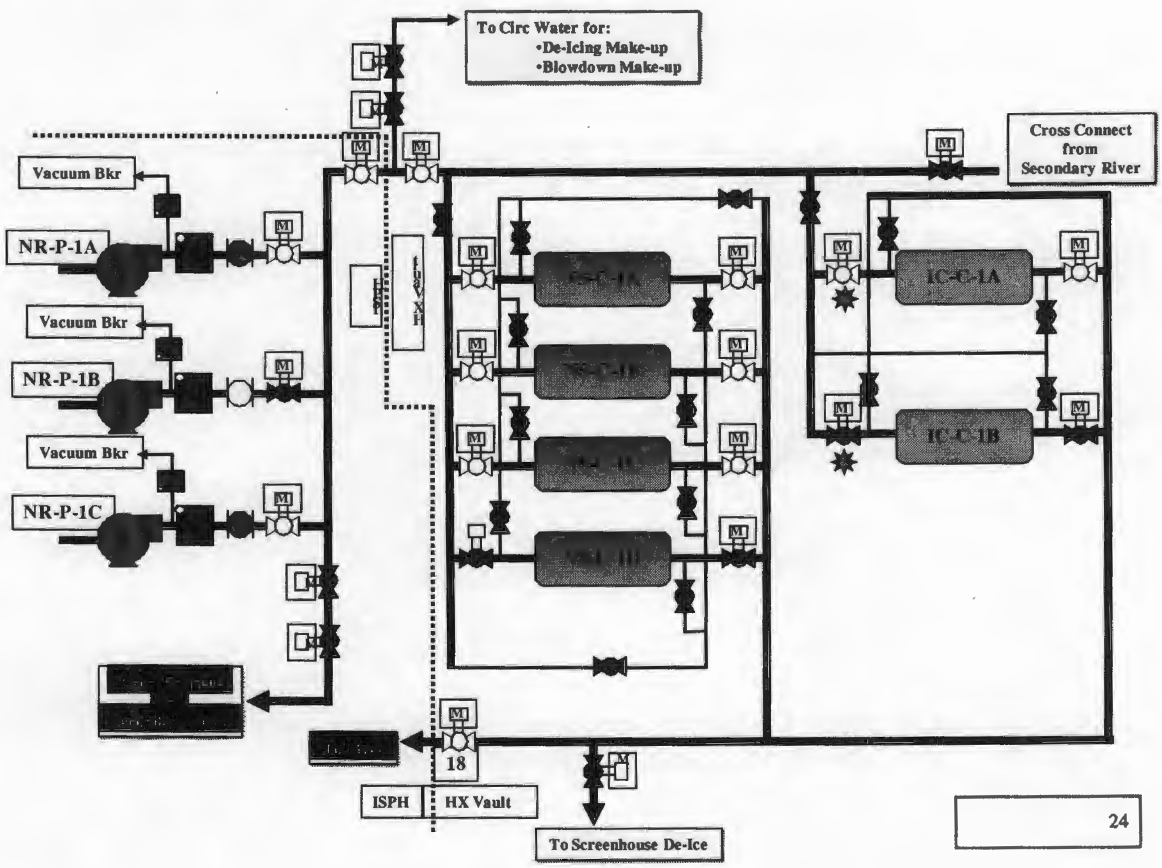
Part 2 is incorrect.
Although the "B" Makeup Pump is cooled by the Nuclear Services Closed Cooling Water System (and therefore affected by NR-V-18 going closed), it is providing RCP seal injection and therefore, the "C" Makeup Pump is aligned to be cooled by the Decay Heat Removal System. One Makeup Pump is sufficient to supply HPI during a LOCA. Plausible if the candidate is not familiar with the Tech Spec Bases for Makeup (HPI) Pumps. IAW TS 3.3.1 Bases:

- The requirements of Specification 3.3.1 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two engineered safeguards makeup pumps, two decay heat removal pumps and two decay heat removal coolers (along with their respective cooling water systems components) are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both CFTs are required because a single CFT has insufficient inventory to reflood the core for hot and cold line breaks (Reference 1).

Additionally, IAW TQ-TM-104-211-C001, Makeup System:

- Cooling water supplies
- Makeup Pump 1A/C coolers
- Normal: Decay Heat Closed Cooling Water (DHCCW)
- Makeup pump 1A cooled by 1A Decay Heat Closed Cooling Water
- Makeup pump 1C cooled by 1B Decay Heat Closed Cooling Water
- Back up: Nuclear Services Closed Cooling Water (NSCCW)
- Makeup Pump1B coolers
- Nuclear Services Closed Cooling Water (NSCCW)

B. Correct.
Part 1 is correct. Since NR-V-18 is the discharge valve for the NSRW System, it will stop most NSRW flow when closed. Therefore, NSRW flow will lower to the ICCW and the NSCCW heat exchangers. IAW TQ-TM-104-531-C001, Primary Cooling Systems:



Part 2 is correct. Tech Specs require that two Nuclear Services Closed Cooling Water Pumps be Operable. If the cooling supply to those pumps is not functioning correctly, the pumps may not be considered Inoperable. IAW Tech Spec 3.3.1.4 and the associated Bases:

3.3.1.4 Cooling Water Systems - Specification 3.0.1 applies.
a. Two nuclear service closed cycle cooling water pumps must be OPERABLE.
b. Two nuclear service river water pumps must be OPERABLE.

- Bases:
  - Two nuclear service river water pumps and two nuclear service closed cycle cooling pumps are required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant.

Additionally, IAW Tech Spec 1.3:

1.3 OPERABLE
- A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

C. Incorrect.
Part 1 is incorrect. Since NR-V-18 is the discharge valve for the NSRW System, it will stop most NSRW flow when closed. Therefore, NSRW flow will lower to the ICCW and the NSCCW heat exchangers. Plausible if the candidate believes that NR-V-18 is a parallel path to the Intermediate Closed Cooling Water Coolers. If this is the train of thought, then the logic would be that if one of three paths closes off, the flow through the Intermediate Closed Cooling Water Coolers will rise.

Part 2 is incorrect.
Although the "B" Makeup Pump is cooled by the Nuclear Services Closed Cooling Water System (and therefore affected by NR-V-18 going closed), it is providing RCP seal injection and therefore, the "C" Makeup Pump is aligned to be cooled by the Decay Heat Removal System. One Makeup Pump is sufficient to supply HPI during a LOCA. Plausible if the candidate is not familiar with the Tech Spec Bases for Makeup (HPI) Pumps. IAW TS 3.3.1 Bases:

- The requirements of Specification 3.3.1 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two engineered safeguards makeup pumps, two decay heat removal pumps and two decay heat removal coolers (along with their respective cooling water systems components) are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both CFTs are required because a single CFT has insufficient inventory to reflood the core for hot and cold line breaks (Reference 1).

Additionally, IAW TQ-TM-104-211-C001, Makeup System:

- Cooling water supplies
- Makeup Pump 1A/C coolers
- Normal: Decay Heat Closed Cooling Water (DHCCW)
- Makeup pump 1A cooled by 1A Decay Heat Closed Cooling Water
- Makeup pump 1C cooled by 1B Decay Heat Closed Cooling Water
- Back up: Nuclear Services Closed Cooling Water (NSCCW)
- Makeup Pump1B coolers
- Nuclear Services Closed Cooling Water (NSCCW)

D. Incorrect.
Part 1 is incorrect. Since NR-V-18 is the discharge valve for the NSRW System, it will stop most NSRW flow when closed. Therefore, NSRW flow will lower to the ICCW and the NSCCW heat exchangers. Plausible if the candidate believes that NR-V-18 is a parallel path to the Intermediate Closed Cooling Water Coolers. If this is the train of thought, then the logic would be that if one of three paths closes off, the flow through the Intermediate Closed Cooling Water Coolers will rise.

Part 2 is correct, but for the wrong reason. Plausible choice with the wrong answer because the Intermediate Closed Cooling Water Coolers are in parallel with the Nuclear Services Closed Cooling Water Coolers. If the candidate believes in Part (1) that NR-V-18 will raise flow through the Intermediate Closed Cooling Water Coolers, then the parallel path through the Nuclear Services Closed Cooling Water Coolers would lower. Tech Specs require that two Nuclear Services Closed Cooling Water Pumps be Operable. If the cooling supply to those pumps is not functioning correctly, the pumps may not be considered Inoperable. IAW Tech Spec 3.3.1.4 and the associated Bases:

3.3.1.4 Cooling Water Systems - Specification 3.0.1 applies.
- a. Two nuclear service closed cycle cooling water pumps must be OPERABLE.
- b. Two nuclear service river water pumps must be OPERABLE.

- Bases:
  - Two nuclear service river water pumps and two nuclear service closed cycle cooling pumps are required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant.

Additionally, IAW Tech Spec 1.3:

1.3 OPERABLE
- A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	062	AA2.05
	Importance Rating		2.5

K/A: Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The normal values for SWS-header flow rate and the flow rates to the components cooled by the SWS.

Proposed Question: SRO Question # 78

Technical Reference(s):
TQ-TM-104-531-C001, Sld 24, Rev 007
T.S. 3.3, pg 3-22, Rev 278, pg 3-24, Rev ECR TM 09-00160
T.S 1.0, pg 1-2, Rev 175

Proposed References to be provided to applicants during examination: None

Learning Objective: 531-GLO-14

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:
The KA is matched because the candidate must interpret and compare SWS-header flow rate and the flow rates to the components cooled by the SWS upon a Loss of Nuclear Service Water to the normal values for SWS-header flow rate and the flow rates to the components cooled by the SWS.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to understand the Nuclear Services River Water System (NSRWS) flowpath and interpret what the outcome will be to components cooled by the NSRWS when given an abnormal event.

The question is at the SRO level because the candidates must provide knowledge of Tech Spec bases that are required to analyze TS required actions and terminology. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

4 **ID: 984388** **Points: 1.00**

Plant Conditions:

- The reactor is operating at 100% power.
- "A" Emergency Feedwater Pump, EF-P-2A, is Out of Service for repairs, expected to Return to Service in 8 hours.

Sequence of Events:

- A Large Break Loss of Coolant Accident (LOCA) has occurred.
- RCP's have been secured.
- 4# ES fails to operate automatically **and** manually.
- 1600# and 500# ESAS have failed to actuate and have **not** been manually initiated.
- "B" Emergency Feedwater Pump, EF-P-2B, trips and cannot be restarted.
- OP-TM-EOP-006, LOCA Cooldown, Section 4.0, Inadequate RCS Cooldown has been entered.
- Reactor Building pressure has peaked at 26 psig and is steady.
- Currently:
    - Auxiliary Steam is **not** available.
    - Emergency Feedwater Pump, EF-P-1, is the only operating EFW pump.
    - Incore Thermocouple Temperatures indicate 376°F and steady.
    - "A" and "B" OTSG Pressures indicate 170 psig each and steady.
    - OP-TM-EOP-006, Step 4.3 and the associated CAUTION statement have been reached.
        - Step 4.3 reads "REDUCE OTSG Pressure so that secondary Tsat is 40 to 60°F lower than incore thermocouple temperature".

Given the above information, the ARO will reduce OTSG pressure to be no lower than ___(1)___ psig, and the URO will manually initiate Emergency Core Cooling (ESAS) to prevent ___(2)___ from occurring.

A. 1) 85 psig
2) peak clad temperature exceeding 2200°F

B. 1) 85 psig
2) excessive Iodine concentrations in the RB atmosphere

C. 1) 150 psig
2) peak clad temperature exceeding 2200°F

D. 1) 150 psig
2) excessive Iodine concentrations in the RB atmosphere

Answer: C

Answer Explanation

A. Incorrect.
Part 1 is incorrect. Although 85 psig would achieve 40-60 degrees lower than incore thermocouple temperature, there is a Caution prior to the step that states:

- If EF-P-2A, EF-P-2B, and auxiliary steam to EF-P-1 are unavailable, then to prevent loss of EFW, do not lower steam pressure below 150 psig.

According to OP-TM-EOP-0061, LOCA Cooldown Basis Document, the caution statement alerts against possible loss of the steam driven EFW pump if the OTSGs are the selected steam supply. The given information shows that the OTSG's are the selected steam supply to EF-P-1 since Auxiliary Steam is not available. Furthermore, EF-P-1 is the only available Emergency Feedwater Pump and losing it would lead to a loss of Emergency Feedwater.

Part 2 is correct. IAW the Basis for Tech Spec 3.3:

- In the event that the need for emergency core cooling should occur, operation of one Makeup Pump, one Decay Heat Removal Pump, and both Core Flood Tanks will protect the core. In the event of a Reactor Coolant System Rupture, their operation will limit the peak clad temperature to less than 2200F and the metal-water reaction to that representing less than 1 percent of the clad.

B. Incorrect.
Part 1 is incorrect. Although 85 psig would achieve 40-60 degrees lower than incore thermocouple temperature, there is a Caution prior to the step that states:

- If EF-P-2A, EF-P-2B, and auxiliary steam to EF-P-1 are unavailable, then to prevent loss of EFW, do not lower steam pressure below 150 psig.

According to OP-TM-EOP-0061, LOCA Cooldown Basis Document, the caution statement alerts against possible loss of the steam driven EFW pump if the OTSGs are the selected steam supply. The given information shows that the OTSG's are the selected steam supply to EF-P-1 since Auxiliary Steam is not available. Furthermore, EF-P-1 is the only available Emergency Feedwater Pump and losing it would lead to a loss of Emergency Feedwater.

Part 2 is incorrect. According to T.S. 3.3, The iodine removal function of the reactor building spray system requires one spray pump and TSP in baskets located in the Reactor Building Basement. The Building Spray System becomes functional only after a Block 4 permissive signal from either a 1600# RCS, 500# RCS, or 4# RB signal AND a 30# RB signal is received. Since the maximum RB pressure attained was 26#, a 30# RB signal was not reached. Therefore, the Building Spray Pumps did not receive a start signal. Plausible if the candidate does not recognize this fact.

C. Correct.
Part 1 is correct. Although 85 psig would achieve 40-60 degrees lower than incore thermocouple temperature, IAW OP-TM-EOP-006, there is a Caution prior to step 4.3 that states:

- If EF-P-2A, EF-P-2B, and auxiliary steam to EF-P-1 are unavailable, then to prevent loss of EFW, do not lower steam pressure below 150 psig.

Part 2 is correct. IAW the Basis for Tech Spec 3.3:

- In the event that the need for emergency core cooling should occur, operation of one Makeup Pump, one Decay Heat Removal Pump, and both Core Flood Tanks will protect the core. In the event of a Reactor Coolant System Rupture, their operation will limit the peak clad temperature to less than 2200F and the metal-water reaction to that representing less than 1 percent of the clad.

D. Incorrect.
Part 1 is correct. Although 85 psig would achieve 40-60 degrees lower than incore thermocouple temperature, IAW OP-TM-EOP-006, there is a Caution prior to step 4.3 that states:

- If EF-P-2A, EF-P-2B, and auxiliary steam to EF-P-1 are unavailable, then to prevent loss of EFW, do not lower steam pressure below 150 psig.

Part 2 is incorrect. According to T.S. 3.3, The iodine removal function of the reactor building spray system requires one spray pump and TSP in baskets located in the Reactor Building Basement. The Building Spray System becomes functional only after a Block 4 permissive signal from either a 1600# RCS, 500# RCS, or 4# RB signal AND a 30# RB signal is received. Since the maximum RB pressure attained was 26#, a 30# RB signal was not reached. Therefore, the Building Spray Pumps did not receive a start signal. Plausible if the candidate does not recognize this fact.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	011	2.4.20	
	Importance Rating			4.3

K/A: Large Break LOCA: Knowledge of the operational implications of EOP warnings, cautions, and notes.

Proposed Question: SRO Question # 79

Technical Reference(s): OP-TM-EOP-006, pg 19, Rev 11
T.S. 3.3, pg 3-22, Rev 278, pg 3-24, Rev ECR TM 09-00160

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP006-PCO-5

Question Source:
Bank #
Modified Bank #
New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.41
55.43 2

Comments:

The KA is matched because the candidate must analyze a given set of conditions and then make a decision while incorporating the operational implications of an EOP cautions during a Large Break LOCA.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze and interpret plant conditions, calculate required OTSG pressure based on procedural guidance and using steam tables, knowledge of plant procedure direction during abnormal plant lineups, and tech spec bases for operation of ESAS during LOCA conditions.

The question is at the SRO level because the candidates must provide knowledge of Tech Spec bases that are required to analyze TS required actions and terminology. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

5 **ID: 984389** **Points: 1.00**

Plant Conditions:

- The Reactor is operating at 100% power.

Event:

- A large Steam Line Rupture occurs on the "A" OTSG.
- The reactor has tripped on low RCS pressure.
- The Main Turbine has **not** been tripped.
- Tave is 525F and lowering.
- "A" OTSG Pressure is 500 psig and lowering.
- Subcooling Margin is 26F and steady.
- All 4 RCPs are operating.

Based on these conditions, the CRS will order isolation of the "A" OTSG IAW OP-TM-EOP-010, Rule 3, Excessive Heat Transfer, ___(1)___, and the basis for isolating the OTSG IAW Rule 3 is to prevent the potential for __(2)___.

A. (1) immediately
(2) a loss of pressurizer level which can result in a saturated RCS

B. (1) immediately
(2) loop voids which can interrupt primary to secondary heat transfer on the "A" OTSG

C. (1) after the Main Turbine Stop Valves have been closed
(2) a loss of pressurizer level which can result in a saturated RCS

D. (1) after the Main Turbine Stop Valves have been closed
(2) loop voids which can interrupt primary to secondary heat transfer on the "A" OTSG

Answer: C

Answer Explanation

A. Incorrect.
Part 1 is incorrect. Before the OTSG may be isolated, the Main Turbine must be tripped, which is described in Rule 3 Entry statement:

- IAAT Primary to Secondary Heat Transfer is excessive and the turbine is tripped, then perform the following.

Additionally, The Basis document for Rule 3, which is OP-TM-EOP-0101, Emergency Procedure Rules Guides and Graphs Basis Document,states:

- Overcooling can result from failed open TBVs, ADVs, MSSVs, or steam line breaks. Excessive Main or Emergency Feedwater may also cause overcooling. A Main Turbine failure to trip also result in a rapid overcooling. Rule 3, however, should only be implemented after the Turbine stop valves have closed.

Part 2 is correct. IAW OP-TM-EOP-010, Rule 3 Basis document:

- Use of the Rule is governed by a shutdown reactor and Turbine and recognition of cooldown beyond normal post trip parameters, and is followed up as the first step in the Emergency Operating Procedure (EOP) for Excessive Heat Transfer (XHT) to ensure correct and timely completion of action(s) in order to prevent an extended overcooling event.
- An extended overcooling may result in:
  - OTSG damage from tube vibration due to high steam flow and excessive feedwater rates,
  - Thermal shock to the OTSG,
  - Excessive OTSG tube tensile loads due to exceeding tube-to-shell differential temperature limits,
  - – Loss of pressurizer level which can result in a saturated RCS,
  - – Reactor Coolant system pressurized thermal shock (PTS)

B. Incorrect.
Part 1 is incorrect. Before the OTSG may be isolated, the Main Turbine must be tripped, which is described in Rule 3 Entry statement:

- IAAT Primary to Secondary Heat Transfer is excessive and the turbine is tripped, then perform the following.

Additionally, The Basis document for Rule 3, which is OP-TM-EOP-0101, Emergency Procedure Rules Guides and Graphs Basis Document,states:

- Overcooling can result from failed open TBVs, ADVs, MSSVs, or steam line breaks. Excessive Main or Emergency Feedwater may also cause overcooling. A Main Turbine failure to trip also result in a rapid overcooling. Rule 3, however, should only be implemented after the Turbine stop valves have closed.

Part 2 is incorrect. This choice would be the correct reason for a Loss of Subcooling Margin occurred (Rule 1). Additionally, OP-TM-EOP-010, Rule 3 states that if the the leak is in the Reactor Building or the Intermediate Building, then perform Phase 1 and 2 isolation. If the leak was in the Turbine Building, then only Phase 1 isolation would occur, leaving EFW available to the OTSG and the atmospheric dump valve would be available to lift automatically at 1026 psig. Therefore, it would be possible to have primary to secondary heat transfer exist in the affected OTSG. Plausible if the candidate believes that only Phase 1 isolation has occurred on the "A" OTSG and/or if the candidate confuses the basis for Rule 3 with that for Rule 1.

C. Correct.
Part 1 is correct. Before the OTSG may be isolated, the Main Turbine must be tripped, which is described in Rule 3 Entry statement:

- IAAT Primary to Secondary Heat Transfer is excessive and the turbine is tripped, then perform the following.

Additionally, The Basis document for Rule 3, which is OP-TM-EOP-0101, Emergency Procedure Rules Guides and Graphs Basis Document,states:

- Overcooling can result from failed open TBVs, ADVs, MSSVs, or steam line breaks. Excessive Main or Emergency Feedwater may also cause overcooling. A Main Turbine failure to trip also result in a rapid overcooling. Rule 3, however, should only be implemented after the Turbine stop valves have closed.

Part 2 is correct. IAW OP-TM-EOP-010, Rule 3 Basis document:

- Use of the Rule is governed by a shutdown reactor and Turbine and recognition of cooldown beyond normal post trip parameters, and is followed up as the first step in the Emergency Operating Procedure (EOP) for Excessive Heat Transfer (XHT) to ensure correct and timely completion of action(s) in order to prevent an extended overcooling event.
- An extended overcooling may result in:
  - – OTSG damage from tube vibration due to high steam flow and excessive feedwater rates,
  - – Thermal shock to the OTSG,
  - – Excessive OTSG tube tensile loads due to exceeding tube-to-shell differential temperature limits,
  - – Loss of pressurizer level which can result in a saturated RCS,
  - – Reactor Coolant system pressurized thermal shock (PTS)

D. Incorrect.
Part 1 is correct. Before the OTSG may be isolated, the Main Turbine must be tripped, which is described in Rule 3 Entry statement:

- IAAT Primary to Secondary Heat Transfer is excessive and the turbine is tripped, then perform the following.

Additionally, The Basis document for Rule 3, which is OP-TM-EOP-0101, Emergency Procedure Rules Guides and Graphs Basis Document,states:

- Overcooling can result from failed open TBVs, ADVs, MSSVs, or steam line breaks. Excessive Main or Emergency Feedwater may also cause overcooling. A Main Turbine failure to trip also result in a rapid overcooling. Rule 3, however, should only be implemented after the Turbine stop valves have closed.

Part 2 is incorrect. This choice would be the correct reason for a Loss of Subcooling Margin occurred (Rule 1). Additionally, OP-TM-EOP-010, Rule 3 states that if the the leak is in the Reactor Building or the Intermediate Building, then perform Phase 1 and 2 isolation. If the leak was in the Turbine Building, then only Phase 1 isolation would occur, leaving EFW available to the OTSG and the atmospheric dump valve would be available to lift automatically at 1026 psig. Therefore, it would be possible to have primary to secondary heat transfer exist in the affected OTSG. Plausible if the candidate believes that only Phase 1 isolation has occurred on the "A" OTSG and/or if the candidate confuses the basis for Rule 3 with that for Rule 1.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E05	2.2.44
	Importance Rating		4.3

K/A: Excessive Heat Transfer: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question: SRO Question # 80

Technical Reference(s): OP-TM-EOP-010, pg 6, Rev 016
OP-TM-EOP-0101, pg 17, Rev 008

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP003-PCO-1

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 5

Comments:
The KA is matched because the candidate must interpret control room indications to verify the status of the Control Rod System, and then must understand how operator actions and directives affect plant and system conditions during an Excessive Heat Transfer event.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to understand procedural requirements for determining excessive heat transfer and basis for which the OTSG is isolated due to a steam rupture.

The question is at the SRO level because the candidates must provide knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 7).

6 **ID: 984390** **Points: 1.00**

Plant Conditions:

- The Reactor is operating at 100% power.
- The "C" 4kV bus is OOS due to an electrical fault on the bus.

Sequence of Events:

- Time = 1530:
  - A Loss of Offsite Power (LOOP) has occurred.
  - OP-TM-AOP-020, Loss of Station Power, and OP-TM-EOP-001, Reactor Trip, have been entered.
  - "A" and "B" Emergency Diesel Generators, EG-Y-1A and EG-Y-1B, are operating as expected.

- Time = 1900:
  - The Plant is stable in Hot Shutdown.
  - EG-Y-1A has tripped and cannot be restarted. The trip has been verified not to be a common mode failure.
  - The ARO attempts to place the SBO Diesel, EG-Y-4, on the ES Bus but it will **not** start.
  - A report is received that offsite power will not be restored for another 4 hours.

Given the above information and IAW OP-TM-AOP-0201, Loss of Station Power Basis Document, the reason the SBO Diesel will not start is because ___(1)___ and the appropriate action to be taken by the CRS is ___(2)___.

A. (1) the SBO batteries do not have enough power to allow diesel start and load
(2) to maintain Hot Shutdown conditions

B. (1) the SBO batteries do not have enough power to allow diesel start and load
(2) place the plant in Cold Shutdown within 24 hours of EG-Y-1A tripping

C. (1) FS-V-646, SBO D/G Cooling Water Supply Isolation Valve, has been blocked closed and cannot open in Automatic
(2) to maintain Hot Shutdown conditions

D. (1) FS-V-646, SBO D/G Cooling Water Supply Isolation Valve, has been blocked closed and cannot open in Automatic
(2) place the plant in Cold Shutdown within 24 hours of EG-Y-1A tripping

Answer: A

Answer Explanation

A. Correct.

Part 1 is correct. IAW OP-TM-AOP-0201, Loss of Station Power Basis Document, the note prior to Step 3.24:

- This note informs operators that the SBO must be promptly energized to ensure the SBO batteries have enough power to allow diesel start and load. C-1101-864-E420-001 describes the capacity of SBO batteries.

Part 2 is correct. According to Tech Spec 3.7, none of the criteria are met to change the current plant status:

- Both diesel generators shall be operable except that from the date that one of the diesel generators is made or found to be inoperable for any reason, reactor operation is permissible for the succeeding seven days provided that the redundant diesel generator is:
    1. verified to be operable immediately;
    2. within 24 hours, either:
        a. determine the redundant diesel generator is not inoperable due to a common mode failure; or,
        b. test redundant diesel generator in accordance with surveillance requirement 4.6.1.a.
- In the event two diesel generators are inoperable, the unit shall be placed in HOT SHUTDOWN in 12 hours. If one diesel is not operable within an additional 24 hour period the plant shall be placed in COLD SHUTDOWN within an additional 24 hours thereafter.

- With one diesel generator inoperable, in addition to the above, verify that:
    - All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE or follow specifications 3.0.1.

B. Incorrect.
Part 1 is correct. IAW OP-TM-AOP-0201, Loss of Station Power Basis Document, the note prior to Step 3.24:

- This note informs operators that the SBO must be promptly energized to ensure the SBO batteries have enough power to allow diesel start and load. C-1101-864-E420-001 describes the capacity of SBO batteries.

Part 2 is incorrect. Cold Shutdown requirements are met if both Emergency Diesel Generators are inoperable for up to 24 hours after achieving a Hot Shutdown condition. With EG-Y-1B running, it is operable and therefore the criteria is not met. Plausible if the candidate believes that EG-Y-4 counts as one of the 2 inoperable Emergency Diesel Generators.

C. Incorrect.
Part 1 is incorrect. Referencing 2014 OPEX, FS-V-646, SBO D/G Cooling Water Supply Isolation Valve, was blocked closed as a result of an outage lineup that did not restore the valve back to normal configuration. When performing a surveillance run on the SBO after the outage, the SBO started and ran for a few minutes before the Control Room received a high temperature alarm. Investigation found that the valve was blocked closed and therefore could not open automatically. Plausible if the candidate is not familiar with recent OPEX and believes that the SBO will not start as a result of the valve being out of place.

Part 2 is correct. According to Tech Spec 3.7, none of the criteria are met to change the current plant status:

- Both diesel generators shall be operable except that from the date that one of the diesel generators is made or found to be inoperable for any reason, reactor operation is permissible for the succeeding seven days provided that the redundant diesel generator is:
    1. verified to be operable immediately;
    2. within 24 hours, either:
        a. determine the redundant diesel generator is not inoperable due to a common mode failure; or,
        b. test redundant diesel generator in accordance with surveillance requirement 4.6.1.a.

- In the event two diesel generators are inoperable, the unit shall be placed in HOT SHUTDOWN in 12 hours. If one diesel is not operable within an additional 24 hour period the plant shall be placed in COLD SHUTDOWN within an additional 24 hours thereafter.

- With one diesel generator inoperable, in addition to the above, verify that:
  - All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE or follow specifications 3.0.1.

D. Incorrect.
Part 1 is incorrect. Referencing 2014 OPEX, FS-V-646, SBO D/G Cooling Water Supply Isolation Valve, was blocked closed as a result of an outage lineup that did not restore the valve back to normal configuration. When performing a surveillance run on the SBO after the outage, the SBO started and ran for a few minutes before the Control Room received a high temperature alarm. Investigation found that the valve was blocked closed and therefore could not open automatically. Plausible if the candidate is not familiar with recent OPEX and believes that the SBO will not start as a result of the valve being out of place.

Part 2 is incorrect. Cold Shutdown requirements are met if both Emergency Diesel Generators are inoperable for up to 24 hours after achieving a Hot Shutdown condition. With EG-Y-1B running, it is operable and therefore the criteria is not met. Plausible if the candidate believes that EG-Y-4 counts as one of the 2 inoperable Emergency Diesel Generators.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	056	2.4.18
	Importance Rating		4.3

K/A: Loss of Offsite Power: Knowledge of the specific bases for EOPs.

Proposed Question: SRO Question # 81

Technical Reference(s): OP-TM-AOP-0201, pg 4, Rev 0
T.S. 3.7, pg 3-43, Rev 278

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP020-PCO-3

Question Source: Bank #

Modified Bank #

New X

Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	2	

Comments:
The KA is matched because, although a Loss of Offsite Power is listed as an Abnormal Operating Procedure, it is as severe as an Emergency Operating Procedure and is identified by the NRC as a Major Event in scenarios, on par with Excessive Heat Transfer, Lack of Heat Transfer and OTSG Tube Leak. So treating the procedure as an EOP, the candidate must have a knowledge of the specific bases for a Loss of Offsite Power.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to understand procedural actions, including notes, and the Tech Spec actions for electrical malfunctions in the plant.

The question is at the SRO level because the candidates must demonstrate application of required Tech Spec actions in accordance with rules of application requirements. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

7 **ID: 984405** **Points: 1.00**

Plant Conditions:

- The plant is currently in an outage.
- Refueling operations are in progress.
  - Fuel is currently being moved in the Fuel Transfer Canal.
  - Refueling is the outage priority.
- Fuel Transfer Canal Level is 344.5 feet and steady.
- Several large loads are ready to be moved with the Polar Crane.

Event:

- Fuel Transfer Canal Level has dropped 2 feet.
- OP-TM-AOP-060, Leakage While on Decay Heat Removal, has been entered.

Given the above information, the CRS will direct that fuel handling be suspended IAW ___(1)___ because the accident analysis assumption for the ___(2)___.

A. (1) 1505-3, Fuel Handling Problems
(2) release to the containment atmosphere from the postulated damaged fuel rods located on top of the fuel core seated in the reactor vessel is not met

B. (1) 1505-3, Fuel Handling Problems
(2) dropping of materials or equipment from the Polar Crane into the reactor vessel and possibly damaging the fuel to the extent that any escape of fission products would result

C. (1) 1505-1, Fuel and Component Shuffles
(2) release to the containment atmosphere from the postulated damaged fuel rods located on top of the fuel core seated in the reactor vessel is not met

D. (1) 1505-1, Fuel and Component Shuffles
(2) dropping of materials or equipment from the Polar Crane into the reactor vessel and possibly damaging the fuel to the extent that any escape of fission products would result

Answer: C

Answer Explanation

A. Incorrect.

Part 1 is incorrect. OP-TM-AOP-060, Leakage While on Decay Heat Removal, requirements are met to suspend fuel handling. Plausible if the candidate is not familiar with the Fuel Handling Procedures or steps within OP-TM-AOP-060 and chooses based on procedure title only. OP-TM-AOP-060, Step 3.1:

- If fuel movement is in progress, then NOTIFY the fuel handling SRO to suspend fuel handling IAW 1505-1, "Fuel And Control Component Shuffles".

Part 2 is correct. Tech Spec 3.8:

3.8 FUEL LOADING AND REFUELING
Applicability: Applies to fuel loading and refueling operations.
Objective: To assure that fuel loading and refueling operations are performed in a responsible manner.

3.8.11 During the handling of irradiated fuel in the Reactor Building at least 23 feet of water shall be maintained above the level of the reactor pressure vessel flange, as determined by a shiftly check and a daily verification. If the water level is less than 23 feet above the reactor pressure vessel flange, place the fuel assembly(s) being handled into a safe position, then cease fuel handling until the water level has been restored to 23 feet or greater above the reactor pressure vessel flange.

Additionally, IAW Tech Spec 3.8 Bases:

- The minimum water level specified is the basis for the accident analysis assumption of a decontamination factor of 200 for the release to the containment atmosphere from the postulated damaged fuel rods located on top of the fuel core seated in the reactor vessel.

Drawing 308-946, Reactor Coolant Systems RCS Benchmark Elevations Schematic Layout, shows that the RV Flange is at the 321'0" elevation. 23 feet of water above that level would be at the 344'0" level. The Fuel Transfer Canal and RCS are connected and therefore, the same body of water while refueling is occurring. Thus, if Fuel Transfer Canal level was 344.5' and dropped 2', it would result in a Fuel Transfer Canal and RCS level of 342.5', which would be less than the required 23 feet of water above the RV Flange.

B. Incorrect.
Part 1 is incorrect. OP-TM-AOP-060, Leakage While on Decay Heat Removal, requirements are met to suspend fuel handling. Plausible if the candidate is not familiar with the Fuel Handling Procedures or steps within OP-TM-AOP-060 and chooses based on procedure title only. OP-TM-AOP-060, Step 3.1:

- If fuel movement is in progress, then NOTIFY the fuel handling SRO to suspend fuel handling IAW 1505-1, "Fuel And Control Component Shuffles".

Part 2 is incorrect. Polar Crane operations are not allowed during fuel handling. Therefore that is not the reason why fuel handling must be suspended. Although Tech Spec 3.12 allows Polar Crane operation while fuel is not being moved, this is not the reason to secure the Fuel movement. Plausible if: 1) the candidate is not familiar with the Polar Crane Tech Spec and believes that Polar Crane operations can occur during fuel handling, and 2) assumes that a dropped load from the polar crane will cause more damage when striking fuel due to a more shallow pool of water to slow down the inertia of the load. IAW Tech Spec 3.12:

3.12 REACTOR BUILDING POLAR CRANE

Applicability: Applies to the use of the reactor building polar crane hoists over the steam generator compartments and the fuel transfer canal.
Objective: To identify those conditions for which the operation of the reactor building polar crane hoists are restricted.

3.12.1 The reactor building polar crane hoists shall not be operated over the fuel transfer canal when any fuel assembly is being moved.
3.12.2 During the period when the reactor vessel head is removed and irradiated fuel is in the reactor building and fuel is not being moved, the reactor building polar crane hoist shall be operated over the fuel transfer canal only where necessary and in accordance with approved operating procedures stating the purpose of such use.

C. Correct.
Part 1 is correct. OP-TM-AOP-060, Leakage While on Decay Heat Removal, requirements are met to suspend fuel handling. OP-TM-AOP-060, Step 3.1:

- If fuel movement is in progress, then NOTIFY the fuel handling SRO to suspend fuel handling IAW 1505-1, "Fuel And Control Component Shuffles".

Part 2 is correct. Tech Spec 3.8:

> 3.8 FUEL LOADING AND REFUELING
> Applicability: Applies to fuel loading and refueling operations.
> Objective: To assure that fuel loading and refueling operations are performed in a responsible manner.
>
> 3.8.11 During the handling of irradiated fuel in the Reactor Building at least 23 feet of water shall be maintained above the level of the reactor pressure vessel flange, as determined by a shiftly check and a daily verification. If the water level is less than 23 feet above the reactor pressure vessel flange, place the fuel assembly(s) being handled into a safe position, then cease fuel handling until the water level has been restored to 23 feet or greater above the reactor pressure vessel flange.

Additionally, IAW Tech Spec 3.8 Bases:

- The minimum water level specified is the basis for the accident analysis assumption of a decontamination factor of 200 for the release to the containment atmosphere from the postulated damaged fuel rods located on top of the fuel core seated in the reactor vessel.

Drawing 308-946, Reactor Coolant Systems RCS Benchmark Elevations Schematic Layout, shows that the RV Flange is at the 321'0" elevation. 23 feet of water above that level would be at the 344'0" level. The Fuel Transfer Canal and RCS are connected and therefore, the same body of water while refueling is occurring. Thus, if Fuel Transfer Canal level was 344.5' and dropped 2', it would result in a Fuel Transfer Canal and RCS level of 342.5', which would be less than the required 23 feet of water above the RV Flange.

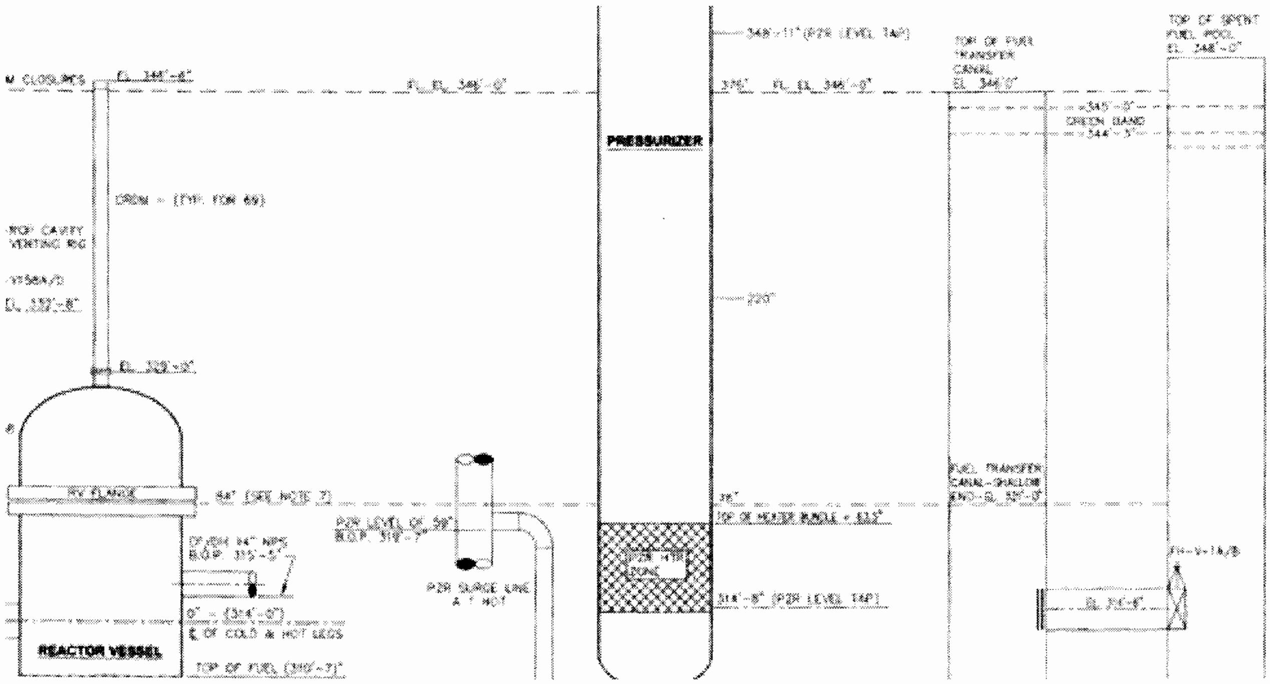
D. Incorrect.
Part 1 is incorrect. OP-TM-AOP-060, Leakage While on Decay Heat Removal, requirements are met to suspend fuel handling. Plausible if the candidate is not familiar with the Fuel Handling Procedures or steps within OP-TM-AOP-060 and chooses based on procedure title only. OP-TM-AOP-060, Step 3.1:

> If fuel movement is in progress, then NOTIFY the fuel handling SRO to suspend fuel handling IAW 1505-1, "Fuel And Control Component Shuffles".

Part 2 is incorrect. Polar Crane operations are not allowed during fuel handling. Therefore that is not the reason why fuel handling must be suspended. Although Tech Spec 3.12 allows Polar Crane operation while fuel is not being moved, this is not the reason to secure the Fuel movement. Plausible if: 1) the candidate is not familiar with the Polar Crane Tech Spec and believes that Polar Crane operations can occur during fuel handling, and 2) assumes that a dropped load from the polar crane will cause more damage when striking fuel due to a more shallow pool of water to slow down the inertia of the load. IAW Tech Spec 3.12:

> 3.12 REACTOR BUILDING POLAR CRANE
>> Applicability: Applies to the use of the reactor building polar crane hoists over the steam generator compartments and the fuel transfer canal.
>> Objective: To identify those conditions for which the operation of the reactor building polar crane hoists are restricted.
>>
>> 3.12.1 The reactor building polar crane hoists shall not be operated over the fuel transfer canal when any fuel assembly is being moved.

3.12.2 During the period when the reactor vessel head is removed and irradiated fuel is in the reactor building and fuel is not being moved, the reactor building polar crane hoist shall be operated over the fuel transfer canal only where necessary and in accordance with approved operating procedures stating the purpose of such use.



Examination Outline Cross-reference:

	RO		SRO
Level	RO		SRO
Tier #	1		
Group #	2		
K/A #	A08	AA2.1	
Importance Rating			4.0

K/A: Ability to determine and interpret the following as they apply to the (Refueling Canal Level Decrease): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: SRO Question # 82

Technical Reference(s):
OP-TM-AOP-060, pg 3, Rev 006
Tech Spec 3.8, pg 3-45, Rev 260
308946, Rev 005

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP060-PCO-3

Question Source:	Bank #		
	Modified Bank #		
	New	X	
Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	2	

Comments:
The KA is matched because, given a Refueling Canal Level Decrease and other facility conditions, the candidate must select the appropriate procedure.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to interpret refueling canal level and then have knowledge of procedural actions and information regarding fuel handling activities.

The question is at the SRO level because the candidates must provide knowledge of Tech Spec bases that are required to analyze TS required actions and terminology. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

8 **ID: 984408** **Points: 1.00**

Plant Conditions:

- The Reactor is operating at 100% power.

Sequence of Events:

- Loss of Offsite Power (LOOP) occurs with an EG-Y-1B start failure.
- OP-TM-EOP-001, Reactor Trip, Immediate Manual Actions are complete.
- One control rod remains stuck out at the 100% withdrawn position.
- 1600 psig ES Actuation has occurred.
- Control room staffing is limited to the minimum required by Technical Specifications.
- Current Conditions:

Parameter	Value	Trend
RCS Pressure	975 psig	Rising
Core exit T/C temperatures	514°F	Rising
Loop A/B cold leg temperatures	500°F	Lowering
Pressurizer level	85 inches	Steady
Total HPI Flow	515 gpm	Steady
RCP Seal Injection Flow	30 gpm	Steady
OTSG pressures	875 psig	Lowering
OTSG levels	25%	Rising
Containment Building Pressure	4.6 psig	Rising

Based on the above conditions, identify the highest priority action to be directed by the CRS.

A. Reduce HPI flow IAW OP-TM-EOP-010 Rule 2, HPI Throttling.

B. Emergency borate IAW OP-TM-EOP-010, Rule 5, Emergency Boration.

C. Minimize subcooled margin IAW OP-TM-EOP-010, Guide 8, RCS Pressure Control.

D. Start the SBO Diesel Generator IAW OP-TM-864-901, SBO Diesel Generator (EG-Y-4) Operations.

Answer: A

Answer Explanation

A. Correct - Reducing HPI flow is the correct answer, since Rule 2, HPI/LPI Throttling, HPI flow limit is exceeded when RCP seal injection flow is considered. HPI pump flows are required to be maintained between 500 and 515 gpm. OS-24, Conduct of Operations During Abnormal; and Emergency Operations, Section 4.1.4.D states that rules are numbered according to priority, and if multiple rule based actions required, the highest priority rule is performed first. These actions are directed by Rule 2, the highest listed Rule applicable to the stem conditions.

B. Incorrect - Initiation of alternate means of RCS boration presents a plausible distracter, since the question stem contains a stuck rod at the 100% withdrawn position and inability to open MU-V-14B (EG-Y-1B start failure). Rule 5, EB, actions for the stuck control rod include initiation of Guide 1, Emergency Boration Backup Methods. This answer is incorrect (not highest priority) since stem conditions also include HPI flow in excess of the 515 gpm/pump HPI flow limit in Rule 2. OS-24, Conduct of Operations During Abnormal; and Emergency Operations, Section 4.1.4.D states that rules are numbered according to priority, and if multiple rule based actions required, the highest priority rule is performed first. Plausible if the candidate is not familiar with the priority of Rules and Guides.

C. Incorrect - Minimizing SCM presents a plausible distracter, since this is a PTS event and Rule 6, PTS, directs the operator to minimize SCM. This answer is not highest priority since stem conditions also include HPI flow in excess of the 515 gpm/pump HPI flow limit in Rule 2. OS-24, Conduct of Operations During Abnormal; and Emergency Operations, Section 4.1.4.D states that rules are numbered according to priority, and if multiple rule based actions required, the highest priority rule is performed first. Plausible if the candidate is not familiar with the priority of Rules and Guides.

D. Incorrect - Start up of the SBO Diesel is a plausible distracter due to EG-Y-1B start failure following loss of offsite power. This action is directed by OP-TM-864-901, SBO Diesel Generator (EG-Y-4) Operations. OS-24 Section 4.3.1.A prioritizes performance ofactions based on safety significance of the mitigation actions. Based on the question stem, EOP-010 Rule 2 is required to be performed to prevent MU-P-1A runout. Therefore EOP-010 performance takes precedence - which makes this an incorrect answer. Plausible if the candidate is not familiar with the priority of Rules and Guides.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	E14	EA2.1
	Importance Rating		4.0

K/A: Ability to determine and interpret the following as they apply to the (EOP Enclosures): Facility conditions and selection of appropriated procedures during abnormal and emergency operations.

Proposed Question: SRO Question # 83

Technical Reference(s): OS-24, pg 7, Rev 024

Proposed References to be provided to applicants during examination: None

Learning Objective: HPI001-PCO-4

Question Source: Bank # IR-HPI001-PCO-4-Q01

Modified Bank #

New

Question History:		Last NRC Exam:	N/A
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43	5	

Comments:
The KA is matched because the candidate must demonstrate the ability to determine and interpret facility conditions and select the appropriate procedure during an emergency operation situation .

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze plant conditions to determine priorites, understand precautions and limitations with the Makeup System, knowledge of EOP Enclosures for emergency boration, RCS pressure control, and HPI throttling, and emergency procedural actions for the SBO Diesel generator.

The question is at the SRO level because the candidates must provide knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 7).

9 **ID: 984103** **Points: 1.00**

Plant Conditions (T= 0900):

- The plant is operating at 100% power.
- No discharges are in progress.
- The setpoints for RM-A-7 are as follows:
  - Alert: 2.0E5 cpm
  - Alarm: 4.0E5 cpm

Sequence of Events:

- T= 1000:
  - An accidental Gaseous waste discharge commences.
  - The following indications are available in the Control Room:

Radiation Monitor	Indication
RM-A-7, Waste Gas Tank Discharge	6.0E7 cpm
RM-A-8 GH, Aux. and Fuel Handling Building Exhaust Duct	4.0E3 cpm
RM-A-14, Fuel Handling (ESF) Vent Radiation Monitor	7.7E-1 μCi/cc

- T= 1010:
  - The following indications are available in the Control Room:

Radiation Monitor	Indication
RM-A-7, Waste Gas Tank Discharge	9.0E7 cpm
RM-A-8 GH, Aux. and Fuel Handling Building Exhaust Duct	3.1E4 cpm
RM-A-14, Fuel Handling (ESF) Vent Radiation Monitor	7.8E-1 μCi/cc

- T= 1020:
  - The following indications are available in the Control Room:

Radiation Monitor	Indication
RM-A-7, Waste Gas Tank Discharge	9.2E7 cpm
RM-A-8 GH, Aux. and Fuel Handling Building Exhaust Duct	3.15E4 cpm
RM-A-14, Fuel Handling (ESF) Vent Radiation Monitor	7.5E0 μCi/cc

Given the above information and assuming a linear rise on all of the radiation monitors, which one of the following identifies the highest EAL whose criteria is met at T=1020?

A. RU1

B. RA1

C. RS1

D. RG1

Answer: B

Answer Explanation

A. Incorrect.
Plausible if the candidate is not familiar with the EAL matrix and recognizes that the values given are greater than those listed for an Unusual Event. The time requirement of greater than 60 minutes is plausible since the Initial Conditions start over 60 minutes from the end time.

B. Correct.
Referring to EP-AA-1009, EXELON NUCLEAR RADIOLOGICAL EMERGENCY PLAN ANNEX FOR THREE MILE ISLAND (TMI) STATION, Page 3-11, The following logic is used for each indication:

- RM-A-7:
- At time 1000, RM-A-7 exceeds threshold #1 for RU1 in value, but not in length of time.
- At time 1010, RM-A-7 exceeds threshold #1 for RA1 and RU1 in value, but not in length of time.
- At time 1020, RM-A-7 still exceeds threshold #1 for RA1 and RU1 in value, but not in length of time. Even if the candidate calculates out the exact time that the threshold was exceeded between 1000 and 1010, it still calculates out to be less than 15 minutes.
- Conclusion is that no call is to be made based on RM-A-7 at time 1020.

- RM-A-8:
- At time 1000, RM-A-8 exceeds threshold #2 for RA1 in value, but not in length of time.
- At time 1010, RM-A-8 exceeds threshold #1 for RS1 and threshold #2 for RA1 in value, but not in length of time.
- At time 1020, RM-A-8 still exceeds threshold #1 for RS1 and threshold #2 for RA1 in value, but only RA1 is met in length of time. Even if the candidate calculates out the exact time that the RS1 threshold was exceeded between 1000 and 1010, it still calculates out to be less than 15 minutes.
- Conclusion is that RA1 is to be made based on RM-A-8 at time 1020.

- RM-A-14:
- At time 1000, RM-A-14 exceeds no threshold for RU1 in value, but not in length of time.
- At time 1010, RM-A-14 exceeds the threshold for RU1 in value, but not in length of time.
- At time 1020, RM-A-14 exceeds the threshold for RU1 in value and in length of time.
- Conclusion is that RU1 is met based on RM-A-14 at time 1020 but the Alert associated with RM-A-8 is a higher EAL and should be declared.

C. Incorrect.
Plausible if the candidate is not familiar with the EAL matrix and believes that the values given are greater than those listed for a Site Area Emergency. At time 1010, RM-A-8 exceeds the threshold for RS1 and RA1 in value, but not in length of time. At time 1020, RM-A-8 still exceeds the threshold for RS1 and RA1 in value, but only RA1 is met in length of time. Even if the candidate calculates out the exact time that the RS1 threshold was exceeded between 1000 and 1010, it still calculates out to be less than 15 minutes.

D. Incorrect.
Plausible if the candidate does not recognize the numbers in Table R1 and confuses the limits at xE4 (Site Area Emergency) with the limits at xE5 (General Emergency).

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	060	2.4.47
Importance Rating		4.2

K/A: Accidental Gaseous-Waste Release: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: SRO Question # 84

Technical Reference(s): EP-AA-1009, pg 3-11, Rev 21

Proposed References to be provided to applicants during examination: EP-AA-1009, page 3-11

Learning Objective: 231-GLO-14

Question Source:
Bank #
Modified Bank #
New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.41
55.43 4

Comments:
The KA is matched because the question requires the candidates to diagnose trends given from Control Room indications associated with an Accidental Gaseous Waste Release.

The question is at the Comprehension/Analysis cognitive level because the candidate must interpret data given, calculate a rise in radiation monitors, and analyze an EAL matrix.

The question is at the SRO level because the candidates must demonstrate analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 6).

10 **ID: 984410** **Points: 1.00**

Plant Conditions:

- The Reactor is operating at 85% power.

Sequence of Events:

- A Fire in the Relay Room has occurred.
- The crew has initiated OP-TM-EOP-020, Cooldown From Outside of Control Room.
    - IMA's are complete and Control Room Evacuation has just commenced.
- Intermediate Cooling Water Pump, IC-P-1A, has tripped.
- Due to an open circuit at the fire location, Intermediate Cooling Water Pump, IC-P-1B, did not automatically start.
- MU-V-32, RCP Seal Injection Control Valve, inlet flow is inadequate.
- 1E 4KV bus was deenergized but is now reenergized IAW OP-TM-EOP-020, Attachment 9, Starting EG-Y-1B and Loading 1E 4160V Bus.

Given the above information, which ONE of the following describes:
(1) The action that is required concerning Intermediate Closed Cooling Water, and
(2) The procedure that directs this action?

A. (1) After ES/UV lockouts are reset, IC-P-1B is required to be started to prevent CRD stator damage.
(2) OP-TM-EOP-020, Cooldown From Outside of Control Room.

B. (1) After ES/UV lockouts are reset, IC-P-1B is required to be started to prevent CRD stator damage.
(2) OP-TM-AOP-032, Loss of Intermediate Closed Cooling Water.

C. (1) After establishing control at the RSD panels, IC-P-1B is required to be shutdown to prevent RCP thermal barrier cooler damage.
(2) OP-TM-EOP-020, Cooldown From Outside of Control Room

D. (1) After establishing control at the RSD panels, IC-P-1B is required to be shutdown to prevent RCP thermal barrier cooler damage.
(2) OP-TM-AOP-032, Loss of Intermediate Closed Cooling Water.

Answer: C

Answer Explanation

A. Incorrect.

Part 1 is incorrect. The reactor is tripped in the event of loss of cooling to the CRD stators in order to prevent stator damage. Loss of CRD stator cooling would not prevent CRD insertion on RPS actuation. Maintaining reactor shutdown is not affected by loss of IC component cooling.

Part 2 is correct. OP-TM-AOP-020 Attachment 10, Step 1.4 addresses the situation from the RSD Panel.

B. Incorrect.
Part 1 is incorrect. The reactor is tripped in the event of loss of cooling to the CRD stators in order to prevent stator damage. Loss of CRD stator cooling would not prevent CRD insertion on RPS actuation. Maintaining reactor shutdown is not affected by loss of IC component cooling.

Part 2 is incorrect. OP-TM-AOP-032 does not address actions from outside of the Control Room.

C. Correct.
Part 1 is correct. EOP-0201 Step 3.2:

- If EG-Y-1B was required to be started IAW Attachment 9 (i.e., the bus was de-energized), then it is assumed that RCP seals may be overheated. Seal return would be isolated, IC-P-1B stopped to prevent thermal barrier cooler damage, and OP-TM-226-901 is initiated.

Part 2 is correct. OP-TM-AOP-020 Attachment 10, Step 1.4 addresses the situation from the RSD Panel.

D. Incorrect.
Part 1 is correct. EOP-0201 Step 3.2:

- If EG-Y-1B was required to be started IAW Attachment 9 (i.e., the bus was de-energized), then it is assumed that RCP seals may be overheated. Seal return would be isolated, IC-P-1B stopped to prevent thermal barrier cooler damage, and OP-TM-226-901 is initiated.

Part 2 is not correct as OP-TM-AOP-032 does not address actions from outside of the Control Room.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	068	2.1.23	
	Importance Rating			4.4

K/A: Control Room Evacuation: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: SRO Question # 85

Technical Reference(s): OP-TM-EOP-020, pg 81, Rev 15A
OP-TM-EOP-0201, pg 18, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-020-PCO-1

Question Source:	Bank #	ILT 10-02 Question #82
	Modified Bank #	
	New	

Question History:	Last NRC Exam:	2012 (TMI 10-02)

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:
The KA is matched because the candidate must demonstrate the ability to perform specific system and integrated plant procedures during a Control Room Evacuation.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze the plant conditions to determine priorities, have knowledge of the ICCW system with regards to RCP seal cooling, operational knowledge of RCP seals and actions to take when all cooling is lost or inadequate cooling exists, and knowledge of applicable normal and abnormal procedures to determine actions to be taken.

The question is at the SRO level because the candidates must provide knowledge of when to implement attachments and appendices, including hot to coordinate these items with procedure steps. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 7).

11 **ID: 984413** **Points: 1.00**

Plant Conditions:

- The Reactor is operating at 100% power.

Sequence of Events:

- ICS is placed in Manual Control IAW OP-TM-621-471, ICS Manual Control.
- The Shutdown Bypass switch in the "A" RPS cabinet is taken to the "Bypass" position by an I&C Technician for maintenance.
- The URO verifies that the plant is stable at 100% power with ICS in Manual Control.
- The Shutdown Bypass switch in the "A" RPS cabinet is taken to the "Normal" position by an I&C Technician.
- A Manual Channel Reset was NOT performed on the "A" RPS cabinet.
- The Shutdown Bypass switch in the "B" RPS cabinet is taken to the "Bypass" position by an I&C Technician for maintenance.
- The URO verifies that the plant is stable at 100% power with ICS in Manual Control.

Given the above information, the CRS will direct the crew to ___(1)___ because a high ___(2)___ condition exists within RPS.

A. (1) trip the Reactor and Main Turbine IAW OP-TM-EOP-001, Reactor Trip
(2) Nuclear Power

B. (1) trip the Reactor and Main Turbine IAW OP-TM-EOP-001, Reactor Trip
(2) Reactor Coolant System Pressure

C. (1) Place the unit in Hot Shutdown within 6 Hours IAW 1102-4, Power Operation and 1102-10, Plant Shutdown
(2) Nuclear Power

D. (1) place the unit in Hot Shutdown within 6 Hours IAW 1102-4, Power Operation and 1102-10, Plant Shutdown
(2) Reactor Coolant System Pressure

Answer: B

Answer Explanation

A. Incorrect.
Part 1 is correct but part 2 is incorrect. IAW T.S. 2.3.1 Bases:

f. Shutdown Bypass - In order to provide for control rod tests, zero power physics testings, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to value ≤ 5.0 percent of rated power during reactor shutdown.
2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

B. Correct.
The candidate must first recognize that when the Shutdown Bypass switch in the "A" RPS cabinet is taken to "Bypass", it will automatically set the RCS High Pressure setpoint to 1720 psig. Since we are at Normal Operating Pressure, a channel trip signal is sent to all of the remaining cabinets. IAW T.S. 2.3.1 Bases:

> f. Shutdown Bypass - In order to provide for control rod tests, zero power physics testings, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:
>
> 1. By administrative control the nuclear overpower trip set point must be reduced to value ≤ 5.0 percent of rated power during reactor shutdown.
> 2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

Then the candidate must recognize that since the Manual Channel Reset Pushbutton was not pressed, the RCS High Pressure signal is still present on all 4 cabinets. When the Shutdown Bypass switch in the "B" RPS cabinet is taken to "Bypass", it will also automatically set the RCS High Pressure setpoint to 1720 psig. Since we are at Normal Operating Pressure, a channel trip signal is sent to all of the remaining cabinets. Since two channels have sent a trip signal, an automatic reactor trip will occur. IAW Tech Spec 3.5.1 bases:

- There are four reactor protection channels. Normal trip logic is two out of four. Minimum required trip logic is one out of two.

Since two channels have sent a trip signal, an automatic reactor trip should have occurred. IAW Tech Spec 3.5.1 bases:

- There are four reactor protection channels. Normal trip logic is two out of four. Minimum required trip logic is one out of two.

Therefore a manual reactor trip should occur IAW OS-24, Conduct of Operations During Abnormal and Emergency Events:

> E. Initiate reactor trip or actuate safety systems if plant conditions degrade or system setpoints are exceeded without automatic actuation. Control Room Supervisor concurrence should be obtained for actuation of protective systems.

C. Incorrect.
Part 1 is incorrect. If the candidate does not recognize that an ATWS has occurred, the candidate will believe that there are two channels that cannot perform their function and that would not meet the requirements for Minimum channels for redundancy. Plausible if the candidate does not recognize that the "A" and/or "B" RPS cabinet has sent a trip signal by believing that the Shutdown Bypass switch has no effect at power.

Part 2 is incorrect. IAW T.S. 2.3.1 Bases:

> f. Shutdown Bypass - In order to provide for control rod tests, zero power physics testings, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:
>
> 1. By administrative control the nuclear overpower trip set point must be reduced to value ≤ 5.0 percent of rated power during reactor shutdown.
> 2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

D. Incorrect.
Part 1 is incorrect. If the candidate does not recognize that an ATWS has occurred, the candidate will believe that there are two channels that cannot perform their function and that would not meet the requirements for Minimum channels for redundancy. Plausible if the candidate does not recognize that the "A" and/or "B" RPS cabinet has sent a trip signal by believing that the Shutdown Bypass switch has no effect at power.

Part 2 is correct. IAW T.S. 2.3.1 Bases:

> f. Shutdown Bypass - In order to provide for control rod tests, zero power physics testings, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:
>
> 1. By administrative control the nuclear overpower trip set point must be reduced to value ≤ 5.0 percent of rated power during reactor shutdown.
> 2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012	A2.03
	Importance Rating		3.7

K/A: Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Incorrect channel bypassing.

Proposed Question: SRO Question # 86

Technical Reference(s): Tech Spec, pg 2-9, Rev 157

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO-14

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	.
	55.43	2

Comments:
The KA is matched because the candidate must predict the impacts of incorrect channel bypassing associated with the Reactor Protection System and the select the correct procedure to correct, control, or mitigate the consequences of the incorrect channel bypassing.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze plant conditions to determine symptoms and priorities, detailed knowledge of the Reactor Protection System Shutdown Bypass features, and knowledge of procedural and Tech Spec actions to mitigate the event.

The question is at the SRO level because the candidates must provide knowledge of Tech Spec bases that are required to analyze TS required actions and terminology. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

12 **ID: 984415** **Points: 1.00**

Plant Conditions:

- A reactor startup is in progress.
- The ECP is for criticality to be achieved at a rod position of 65% withdrawn on Group 5 rods. The + position being 25% withdrawn on group 6 and the - position being 30% withdrawn on group 5.

Sequence of Events:

- T = 0 minutes:
  - Main Feedwater Pump (FW-P-1B) is in service on Aux steam.
  - Once Through Steam Generator (OTSG) pressure is at 890 psig.
  - T ave is at 530°F.
  - Group 5 rods have been withdrawn to 25%.
  - Counts Per Second (CPS) = 240 CPS.
  - Start Up Rate (SUR) = 0.0 DPM.
- T = 2 minutes:
  - CPS = 250.
- T = 3 minutes:
  - "B" OTSG pressure 870 psig.
  - T ave at 528°F.
- T = 4 minutes:
  - CPS = 315.
  - SUR = 0.2 DPM.
- T = 6 minutes:
  - CPS = 790.
  - SUR = 0.4 DPM.
- T = 7 minutes:
  - T ave at 526 F by console digital indication.
  - "A" OTSG pressure at 880 psig.
  - "B" OTSG pressure at 800 psig.
  - MS-V-3A thru F and MS-V-4A/B are all verified closed.
  - CPS = 1125.
  - SUR = 0.6 DPM.
  - Both OTSG levels are 25" on the Startup Range.
  - FW flow is high to the "B" OTSG.
  - RB Pressure is 2.1 psig and rising.
  - RB Temperature is 113F and rising.

Given the above information, which ONE of the following describes:
(1) The event taking place, and
(2) The FIRST action, based on the event, that the CRS will direct IAW procedural guidance?

A. (1) A steam leak.
(2) Trip the Reactor IAW OP-TM-AOP-051, Secondary Side High Energy Leak.

B. (1) A steam leak.
(2) Insert Groups 5 and 4 Control Rods in sequence IAW 1103-8, Approach to Criticality.

C. (1) Excessive feed water condition.
(2) Trip the Reactor IAW OP-TM-AOP-051, Secondary Side High Energy Leak.

D. (1) Excessive feed water condition.
(2) Insert Groups 5 and 4 Control Rods in sequence IAW 1103-8, Approach to Criticality.

Answer: A

Answer Explanation

A. Correct.
Part 1 is correct. The conditions are such that a steam leak is occurring and has caused criticality. No rod movement has occurred but there is a rise in source range counts and Startup Rate. Additionally, Reactor Building pressure and temperatures are rising with OTSG levels remaining constant.

Part 2 is correct. OP-TM-AOP-051 entry criteria is met, and because RB pressure is greater than 2 psig, a reactor trip is warranted. OP-TM-AOP-051, Step 4.1:

- IAAT RB Pressure greater than 2 psig, THEN trip the reactor and Go To EOP-001.

B. Incorrect.
Part 1 is correct. The conditions are such that a steam leak is occurring and has caused criticality. No rod movement has occurred but there is a rise in source range counts and Startup Rate. Additionally, Reactor Building pressure and temperatures are rising with OTSG levels remaining constant.

Part 2 is incorrect. Although 1103-8, Approach to Criticality, states:

3.2.15 WAAT a positive stable SUR exists without rod motion, then
1. Declare the reactor is critical
2. If ROD INDEX is less than the MINIMUM ROD WITHDRAW LIMIT then GO TO Step 3.3 "Missed ECP".

3.3 If the ECP was missed or ECB/ECP becomes INVALID, then perform the following:
NOTE If the reactor is critical on Group 5, then Group 4 will be fully inserted.

3.3.1. INSERT control rods in sequence, until the rod group which was being withdrawn is fully inserted and one additional group is fully inserted.

OP-TM-AOP-051 is also entered and requires a Reactor Trip, which is a "more" conservative decision. Plausible if the candidate does not recognize the need for a Reactor Trip or reverses which is the more conservative decision.

C. Incorrect.
Part 1 is incorrect. An excessive feedwater condition is not supported by the OTSG water levels. Since the OTSG water levels are steady at Low Level Limits, a excessive feedwater condition cannot be occurring. The raised feedwater to the "B" OTSG can be explained by the need to account for the inventory being lost through the steam leak. Plausible if the candidate does not combine the knowledge of the raised feedwater and the OTSG levels.

Part 2 is correct. OP-TM-AOP-051 entry criteria is met, and because RB pressure is greater than 2 psig, a reactor trip is warranted. OP-TM-AOP-051, Step 4.1:

- IAAT RB Pressure greater than 2 psig, THEN trip the reactor and Go To EOP-001.

D. Incorrect.
Part 1 is incorrect. An excessive feedwater condition is not supported by the OTSG water levels. Since the OTSG water levels are steady at Low Level Limits, a excessive feedwater condition cannot be occurring. The raised feedwater to the "B" OTSG can be explained by the need to account for the inventory being lost through the steam leak. Plausible if the candidate does not combine the knowledge of the raised feedwater and the OTSG levels.

Part 2 is incorrect. Although 1103-8, Approach to Criticality, states:

3.2.15 WAAT a positive stable SUR exists without rod motion, then
1. Declare the reactor is critical
2. If ROD INDEX is less than the MINIMUM ROD WITHDRAW LIMIT then GO TO Step 3.3 "Missed ECP".

3.3 If the ECP was missed or ECB/ECP becomes INVALID, then perform the following:
NOTE If the reactor is critical on Group 5, then Group 4 will be fully inserted.

3.3.1. INSERT control rods in sequence, until the rod group which was being withdrawn is fully inserted and one additional group is fully inserted.

OP-TM-AOP-051 is also entered and requires a Reactor Trip, which is a "more" conservative decision. Plausible if the candidate does not recognize the need for a Reactor Trip or reverses which is the more conservative decision.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039	A2.05
	Importance Rating		3.6

K/A: Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Increasing steam demand, its relationship to increases in reactor power.

Proposed Question: SRO Question # 87

Technical Reference(s): OP-TM-AOP-051, pg 9, Rev 0A

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP062-PCO-3

Question Source: Bank #

Modified Bank # SR5C10-03-Q02 (Note changes or attach parent)

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:
The KA is matched because the candidate must predict the impact of increased steam demand from the MRSS, and its relationship to increases in reactor power. Then, based on the prediction, the candidate must decide the correct procedure to use to correct, control, or mitigate the consequences of the increased steam demand.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze and interpret plant conditions and priorities, have knowledge of steam flow affects on primary parameters and their affect on reactivity, and knowledge of procedural actions required for unanticipated criticality.

The question is at the SRO level because the candidates must provide knowledge of administration procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 7). This is not RO level of knowledge for the following reason: The CRS is in both OP-TM-AOP-051 and 1103-8. Within OP-TM-AOP-051, the routing is through Section 1, 2 and into 3. Section 3 routes the CRS to Section 4 if the steam leak is in the Reactor Building. Within Section 4, the IAAT statement lies to trip the reactor if RB Pressure is greater than 2 psig. The required RO knowledge associated with RB Pressure is to trip the Reactor if RB Pressure is greater than 4 psig. Therefore, this is an SRO decision of which action to take.

Original Question: SR5C10-03-Q02

- A reactor startup is in progress.
- The ECP is for criticality to be achieved at a rod position of 65% withdrawn on Group 5 rods.
- The + and - positions for the rods on this ECP are 30% withdrawn on group 5 and 25% withdrawn on group 6.
- The following conditions and indications have been received:

T = 0 minute
- All safety rods are fully withdrawn
- RCS on daisy chain cleanup
- Main Feedwater Pump (FW-P-1B) is in service on Aux steam
- Gland Seal Steam is being supplied from Aux steam
- Main Turbine is on the turning gear
- Once Through Steam Generator (OTSG) pressure is at 890 psig
- T ave is at 530°F
- Group 5 rods have been withdrawn to 25%
- Counts Per Second (CPS) = 240 CPS
- Start Up Rate (SUR) = 0 DPM

T = 2 minutes
- CPS = 250

T = 3 minutes
- "B" OTSG pressure 870 psig
- T ave at 526°F

T = 4 minutes
- CPS = 315
- SUR = .2 DPM

T = 5 minutes
- "A" OTSG pressure at 880 psig
- MS-V-3A thru F indicate closed (verified)

T = 6 minutes
- CPS = 790
- SUR = .4 DPM

T = 7 minutes
- T ave at 520 F by console digital indication
- "A" OTSG pressure at 880 psig
- "B" OTSG pressure at 800 psig
- MS-V-3A thru F indicate closed (verified)
- MS-V-4A/B verified closed
- CPS = 1125
- SUR = .6 DPM
- Both OTSG levels are normal
- FW flow is high to the "B" OTSG

Which of the following describes the event or events taking place?

A. Dilution accident.

B. Excessive feed water condition.

C. Unanticipated criticality from a steam leak.

D. Unanticipated criticality from a feed line leak.

Answer: C

13 **ID: 984420** **Points: 1.00**

Plant Conditions:

- The plant is in Hot Shutdown in preparation for a Refueling Outage.

Sequence of Events:

- "A" and "B" Nuclear River Service Water Pumps, NR-P-1A and NR-P-1B, have tripped.
- Intermediate Closed Cooling Water (ICCW) and Nuclear Service Closed Cooling Water (NSCCW) temperatures are rising.
- Secondary River (SR) and Nuclear River Water (NR) systems are in the process of being cross-connected IAW OP-TM-541-901, Cross-tie Secondary River to Supply Nuclear River.
    - NR-V-2 is OPEN.
    - NR-V-7 is OPEN.
    - NR-V-6 is OPEN.
- ICCW and NSCCW temperatures are slowly lowering.
- Secondary River Discharge Pressure Indicator SR-PI-134 indicates 20 psig and slowly lowering.

Given the above information, which one of the following identifies:
(1) The procedural action to be directed by the CRS, and
(2) The applicable Technical Specification LCO Time Clock, if any, related to the Nuclear River System?

A. (1) Start an additional Secondary River pump.
(2) No Tech Spec LCO Time Clock is in effect.

B. (1) Start an additional Secondary River pump.
(2) Restore the Nuclear Services River Water System to Operable within 72 hours.

C. (1) Throttle SR-V-2, SR Pumps Discharge Header Isolation Valve.
(2) No Tech Spec LCO Time Clock is in effect.

D. (1) Throttle SR-V-2, SR Pumps Discharge Header Isolation Valve.
(2) Restore the Nuclear Services River Water System to Operable within 72 hours.

Answer: C

Answer Explanation

A. Incorrect.

Part 1 is incorrect. The given conditions state that Nuclear River and Secondary River have been cross connected. NR-V-6 being OPEN indicates that Step 4.10 of OP-TM-541-901, Cross-Tie Secondary River to Supply Nuclear River, has been completed. Therefore step 4.6 has already been completed, which states:

4.6 ENSURE the third Secondary River pump is operating.

Plausible if the candidate is not familiar with the procedure logic and does not recognize that a third SR pump is started prior to opening the cross connect valves.

Part 2 is correct. IAW Tech Spec 3.3.1:

3.3.1 The reactor shall not be made critical unless the following conditions are met:
3.3.1.4 Cooling Water Systems - Specification 3.0.1 applies.
b. Two nuclear service river water pumps must be OPERABLE.

Additionally, Tech Spec 3.0.1 states:

3.0.1 When a Limiting Condition for Operation is not met, except as provided in action called for in the specification, within one hour action shall be initiated to place the unit in a condition in which the specification does not apply by placing it, as applicable, in:
1. At least HOT STANDBY within the next 6 hours.
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the action requirements, the action may be taken in accordance with the time limits of the specification as measured from the time of failure to meet the Limiting Condition for Operation. Applicability of these requirements is stated in the individual specifications.

Specification 3.0.1 is not applicable in COLD SHUTDOWN OR REFUELING SHUTDOWN.

Since the reactor is not critical, the Limiting Condition for Operation is not met, and therefore the TS action statement is not in effect.

B. Incorrect.
Part 1 is incorrect. The given conditions state that Nuclear River and Secondary River have been cross connected. NR-V-6 being OPEN indicates that Step 4.10 of OP-TM-541-901, Cross-Tie Secondary River to Supply Nuclear River, has been completed. Therefore step 4.6 has already been completed, which states:

4.6 ENSURE the third Secondary River pump is operating.

Plausible if the candidate is not familiar with the procedure logic and does not recognize that a third SR pump is started prior to opening the cross connect valves.

Part 2 is incorrect. Tech Spec 3.3.2 is applicable while critical. T.S. 3.3.2 states:

3.3.2 Maintenance or testing shall be allowed during reactor operation on any component(s) in the makeup and purification, decay heat, RB emergency cooling water, RB spray, BWST level instrumentation, or cooling water systems which will not remove more than one train of each system from service. Components shall not be removed from service so that the affected system train is inoperable for more than 72 consecutive hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.

Plausible if the candidate does not recognize that either: a) Tech Spec 3.3.2 applies only while critical, or b) believes that T.S. 3.0.1 is incorrect and chooses this by default.

C. Correct.
Part 1 is correct. The given conditions state that Nuclear River and Secondary River have been cross connected. NR-V-6 being OPEN indicates that Step 4.10 of OP-TM-541-901, Cross-Tie Secondary River to Supply Nuclear River, has been completed. The next step, step 4.11 states:

> 4.11 THROTTLE SR-V-2 to maintain Secondary River discharge pressure (SR-PI-134) above 21 psig.

Part 2 is correct. IAW Tech Spec 3.3.1:

> 3.3.1 The reactor shall not be made critical unless the following conditions are met:
> 3.3.1.4 Cooling Water Systems - Specification 3.0.1 applies.
> b. Two nuclear service river water pumps must be OPERABLE.

Additionally, Tech Spec 3.0.1 states:

> 3.0.1 When a Limiting Condition for Operation is not met, except as provided in action called for in the specification, within one hour action shall be initiated to place the unit in a condition in which the specification does not apply by placing it, as applicable, in:
> 1. At least HOT STANDBY within the next 6 hours.
> 2. At least HOT SHUTDOWN within the following 6 hours, and
> 3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the action requirements, the action may be taken in accordance with the time limits of the specification as measured from the time of failure to meet the Limiting Condition for Operation. Applicability of these requirements is stated in the individual specifications.

Specification 3.0.1 is not applicable in COLD SHUTDOWN OR REFUELING SHUTDOWN.

Since the reactor is not critical, the Limiting Condition for Operation is not met, and therefore the TS action statement is not in effect.

D. Incorrect.
Part 1 is correct. The given conditions state that Nuclear River and Secondary River have been cross connected. NR-V-6 being OPEN indicates that Step 4.10 of OP-TM-541-901, Cross-Tie Secondary River to Supply Nuclear River, has been completed. The next step, step 4.11 states:

> 4.11 THROTTLE SR-V-2 to maintain Secondary River discharge pressure (SR-PI-134) above 21 psig.

Part 2 is incorrect. Tech Spec 3.3.2 is applicable while critical. T.S. 3.3.2 states:

> 3.3.2 Maintenance or testing shall be allowed during reactor operation on any component(s) in the makeup and purification, decay heat, RB emergency cooling water, RB spray, BWST level instrumentation, or cooling water systems which will not remove more than one train of each system from service. Components shall not be removed from service so that the affected system train is inoperable for more than 72 consecutive hours. If the system is not restored to meet the requirements of Specification 3.3.1 within 72 hours, the reactor shall be placed in a HOT SHUTDOWN condition within six hours.

Plausible if the candidate does not recognize that either: a) Tech Spec 3.3.2 applies only while critical, or b) believes that T.S. 3.0.1 is incorrect and chooses this by default.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076	A2.02
	Importance Rating		3.1

K/A: 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

Proposed Question: SRO Question # 88

Technical Reference(s):
OP-TM-541-901, pg 2, Rev 0
TS 3.3 pg 3-22, Rev 278
TS 3.0 pg 3-1, Rev 98

Proposed References to be provided to applicants during examination: None

Learning Objective: 531-GLO-14

Question Source:
Bank #
Modified Bank #
New X

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:
55.41
55.43 2

Comments:

The KA is matched because the candidate must predict the impact of low Service Water header pressure on the SWS. Then, based on those predictions, the candidate must use the correct procedure to correct, control, or mitigate the consequences of low Service Water header pressure.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze and interpret plant conditions to determine priorities, have knowledge and understand the operation and flowpath with Secondary River and Nuclear River water systems cross-connected, have knowledge of the procedural actions to take when river water systems are cross-connected, and Tech spec implications with River water systems cross-connected.

The question is at the SRO level because the candidates must demonstrate application of required Tech Spec actions in accordance with rules of application requirements. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

14 **ID: 984417** **Points: 1.00**

Plant Conditions:

- The Reactor has been tripped.
- RCS pressure is 1985 psig and steady.
- Pressurizer Relief Valve, RC-RV-1A, leakage is occurring to the RC Drain Tank.
- RC Drain Tank pressure rising.

Event:

- RC Drain Tank Relief Valve, WDG-V-1, fails to open.
- RCDT pressure continues to rise.
- The RCDT rupture disc bursts at 65 psig, reducing RCDT pressure to 15 psig.

Given the above information, identify the ONE selection below that describes:

(1) The Initial (i.e., immediately prior to the burst rupture disc.) tailpipe conditions and Final (i.e., after the burst rupture disc.) tailpipe conditions, and

(2) If achievable, whether gagging RC-RV-1A Out of Service for corrective maintenance is allowed.

A. (1) 300°F initial, 213°F final.
(2) RC-RV-1A may NOT be taken out of service because both pressurizer code safety valves are required to be in service to conform to the system design relief capabilities (i.e the code safety valves prevent overpressure for a rod withdrawal or feedwater line break accidents).

B. (1) 300°F initial, 213°F final.
(2) RC-RV-1A may be taken out of service because the remaining pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat.

C. (1) 312°F initial, 250°F final.
(2) RC-RV-1A may NOT be taken out of service because both pressurizer code safety valves are required to be in service to conform to the system design relief capabilities (i.e the code safety valves prevent overpressure for a rod withdrawal or feedwater line break accidents).

D. (1) 312°F initial, 250°F final.
(2) RC-RV-1A may be taken out of service because the remaining pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat.

Answer: D

Answer Explanation

A. Incorrect.

Part 1 is incorrect. These answers are based on 65 psia and 15 psia, rather than using 65 psig and 15 psig. Plausible because it provides correct answers for absolute pressures as listed and plotted in the Steam Table Book.

Note:

213 degrees = 15 psia saturation temperature.
300 degrees = 65 psia saturation temperature.

Part 2 is incorrect. This bases is applicable to make the reactor critical. Plausible if the candidate is not familiar with Tech Specs associated with the Pressurizer Code Safety Valves. IAW Tech Spec 3.1.1.3 Bases:

- Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal or feedwater line break accidents.

B. Incorrect.

Part 1 is incorrect. These answers are based on 65 psia and 15 psia, rather than using 65 psig and 15 psig. Plausible because it provides correct answers for absolute pressures as listed and plotted in the Steam Table Book.

Note:

213 degrees = 15 psia saturation temperature.
300 degrees = 65 psia saturation temperature.

Part 2 is correct. IAW Tech Spec 3.1.1.3 Bases:

- One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat.

C. Incorrect.

Part 1 is correct, described as follows:

- 312 degrees F is saturation temperature for 65 psig. Initial vapor enthalpy at 1138 BTU/lbm (corresponding to 2000 psia in Pressurizer) expanding down to 65 psig produces wet vapor at 312 degrees F with 4.9% moisture.
- 250 degrees F is saturation temperature for 15 psig. Initial vapor enthalpy at 1138 BTU/lbm (corresponding to 2000 psia in Pressurizer) expanding down to 15 psig produces wet vapor with 2.8% moisture.

Part 2 is incorrect. This bases is applicable to make the reactor critical. Plausible if the candidate is not familiar with Tech Specs associated with the Pressurizer Code Safety Valves. IAW Tech Spec 3.1.1.3 Bases:

- Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal or feedwater line break accidents.

D. Correct.
Part 1 is correct, described as follows:

- 312 degrees F is saturation temperature for 65 psig. Initial vapor enthalpy at 1138 BTU/lbm (corresponding to 2000 psia in Pressurizer) expanding down to 65 psig produces wet vapor at 312 degrees F with 4.9% moisture.
- 250 degrees F is saturation temperature for 15 psig. Initial vapor enthalpy at 1138 BTU/lbm (corresponding to 2000 psia in Pressurizer) expanding down to 15 psig produces wet vapor with 2.8% moisture.

Part 2 is correct. IAW Tech Spec 3.1.1.3 Bases:

- One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	007 2.2.36	
Importance Rating		4.2

K/A: Pressurizer Relief Tank / Quench Tank System: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Proposed Question: SRO Question # 89

Technical Reference(s): Steam Tables
TS 3.1, pg 3-2, Rev 266

Proposed References to be provided to applicants during examination: None

Learning Objective: 223-GLO-7

Question Source: Bank #

Modified Bank # IR-223-GLO-7-Q01

New

Question History: Last NRC Exam: N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:
The KA is matched because the candidate must analyze the effect of maintenance activities on the Pressurizer Relief Tank / Quench Tank System, such as degraded power sources, on the status of limiting conditions for operations.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to calculate the initial and final temperatures of the saturated steam at various pressures and knowledge of Tech Specs and bases to allow RC drain tank relief valve maintenance with the current plant conditions.

The question is at the SRO level because the candidates must provide knowledge of Tech Spec bases that are required to analyze TS required actions and terminology. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

Original Question: IR-223-GLO-7-Q01

Plant Conditions:
- Reactor tripped.
- RCS pressure 1985 psig.
- Pressurizer Relief Valve RC-RV-1A leakage to the RC Drain Tank.
- RC Drain Tank pressure rising.

Event:
- RC Drain Tank relief valve WDG -V-1 fails to open.
- RCDT pressure continues to rise.
- Rupture disc bursts at 65 psig, reducing RCDT pressure to 15 psig.

Based on these conditions identify the ONE selection below that describes the change in fluid conditions **downstream** of the Pressurizer relief valve when the RC Drain Tank rupture disc bursts:
(1) Initial tailpipe conditions. (i.e., immediately prior to the burst rupture disk.), and
(2) Final tailpipe conditions. (i.e., after the burst rupture disk.)

A. (1) 636 degrees F.
(2) 636 degrees F.

B. (1) 368 degrees F.
(2) 282 degrees F.

C. (1) 312 degrees F.
(2) 250 degrees F.

D. (1) 300 degrees F.
(2) 213 degrees F.

Answer: C

15 **ID: 984418** **Points: 1.00**

Plant Conditions:

- A Plant Heatup is in progress IAW 1102-1, Plant Heatup to 525°F.
- RCS Pressure is 250 psig and steady.
- Tave is 290°F and steady, controlled with OTSG's.
- Both trains of DHR are in Standby.
- DC Distribution Panel 1M transfer switch is selected to the "A" DC Distribution System position.

Event:

- The following alarms have actuated simultaneously:
  - AA-3-2 7 KV Bus Trouble.
  - AA-3-3 4 KV BOP Bus Trouble.
  - AA-3-5 480V BOP Bus Trouble.
  - A-1-7 Battery 1A Discharging.
  - A-2-7 Battery Charger 1A/1C/1E Trouble.
  - A-3-7 Inverter 1A/1C/1E Trouble.
  - B-3-1 4KV ES Bus Trouble.
  - PRF1-1-1 CRDM Brkr Test Trouble.
  - H&V A 4-2 Contr Bldg Bat Chgrs A Damper Tbl Fire-Smoke.
- A loss of breaker status lights at multiple control switches occurs.
- An NLO reports that "A" DC Voltage is 103VDC.

Based on these conditions, identify the ONE selection below that describes the:
(1) Applicable procedure to respond to these conditions, and
(2) Impact on the ability to operate plant equipment.

A. (1) OP-TM-AOP-023, "A" DC System Failure.
(2) The PORV Isolation Valve, RC-V-2, is not operable which reduces the ability to provide isolation of the PORV discharge line to positively control potential RCS depressurization.

B. (1) OP-TM-AOP-023, "A" DC System Failure.
(2) The Pressurizer Pilot Operated Relief Valve (PORV), RC-RV-2, is not operable which reduces the ability to provide the required overpressure relief when high pressure sources and flowpaths are in service.

C. (1) OP-TM-734-901, Energize "A" DC Using Panel Cross Tie.
(2) The PORV Isolation Valve, RC-V-2, is not operable which reduces the ability to provide isolation of the PORV discharge line to positively control potential RCS depressurization.

D. (1) OP-TM-734-901, Energize "A" DC Using Panel Cross Tie.
(2) The Pressurizer Pilot Operated Relief Valve (PORV), RC-RV-2, is not operable which reduces the ability to provide the required overpressure relief when high pressure sources and flowpaths are in service.

Answer: B

Answer Explanation

A. Incorrect.

Part 1 is correct. Entry conditions for OP-TM-AOP-023, "A" DC System Failure, have been met and therefore, the actions contained within the procedure are applicable. IAW OP-TM-AOP-023:

1.0 ENTRY CONDITIONS: ALL of the following:
- The OTSGs are being used for RCS heat removal.
- "A" or "C" "Battery Voltage" is less than 125 VDC and lowering.

NOTE: "A" or "C" battery voltage is indicated on battery ground detectors on 1A DC Distribution Panel. Loss of 1A DC Distribution panel is evident in the control room by the following simultaneous alarms:
- A-3-7, INVERTER 1A/1C/1E TROUBLE
- L-1-3, VOLTAGE REGULATOR DC LOSS
- PRF1-1-1, CRD BREAKER TEST TROUBLE

Part 2 is incorrect. Although the Bases is correct IAW Tech Spec 3.1.12 for the PORV Block Valve, RC-V-2, the conditions given do not match that required for the PORV Block Valve, RC-V-2, to be operable IAW Tech Spec 3.1.12. Additionally, a loss of the "A" DC system renders the PORV, RC-RV-2, inoperable, not the PORV Block Valve, RC-V-2. Plausible if the candidate is not familiar with the conditions necessary for the PORV Block Valve requirement to be operable or if the candidate does not recognize the loads off of the "A" DC System.

B. Correct.

Part 1 is correct. Entry conditions for OP-TM-AOP-023, "A" DC System Failure, have been met and therefore, the actions contained within the procedure are applicable. IAW OP-TM-AOP-023:

1.0 ENTRY CONDITIONS: ALL of the following:
- The OTSGs are being used for RCS heat removal.
- "A" or "C" "Battery Voltage" is less than 125 VDC and lowering.

NOTE: "A" or "C" battery voltage is indicated on battery ground detectors on 1A DC Distribution Panel. Loss of 1A DC Distribution panel is evident in the control room by the following simultaneous alarms:
- A-3-7, INVERTER 1A/1C/1E TROUBLE
- L-1-3, VOLTAGE REGULATOR DC LOSS
- PRF1-1-1, CRD BREAKER TEST TROUBLE

Part 2 is correct. OP-TM-AOP-0231, "A" DC System Failure Basis Document, states that the PORV is rendered inoperable. IAW OP-TM-AOP-0231:

Step 4.12: The PORV is inoperable. It is a 250 VDC load and is affected by loss of either battery bank. This step provides guidance to ensure compliance with TS 3.1.12 if RCS temperature is ≤329°F.

Since the RCS temperature given was 290°F, Tech Spec 3.1.12 applies. The bases for Tech Spec 3.1.12 states:

Below 329°F, the PORV setpoint is reduced to provide the required low temperature overpressure relief when high pressure sources and flowpaths are in service. There is no lower limit on the pressure actuation specified as lower setpoints also provide this same protection.

C. Incorrect.
Part 1 is incorrect. OP-TM-734-901, Energize "A" DC Using DC Panel Cross Tie, is used for an event where a fire has caused the loss of 1P 480V bus. There is no indication that a fire has occurred and therefore the prerequisites for OP-TM-734-901 are not met. Plausible if the candidate is not familiar with prerequisites for DC Electrical procedures. IAW OP-TM-734-901:

3.3 Prerequisites:
- 3.3.1 Verify the reactor is shutdown.
- 3.3.2 Verfiy battery chargers 1A, 1C, and 1E are not available.
- 3.3.3 Verify two B train battery chargers are in service.
- 3.3.4 Verify the procedure was initiated by Fire Mitigation procedure.
- 3.3.5 Verify A train battery voltage > 105 VDC.

Part 2 is incorrect. Although the Bases is correct IAW Tech Spec 3.1.12 for the PORV Block Valve, RC-V-2, the conditions given do not match that required for the PORV Block Valve, RC-V-2, to be operable IAW Tech Spec 3.1.12. Additionally, a loss of the "A" DC system renders the PORV, RC-RV-2, inoperable, not the PORV Block Valve, RC-V-2. Plausible if the candidate is not familiar with the conditions necessary for the PORV Block Valve requirement to be operable or if the candidate does not recognize the loads off of the "A" DC System.

D. Incorrect.
Part 1 is incorrect. OP-TM-734-901, Energize "A" DC Using DC Panel Cross Tie, is used for an event where a fire has caused the loss of 1P 480V bus. There is no indication that a fire has occurred and therefore the prerequisites for OP-TM-734-901 are not met. Plausible if the candidate is not familiar with prerequisites for DC Electrical procedures. IAW OP-TM-734-901:

3.3 Prerequisites:
- 3.3.1 Verify the reactor is shutdown.
- 3.3.2 Verfiy battery chargers 1A, 1C, and 1E are not available.
- 3.3.3 Verify two B train battery chargers are in service.
- 3.3.4 Verify the procedure was initiated by Fire Mitigation procedure.
- 3.3.5 Verify A train battery voltage > 105 VDC.

Part 2 is correct. OP-TM-AOP-0231, "A" DC System Failure Basis Document, states that the PORV is rendered inoperable. IAW OP-TM-AOP-0231:

> Step 4.12: The PORV is inoperable. It is a 250 VDC load and is affected by loss of either battery bank. This step provides guidance to ensure compliance with TS 3.1.12 if RCS temperature is ≤329°F.

Since the RCS temperature given was 290°F, Tech Spec 3.1.12 applies. The bases for Tech Spec 3.1.12 states:

Below 329°F, the PORV setpoint is reduced to provide the required low temperature overpressure relief when high pressure sources and flowpaths are in service. There is no lower limit on the pressure actuation specified as lower setpoints also provide this same protection.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	063	2.2.25	

Importance Rating 4.2

K/A: DC Electrical Distribution System: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Proposed Question: SRO Question # 90

Technical Reference(s): TS 3.1 pg 3-18e, Rev 234
OP-TM-AOP-023, pg 1, Rev 4
OP-TM-AOP-0231, pg 10, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP023-PCO-2

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2

Comments:
The KA is matched because the candidate must demonstrate a knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits associated with the DC Electrical Distribution System.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to know entry conditions for abnormal and emergency operating procedures for loss of the DC system and have knowledge of the affected DC loads on a malfunction or loss of the system.

The question is at the SRO level because the candidates must provide knowledge of Tech Spec bases that are required to analyze TS required actions and terminology. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

16 **ID: 984419** **Points: 1.00**

Plant Conditions:

- A plant startup is in progress IAW 1102-2, Plant Startup.
- NI-3, Intermediate Range Nuclear Instrument, is INOPERABLE due to a log amplifier failure.
    - Repairs are scheduled for next shift.
- The reactor is critical with all Power Range Nuclear Instruments reading 7%.

Event:

- NI-4, Intermediate Range Nuclear Instrument, indication dropped suddenly and is stable at 10E-11 amps.

Given the above information, which one of the following describes the action to be directed by the CRS and the correct basis for this action?

A. Continue the plant startup IAW 1102-2, Plant Startup, because **overlap** with at least 2 of 4 Power Range Nuclear Instruments has been verified.

B. Continue the plant startup IAW 1102-2, Plant Startup, because startup rate (SUR) protection is not required after reactor power is past the **point of adding heat**.

C. Restore NI-3 or NI-4 to operable within one hour or be in hot shutdown within 6 hours IAW 1102-10, Plant Shutdown, because the plant is outside the Tech Spec envelope with zero **redundancy** of Intermediate Range Nuclear Instruments.

D. Restore NI-3 or NI-4 to operable within one hour or be in hot shutdown within 6 hours IAW 1102-10, Plant Shutdown, because no Intermediate Range instrumentation channels are **operable** in the event of a Control Rod Withdraw accident.

Answer: D

Answer Explanation

A. **Incorrect.** This is plausible because it would be correct if 2/4 PRNI are >10% power [Table 3.5-1 (Note (b)].

B. **Incorrect.** This is plausible because the protection provided from IRNI is Control Rod OUT INHIBIT at 3 DPM and SUR decays rapidly after the POAH is reached.

C. **Incorrect.** This is plausible because there are no operable IRNI's. If the applicant does not fully understand Table 3.5-1 and the notes associated with IRNI operability then it would appear that TS 3.0.1 applies.

D. **Correct.** According to Technical Specification 3.5.1 (p3-27; Amendment 189), the reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5-1, Column "A" and "B" are met. According to Technical Specification 3.5.1, Table 3.5-1 (p3-29/30; Amendment 247/189), Functional Unit A.3 requires that a Minimum of one Intermediate Range Channel be operable. Since this condition is not met, Column "A" is not met. A note is provided (b) to indicate that when 2 of 4 Power Range instrument channels are > 10% Full Power, IR instrumentation is NOT required. This is NOT the case in the stated conditions.

When the requirements of Column A and B are not met, another note is provided (a) to indicate the required action:

- Restore within one hour or place the unit in Hot Standby within an additional six hours.

According to FSAR Section 7.3.1.2, System Design:

- The intermediate range instrumentation has two identical log channels originating in two electrically adjustable gamma compensated ion chambers. Each channel provides eight decades of flux level information in terms of the log of ion chamber current and startup rate. The ion chamber measuring range is from 10-11 to 10-3 ampere. The startup rate range is from -0.5 to +5 decades per minute. A high startup rate of +3 decades per minute in either channel will initiate a control rod withdraw inhibit.

Additionally, IAW FSAR Section 14.1.2.2:

14.1.2.2 Startup Accident
a. Identification of Cause: During a startup, an uncontrolled reactivity addition could cause a nuclear excursion. The uncontrolled reactivity addition, through rod withdrawal from zero power, is a startup accident.
The following design provisions minimize the possibility of inadvertent continuous rod withdrawal and limit the potential for power excursions:

4) A startup withdrawal stop and alarm are provided in the intermediate range.

The criteria for the analysis of this accident is that the Reactor Protection System shall be designed to limit (1) the reactor thermal power to the design overpower condition (112 percent rated power) and (2) the Reactor Coolant System pressure so as not to exceed the ASME Code allowable pressure limit of 2750 psig (110 percent of Design Pressure).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	015	A2.02
	Importance Rating		3.5

K/A: Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulty or erratic operation of detectors or compensating components.

Proposed Question: SRO Question # 91

Technical Reference(s):	TS 3.5, pg 3-27, Rev 189 TS 3.5, pg 3-27a, Rev 273 TS Table 3.5-1, pg 3-29, Rev 247 TS Table 3.5-1, pg 3-30 Rev 189

Proposed References to be provided to applicants during examination: None

Learning Objective: 623-GLO-14

Question Source:	Bank #	IS-623-GLO-14-Q03
	Modified Bank #	
	New	

Question History: NA   Last NRC Exam: NA

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

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:

The KA is matched because the candidate must predict the impact of the faulty or erratic operation of detectors or compensating components on the NIS. Then, based on those predictions, the candidate must use the correct procedure to correct, control, or mitigate the consequences of the faulty or erratic operation of detectors or compensating components.

The question is at the Comprehension/Analysis cognitive level because the operator must evaluate the stated conditions, and determine a correct course of action as required by Technical Specifications, where each action could be made to be correct if conditions were changed. In doing so, the operator demonstrates that the TS can be correctly applied, demonstrating understanding of the overall effect on plant operation.

The question is at the SRO level because the candidates must provide knowledge of Tech Spec bases that are required to analyze TS required actions and terminology. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

17 **ID: 984421** **Points: 1.00**

Plant conditions:

- Reactor is operating at 100%, with ICS in full automatic.
- RPS Cabinet B is de-energized for maintenance.

Sequence of events:

- Due to a slow failure, a false signal output reduction is occurring from RPS Channel A RCS Pressure transmitter.
- RPS Channel A is placed in Manual Bypass, RPS Channel A RCS pressure indication continues to LOWER slowly.
- ACTUAL RCS pressure is now 2260 psig, RISING slowly due to automatic Pressurizer heater operation.

Based on these conditions, identify the ONE selection below that describes:
(1) Impact of the malfunction on operation of the Spray Valve **and** PORV.
(2) The procedural actions required to mitigate the consequences of the failure.

A. (1) AUTOMATIC operation of the Spray Valve ONLY is Failed.
(2) Turn off Pressurizer heaters IAW OP-TM-AOP-043, Loss of Pressurizer.

B. (1) AUTOMATIC operation of the Spray Valve ONLY is Failed.
(2) Turn off Pressurizer heaters and fully open the Spray Valve IAW OP-TM-MAP-G0308, RC PRESS NARROW RNG HI/LO.

C. (1) AUTOMATIC operation of the Spray Valve and the PORV is Failed.
(2) Turn off Pressurizer heaters IAW OP-TM-AOP-043, Loss of Pressurizer.

D. (1) AUTOMATIC operation of the Spray Valve and the PORV is Failed.
(2) Turn off Pressurizer heaters and fully open the Spray Valve IAW OP-TM-MAP-G0308, RC PRESS NARROW RNG HI/LO.

Answer: D

Answer Explanation

A. Incorrect.

Part 1 is incorrect. Continued reduction in false pressure signal will prevent automatic open operation for the Spray Valve and the PORV. Distracter is plausible because the bistables that operate the Spray Valve and the PORV are still operable, even though the signal input is failing.

Part 2 is incorrect. OP-TM-MAP-G0308 is the correct procedure to be implemented from the choices provided. Distracter is plausible if the candidate is not familiar with OP-TM-AOP-043, Loss of Pressurizer and selects this procedure because automatic open operations are failed. OP-TM-AOP-043 is selected for the following issues:

- Inadequate pressurizer heater capacity to maintain RCS pressure.
- No valid pressurizer level indication.
- Pressurizer level > 370" and RCS temperature stable or lowering.
- RCS <329°F and pressurizer level > 100" for more than 30 minutes.

B. Incorrect.
Part 1 is incorrect. Continued reduction in false pressure signal will prevent automatic open operation for the Spray Valve and the PORV. Distracter is plausible because the bistables that operate the Spray Valve and the PORV are still operable, even though the signal input is failing.

Part 2 is correct. Although the setpoint for MAP G-3-8 is >2255 psig, the current state of instrument failure will lead to the alarm coming in, therefore the CRS will enter OP-TM-MAP-G0308, RCS Press Narrow Rng Hi/Lo, to take manual control of Pressurizer heaters and spray, based on "approaching" criteria, as allowed in OS-24, Conduct of Operations During Abnormal and Emergency Events, Section 4.1.14:

4.1.14 Guidance on using APPROACHING

A. If it is clear that the plant trend is going to reach a setpoint requiring action, Shift Management may elect to perform the action before the setpoint is reached. This applies to EOP and AOP entry, safety system actuation, and the performance of emergency response procedure steps.

C. Incorrect.
Part 1 is correct. IAW TQ-TM-104-624-C001, Non Nuclear Instrumentation System, page 29 of 99:

Narrow Range RC Pressure, 1700 – 2500 psig:
Rosemount Capacitance Type Detectors.

1) 2 detectors per loop: RC3A-PT1/2 for the "A" loop (A0586); and RC3B-PT1/2 for the "B" loop (A0587).
2) RC3A-PT2 feeds only the "C" RPS cabinet, and RC3B-PT2 feeds only the "D" RPS cabinet.
3) RC3A-PT1 feeds the "A" RPS cabinet and RC3B-PT1 feeds the "B" RPS cabinet.
    a) Both PTs feed RC3-PR paperless recorder narrow range and a bar graph meter on the console. Hi/Lo pressure alarm G-3-8 comes from a relay in RC3-PR for the "A" Hotleg and the bar graph meter for the "B" Hotleg (setpoint 2255/2055).
    b) Both PT's are SASS monitored and are capable of providing an input to:
    - PZR Heaters, both modulating and bistable controlled.
    - Spray Valve control.
    - PORV High Pressure setpoint control.

The Narrow Range Pressure instruments will not swap due to SASS sensing a slow failure of the detector. Therefore, the false rising signal will be the one inputed to the associated Pressurizer heaters and valves. IAW TQ-TM-104-624-C001, Non Nuclear Instrumentation System, page 24 of 99:

NNI System and Assignment of SASS

a. If SASS senses one of the parallel instruments more than 3% of full scale away from the other, it will announce a MISMATCH (MAP H-3-2). An AUTOMATIC transfer will not occur if a SASS monitored channel is in MISMATCH.
b. If SASS senses one of the parallel instruments changing more than 8%/sec. (SASS ACTUATION), it will automatically select the other instrument and provide a computer alarm.
c. After an automatic transfer, SASS will not allow a manual transfer back to the failed instrument. It must be reset down in the SASS modules in the ICS/NNI cabinets.
d. A manual transfer with two instruments in a mismatch is allowed.

Part 2 is incorrect. OP-TM-MAP-G0308 is the correct procedure to be implemented from the choices provided. Distracter is plausible if the candidate is not familiar with OP-TM-AOP-043, Loss of Pressurizer and selects this procedure because automatic open operations are failed. OP-TM-AOP-043 is selected for the following issues:

- Inadequate pressurizer heater capacity to maintain RCS pressure.
- No valid pressurizer level indication.
- Pressurizer level > 370" and RCS temperature stable or lowering.
- RCS <329□F and pressurizer level > 100" for more than 30 minutes.

D. Correct.
Part 1 is correct. IAW TQ-TM-104-624-C001, Non Nuclear Instrumentation System, page 29 of 99:

Narrow Range RC Pressure, 1700 – 2500 psig:
Rosemount Capacitance Type Detectors.
1) 2 detectors per loop: RC3A-PT1/2 for the "A" loop (A0586); and RC3B-PT1/2 for the "B" loop (A0587).
2) RC3A-PT2 feeds only the "C" RPS cabinet, and RC3B-PT2 feeds only the "D" RPS cabinet.
3) RC3A-PT1 feeds the "A" RPS cabinet and RC3B-PT1 feeds the "B" RPS cabinet.
a) Both PTs feed RC3-PR paperless recorder narrow range and a bar graph meter on the console. Hi/Lo pressure alarm G-3-8 comes from a relay in RC3-PR for the “A” Hotleg and the bar graph meter for the “B” Hotleg (setpoint 2255/2055).
b) Both PT's are SASS monitored and are capable of providing an input to:
- PZR Heaters, both modulating and bistable controlled.
- Spray Valve control.
- PORV High Pressure setpoint control.

The Narrow Range Pressure instruments will not swap due to SASS sensing a slow failure of the detector. Therefore, the false rising signal will be the one inputed to the associated Pressurizer heaters and valves. IAW TQ-TM-104-624-C001, Non Nuclear Instrumentation System, page 24 of 99:

NNI System and Assignment of SASS
a. If SASS senses one of the parallel instruments more than 3% of full scale away from the other, it will announce a MISMATCH (MAP H-3-2). An AUTOMATIC transfer will not occur if a SASS monitored channel is in MISMATCH.
b. If SASS senses one of the parallel instruments changing more than 8%/sec. (SASS ACTUATION), it will automatically select the other instrument and provide a computer alarm.
c. After an automatic transfer, SASS will not allow a manual transfer back to the failed instrument. It must be reset down in the SASS modules in the ICS/NNI cabinets.
d. A manual transfer with two instruments in a mismatch is allowed.

Part 2 is correct. Although the setpoint for MAP G-3-8 is >2255 psig, the current state of instrument failure will lead to the alarm coming in, therefore the CRS will enter OP-TM-MAP-G0308, RCS Press Narrow Rng Hi/Lo, to take manual control of Pressurizer heaters and spray, based on "approaching" criteria, as allowed in OS-24, Conduct of Operations During Abnormal and Emergency Events, Section 4.1.14:

4.1.14 Guidance on using APPROACHING
A. If it is clear that the plant trend is going to reach a setpoint requiring action, Shift Management may elect to perform the action before the setpoint is reached. This applies to EOP and AOP entry, safety system actuation, and the performance of emergency response procedure steps.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	016	A2.01	
	Importance Rating			3.1

K/A: Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure

Proposed Question: SRO Question # 92

Technical Reference(s):
TQ-TM-104-624-C001, pg 24, 29, Rev 002
OP-TM-MAP-G0308, pg 1, Rev 003
OS-24, pg 12, Rev 024

Proposed References to be provided to applicants during examination: None

Learning Objective: 624-GLO-14

Question Source:
Bank # IS-624-GLO-14-Q01
Modified Bank #
New

Question History: Last NRC Exam: ILT 10-02

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

10 CFR Part 55 Content:
55.41
55.43 5

Comments:

The KA is matched because the candidate must predict the impact of a detector failure on the NNIS. Then, based on the prediction, the candidate must use the correct procedure to correct, control, or mitigate the consequences of the detector failure.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to understand the function and operation of the NNI and RPS system to include indications, control signals, and failure modes, be able to analzye and interpret the pant conditions to determine priorities, and have knowledge of primary plant parameters, limits, and alarms to determine when actions must be taken.

The question is at the SRO level because the candidates must provide knowledge of administration procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 7).

18 **ID: 984422** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- The Containment Fire Service Spool Piece is **NOT** installed.

Event:

- PLF-1-1, Reactor BLDG fire, alarm is received.
- An NLO reports the panel fire alarm is for RB-FZ-1A, Reactor Building 281' North Outside "D" ring.
- The CRS announces entry into OP-TM-AOP-001, Fire, based on confirmation of fire.
- A spurious 1600 psig ES actuation has occurred.
- RB Pressure is currently 0.2 psig and slowly rising.
- Smoke and Fire have NOT entered the Control Tower.
- Temperatures in the Reactor Building are as follows and are rising at 0.2°F/minute.

Location	Temp F	Location	Temp F	Location	Temp F
SE Wall Elev. 352'	INVAL	NW Wall Elev 352'	131.2	S Rx Wall Elev 321'	100.9
NW Sec Shield Elev 352'	134.3	E Wall Elev 400'	103.9	NE Wall Elev 287'	127.0
NE Sec Shield Elev 352'	129.7	S Sec Shield Elev 352'	INVAL	S Wall Elev 287'	99.6
E Wall Elev 382'	108.5	NW Sec Shield Elev 352'	129.4	NW Wall Elev 287'	128.8
NE Sec Shield Elev 352'	130.0	NE Wall Elev 314'	127.4	E Sec Shield Elev 352'	INVAL
NW Sec Shield Elev 352'	134.2	S Wall Elev 314'	INVAL	NW Sec Shield Elev 287'	INVAL
NE Sec Shield Elev 352'	129.9	NW Wall Elev 314'	129.1	NE Sec Shield Elev 364'	127.6
NW Sec Shield Elev 352'	134.5	E Sec Shield Elev 352'	110.1	N Sec Shield Elev 364'	144.2

Given the above information and assuming no operator actions have occurred, which ONE of the following describes:

(1) The procedures to be entered in addition to OP-TM-AOP-001, Fire, and

(2) The applicable timeclock action that applies IAW Tech Spec 3.17, Reactor Building Air Temperature?

A. (1) OP-TM-AOP-046, Inadvertant ESAS, and OP-TM-811-901, Containment Fire Service.
(2) Immediately initiate a Plant Shutdown and Cooldown.

B. (1) OP-TM-AOP-046, Inadvertant ESAS, and OP-TM-811-901, Containment Fire Service.
(2) Reduce average temperature to below Tech Spec limits within 8 hours.

C. (1) OP-TM-AOP-058, Toxic Gas Release, and OP-TM-534-901, RB Emergency Cooling Operations.
(2) Immediately initiate a Plant Shutdown and Cooldown.

D. (1) OP-TM-AOP-058, Toxic Gas Release, and OP-TM-534-901, RB Emergency Cooling Operations.
(2) Reduce average temperature to below Tech Spec limits within 8 hours.

Answer: B

Answer Explanation

A. Incorrect.

Part 1 is correct. OP-TM-AOP-001, Fire, states to initiate OP-TM-811-901, Containment Fire Service, if the Containment Fire Service Spool Pice is not installed. This is the condition given in the stem and therefore OP-TM-811-901 is applicable. OP-TM-AOP-001, Step 3.13:

If fire is in Unit 1 containment building, then perform the following:

A. ACTUATE RB Evacuation alarm.

B. If containment fire service spool piece is installed, then OPEN FS-V-367 (TB 305' west of CO-P-2s, 13' above floor)

C. If containment fire service spool piece is not installed, then INITIATE OP-TM-811-901, "Containment Fire Service," to place containment fire service system in service.

Additionally, OP-TM-AOP-001, Fire, states in Attachment 34, RB-FZ-1A - Reactor Building 281' North Outside D-Ring, that a fire in this area may cause a spurious 1600 psig ES actuation. OP-TM-AOP-0011, Fire Basis Document, states in Attachment 34, RB-FZ-1A Reactor Building 281' North Outside D-Ring, that a spurious HPI actuation may occur because RC3A-PT-3 and RC3A-PT-4 are affected by a fire in this area and that the implementing procedure is OP-TM-AOP-046, Inadvertant ESAS.

Part 2 is incorrect. Plausible if the candidate does not understand the Tech Spec Action Statement. IAW Tech Spec 3.17:

3.17 REACTOR BUILDING AIR TEMPERATURE

Applicability: This specification applies to the average air temperature of the primary containment during power operations.

Specification

3.17.1 Primary containment average air temperature above Elev. 320 shall not exceed 130°F and average air temperature below Elev. 320 shall not exceed 120°F.

3.17.2 If, while the reactor is critical, the above stated temperature limits are exceeded, the average temperature shall be reduced to the above limits within 8 hours, or be in at least HOT STANDBY within the next six (6) hours and in COLD SHUTDOWN within the following thirty (30) hours.

3.17.3 The primary containment average air temperature shall be calculated as follows:

a) The average temperature above elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 13 locations above elevation 320. A list of locations is given below.

b) The average temperatures below elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 4 locations below Elev. 320

B. Correct.
Part 1 is correct. OP-TM-AOP-001, Fire, states to initiate OP-TM-811-901, Containment Fire Service, if the Containment Fire Service Spool Pice is not installed. This is the condition given in the stem and therefore OP-TM-811-901 is applicable. OP-TM-AOP-001, Step 3.13:

> If fire is in Unit 1 containment building, then perform the following:
> A. ACTUATE RB Evacuation alarm.
> B. If containment fire service spool piece is installed, then OPEN FS-V-367 (TB 305' west of CO-P-2s, 13' above floor)
> C. If containment fire service spool piece is not installed, then INITIATE OP-TM-811-901, "Containment Fire Service," to place containment fire service system in service.

Additionally, OP-TM-AOP-001, Fire, states in Attachment 34, RB-FZ-1A - Reactor Building 281' North Outside D-Ring, that a fire in this area may cause a spurious 1600 psig ES actuation. OP-TM-AOP-0011, Fire Basis Document, states in Attachment 34, RB-FZ-1A Reactor Building 281' North Outside D-Ring, that a spurious HPI actuation may occur because RC3A-PT-3 and RC3A-PT-4 are affected by a fire in this area and that the implementing procedure is OP-TM-AOP-046, Inadvertant ESAS.

Part 2 is correct. Average temperature exceeds the listed Tech Spec limit of 120F below 320 ft. IAW Tech Spec 3.17:

> 3.17 REACTOR BUILDING AIR TEMPERATURE
> Applicability: This specification applies to the average air temperature of the primary containment during power operations.
>
> Specification
> 3.17.1 Primary containment average air temperature above Elev. 320 shall not exceed 130°F and average air temperature below Elev. 320 shall not exceed 120°F.
> 3.17.2 If, while the reactor is critical, the above stated temperature limits are exceeded, the average temperature shall be reduced to the above limits within 8 hours, or be in at least HOT STANDBY within the next six (6) hours and in COLD SHUTDOWN within the following thirty (30) hours.
> 3.17.3 The primary containment average air temperature shall be calculated as follows:
> a) The average temperature above elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 13 locations above elevation 320. A list of locations is given below.
> b) The average temperatures below elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 4 locations below Elev. 320

C. Incorrect.
Part 1 is incorrect. Although OP-TM-534-901 may be entered either because of the high RB temperatures below 320 ft or because of the inadvertant ESAS, OP-TM-AOP-058, Toxic Gas Release, will not be entered at this time. None of the entry criteria are given in the stem. Plausible if the candidate believes that entry is required. IAW OP-TM-AOP-058:

> 1.0 ENTRY CONDITIONS
> - Any of the following conditions occur:
>   – Notification is received from Dauphin County Emergency Operations Center of a large toxic gas release external to the site,
>   – Confirmed report of toxic gas at or near the site from an offsite source.
>   – Smell of toxic gas in the Control Room.

Part 2 is incorrect. Plausible if the candidate does not understand the Tech Spec Action Statement. IAW Tech Spec 3.17:

3.17 REACTOR BUILDING AIR TEMPERATURE

Applicability: This specification applies to the average air temperature of the primary containment during power operations.

Specification

3.17.1 Primary containment average air temperature above Elev. 320 shall not exceed 130°F and average air temperature below Elev. 320 shall not exceed 120°F.

3.17.2 If, while the reactor is critical, the above stated temperature limits are exceeded, the average temperature shall be reduced to the above limits within 8 hours, or be in at least HOT STANDBY within the next six (6) hours and in COLD SHUTDOWN within the following thirty (30) hours.

3.17.3 The primary containment average air temperature shall be calculated as follows:

a) The average temperature above elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 13 locations above elevation 320. A list of locations is given below.

b) The average temperatures below elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 4 locations below Elev. 320

D. Incorrect.

Part 1 is incorrect. Although OP-TM-534-901 may be entered either because of the high RB temperatures below 320 ft or because of the inadvertant ESAS, OP-TM-AOP-058, Toxic Gas Release, will not be entered at this time. None of the entry criteria are given in the stem. Plausible if the candidate believes that entry is required. IAW OP-TM-AOP-058:

1.0 ENTRY CONDITIONS

- Any of the following conditions occur:
  - Notification is received from Dauphin County Emergency Operations Center of a large toxic gas release external to the site,
  - Confirmed report of toxic gas at or near the site from an offsite source.
  - Smell of toxic gas in the Control Room.

Part 2 is correct. Average temperature exceeds the listed Tech Spec limit of 120F below 320 ft. IAW Tech Spec 3.17:

3.17 REACTOR BUILDING AIR TEMPERATURE

Applicability: This specification applies to the average air temperature of the primary containment during power operations.

Specification

3.17.1 Primary containment average air temperature above Elev. 320 shall not exceed 130°F and average air temperature below Elev. 320 shall not exceed 120°F.

3.17.2 If, while the reactor is critical, the above stated temperature limits are exceeded, the average temperature shall be reduced to the above limits within 8 hours, or be in at least HOT STANDBY within the next six (6) hours and in COLD SHUTDOWN within the following thirty (30) hours.

3.17.3 The primary containment average air temperature shall be calculated as follows:

a) The average temperature above elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 13 locations above elevation 320. A list of locations is given below.

b) The average temperatures below elevation 320 will be calculated by taking the arithmetic average of the temperatures from at least 4 locations below Elev. 320

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	086	2.4.47	
	Importance Rating			4.2

K/A: Fire Protection System: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: SRO Question # 93

Technical Reference(s): OP-TM-AOP-001, pg 5, 119, Rev 9
TS 3.17, pg 3-80, Rev 157

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP001-PCO-2

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Comments:
The KA is matched because the candidate must diagnose and recognize trends associated with the Fire Protection System in an accurate and timely manner utilizing the appropriate control room reference material.

The question is at the analysis cognitive level because the candidate is required to analyze and interpret plant conditions and determine priorities, have an understanding of entry conditions for abnormal and emergency operating procedurs, and have knowledge of Tech Spec LCO's for containment parameters.

The question is at the SRO level because the candidates must demonstrate application of required Tech Spec actions in accordance with rules of application requirements. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

19 **ID: 984423** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power.
- The specific activity of the primary and secondary coolant are as follows:

	Two days ago at 1200	Yesterday at 1200	Today at 1200
Combined Primary to Secondary Steam Generator Leakage	0.4 GPM	0.6 GPM	0.6 GPM
RCS Dose Equivalent I-131	0.50 microcuries/gram	0.53 microcuries/gram	0.52 microcuries/gram
OTSG Dose Equivalent I-131	0.05 microcuries/gram	0.053 microcuries/gram	0.052 microcuries/gram

Given the above information, identify the ONE choice that identifies:
(1) Which specific activity is out of specification, and
(2) The proper basis.

A. (1) RCS.
(2) Dose from normal condenser Offgas will exceed the 10CFR acceptance criteria.

B. (1) RCS.
(2) Doses exceeding 10CFR acceptance criteria could occur in the event of a Steam Generator Tube Rupture.

C. (1) OTSG.
(2) Dose from normal condenser Offgas will exceed the 10CFR acceptance criteria.

D. (1) OTSG
(2) Doses exceeding 10CFR acceptance criteria could occur in the event of a Steam Generator Tube Rupture..

Answer: B

Answer Explanation

A. Incorrect.
Part 1 is correct. IAW TS 3.1.4:

3.1.4.1 LIMITING CONDITION FOR OPERATION: RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT Xe-133 specific activity shall be limited to:
a. Less than or equal to 0.35 microcuries/gram DOSE EQUIVALENT I-131, and
b. Less than or equal to 797 microcuries/gram DOSE EQUIVALENT Xe-133.

3.1.4.2 APPLICABILITY: At all times except REFUELING SHUTDOWN and COLD SHUTDOWN.

3.1.4.3 ACTION:

a.1 With DOSE EQUIVALENT I-131 not within limit, perform the sampling and analysis requirements of Table 4.1.3 until the RCS DOSE EQUIVALENT I-131 is restored to within limit, AND

a.2 Verify that DOSE EQUIVALENT I-131 is less than or equal to 60 microcuries/gram, AND

a.3 Restore DOSE EQUIVALENT I-131 to within limit within 48 hours.

a.4 If the requirements of a.1, a.2 or a.3 cannot be met, be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within 36 hours.

Part 2 is incorrect. Plausible if the candidate confuses the basis for activity with that of RCS or Makeup leakage. TS 3.1.6 Basis:

- Buildup of radioactive solids in the secondary side of a steam generator and the presence of radioactive ions in the condensate can be tolerated to only a small degree. Therefore, the appearance of activity in the condenser off-gas, or any other possible indications of primary to secondary leakage such as water inventories, condensate demineralizer activity, etc., shall be considered positive indication of primary to secondary leakage and steps shall be taken to determine the source and quantity of the leakage.

Plausible if the candidate does not recognize that a steam generator tube leak does not exist. Plausible since T.S. 3.1.4 includes a tube rutpure.

B. Correct.
Part 1 is correct. IAW TS 3.1.4:

3.1.4.1 LIMITING CONDITION FOR OPERATION: RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT Xe-133 specific activity shall be limited to:

a. Less than or equal to 0.35 microcuries/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 797 microcuries/gram DOSE EQUIVALENT Xe-133.

3.1.4.2 APPLICABILITY: At all times except REFUELING SHUTDOWN and COLD SHUTDOWN.

3.1.4.3 ACTION:

a.1 With DOSE EQUIVALENT I-131 not within limit, perform the sampling and analysis requirements of Table 4.1.3 until the RCS DOSE EQUIVALENT I-131 is restored to within limit, AND

a.2 Verify that DOSE EQUIVALENT I-131 is less than or equal to 60 microcuries/gram, AND

a.3 Restore DOSE EQUIVALENT I-131 to within limit within 48 hours.

a.4 If the requirements of a.1, a.2 or a.3 cannot be met, be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within 36 hours.

Part 2 is correct. IAW TS 3.1.4 Basis:

- The iodine specific activity in the reactor coolant is limited to 0.35 □Ci/gm DOSE EQUIVALENT I 131, and the noble gas specific activity in the reactor coolant is limited to 797 μCi/gm DOSE EQUIVALENT Xe-133.
- The limits on specific activity ensure that offsite and control room doses will meet the appropriate 10CFR100.11 and 10CFR50 Appendix A GDC19 acceptance criteria.
- The SLB and SGTR accident analyses show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the 10CFR100.11 and 10CFR50 Appendix A GDC19 acceptance criteria.

C. Incorrect.
Part 1 is incorrect. T.S. 3.13.1 states:

- The specific activity of the secondary coolant system shall be ≤ 0.10 μ Ci/gram DOSE EQUIVALENT I-131.

This is the case in the given conditions. Plausible if the candidate is not familiar with the limit.

Part 2 is incorrect. Plausible if the candidate confuses the basis for activity with that of RCS or Makeup leakage. TS 3.1.6 Basis:

- Buildup of radioactive solids in the secondary side of a steam generator and the presence of radioactive ions in the condensate can be tolerated to only a small degree. Therefore, the appearance of activity in the condenser off-gas, or any other possible indications of primary to secondary leakage such as water inventories, condensate demineralizer activity, etc., shall be considered positive indication of primary to secondary leakage and steps shall be taken to determine the source and quantity of the leakage.

Plausible if the candidate does not recognize that a steam generator tube leak does not exist. Plausible since T.S. 3.1.4 includes a tube rutpure.

D. Incorrect.
Part 1 is incorrect. T.S. 3.13.1 states:

- The specific activity of the secondary coolant system shall be ≤ 0.10 μ Ci/gram DOSE EQUIVALENT I-131.

This is the case in the given conditions. Plausible if the candidate is not familiar with the limit.

Part 2 is correct. IAW TS 3.1.4 Basis:

- The iodine specific activity in the reactor coolant is limited to 0.35 □Ci/gm DOSE EQUIVALENT I 131, and the noble gas specific activity in the reactor coolant is limited to 797 μCi/gm DOSE EQUIVALENT Xe-133.
- The limits on specific activity ensure that offsite and control room doses will meet the appropriate 10CFR100.11 and 10CFR50 Appendix A GDC19 acceptance criteria.
- The SLB and SGTR accident analyses show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the 10CFR100.11 and 10CFR50 Appendix A GDC19 acceptance criteria.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.34	
	Importance Rating		3.1

K/A: Knowledge of primary and secondary plant chemistry limits.

Proposed Question: SRO Question # 94

Technical Reference(s): TS 3.1, pg 3-8, Rev 272

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO-14

Question Source:	Bank #	IS-220-GLO-14-Q04
	Modified Bank #	
	New	

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	
	55.43	4

Comments:
The KA is matched because the candidate must demonstrate knowledge of primary and secondary plant chemistry limits.

The question is at the Memory/Fundamental Knoiwledge cognitive level because the candidate is required to have knowledge of primary and secondary plant chemistry limits, and knowledge of Tech Spec and 10CFR limits requiring action to be taken and the bases for that action.

The question is at the SRO level because the candidate must demonstrate analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 6).

20 **ID: 984424** **Points: 1.00**

Plant Conditions:

- Transfer of an irradiated fuel assembly is in progress using the Main Fuel Handling Bridge.

Event:

- The bridge stops travel and will NOT move south to complete the operation.
- Investigation reveals that the south wall interlock has failed in the actuated position.
- It is decided to bypass the south wall interlock.
    - This action is not addressed specifically in any procedure.

Based on the above conditions and IAW 1507-3, Main Fuel Handling Bridge Operating Instructions, identify the individuals required to approve bypassing this interlock to continue the transfer.

A. Core Load Engineer,
Station Duty Manager, and
Operations Shift Manager.

B. Core Load Engineer,
Refueling Outage Manager, and
Licensed Fuel Handling Supervisor.

C. Station Duty Manager,
Operations Shift Manager, and
Licensed Fuel Handling Supervisor.

D. Station Duty Manager,
Refueling Outage Manager, and
Licensed Fuel Handling Supervisor.

Answer: C

Answer Explanation

A. Incorrect. These are not the required personnel IAW 1507-3, Main Fuel Handling Bridge Operating Instructions. Plausible if the candidate is not familiar with the required permission needed to bypass interlocks. IAW 1507-3, Limits and Precautions:

5.1.1 Interlocks that are bypassed for non-procedure driven events shall have the following requirements.

- Licensed Fuel Handling Supervisor (LFHS), Station Duty Manager and Operations Shift Manager have given permission.

B. Incorrect. These are not the required personnel IAW 1507-3, Main Fuel Handling Bridge Operating Instructions. Plausible if the candidate is not familiar with the required permission needed to bypass interlocks. IAW 1507-3, Limits and Precautions:

5.1.1 Interlocks that are bypassed for non-procedure driven events shall have the following requirements.

- Licensed Fuel Handling Supervisor (LFHS), Station Duty Manager and Operations Shift Manager have given permission.

C. Correct. These are the required personnel IAW 1507-3, Main Fuel Handling Bridge Operating Instructions. Plausible if the candidate is not familiar with the required permission needed to bypass interlocks. IAW 1507-3, Limits and Precautions:

> 5.1.1 Interlocks that are bypassed for non-procedure driven events shall have the following requirements.
> - Licensed Fuel Handling Supervisor (LFHS), Station Duty Manager and Operations Shift Manager have given permission.

D. Incorrect. These are not the required personnel IAW 1507-3, Main Fuel Handling Bridge Operating Instructions. Plausible if the candidate is not familiar with the required permission needed to bypass interlocks. IAW 1507-3, Limits and Precautions:

> 5.1.1 Interlocks that are bypassed for non-procedure driven events shall have the following requirements.
> - Licensed Fuel Handling Supervisor (LFHS), Station Duty Manager and Operations Shift Manager have given permission.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	2.1.41	
Importance Rating		3.7

K/A: Knowledge of the refueling process.

Proposed Question: SRO Question # 95

Technical Reference(s): 1507-3, pg 5, Rev 31A

Proposed References to be provided to applicants during examination: None

Learning Objective: 252-GLO-12

Question Source: Bank # SR5B16-03-Q01

Modified Bank #

New

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	
	55.43	7

Comments:
The KA is matched because the candidate must demonstrate knowledge of the refueling process.

The question is at the memory cognitive level because the candidate is required to have procedural knowledge of fuel handling activities to include licensed operator responsibilities.

The question is at the SRO level because the candidate must demonstrate knowledge of Refuel Floor SRO responsibilities. This is a job of the SRO only and therefore is not RO level knowledge. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 9).

21 **ID: 984425** **Points: 1.00**

Plant Conditions:

- The reactor is at 8% power and holding following isolation of an Instrument Air (IA) leak.
- The section of the IA header that feeds AH-E-29A, Diesel Generator Room "A" Fan Damper, has been isolated and completely depressurized to facilitate repair of a tubing failure.
- Operators have performed OP-TM-861-910, Emergency Ventilation of EG-Y-1A Room, blocking open the EG-Y-1A space Security Door and Fire Door.

Given the above information, which ONE of the following identifies:
(1) The current operability status of EG-Y-1A, and
(2) The requirements with respect to the doors blocked open for completion of OP-TM-861-910?

A. (1) OPERABLE.
(2) Compensatory actions for a blocked open Fire Door are required, ONLY.

B. (1) OPERABLE.
(2) Compensatory actions for a blocked open Security Door and a blocked open Fire Door are required.

C. (1) INOPERABLE.
(2) Compensatory actions for a blocked open Fire Door are required, ONLY.

D. (1) INOPERABLE.
(2) Compensatory actions for a blocked open Security Door and a blocked open Fire Door are required.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**
Part 1 Incorrect - Plausible if the candidate believes that opening the security and fire door for EG-Y-1A and supplying an emergency ventilation source will return EG-Y-1A to an operable status.

Part 2 Incorrect - This is plausible because there are two separate doors that must be opened, and the operator may incorrectly believe that only the fire door, which is closest to the Diesel Engine, need be open to establish emergency cooling of the room.

B. **Incorrect.**
Part 1 Incorrect - Plausible if the candidate believes that opening the security and fire door for EG-Y-1A and supplying an emergency ventilation source will return EG-Y-1A to an operable status.

Part 2 Correct - According to OP-TM-861-910, this procedure will open the EG-Y-1A Room doors D-106 and D-107. D-106 is a Security Door, and D-107 is a Fire Door. Because both doors are needed to be opened compensatory measures for a Security and Fire Impairment standpoint are required.

C. **Incorrect.**

Part 1 Correct - The dampers on AH-E-29A & B outlet will fail closed on loss of "out bldgs" IA. The diesel would operate if required. The doors can be opened IAW OP-TM-861-910, 911 if EG-Y-1A or B is operating. With the normal EDG room ventilation inoperable, EG-Y-1A & B will be declared inoperable and Tech Spec 3.7.2 action statement entered.

Part 2 Incorrect - This is plausible because there are two separate doors that must be opened, and the operator may incorrectly believe that only the fire door, which is closest to the Diesel Engine, need be open to establish emergency cooling of the room.

D. **Correct.**

Part 1 Correct - The dampers on AH-E-29A & B outlet will fail closed on loss of "out bldgs" IA. The diesel would operate if required. The doors can be opened IAW OP-TM-861-910, 911 if EG-Y-1A or B is operating. With the normal EDG room ventilation inoperable, EG-Y-1A & B will be declared inoperable and Tech Spec 3.7.2 action statement entered.

Part 2 Correct - According to OP-TM-861-910, this procedure will open the EG-Y-1A Room doors D-106 and D-107. D-106 is a Security Door, and D-107 is a Fire Door. Because both doors are needed to be opened compensatory measures for a Security and Fire Impairment standpoint are required.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.36	
	Importance Rating		4.2

K/A: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Proposed Question: SRO Question # 96

Technical Reference(s): OP-TM-AOP-028, pg 31, Rev 6

TS 1.3, pg 1-2, Rev 175

OP-TM-861-910, pg 1, Rev 1

Proposed References to be provided to applicants during examination: None

Learning Objective: 861-GLO-14

Question Source: Bank # IS-861-GLO-14-Q01

Modified Bank #

New

Question History:		Last NRC Exam:	NA
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	1	

Comments:

The KA is matched because the operator must demonstrate the ability to analyze the effect of maintenance activities due to the loss of IA, such as degraded power sources, on the status of limiting conditions for operations (i.e. by considering operability of the EDG with a failed damper in a support system and blocked open but normally closed doors that are compensatory actions resulting from isolation of an IA leak).

The question is at the Comprehension/Analysis cognitive level because the operator must evaluate the stated conditions, and determine a correct course of action from a procedure that is written to cover more than one failure mechanism. In doing so, the operator demonstrates that the procedure can be correctly applied, demonstrating understanding of the overall effect on plant operation.

The question is at the SRO level because the candidate must demonstrate knowledge of the administration of fire protection program requirements such as compensatory actions associated with inoperable sprinkler systems, fire doors, etc. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3). Additionally, the question is SRO-Only because it involves a TS operability determination with multiple conditions to consider, and recalling the strategy within a System operating procedure that would be implemented in an abnormal situation.

22 **ID: 984426** **Points: 1.00**

Plant Conditions:

- Tcold is 352°F.
- A plant Heatup is in progress IAW 1102-1, Plant Heatup to 525°F.
- The Plant Computer is NOT available.
- The following data was taken per Enclosure IV of the Plant Heat up Procedure:

TIME	TCOLD	TPZR
1730	334°F	430°F
1735	335°F	434°F
1740	337°F	438°F
1745	338°F	441°F
1750	340°F	445°F
1755	342°F	449°F
1800	358°F	460°F

Based upon the above data, the appropriate action that the CRS will direct at time 1800 is to ___(1)___ and the Tech Spec basis for that action is to ___(2)___.

A. (1) commence an 18 minute temperature and pressure hold
(2) avoid exceeding the stress limits for cyclic operations on RCS components

B. (1) commence an 18 minute temperature and pressure hold
(2) maintain the thermal stresses at the pressurizer spray line nozzle below the design limit

C. (1) lower the temperature difference between the RCS and Pressurizer to less than 100°F
(2) avoid exceeding the stress limits for cyclic operations on RCS components

D. (1) lower the temperature difference between the RCS and Pressurizer to less than 100°F
(2) maintain the thermal stresses at the pressurizer spray line nozzle below the design limit

Answer: A

Answer Explanation

A. Correct.
Part 1 is correct. Since a step change of >15°F from Time 1755 and 1800, a hold must be performed. IAW 1102-1 Limits and Precautions:

2. RCS heatup rate shall be limited to 50°F/Hr. A step change in RCS temp of up to 15°F is allowable if followed by an 18 minute temperature and pressure hold.

Additionally, IAW 1102-1, Enclosure 4:

- These requirements are for compliance with Tech. Spec. 3.1.2.
  1. Each 30 minutes, plot a point and record the time on the P/T curves (Figure 1 or 1A). If the plant computer calculated heatup rate is unavailable then complete Encl 4 Data Sheet every 5 min.

Part 2 is correct. IAW Tech Specs 3.1.2 Bases:

- All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes (Reference 1). These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation (Reference 2).

- The heatup and cooldown rate limits in this specification are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold) times.

Additionally, IAW TMI-1 UFSAR:

- Based on the predicted RTNDT after 29 effective full power years of operation, the pressure-temperature limits have been established in accordance with the requirements of 10CFR50, Appendix G, as described in Reference 25, and are based on the information obtained in the analysis of surveillance capsule TMI-1C, described in Reference 24.

- The heatup and cooldown rate limits shown in T.S. 3.1.2, Figure Nos. 3.1-1 and 3.1-2 are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold).

B. Incorrect.
Part 1 is correct. Since a step change of >15°F from Time 1755 and 1800, a hold must be performed. IAW 1102-1 Limits and Precautions:

2. RCS heatup rate shall be limited to 50°F/Hr. A step change in RCS temp of up to 15°F is allowable if followed by an 18 minute temperature and pressure hold.

Additionally, IAW 1102-1, Enclosure 4:

- These requirements are for compliance with Tech. Spec. 3.1.2.
  1. Each 30 minutes, plot a point and record the time on the P/T curves (Figure 1 or 1A). If the plant computer calculated heatup rate is unavailable then complete Encl 4 Data Sheet every 5 min.

Part 2 is incorrect. Plausible if the candidate confuses the bases for heatup rate with that of the differential temperature across the Pressurizer Spray Valve. IAW T.S. 3.1.2

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

- Bases:
  - The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit.

C. Incorrect.
Part 1 is incorrect. A step change of >15°F from Time 1755 and 1800, a hold must be performed. Plausible if the candidate performs a 30 minute average for the Heatup Rate. However, since the Plant Computer is out, a 5 minute Heatup Rate must be performed. IAW 1102-1 Limits and Precautions:

2. RCS heatup rate shall be limited to 50°F/Hr. A step change in RCS temp of up to 15°F is allowable if followed by an 18 minute temperature and pressure hold.

Additionally, IAW 1102-1, Enclosure 4:

- These requirements are for compliance with Tech. Spec. 3.1.2.
  1. Each 30 minutes, plot a point and record the time on the P/T curves (Figure 1 or 1A). If the plant computer calculated heatup rate is unavailable then complete Encl 4 Data Sheet every 5 min.

Part 2 is correct. IAW Tech Specs 3.1.2 Bases:

- All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes (Reference 1). These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation (Reference 2).

- The heatup and cooldown rate limits in this specification are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold) times.

Additionally, IAW TMI-1 UFSAR:

- Based on the predicted RTNDT after 29 effective full power years of operation, the pressure-temperature limits have been established in accordance with the requirements of 10CFR50, Appendix G, as described in Reference 25, and are based on the information obtained in the analysis of surveillance capsule TMI-1C, described in Reference 24.

- The heatup and cooldown rate limits shown in T.S. 3.1.2, Figure Nos. 3.1-1 and 3.1-2 are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold).

D. Incorrect.
Part 1 is incorrect. A step change of >15°F from Time 1755 and 1800, a hold must be performed. Plausible if the candidate performs a 30 minute average for the Heatup Rate. However, since the Plant Computer is out, a 5 minute Heatup Rate must be performed. IAW 1102-1 Limits and Precautions:

2. RCS heatup rate shall be limited to 50°F/Hr. A step change in RCS temp of up to 15°F is allowable if followed by an 18 minute temperature and pressure hold.

Additionally, IAW 1102-1, Enclosure 4:

- These requirements are for compliance with Tech. Spec. 3.1.2.
  1. Each 30 minutes, plot a point and record the time on the P/T curves (Figure 1 or 1A). If the plant computer calculated heatup rate is unavailable then complete Encl 4 Data Sheet every 5 min.

Part 2 is incorrect. Plausible if the candidate confuses the bases for heatup rate with that of the differential temperature across the Pressurizer Spray Valve. IAW T.S. 3.1.2

> 3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

- Bases:
  - The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	2	
K/A #	2.2.44	
Importance Rating		4.4

K/A: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question: SRO Question # 97

Technical Reference(s): TS 3.1, pg 3-3, Rev 278
1102-1, pg 3, 35, Rev 172C

Proposed References to be provided to applicants during examination: None

Learning Objective: GOP-001-PCO-3

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:
The KA is matched because the candidate must interpret control room indications associated with various systems to verify the status and operation of a Plant Heatup. Additionally, the candidate must demonstrate understanding of how operator actions and directives affect plant and system conditions.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze the current heat-up rates of the RCS and Pressurizer and calculated 5 minute rates. The candidate must also determine if the heat-up rates exceed the procedural limits and if so, the correct action to take to mitigate the event.

The question is at the SRO level because the candidates must provide knowledge of Tech Spec bases that are required to analyze TS required actions and terminology. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

23 **ID: 984427** **Points: 1.00**

Plant Conditions:

- The Reactor is operating at 30% power.
- A female Exelon employee has 1970 mr TEDE exposure for the current calendar year.

Event:

- She is assigned to a **5 hour** task that will result in approximately **225 mr/hr** additional TEDE exposure.
- The task is to be performed in the Auxiliary Building.
- Her work group supervisor has authorized her participation in the work.

Given the above information, which ONE of the following describes:
(1) The additional authorization **required** for this exposure in accordance with RP-AA-203, Exposure Limits and Controls, and
(2) The additional authorization **required** for this exposure in accordance with RP-TM-460-1007, Access to TMI-1 Reactor Building, **IF** the task was inside the gates on top of the "A" D-ring instead?

A. (1) Radiation Protection Manager, ONLY.
(2) Shift Manager.

B. (1) Radiation Protection Manager, ONLY.
(2) Site Vice President.

C. (1) Radiation Protection Manager and Plant Manager.
(2) Shift Manager.

D. (1) Radiation Protection Manager and Plant Manager.
(2) Site Vice President.

Answer: C

Answer Explanation

A. Incorrect.
Part 1 is incorrect. 225 mrem/hr times 5 hours equals 1125 mrem. 1125 mrem plus 1970 mrem equals 3095 mrem. RP Manager written approval is required for 2500 – 3000 mr. Plausible if the candidate adds 225 mrem to the beginning TEDE and does not multiply the 225 by the 5 hours. IAW RP-AA-203, Exposure Limits and Controls, Section 4.2.6:

> 4.2.6. To raise the ADCL up to and including 3000 mrem TEDE in a calendar year, written approval is required by the Radiation Protection Manager and the work group supervisor.

Part 2 is correct. IAW RP-TM-460-1007, Access to TMI-1 Reactor Building, Section 3.3: Precautions:

- The approval of the Shift Manager, Plant Manager, Radiation Protection Manager or designees, are required for entries within secondary shielding, inside the gates on top of the "D" rings, or within the Fuel Transfer Canal (including the reactor head area and along the east and west walkways adjacent to the Fuel Transfer Canal) when reactor power level is 1 percent or greater. The Polar Crane is considered as being included on the top of the D-Ring area. Approvals will be recorded on Attachment 6. Each individual will sign Attachment 6 (Approval for D-Ring Entries at Power) or give permission via telephone. The completed form will be filed with the original ALARA plan.

B. Incorrect.
Part 1 is incorrect. 225 mrem/hr times 5 hours equals 1125 mrem. 1125 mrem plus 1970 mrem equals 3095 mrem. RP Manager written approval is required for 2500 – 3000 mr. Plausible if the candidate adds 225 mrem to the beginning TEDE and does not multiply the 225 by the 5 hours. IAW RP-AA-203, Exposure Limits and Controls, Section 4.2.6:

> 4.2.6. To raise the ADCL up to and including 3000 mrem TEDE in a calendar year, written approval is required by the Radiation Protection Manager and the work group supervisor.

Part 2 is incorrect. Plausible if the candidate is not familiar with the requirements for RB entry while at power. IAW RP-TM-460-1007, Access to TMI-1 Reactor Building, Section 3.3: Precautions:

- The approval of the Shift Manager, Plant Manager, Radiation Protection Manager or designees, are required for entries within secondary shielding, inside the gates on top of the "D" rings, or within the Fuel Transfer Canal (including the reactor head area and along the east and west walkways adjacent to the Fuel Transfer Canal) when reactor power level is 1 percent or greater. The Polar Crane is considered as being included on the top of the D-Ring area. Approvals will be recorded on Attachment 6. Each individual will sign Attachment 6 (Approval for D-Ring Entries at Power) or give permission via telephone. The completed form will be filed with the original ALARA plan.

C. Correct.
Part 1 is correct. Plant Manager written approval is required for 3000 – 4000 mr. 225 mrem/hr times 5 hours equals 1125 mrem. 1125 mrem plus 1970 mrem equals 3095 mrem. IAW RP-AA-203, Exposure Limits and Controls, Section 4.2.7:

> 4.2.7. To raise the ADCL to between 3001 and 4000 mrem TEDE in a calendar year, written approval is required by the Radiation Protection Manger, a work group supervisor, and the Station/Plant Manager.

Part 2 is correct. IAW RP-TM-460-1007, Access to TMI-1 Reactor Building, Section 3.3: Precautions:

- The approval of the Shift Manager, Plant Manager, Radiation Protection Manager or designees, are required for entries within secondary shielding, inside the gates on top of the "D" rings, or within the Fuel Transfer Canal (including the reactor head area and along the east and west walkways adjacent to the Fuel Transfer Canal) when reactor power level is 1 percent or greater. The Polar Crane is considered as being included on the top of the D-Ring area. Approvals will be recorded on Attachment 6. Each individual will sign Attachment 6 (Approval for D-Ring Entries at Power) or give permission via telephone. The completed form will be filed with the original ALARA plan.

D. Incorrect.
Part 1 is correct. Plant Manager written approval is required for 3000 – 4000 mr. 225 mrem/hr times 5 hours equals 1125 mrem. 1125 mrem plus 1970 mrem equals 3095 mrem. IAW RP-AA-203, Exposure Limits and Controls, Section 4.2.7:

4.2.7. To raise the ADCL to between 3001 and 4000 mrem TEDE in a calendar year, written approval is required by the Radiation Protection Manger, a work group supervisor, and the Station/Plant Manager.

Part 2 is incorrect. Plausible if the candidate is not familiar with the requirements for RB entry while at power. IAW RP-TM-460-1007, Access to TMI-1 Reactor Building, Section 3.3: Precautions:

- The approval of the Shift Manager, Plant Manager, Radiation Protection Manager or designees, are required for entries within secondary shielding, inside the gates on top of the "D" rings, or within the Fuel Transfer Canal (including the reactor head area and along the east and west walkways adjacent to the Fuel Transfer Canal) when reactor power level is 1 percent or greater. The Polar Crane is considered as being included on the top of the D-Ring area. Approvals will be recorded on Attachment 6. Each individual will sign Attachment 6 (Approval for D-Ring Entries at Power) or give permission via telephone. The completed form will be filed with the original ALARA plan.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.4	
	Importance Rating		3.7

K/A: Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: SRO Question # 98

Technical Reference(s): RP-AA-203, pg 4, Rev 3
RP-TM-460-1007, pg 4, Rev 6

Proposed References to be provided to applicants during examination: None

Learning Objective: N-TM-TQ-104-NOP-DBIG-APCO-1

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: N/A

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	4

Comments:
The KA is matched because the candidate must demonstrate knowledge of radiation exposure limits under normal or emergency conditions.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to calculate stay times and total dose that will be received by the employee and based on the calculation determine the type of dose extension and the required approvals.

The question is at the SRO level because the candidate must demonstrate analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures. This is a job of the SRO only and therefore is not RO level knowledge. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 6).

24 **ID: 984428** **Points: 1.00**

According to RP-AA-1004, Radiation Protection Stop Work Authority and Corporate RPM Event Notification, which ONE of the following identifies an event, or events, that require(s) the Shift Manager to conduct a prompt investigation in accordance with OP-AA-106-101-1001, Event Response Guidelines?

A. An individual worker exceeds their RWP dose limit by 28 mrem, ONLY.

B. Three unplanned personal contamination events ranging from 10-20K cpm each, occur for a specific job during the course of one shift, ONLY.

C. An unplanned Radiation Worker uptake 7 mrem AND loose surface contamination in a controlled and contained area is spread about 800 ft^2 beyond its boundary.

D. An unplanned Radiation Worker uptake of 7 mrem AND three unplanned personal contamination events ranging from 10-20K cpm each, occur for a specific job during the course of one shift.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.** According to RP-AA-1004 (p3; Rev 8), Step 3.5, the Shift Manager must initiate a prompt investigation in accordance with OP-AA-106-101-1001 for all events that require a four hour notification of the Corporate RPM. According to RP-AA-1004 (p4; Rev 8), Attachment 1, the Administrative Exposure Event threshold for a four hour notification is > 25 mrem DDE over the RWP limit. Since the stated event has resulted in 28 mrem over the RWP limit, the four hour notification threshold has been exceeded.

B. **Incorrect.** This is plausible because the operator may misinterpret the Attachment, and incorrectly determines that Personnel Contamination Events four hour reporting threshold has been exceeded. However, according to RP-AA-1004 (p4; Rev 8), Attachment 1, the Personnel Contamination Events for a four hour notification is calculated SDE > 1 Rem or >3 Unplanned PCEs in a single job during one shift. Since the stated event has resulted in only three PCEs. In fact, this number of PCEs doesn't even arise to the 12 hour reporting limit.

C. **Incorrect.** This is plausible if the operator incorrectly believes that both four hour and 12 hour notifications to the Corporate RPM result in requiring a prompt investigation. According to RP-AA-1004 (p4; Rev 8), Attachment 1, the Unplanned Internal Contamination Event threshold for a four hour notification is > 10 mrem unplanned uptake. Since the stated event has resulted in 230 nanocuries, the four hour notification threshold has NOT been exceeded. Additionally, according to RP-AA-1004 (p4; Rev 8), Attachment 1, the Loss of Contamination Control Event threshold for a four hour notification is a spread of contamination > 1000 ft^2. Since the stated event has resulted in a spread of only 800 ft^2, the four hour notification threshold has NOT been exceeded. However, the Loss of Contamination Control Event threshold for a 12 hour notification is a spread of contamination > 500 but < 1000 ft^2 and therefore, the 12 hour threshold has been exceeded.

D. **Incorrect.** This is plausible if the operator misinterprets the Attachment, and incorrectly determines that Personnel Contamination Events four hour reporting threshold has been exceeded (See B). Under this situation the operator would determine that both events result in exceeding the four hour reporting threshold (See C) and require a prompt investigation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.14	
	Importance Rating		3.8

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: SRO Question # 99

Technical Reference(s): RP-AA-1004, pg1-4, Rev 8

Proposed References to be provided to applicants during examination: RP-AA-1004 (p4of5; Rev 8) Attachment 1

Learning Objective: ADM08005

Question Source: Bank # IS-OP-AA-RPT-Q07

Modified Bank #

New

Question History: NA Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 4

Comments:

The KA is matched because the operator must demonstrate knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities (i.e. that events requiring a four hour report to the Corporate RPM require a prompt investigation by the Shift Manager).

The question is at the Comprehension/Analysis cognitive level because the operator must evaluate the stated conditions, and compare these conditions to an Attachment to determine whether or not reporting thresholds have been exceeded. Once determined, the operator must apply knowledge regarding their job responsibility and draw a conclusion regarding action that must be taken. In doing so, the operator demonstrates that the procedure can be correctly applied, demonstrating understanding of the overall effect on plant operation.

The question is at the SRO level because the candidate must demonstrate analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures. This is a job of the SRO only and therefore is not RO level knowledge. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 6).

25 **ID: 984429** **Points: 1.00**

Plant Conditions:

- The Plant is in a refueling outage.

Sequence of Events:

- Time = 1110
  - An unplanned valid reading on RM-G-7 is identified.
- Time = 1115
  - The Shift Emergency Director classifies the event as an UNUSUAL EVENT.
- Time = 1119
  - Prior to state and local notifications being made for the UNUSUAL EVENT, conditions change such that the Shift Emergency Director upgrades the classification to an ALERT.
  - The Shift Emergency Director informs the Communicator to make notifications of the ALERT, ONLY.

Given the above information, which one of the following identifies:
(1) The latest time that the state and local notifications must be made, and
(2) What action is required to stop the clock?

A. (1) 1130.
(2) After completion of the initial roll call.

B. (1) 1130.
(2) After completion of the follow-up notifications.

C. (1) 1134.
(2) After completion of the initial roll call.

D. (1) 1134.
(2) After completion of the follow-up notifications.

Answer: A

Answer Explanation

A. Correct.
Part 1 is Correct - IAW EP-AA-111, Emergency Classification and Protective Action Recommendations:

- If a higher classification is made prior to transmitting an event notification, then notification for the higher classification can supersede the previous event notification, provided that it can be performed within the 15 minute timeframe of the previous event.
- If the notification of a higher classification cannot be performed within the 15-minute timeframe of the previous event, the previous event notification is required within its 15 minute timeframe, and the subsequent event notification is required within its 15-minute timeframe.

Part 2 is Correct - IAW EP-MA-114-100, Mid-Atlantic State / Local Notifications:

4.2.3 Completion of the initial Roll Call (contact made via dedicated or commercial line with agencies listed) must be performed within 15 minutes of initial classification, re-classification, or PAR change.

B. Incorrect.
Part 1 is Correct - IAW EP-AA-111, Emergency Classification and Protective Action Recommendations:

- If a higher classification is made prior to transmitting an event notification, then notification for the higher classification can supersede the previous event notification, provided that it can be performed within the 15 minute timeframe of the previous event.
- If the notification of a higher classification cannot be performed within the 15-minute timeframe of the previous event, the previous event notification is required within its 15 minute timeframe, and the subsequent event notification is required within its 15-minute timeframe.

Part 2 is Incorrect - Plausible if the candidate believes that all agencies must acknowledge the message prior to 15 minutes following the declaration. York Haven Power Station is a Follow-Up Notification that is not required to be notified within 15 minutes. IAW EP-MA-114-100-F-01:

- Follow-Up Notifications* (TMI)
- * NOT required within 15 minutes of Classification.

C. Incorrect.
Part 1 is Incorrect -Plausible if the candidate believes the state and local notification can be made 15 minutes following the escalation since it occurred prior to making the notification for the initial declaration.

Part 2 is Correct - IAW EP-MA-114-100, Mid-Atlantic State / Local Notifications:

4.2.3 Completion of the initial Roll Call (contact made via dedicated or commercial line with agencies listed) must be performed within 15 minutes of initial classification, re-classification, or PAR change.

D. Incorrect.
Part 1 is Incorrect - Plausible if the candidate believes the state and local notification can be made 15 minutes following the escalation since it occurred prior to making the notification for the initial declaration.

Part 2 is Incorrect - Plausible if the candidate believes that all agencies must acknowledge the message prior to 15 minutes following the declaration. York Haven Power Station is a Follow-Up Notification that is not required to be notified within 15 minutes. IAW EP-MA-114-100-F-01:

- Follow-Up Notifications* (TMI)
- * NOT required within 15 minutes of Classification.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.40	
	Importance Rating		4.5

K/A: Knowledge of SRO responsibilities in emergency plan implementation.

Proposed Question:	SRO Question # 100		
Technical Reference(s):	EP-MA-114-100, pg 8, Rev 19 EP-MA-114-100-F-01, pg 3, Rev N EP-AA-111, pg 5, Rev 18		

Proposed References to be provided to applicants during examination: None

Learning Objective: EP101007

Question Source:	Bank #		
	Modified Bank #	IR-EP101007-Q04	(Note changes or attach parent)
	New		

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	
	55.43	7

Comments:
The KA is matched because the candidate must demonstrate knowledge of SRO responsibilities in emergency plan implementation.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze plant conditions to determine the times for each classification and based on knowledge of the EP procedures, calculate the times required for the notifications to be completed and what actions need to be performed by the communicator in order to meet the notification times.

The question is at the SRO level because the candidate must demonstrate reporting requirements This is a job of the SRO only and therefore is not RO level knowledge. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 9).

Original Question: IR-EP101007-Q04

Plant conditions:
- An OTSG tube rupture has resulted in a reactor trip and ESAS actuation.
- The Shift Emergency Director initially classified the event as an ALERT 1115 hours.
- Local notifications of the ALERT have been made.
- The Shift Emergency Director upgraded the classification to SITE AREA EMERGENCY at 1148.

Which ONE of the following identifies the latest time that the upgrade notification to required local agencies can be made and what action stops the clock?

A. 1203; after completion of the initial roll call.

B. 1203; after all required contacts have acknowledged the message.

C. 1248; after completion of the initial roll call.

D. 1248; after all required contacts have acknowledged the message.

Answer: A