1 **ID: 1142054** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- Both Main Feedwater Pumps trip.
- The automatic reactor trip fails.
- The automatic turbine trip fails.

Given the above information and assuming that the reactor remains untripped, Tcold will ___(1)___ and Power Range Nuclear Instrumentation will indicate ___(2)___ than actual Core Thermal Power.

A. (1) rise
(2) lower

B. (1) rise
(2) higher

C. (1) lower
(2) lower

D. (1) lower
(2) higher

Answer: B

Answer Explanation

A. Incorrect.

Part 1 is correct. IAW TQ-TM-104-401-C001, Main Feedwater System:

- The MFW System does not have a substantial impact on plant safety. However, at full power, the plant operation cannot be sustained without full power capacity of the Main Feedwater System. Loss of feedwater will cause the reactor plant to overheat/trip.

A Loss of Main Feedwater is addressed in the FSAR. IAW TMI-1 UFSAR:

> 14.2.2.7 Loss Of Feedwater Accident: A loss of feedwater may result from abnormal closure of the feedwater isolation valves, control valve failure, or pump failure. The loss of feedwater flow results in a loss of heat sink, primary system heatup, increased pressurizer level and pressure, and reactor trip on high RCS pressure.

Emergency Feedwater will initate on the Loss of Both Main Feedwater signal. IAW TQ-TM-104-424, Emergency Feedwater System:

a. EFW is auto initiated for the following HSPS signals
  1) Loss of both Main Feedwater Pumps
  2) Loss of all Reactor Coolant Pumps (RCPs)
  3) 4 PSIG RB pressure
  4) Low OTSG level (10 inches S/U range)

Although Emergency Feedwater initates, it will only maintain the OTSG's to a level of 25" in the Startup Range. IAW TQ-TM-104-644, Heat Sink Protection System:

3) EFW Valve Control
  a) All EF-V-30's receive a 0 inch setpoint when Heat Sink Protection System is not actuated.
  b) If Heat Sink Protection System is actuated the valves control at the following setpoints:
    (3) If initiated on trip of both MFW pumps, then all valves control at 25 inches on the Startup Range level.

Typically at 100% Reactor Power, the OTSG levels are approximately 75% in the Operating Range. Therefore, there is not enough OTSG inventory to remove the heat being generated from the reactor.

Part 2 is incorrect. IAW OPM N-06, Reactor Theory:

- If the moderator temperature increases, the volume occupied by the moderator will increase and the density of moderator atoms N exposed to the fast flux will decrease. A moderator density decrease (moderator temperature increase or moderator to fuel ratio decrease) will decrease the Macroscopic Cross Section for Absorption, and the Macroscopic Cross Section for Scattering of the moderator due to the drop in atom density. This will result in an increase in neutron slowing down length, Fermi Age, slowing down time, and a decrease in absorption within the moderator resulting in an increase in fast neutron leakage.

Plausible if the candidate believes that the Nuclear Instruments are located within the core or if the candidate believes that a higher moderator temperature will have a better effect of slowing down neutrons.

B. Correct.
Part 1 is correct. IAW TQ-TM-104-401-C001, Main Feedwater System:

- The MFW System does not have a substantial impact on plant safety. However, at full power, the plant operation cannot be sustained without full power capacity of the Main Feedwater System. Loss of feedwater will cause the reactor plant to overheat/trip.

A Loss of Main Feedwater is addressed in the FSAR. IAW TMI-1 UFSAR:

> 14.2.2.7 Loss Of Feedwater Accident: A loss of feedwater may result from abnormal closure of the feedwater isolation valves, control valve failure, or pump failure. The loss of feedwater flow results in a loss of heat sink, primary system heatup, increased pressurizer level and pressure, and reactor trip on high RCS pressure.

Emergency Feedwater will initate on the Loss of Both Main Feedwater signal. IAW TQ-TM-104-424, Emergency Feedwater System:

a. EFW is auto initiated for the following HSPS signals
   1) Loss of both Main Feedwater Pumps
   2) Loss of all Reactor Coolant Pumps (RCPs)
   3) 4 PSIG RB pressure
   4) Low OTSG level (10 inches S/U range)

Although Emergency Feedwater initates, it will only maintain the OTSG's to a level of 25" in the Startup Range. IAW TQ-TM-104-644-C001, Heat Sink Protection System:

3) EFW Valve Control
   a) All EF-V-30's receive a 0 inch setpoint when Heat Sink Protection System is not actuated.
   b) If Heat Sink Protection System is actuated the valves control at the following setpoints:
      (3) If initiated on trip of both MFW pumps, then all valves control at 25 inches on the Startup Range level.

Typically at 100% Reactor Power, the OTSG levels are approximately 75% in the Operating Range. Therefore, there is not enough OTSG inventory to remove the heat being generated from the reactor.

Part 2 is correct. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

Detector Locations
   a. External to the core.
   b. Measure fast neutrons that leak out of core, and are thermalized in the concrete of the biological shield.
   d. One Power Range detector string in each quadrant

IAW OPM N-06, Reactor Theory:

> If the moderator temperature increases, the volume occupied by the moderator will increase and the density of moderator atoms N exposed to the fast flux will decrease. A moderator density decrease (moderator temperature increase or moderator to fuel ratio decrease) will decrease the Macroscopic Cross Section for Absorption, and the Macroscopic Cross Section for Scattering of the moderator due to the drop in atom density. This will result in an increase in neutron slowing down length, Fermi Age, slowing down time, and a decrease in absorption within the moderator resulting in an increase in fast neutron leakage.

C. Incorrect.
Part 1 is incorrect. See explanation for choice A.

Plausible if the candidate believes that Emergency Feedwater will be sufficient to remove the heat being generated by the reactor.

Part 2 is incorrect. IAW OPM N-06, Reactor Theory:

> If the moderator temperature increases, the volume occupied by the moderator will increase and the density of moderator atoms N exposed to the fast flux will decrease. A moderator density decrease (moderator temperature increase or moderator to fuel ratio decrease) will decrease the Macroscopic Cross Section for Absorption, and the Macroscopic Cross Section for Scattering of the moderator due to the drop in atom density. This will result in an increase in neutron slowing down length, Fermi Age, slowing down time, and a decrease in absorption within the moderator resulting in an increase in fast neutron leakage.

Plausible if the candidate believes that the Nuclear Instruments are located within the core or if the candidate believes that a higher moderator temperature will have a better effect of slowing down neutrons.

D. Incorrect.
Part 1 is incorrect. See explanation for choice A.

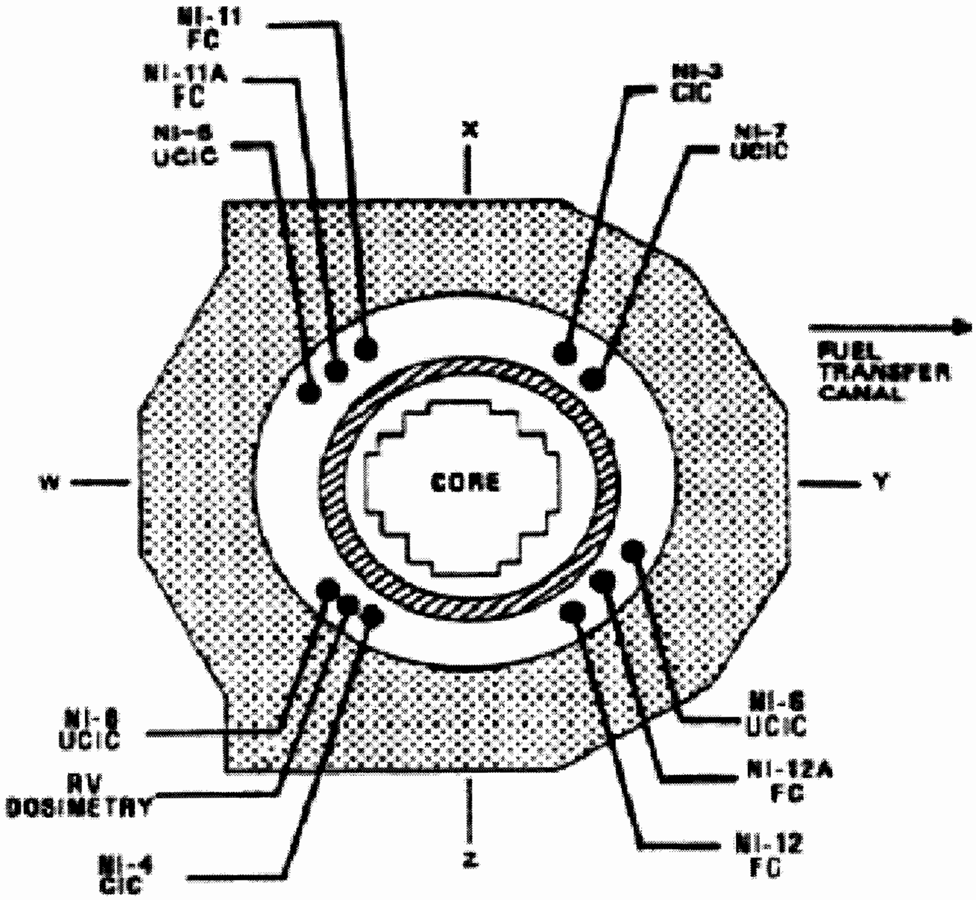
Plausible if the candidate believes that Emergency Feedwater will be sufficient to remove the heat being generated by the reactor.

Part 2 is correct. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

> Detector Locations
> - a. External to the core.
> - b. Measure fast neutrons that leak out of core, and are thermalized in the concrete of the biological shield.
> - d. One Power Range detector string in each quadrant

IAW OPM N-06, Reactor Theory:

> If the moderator temperature increases, the volume occupied by the moderator will increase and the density of moderator atoms N exposed to the fast flux will decrease. A moderator density decrease (moderator temperature increase or moderator to fuel ratio decrease) will decrease the Macroscopic Cross Section for Absorption, and the Macroscopic Cross Section for Scattering of the moderator due to the drop in atom density. This will result in an increase in neutron slowing down length, Fermi Age, slowing down time, and a decrease in absorption within the moderator resulting in an increase in fast neutron leakage.



CIC COMPENSATED ION CHAMBER - INTERMEDIATE RANGE DETECTOR

UCIC UNCOMPENSATED ION CHAMBER - POWER RANGE DETECTOR

FC FISSION CHAMBER - WIDE RANGE DETECTOR

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	029	EK1.01	
	Importance Rating	2.8		

Knowledge of the operational implications of the following concepts as they apply to the ATWS: Reactor nucleonics and thermo-hydraulics behavior.

Proposed Question:	RO Question # 1
Technical Reference(s):	TQ-TM-623-C001, p 9, Rev 4 TQ-TM-644-C001, p 29-30, Rev 2 TQ-TM-424-C001, p 5-6, Rev 10 TQ-TM-104-401-C001, pg 95, Rev 8 TMI-1 UFSAR, pg 14.2-43, Rev 22 OPM N-06, p 51, Rev 6

Proposed References to be provided to applicants during examination: None

Learning Objective: 401-GLO-11

Question Source:	Bank #	ID# 714454
	Modified Bank #	
	New	

Question History: Last NRC Exam: ILT 05-1 (2007)

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

10 CFR Part 55 Content:	55.41	2
	55.43	

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of reactor nucleonics and thermo-hydraulic behavior associated with an ATWS.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the conditions to determine the cause-effect relationship between secondary and primary components.

What MUST be known:
1. The physical location of Power Range Nuclear Instruments at Three Mile Island. 2. How a colder moderator affects neutron leakage out of the reactor vessel. 3. What the end result for reactor coolant temperatures will be upon a loss of Main Feedwater.

2 **ID: 1142290** **Points: 1.00**

Initial Conditions:

- The plant is operating at 100% reactor power.
- The reactor is at the End of Life portion of the cycle.

Sequence of Events:

- A Steam Line Rupture occurs in the Reactor Building from the "A" OTSG.
- A Reactor Trip occurs.
- The Reactor Coolant Pumps have been secured.
- The reactor has returned to a Critical state (the reactor is critical).

Given the above information and IAW the TMI-1 UFSAR, natural circulation:

A. Will not exist because steam generator tube failures will occur due to cross-flow steam velocity pressure loads.

B. Will exist but may not be sufficient to remove heat and avoid core damage due to the recriticality of the reactor.

C. Will not exist because there will be a large amount of sustained void formations in the hot legs due to tripped RCP's.

D. Will exist and will be sufficient to remove heat and avoid core damage due to the rupture occurring at EOL conditions.

Answer: B

Answer Explanation

A. Incorrect.
IAW TMI-1 UFSAR, Chapter 14, Pages 14.1-30/31:

> The second reactor protection criterion for the MSLB event states that the steam generator tubes shall not fail as a result of the loss of secondary side pressure or the resultant temperature gradients. The analyses performed for the replacement OTSG unflawed and flawed tube scenarios considered the pressure differential load, axial load, and tensile loads resulting from the MSLB conditions predicted in Reference 130.
>
> The analyses performed for the unflawed tube considered structural loadings based on normal, upset, emergency, and faulted transients, including the main steam line break accident. The results of these analyses (References 133, 135, and 136) show that the tube design meets the requirements of the ASME code for the design loadings identified in the TMI-1 Functional Specification. Structural analyses that considered various transient loadings were also performed to determine allowable flaw configurations based on the structural integrity requirements outlined in current NEI 97-06 documents. The loads evaluated for the replacement OTSG included normal operating transients at 100 percent power as well as conservative accident (faulted) conditions for the MSLB and small break loss-of-coolant accidents (SBLOCAs) (Reference 134). The MSLB defines the limiting primary-secondary pressure differential load as well as the limiting bending loads. The limiting tube axial loads occur during the SBLOCA and bound those of the MSLB. Since the MSLB and SBLOCA tube loads do not result in tube failure (Reference 134), a flawed steam generator tube will not fail as a result of a steam line break as analyzed for limiting core response.
>
> In addition to the evaluation of the tensile, compressive, and pressure loads on flawed and unflawed tubes, evaluations of the potential for tube failures caused by high cross flow velocities were also performed (Reference 135). These evaluations assess the effects of the high cross flow loading for stress on the steam generator tubes during a steam line break. These evaluations conclude that steam generator tube failures will not occur as a result of cross-flow steam velocities during a steam line break accident with the replacement OTSGs.

Plausible if the candidate is not familiar with the FSAR analysis data for a Steam Line Break.

B. Correct.
IAW TMI-1 UFSAR, Chapter 14, Page 14.1-31:

> A more general concern exists with a large steam line break at EOL conditions and whether or not a return-to-power is experienced following the Reactor Coolant pump trip. If a return-to-criticality is experienced, natural circulation flow may not be sufficient to remove heat and to avoid core damage. B&W has analyzed and concluded that the subcritical return to power condition is bounded by the case where RCPs are not tripped. The tripped pump case results in a reduced overcooling effect.

C. Incorrect.
IAW TMI-1 UFSAR, Chapter 14, Page 14.1-31:

> The analysis of pump trip in a large steam line break has been performed in Reference 8. The assessment of consequences of an imposed reactor coolant pump trip, upon initiation of the low reactor coolant pressure ESAS, was made there. In the event of a large steam line break (maximum overcooling) the blowdown may induce a steam bubble in the Reactor Coolant System. The maximum overcooling case has been analyzed by B&W, the Reactor Coolant Pump trip increased the amount of void formation in the hot leg "candy cane" of the pressurizer loop, however, natural circulation was not completely blocked. The steam bubble was collapsed and full natural circulation was restored. Core cooling was maintained throughout the transient and no void formation occurred in the core.

Plausible if the candidate is not familiar with the FSAR analysis data for a Steam Line Break.

D. Incorrect.
IAW TMI-1 UFSAR, Chapter 14, Page 14.1-31:

> A more general concern exists with a large steam line break at EOL conditions and whether or not a return-to-power is experienced following the Reactor Coolant pump trip. If a return-to-criticality is experienced, natural circulation flow may not be sufficient to remove heat and to avoid core damage. B&W has analyzed and concluded that the subcritical return to power condition is bounded by the case where RCPs are not tripped. The tripped pump case results in a reduced overcooling effect.

Plausible if the candidate is not familiar with the FSAR analysis data for a Steam Line Break.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	040	AK1.06
	Importance Rating	3.7	

Knowledge of the operational implications of the following concepts as they apply to the Steam Line Rupture: High-energy steam line break considerations.

Proposed Question: RO Question # 2

Technical Reference(s): TMI-1 UFSAR, pg 14.1-31, Rev 22

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP003-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	14
	55.43	

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the high-energy steam line break considerations as assumed and analyzed within the Unit Final Safety Analysis Report.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall bits of information from a documented source.

What MUST be known:
1. IAW FSAR analysis, how effective will natural circulation be at the end-of-life given that the reactor has returned to a critical state?

3 **ID: 1142077** **Points: 1.00**

Plant Conditions:

- The Plant is operating at 100% reactor power.

Sequence of Events:

- Reactor trips due to a Steam Line Rupture in Containment.
- HPI has actuated due to a severe RCS cooldown rate.
- Currently:
  - Containment Pressure is 2 psig and lowering slowly.
  - Turbine Bypass Valves, MS-V-3A, 3B & 3C are set at 10% open in manual control.
  - OTSG 1A has been isolated IAW OP-TM-EOP-010, Rule 3 and is DRY.
  - OTSG 1B pressure is 850 psig and lowering slowly.
  - OTSG 1B level is 50% OPERATE range and relatively steady.
  - OTSG Tube-to-Shell Delta-T is + 20°F and rising very slowly.
  - HPI has been terminated.
  - RCS T_{cold} is 515°F and rising very slowly (both loops).
  - RCS T_{hot} is 517°F and rising very slowly (both loops).
  - RCS pressure is 1100 psig and rising very slowly.

Given the above information, select the FIRST applicable procedure to be implemented that requires action to be taken by an operator.

A. Guide 11, Cooldown Rate Limits.

B. Guide 12, RCS Stabilization following OTSG Isolation.

C. Guide 13, Feeding a Dry or Depressurized OTSG.

D. Guide 14, Tube-to-Shell Delta-T Limit/Control.

Answer: B

Answer Explanation

A. Incorrect.

Guide 11 is not referenced in OP-TM-EOP-001, Reactor Trip, or in OP-TM-EOP-003, Excessive Primary to Secondary Heat Transfer. Additionally, Guide 11 is for information purposes only, stating what the cooldown rate is. Rule 4 would direct action associated with not being within Guide 11 limits. Lastly, Guide 11 would show that the cooldown rate is 100F/hr. With a slight heatup rate given in the stem, the cooldown rate is not being violated and no immediate action must be taken. Plausible if the candidate believes that action must be taken to maintain the cooldown rate.

B. Correct.

To answer/understand this question, a complete analysis of the conditions given along with what procedures have already been entered must be understood. The routing for a steam line rupture would be as follows: OP-TM-EOP-001, Reactor Trip through the Immediate Manual Actions. The symptom check would then lead to OP-TM-EOP-003, Excessive Primary to Secondary Heat Transfer. The Immediate Manual Actions include Rule 3 and Guide 9. The first follow-up step is to perform Guide 12 if temperature reduction has been terminated, as is the case given in the stem. As a backup, Rule 3 also provides direction to initiate Guide 12.

Rule 3 actions are as follows:

1. VERIFY OTSG level < 97.5%.
2. VERIFY primary to secondary heat transfer is excessive.
3. PERFORM Phase 1 Isolation of the affected OTSG(s).
4. PERFORM Phase 2 Isolation of the affected OTSG(s). GO TO Step 7.
7. INITIATE Guide 12, "RCS Stabilization".

Guide 12, RCS Stabilization following OTSG Isolation entry criteria is as follows:

- When directed by EOP-001, EOP-005 or Rule 3, then perform the following:

C. Incorrect.

Guide 13, Feeding a Dry or Depressurized OTSG, entry criteria is as follows:

- Any one of the following:
  - It is anticipated that Tube to Shell delta T limits will be exceeded, or
  - Neither OTSG is available, or
  - Single Loop Natural Circulation cooldown where the Thot on the loop with the isolated OTSG is more than 50 degF hotter than the active loop Thot or the subcooling margin in the hot leg of the isolated OTSG is < 25 degF

Tube to Shell delta T is neither at the limits nor near the point of exceeding the limits.
The "B" OTSG is available.
The Thots are equal in both loops.

Therefore, the entry criteria are not met. Plausible if the candidate is not familiar with the entry criteria.

D. Incorrect.
Guide 14, Tube-to-Shell Delta-T Limit/Control entry criteria is as follows:

- IAAT the reactor is shutdown, then perform the following:

Guide 14 will be entered, but will NOT be performed since none of the limits have been exceeded in the given conditions (-70 tensile limit, +50 compressive limit)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E10	EK1.2
	Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as they apply to the Post-Trip Stabilization: Normal, abnormal, and emergency operating procedures associated with (Post-Trip Stabilization).

Proposed Question: RO Question # 3

Technical Reference(s): OP-TM-EOP-010, pg 6, 24, Rev 17

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP001-PCO-4

Question Source: Bank # ID# 303177

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	

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the emergency operating procedures associated with stabilizing the plant post-trip.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the given information, utilize steam tables, and determine which procedure is applicable.

What MUST be known:
1. Has Subcooling Margin been lost, as analyzed using Steam Tables?
2. What are the conditions required for natural circulation to occur?
3. What are the tube-to-shell Delta-T limits?
4. What are the entry criteria for Guides 11/12/13/14?

4 **ID: 1149467** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- No evolutions are in progress.

Event:

- A Pressurizer steam space leak occurs through a Pressurizer Safety Valve.
- PPC point A0459, RC Drain Tank WDL-T-3 Temp, is in alarm.

Given the above information, MU-V-17, Normal Makeup to RCS Control Valve, will travel to full ___(1)___ and WDL-P-8, Reactor Coolant Drain Tank Pump, ___(2)___ running.

A. (1) open
(2) is

B. (1) open
(2) is NOT

C. (1) closed
(2) is

D. (1) closed
(2) is NOT

Answer: C

Answer Explanation

A. Incorrect.
Part 1 is incorrect. Plausible if the candidate believes that Pressurizer level lowers due to a loss of inventory.

Part 2 is correct. WDL-P-8 will automatically start when the Reactor Coolant Drain Tank temperature reaches 110°F. Since PPC Alarm A0459 comes in to alarm at a Reactor Coolant Drain Tank temperature of >120°F, the candidate must realize that WDL-P-8 has started. IAW OP-TM-PPC-A0459, RC Drain Tank WDL-T-3 Temp:

1.0 SETPOINTS
- RC Drain Tank temperature >120 Deg F

2.0 CAUSES
- Open or leaking PORV
- Open or leaking Code safety
- Open or leaking Pressurizer vent valves

3.0 AUTOMATIC ACTIONS
3.1 IC-V-20 opens and WDL-P-8 start at 110°F.

B. Incorrect.
Part 1 is incorrect. Plausible if the candidate believes that Pressurizer level lowers due to a loss of inventory.

Part 2 is incorrect. Plausible if the candidate does not recognize that Reactor Coolant Drain Tank temperature has exceeded the setpoint to start WDL-P-8. IAW OP-TM-PPC-A0459, RC Drain Tank WDL-T-3 Temp:

1.0 SETPOINTS
- RC Drain Tank temperature >120 Deg F

2.0 CAUSES
- Open or leaking PORV
- Open or leaking Code safety
- Open or leaking Pressurizer vent valves

3.0 AUTOMATIC ACTIONS
3.1 IC-V-20 opens and WDL-P-8 start at 110°F.

C. Correct.
Part 1 is correct. IAW TQ-TM-104-220-C001, Reactor Coolant System:

2) Pressurizer level is maintained at 220" when the Plant is operating at 100% full power. The level is sensed and applied to a level control circuit. The level control circuit when in the automatic mode will control the position of the normal makeup valve (MU-V-17). If the level drops below 220" the circuit will open the normal makeup valve until the levels returns to normal. The Pressurizer level control setpoint is set at 55% corresponding to 220".

Since MU-V-17 is set to maintain 220 inches in the Pressurizer, it will close when Pressurizer level indicates high. As the Pressurizer Steam Space depressurizes, the water volume in the Pressurizer will expand and level will indicate higher. IAW OP-TM-MAP-G0105:

PZR LEVEL HI-HI

2.0 CAUSES

- Loss of Letdown
- Excessive Makeup Flow
- Large pressurizer Steam Space Leak
- Faulty level instrumentation
- Loss of ICS Auto power

3.0 AUTOMATIC ACTIONS

3.1 If in Auto, then MU-V-17 Closes.

Part 2 is correct. WDL-P-8 will automatically start when the Reactor Coolant Drain Tank temperature reaches 110°F. Since PPC Alarm A0459 comes in to alarm at a Reactor Coolant Drain Tank temperature of >120°F, the candidate must realize that WDL-P-8 has started. IAW OP-TM-PPC-A0459, RC Drain Tank WDL-T-3 Temp:

1.0 SETPOINTS

- RC Drain Tank temperature >120 Deg F

2.0 CAUSES

- Open or leaking PORV
- Open or leaking Code safety
- Open or leaking Pressurizer vent valves

3.0 AUTOMATIC ACTIONS

3.1 IC-V-20 opens and WDL-P-8 start at 110°F.

D. Incorrect.
Part 1 is correct. IAW TQ-TM-104-220-C001, Reactor Coolant System:

2) Pressurizer level is maintained at 220" when the Plant is operating at 100% full power. The level is sensed and applied to a level control circuit. The level control circuit when in the automatic mode will control the position of the normal makeup valve (MU-V-17). If the level drops below 220" the circuit will open the normal makeup valve until the levels returns to normal. The Pressurizer level control setpoint is set at 55% corresponding to 220".

Since MU-V-17 is set to maintain 220 inches in the Pressurizer, it will close when Pressurizer level indicates high. As the Pressurizer Steam Space depressurizes, the water volume in the Pressurizer will expand and level will indicate higher. IAW OP-TM-MAP-G0105:

PZR LEVEL HI-HI

2.0 CAUSES

- Loss of Letdown
- Excessive Makeup Flow
- Large pressurizer Steam Space Leak
- Faulty level instrumentation
- Loss of ICS Auto power

3.0 AUTOMATIC ACTIONS

3.1 If in Auto, then MU-V-17 Closes.

Part 2 is incorrect. Plausible if the candidate does not recognize that Reactor Coolant Drain Tank temperature has exceeded the setpoint to start WDL-P-8. IAW OP-TM-PPC-A0459, RC Drain Tank WDL-T-3 Temp:

1.0 SETPOINTS

- RC Drain Tank temperature >120 Deg F

2.0 CAUSES

- Open or leaking PORV
- Open or leaking Code safety
- Open or leaking Pressurizer vent valves

3.0 AUTOMATIC ACTIONS

3.1 IC-V-20 opens and WDL-P-8 start at 110°F.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	008	AK2.03
	Importance Rating	2.5	

Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Controllers and Positioners.

Proposed Question: RO Question # 4

Technical Reference(s):
TQ-TM-104-220-C001, p 26, Rev 8
OP-TM-MAP-G0105, p 1, Rev 2
OP-TM-PPC-A0459, p 1, Rev 0

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-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 7

55.43

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the interrelation between a Pressurizer Steam Space Accident with both the control of the Pressurizer Makeup Valve and Reactor Coolant Drain Tank Pump.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the given information and determine the multi-step cause-effect relationship from the given event.

What MUST be known:
1. How MU-V-17 is controlled upon a Pressurizer Steam Space Accident. 2. How WDL-P-8 is affected upon a Pressurizer Steam Space Accident.

5 **ID: 1142309** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- The following pumps are running:
  - NS-P-1A
  - NS-P-1C
  - SR-P-1A
  - SR-P-1C
  - NR-P-1A
  - NR-P-1B (on "T" 480v Bus)
  - IC-P-1
- The following pumps are ES selected:
  - NS-P-1A
  - NS-P-1C
  - NR-P-1A
  - NR-P-1B (on "T" 480v Bus)

Sequence of Events:

- A large break LOCA occurs.
- RB pressure peaks at 40 PSIG.

Given the above information and assuming no operator actions, which of the following actions will occur?

A. NR-P-1B trips and then restarts.

B. IC-P-1B starts and remains running.

C. NS-P-1B starts and remains running.

D. SR-P-1A trips and remains shutdown.

Answer: B

Answer Explanation

A. Incorrect.
NR-P-1B will remain running throughout the event. This would be a correct choice due to the block loading sequence. if the event was combined with a loss of off-site power, a casualty that is often associated with the ESAS actuation due to design criteria.
Plausible if the candidate does not recognize that a Loss of Offsite Power did not occur.

B. Correct.
The 30# isolation causes a low-flow condition in the ICCW system which then causes the standby pump to start. IAW OP-TM-541-000, Primary Component Cooling:

6.0 SYSTEM INFORMATION
6.2 The IC pumps, unless in Pull-To-Lock, will start on a low system flow of 550 gpm from IC-5-FS or from a trip of the running pump.

Additionally, IAW TQ-TM-104-531-C001, Primary Cooling Systems:

7) 30 psig Reactor Building Pressure Isolation
a) IC-V-2, IC-V-3, IC-V-4 and IC-V-6 automatically close. IC-V-74 will open to provide a recirculation path. Standby IC-P-1 will automatically start on low ICCW flow (<550 gpm).

C. Incorrect.
NS-P-1B is the standby pump. Nothing in the stem leads to a standby pump start signal being received. Plausible since this would be the response of NS-P-1B on a loss of off-site power, a casualty that is often associated with the ESAS actuation due to design criteria.

D. Incorrect.
SR-P-1A will remain running throughout the event. This would be a correct choice due to a 27/86 lockout if the event was combined with a loss of off-site power, a casualty that is often associated with the ESAS actuation due to design criteria.
Plausible if the candidate does not recognize that a Loss of Offsite Power did not occur.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	011	EK2.02
	Importance Rating	2.6	

Knowledge of the interrelations between the Large Break LOCA and the following: Pumps.

Proposed Question:	RO Question # 5		
Technical Reference(s):	TQ-TM-104-531-C001, p 41, Rev 8 OP-TM-541-000, p 12, Rev 20		
Proposed References to be provided to applicants during examination:			None
Learning Objective:	531-GLO-5		
Question Source:	Bank #	ID# 909424	
	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	8	
	55.43		

Comments:
The KA is matched because the question requires the candidates to demonstrate knowledge of the interrelation between a Large Break LOCA and the logic of various associated pumps.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the given information and determine the cause-effect resulting from the given event.

What MUST be known:
1. How the ICCW System responds to a large break LOCA (>30# RB pressure). 2. How the NSRW System responds to a large break LOCA (>30# RB pressure). 3. How the NSCCW System responds to a large break LOCA (>30# RB pressure). 4. How the SSRW System responds to a large break LOCA (>30# RB pressure).

6 **ID: 1142075** **Points: 1.00**

Plant Conditions:

- The plant is operating at 50% reactor power.
- "D" Emergency Feedwater Regulating Valve, EF-V-30D, is in manual due to an identified fault in the automatic circuitry.
- "A/B/C" Emergency Feedwater Regulating Valves, EF-V-30A-C, are in automatic.

Sequence of Events:

- Loss of VBA and D-16 power supply to HSPS.
- Reactor trip occurs due to high RCS pressure.
- EFW actuation due to low levels in both OTSG 1A and OTSG 1B.

Given the above information, identify the one statement below that describes the effect on the operation of the Emergency Feedwater System.

A. EF-V-30A automatic control by HSPS is not functional.

B. EF-V-30B automatic control by HSPS is not functional.

C. MS-V-13A, Main Steam to EF-P-1, will not open automatically.

D. EF-V-30D cannot be controlled in manual from the Control Room.

Answer: A

Answer Explanation

A. Correct.
EF-V-30A is controlled by Train A, which will not actuate due to loss of all train power. IAW OP-TM-AOP-015:

3.12 SELECT the following instruments in HSPS Cabinet Section A1 Rack 4 for EF-V-30A and EF-V-30C control:

- LT-1047
- LT-1045
- LT-1055
- LT-1053

IAW OP-TM-AOP-0151:

Step 3.12 This step provides guidance to select operable steam generator level instruments in HSPS for EF-V-30A and EF-V-30C control. This action will allow automatic operation of EF-V-30A and EF-V-30C in the event of an EFW actuation.

Attachment 1: EF-V-30A: Select LT-1045 and LT-1047.

Additionally, IAW TQ-TM-104-644-C001:

4. The Heat Sink Protection System instruments are all vital powered through their respective Heat Sink Protection System channel or signal conditioning cabinet.
   a. Train A contains 2 power supplies
      1) Power Supply 1 (PS1)
         a) Fed from VBA breaker #11
            (1) The 120 VAC transformer that supplies Power Supply 1 also supplies 24 VAC for CC recorders.
            (2) The power supply to the OTSG Train A startup range digital level recorder (FW -LR-1083) is PS2 (VBA Bkr 8).
         b) The output of power supply 1 is +/–15 VDC
      2) Power supply 2 (PS2)
         a) Normally fed from VBA breaker #8
         b) Backup feed is from a 24 VDC supply (PS3) powered from 115 VAC panel D-16.

B. Incorrect.
EF-V-30B is controlled by Train B, which will actuate as designed. EF-V-30A and EF-V-30C will be inoperable in automatic and manual. This would be 1 EF-V-30 valve to each OTSG as EF-V-30A and EF-V-30D feed the "A" OTSG while EF-V-30B and EF-V-30C feed the "B" OTSG.
Distracter is plausible if the candidate confuses which EFW control valves are associated with which Train and/or OTSG.

C. Incorrect.
EF-P-1 will auto start due to Train B actuation on low OTSG levels. Train B actuation opens both MS-V-13A and MS-V-13B.
Distracter is plausible because loss of Train A power will prevent Train A start of EF-P-1.

D. Incorrect.
EF-V-30D is controlled by Train B, and will function as designed. EF-V-30A and EF-V-30C will be inoperable in automatic and manual. This would be 1 EF-V-30 valve to each OTSG as EF-V-30A and EF-V-30D feed the "A" OTSG while EF-V-30B and EF-V-30C feed the "B" OTSG.
Distracter is plausible if the candidate confuses the "A" Train EFW control valves with the "A" OTSG EFW control valves.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E04 EK2.1	
	Importance Rating	3.8	

Knowledge of the interrelations between the (Inadequate Heat Transfer) 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 # 6

Technical Reference(s):
TQ-TM-104-644-C001, p 18, Rev 2
OP-TM-AOP-015, p 5, Rev 8
OP-TM-AOP-0151, p 6, Rev 6

Proposed References to be provided to applicants during examination: None

Learning Objective: 644-GLO-11

Question Source: Bank # ID# 363944
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 4
55.43

Comments:
The KA is matched because the operator must demonstrate knowledge of the interrelation between an inadequate heat transfer condition and components and signals. The EFW valves associated with the "A" OTSG (and also therefore the "A" RCS Loop) are EF-V-30A and EF-V-30D. With EF-V-30D in manual, it will remain at 0 until manipulated by an operator. Additionally, EF-V-30A control has been lost, so there is no feedwater available to the "A" OTSG (causing an inadequate heat transfer condition).

The question is at the Comprehension/Analysis cognitive level because the operator must analyze the conditions and apply knowledge of HSPS logic to the analysis.

What MUST be known:
1. The power supplies for HSPS components. 2. The effect on HSPS upon a loss of a power supply. 3. Which HSPS Trains control each EFW Control Valve.

7 **ID: 1142051** **Points: 1.00**

Plant Conditions:

- A Plant Cooldown in progress.
- A Decay Heat Removal Train is in service.
- RCS temperature is 295 degrees.
- RCS cooldown rate is 20 degrees F per hour.
- All actions have been taken for the Plant Cooldown.

Event:

- A transient occurs, resulting in the following conditions:
  - o RCS temperature 300 degrees F and rising.
  - o RCS pressure 380 psig and rising.

Given the above information and assuming no operator action, select the first automatic action that will take place and the reason for the action.

A. RC-RV-2, Pressurizer Pilot Operated Relief Valve, will open at 400 psig to prevent RCS overpressure (NDTT concern).

B. RC-RV-2, Pressurizer Pilot Operated Relief Valve, will open at 592 psig to prevent RCS overpressure (NDTT concern).

C. DH-V-1 and DH-V-2, RCS Drop Line to Decay Heat Isolation Valves, will close at 400 psig to protect the low pressure Decay Heat Removal piping.

D. DH-V-1 and DH-V-2, RCS Drop Line to Decay Heat Isolation Valves, will close at 592 psig to protect the low pressure Decay Heat Removal piping.

Answer: C

Answer Explanation

A. Incorrect.

Although there is a pressure associated with the PORV for NDTT concerns, the pressure is 592 psig, therefore the Decay Heat Valves closing would be the first action that would occur based on pressure. Plausible if the candidate believes that the PORV LTOP setpoint is 400 psig. IAW TQ-TM-104-220-C001, Reactor Coolant System:

> a. The PORV (RC-RV-2) is mounted on a separate nozzle on the top of the Pressurizer. Its function when RCS temperature is less than 313°F is to relieve overpressure if RCS pressure increases to 592 psig. This provides part of the Low Temperature Overpressure Protection (LTOP).

B. Incorrect.

Although there is a pressure associated with the PORV for NDTT concerns, the pressure is 592 psig, therefore the Decay Heat Valves closing would be the first action that would occur based on pressure. Plausible if the candidate is not familiar with the pressure associated with Decay Heat Valves. IAW TQ-TM-104-220-C001, Reactor Coolant System:

> a. The PORV (RC-RV-2) is mounted on a separate nozzle on the top of the Pressurizer. Its function when RCS temperature is less than 313°F is to relieve overpressure if RCS pressure increases to 592 psig. This provides part of the Low Temperature Overpressure Protection (LTOP).

C. Correct.

IAW TQ-TM-104-212-C001, Decay Heat Removal System:

1) Valve DH-V-1 and DH-V-2 Controls
   a) Auto Close Interlock: The valves will automatically close and cannot be opened when the RCS pressure is above 400 psig. The automatic closure of DH-V-1 & 2 is designed to isolate the DH System when the RCS pressure exceeds the DH System design pressure.

The 400 psig setpoint protects the DH system from significant overpressure.

The interlock evolved from an NRC concern about the necessity to keep the low pressure DH System isolated from the high pressure RCS in order to avoid damage by over-pressurization or the potential for loss of integrity of the low pressure DH System and possible radioactive releases.

D. Incorrect.
Although there is a pressure associated with the Auto-close of DH-V-1 and DH-V-2, the pressure is 400 psig. Plausible if the candidate is not familiar with the pressure associated with Decay Heat Valves. IAW TQ-TM-104-212-C001, Decay Heat Removal System:

1) Valve DH-V-1 and DH-V-2 Controls
   a) Auto Close Interlock: The valves will automatically close and cannot be opened when the RCS pressure is above 400 psig. The automatic closure of DH-V-1 & 2 is designed to isolate the DH System when the RCS pressure exceeds the DH System design pressure.

The 400 psig setpoint protects the DH system from significant overpressure.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	025	AK3.02
	Importance Rating	3.3	

Knowledge of the reasons for the following responses as they apply to the Loss of Residual Heat Removal System: Isolation of RHR low-pressure piping prior to pressure increase above specified level.

Proposed Question: RO Question # 7

Technical Reference(s): TQ-TM-104-212-C001, p 42, Rev 14

Proposed References to be provided to applicants during examination: None

Learning Objective: 212-GLO-10

Question Source: Bank # ID# 862928

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	3
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge of the isolation criteria of Decay Heat Removal piping to avoid overpressurization.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall pieces of information and determine which event comes first.

What MUST be known:
1. What pressure the RCS Drop Line Isolation Valves to the Decay Heat System closes at. 2. What pressure the PORV opens at while in LTOP mode.

8 **ID: 1142053** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- A Small Break Loss of Coolant Accident (LOCA) is in progress.
- The leak is in the Reactor Building (RB).
- Reactor Coolant Pressure is 1885 psig.
- RB pressure is 5.7 psig.
- Map F-1-8, NS Surge Tank Level Hi/Lo, is in alarm
- The running Nuclear Services Closed Cooling Pumps currents are erratic.

Given the above information, which of the following automatic actions has occurred?

1. Reactor Trip Isolation.
2. Engineered Safeguards Actuation.
3. Nuclear Services Closed Cooling Line Break Isolation.

A. 1 and 2, ONLY.

B. 1 and 3, ONLY.

C. 2 and 3, ONLY.

D. 1, 2 and 3.

Answer: D

Answer Explanation

The following information on what signals the three choices must be known to answer the question.

Reactor Trip Isolation:

IAW OS-24, Conduct of Operations During Abnormal and Emergency Events

2. Reactor Trip Requirements:

2.1 A reactor trip is required (automatic or manual) if any of the following limits are exceeded:

- Reactor power is > 105.1%
- RCS Thot is > 618.8F
- RCS pressure > 2355 psig
- RCS pressure < 1900 psig
- Containment pressure > 4 psig
- Reactor Power >55% with less than 3 RCPs operating
- No RCP operating in one loop
- Reactor power above flux/flow/axial imbalance limit
- Turbine trip and >45% reactor power
- Both A and B Main Feedwater pump turbines trip and > 7 % reactor power.

Upon a reactor trip, Reactor Trip Isolation occurs. Step 3.15 of OP-TM-EOP-001, Reactor Trip, states:

3.15 Initiate OP-TM-642-904 "Reactor Trip Isolation ESAS Actuation"

Furthermore, OP-TM-EOP-0011, Reactor Trip Basis Document, gives the following explanation:

2.0 MITIGATION STRATEGY

- The expected reactor trip response includes all rods have fully inserted, feedwater is properly controlled, minimum required instrument and ES power supplies are energized, pressurizer level is being controlled, OTSG pressure and RCS temperature are being controlled, RCS pressure is being controlled, instrument air pressure is available, the main generator is properly disconnected, reactor trip containment isolation has occurred, containment pressure is not rising, RCS leakage is not excessive, and there is no indication of significant secondary leak in the Intermediate Building.

Step 3.15 The step intent is to ensure reactor trip specific containment isolation valves are closed and initiate OP-TM-642-904. This procedure provides direction for verifying complete actuation, for defeating ESAS and for restoring the affected functions when desired. There is no GEOG step equivalent in III.A. ESAS "Reactor Trip Isolation" is a TMI specific design feature.

Engineered Safeguards Actuation:

IAW TQ-TM-104-642-C001, Engineered Safeguards Actuation System:

C. Explanation

1. Operating Mode Flow Paths
    a. Engineered Safeguards Actuation System is normally in Standby
    b. Automatic actuation signal sequence for a major LOCA
        1) 1900 psig Reactor Coolant System pressure – Reactor Trips
        2) 1600 psig Reactor Coolant System pressure actuation
        3) 4 psig Reactor Building pressure actuation
        4) 30 psig Reactor Building pressure actuation
        5) 500 psig Reactor Coolant System pressure actuation

Nuclear Services Closed Cooling Line Break Isolation:
IAW OP-TM-MAP-F0108, NS Surge Tank Level Hi/Lo:

3.0 AUTOMATIC ACTIONS

- Lo Level - No auto action occurs unless an ES Actuation occurs simultaneously. ES Actuation Channel "A" and Lo level Closes NS-V-4 and NS-V-15. ES Actuation Channel "B" and Lo level Closes NS-V-15 and NS-V-35.

Reactor Trip Isolation is initiated from any automatic or manual reactor trip. At > 4psig in the RB all channels of RPS will trip and cause an automatic reactor trip.

RB Isolation and Cooling will initiate on RB pressure greater than 4 psig.

NSCCW Line Break Isolation occurs on any 1600# / 500# / 4# ES signal and a low level in the NSCCW Surge tank.

The Main Annunciator in alarm, by itself, does not tell the candidate if the NSCCW Surge Tank level is high or low. However, since the NSCCW Pumps have eraatic amps, it gives the indication of pump cavitation, which indicates low NPSH, and therefore the candidate can determine that the NSCCW Surge Tank has a low level condition.

A. Incorrect.
Plausible if candidate does not understand requirements for a Nuclear Services Closed Cooling Line Break Isolation.

B. Incorrect.
Plausible if candidate does not understand requirements for an Engineered Safeguards Actuation.

C. Incorrect.
Plausible if candidate does not understand requirements for a Reactor Trip Isolation.

D. Correct.

- Reactor Trip Isolation has automatically occurred on the reactor trip (which occurred when the Reactor Coolant System went below 1900 psig).

- Engineered Safety Actuation has automatically occurred when the Reactor Building Pressure exceeded 4 psig.

- Nuclear Services Closed Cooling Line Break Isolation has occurred with the combination of the ESAS signal coming in and a low Nuclear Services Closed Cooling Water Surge Tank level.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	026	AK3.02
	Importance Rating	3.6	

Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS.

Proposed Question: RO Question # 8

Technical Reference(s):
OP-TM-EOP-0011, pgs 2,12, Rev 5
OP-TM-MAP-F0108, p 1, Rev 2
TQ-TM-104-642-C001, p 28, Rev 6
OS-24, p 29, Rev 25

Proposed References to be provided to applicants during examination: None

Learning Objective: 531-GLO-5

Question Source: Bank # ID# 356966

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

Comments:
The KA is matched because the operator must demonstrate knowledge of the automatic actions for the Nuclear Services Closed Cooling Water upon a Engineered Safeguards signal.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall pieces of information and determine whether an action has occurred.

What MUST be known:
1. The conditions required for a Reactor Trip Isolation signal to occur.
2. The conditions required for an Engineered Safeguards Actuation signal to occur.
3. The conditions required for a Nuclear Services Closed Cooling Water Line Break Isolation signal to occur.

9 **ID: 1142335** **Points: 1.00**

Plant Conditions:

- Plant is operating at 100% reactor power.
- The "A" Emergency Diesel, EG-Y-1A, is in ES standby.
- The "B" Emergency Diesel, EG-Y-1B, is running, loaded onto its associated ES bus, in parallel with offsite power for a surveillance run.

Event:

- All DC power to both EG-Y-1A and EG-Y-1B is lost.
- DC power is simultaneously lost to the generator breakers associated with EG-Y-1A and EG-Y-1B.

Given the above information and assuming no operator actions, EG-Y-1A is ___(1)___ and EG-Y-1B is ___(2)___.

A. (1) running
(2) running

B. (1) running
(2) secured

C. (1) secured
(2) running

D. (1) secured
(2) secured

Answer: A

Answer Explanation

A. Correct.

Part 1 is correct. EG-Y-1A was in ES standby when the loss of DC occurred, therefore, EG-Y-1A will start since the air start solenoid fails open on a loss of DC. IAW TQ-TM-104-861-C001, Emergency Diesel Generators and Auxiliaries (Including SBO Generator):

d. Loss of DC to diesel generators.
1) With diesel generator in standby:
a) The D/G will auto start due to loss of DC to the air start solenoid.
b) The exciter will remain shorted due to lack of DC (K1 relay).
c) The engine can not be stopped by using any of the electrical pushbuttons (control room 1E bus or EMIP).
d) Protective trips O.O.S. (except overspeed).
e) Annunciators, O.O.S.
f) Operating Procedures AOP-023 and AOP-024 address RO/AO actions to shut down diesels.

Part 2 is correct. EG-Y-1B was in parallel when the loss of DC occurred; therefore EG-Y-1B will remain running since there is no trip signal received. IAW TQ-TM-104-861-C001, Emergency Diesel Generators and Auxiliaries (Including SBO Generator):

d. Loss of DC to diesel generators.
2) With diesel generator operating in parallel:
a) Exciter shutdown due to K1 relay action.
b) Depending on how fast D.C. is lost and if DC is lost to 1D/E 4160V ES busses, generator breaker(s) may open.
c) 1.c,d,e, and f above apply here.
d) Failure of generator breaker to open if unit is in parallel operations would cause operation (with field shorted) as an inductive load and damage to the generator will occur. Trip generator breaker with push-button at the generator breaker.

B. Incorrect.

Part 1 is correct. EG-Y-1A was in ES standby when the loss of DC occurred, therefore, EG-Y-1A will start since the air start solenoid fails open on a loss of DC. IAW TQ-TM-104-861-C001, Emergency Diesel Generators and Auxiliaries (Including SBO Generator):

d. Loss of DC to diesel generators.
1) With diesel generator in standby:
a) The D/G will auto start due to loss of DC to the air start solenoid.
b) The exciter will remain shorted due to lack of DC (K1 relay).
c) The engine can not be stopped by using any of the electrical pushbuttons (control room 1E bus or EMIP).
d) Protective trips O.O.S. (except overspeed).
e) Annunciators, O.O.S.
f) Operating Procedures AOP-023 and AOP-024 address RO/AO actions to shut down diesels.

Part 2 is incorrect but plausible if the candidate believes that the running Emergency Diesel Generator will shutdown with a loss of the exciter.

C. Incorrect.
Part 1 is incorrect but plausible if the candidate is not familiar with the failure modes of the Emergency Diesel Generator air start solenoids.

Part 2 is correct. EG-Y-1B was in parallel when the loss of DC occurred, therefore EG-Y-1B will remain running since there is no trip signal received. IAW TQ-TM-104-861-C001, Emergency Diesel Generators and Auxiliaries (Including SBO Generator):

d. Loss of DC to diesel generators.
  2) With diesel generator operating in parallel:
    a) Exciter shutdown due to K1 relay action.
    b) Depending on how fast D.C. is lost and if DC is lost to 1D/E 4160V ES busses, generator breaker(s) may open.
    c) 1.c,d,e, and f above apply here.
    d) Failure of generator breaker to open if unit is in parallel operations would cause operation (with field shorted) as an inductive load and damage to the generator will occur. Trip generator breaker with push-button at the generator breaker.

D. Incorrect.
Part 1 is incorrect but plausible if the candidate is not familiar with the failure modes of the Emergency Diesel Generator air start solenoids.

Part 2 is incorrect but plausible if the candidate believes that the running Emergency Diesel Generator will shutdown with a loss of the exciter.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	058	AK3.01	
	Importance Rating	3.4		

Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Use of dc control power by D/Gs.

Proposed Question: RO Question # 9

Technical Reference(s): TQ-TM-104-861-C001, p 68, Rev 11

Proposed References to be provided to applicants during examination: None

Learning Objective:	861-GLO-6		
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	8	
	55.43		

Comments:

The KA is matched because the operator must demonstrate knowledge of the response of the Emergency Diesel Generators upon a loss of DC Power.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall pieces of information and determine whether an action has occurred.

What MUST be known:
1. What the effect is to an Emergency Diesel Generator that is in an ES Standby condition. 2. What the effect is to an Emergency Diesel Generator that is currently running.

10 **ID: 1142336** **Points: 1.00**

Plant Conditions:

- The plant is operating at 70% reactor power, MOL.
- The "A" Reactor Coolant Pump, RC-P-1A, has had an oil leak for 6 months.
- Oil additions have been made every 5 days IAW OP-TM-226-401, Remote Oil Addition to RC-P-1A.

Event:

- A large failure of the RC-P-1A oil system has occurred (a rapid drop).
- PPC Point A0961, RC-P-1A Upper BRG Lube Oil Level - Hi Lo, is in alarm.
  - RC-T-7, RC-P-1A/B East Oil Drain Tank, is filled and oil is overflowing uncontained into the Reactor Compartment.
- PPC Point A0701, RC-P-1A MTR UP Thrust BRG Temp, is in alarm.
  - RC-P-1A Motor Upper Thrust Bearing temperature is 220°F and rising.

Given the above information, RC-P-1A must ___(1)___ and the procedural action to take next is to ___(2)___.

A. (1) be secured IAW OP-TM-226-151, Shutdown RC-P-1A
(2) maintain the reactor critical, below 75%

B. (1) be secured IAW OP-TM-226-151, Shutdown RC-P-1A
(2) initiate a plant shutdown and cooldown to less than 400°F RCS temperature

C. (1) have oil added immediately IAW OP-TM-226-401, Remote Oil Addition to RC-P-1A
(2) maintain the reactor critical, below 75%

D. (1) have oil added immediately IAW OP-TM-226-401, Remote Oil Addition to RC-P-1A
(2) initiate a plant shutdown and cooldown to less than 400°F RCS temperature

Answer: B

Answer Explanation

A. Incorrect.
Part 1 is correct. IAW OP-TM-PPC-A071, RC-P-1A MTR UP Thrust BRG Temp:

- IAAT RC-P-1A Thrust Bearing Temperature (Up or Down) is > 200F, then PERFORM OP-TM-226-151 to place RC-P-1A in Standby mode.

Part 2 is incorrect. Plausible if the candidate does not recognize the fire hazard associated with oil in a hot environment. Without the concern of the oil, maintaining the reactor critical would be the correct answer.

B. Correct.
Part 1 is correct. IAW OP-TM-PPC-A0701, RC-P-1A MTR UP Thrust BRG Temp:

- IAAT RC-P-1A Thrust Bearing Temperature (Up or Down) is > 200F, then PERFORM OP-TM-226-151 to place RC-P-1A in Standby mode.

Part 2 is correct. IAW OP-TM-PPC-A0961, RC-P-1A Upper BRG Lube Oil Level - Hi Lo:

- If a large failure of RC-P-1A Oil system is suspected (rapid drop), and it is suspected that the oil is not being contained / collected, then INITIATE Plant Shutdown and Cooldown to lower RCS temperature to < 400°F (below the flashpoint of the oil).

C. Incorrect.
Part 1 is incorrect but plausible if the candidate is not familiar with the temperature setpoints associated with the Reactor Coolant Pumps (there are several temperature requirements that are above 220F). IAW OP-TM-226-000, Reactor Coolant Pumps:

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.

Part 2 is incorrect. Plausible if the candidate does not recognize the fire hazard associated with oil in a hot environment. Without the concern of the oil, maintaining the reactor critical would be the correct answer.

D. Incorrect.
Part 1 is incorrect but plausible if the candidate is not familiar with the temperature setpoints associated with the Reactor Coolant Pumps (there are several temperature requirements that are above 220F). IAW OP-TM-226-000, Reactor Coolant Pumps:

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.

Part 2 is correct. IAW OP-TM-PPC-A0961, RC-P-1A Upper BRG Lube Oil Level - Hi Lo:

- If a large failure of RC-P-1A Oil system is suspected (rapid drop), and it is suspected that the oil is not being contained / collected, then INITIATE Plant Shutdown and Cooldown to lower RCS temperature to < 400°F (below the flashpoint of the oil).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015/017 AA1.02	
	Importance Rating	2.8	

Ability to operate and/or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): RCP oil reservoir level and alarm indicators.

Proposed Question: RO Question # 10

Technical Reference(s): OP-TM-PPC-A0961, p 1, Rev 1
OP-TM-PPC-A0701, p 1, Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-11

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

Comments:

The KA is matched because the operator must demonstrate knowledge of the RCP oil reservoir levels and alarms associated with the Reactor Coolant Pump Malfunctions.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall pieces of information and determine what action to take.

What MUST be known:
1. What the procedural requirement is with regards to a reactor Coolant Pump, upon a complete loss of lube oil to that Reactor Coolant Pump. 2. What the procedural requirement is with regards to RCS pressure and temperature, upon a discovery of free-standing oil in the Reactor Building.

11 **ID: 1146606** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Sequence of Events:

- A Reactor Trip occurred due to a loss of both Main Feedwater Pumps.
- EF-P-1 failed to start and cannot be started manually.
- Currently:
    - CO-T-1A/B Condensate Storage Tank levels are both lowering, currently 4.9 feet.

Based on the above information, identify the ONE selection below that describes, in order of preference, the alternate sources of water for the Emergency Feedwater System that have operator lineup actions.

A. (1) Main Condenser Hotwell, then
(2) Fire Service Altitude Tank, FS-T-1, then
(3) Million Gallon Demin Water Tank, DW-T-2.

B. (1) Main Condenser Hotwell, then
(2) Million Gallon Demin Water Tank, DW-T-2, then
(3) Susquehanna River.

C. (1) Fire Service Altitude Tank, FS-T-1, then
(2) Susquehanna River, then
(3) Million Gallon Demin Water Tank, DW-T-2.

D. (1) Million Gallon Demin Water Tank, DW-T-2, then
(2) Main Condenser Hotwell, then
(3) Susquehanna River.

Answer: B

Answer Explanation

A. Incorrect.

Plausible if the candidate is not familiar with the order of sources that supply water to the Emergency Feedwater System or determines that one of the sources in the order does not have operator actions associated with it. Fire Service System, although a possible source to Emergency Feedwater, is only used during an event that is beyobnd design basis. IAW TQ-TM-104-424-C001, Emergency Feedwater System:

12. EFW from Fire Services: OP-TM-424-921
    a. Severe security event, which results in catastrophic damage to the facility that is beyond the design basis
    b. Emergency Director permission obtained to align Fire Services for OTSG fill.

B. Correct.

IAW TQ-TM-104-424-C001, Emergency Feedwater System:

10. Condensate Storage Tanks are discussed in detail in the Condensate lesson plan. These two tanks provide the normal source of water to the EF pumps. Each tank has a 265,000 gallon capacity. The tanks are located outside opposite sides of the service building (east and west) for single missile protection.
    a. Alternate sources of water for EFW, in order of use, are as follows:
        1) Hotwell
            a) Approximately 165,000 gal capacity.
            b) Used when CSTs reaches 5 feet.
            c) Must break vacuum to use.
        2) Million Gallon Demin Water Tank
            a) The Million Gallon Demin tank, is pumped (or gravity fed) to the condensate storage tanks.
            b) Hotwell level is maintained at 9 feet by makeup from the million gallon tank (DW-T2)
            c) 22,000 gallons per foot.
            d) Maintain EFW flow < 600 gpm.
            e) Used until <2# suction pressure or cavitation occurs on running EFW Pumps AND no other sources of Condensate are available then shift to river water.
        3) River (via Reactor River connection and EF-V-4/5)
            a) Used as last resort

C. Incorrect.

Plausible if the candidate is not familiar with the order of sources that supply water to the Emergency Feedwater System or determines that one of the sources in the order does not have operator actions associated with it. Hotwell is the first alternate source (see Explanation for the correct choice). Additionally, the Fire Service System, although a possible source to Emergency Feedwater, is only used during an event that is beyobnd design basis. IAW TQ-TM-104-424-C001, Emergency Feedwater System:

12. EFW from Fire Services: OP-TM-424-921
    a. Severe security event, which results in catastrophic damage to the facility that is beyond the design basis
    b. Emergency Director permission obtained to align Fire Services for OTSG fill.

D. Incorrect.
Plausible if the candidate is not familiar with the order of sources that supply water to the Emergency Feedwater System or determines that one of the sources in the order does not have operator actions associated with it. Hotwell is the first alternate source. IAW TQ-TM-104-424-C001, Emergency Feedwater System:

10. Condensate Storage Tanks are discussed in detail in the Condensate lesson plan. These two tanks provide the normal source of water to the EF pumps. Each tank has a 265,000 gallon capacity. The tanks are located outside opposite sides of the service building (east and west) for single missile protection.
    a. Alternate sources of water for EFW, in order of use, are as follows:
        1) Hotwell
            a) Approximately 165,000 gal capacity.
            b) Used when CSTs reaches 5 feet.
            c) Must break vacuum to use.
        2) Million Gallon Demin Water Tank
            a) The Million Gallon Demin tank, is pumped (or gravity fed) to the condensate storage tanks.
            b) Hotwell level is maintained at 9 feet by makeup from the million gallon tank (DW-T2)
            c) 22,000 gallons per foot.
            d) Maintain EFW flow < 600 gpm.
            e) Used until <2# suction pressure or cavitation occurs on running EFW Pumps AND no other sources of Condensate are available then shift to river water.
        3) River (via Reactor River connection and EF-V-4/5)
            a) Used as last resort

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	054 AA1.01	
	Importance Rating	4.5	

Ability to operate and/or monitor the following as they apply to the Loss of Main Feedwater (MFW): AFW controls, including the use of alternate AFW sources.

Proposed Question: RO Question # 11

Technical Reference(s): TQ-TM-104-424-C001, p 12-13, Rev 10

Proposed References to be provided to applicants during examination: None

Learning Objective: 424-GLO-2

Question Source: Bank #

Modified Bank # ID# 677544

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

Comments:

The KA is matched because the operator must demonstrate knowledge of the operations of Emergency Feedwater (AFW) alternate sources that involve manipulation of controls.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall pieces of information and determine the order of Emergency Feedwater (AFW) sources.

What MUST be known:
1. The possible sources of water for the Emergency Feedwater System. 2. What the preferred order of alternate sources of water for the Emergency Feedwater System is.

Original Question:

Plant conditions:

Reactor was tripped due to low Main Condenser vacuum conditions.
Both Main Feedwater Pumps are tripped.
All three Emergency Feedwater Pumps are operating.
CO-T-1A/B Condensate Storage Tank levels are both lowering, approaching 5 feet.
Main Condenser is at atmospheric pressure.

Based on these conditions, identify the ONE selection below that describes the remaining sources of supply water for the Emergency Feedwater System.

A. (1) Main Condenser Hotwell
(2) DW-T-2 Million Gallon Tank
(3) Reactor River Water System

B. (1) Main Condenser Hotwell
(2) Pretreatment Building Clearwell
(3) Reactor River Water System

C. (1) DW-T-2 Million Gallon Tank
(2) FS-T-1 Fire Service Altitude Tank
(3) Reactor River Water System

D. (1) Main Condenser Hotwell
(2) DW-T-2 Million Gallon Tank
(3) FS-T-1 Fire Service Altitude Tank

Answer: A

12 **ID: 1142379** **Points: 1.00**

Plant Conditions:

- The plant is operating at 50% reactor power.

Event:

- All 230 KV breakers from switchyard to the grid OPEN.
- All four RCP's remain running.
- Plant power is lowering in ICS Track mode.
- "B" main feedwater pump is NOT responding to demand changes (constant speed).
- RCS pressure is 2230 psig and rising.

Given the above information, which of the following is the FIRST required IAW OP-TM-AOP-022, Load Rejection?

A. Trip the reactor.

B. Initiate Emergency Feedwater.

C. Trip "B" main feedwater pump.

D. Open the Spray Valve in Manual, then place in Auto.

Answer: D

Answer Explanation

A. Incorrect.

This step is prior to opening the Spray Valve in OP-TM-AOP-022, but it is an IAAT statement that is only performed if the Main Generator breakers have opened. The information given gives no indication that the Main Generator breakers have opened. Tripping the reactor may seem appropriate given that multiple events are occurring, however a trip at this time will induce a blackout, which is undesirable. Plausible if the candidate is not familiar with the requirements to trip the reactor IAW the load rejection procedure. IAW OP-TM-AOP-022, Load Rejection:

3.1 IAAT GB1-02 and GB1-12 OPEN, then TRIP the reactor and GO TO OP-TM-EOP-001 Reactor Trip.

B. Incorrect.

Initiating EFW is plausible since more cooling is (for the moment) needed to help curb the RCS pressure rise, however no setpoints have been reached that require its actuation. Plausible if the candidate is not familiar with the Load Rejection procedure. IAW TQ-TM-104-424-C001, Emergency Feedwater System:

Explanation

1. The EFW system receives automatic initiation and control signals from the Heat Sink Protection System (HSPS).
   a. EFW is auto initiated for the following HSPS signals
      1) Loss of both Main Feedwater Pumps
      2) EFW is backup supply system to MFW. Loss of all Reactor Coolant Pumps (RCPs)
         a) EFW flow to OTSG provides for smooth natural circulation conditions. (boiler-condenser mode)
      3) 4 PSIG RB pressure
         a) Initiates EFW in case of MFW line rupture inside containment.
      4) Low OTSG level (10 inches S/U range)
         a) Backup for loss of MFW.

C. Incorrect.

Since "B" feedwater pump is having a problem, it is plausible that a student may think that tripping it is required. Once power level is lowered, tripping the stuck feedpump may be appropriate in order to prevent an RCS low pressure trip due to cooldown (overfeed) and insurge reducing pressurizer saturation temperature. However, feedwater may be controlled at the valve level, and at this moment in time, heat source still exceeds heat sink since RCS pressure is rising (due to temperature rise). Plausible if the candidate is not familair with the Load Rejection procedure.

D. Correct.

According to the OP-TM-AOP-022, if RCS pressure rises to greater than 2200 psig, manually open spray valve, then place back to auto (so that it closes on setpoint). This is the correct answer since RCS Pressure is above 2000# in the information given and there is no other applicable step before this one. IAW OP-TM-AOP-022, Load Rejection:

3.2 If RCS pressure > 2200 psig, then manually OPEN RC-V-1 and then PLACE RC-V-1 to AUTO.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	077 AA1.04	
	Importance Rating	4.1	

Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: Reactor Controls.

Proposed Question: RO Question # 12

Technical Reference(s): OP-TM-AOP-022, p 1, Rev 5

Proposed References to be provided to applicants during examination: None

Learning Objective: 022-PCO-4

Question Source: Bank # ID # 304505

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

Comments:

The KA is matched because the operator must demonstrate the ability to recognize plant conditions that require manipulation of Reactor Controls.

The question is at the Comprehension/Analysis cognitive level because the operator must analyze multiple pieces of information and then select the correct action to be taken first.

What MUST be known:
1. The conditional procedural steps associated with the Load Rejection procedure.

13 **ID: 1142387** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- A small RCS leak has been identified.

Sequence of Events:

- T=1625:
  - A water addition begins.
  - Pressurizer Level is 215 inches.
  - Makeup Tank Level is 84 inches
  - RCS Tave is 579.3?F.

- T=1628
  - A **step change** in leakage occurs.
  - The water addition rate was increased on the batch controller.
  - Pressurizer Level is 215 inches.
  - Makeup Tank Level is 83 inches
  - RCS Tave is 579.2?F.
  - **Total** water added to the Makeup Tank is 60 gallons.

- T=1635:
  - Pressurizer Level is 205 inches.
  - Makeup Tank Level is 80 inches
  - RCS Tave is 578.8?F.
  - **Total** water added to the Makeup Tank is 420 gallons.

Given the above information, select the **most accurate** RCS leak rate at T=1635.

A. 26.8 gpm

B. 61.2 gpm

C. 76.0 gpm

D. 84.5 gpm

Answer: C

Answer Explanation

A. Incorrect.

Distractor will be selected if the candidate calculates the RCS leak rate from time 1625 to 1628. This is not the most accurate time frame to calculate, however, since it only calculates the original leak rate. Plausible if the candidate believes that the most accurate leak rate will involve the time period where no water addition was occurring.

B. Incorrect.

Distractor will be selected if the candidate calculates the RCS leak rate from time 1625 to 1635. This is not the most accurate time frame to calculate, however, since it combines both the original leak rate and the higher leak rate. Plausible if the candidate believes that calculating over a ten minute period shall give a more accurate leak rate. That would be a correct train of thought if the leak rate had not changed. IAW OS-24, Conduct of Operations During Abnormal and Emergency Events:

ATTACHMENT F, Transient RCS Leakrate Calculation Methodology

b. The longer the time interval between collecting data points, the more reliable the leakrate estimate will be. At a minimum leak rates should not be calculated for time intervals of < 5 minutes.

C. Correct.

The leak rate increased at T=1628. Therefore, OS-24, Conduct of Operations During Abnormal and Emergency Events, Attachment F, Transient RCS Leakrate Calculation Methodology, will be used from T=1628 through T=1635. This removes the lower leak rate from the calculation, while giving an accurate time frame of data (>5 minutes). Additionally, the candidate will need to recognize that the total water added to the Makeup Tank is 420 gallons, but there was 60 gallons added prior to T=1628. Therefore, the candidate will need to subtract 60 from 420 and use 360 gallons in the Attachment's calculation. IAW OS-24:

ATTACHMENT F, Transient RCS Leakrate Calculation Methodology

b. The longer the time interval between collecting data points, the more reliable the leakrate estimate will be. At a minimum leak rates should not be calculated for time intervals of < 5 minutes.

RCS leakrate (GPM) = [(ΔPL)*(12) + (ΔMTL)*(30) - (ΔTavg)*(COEFF) + GAL ADD] / ΔTIME

Where:

ΔPL = change in Pressurizer Level (Computer Point C4017) (initial - final inches)
ΔMTL = change in Makeup Tank Level (Computer Point A0498) (initial - final inches)
ΔTavg = change in RCS Average Temperature (Computer Point A5066) (initial - final °F)
GAL ADD = gallons added to the MU/RCS systems during the observation period
ΔTIME = change in time (final - initial minutes)
COEFF = 95 gal/°F if Tavg is 579°F otherwise use the table below

Solution:

RCS leakrate (GPM) = [(ΔPL)*(12) + (ΔMTL)*(30) - (ΔTavg)*(COEFF) + GAL ADD] / ΔTIME
= [(10) *(12) + (3) *(30) - (0.4) * (95) + 360] / 7
= [120 + 90 - 38 + 360] / 7
= 532 / 7
= 76.0 gpm

D. Incorrect.
Distractor will be selected if the candidate calculates the RCS leak rate from time 1628 to 1638 but uses 420 gallons as the amount of water added to the Makeup Tank. Plausible if the candidate does not recognize that the time frame began with 60 gallons added to the Makeup Tank and therefore that volume must be subtracted from the 420 gallons given at T=1638.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	009	EA2.04
	Importance Rating	3.8	

Ability to determine or interpret the following as they apply to a small break LOCA: PZR level.

Proposed Question: RO Question # 13

Technical Reference(s): OS-24, p 35, Rev 25

Proposed References to be provided to applicants during examination: OS-24, Attachment F

Learning Objective: AOP050-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	

Comments:
The KA is matched because the operator must interpret PZR Level and other points of data and determine the size of the small break LOCA.

The question is at the Comprehension/Analysis cognitive level because the operator must analyze multiple pieces of information and then perform a calculation to arrive at the correct answer.

What MUST be known:
1. What the minimum recommended time is for an accurate RCS leak rate calculation. 2. How to perform an RCS leak rate calculation.

14 **ID: 1142408** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- An earthquake is felt on Three Mile Island.
- A loss of Vital Bus "B" occurs.
  - OP-TM-AOP-016, Loss of VBB, is entered.
- An ES actuation occurs due to a LOCA.
  - OP-TM-EOP-001, Reactor Trip, is entered.

Given the above information, which of the following choices correctly completes the sentence?

MU-FT-1128, HPI Flow Loop C Transmitter is ___(1)___ and MU-V-16C, "C" HPI Control Valve, indication is ___(2)___ on Console Right.

A. (1) operable
(2) available

B. (1) operable
(2) NOT available

C. (1) inoperable
(2) available

D. (1) inoperable
(2) NOT available

Answer: C

Answer Explanation

A. Incorrect.
Part 1 is incorrect. According to OP-TM-AOP-016, Loss of VBB, Attachment 3, MU-FT-1128 is an Inoperable Instrument.

Part 2 is correct. According to 1107-4, Electrical Distribution Panel Listing, MU-V-16C is powered from the 1A Engineered Safeguards Valves Bus, which is powered from the 1S 480V Engineered Safeguards Bus, which is in turn, powered from the 1E 4160V Engineered Safeguards Bus. Although Vital Bus "B" is powered from the 1B Engineered Safeguards Motor Control Center, which in turn is powered from the 1S 480V Engineered Safeguards Bus, it is powered off of a different breaker than that for MU-V-16C. Therefore, since the loss of Vital Bus Bravo does not affect MU-V-16C, the valve indication is available.

B. Incorrect.
Part 1 is incorrect. According to OP-TM-AOP-016, Loss of VBB, Attachment 3, MU-FT-1128 is an Inoperable Instrument.

Part 2 is incorrect. According to 1107-4, Electrical Distribution Panel Listing, MU-V-16C is powered from the 1A Engineered Safeguards Valves Bus, which is powered from the 1S 480V Engineered Safeguards Bus, which is in turn, powered from the 1E 4160V Engineered Safeguards Bus. Although Vital Bus "B" is powered from the 1B Engineered Safeguards Motor Control Center, which in turn is powered from the 1S 480V Engineered Safeguards Bus, it is powered off of a different breaker than that for MU-V-16C. Therefore, since the loss of Vital Bus Bravo does not affect MU-V-16C, the valve indication is available. Plausible if the candidate associates the "B" Train HPI Valve with Vital Bus "B".

C. Correct.
Part 1 is correct. According to OP-TM-AOP-016, Loss of VBB, Attachment 3, MU-FT-1128 is an Inoperable Instrument.

Part 2 is correct. According to 1107-4, Electrical Distribution Panel Listing, MU-V-16C is powered from the 1A Engineered Safeguards Valves Bus, which is powered from the 1S 480V Engineered Safeguards Bus, which is in turn, powered from the 1E 4160V Engineered Safeguards Bus. Although Vital Bus "B" is powered from the 1B Engineered Safeguards Motor Control Center, which in turn is powered from the 1S 480V Engineered Safeguards Bus, it is powered off of a different breaker than that for MU-V-16C. Therefore, since the loss of Vital Bus Bravo does not affect MU-V-16C, the valve indication is available.

D. Incorrect.
Part 1 is correct. According to OP-TM-AOP-016, Loss of VBB, Attachment 3, MU-FT-1128 is an Inoperable Instrument.

Part 2 is incorrect. According to 1107-4, Electrical Distribution Panel Listing, MU-V-16C is powered from the 1A Engineered Safeguards Valves Bus, which is powered from the 1S 480V Engineered Safeguards Bus, which is in turn, powered from the 1E 4160V Engineered Safeguards Bus. Although Vital Bus "B" is powered from the 1B Engineered Safeguards Motor Control Center, which in turn is powered from the 1S 480V Engineered Safeguards Bus, it is powered off of a different breaker than that for MU-V-16C. Therefore, since the loss of Vital Bus Bravo does not affect MU-V-16C, the valve indication is available. Plausible if the candidate associates the "B" Train HPI Valve with Vital Bus "B".

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	057	AA2.07	
	Importance Rating	3.3		

Ability to determine or interpret the following as they apply to the Loss of Vital AC Instrument Bus: Valve indicator of charging pump suction valve from RWST.

Proposed Question: RO Question # 14

Technical Reference(s): 1107-4, p 34, 53, 263, Rev 225
OP-TM-AOP-016, p 23, Rev 76

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP016-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 7

55.43

Comments:

The KA is matched because the operator must determine the status of a Charging Pump suction valve from the Borated Water Storage Tank (RWST) upon a Loss of a Vital Bus event.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall bits of information to answer the question.

What MUST be known:
1. The status of MU-FT-1128 upon a loss of Vital Bus "B".
2. The status of MU-V-16C inidication on Console Right upon a loss of Vital Bus "B".

15 **ID: 1142073** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- An Auxiliary Operator reports the following from performing his rounds:
  - River water temperature is 96F.
  - ISPH Pump Bay water level is 278ft.

Given the above information, OP-TM-AOP-005, River Water Systems Failures, is entered based on ___(1)___ and the correct action to take is to ___(2)___.

A. (1) River Water temperature is too high
(2) initiate a plant shutdown IAW 1102-4, Power Operations

B. (1) River Water temperature is too high
(2) initiate OP-TM-EOP-001, Reactor Trip, and then trip all four Reactor Coolant Pumps

C. (1) ISPH Pump Bay water level is too low
(2) initiate a plant shutdown IAW 1102-4, Power Operations

D. (1) ISPH Pump Bay water level is too low
(2) initiate OP-TM-EOP-001, Reactor Trip, and then trip all four Reactor Coolant Pumps

Answer: A

Answer Explanation

A. **Correct.**

Part 1 is correct. The only entry criteria given in the stem that would lead to OP-TM-AOP-005, River Water Systems Failures, is river water temperature. IAW OP-TM-AOP-005:

1.0 ENTRY CONDITIONS
1D or 1E 4160V bus is energized and any of the following conditions exist:
ISPH pump bay level < 277' or
Failure of York Haven Dam or
River water temperature > 90°F or
Failure of all secondary and nuclear services river water pumps.

Part 2 is correct. Procedural guidance is given for the case in which river water temperature has risen to greater than 95F. IAW OP-TM-AOP-005, River Water Systems Failures:

3.4 IAAT river water temperature >/=95°F, then INITIATE a plant shutdown IAW 1102-4 "Power Operations" to be at CSD within 36 hours.

B. **Incorrect.**

Part 1 is correct. The only entry criteria given in the stem that would lead to OP-TM-AOP-005, River Water Systems Failures, is river water temperature. IAW OP-TM-AOP-005:

1.0 ENTRY CONDITIONS
1D or 1E 4160V bus is energized and any of the following conditions exist:
ISPH pump bay level < 277' or
Failure of York Haven Dam or
River water temperature > 90°F or
Failure of all secondary and nuclear services river water pumps.

Part 2 in incorrect. The procedural guidance within OP-TM-AOP-005 to initate OP-TM-EOP-001, Reactor Trip and secure Reactor Coolant Pumps is given only if all Nuclear Services River Water Pumps and Secondary Services River Water Pumps are inoperable or if river water level has lowered below 271". Neither case is described in the question. Plausible if the candidate is not familiar with the procedural action for high river water temperature. IAW OP-TM-AOP-005, River Water Systems Failures:

3.1 IAAT all NR and SR pumps are inoperable or ISPH pump bay water level < 271', then TRIP the reactor.
INITIATE OP-TM-EOP-001 Reactor Trip.
GO TO Section 4.3.

C. **Incorrect.**

Part 1 is incorrect. The only entry criteria given in the stem that would lead to OP-TM-AOP-005, River Water Systems Failures, is river water temperature. IAW OP-TM-AOP-005:

1.0 ENTRY CONDITIONS
1D or 1E 4160V bus is energized and any of the following conditions exist:
ISPH pump bay level < 277' or
Failure of York Haven Dam or
River water temperature > 90°F or
Failure of all secondary and nuclear services river water pumps.

Part 2 is correct. Procedural guidance is given for the case in which river water temperature has risen to greater than 95F. IAW OP-TM-AOP-005, River Water Systems Failures:

3.4 IAAT river water temperature ?>/=95°F, then INITIATE a plant shutdown IAW 1102-4 "Power Operations" to be at CSD within 36 hours.

D. **Incorrect.**

Part 1 is incorrect. The only entry criteria given in the stem that would lead to OP-TM-AOP-005, River Water Systems Failures, is river water temperature. IAW OP-TM-AOP-005:

1.0 ENTRY CONDITIONS
1D or 1E 4160V bus is energized and any of the following conditions exist:
ISPH pump bay level < 277' or
Failure of York Haven Dam or
River water temperature > 90°F or
Failure of all secondary and nuclear services river water pumps.

Part 2 in incorrect. The procedural guidance within OP-TM-AOP-005 to initate OP-TM-EOP-001, Reactor Trip and secure Reactor Coolant Pumps is given only if all Nuclear Services River Water Pumps and Secondary Services River Water Pumps are inoperable or if river water level has lowered below 271". Neither case is described in the question. Plausible if the candidate is not familiar with the procedural action for high river water temperature. IAW OP-TM-AOP-005, River Water Systems Failures:

3.1 IAAT all NR and SR pumps are inoperable or ISPH pump bay water level < 271', then TRIP the reactor.
INITIATE OP-TM-EOP-001 Reactor Trip.
GO TO Section 4.3.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	062 AA2.02	
	Importance Rating	2.9	

Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The cause of possible SWS loss.

Proposed Question: RO Question # 15

Technical Reference(s): OP-TM-AOP-005, p 1, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-005-PCO-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 10

55.43

Comments:
The KA is matched because the operator must demonstrate the ability to determine and interpret the loss of the River Water system as it applies to a loss of Nuclear Service Water. Although the Nuclear Service Water Pumps are operating in the stem, the question is considered a loss of Nuclear Service Water because river water temperature is too high and therefore the Nuclear Service Water System cannot properly perform its function. IAW OP-TM-AOP-0051, River Water Systems Failures Basis Document:

1.0 DESIGN OR LICENSING BASIS REQUIREMENTS
1.1 UFSAR:
Section 2.6.3 – Low Flow Studies
This section discusses the minimum expected river water levels and the ability of the river water pumps to access water at elevations of 270' or above. The UFSAR states that an emergency procedure will direct operational response to low river water levels.
Section 9.6 – Cooling Water Systems
River water temperatures of 95F and below will support operation of all safe shutdown equipment, with the exception of the Control Building chillers, which require river water temperatures of 92F or below.

Temperature was selected to avoid overlap with previous recent NRC examinations at Three Mile Island.

The question is at the Analysis cognitive level because the candidate must be able to analyze plant conditions and system requirements associated with the river water and nuclear service water systems. Based on the candidate's analysis determine appropriate action that must be procedurally taken.

What MUST be known:
1. Entry criteria for the River Water Systems Failures abnormal operating procedure.
2. abnormal operating procedure directions based on conditions given.

16 **ID: 1142048** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- Normal Plant lineups with the following exceptions:
  - MU-P-1B is OOS due to an oil leak.
  - MU-P-1A is running and selected for ES.
  - EG-Y-1B is OOS for a scheduled inspection.

Event:

- 230KV Bus 4 trips due to a fault on the Grid.

Given the above information, a failure of _________ would require declaration of a ONE HOUR Tech Spec timeclock?

A. the 1C 4160V bus to fast transfer

B. NR-P-1C to remain running

C. NS-P-1B to auto start

D. MU-P-1A to restart

Answer: D

Answer Explanation

A. Incorrect.
Plausible since this bus is initially lost during the event; however it does not affect any Tech Spec equipment.

B. Incorrect.
Plausible since this is a normally selected ES component; however it is supplied by EG-Y-1B in an emergency and is only a 72 hour timeclock.

C. Incorrect.
Plausible since this pump should have auto started during the event; however it would only be a one hour timeclock if it was selected for ES on the 1P 480V bus, which is not the normal equipment lineup.

D. Correct.
IAW Technical Specifications:

> TS 3.7.2.c: "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."
>
> TS 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 STANBY 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022 2.2.39	
	Importance Rating	3.9	

Loss of Reactor Coolant Makeup: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Proposed Question: RO Question # 16

Technical Reference(s): Tech Spec, p 3-43, Amend 278
Tech Spec, p 3-1, Amend 98

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO-14

Question Source: Bank # ID# 677575

Modified Bank #

New

Question History: Last NRC Exam: 08-01 (2010)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Comments:
The KA is matched because the operator must demonstrate the ability to analyze the effect of maintenance on the Makeup (HPI) System on the status of Tech Spec Limiting Conditions for Operations.

The question is at the Analysis cognitive level because the candidate must be able to analyze plant conditions and system requirements associated with the Makeup System, and based on the candidate's analysis, determine the appropriate Tech Spec Limiting Conditions for Operations.

What MUST be known:
1. What one-hour Tech Spec is associated with Engineered Safeguards equipment. 2. What equipment is associated with each train of the Engineered Safguards System.

17 **ID: 1149468** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Sequence of Events:

- A Loss of 1C ESV MCC occurs.
- RC-V-1, Pressurizer Spray Control Valve, has failed open.
- RC-V-3, Pressurizer Spray Line Isolation Valve, has failed partially open.
- Reactor Trip on low RCS Pressure, OP-TM-EOP-001, Reactor Trip, has been entered.
- Currently:
  - o RCS Pressure is 1825 psig and lowering.
  - o The cooldown rate is within the desired band.
  - o Pressurizer Level is 75 inches and lowering slowly.
  - o It is desired to raise RCS Pressure IAW Guide 8, RCS Pressure Control.

Given the above information, which ONE of the following correctly describes:
(1) The action to take to raise RCS Pressure, and
(2) The basis for the action?

A. (1) Ensure RC-P-1A, RC-P-1B, and either RC-P-1C or RC-P-1D are shutdown.
(2) To allow Pressurizer Heaters to raise RCS Pressure.

B. (1) Ensure RC-P-1A, RC-P-1B, and either RC-P-1C or RC-P-1D are shutdown.
(2) To minimize the effect of spray flow to the Pressurizer.

C. (1) Energize Pressurizer heater banks and/or adjust heater demand for the SCR controlled heaters.
(2) To allow Pressurizer Heaters to raise RCS Pressure.

D. (1) Energize Pressurizer heater banks and/or adjust heater demand for the SCR controlled heaters.
(2) To minimize the effect of spray flow to the Pressurizer.

Answer: B

Answer Explanation

A. Incorrect.
Part 1 is correct. Once OP-TM-EOP-001, Reactor Trip, is entered, Step 3.11 states:

3.11 Initiate Guide 8, "RCS Pressure Control".

Once OP-TM-EOP-010, Emergency Procedures Rules Guides and Graphs, is entered (Specifically Guide 8, RCS Pressure Control), Step A.2 states:

A. To RAISE RCS Pressure:
2. Ensure either RC-V-1, or RC-V-3 is closed.
RNO:
- Ensure both RC-P-1A and RC-P-1B are shutdown:
- Ensure either RC-P-1C or RC-P-1D are shutdown.

Part 2 is incorrect. Upon a loss of the 1C ESV MCC, the Pressurizer Lo Lo Level interlock loses power and cannot be overriden, and the Pressurizer heaters will not energize automatically or Manually. Therefore, lowering Pressurizer spray flow, while minimizing RCS pressure reduction, will not be less of a pressure reduction than the pressure rise of the Pressurizer heaters. Plausible if the candidate does not recognize that the Pressurizer low level interlock has actuated. IAW OP-TM-AOP-013, Loss of 1D 4160V Bus:

Table 1: Effects of Loss of 1C ESV MCC
Loss of all Pressurizer heaters due to loss of power to the LO LO level interlock. Auto and Manual control not functional, ICS/NNI loss of power keyswitch has no effect to override low level interlock.

B. Correct.
Part 1 is correct. Once OP-TM-EOP-001, Reactor Trip, is entered, Step 3.11 states:

3.11 Initiate Guide 8, "RCS Pressure Control".

Once OP-TM-EOP-010, Emergency Procedures Rules Guides and Graphs, is entered (Specifically Guide 8, RCS Pressure Control), Step A.2 states:

A. To RAISE RCS Pressure:
2. Ensure either RC-V-1, or RC-V-3 is closed.
RNO:
- Ensure both RC-P-1A and RC-P-1B are shutdown:
- Ensure either RC-P-1C or RC-P-1D are shutdown.

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

Step A2. The step intent is to minimize the effect of Pressurizer spray if both the spray and block valves are open. Shutdown of all but one RC pump will minimize the spray flow to the Presusurizer and allow pressure recovery efforts to continue.

C. Incorrect.
Part 1 is incorrect. With a loss of the 1C ESV MCC, the Pressurizer heaters will not energize. Therefore, lowering Pressurizer spray flow, while minimizing RCS pressure reduction, will not be less of a pressure reduction than the pressure rise of the Pressurizer heaters. Plausible if the candidate does not recognize that the Pressurizer low level interlock has actuated. IAW OP-TM-EOP-0101, Emergency Procedure Rules Guides and Graphs Basis Document:

> Step A5. The step intent is to determine if Pressurizer heaters are available. The Pressurizer heater low level interlock is actuated when level is below 80 inches.

Part 2 is incorrect. Upon a loss of the 1C ESV MCC, the Pressurizer Lo Lo Level interlock loses power and cannot be overriden, and the Pressurizer heaters will not energize Automatically or Manually. Therefore, lowering Pressurizer spray flow, while minimizing RCS pressure reduction, will not be less of a pressure reduction than the pressure rise of the Pressurizer heaters. Plausible if the candidate does not recognize that the Pressurizer low level interlock has actuated. IAW OP-TM-AOP-013, Loss of 1D 4160V Bus:

> Table 1: Effects of Loss of 1C ESV MCC
>
> Loss of all Pressurizer heaters due to loss of power to the LO LO level interlock. Auto and Manual control not functional, ICS/NNI loss of power keyswitch has no effect to override low level interlock.

D. Incorrect.
Part 1 is incorrect. With a loss of the 1C ESV MCC, the Pressurizer heaters will not energize. Therefore, lowering Pressurizer spray flow, while minimizing RCS pressure reduction, will not be less of a pressure reduction than the pressure rise of the Pressurizer heaters. Plausible if the candidate does not recognize that the Pressurizer low level interlock has actuated. IAW OP-TM-EOP-0101, Emergency Procedure Rules Guides and Graphs Basis Document:

> Step A5. The step intent is to determine if Pressurizer heaters are available. The Pressurizer heater low level interlock is actuated when level is below 80 inches.

Part 2 is incorrect. This is the correct basis for minimizing the number of operating RCP's. Plausible if the candidate believes that Pressurizer heaters may overcome the spray flow.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027 2.4.18	
	Importance Rating	3.3	

Pressurizer Pressure Control System (PZR PCS) Malfunction: Knowledge of the specific bases for EOPs.

Proposed Question: RO Question # 17

Technical Reference(s): OP-TM-EOP-001, p 7, Rev 12
OP-TM-EOP-010, p 18, Rev 17
OP-TM-EOP-0101, p38, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-G08-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 2

55.43

Comments:
The KA is matched because the operator must demonstrate the knowledge of a specific basis within Emergency Operating Procedures associated Pressurizer Pressure Control Malfunctions.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the conditions given and determine the correct action to take, as well as recalling the basis for why the action is taken.

What MUST be known:
1. What the effect is on Pressurizer heaters upon a loss of the 1C ES Valves MCC. 2. What the Guide 8 procedural guidance is to raise RCS pressure upon a stuck open Spray Valve. 3. The reason for the Guide 8 action to secure 3 out of 4 Reactor Coolant Pumps.

18 **ID: 1142450** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- Loss of Offsite Power occurs.
- EG-Y-1A and EG-Y-1B fail to start.
- EG-Y-4 trips as soon as it is started and will not restart.

Given the above information, which of the following systems require Tech Spec entry?

1. Core Flood System
2. Heat Sink Protection System
3. Emergency Feedwater System

A. 2, ONLY.

B. 3, ONLY.

C. 1 and 2.

D. 1 and 3.

Answer: B

Answer Explanation

The key to answering the question is to assess all three systems independently against a Station Blackout and determine how each system is affect by a Station Blackout. System explanations:

Core Flood System: The Core Flood System is a passive system. The Core Flood Tank discharge valves, while electrically powered, are locked open with the associated breakers opened. This means that a Station Blackout would not affect the Core Flood System. IAW TQ-TM-104-213-C001, Core Flood System:

- The Core Flood System is a passive Engineered Safeguards System designed to inject borated water into the reactor vessel to:
  - Prevent gross fuel damage following a loss of coolant accident (LOCA)
  - Minimize Zirconium cladding – water reaction
  - Maintain fuel integrity
  - Ensure a coolable core geometry exists
- The electrical requirements for the Core Flood System are limited to motor operated valves, instrumentation and controls.
- The Core Flood System is self-contained and selfactuated to perform its Emergency Core Cooling System function without relying on electrical power sources.
- Electrical Distribution
  - CF-V-1A and CF-V-1B are powered from 1C ES Valves MCC.
  - The 1C ES Valve MCC also has an AC transfer switch that allows the MCC to be fed from either the 1P or 1S 480 volt bus.
  - 1C ES Valves MCC breaker is open and deenergized when the reactor is critical
  - The breaker is opened and an Equipment Status tag is hung at the breaker, as an administrative control to ensure the valves position is not altered which would isolate the respective tank thereby reducing Emergency Core Cooling System availability.
- ES Standby Mode: CF-V-1A & 1B all open with Bkr. open

Emergency Feedwater System: Two of the three Emergency Feedwater Pumps are electrically powered. Upon a Station Blackout condition, only EF-P-1 (steam driven) will be running. This is entry criteria into Tech Spec 3.4. IAW Tech Spec 3.4:

3.4 DECAY HEAT REMOVAL (DHR) CAPABILITY

3.4.1 Reactor Coolant System (RCS) temperature greater than 250 degrees F.

3.4.1.1 Three independent Emergency Feedwater (EFW) Pumps and two redundant flowpaths to each Once Through Steam Generator (OTSG) shall be OPERABLE with:

Two EFW Pumps, each capable of being powered from an OPERABLE emergency bus, and one EFW Pump capable of being powered from two OPERABLE main steam supply paths.

With more than one EFW Pump or both flowpaths to either OTSG inoperable, initiate action immediately to restore at least two EFW Pumps and one flowpath to each OTSG:

Heat Sink Protection System: The Heat Sink Protection System has two trains. Both trains have Vital AC power supplies as their primary power supply. However, there is a BOP backup power supply for each train. IAW TQ-TM-104-644-C001, Heat Sink Protection System:

- The Heat Sink Protection System instruments are all vital powered through their respective Heat Sink Protection System channel or signal conditioning cabinet.
    - Train A contains 2 power supplies
        - Power Supply 1 (PS1) fed from VBA breaker #11
        - Power Supply 2 (PS2) normally fed from VBA breaker #8
        - Backup feed is from a 24 VDC supply (PS3) powered from 115 VAC panel D-16.
    - Train B contains two power supplies (similar to Train A)
        - PS1 input: 120 VAC from VBB breaker #26
        - PS2 input: 120 VAC from VBB breaker #20
        - Backup: 24 VDC from PS3
- Panel D-16
    - Powered from 1B Reactor Plant MCC, which is powered from 1L 480V Reactor Plant switchgear breaker #4
    - Supplies 24 VDC backup power (PS3)
    - A "Backup 24 VDC Avail." indicating light is located on the outside of each train cabinet at the bottom of the existing light column.
    - The light is lit if the 24 VDC power supply (fed from 120 VAC Panel D-16) is energized and ready to backup the normal train power supply source (VBA or VBB).
    - If the train power supply is using the DC backup a "Battery B/U" indicating light on the power supply in the train cabinet will be lit.
    - Panel D-16 also provides 28 VDC to the local alarm panel and console indicating lights.

That Backup power supply is lost upon a Station Blackout and therefore, IAW Tech Spec 1.3, HSPS could be considered inoperable. Tech Spec 1.3 states:

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).

However, Tech Spec 3.5.1.9.1 states:

3.5.1.9 The reactor shall not be in the Startup mode or in a critical state unless both HSPS actuation logic trains associated with the Functional units listed in Table 3.5-1 are operable except as provided in Table 3.5-1,D.

3.5.1.9.1 With one HSPS actuation logic train inoperable, restore the train to OPERABLE or place the inoperable device in an actuated state within 72 hours or be in HOT SHUTDOWN within the next 12 hours. With both HSPS actuation logic trains inoperable, restore one train to OPERABLE within 1 hour or be in HOT SHUTDOWN within the next 6 hours.

Since the reactor is neither in a critical state nor a startup mode, there is no Tech Spec LCO action required.

A. Incorrect.
See the Heat Sink Protection System explanations above.
Plausible if the candidate does not recognize that the reactor is not in a critical state nor a startup mode and therefore there is no LCO action associated with HSPS.

B. Correct.
See the Core Flood System, Heat Sink Protection System, and Emergency Feedwater System explanations above.

C. Incorrect.
See the Core Flood System and Heat Sink Protection System explanations above.
Plausible if the candidate does not recognize that the breakers associated with the key Core Flood Tank System are open and therefore the system is not affected upon a Station Blackout and misses that the Emergency Feedwater system does have Tech Spec entry criteria.

D. Incorrect.
See the Core Flood System and Emergency Feedwater System explanations above.
Plausible if the candidate does not recognize that the breakers associated with the key Core Flood Tank System are open and therefore the system is not affected upon a Station Blackout.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	055 2.2.42	
	Importance Rating	3.9	

Station Blackout: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Proposed Question: RO Question # 18

Technical Reference(s): Tech Spec, p 3-25, Amend 242
Tech Spec, p 1-2, Amend 175

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-020-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	8
	55.43	

Comments:
The KA is matched because the operator must identify system parameters that are entry-level conditions for Technical Specifications associated with a Station Blackout.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall Tech Spec entry criteria.

What MUST be known:
1. The Tech Spec entry criteria associated with the Core Flood System. 2. The Tech Spec entry criteria associated with the Heat Sink Protection System. 3. The Tech Spec entry criteria associated with the Emergency Feedwater System.

19 **ID: 1146787** **Points: 1.00**

Plant Conditions:

- The plant is in a Refueling Shutdown.
- Fuel offload is 1/3 of the way complete.
- Spent Fuel Pool Level is 344.6 feet and steady.

Event:

- A major earthquake is felt.
- Spent Fuel Pool Level is currently 342 feet and lowering.
- PLB-2-9, Spent Fuel Pool A Level Low, is in alarm.
- PLB-2-10, Spent Fuel Pool B Level Low, is in alarm.
- The Fuel Transfer Canal level is lowering consistent with the Spent Fuel Pools.

Given the above information, which of the following describes the MINIMUM action required to arrest the lowering of the Fuel Transfer Canal Level?

A. Place one train of Spent Fuel Cooling in Service.

B. Place both trains of Spent Fuel Cooling in Service.

C. Ensure closed either Fuel Transfer Canal Isolation Valve, FH-V-1A or FH-V-1B.

D. Ensure closed both Fuel Transfer Canal Isolation Valves, FH-V-1A and FH-V-1B.

Answer: D

Answer Explanation

A. Incorrect.

Plausible if the candidate is not familiar with the procedure routing of OP-TM-AOP-035, Loss of Spent Fuel Cooling, or does not understand the reasons for each of the steps taken in the procedure. OP-TM-AOP-035 directs actions based on low level or high temperature.

Step 3.5 is an IAAT step that directs routing to Section 4.0. Therefore, section 3.0 is exited. If the candidate does not recognize this, then Section 3.0 will be continued. IAW OP-TM-AOP-035:

> 3.11 If Spent Fuel Pool temperature is > 160°, then PLACE both Spent Fuel Cooling trains in service IAW 1104-6.

B. Incorrect.

Plausible if the candidate is not familiar with the procedure routing of OP-TM-AOP-035, Loss of Spent Fuel Cooling, or does not understand the reasons for each of the steps taken in the procedure. OP-TM-AOP-035 directs actions based on low level or high temperature.

Step 3.5 is an IAAT step that directs routing to Section 4.0. Therefore, section 3.0 is exited. If the candidate does not recognize this, then Section 3.0 will be continued. Furthermore, this distractor is not correct as step 3.11 directs placing both Spent Fuel Cooling Trains in service. IAW OP-TM-AOP-035:

> 3.11 If Spent Fuel Pool temperature is > 160°, then PLACE both Spent Fuel Cooling trains in service IAW 1104-6.

C. Incorrect.

Plausible if the candidate understands the reason for the step but is unfamiliar with the physical alignment of the Fuel Handling System and believes that FH-V-1A and FH-V-1B are in series. This logic would lend to the conclusion that, like Containment Isolation Valves, only one of the two need to be closed. However, IAW OP-TM-AOP-035, Loss of Spent Fuel Cooling:

> 4.6 If irradiated fuel is in the Reactor Vessel, then ENSURE the following valves are closed:
> A. FH-V-1A (N end of SF pool)
> B. FH-V-1B (N end of SF pool)

D. Correct.

The routing through OP-TM-AOP-035, Loss of Spent Fuel Cooling, is as follows: First, Step 3.5 is an IAAT statement that is met based on conditions given in the stem:

> 3.5 IAAT Spent Fuel Pool level is < 343' 6" (low level alarm) then GO TO Section 4.0.

Once in section 4.0, Step 4.6 is applicable based on conditions given in the stem (only 1/3 of the fuel has been removed from the core, indicating that there is irradiated fuel in the core still):

> 4.6 If irradiated fuel is in the Reactor Vessel, then ENSURE the following valves are closed:
> A. FH-V-1A (N end of SF pool)
> B. FH-V-1B (N end of SF pool)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	A08 AK1.1	
	Importance Rating	3.7	

Knowledge of the operational implications of the following concepts as they apply to the (Refueling Canal Level Decrease): Components, capacity, and function of emergency systems.

Proposed Question: RO Question # 19

Technical Reference(s): OP-TM-AOP-035, p 1,9, Rev 4

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-035-PCO-4

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 12

55.43

Comments:
The KA is matched because the operator must demonstrate knowledge of components and functions of the components associated with a Refueling Canal Level Decrease.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall the physical layout of a system and also recall a procedure step.

What MUST be known:
1. The procedural guidance contained within the Loss of Spent Fuel Cooling abnormal operating procedure. 2. The criteria that must be exist to meet conditional procedure steps within the Loss of Spent Fuel Cooling abnormal operating procedure.

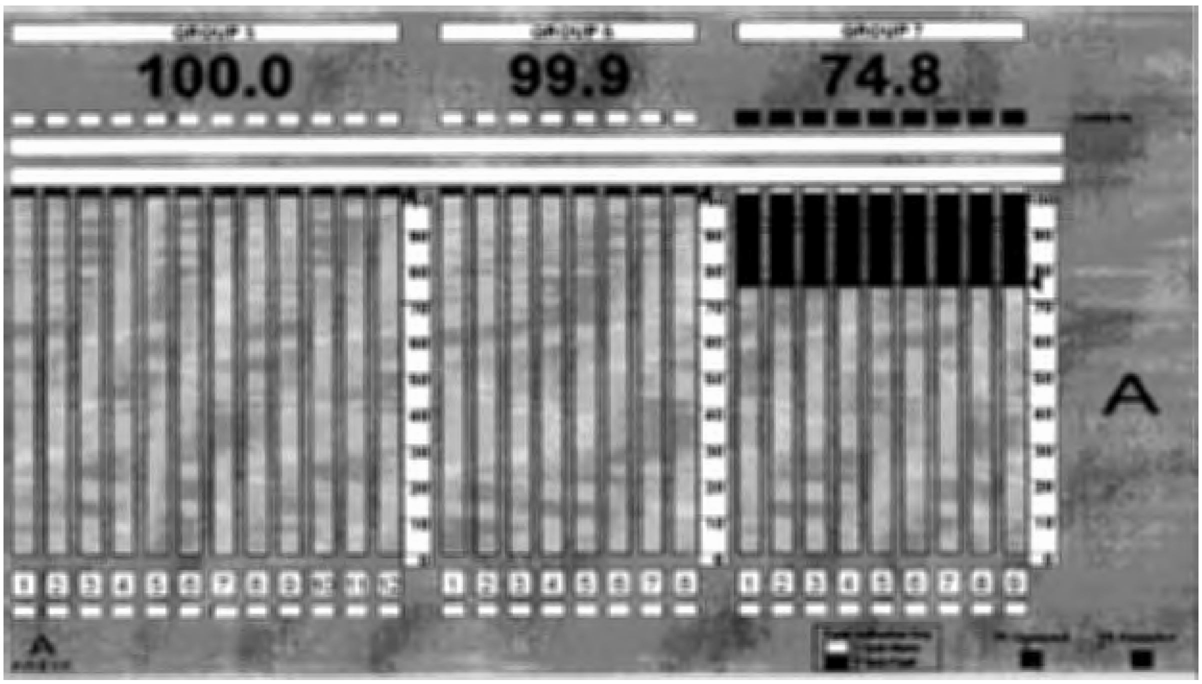
20 **ID: 1142080** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor 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

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		

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

Proposed Question: RO Question # 20

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 # ID# 907081

Modified Bank #

New

Question History: Last NRC Exam: 2014

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

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?

21 **ID: 1142097** **Points: 1.00**

At full power, which ONE of the following represents a violation of containment integrity?

A. Both doors of the personnel hatch/airlock are OPEN.

B. The isolation for a relief valve located between Containment isolation valves is LOCKED OPEN.

C. A drain valve located between Containment isolation valves is capped, CLOSED, with a sign posted.

D. Both the motor operated (internal RB) and air operated (external RB) pressurizer sample valves are OPEN.

Answer: A

Answer Explanation

A. Correct.
Containment integrity per TS 3.6 allows only one door to be open.

3.6.12.a: At least one door in each of the personnel or emergency air locks shall be closed and sealed during personnel passage through these air locks.

B. Incorrect.
The isolation valve for a relief valve located between Containment Isolation valves is OPEN. For the relief valve to be operable the isolation has to be open. The position of the relief valve isolations on the enclosure in 1101-3 is LOCKED OPEN.
Distractor is plausible for an Operator unfamiliar with the requirements of 1101-3 Containment Integrity Access Limits.

C. Incorrect.
A drain valve located between Containment isolation valves is capped and closed- 1101-3, 2.6b Criteria for locking valves: Manual Containment Isolation Valves with caps/flanges directly downstream will be closed and capped and a sign to that effect will be prominently posted thereon. The position of drain valves in the 1101-3 enclosure is sign posted/closed/capped.
Distractor is plausible for an Operator unfamiliar with the requirements of 1101-3 Containment Integrity Access Limits.

D. Incorrect.
Both the motor operated (internal RB) and air operated (external RB) pressurizer sample valves are OPEN- These valves are allowed to be open and controlled by an administrative procedure, as allowed by the CONTAINMENT INTEGRITY definition.
Distractor is plausible in that BOTH valves are open, but this is allowed for any CIV as long as CIVs are operable and system is not breached.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	069 AK2.03	
	Importance Rating	2.8	

Knowledge of the interrelations between the Loss of Containment Integrity and the following: Personnel access hatch and emergency access hatch.

Proposed Question: RO Question # 21

Technical Reference(s): Tech Spec, p 3-41b, Amend 201

Proposed References to be provided to applicants during examination: None

Learning Objective: 244-GLO-14

Question Source: Bank # ID# 468905

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

Comments:
The KA is matched because the operator must identify the Loss of Containment Integrity and the RB access hatches.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall Tech Spec entry criteria.

What MUST be known:
1. What the Tech Spec requirements are to maintain containment integrity.

22 **ID: 1142088** **Points: 1.00**

Plant Conditions:

- A Plant Startup is in progress.
- Reactor power is 6% rated thermal power.
- Turbine Bypass Valves MS-V-3A/B/C/D/E/F are in AUTOMATIC.

Event:

- Loss of all CW Pumps due to flooding inside the CW Pump House.

Given the above information and assuming no operator action, identify the ONE selection below that describes the pressure control scheme selected for control of the Main Steam atmospheric dump valves MS-V-4A/B.

A. Manual control using the Back-Up Loaders.

B. 895 psig setpoint, proportional plus bias pressure control.

C. 1010 psig setpoint, proportional plus bias pressure control.

D. 1026-1052 proportional control high pressure relief.

Answer: B

Answer Explanation

A. Incorrect.

Back-up Loaders are only used following loss of ICS/NNI Auto Power. Distracter is plausible because this would be a correct answer if loss of ICS Auto Power was lost. Additional plausibility is merited because the TBVs will be latched closed due to loss of vacuum. IAW OP-TM-411-000, Main Steam OTSG:

6.1.3 Loss of ICS Auto Power

1. Turbine Bypass Valves (MS-V-3A-F) revert to HAND and respond only to demand changes from the ICS "MS-V-3D, E, F or 4A (MS-V-3A, B, C or 4B)" HAND/AUTO control stations.
2. Atmospheric Dump Valves (MS-V-4A/B) automatically transfer to "BU Loader" and may be positioned as the operator desires from the "MS-V-4A(B) BACKUP CTRL" stations.

B. Correct.

TBV control scheme and setpoint transfer to the ADVs on loss of CW, low vacuum. IAW OP-TM-411-000, Main Steam OTSG:

6.1.2 Loss of Condenser Vacuum

1. Turbine Bypass Valves (MS-V-3A-F) are closed and latched. The ICS "MS-V-3D, E, F or 4A (MS-V-3A, B, C or 4B)" HAND/AUTO control stations have no effect on their position.
2. Atmospheric Dump Valves operated in auto control at Turbine Hdr Pressure setpoint plus appropriate bias (+10, +75, +125 psig) or open at 1040 whichever demand is greater.
3. Atmospheric Dump Valves (MS-V-4A/B) operated in HAND respond only to demand changes from the ICS "MS-V-3D, E, F or 4A (MS-V-3A, B, C or 4B)" HAND/AUTO control stations.
4. Operator may select Atmospheric Dump Valves control from "MS-V-4A(B) BACKUP CTRL" stations by depressing the "B/U LOADER" pushbutton and position valves as desired.

Furthermore, IAW TQ-TM-104-411-C001, Main Steam:

- 895 psig Automatic Control Setpoint
    - This setpoint provides a 10 psi control band allowance to prevent Turbine Bypass Valves and Turbine Control Valves from fighting each other during Turbine startup and low load operation when both flow paths are in operation at the same time.

C. Incorrect.
Stem conditions do not present an automatic reactor trip. Distracter is plausible because this would be the setpoint and control scheme for the ADVs if the reactor tripped under these conditions. IAW OP-TM-411-000, Main Steam OTSG:

6.1.2 Loss of Condenser Vacuum

1. Turbine Bypass Valves (MS-V-3A-F) are closed and latched. The ICS "MS-V-3D, E, F or 4A (MS-V-3A, B, C or 4B)" HAND/AUTO control stations have no effect on their position.
2. Atmospheric Dump Valves operated in auto control at Turbine Hdr Pressure setpoint plus appropriate bias (+10, +75, +125 psig) or open at 1040 whichever demand is greater.
3. Atmospheric Dump Valves (MS-V-4A/B) operated in HAND respond only to demand changes from the ICS "MS-V-3D, E, F or 4A (MS-V-3A, B, C or 4B)" HAND/AUTO control stations.
4. Operator may select Atmospheric Dump Valves control from "MS-V-4A(B) BACKUP CTRL" stations by depressing the "B/U LOADER" pushbutton and position valves as desired.

Furthermore, IAW TQ-TM-104-411-C001, Main Steam:

f) 1010 psig Automatic Control Setpoint

- This setpoint is to prevent excessive Pressurizer Level decrease on reactor trip.
- Raising the OTSG pressure setpoint on a reactor trip limits RCS Cooldown and shrink, since this sets post trip OTSG saturated steam temperature at 555°F.

D. Incorrect.
This control scheme is blocked from the ADVs if low vacuum or loss of CW condition exists. Distracter is plausible because this is the ADV control scheme used in the initial plant conditions presented in the question stem. IAW OP-TM-411-000, Main Steam OTSG:

6.1.1 Normal Operations

1. Turbine Bypass Valves (MS-V-3A-F) operated in AUTO control at Turbine Hdr Pressure setpoint plus appropriate bias (+10, +75, +125 psig) or open at 1040 whichever is the greater open signal.
2. Turbine Bypass Valves (MS-V-3A-F) operated in HAND respond only to demand changes from the ICS "MS-V-3D, E, F or 4A (MS-V-3A, B, C or 4B)" HAND/AUTO control stations.
3. Atmospheric Dump Valves (MS-V-4A/B) will open at 1026 - 1052 psig.
4. Operator may select Atmospheric Dump Valves control from "MS-V-4A(B) BACKUP CTRL" stations by depressing the "B/U LOADER" pushbutton and position valves as desired.

Furthermore, IAW TQ-TM-104-411-C001, Main Steam:

g) 1040 psig Automatic (Fixed) Control Setpoint

- This circuit provides an independent high-pressure relief that will open proportional to OTSG pressure.
- Transfers to MS-V-4A/4B controls on low vacuum or loss of CW Pumps.
- 1026-1052 psig Automatic (Fixed) Proportional Control Setpoint (ADV only)
- ADVs will modulate open proportionally to control steam flow as a function of pressure in the range 1026 to 1052 psig.
- This circuit is never used by the TBVs, and is blocked (not used) by the ADVs if Main Condenser vacuum is <23" Hg or less than 2 CW Pumps are running.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	051 AK3.01	
	Importance Rating	2.8	

Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capability upon loss of condenser vacuum.

Proposed Question: RO Question # 22

Technical Reference(s): TQ-TM-104-411-C001, p 29-30, Rev 7
OP-TM-411-000, p 10-11, Rev 19

Proposed References to be provided to applicants during examination: None

Learning Objective: 621-GLO-8

Question Source: Bank # ID# 371322

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	7
	55.43	

Comments:
The KA is matched because the operator must demonstrate knowledge of the steam dump capability during a Loss of Condenser Vacuum

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must identify setpoints based on the conditions given.

What MUST be known:
1. What the setpoint is for the Atmospheric Dump Valves under given conditions. 2. The relationship between the Turbine Bypass Valves and the Atmospheric Dump Valves upon a loss of Circulating Water Pumps.

23 **ID: 1144393** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- Boric Acid Pumps, CA-P-1A and CA-P-1B, are out of service.

Event:

- A LOCA and reactor trip occur.
- MU-V-14A and MU-V-14B fail to open and cannot be opened manually.
- Rod 3-4 is 100% withdrawn.
- Reactor Power is 1% and lowering.
- Symptom Check is negative.

Based on the above information, a/the ___(1)___ will be the Emergency Boration source, and the Boric Acid Injection Switch will be placed in the ___(2)___ position.

A. (1) Boric Acid Mix Tank (BAMT)
(2) off

B. (1) Boric Acid Mix Tank (BAMT)
(2) inject

C. (1) Reclaimed Boric Acid Tank (RBAT)
(2) off

D. (1) Reclaimed Boric Acid Tank (RBAT)
(2) inject

Answer: D

Answer Explanation

A. Incorrect.

Part 1 is incorrect. Guide 1 gives the operator three options for Emergency Boration Backup methods. One of the three methods (using the Boric Acid Mix Tank) is not possible since CA-P-1A and CA-P-1B are out of service. That leaves two Emergency Boration backup methods: the "A" Reclaimed Boric Acid Tank (RBAT) and the "B" Reclaimed Boric Acid Tank RBAT. Plausible if the candidate is not familiar with the components associated with the Reclaimed Boric Acid Tanks and the Boric Acid Mix Tank.

Part 2 is incorrect. Plausible if the candidate correctly associates the Boric Acid Injection Switch with the Reclaimed Boron Acid Tanks but believes that the Boric Acid Mix Tank is the correct Tank that will be used.

B. Incorrect.

Part 1 is incorrect. Guide 1 gives the operator three options for Emergency Boration Backup methods. One of the three methods (using the Boric Acid Mix Tank) is not possible since CA-P-1A and CA-P-1B are out of service. That leaves two Emergency Boration backup methods: the "A" Reclaimed Boric Acid Tank (RBAT) and the "B" Reclaimed Boric Acid Tank RBAT. Plausible if the candidate is not familiar with the components associated with the Reclaimed Boric Acid Tanks and the Boric Acid Mix Tank.

Part 2 is correct. Guide 1 gives the operator three options for Emergency Boration Backup methods. One of the three methods (using the Boric Acid Mix Tank) is not possible since CA-P-1A and CA-P-1B are out of service. That leaves two Emergency Boration backup methods, both involving the same Boric Acid Injection Switch. IAW OP-TM-EOP-010, Emergency Procedure Rules Guides and Graphs, Guide 1, Emergency Boration Backup Methods:

B. If Backup Emergency Boration source is the RBAT, then perform the following:
   1. Position the "Boric Acid Injection" switch to "INJECT" (opens WDL-V-61).

C. Incorrect.

Part 1 is correct. Upon a reactor trip, the Immediate Manual Actions of OP-TM-EOP-001 have been performed, as evidenced by the completion of a symptom check. IAW OP-TM-EOP-001, Reactor Trip:

3.3 Verify control rod groups 1 through 7 are fully inserted.
   RNO Initiate Rule 5, "Emergency Boration".

Rule 5 gives the operator three options to perform Emergency Boration. Two of the three methods are not possible since MU-V-14A and MU-V-14B will not open. IAW OP-TM-EOP-010, Emergency Procedure Rules Guides and Graphs, Rule 5, Emergency Boration:

3. Perform one of the following:
   - Open MU-V-14A
   - Open MU-V-14B
   - Perform Guide 1 "Emergency Boration Backup Methods".

Guide 1 gives the operator three options for Emergency Boration Backup methods. One of the three methods (using the Boric Acid Mix Tank) is not possible since CA-P-1A and CA-P-1B are out of service. That leaves two Emergency Boration backup methods: the "A" Reclaimed Boric Acid Tank (RBAT) and the "B" Reclaimed Boric Acid Tank RBAT.

Part 2 is incorrect. Plausible if the candidate incorrectly associates the Boric Acid Injection Switch with the Boric Acid Mix Tank. IAW OP-TM-EOP-010, Emergency Procedure Rules Guides and Graphs, Guide 1, Emergency Boration Backup Methods:

B. If Backup Emergency Boration source is the RBAT, then perform the following:
   1. Position the "Boric Acid Injection" switch to "INJECT" (opens WDL-V-61).

D. Correct.
Part 1 is correct. Upon a reactor trip, the Immediate Manual Actions of OP-TM-EOP-001 have been performed, as evidenced by the completion of a symptom check. IAW OP-TM-EOP-001, Reactor Trip:

3.3 Verify control rod groups 1 through 7 are fully inserted.
   RNO Initiate Rule 5, "Emergency Boration".

Rule 5 gives the operator three options to perform Emergency Boration. Two of the three methods are not possible since MU-V-14A and MU-V-14B will not open. IAW OP-TM-EOP-010, Emergency Procedure Rules Guides and Graphs, Rule 5, Emergency Boration:

3. Perform one of the following:
   - Open MU-V-14A
   - Open MU-V-14B
   - Perform Guide 1 "Emergency Boration Backup Methods".

Guide 1 gives the operator three options for Emergency Boration Backup methods. One of the three methods (using the Boric Acid Mix Tank) is not possible since CA-P-1A and CA-P-1B are out of service. That leaves two Emergency Boration backup methods: the "A" Reclaimed Boric Acid Tank (RBAT) and the "B" Reclaimed Boric Acid Tank RBAT.

Part 2 is correct. Guide 1 gives the operator three options for Emergency Boration Backup methods. One of the three methods (using the Boric Acid Mix Tank) is not possible since CA-P-1A and CA-P-1B are out of service. That leaves two Emergency Boration backup methods, both involving the same Boric Acid Injection Switch. IAW OP-TM-EOP-010, Emergency Procedure Rules Guides and Graphs, Guide 1, Emergency Boration Backup Methods:

B. If Backup Emergency Boration source is the RBAT, then perform the following:
   1. Position the "Boric Acid Injection" switch to "INJECT" (opens WDL-V-61).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	024 AA1.19	
	Importance Rating	3.2	

Ability to operate and / or monitor the following as they apply to Emergency Boration: Makeup control system selector switch for CVCS isolation valve.

Proposed Question:	RO Question # 23
Technical Reference(s):	OP-TM-EOP-010, p 11,13, Rev 17 OP-TM-EOP-001, p 5, Rev 12

Proposed References to be provided to applicants during examination: None

Learning Objective: 621-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	6
	55.43	

Comments:
The KA is matched because the operator must demonstrate the ability to monitor a makeup control switch for a CVCS isolation valve. Although WDL-V-61 is a Liquid Waste Disposal Valve, it it the isolation valve that connects to the Makeup (CVCS) System.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze conditions given and then decide on what components will be used/positioned.

What MUST be known:
1. The available methods to perform Emergency Boration. 2. The procedural guidance contained within Guide 1.

24 **ID: 1144399** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Sequence of Events:

- A Loss of ICS Auto Power occurs.
- A trip of all three Condensate Booster Pumps occurs.
- The reactor trips.
- A Main Steam Safety Valve has failed to reseat.

Given the above information:

(1) What is the source of water to the OTSG's, and

(2) What is the consequence if the Main Steam Safety Valve is not reseated IAW Guide 6, OTSG Pressure Control?

A. (1) Main Feedwater.
(2) A cooldown will occur since feedwater and steam flows are not matched.

B. (1) Main Feedwater.
(2) The Atmospheric Dump Valves will automatically adjust to match feedwater and steam flows.

C. (1) Emergency Feedwater.
(2) A cooldown will occur since feedwater and steam flows are not matched.

D. (1) Emergency Feedwater.
(2) The Turbine Bypass Valves will automatically adjust to match feedwater and steam flows.

Answer: C

Answer Explanation

A. Incorrect.

Part 1 is incorrect. When the Condensate Booster Pumps trip, the counting circuit recognizes zero Booster Pumps and will trip both Main Feedwater Pumps. This Loss of Main Feedwater will result in HSPS starting Emergency Feedwater. Additionally, since Condenser vacuum is present, the Turbine Bypass Valves will be used to control OTSG pressure. Atmospheric Dump Valves are only used when the Turbine Bypass Valves cannot be used. Plausible if the candidate does not recognize that a total Loss of Main Feedwater has occurred.

Part 2 is correct. IAW OP-TM-EOP-0101, Guide 6 Basis:

Step 6 The step intent is to reseat any leaking MSSV.

Following reactor trip, OTSG pressure goes to saturation for the existing RCS temperature. If pre-trip reactor power was > ~15 %, MSSV actuation is likely and affords the opportunity for MSSV failure. Industry experience has shown the most likely failure mechanism is failure to reseat following actuation and blowdown. The narrow band between accumulation and blowdown values contributes to this mechanism, but continued lowering of steam pressure is usually successful in reseating the MSSV. For this reason, guidance on lowering OTSG pressure to reseat the valve is provided. Main steam pressure is maintained greater than 625 psig to prevent actuation of the HSPS feature and MSSV reseat usually occurs prior to header pressure lowering to this value. If OTSG pressure is quickly lowered and allowed to recover, RCS temperature impact is minimized but should not be allowed to go below 525F or exceed the Technical Specification cooldown rates during this effort. Failure to reseat following this guidance will result in cooldown once decay heat has lowered and if the safety valve cannot be reseated (including maintenance efforts if possible), isolation in accordance with Rule 3 is required. If OTSG pressure is reduced to 625 psig, it will need to be quickly recovered to > 800 psig to prevent RCS Tc from dropping below 525F.

B. Incorrect.

Part 1 is incorrect. When the Condensate Booster Pumps trip, the counting circuit recognizes zero Booster Pumps and will trip both Main Feedwater Pumps. This Loss of Main Feedwater will result in HSPS starting Emergency Feedwater. Additionally, since Condenser vacuum is present, the Turbine Bypass Valves will be used to control OTSG pressure. Atmospheric Dump Valves are only used when the Turbine Bypass Valves cannot be used. Plausible if the candidate does not recognize that a total Loss of Main Feedwater has occurred.

Part 2 is incorrect. With a Loss of ICS Auto Power, the Atmospheric Dump Valves will be controlled in HAND (manually). Therefore, no automatic adjustments can be made. Additionally, since there is vacuum, the TBV's will be used to control OTSG pressure. Plausible if the candidate does not recognize that the valves are not under AUTO control and/or that vacuum exists.

C. Correct.
Part 1 is correct. When the Condensate Booster Pumps trip, the counting circuit recognizes zero Booster Pumps and will trip both Main Feedwater Pumps. This Loss of Main Feedwater will result in HSPS starting Emergency Feedwater. Additionally, since Condenser vacuum is present, the Turbine Bypass Valves will be used to control OTSG pressure. Atmospheric Dump Valves are only used when the Turbine Bypass Valves cannot be used. IAW TQ-TM-104-421-C001, Condensate System:

f. Because the Condensate, Condensate Booster, and Main Feedwater Pumps are connected in series, protection is provided to assure adequate suction pressure to all pumps.
1) A counting circuit ensures that an adequate number of pumps are running upstream before certain pumps may be started.
6) In the event a Booster Pump trips and Stand-by pump fails to start within 0.5 seconds, one Feedwater Pump will be tripped and load will runback as stated above.

Part 2 is correct. IAW OP-TM-EOP-0101, Guide 6 Basis:

Step 6 The step intent is to reseat any leaking MSSV.
Following reactor trip, OTSG pressure goes to saturation for the existing RCS temperature. If pre-trip reactor power was > ~15 %, MSSV actuation is likely and affords the opportunity for MSSV failure. Industry experience has shown the most likely failure mechanism is failure to reseat following actuation and blowdown. The narrow band between accumulation and blowdown values contributes to this mechanism, but continued lowering of steam pressure is usually successful in reseating the MSSV. For this reason, guidance on lowering OTSG pressure to reseat the valve is provided. Main steam pressure is maintained greater than 625 psig to prevent actuation of the HSPS feature and MSSV reseat usually occurs prior to header pressure lowering to this value. If OTSG pressure is quickly lowered and allowed to recover, RCS temperature impact is minimized but should not be allowed to go below 525F or exceed the Technical Specification cooldown rates during this effort. Failure to reseat following this guidance will result in cooldown once decay heat has lowered and if the safety valve cannot be reseated (including maintenance efforts if possible), isolation in accordance with Rule 3 is required. If OTSG pressure is reduced to 625 psig, it will need to be quickly recovered to > 800 psig to prevent RCS Tc from dropping below 525F.

D. Incorrect.
Part 1 is correct. When the Condensate Booster Pumps trip, the counting circuit recognizes zero Booster Pumps and will trip both Main Feedwater Pumps. This Loss of Main Feedwater will result in HSPS starting Emergency Feedwater. Additionally, since Condenser vacuum is present, the Turbine Bypass Valves will be used to control OTSG pressure. Atmospheric Dump Valves are only used when the Turbine Bypass Valves cannot be used. IAW TQ-TM-104-421-C001, Condensate System:

f. Because the Condensate, Condensate Booster, and Main Feedwater Pumps are connected in series, protection is provided to assure adequate suction pressure to all pumps.
1) A counting circuit ensures that an adequate number of pumps are running upstream before certain pumps may be started.
6) In the event a Booster Pump trips and Stand-by pump fails to start within 0.5 seconds, one Feedwater Pump will be tripped and load will runback as stated above.

Part 2 is incorrect. With a Loss of ICS Auto Power, the Turbine Bypass Valves will be controlled in HAND (manually). Therefore, no automatic adjustments can be made. Plausible if the candidate does not recognize that the valves are not under AUTO control.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	E14 EA1.2	
	Importance Rating	2.8	

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

Proposed Question: RO Question # 24

Technical Reference(s): TQ-TM-104-421-C001, p 47, Rev 11
OP-TM-EOP-0101, p 32, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: 421-GLO-5

Question Source:		
	Bank #	ID #1103810
	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	

Comments:
The KA is matched because the operator must demonstrate the ability to monitor and/or predict the operating charachteristic if the facility while performing an EOP Guide Enclosure.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze conditions given and then decide on what the automatic results would be.

What MUST be known:
1. What the effect is to the Main Feedwater System upon a loss of all Condensate Booster Pumps. 2. What the effect is to the Emergency Feedwater System upon a loss of all Condensate Booster Pumps. 3. The effect to the Atmospheric Dump Valves upon a loss of ICS AUTO Power.

25 **ID: 1144407** **Points: 1.00**

Plant Conditions:

- A reactor and plant startup are in progress.
- The reactor is operating at 15% reactor power.
- Intermediate Range Nuclear Instrument, NI-4, is Out of Service.

Event:

- Intermediate Range Nuclear Instrument, NI-3, indication instantly drops to zero.

Given the above information, the change in indication was caused by ___(1)___, and ___(2)___ IAW Tech Specs.

A. (1) blown fuse
(2) the plant may continue the power ascension

B. (1) blown fuse
(2) an Intermediate Range channel must be returned to operable status within 1 hour to avoid having to place the plant in Hot Shutdown

C. (1) loss of the "A" Battery
(2) the plant may continue the power ascension

D. (1) loss of the "A" Battery
(2) an Intermediate Range channel must be returned to operable status within 1 hour to avoid having to place the plant in Hot Shutdown

Answer: A

Answer Explanation

A. Correct.

Part 1 is correct. A blown fuse will cause electrical current to stop flowing through the Nuclear Instrument circuitry. This will cause the indication in the Control Room to fail low, which would read as zero. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

B. Main Idea

- All nuclear instrument strings are powered from reliable and high quality Vital AC sources. These sources are reliable as they are both Vital DC (site battery) and Emergency Diesel (via ES buses through inverters or DC through battery chargers and inverters). The Vital AC system itself has redundancies and cross-tie capabilities, and even backup sources from the BOP AC electrical system via regulating transformers.
- Loss of instrument power (via any means such as Inverter failure or blown fuse) will fail the instrument signal low.

Part 2 is correct. Tech Specs have a caveat that states Intermediate Range Nuclear Instruments are not required above 10%. So the Tech Spec action is not valid since the stem stated the plant is at 15%. IAW Tech Spec Table 3.5-1:

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met
3. Intermediate range instrument channels	1	0	(a) (b)

(a) Restore the conditions of Column (A) and Column (B) within one hour or place the unit in HOT SHUTDOWN within an additional 6 hours.

(b) When 2 of 4 power range instrument channels are greater than 10 percent full power, intermediate range instrumentation is not required.

B. Incorrect.

Part 1 is correct. A blown fuse will cause electrical current to stop flowing through the Nuclear Instrument circuitry. This will cause the indication in the Control Room to fail low, which would read as zero. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

B. Main Idea

- All nuclear instrument strings are powered from reliable and high quality Vital AC sources. These sources are reliable as they are both Vital DC (site battery) and Emergency Diesel (via ES buses through inverters or DC through battery chargers and inverters). The Vital AC system itself has redundancies and cross-tie capabilities, and even backup sources from the BOP AC electrical system via regulating transformers.
- Loss of instrument power (via any means such as Inverter failure or blown fuse) will fail the instrument signal low.

Part 2 is incorrect. Tech Specs have a caveat that states Intermediate Range Nuclear Instruments are not required above 10%. So the Tech Spec action is not valid since the stem stated the plant is at 15%. Plausible if the candidate does not recognize the caveat within Tech Specs.

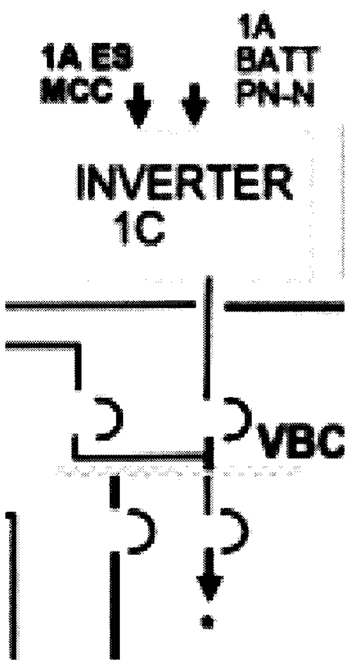
C. Incorrect.
Part 1 is incorrect. A loss of the "A" Battery will not affect the Intermediate Range Nuclear Instruments. The "A" Battery is one of two sources that feed the inverter that ultimately powers NI-3. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

3. 120V Vital AC Distribution Panel VBC fed from 1A ES MCC Unit 1AR via Inverter 1C
    a. Intermediate Range NI-3
        1) 600v - provides potential field.
        2) -10v to -80v provides compensating voltage from the auxiliary power supply.

B. Main Idea

- All nuclear instrument strings are powered from reliable and high quality Vital AC sources. These sources are reliable as they are both Vital DC (site battery) and Emergency Diesel (via ES buses through inverters or DC through battery chargers and inverters). The Vital AC system itself has redundancies and cross-tie capabilities, and even backup sources from the BOP AC electrical system via regulating transformers.

Additionally, IAW TQ-TM-104-734-C001, Vital ACDC System:



Part 2 is correct. Tech Specs have a caveat that states Intermediate Range Nuclear Instruments are not required above 10%. So the Tech Spec action is not valid since the stem stated the plant is at 15%. IAW Tech Spec Table 3.5-1:

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met
3. Intermediate range instrument channels	1	0	(a) (b)

(a) Restore the conditions of Column (A) and Column (B) within one hour or place the unit in HOT SHUTDOWN within an additional 6 hours.

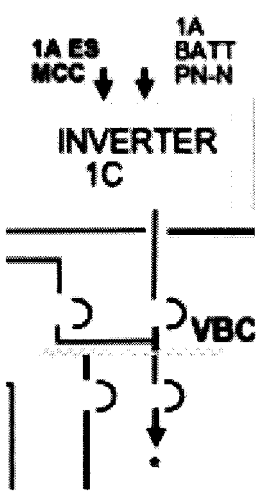
(b) When 2 of 4 power range instrument channels are greater than 10 percent full power, intermediate range instrumentation is not required.

D. Incorrect.
Part 1 is incorrect. A loss of the "A" Battery will not affect the Intermediate Range Nuclear Instruments. The "A" Battery is one of two sources that feed the inverter that ultimately powers NI-3. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

3. 120V Vital AC Distribution Panel VBC fed from 1A ES MCC Unit 1AR via Inverter 1C
    a. Intermediate Range NI-3
        1) 600v - provides potential field.
        2) -10v to -80v provides compensating voltage from the auxiliary power supply.

B. Main Idea
All nuclear instrument strings are powered from reliable and high quality Vital AC sources. These sources are reliable as they are both Vital DC (site battery) and Emergency Diesel (via ES buses through inverters or DC through battery chargers and inverters). The Vital AC system itself has redundancies and cross-tie capabilities, and even backup sources from the BOP AC electrical system via regulating transformers.

Additionally, IAW TQ-TM-104-734-C001, Vital ACDC System:



Part 2 is incorrect. Tech Specs have a caveat that states Intermediate Range Nuclear Instruments are not required above 10%. So the Tech Spec action is not valid since the stem stated the plant is at 15%. Plausible if the candidate does not recognize the caveat within Tech Specs.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	033 AA2.03	
	Importance Rating	2.8	

Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Indication of blown fuse.

Proposed Question: RO Question # 25

Technical Reference(s):
TQ-TM-104-623-C001, p 22, Rev 4
Tech Spec 3.05, p 3-29, Amend 247
Tech Spec 3.05, p 3-30, Amend 189

Proposed References to be provided to applicants during examination: None

Learning Objective: 623-GLO-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 6
55.43

Comments:

The KA is matched because the operator must determine that a blown fuse has occurred on an Intermediate Range Nuclear Instrument from the given indications.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze conditions given and then decide on what the result would be.

What MUST be known:
1. What the effect is to the Intermediate Range Nuclear Instruments upon a blown fuse.
2. What the effect is to the Intermediate Range Nuclear Instruments upon a loss of the station battery.
3. The Tech Spec requirement for the number of operable Intermediate Range Nuclear Instruments.

26 **ID: 1142101** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- Surveillance procedure 1303-4.16 Emergency Power System has been initiated to run an Emergency Diesel Generator for the monthly surveillance.

To prevent Load Tap changer from adversely affecting diesel generator VAR loading, the associated Auxiliary Transformer Load Tap Changer will be placed in manual prior to paralleling the diesel to the grid and ___(1)___, and the Emergency Diesel Generator load will be controlled at ___(2)___ MVAR.

A. (1) remain there during the entire run
(2) 0.5 to 1.1

B. (1) remain there during the entire run
(2) 2.9 ± 0.1

C. (1) be returned to auto after closing the diesel breaker
(2) 0.5 to 1.1

D. (1) be returned to auto after closing the diesel breaker
(2) 2.9 ± 0.1

Answer: A

Answer Explanation

A. Correct.

Part 1 is correct. The Load Tap Changer is taken to Manual either in section 8.1, Monthly Surveillance Testing, or Section 8.3, Yearly Testing. The Load Tap Changer is not restored to automatic control until the section 8.6, Return of Diesel Generator to Standby. IAW SP-1303-4.16, Emergency Power System:

> 8.1.10 Place the Load Tap Changer in Manual for the Auxiliary Transformer that is in parallel with the diesel to be tested as follows:
>
> 8.3.9 Place the Load Tap Changer in Manual for the Auxiliary Transformer that is in parallel with the diesel to be tested as follows:
>
> 8.6.23 Restore L TC to automatic control on Aux. Transformer that was paralleled with the diesel, as follows:

Part 2 is correct. IAW SP-1303-4.16, Emergency Power System:

> 8.3.25. VERIFY the following conditions and RECORD the time:
> a. 2.9 ± 0.1 MW output
> b. 0.5 to 1.1 MVAR output.

B. Incorrect.

Part 1 is correct. The Load Tap Changer is taken to Manual either in section 8.1, Monthly Surveillance Testing, or Section 8.3, Yearly Testing. The Load Tap Changer is not restored to automatic control until the section 8.6, Return of Diesel Generator to Standby. IAW SP-1303-4.16, Emergency Power System:

> 8.1.10 Place the Load Tap Changer in Manual for the Auxiliary Transformer that is in parallel with the diesel to be tested as follows:
>
> 8.3.9 Place the Load Tap Changer in Manual for the Auxiliary Transformer that is in parallel with the diesel to be tested as follows:
>
> 8.6.23 Restore L TC to automatic control on Aux. Transformer that was paralleled with the diesel, as follows:

Part 2 is incorrect. Plausible if the candidate picks the real load (MW) instead of the reactive load (MVAR). IAW SP-1303-4.16, Emergency Power System:

> 8.3.25. VERIFY the following conditions and RECORD the time:
> a. 2.9 ± 0.1 MW output
> b. 0.5 to 1.1 MVAR output.

C. Incorrect.

Part 1 is incorrect. The Load Tap Changer is taken to Manual either in section 8.1, Monthly Surveillance Testing, or Section 8.3, Yearly Testing. The Load Tap Changer is not restored to automatic control until the section 8.6, Return of Diesel Generator to Standby. Plausible if the candidate is not familiar with the procedure and believes that the Load Tap Changer will be in Manual only for a portion of the surveillance.

Part 2 is correct. IAW SP-1303-4.16, Emergency Power System:

8.3.25. VERIFY the following conditions and RECORD the time:
a. 2.9 ± 0.1 MW output
b. 0.5 to 1.1 MVAR output.

D. Incorrect.
Part 1 is incorrect. The Load Tap Changer is taken to Manual either in section 8.1, Monthly Surveillance Testing, or Section 8.3, Yearly Testing. The Load Tap Changer is not restored to automatic control until the section 8.6, Return of Diesel Generator to Standby. Plausible if the candidate is not familiar with the procedure and believes that the Load Tap Changer will be in Manual only for a portion of the surveillance.

Part 2 is incorrect. Plausible if the candidate picks the real load (MW) instead of the reactive load (MVAR). IAW SP-1303-4.16, Emergency Power System:

8.3.25. VERIFY the following conditions and RECORD the time:
a. 2.9 ± 0.1 MW output
b. 0.5 to 1.1 MVAR output.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	A05 AA2.2	
	Importance Rating	3.5	

Ability to determine and interpret the following as they apply to the (Emergency Diesel Actuation): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question: RO Question # 26

Technical Reference(s): 1303-4.16, p 9,26,40, 63, Rev 135A

Proposed References to be provided to applicants during examination: none

Learning Objective: 861-GLO-10

Question Source: Bank #

Modified Bank # ID# 679216

New

Question History: Last NRC Exam: ILT 08-01 (2010)

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

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments:
The KA is matched because the operator must demonstrate adherence to procedures and operations within the facility's limitations associated with an Emergency Diesel Generator actuation.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall steps and setpoints from a procedure.

What MUST be known:
1. The procedural guidance associated with the Load Tap Changer during a monthly surveillance run of an Emergency Diesel Generator. 2. The setpoints associated with the monthly surveillance run of an Emergency Diesel Generator.

Original Question:

Plant conditions:

The plant is at 100% power.
Surveillance procedure 1303-4.16 Emergency Power System has been initiated to run EG-Y-1A for the monthly surveillance.

To prevent Load Tap changer from adversely affecting diesel generator VAR loading, ___(1)___ Auxiliary Transformer Load Tap Changer(s) will be placed in manual prior to paralleling the diesel to the grid _____(2)_____ . EG-Y-1A load will be controlled at ___(3)___.

A. (1) the B
(2) and remain there during the entire run
(3) 2.9 ± 0.1 MWe and 0.5 to 1.1 MVAR

B. (1) the B
(2) and be returned to auto after closing the diesel breaker
(3) 3.0 ± 0.1 MWe and 0.5 to 1.1 MVAR

C. (1) both
(2) and remain there during the entire run
(3) 2.9 ± 0.1 MWe and 0.5 to 1.1 MVAR

D. (1) both
(2) and be returned to auto after closing the diesel breaker
(3) 3.0 ± 0.1 MWe and 0.5 to 1.1 MVAR

Answer: A

27 **ID: 1144430** **Points: 1.00**

Plant Conditions:

- The plant is operating at 90% reactor power.
- 500 EFPD.

Event:

- The "D" Reactor Coolant Pump, RC-P-1D, trips.
- A Plant Runback is occurring.
    - o Assume that the approximate reactor power level listed in OP-TM-MAP-H0101 is the correct reactor power level that the runback will stop at.

- The final total indicated rod index will be one of the following:

    - o 200% withdrawn
    - o 230% withdrawn
    - o 255% withdrawn
    - o 270% withdrawn

Given the above information, _________ of the above listed indicated rod indexes are in the Permissible Region.

A. one

B. two

C. three

D. four

Answer: A

Answer Explanation

A. Correct.

There are four Reactor Coolant Pumps operating in the Plant Conditions, which is determined from the fact that the plant is operating at 90% reactor power. The unit will run back to approximately 75% NI power on a loss of 1 Reactor Coolant Pump. IAW OP-TM-MAP-H0101, ICS Runback:

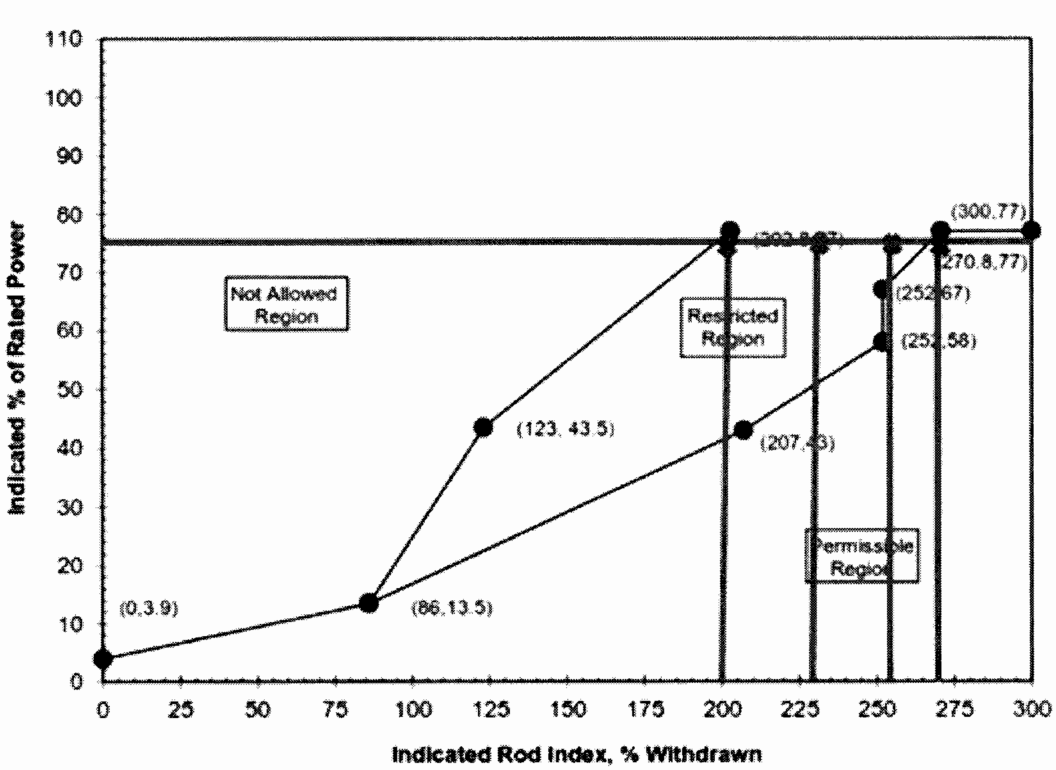
NOTE: Actual reactor power may vary due to plant efficiencies.
NOTE: ULD STAR module will trip to Manual following runback.

Unit runs back to:

665 MWe (approximately 75% NI power) at a rate of 50%/minute on a loss of 1 RCP.
405 MWe (approximately 50% NI power) at a rate of 50%/minute on a loss of 2 RCPs.
560 MWe (approximately 68% NI power) at a rate of 50%t/minute on a loss of 1 MFP.
455 MWe (approximately 55% NI power) at a rate of 30%/minute on an Asymmetric rod fault.
RC flow limited MWe power level at a rate of 20%/minute on RC Flow degradation.

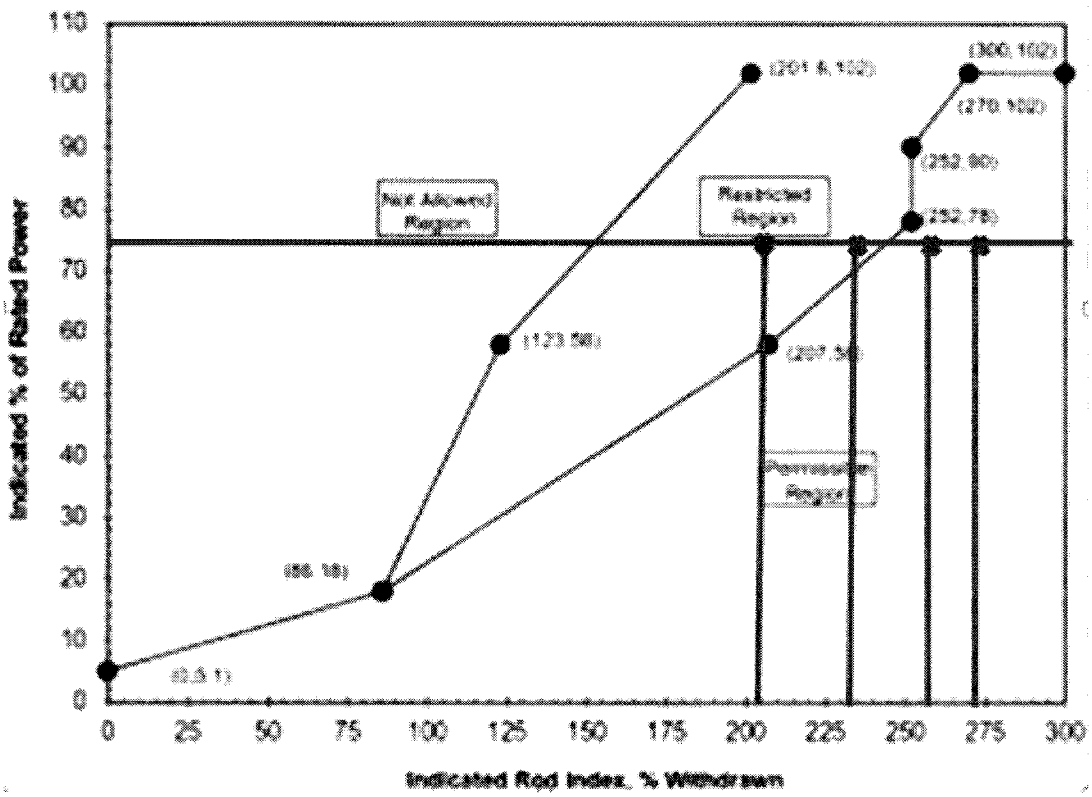
Using TMI-1 Cycle 20 Core Operating Limits Report (COLR), Figure 2 (Page 2 of 2), Error Adjusted Rod Insertion Limits (400 +/-10 EFPD to EOC; 3 Pump Operation, only 270% withdrawn is in the Permissible Region. The other three points are in the Restricted Region.

Figure 2 (Page 2 of 2)
Error Adjusted Rod Insertion Limits
(400 ±10 EFPD to EOC; 3 Pump Operation)



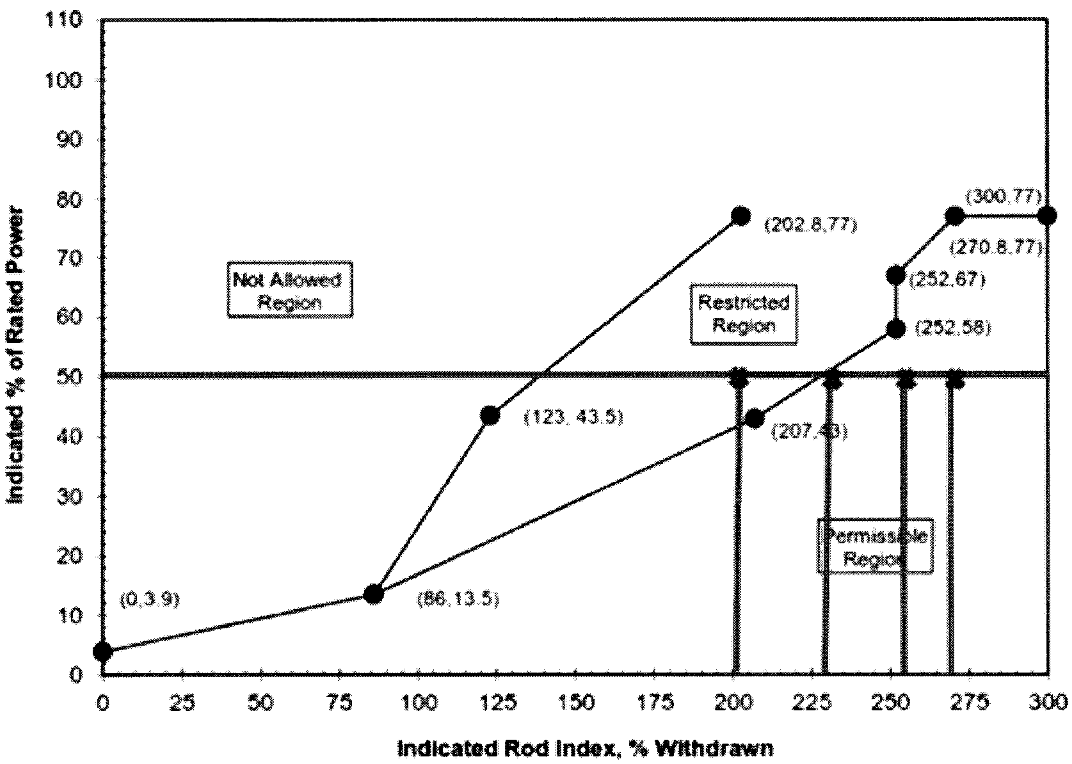
B Incorrect.
Plausible if the candidate uses TMI-1 Cycle 20 Core Operating Limits Report (COLR), Figure 1 (Page 2 of 2), Error Adjusted Rod Insertion Limits (400 +/-10 EFPD to EOC; 4 Pump Operation. Using this graph, 255% and 270% withdrawn are in the Permissible Region. The other two points are in the Restricted Region.

Figure 1 (Page 2 of 2)
Error Adjusted Rod Insertion Limits
(400 ±10 EFPD to EOC; 4 Pump Operation)



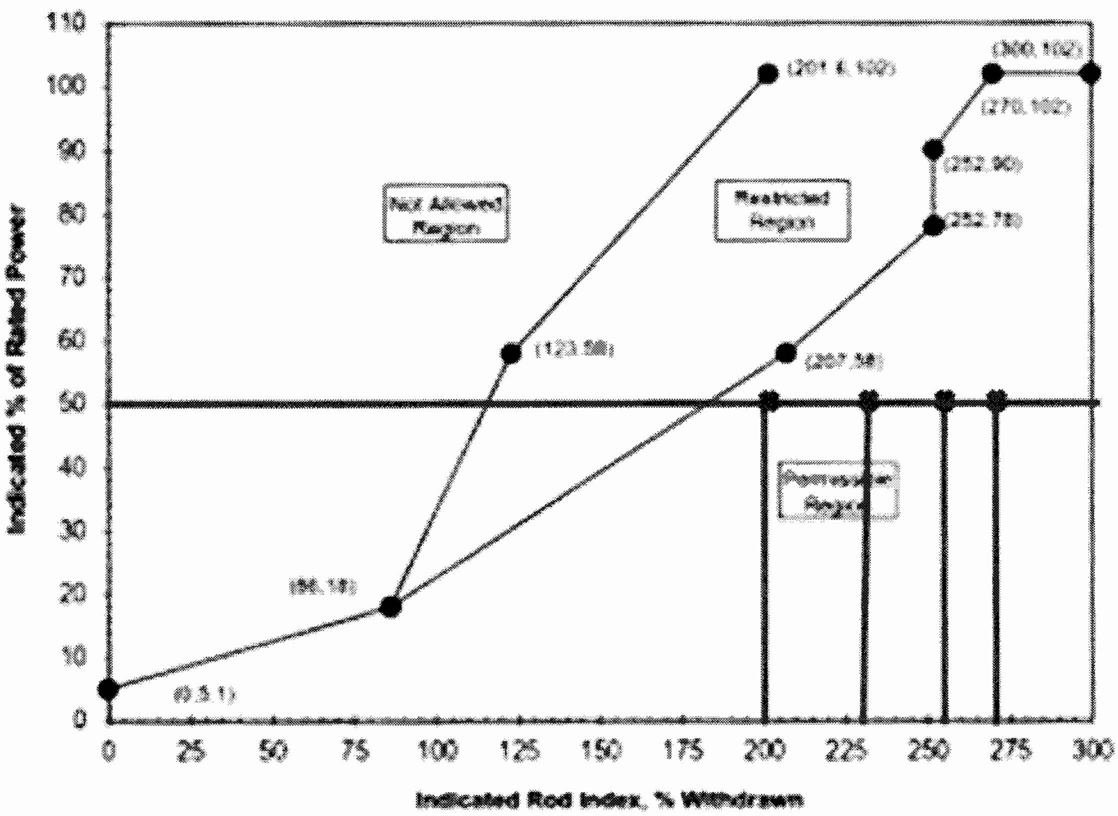
C. Incorrect.
Plausible if the candidate uses 50% NI Power (Loss of 2 Reactor Coolant Pumps) instead of 75% NI Power. In that case, using TMI-1 Cycle 20 Core Operating Limits Report (COLR), Figure 2 (Page 2 of 2), Error Adjusted Rod Insertion Limits (400 +/-10 EFPD to EOC; 3 Pump Operation, 230%, 255%, and 270% withdrawnare in the Permissible Region. The fourth point is in the Restricted Region.

Figure 2 (Page 2 of 2)
Error Adjusted Rod Insertion Limits
(400 ±10 EFPD to EOC; 3 Pump Operation)



D. Incorrect.
Plausible if the candidate uses 50% NI Power (Loss of 2 Reactor Coolant Pumps) instead of 75% NI Power and uses TMI-1 Cycle 20 Core Operating Limits Report (COLR), Figure 1 (Page 2 of 2), Error Adjusted Rod Insertion Limits (400 +/-10 EFPD to EOC; 4 Pump Operation. Using this graph, all of percentages withdrawn are in the Permissible Region.

Figure 1 (Page 2 of 2)
Error Adjusted Rod Insertion Limits
(400 ±10 EFPD to EOC; 4 Pump Operation)



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	A01 2.1.25	
	Importance Rating	3.9	

Plant Runback: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: RO Question # 27

Technical Reference(s): COLR TMI1, p 13-16, Rev 9
OP-TM-MAP-H0101, p 1, Rev 1

Proposed References to be provided to applicants during examination: COLR Figure 1 (2 pages)
COLR Figure 2 (2 pages)

Learning Objective: 621-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 5

55.43

Comments:
The KA is matched because the operator must interpret referance materials associated with a Plant Runback.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze graphs and determine the status of various data points.

What MUST be known:
1. The Permissive, Restricted, and Not Allowed ranges of the Error Adjusted Rod Insertion Limits curve. 2. The axis on the Error Adjusted Rod Insertion Limits curve. 3. How to interpret a graph.

28 **ID: 1146608** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- A large break LOCA occurs.
- Building Spray initiates.

Given the above information, the use of Building Spray will cause ___(1)___ the Borated Water Storage Tank (BWST), which will result in an increase in ___(2)___ dose rates.

A. (1) a more rapid depletion of
(2) off-site thyroid

B. (1) a more rapid depletion of
(2) Auxiliary Building

C. (1) an inadvertant transfer of reactor coolant to
(2) off-site thyroid

D. (1) an inadvertant transfer of reactor coolant to
(2) Auxiliary Building

Answer: B

Answer Explanation

A. Incorrect.

Part 1 is correct. IAW OP-TM-214-901, RB Spray Operation:

> 3.1.2 The use of RB spray will result in a more rapid depletion of the BWST and an earlier use of RB sump recirculation mode. RB sump recirculation mode will increase auxiliary building dose rates and may affect the accessibility of equipment.

Part 2 is incorrect. Plausible if the candidate is not familiar with the purpose of the Building Spray System and believes that a more rapid reduction of BWST level will cause off-site thyroid to be higher, overcoming the effect that Building Spray has within the Reactor Building. IAW OP-TM-214-901, RB Spray Operation:

> NOTES:
>
> (2) RB Spray is expected to reduce reactor building iodine levels by 95% (factor of 20). This will reduce off site thyroid dose.

B. Correct.

Part 1 is correct. IAW OP-TM-214-901, RB Spray Operation:

> 3.1.2 The use of RB spray will result in a more rapid depletion of the BWST and an earlier use of RB sump recirculation mode. RB sump recirculation mode will increase auxiliary building dose rates and may affect the accessibility of equipment.

Part 2 is correct. IAW OP-TM-214-901, RB Spray Operation:

> 3.1.2 The use of RB spray will result in a more rapid depletion of the BWST and an earlier use of RB sump recirculation mode. RB sump recirculation mode will increase auxiliary building dose rates and may affect the accessibility of equipment.

C. Incorrect.

Part 1 is incorrect. Plausible if the candidate confuses the Low Pressure Injection Pumps performance of Low Pressure Injection (LPI mode) with the Low Pressure Injection Pumps performance of Decay Heat Removal (DHR mode), since the same pumps are used for both functions. IAW OP-TM-214-000, Building Spray System:

> 2.1.2 Admin
>
> Inadvertent operation of the Building Spray System is highly undesirable. Avoid unnecessary introduction of spray into Reactor Building.
>
> Do not simultaneously operate the DH system (DHR mode) and the Reactor Building Spray System. This will prevent inadvertent transfer of reactor coolant to the BWST.

Part 2 is incorrect. Plausible if the candidate is not familiar with the purpose of the Building Spray System and believes that a more rapid reduction of BWST level will cause off-site thyroid to be higher, overcoming the effect that Building Spray has within the Reactor Building. IAW OP-TM-214-901, RB Spray Operation:

> NOTES:
>
> (2) RB Spray is expected to reduce reactor building iodine levels by 95% (factor of 20). This will reduce off site thyroid dose.

D. Incorrect.
Part 1 is incorrect. Plausible if the candidate confuses the Low Pressure Injection Pumps performance of Low Pressure Injection (LPI mode) with the Low Pressure Injection Pumps performance of Decay Heat Removal (DHR mode), since the same pumps are used for both functions. IAW OP-TM-214-000, Building Spray System:

2.1.2 Admin
Inadvertent operation of the Building Spray System is highly undesirable. Avoid unnecessary introduction of spray into Reactor Building.
Do not simultaneously operate the DH system (DHR mode) and the Reactor Building Spray System. This will prevent inadvertent transfer of reactor coolant to the BWST.

Part 2 is correct. IAW OP-TM-214-901, RB Spray Operation:

3.1.2 The use of RB spray will result in a more rapid depletion of the BWST and an earlier use of RB sump recirculation mode. RB sump recirculation mode will increase auxiliary building dose rates and may affect the accessibility of equipment.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 K1.13	
	Importance Rating	3.3	

Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems: CSS

Proposed Question: RO Question # 28

Technical Reference(s): OP-TM-214-901, p 1, Rev 4

Proposed References to be provided to applicants during examination: none

Learning Objective: 214-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	5	
	55.43		

Comments:
The KA is matched because the operator must demonstrate knowledge of the cause-effect relationship between High Pressure Injection/Low Pressure Injection Systems (ECCS) and the Building Spray System (CSS).

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the data given and determine, through multiple mental steps, the cause-effect relationship between two systems.

What MUST be known:
1. How use of the Building Spray System affects the Borated Water Storage Tank during a LOCA. 2. What the cause-effect relationship is between dose rates in the Auxiliary Building and a more rapid depletion of the Borated Water Storage Tank.

29 **ID: 1146609** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- An RCS leak occurs.
- RCS Pressure is being maintained with heaters and spray in Auto.
- Pressurizer Level is being maintained in the normal band with MU-V-17 in Auto.
- Main Annunciator C-3-2, IC Surge Tank Level HI/LO, is illuminated.
  - Intermediate Closed Cooling Water Surge Tank, IC-T-1, is still on scale.
- Main Annunciator, C-1-1, Radiation Level HI, is illuminated.
  - RM-L-9, ICCW Radiation Monitor, is in alarm and counts are rising slowly.
- PPC Point A0492, IC Temp Out RC-P-1C CLR, is in alarm.

Given the above information, the RCS is leaking into the Intermediate Closed Cooling System through the "C" Reactor Coolant Pump, RC-P-1C, ___(1)___, and the correct action to take is to ___(2)___ IAW OP-TM-MAP-C0302, IC Surge Tank Level HI/LO.

A. (1) #1 Shaft Seal
(2) secure RC-P-1C

B. (1) #1 Shaft Seal
(2) Close the breaker for IC-V-79C and then close IC-V-79C, RC-P-1C ICCW Outlet Valve.

C. (1) Thermal Barrier
(2) secure RC-P-1C

D. (1) Thermal Barrier
(2) Close the breaker for IC-V-79C and then close IC-V-79C, RC-P-1C ICCW Outlet Valve.

Answer: D

Answer Explanation

A. Incorrect.

Part 1 is incorrect. The #1 Shaft Seal is cooled by seal injection, provided by the Makeup System. Since it was given in the stem that Pressurizer Level is being maintained in the normal band with MU-V-17 in Auto, the candidate needs to determine that MU-P-1B is operating. If normal seal injection were to be lost, then RCS would flow past the thermal barrier cooler (which is cooled by ICCW) and then would cool the seals. Plausible if the candidate does not recognize that normal Makeup exists and does not know which support system provides each function within a RCP. IAW TQ-TM-104-226-C001, Reactor Coolant Pumps and Motors:

h. Shaft Seals
1) Used to control leakage from around the shaft.
2) In three stages with three mechanical seal assemblies.
3) Suitable because of precision tolerance and long wear.
4) Cooled and lubricated from seal injection.

Part 2 is incorrect. RC-P-1C would only be secured under the circumstances listed below IAW OP-TM-C0302, IC Surge Tank Level HI/LO. Since IC-T-1 is on scale and Seal injection is operating normally, none of the criteria are met to perform this action. IAW OP-TM-C0302, IC Surge Tank Level HI/LO:

4.1.3 If RM-L-9 is rising, then PERFORM the following:
1. IAAT IC-T-1 level indication cannot be maintained on scale, then PERFORM the following:
B. TRIP all four RC pumps.
3. If RCP Thermal Barrier is leaking, then PERFORM the following:
C. BRIEF all licensed operators on the following requirement:
- IAAT seal injection flow is < 22 gpm, then REDUCE Rx power within limits and STOP affected RCP IAW OP-TM-226-150 series procedures.

Plausible if the candidate is not familiar with the actions within the alarm response.

B. Incorrect.

Part 1 is incorrect. The #1 Shaft Seal is cooled by seal injection, provided by the Makeup System. Since it was given in the stem that Pressurizer Level is being maintained in the normal band with MU-V-17 in Auto, the candidate needs to determine that MU-P-1B is operating. If normal seal injection were to be lost, then RCS would flow past the thermal barrier cooler (which is cooled by ICCW) and then would cool the seals. Plausible if the candidate does not recognize that normal Makeup exists and does not know which support system provides each function within a RCP. IAW TQ-TM-104-226-C001, Reactor Coolant Pumps and Motors:

h. Shaft Seals
1) Used to control leakage from around the shaft.
2) In three stages with three mechanical seal assemblies.
3) Suitable because of precision tolerance and long wear.
4) Cooled and lubricated from seal injection.

Part 2 is correct. IAW OP-TM-C0302, IC Surge Tank Level HI/LO:

2. COMPARE trends of the following IC temperatures to determine source of leakage into ICCW:
   - A0490 IC TEMP OUT RC-P-1A CLR
   - A0491 IC TEMP OUT RC-P-1B CLR
   - A0492 IC TEMP OUT RC-P-1C CLR
   - A0493 IC TEMP OUT RC-P-1D CLR
   - A0495 IC TEMP OUT LETDOWN CLR A
   - A0496 IC TEMP OUT LETDOWN CLR B

3. If RCP Thermal Barrier is leaking, then PERFORM the following:
   A. CLOSE breaker associated with isolation valve.
      - IC-V-79A (1A ES Valves MCC, Unit 3D)
      - IC-V-79B (1B ES Valves MCC, Unit 3D)
      - IC-V-79C (1A ES Valves MCC, Unit 7B)
      - IC-V-79D (1B ES Valves MCC, Unit 5A)
   B. CLOSE IC-V-79 valve associated with affected RCP.

C. Incorrect.
Part 1 is correct. The Reactor Coolant Pump Thermal Barrier Cooler heat sink is the Intermediate Closed Cooling Water System. This is the cooler associated with PPC Point A0492, IC Temp Out RC-P-1C CLR. IAW TQ-TM-104-226-C001, Reactor Coolant Pumps and Motors:

e. Thermal Barrier/Heat Exchanger
   1) Flanged cylindrical shell and heat exchanger coil in a welded assembly.
   2) Located directly above turning vane-diffuser.
   3) Thermal barrier extends downward along the inside diameter of the turning-vane diffuser.
   4) Heat exchanger portion is welded to the thermal barrier shell near the bottom.
   5) Thermal barrier prevents hear transfer from the Reactor Coolant to the pump internals.
   6) Cooled by ICCW @ approximately 40 gpm.

Part 2 is incorrect. RC-P-1C would only be secured under the circumstances listed below IAW OP-TM-C0302, IC Surge Tank Level HI/LO. Since IC-T-1 is on scale and Seal injection is operating normally, none of the criteria are met to perform this action. IAW OP-TM-C0302, IC Surge Tank Level HI/LO:

4.1.3 If RM-L-9 is rising, then PERFORM the following:
   1. IAAT IC-T-1 level indication cannot be maintained on scale, then PERFORM the following:
      B. TRIP all four RC pumps.
   3. If RCP Thermal Barrier is leaking, then PERFORM the following:
      C. BRIEF all licensed operators on the following requirement:
   - IAAT seal injection flow is < 22 gpm, then REDUCE Rx power within limits and STOP affected RCP IAW OP-TM-226-150 series procedures.

Plausible if the candidate is not familiar with the actions within the alarm response.

D. Correct.
Part 1 is correct. The Reactor Coolant Pump Thermal Barrier Cooler heat sink is the Intermediate Closed Cooling Water System. This is the cooler associated with PPC Point A0492, IC Temp Out RC-P-1C CLR. IAW TQ-TM-104-226-C001, Reactor Coolant Pumps and Motors:

e. Thermal Barrier/Heat Exchanger
   1) Flanged cylindrical shell and heat exchanger coil in a welded assembly.
   2) Located directly above turning vane-diffuser.
   3) Thermal barrier extends downward along the inside diameter of the turning-vane diffuser.
   4) Heat exchanger portion is welded to the thermal barrier shell near the bottom.
   5) Thermal barrier prevents hear transfer from the Reactor Coolant to the pump internals.
   6) Cooled by ICCW @ approximately 40 gpm.

Part 2 is correct. IAW OP-TM-C0302, IC Surge Tank Level HI/LO:

2. COMPARE trends of the following IC temperatures to determine source of leakage into ICCW:
   - A0490 IC TEMP OUT RC-P-1A CLR
   - A0491 IC TEMP OUT RC-P-1B CLR
   - A0492 IC TEMP OUT RC-P-1C CLR
   - A0493 IC TEMP OUT RC-P-1D CLR
   - A0495 IC TEMP OUT LETDOWN CLR A
   - A0496 IC TEMP OUT LETDOWN CLR B

3. If RCP Thermal Barrier is leaking, then PERFORM the following:
   A. CLOSE breaker associated with isolation valve.
      - IC-V-79A (1A ES Valves MCC, Unit 3D)
      - IC-V-79B (1B ES Valves MCC, Unit 3D)
      - IC-V-79C (1A ES Valves MCC, Unit 7B)
      - IC-V-79D (1B ES Valves MCC, Unit 5A)
   B. CLOSE IC-V-79 valve associated with affected RCP.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 K1.04	
	Importance Rating	3.3	

Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: RCS, in order to determine source(s) of RCS leakage into the CCWS.

Proposed Question: RO Question # 29

Technical Reference(s): TQ-TM-104-226-C001, p 7-8, Rev 9
OP-TM-MAP-C0302, p 2-3, Rev 3

Proposed References to be provided to applicants during examination: none

Learning Objective: 531-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	10
	55.43	

Comments:
The KA is matched because the operator must demonstrate knowledge of the physical connections between the Reactor Coolant System (RCS) and the Intermediate Closed Cooling Water System (CCWS).

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the data given and determine what action to take.

What MUST be known:
1. What component within a Reactor Coolant Pump is cooled by the Intermediate Closed Cooling Water System. 2. The procedural guidance within the Main Annunciator C-3-2.

30 **ID: 1142238** **Points: 1.00**

Plant Conditions:

- A reactor startup is in progress.
- The plant is currently operating at 50% reactor power.

Event:

- A loss of the 230 KV 4 bus occurs.

Given the above information, the associated Balance of Plant Busses will Fast Transfer to the ___(1)___ Auxiliary Transformer and the associated Safeguards bus will be powered from the ___(2)___ Emergency Diesel Generator.

A. (1) "A"
(2) "A"

B. (1) "A"
(2) "B"

C. (1) "B"
(2) "A"

D. (1) "B"
(2) "B"

Answer: A

Answer Explanation

A. Correct.

Part 1 is correct. IAW MAP NN, alarm NN-2-1, 230KV Bus 4 Trip:

- AUTOMATIC ACTION:
  - The 1B 6900V, and 1C 4160V buses transfer to the 1A aux. transformer

Part 2 is correct. IAW MAP NN, alarm NN-2-1, 230KV Bus 4 Trip:

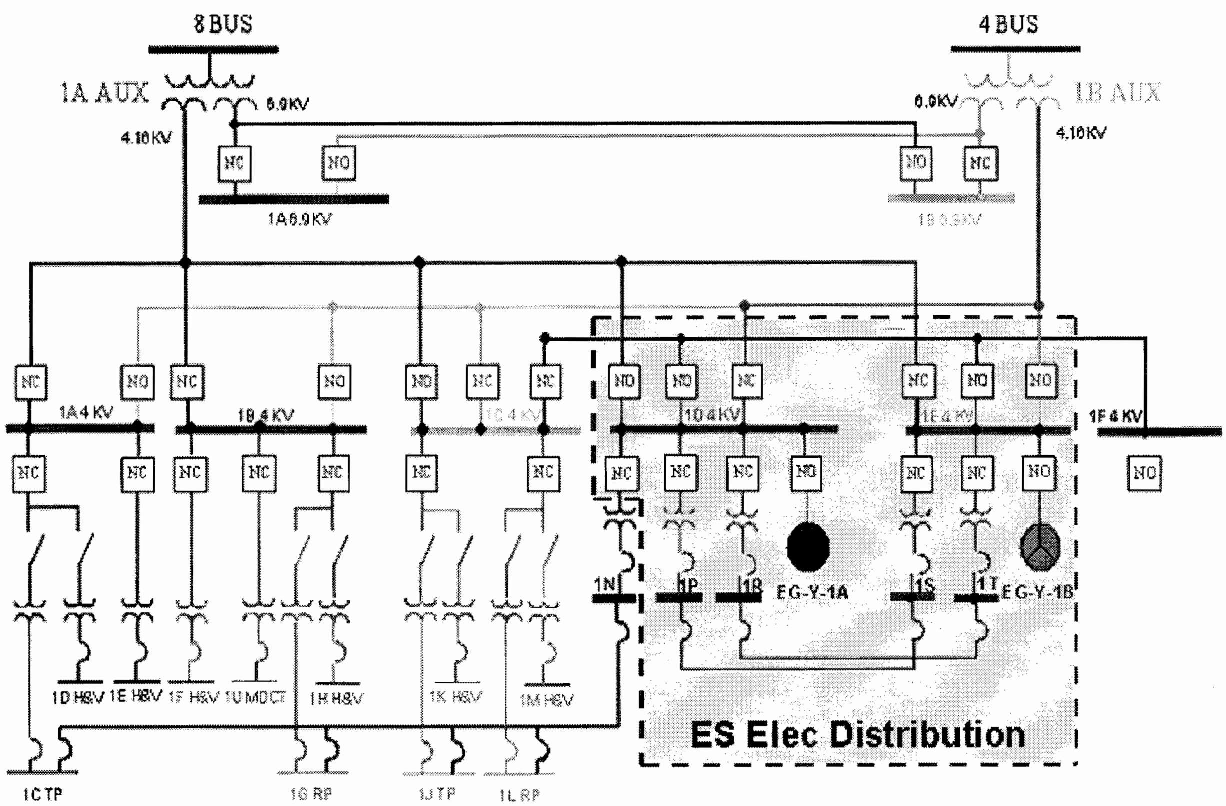
- AUTOMATIC ACTION:
  - 1A Diesel Generator starts and energizes 1D bus

B. Incorrect.

Part 1 is correct. IAW MAP NN, alarm NN-2-1, 230KV Bus 4 Trip:

- AUTOMATIC ACTION:
  - The 1B 6900V, and 1C 4160V buses transfer to the 1A aux. transformer

Part 2 is incorrect but plausible if the candidate associates the "B" Emergency Diesel Generator with the "B" Auxiliary Transformer. However, the 4 Bus powers the "D" 4KV ES Bus, which is associated with the "A" Emergency Diesel Generator. IAW TQ-TM-104-701-C001, Main Electrical Distribution System:



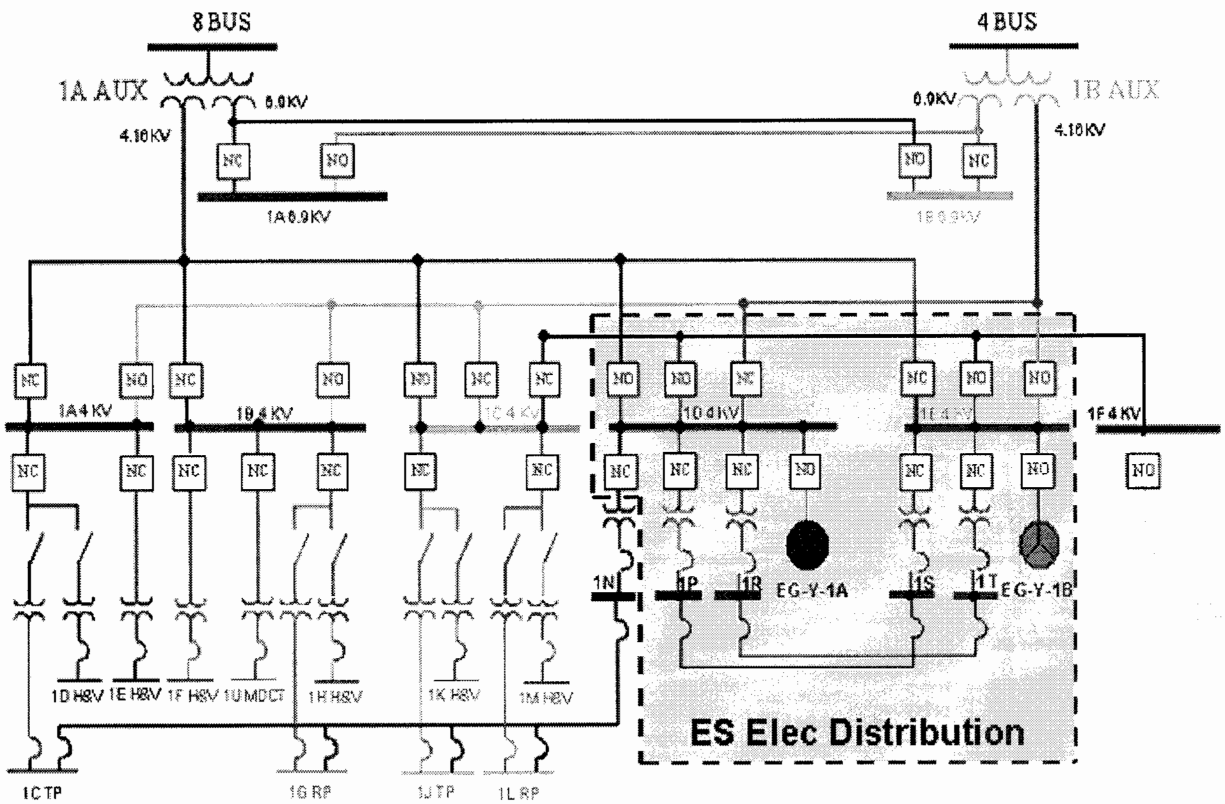
C. Incorrect.
Part 1 is incorrect. Plausible if the candidate associates the 4 Bus with the "A" Auxiliary Transformer by using chronological and alphabetical order association.

Part 2 is correct. IAW MAP NN, alarm NN-2-1, 230KV Bus 4 Trip:

- AUTOMATIC ACTION:
  - 1A Diesel Generator starts and energizes 1D bus

D. Incorrect.
Part 1 is incorrect. Plausible if the candidate associates the 4 Bus with the "A" Auxiliary Transformer by using chronological and alphabetical order association.

Part 2 is incorrect but plausible if the candidate associates the "B" Emergency Diesel Generator with the "B" Auxiliary Transformer. However, the 4 Bus powers the "D" 4KV ES Bus, which is associated with the "A" Emergency Diesel Generator. IAW TQ-TM-104-701-C001, Main Electrical Distribution System:



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062 K1.02	
	Importance Rating	4.1	

Knowledge of the physical connections and/or cause-effect relationships between the ac distribution system and the following systems: ED/G.

Proposed Question: RO Question # 30

Technical Reference(s): MAP NN, NN-2-1, p 1, Rev 9

Proposed References to be provided to applicants during examination: none

Learning Objective: 731-GLO-5

Question Source:		
	Bank #	ID# 587994
	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	

Comments:
The KA is matched because the operator must demonstrate knowledge of the cause-effect relationship between the AC Distribution System and an Emergency Diesel Generator.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall the automatic actions that occur for a loss of power to an Auxiliary Transformer.

What MUST be known:
1. Which of the ES Busses and Emergency Diesel Generators are associated with the 230kV 4 Bus.   2. Which Auxiliary Transformer is associated with the 230kV 4 Bus.

31 **ID: 1142227** **Points: 1.00**

Plant Conditions:

- "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

A. Incorrect.
Plausible if the examinee does not know the MU-P-1C breaker remains closed when the 1E 4160 Volt bus de-energizes.

B. Correct.
Without the ESAS signal NR-P-1B will autostart when the diesel re-energizes the 1T 480 Volt ES bus due to the NR-P-1C breaker disagreement.

C. Incorrect.
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.
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	

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, p 37-38, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 531-GLO-5

Question Source: Bank # ID# 862967

Modified Bank #

New

Question History:		Last NRC Exam:	2014 (ILT 12-01)
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments:
The KA is matched because the operator must demonstrate the knowledge of the power supplies for ESAS equipment control.

The question is at the Comprehension/Analysis cognitive level because the candidate must comprehend the operation of ESAS loads following an ESAS actuation and subsequent LOOP (in that specific order)..

What MUST be known:
1. Component responses within the NSRW System given a sequence combination of ES signals and Loss of power. 2. Component responses within the Makeup System given a sequence combination of ES signals and Loss of power. 3. Component responses within the NSCCW System given a sequence combination of ES signals and Loss of power. 4. Component responses within the DCCW System given a sequence combination of ES signals and Loss of power.

32 **ID: 1146622** **Points: 1.00**

Plant Conditions:

- The 1R ES 480V Bus is out of service for corrective maintenance.
- A reactor plant shutdown is in progress, currently 24% reactor power and lowering.
- The "B" and "C" Nuclear Services River Water Pumps, NR-P-1B and NR-P-1C are operating.
  - NR-P-1B is ES selected on the 1T 480V ES Bus.
- The "B" and "C" Secondary Services River Water Pumps, SR-P-1B and SR-P-1C are operating.

Sequence of Events:

- A Steam Line Rupture has occurred in the Reactor Building.
- A 4 psig Engineered Safeguards Signal has actuated.

With regards to the following valves:

- NR-V-1C: "C" Nuclear Services River Water Pump Discharge Valve.
- NR-V-4A: "A" Makeup to Circ Water Flume Valve.
- NR-V-4B: "B" Makeup to Circ Water Flume Valve.
- SR-V-1C: "C" Secondary Services River Water Pump Discharge Valve.

Correctly complete the following sentence:

Given the above information, ___(1)___ is(are) the open valve(s) and ___(2)___ is(are) the closed valve(s).

A. (1) NR-V-1C
(2) NR-V-4A, NR-V-4B, and SR-V-1C

B. (1) NR-V-1C and NR-V-4A
(2) NR-V-4B and SR-V-1C

C. (1) NR-V-1C and SR-V-1C
(2) NR-V-4A and NR-V-4B

D. (1) NR-V-1C, NR-V-4A, and SR-V-1C
(2) NR-V-4B

Answer: C

Answer Explanation

A. Incorrect.
Part 1 is partially correct. NR-V-1C will be open.

Part 2 is incorrect but plausible if the candidate believes that, since SR-P-1C will be tripped due to load shed, the associated discharge valve SR-V-1C, will close automatically.

B. Incorrect.
Part 1 is incorrect but plausible if the candidate believes that the power supply to NR-V-4A is the 1R 480V ES Bus (or an MCC off of it). This can be confusing since the power supply to NR-V-1A is the 1A ES Screenhouse MCC (which is powered from the 1R 480V ES Bus) and the power supply to NR-V-4A is the 1A ES Valves MCC (which is not powered from the 1R 480B ES Bus).

Part 2 is incorrect but plausible if the candidate believes that, since SR-P-1C will be tripped due to load shed, the associated discharge valve SR-V-1C, will close automatically.

C. Correct.
Part 1 is correct.
NR-V-1C: The Nuclear Services River Water Pump Discharge Valves do have automatic open signals upon an associated pump start but do not have automatic close functions. Therefore, since NR-P-1C was running before the event occurred, NR-V-1C was open. Once the ESAS signal occurs, NR-P-1C will trip due to load shed but NR-V-1C will remain open. IAW TQ-TM-104-531-C001, Primary Cooling Systems:

- ESAS actuation
    - 43 SS - Determines which pump is selected for ES
    - A or B Pump selected on 1 R BUS
    - B or C Pump selected on 1T BUS
    - Non-ES Selected NR pump will trip on an ES actuation.

NR-V-1A/B/C automatically open on pump start signal, and MUST be manually closed on pump stop.

SR-V-1C: The Secondary Services River Water Pump Discharge Valves do not have automatic open or close functions. Therefore, since both pumps were running before the event occurred, both discharge valves were open. Once the ESAS signal occurs, SR-P-1C will be secured but SR-V-1C will remain open. Plausible if IAW TQ-TM-104-544-C001, Secondary Cooling Water Systems:

- Normally all three SR-V-1s are maintained open as these valves do not have automatic functions.

Part 2 is correct.
NR-V-4A/B: The Nuclear Services River Water Makeup to Circ Water Valves are closed on an ES signal to prevent Primary Cooling water from going to a lesser important component. IAW TQ-TM-104-531-C001, Primary Cooling Systems:

- ESAS actuation
    - NR-V-4A/B Close on ES signal

D. Incorrect.
Part 1 is incorrect but plausible if the candidate believes that the power supply to NR-V-4A is the 1R 480V ES Bus (or an MCC off of it). This can be confusing since the power supply to NR-V-1A is the 1A ES Screenhouse MCC (which is powered from the 1R 480V ES Bus) and the power supply to NR-V-4A is the 1A ES Valves MCC (which is not powered from the 1R 480B ES Bus).

Part 2 is incorrect but plausible if the candidate believes that, since SR-P-1C will be tripped due to load shed, the associated discharge valve SR-V-1C, will close automatically.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 K2.08	
	Importance Rating	3.1	

Knowledge of bus power supplies to the following: ESF-actuated MOVs

Proposed Question: RO Question # 32

Technical Reference(s): TQ-TM-104-531-C001, p 37, Rev 8
TQ-TM-104-544-C001, p 24, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: 531-GLO-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 7

55.43

Comments:
The KA is matched because the operator must demonstrate the knowledge of the power supplies for ESF-actuated MOV's associated with Service Water Systems.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the conditions and event, and understand the cause-effect on various plant components.

What MUST be known:
1. The response of NSRW valves given an ES actuation signal.
2. The response of SSRW valves given an ES actuation signal.
3. The relationship between NSRW valves and the associated pumps upon pump start/stop.
4. The relationship between SSRW valves and the associated pumps upon pump start/stop.

33 **ID: 1146674** **Points: 1.00**

Plant Conditions:

- The plant is operating at 75% reactor power.
- All four Reactor Coolant Pumps are operating.
- ICS is in manual control.
- Power Range Nuclear Instrument, NI-8, is failed HIGH.
- The crew has taken the appropriate steps IAW OP-TM-MAP-G0102, RPS Channel Trip.
- A Loss of VBA has occurred.
    - The crew has taken the appropriate steps IAW OP-TM-AOP-015, Loss of VBA.

Event:

- RC3B-PT1, RCS loop "B" Narrow Range Pressure Transmitter, suddenly fails LOW.

Given the above information, the Reactor Protection System is in a:

A. 1 out of 3 logic and the reactor has remained critical.

B. 1 out of 4 logic and the reactor has remained critical.

C. 2 out of 3 logic and a reactor trip has occurred.

D. 2 out of 4 logic and a reactor trip has occurred.

Answer: C

Answer Explanation

A. Incorrect.

Plausible if the candidate believes that there are no RPS channels in a tripped state prior to the event occurring. To get a 1 out of 3 logic conclusion, the candidate will believe that one of the 2 cabinets in the Plant Conditions ("A" or "D") is placed in Manual Bypass and does not recognize that the other has tripped (either that NI-8 failing high does not send a trip signal because it a failed instrument or that the loss of VBA does not send a trip signal to the other three channels).

B. Incorrect.

Plausible if the candidate believes that there are no RPS channels in a tripped state prior to the event occurring. To get a 1 out of 3 logic conclusion, the candidate will not recognize that NI-8 failing high and that the loss of VBA will send a trip signals to the other RPS channels.

C. Correct.

There are four Reactor Protection System (RPS) cabinets. Whenever any two of the four channels agree that a setpoint has been reached or exceeded a reactor trip signal is initiated by each RPS channel. The Control Rod Drive Control System receives the reactor trip signals from RPS and opens the control rod drive breakers to release the control rods, thus, tripping the reactor.

When NI-8 fails high, OP-TM-MAP-G0102, RPS Channel Trip, is entered. This procedure directs that the RPS channel associated with NI-8, RPS "D", be placed in Manual Bypass. IAW OP-TM-MAP-G0102:

4.0 MANUAL ACTIONS REQUIRED

- If trip condition can not be cleared or the reason for the trip is not known, then PLACE RPS channel in Manual Bypass IAW OP-TM-641-455.

Manual Bypass is described in TQ-TM-104-641-C001, Reactor Protection System:

a. Manual Bypass section - The manual bypass section of the reactor trip module allows the RPS module to be placed in test, for on-line testing, without being tripped. A key operated bypass switch on the trip module front plate is used to achieve manual bypassing. When the key is turned to the manual bypass position, the following two functions are initiated:

1) First, two contacts close, applying power to the channel trip relay through an alternate path. With this power applied to the channel trip relay, any trip signal from the RPS channel trip bistables or from the test and critical module relays is over-ridden.

2) Second, three contacts open preventing the activation of the manual bypass function in any of the other RPS channels. Placing a RPS channel in manual bypass automatically changes the RPS from a 2-out-of-4 logic to a 2-out-of-3 logic for tripping.

When the Loss of VBA occurs, it deenergizes RPS cabinet "A" and sends a trip signal to the other three cabinets. IAW OP-TM-AOP-0151, Loss of VBA Basis Document:

- Effects of Loss of Power
  - CRD AC breaker Unit 10 opens. ?A? RPS channel trip signal is sent to all other RPS cabinets.
  - NI-5 power range, RC-3A-PT1 narrow range pressure, RC14A-DPT1, and RC14A-DPT2 signals from RPS Channel "A" will fail. SASS will automatically select NI-6, RC3B-PT1, RC14A-DPT2, and RC14B-DPT2 for ICS input.

Since RPS cannot have more than 1 channel in Manual Bypass, then the "A" RPS channel remains in a tripped state. This now creates a tripped channel and a 1 out of 2 logic for the remaining channels, meaning one trip signal out of two possible channels is required for a reactor trip.

Additionally, when the loss of VBA occurs, Pressure Instrument RC3A-PT1 loses power. This is one of two instruments that are associated with the Pressurizer Pressure Control System. There are two channels, swapped when possible and required by SASS, that control the Pressurizer heaters, Spray Valve, and PORV. RC3A-PT1 is the preferred instrument. When it is lost, SASS will automatically select RC3B-PT1 for Pressurizer Pressure Control. IAW OP-TM-621-000, Integrated Control System (ICS):

SASS Inputs and Indications

Page 1 of 2

SASS		SOURCE INSTRUMENT		INDICATIONS				
Channel No.		A	B	A	B	Non Selected	Selected	Affected ICS/NNI Controls
1-1-1	Reactor Power	NI-5	NI-6	NI-5 Dig RPS A Ind. A0582	NI-6 Dig RPS B Ind. A0583		NI-5 NIR A5022	Diamond Panel, Rx Mstr & FW Loop Masters (Neutron Error)
1-1-2	RCS Pressure	RC3A-PT1	RC3B-PT1	RC3A-PR1 RPS A Ind. A0586	RC3B-PR1 RPS B Ind. A0587		A5071	Pressurizer Heaters, Spray Valve & PORV

Lastly, when the event occurs, RC3B-PT1 fails low. This is associated with the "B" RPS cabinet and provides a low pressure trip signal and a high pressure trip signal. Since the instrument has failed low, the "B" RPS cabinet see a low pressure trip signal. It sends a trip signal to the other cabintes and the RPS logic now recognizes two tripped channels and will initiate a reactor trip.

D. Incorrect.
Plausible if the candidate believes that only one of the listed RPS channels (A and D) are in a tripped state prior to the event occurring and that the other is fully enabled in an untripped state. To get a 2 out of 4 logic conclusion, the candidate will recognize one tripped channel initially and then the "B" RPS pressure instrument as the second channel.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010 K3.02	
	Importance Rating	4.0	

Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: RPS.

Proposed Question: RO Question # 33

Technical Reference(s):
OP-TM-MAP-G0102, p 1, Rev 3
OP-TM-641-000, p 17, Rev 4
OP-TM-AOP-0151, p23, Rev 6
TQ-TM-104-641-C001, p 10, Rev 2

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	6
	55.43	

Comments:
The KA is matched because the operator must demonstrate the knowledge of the effect that a malfunction of the Pressurizer Pressure Control System will have on the Reactor Protection System (RPS).

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the conditions and event associated with a system, and understands the cause-effect on other systems.

What MUST be known:
1. RPS cabinet trip signal logic. 2. The manual bypass feature of the RPS cabinets. 3. RPS trip signals. 4. Power supplies associated with RPS trips.

34 **ID: 1146956** **Points: 1.00**

Plant Conditions:

- The plant is in refueling shutdown with the core off-loaded to the spent fuel pool.
- The following lineup exists for NS pumps:
  - NS-P-1A is in standby and ES selected.
  - NS-P-1B is operating and ES selected.
  - NS-P-1C is in standby.

Event:

- An overcurrent condition on NS-P-1B results in the following actions:
  - Trip of the NS-P-1B breaker.
  - Trip of the associated 480v Hi-Side bus feeder breaker.

Given the above information, _________ is/are running.

A. NS-P-1A, ONLY

B. NS-P-1C, ONLY

C. NS-P-1A and NS-P-1C

D. none of the NSSCW Pumps

Answer: D

Answer Explanation

A. Incorrect.
Plausible if the candidate recognizes that NS-P-1C will not receive power and then incorrectly believes that the counting circuit will then attempt to start NS-P-1A. However, because the breaker for NS-P-1C closes, the counting circuit logic is satisfied and it will not continue on.

B. Incorrect.
Plausible if the candidate does not recognize that NS-P-1B is currently powered from the 1S 480V Bus, which is the same power supply as NS-P-1C. Since NS-P-1B can be powered from either the "A" Train (1P 480V Bus) or the "B" Train (1S 480V Bus), the candidate will have to figure out which bus is powering NS-P-1B. Since NS-P-1A and NS-P-1B are both ES selected, and only one NSCCW pump may be ES selected from any bis (due to the 43 selector switch), the candidate may conclude that NS-P-1A is ES selected on the 1P 480V Bus and NS-P-1B is ES selected on the 1S 480V Bus. Therefore, NS-P-1C cannot be running.

C. Incorrect.
Plausible if the candidate is unfamiliar with the power supplies and/or the counting circuit logic associated with the NSCCW Pumps. Normally, the counting circuit will start the next pump in sequence in 0.5 seconds. If that pump does not indicate breaker closed, then the counting circuit will attempt to start the second pump in sequence (at t=1.0 seconds). The candidate may believe that the counting circuit starts both pumps in 0.5 second increments.

D. Correct.
The logic is such that the NSCCW Pumps operate normally with manual starting and stopping from the control room. Each pump has a standby feature so that, with a loss of a pump (as sensed by breaker positions), another pump will start automatically. In this case, NS-P-1B trips on overload.. The next breaker in line to be closed is NR-P-1C. The successful closing of NR-P-1C breaker (using DC control power) ends the counting circuit sequence. However, NS-P-1C has no 480V AC power to operate the pump, so it will not be running. Bus undervoltage is sensed at the 4kv level, which does not have an undervoltage, so no further check is made to attempt NR-P-1A start. IAW TQ-TM-104-531-C001, Primary Cooling Systems:

2) Standby NS Pump auto-start
   a) If a running pump trips, standby (nonrunning) pump auto-starts.
   b) If preferred STBY pump does not start in 0.5 seconds, other STBY Pump will start, if not running. Sequence is A-B-C-A.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062 K3.01	
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: Major system loads.

Proposed Question: RO Question # 34

Technical Reference(s): TQ-TM-104-531-C001, p 38, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 531-GLO-5

Question Source: Bank #

Modified Bank # ID# 686615

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

Comments:
The KA is matched because the operator must demonstrate the knowledge of the effect that a malfunction of the ac distribution system will have on the NSCCW pumps (a major system load).

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the conditions and event associated with a system, and understands the cause-effect on other systems.

What MUST be known:
1. Standby Pump start logic associated with the NSCCW Pumps. 2. Electrical logic effect on NSCCW Pumps when an High-side breaker trips.

Original Question:

Plant Conditions:

The plant is in refueling shutdown with the core off-loaded to the spent fuel pool.
The following lineup exists for NS pumps:
NS-P-1A is operating and ES selected.
NS-P-1B is shutdown. NS-P-1B is racked in on the "P" bus.
NS-P-1C is shutdown and ES selected.

Event:

Trip of the "P" 480v Hi-Side bus feeder breaker (P1-02) due to an overcurrent condition caused by NS-P-1A, which also tripped.

Which of the following is the correct light and amp indication seen on the console for this event?

A. NS-P-1A green light only. No amps.
NS-P-1B green light. No amps.
NS-P-1C red light. Normal running current.

B. NS-P-1A green and amber lights. No amps.
NS-P-1B red light. No amps.
NS-P-1C green light. No amps.

C. NS-P-1A green and amber lights. No amps.
NS-P-1B red light. No amps.
NS-P-1C red light. Normal running current.

D. NS-P-1A green and amber lights. No amps.
NS-P-1B red and amber lights. No amps.
NS-P-1C green light. No amps.

Answer: B

35 **ID: 1146790** **Points: 1.00**

Plant Conditions:

- Waste Gas Tank, WDG-T-1B, radioactive gas release in progress.
  - Waste Gas Release Stop and Control Valve, WDG-V-47, is open.

Event:

- A blown fuse de-energizes RM-A-7, Waste Gas Release Monitor.

Based on the above conditions, the radioactive gas release will:

A. Continue because RM-A-7 interlocks are "energize to actuate".

B. NOT continue because RM-A-7 and the solenoid for WDG-V-47 have a common power supply.

C. Continue because the RM-A-7 interlock circuit is still energized from RMS interlock power supply.

D. NOT continue because RM-A-7 monitor is de-energized with its interlock defeat switch in NORMAL position.

Answer: D

Answer Explanation

A. Incorrect.
Plausible if the candidate is not familiar with how the interlocks associated with RM-A-7 operate. See the explanation for "D" below.

B. Incorrect.
Because the power supplies are not common. RM-A-7 is powered from Vital Bus "D". WDG-V-47 is powered from the 1B RadWaste MCC, which is "Balance of Plant" powered. Plausible if the candidate is not familiar with the power supplies for RM-A-7 and/or WDG-V-47.

C. Incorrect.
Plausible if the candidate is not familiar with how the interlocks associated with RM-A-7 operate. See the explanation for "D" below.

D. Correct.
The power supply to RM-A-7 is Vital Bus "D". So whether it is a loss of Vital Bus "D" or a blown fuse, the effect on RM-A-7 is the same. The only way that the effect would be negated is if the interlock switch for RM-A-7 is in the "defeat" position. However, nothing in the stem leads the candidate to believe that this is the case. IAW OP-TM-AOP-018, Loss of VBD:

Attachment 1: EFFECTS OF LOSS OF VBD
Loss of power to RM-A-7G will close WDG-V-47.

3.5 If Unit 1 Waste Gas release was in progress, then ENSURE CLOSED WDG-V-47 (Radwaste Panel).

Additionally, IAW OP-TM-AOP-0181, Loss of VBD Basis Document:

Step 3.5 This step provides guidance to ensure that WDG-V-47 is closed if a waste gas release was in progress when VBD de-energized. Loss of power to RM-A-7 will cause WDG-V-47 to close. This step ensures that the release has been terminated to meet the requirements of ODCM Part 1 Table 2.1-2 Item 1.a.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073 K3.01	
	Importance Rating	4.0	

Knowledge of the effect that a loss or malfunction of the PRM system will have on the following:
Radioactive effluent releases.

Proposed Question:	RO Question # 35		
Technical Reference(s):	OP-TM-AOP-018, p 3,13, Rev 6 OP-TM-AOP-0181, p 4, Rev 4		
Proposed References to be provided to applicants during examination:			None
Learning Objective:	661-GLO-11		
Question Source:	Bank #	ID# 363641	
	Modified Bank #		
	New		
Question History:		Last NRC Exam:	03-01
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	13	
	55.43		

Comments:
The KA is matched because the operator must demonstrate the knowledge of the effect that a malfunction of the Process Radiation Monitoring System will have on radioactive effluent releases.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall what happens upon a loss of power to a radiation monitor.

What MUST be known:
1. Electrical logic associated with the RM-A-7 radiation monitor, including how the interlock defeat switch is connected electrically. 2. Interlock associated with RM-A-7 and WDG-V-47.

36 **ID: 1142215** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- Outside air temperature is 5F.
- The Boron content in the Boric Acid Mix Tank is 4000ppm.

Event:

- Main Annunciator D-2-4, Boric Acid Mix Tank Temp LO, is in alarm.

Given the above information, the operators will ensure that the BAMT temperature is greater than:

A. 10F to avoid Tech Spec LCO actions.

B. 40F to avoid Tech Spec LCO actions.

C. 10F above the crystallization temperature for the tank's boron concentration.

D. 40F above the crystallization temperature for the tank's boron concentration.

Answer: C

Answer Explanation

A. Incorrect.

Correct temperature, incorrect reason. The Borated Water Storage Tank is a tank that contains boron and is covered by Tech Specs. Therefore, the choice is plausible if the candidate mistakes the Boric Acid Mix Tank with that of the Borated Water Storage Tank. If so, the candidate will believe that there is a Tech Spec associated with the Boric Acid Mix Tank.

B. Incorrect.

Incorrect temperature, incorrect reason. The Borated Water Storage Tank is a tank that contains boron and is covered by Tech Specs. Therefore, the choice is plausible if the candidate mistakes the Boric Acid Mix Tank with that of the Borated Water Storage Tank, including the BWST temperature requirement of 40F. If so, the candidate will believe that there is a Tech Spec associated with the Boric Acid Mix Tank.

C. Correct.

The setpoint for the Boric Acid Mix Tank Temp LO alarm is 125F. This is to alert the operators that, if left unattended, boron crystallization could occur. IAW MAP D, D-2-4, Boric Acid Mix Tank Temp LO:

MANUAL ACTION REQUIRED:

1. Check tank level and temp. on Chem. Add. Panel
2. Verify power supply to the heater is available, if not check for tripped breaker on 1A E.S. 480V MCC Unit 15D.
3. If possible, monitor (EG-Y-1A Load, verify it is < 3000 kV) bypass HPI and/or LPI and/or defeat 4 psig ES signal and restore heater to service.
4. Insure BAMT is >10°F above crystallization temperature for its boron concentration.

D. Incorrect.

Incorrect temperature, correct reason. The Borated Water Storage Tank is a tank that contains boron and is covered by Tech Specs. Therefore, the choice is plausible if the candidate mistakes the Boric Acid Mix Tanktemperature with that of the Borated Water Storage Tank, including the BWST temperature requirement of 40F. If so, the candidate will remember the correct reason but confuse the temperature requirement. Additionally, the ppm boron in the Boric Acid Mix Tank is given in the stem to avoid a potential second answer. IAW 1301-5.1, BAMT Temp Channels, RBAST Temp Channel, Figure 1, Boric Acid Crystallization Curve, 40F could be a correct answer depending on what the ppm boron is. The number given assures that the temperature could go below 40F, making it an incorrect answer.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 K4.10	
	Importance Rating	3.2	

Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: Minimum temperature requirements on borated systems (prevent crystallization).

Proposed Question: RO Question # 36

Technical Reference(s): MAP D, D-2-4, p 1, Rev 12

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-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	6
	55.43	

Comments:
The KA is matched because the operator must demonstrate the knowledge of the minimum temperature requirements on the Boric Acid Mix Tank (part of the CVCS).

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall a manual action step from an alarm response, including the reason.

What MUST be known:
1. The procedurally guided temperature requirement for the Boric Acid Mix Tank.
2. The reason for maintaining the Boric Acid Mix Tank above the required minimum temperature.

37 **ID: 1146833** **Points: 1.00**

The Building Spray System is designed to reduce idoine levels by ___(1)___ which will reduce ___(2)___.

A. (1) 20%
(2) thyroid dose

B. (1) 20%
(2) stress corrosion cracking

C. (1) 95%
(2) thyroid dose

D. (1) 95%
(2) stress corrosion cracking

Answer: C

Answer Explanation

A. Incorrect.

Part 1 is incorrect. Plausible if the candidate confuses the Building Spray iodine removal percentage of effectiveness (95%) with the Building Spray iodine removal factor. IAW OP-TM-214-901, RB Spray Operation:

> NOTES:
> (1) Manual initiation of RB spray is recommended for any one of the following conditions:
> - o There is a significant RCS leak (> 500 GPM @ 1600 psig) and a significant breech in containment (stuck open purge line, blown seals or open equipment or personnel hatch).
> - o T clad > 1400°F or failed fuel is expected to exceed 5% and RB pressure > 5 psig or a significant breech in containment.
> - o Projected child thyroid dose at site boundary exceeds 5 REM due to containment leakage.
> - o IAW Severe Accident Management Guidelines.
>
> (2) **RB Spray is expected to reduce reactor building iodine levels by 95% (factor of 20).** This will reduce off site thyroid dose.

Part 2 is correct. IAW OP-TM-214-901, RB Spray Operation:

> NOTES:
> (1) Manual initiation of RB spray is recommended for any one of the following conditions:
> - o There is a significant RCS leak (> 500 GPM @ 1600 psig) and a significant breech in containment (stuck open purge line, blown seals or open equipment or personnel hatch).
> - o T clad > 1400°F or failed fuel is expected to exceed 5% and RB pressure > 5 psig or a significant breech in containment.
> - o Projected child thyroid dose at site boundary exceeds 5 REM due to containment leakage.
> - o IAW Severe Accident Management Guidelines.
>
> (2) RB Spray is expected to reduce reactor building iodine levels by 95% (factor of 20). **This will reduce off site thyroid dose.**

B. Incorrect.

Part 1 is incorrect. Plausible if the candidate confuses the Building Spray iodine removal percentage of effectiveness (95%) with the Building Spray iodine removal factor. IAW OP-TM-214-901, RB Spray Operation:

> NOTES:
> (1) Manual initiation of RB spray is recommended for any one of the following conditions:
> - o There is a significant RCS leak (> 500 GPM @ 1600 psig) and a significant breech in containment (stuck open purge line, blown seals or open equipment or personnel hatch).
> - o T clad > 1400°F or failed fuel is expected to exceed 5% and RB pressure > 5 psig or a significant breech in containment.
> - o Projected child thyroid dose at site boundary exceeds 5 REM due to containment leakage.
> - o IAW Severe Accident Management Guidelines.
>
> (2) **RB Spray is expected to reduce reactor building iodine levels by 95% (factor of 20).** This will reduce off site thyroid dose.

Part 2 is incorrect. Plausible if the candidate confuses the reason for the Building Spray Pumps with the reason for the Trisodium Phosphate baskets. IAW TQ-TM-104-214-C001, Reactor Building Spray:

- Addition of TSP to RB will:
  - Raise pH of Reactor Building sump water for long term corrosion control of Safety related components
  - Maintain Iodine in solution as an ion and does not let it assume the Iodine gaseous state.

- The TSP chemical functions to mitigate the consequences of an accident by controlling the RB sump pH. In the event of a LOCA and containment flooding, the pH of the sump water would be maintained between 7.3 and 8.0. Control of post LOCA containment sump pH is important for the following reasons:
  - Too low a pH could result in:
  - Radioactive iodine evolving from the sump water into containment atmosphere thus increasing dose rates inside containment and increasing the potential for control room and offsite doses.
  - Stress corrosion cracking which could lead to failures in safety related equipment or components.

C. Correct.
Part 1 is correct. IAW OP-TM-214-901, RB Spray Operation:

NOTES:
(1) Manual initiation of RB spray is recommended for any one of the following conditions:
- There is a significant RCS leak (> 500 GPM @ 1600 psig) and a significant breech in containment (stuck open purge line, blown seals or open equipment or personnel hatch).
- T clad > 1400°F or failed fuel is expected to exceed 5% and RB pressure > 5 psig or a significant breech in containment.
- Projected child thyroid dose at site boundary exceeds 5 REM due to containment leakage.
- IAW Severe Accident Management Guidelines.

(2) **RB Spray is expected to reduce reactor building iodine levels by 95% (factor of 20).** This will reduce off site thyroid dose.

Part 2 is correct. IAW OP-TM-214-901, RB Spray Operation:

NOTES:
(1) Manual initiation of RB spray is recommended for any one of the following conditions:
- There is a significant RCS leak (> 500 GPM @ 1600 psig) and a significant breech in containment (stuck open purge line, blown seals or open equipment or personnel hatch).
- T clad > 1400°F or failed fuel is expected to exceed 5% and RB pressure > 5 psig or a significant breech in containment.
- Projected child thyroid dose at site boundary exceeds 5 REM due to containment leakage.
- IAW Severe Accident Management Guidelines.

(2) RB Spray is expected to reduce reactor building iodine levels by 95% (factor of 20). **This will reduce off site thyroid dose.**

D. Incorrect.
Part 1 is correct. IAW OP-TM-214-901, RB Spray Operation:

NOTES:
(1) Manual initiation of RB spray is recommended for any one of the following conditions:
- o There is a significant RCS leak (> 500 GPM @ 1600 psig) and a significant breech in containment (stuck open purge line, blown seals or open equipment or personnel hatch).
- o T clad > 1400°F or failed fuel is expected to exceed 5% and RB pressure > 5 psig or a significant breech in containment.
- o Projected child thyroid dose at site boundary exceeds 5 REM due to containment leakage.
- o IAW Severe Accident Management Guidelines.

(2) **RB Spray is expected to reduce reactor building iodine levels by 95% (factor of 20).** This will reduce off site thyroid dose.

Part 2 is incorrect. Plausible if the candidate confuses the reason for the Building Spray Pumps with the reason for the Trisodium Phosphate baskets. IAW TQ-TM-104-214-C001, Reactor Building Spray:

- Addition of TSP to RB will:
  - o Raise pH of Reactor Building sump water for long term corrosion control of Safety related components
  - o Maintain Iodine in solution as an ion and does not let it assume the Iodine gaseous state.
- The TSP chemical functions to mitigate the consequences of an accident by controlling the RB sump pH. In the event of a LOCA and containment flooding, the pH of the sump water would be maintained between 7.3 and 8.0. Control of post LOCA containment sump pH is important for the following reasons:
  - o Too low a pH could result in:
  - o Radioactive iodine evolving from the sump water into containment atmosphere thus increasing dose rates inside containment and increasing the potential for control room and offsite doses.
  - o Stress corrosion cracking which could lead to failures in safety related equipment or components.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026 K4.06	
	Importance Rating	2.8	

Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Iodine scaveging via the CSS.

Proposed Question: RO Question # 37

Technical Reference(s): OP-TM-214-901, p 1, Rev 4

Proposed References to be provided to applicants during examination: None

Learning Objective: 214-GLO-9

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 8

55.43

Comments:
The KA is matched because the operator must demonstrate the knowledge of the Building Spray System design feature for iodine removal.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall a note from an emergency procedure, including the reason.

What MUST be known:
1. The value, in percent, that the Building Spray System is designed to lower iodine levels by. 2. The result of lowering idoine levels by using the Building Spray System.

38 **ID: 1146834** **Points: 1.00**

The Main Steam Isolation Valves have a long-term closure function for containment integrity purposes for which of the following events?

1. Extended Loss of Instrument Air
2. Loss of Coolant Accident
3. OTSG Tube Rupture

A. 1 and 2.

B. 1 and 3.

C. 2 and 3.

D. 1, 2 and 3.

Answer: C

Answer Explanation

To answer the question, the candidate must recognize the following information:

- IAW TQ-TM-104-411-C001, Main Steam:

  e. The Main Steam Isolation Valves (MSIVs) have a long-term closure function for containment integrity for the following design basis accidents:
  1) Large Break LOCA
  2) Small Break LOCA
  3) Main Steam Line Break
  4) Steam Generator Tube Rupture

  f. Time required for the MSIV to close is not critical, is not relied upon in any accident analysis.
  g. The stop check feature of the MSIV will prevent blowdown of the opposite OTSG in the event of a main steam line break upstream of a MSIV.

OTSG Tube Rupture:
IAW OP-TM-EOP-005, OTSG Tube Leakage:

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.

3.35 IAAT both OTSGs are available, and projected or actual offsite integrated dose approaches 500 mRem TEDE or 1500 mRem CDE (thy), then perform the following:
1. When RCS pressure < 1000 psig, then INITIATE Attachment 1A or 1B to isolate the most affected OTSG.

3.59 IAAT BWST level < 22 ft, then perform the following:
1. When RCS pressure < 1000 psig, then INITIATE Attachment 1A or 1B to isolate the most affected OTSG.

3.60 IAAT BWST level < 15 ft, then perform the following:
1. If both OTSGs are affected, then INITIATE Attachment 1A or 1B to isolate the second OTSG.

Attachment 1A OTSG A Isolation

- **OTSG A**
  - MS-V-1A
  - MS-V-1B
  - FW-V-17A
  - FW-V-5A
  - FW-V-16A
  - FW-V-92A
  - EF-V-30A
  - EF-V-30D
  - MS-V-3D
  - MS-V-3E
  - MS-V-3F
  - ME-V-4A
  - MS-V-13A
  - CA-V-4A or CA-V-5A

Loss of Instrument Air:
There are many important Main Steam Valves associated with an extended Loss of Instrument air, but the Main Steam Isolation Valves are not among them. Plausible if the candidate believes that the Main Steam Isolation Valves should be closed on an extended Loss of Instrument Air either to conserve energy within the primary or to initiate containment isolation due to potential issues with the extended loss.

Loss of Coolant Accident:
IAW OP-TM-EOP-006, LOCA Cooldown:

3.32 ENSURE the following valves are Closed: (not more than two at a time)

- MS-V-1A
- MS-V-1B
- MS-V-1C
- MS-V-1D

Additionally, IAW OP-TM-EOP-0061, LOCA Cooldown Basis Document:

> Step 3.32 The step intent is to isolate containment penetrations. The Main Turbine is not needed during accident mitigation and the Main Steam isolation valves are closed to provide containment isolation for the large steam line penetrations.

A. Incorrect.
Although the Loss of Coolant Accident is correct, the Extended Loss of Instrument Air is not correct. See the explanations above.

B. Incorrect.
Although the OTSG Tube Rupture is correct, the Extended Loss of Instrument Air is not correct. See the explanations above.

C. Correct.
The OTSG Tube Rupture and the Loss of Coolant Accident are correct. See the explanations above.

D. Incorrect.
Although the OTSG Tube Rupture and the Loss of Coolant Accident are both correct, the Extended Loss of Instrument Air is not correct. See the explanations above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 K4.07	
	Importance Rating	3.4	

Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Reactor building isolation.

Proposed Question:	RO Question # 38
Technical Reference(s):	OP-TM-EOP-006, p 13, Rev 12 OP-TM-EOP-005, p 17,31,37, Rev 9 OP-TM-EOP-0061, p 13, Rev 7 TQ-TM-104-411-C001, p50, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-7

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	9
	55.43	

Comments:
The KA is matched because the operator must demonstrate the knowledge of the MainSteam System design feature for Reactor Building isolation.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall the reasons for isolating the Reactor Building portion of the Main Steam System.

What MUST be known:
1. The events that require the Main Steam Isolation Valves (MSIVs) to have a long-term closure function for containment integrity.

39 ID: 1146965 Points: 1.00

Which ONE of the following describes a function of the flywheel on the RCPs?

A. Prolongs RCP coastdown time to aid in maintaining loop flow thus maintaining DNBR within acceptable limits during certain loss of RCS flow events.

B. Counters excess acceleration on pump start to minimize the effects of core lift when the first RCP is started during an RCS heatup from Cold Shutdown.

C. Prolongs RCP coastdown time to aid in maintaining loop flow thus maintaining hot channel factors at an acceptable level during certain loss of RCS flow events.

D. Maintains constant RCP speed, minimizing the potential for spurious RCS low flow reactor trips and maintaining hot channel factors at an acceptable level during power operation.

Answer: A

Answer Explanation

A. Correct Answer.
Flywheel designed to provide inertia to aid DNBR. IAW TQ-TM-104-226-C001, Reactor Coolant Pumps and Motors:

e. Flywheels
1) Two per pump (upper and lower)
2) Add significant mass to the RCP motor which in turn provides a moment of inertia designed to increase coastdown time, which provides adequate flow (DNB Consideration) following a loss of power to the RCPs.

B. Incorrect.
Core lift is a concern during Cold Shutdown Pump starts, but not a reason for the flywheel

C. Incorrect.
Flywheel designed to provide inertia to aid DNBR, not specifically for hot channel factors. Hot channel factors are affected by control rods.

D. Incorrect.
Flywheel more important for loss of flow, where RCP coastdown time is important for heat removal. Hot channel factors are affected by control rods

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003	K5.02
	Importance Rating	2.8	

Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP coastdown on RCS parameters.

Proposed Question: RO Question # 39

Technical Reference(s): TQ-TM-104-226-C001, p 17, rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-2

Question Source:	Bank #	ID# 464198
	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	7
	55.43	

Comments:
The KA is matched because the operator must demonstrate the knowledge of the reason for extending the RCP coastdown time with regards to RCS parameters (Departure From Nucleate Boiling Ratio).

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recall the function of the RCP flywheels.

What MUST be known:
1. The function of the installed flywheels as they pertain to the Reactor Coolant Pumps.

40 **ID: 1142189** **Points: 1.00**

Plant Conditions (T=0 minutes):

- 1103-11, RCS Water Level Control, is controlling procedure preparing for RCS heatup.
- RCS final fill operation has been completed, with flow into the RCS terminated.
- Pressurizer level is 390 inches.
- Pressurizer temperature is 190 degrees F.
- Reactor Coolant Drain Tank (RCDT) pressure is 2.0 psig.
- Operator energizes Pressurizer heaters to form a steam bubble in the Pressurizer.

Event (T=60 minutes):

- RCS Pressure indication reaches 22 psig.
- RCDT level begins to rise.

Based on these conditions, which ONE of the following describes the source of the water flowing into the RCDT?

A. Hot leg vent(s).

B. Pressurizer vent.

C. RCP Seal Standpipe(s).

D. Center Control Rod Drive mechanism vent.

Answer: A

Answer Explanation

A. Correct.
RCDT will begin to rise when pressure is high enough to overfill one or both hot legs (flow will initiate out the hot leg vent(s). According to Steps 3.3.2.11-15 of 1103-11 when Pzr level is 390 ±2 inches RCS fill is terminated, and the manual Pzr vent valves are closed. When Pzr temperature has been > 230F for > 30 minutes, the Pzr Vent valves to the RCDT Sparger (RC-V-28 and 44) are closed. With a steam bubble in the Pzr, maintaining RCS pressure at 22 psig water will be forced out of the Pzr and into the RCS causing water to issue from the Hot Leg vents (RC-V15A/B, RC-V46A/B and RC-V-14A/B) and into the RDCT. At this point a Steam Bubble exists in the Pzr.

B. Incorrect.
This is plausible because the Pzr vents to the RCDT, and the Pzr manual vent valves are open during the RCS filling process, even at the point that the Pzr heaters are energized (Steps 10-11 of 1103-11).

C. Incorrect.
This is plausible because the RCP Seal standpipes overflow and drain to the RCDT. However, flow from the RCP standpipes is not possible at these conditions, and the standpipe bypass valves, RC-V-33A-D, are closed in accordance with OP-TM-220-000, Reactor Coolant System:

D. Incorrect.
This is plausible because the CRD venting system vents to the RCDT. However, In accordance with 1103-11, Enclosure 3A, the CRDM are vented and closed when the RCS is being filled, and Pzr level is between 280-360 inches. This would have already occurred at this point in the procedure.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 K5.02	
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as the apply to PRTS: Method of forming a steam bubble in the PZR.

Proposed Question: RO Question # 40

Technical Reference(s): 1103-11, p 17-21,49, Rev 71

Proposed References to be provided to applicants during examination: None

Learning Objective: GOP-012-PCO-5

Question Source:	Bank #	ID# 371272	
	Modified Bank #		
	New		
Question History:		Last NRC Exam:	2010
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	3	
	55.43		

Comments:
The KA is matched because the operator must demonstrate knowledge (i.e. indications of a steam bubble in the Pzr such as vent valves closed, 20-25 psig in the RCS and water issuing from the high point vents) of the operational implications of the method of forming a steam bubble in the PZR as it applies to PRTS.

The question is at the Comprehension/Analysis cognitive level because the operator must understand the process of drawing a bubble in the Pzr (i.e. with water level above the heaters but below the top of the Pzr, the heaters are energized, the water is brought to saturation and boiling occurs, and then the Pzr Vents are closed), and then relate it to its consequence (i.e. once the vents are close, water will back up into the system and ultimately issue from the high point, which must be known as well).

What MUST be known:
1. 1103-11 procedural guidance for drawing a steam bubble in the Pressurizer. 2. Indications that a steam bubble has been drawn in the Pressurizer. 3. Possible location sources for reactor coolant to exit the RCS while drawing a bubble in the Pressurizer.

41 **ID: 1142237** **Points: 1.00**

EFW is injected into the OTSG near the top of the tube bundle because the ___(1)___, and also to reduce the thermal stress on the ___(2)___ tubesheet.

A. (1) thermal center of the OTSG is elevated
(2) lower

B. (1) thermal center of the OTSG is elevated
(2) upper

C. (1) upper tube sheet can withstand the same amount of stress than the lower tube sheet
(2) lower

D. (1) upper tube sheet can withstand the same amount of stress than the lower tube sheet
(2) upper

Answer: A

Answer Explanation

A. Correct.

Part 1 is correct. IAW TQ-TM-104-424-C001, Emergency Feedwater System:

- The EFW nozzle ring is located on the outside of upper half of each OTSG. There are seven nozzle connections on the ring header, which supply flow to the upper tube sheet through spray nozzles.
  a. Reasons for discharging high in the OTSG:
     1) **Elevates the thermal center of the OTSGs to promote the establishment of Natural Circulation**
     2) Prevents thermal shock of the lower tube sheet.
  b. Thermal sleeves minimize thermal shock to the OTSG penetrations.

Part 2 is correct. IAW TQ-TM-104-424-C001, Emergency Feedwater System:

- The EFW nozzle ring is located on the outside of upper half of each OTSG. There are seven nozzle connections on the ring header, which supply flow to the upper tube sheet through spray nozzles.
  a. Reasons for discharging high in the OTSG:
     1) Elevates the thermal center of the OTSGs to promote the establishment of Natural Circulation
     2) **Prevents thermal shock of the lower tube sheet.**
  b. Thermal sleeves minimize thermal shock to the OTSG penetrations.

B. Incorrect.

Part 1 is correct. IAW TQ-TM-104-424-C001, Emergency Feedwater System:

- The EFW nozzle ring is located on the outside of upper half of each OTSG. There are seven nozzle connections on the ring header, which supply flow to the upper tube sheet through spray nozzles.
  a. Reasons for discharging high in the OTSG:
     1) **Elevates the thermal center of the OTSGs to promote the establishment of Natural Circulation**
     2) Prevents thermal shock of the lower tube sheet.
  b. Thermal sleeves minimize thermal shock to the OTSG penetrations.

Part 2 is incorrect. Plausible if the candidate confuses which tube sheet is prevented from experiencing thermal shock. IAW TQ-TM-104-424-C001, Emergency Feedwater System:

- The EFW nozzle ring is located on the outside of upper half of each OTSG. There are seven nozzle connections on the ring header, which supply flow to the upper tube sheet through spray nozzles.
  a. Reasons for discharging high in the OTSG:
     1) Elevates the thermal center of the OTSGs to promote the establishment of Natural Circulation
     2) **Prevents thermal shock of the lower tube sheet.**
  b. Thermal sleeves minimize thermal shock to the OTSG penetrations.

C. Incorrect.
Part 1 is incorrect. Plausible if the candidate believes that the tube sheet is the same design throughout. IAW TQ-TM-104-411-C001, Main Steam:

f) The water coming from the emergency feedwater system could be very cold. If this cold water were to enter the OTSG via the normal feedwater nozzles, severe thermal stress would occur to the lower tube sheet. The tubes, being smaller in size compared to the two foot thick lower tube sheet, can accept this thermal shock.

Part 2 is correct. IAW TQ-TM-104-424-C001, Emergency Feedwater System:

- The EFW nozzle ring is located on the outside of upper half of each OTSG. There are seven nozzle connections on the ring header, which supply flow to the upper tube sheet through spray nozzles.

a. Reasons for discharging high in the OTSG:
1) Elevates the thermal center of the OTSGs to promote the establishment of Natural Circulation
2) **Prevents thermal shock of the lower tube sheet.**
b. Thermal sleeves minimize thermal shock to the OTSG penetrations.

D. Incorrect.
Part 1 is incorrect. Plausible if the candidate believes that the tube sheet is the same design throughout. IAW TQ-TM-104-411-C001, Main Steam:

f) The water coming from the emergency feedwater system could be very cold. If this cold water were to enter the OTSG via the normal feedwater nozzles, severe thermal stress would occur to the lower tube sheet. The tubes, being smaller in size compared to the two foot thick lower tube sheet, can accept this thermal shock.

Part 2 is incorrect. Plausible if the candidate confuses which tube sheet is prevented from experiencing thermal shock. IAW TQ-TM-104-424-C001, Emergency Feedwater System:

- The EFW nozzle ring is located on the outside of upper half of each OTSG. There are seven nozzle connections on the ring header, which supply flow to the upper tube sheet through spray nozzles.

a. Reasons for discharging high in the OTSG:
1) Elevates the thermal center of the OTSGs to promote the establishment of Natural Circulation
2) **Prevents thermal shock of the lower tube sheet.**
b. Thermal sleeves minimize thermal shock to the OTSG penetrations.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 K5.01	
	Importance Rating	3.6	

Knowledge of the operational implications of the following concepts as the apply to AFW: Relationship between AFW flow and RCS heat transfer.

Proposed Question: RO Question # 41

Technical Reference(s): TQ-TM-104-424-C001, p 11-12, Rev 10

Proposed References to be provided to applicants during examination: None

Learning Objective: 424-GLO-2

Question Source:	Bank #	ID# 355467
	Modified Bank #	
	New	

Question History: Last NRC Exam: 1998

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:
The KA is matched because the operator must demonstrate knowledge of the relationship between EMergency Feedwater (AFW) flow and RCS heat transfer.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall the reasons for Emergency Feedwater entering the OTSG's higher than Main Feedwater does.

What MUST be known:
1. The reasons for Emergency Feedwater to be injected near the top of the tube bundles in the OTSG's.

42 **ID: 1148323** **Points: 1.00**

Plant Conditions:

- In preparation for initial Reactor Coolant Pump start following a refueling outage, interlocks are checked for RC-P-1A, resulting in the following information:
    - 0 RCPs are already running.
    - Bus voltage is 6.51 KV.
    - Total Seal Injection flow is 27 gpm.
    - Seal #1 delta-P is 215 psid.
    - Total ICCW flow is 625 gpm.
    - All RCP oil levels are OK.
    - Backstop oil flow is 1 gpm.
    - Lift oil pressure is 1200 psig.
    - RCS Temperature is 457F.
    - NS-P-1A is running.

Event:

- The CRO attempts to start RC-P-1A but it fails to start.

Given the above information, the Reactor Coolant Pump start failure was caused by _________ being too low.

A. ICCW flow

B. Bus Voltage

C. #1 Seal delta-P

D. Seal Injection flow

Answer: B

Answer Explanation

A. Incorrect.
Total ICCW flow is given as 625 gpm in the stem. The Total ICCW flow interlock setpoint is 550 gpm. The starting interlock is met for any value greater than than 550 gpm. Therefore, the contact associated with Total ICCW flow is closed.
Plausible if the candidate is not familiar with the Total ICCW flow setpoint associated with RCP starting circuitry.

B. Correct.
Bus voltage is given as 6.51 KV in the stem. However, the Bus voltage interlock setpoint is 6.62 KV. The starting interlock is NOT met for any value less than 6.62 KV. Therefore, the contact associated with Bus Voltage is not closed and the starting circuit does not have a path for current.

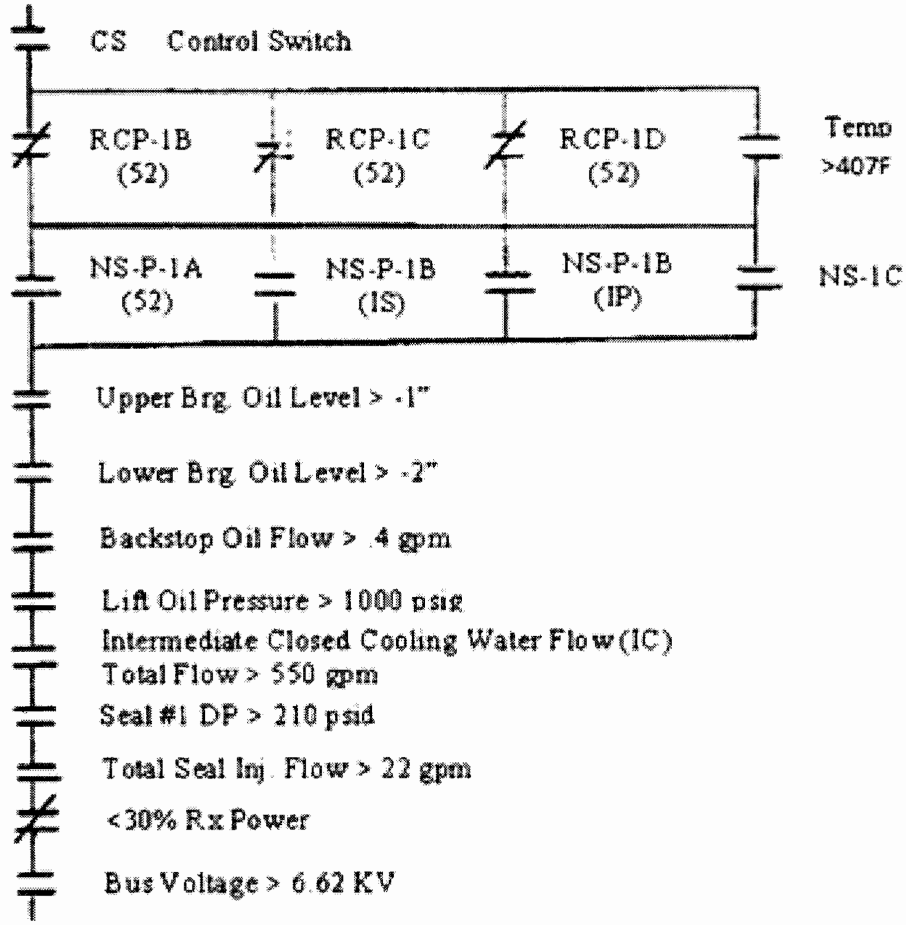
C. Incorrect.
#1 Seal differential pressure is given as 215 psid in the stem. The #1 Seal differential pressure interlock setpoint is 210 psid. The starting interlock is met for any value greater than than 210 psid. Therefore, the contact associated with #1 Seal differential pressure is closed.
Plausible if the candidate is not familiar with the #1 Seal differential pressure setpoint associated with RCP starting circuitry.

D. Incorrect.
Total Seal Injection flow is given as 27 gpm in the stem. The Total Seal Injection flow interlock setpoint is 22 gpm. The starting interlock is met for any value greater than than 22 gpm. Therefore, the contact associated with Total Seal Injection flow is closed.
Plausible if the candidate is not familiar with the Total Seal Injection flow setpoint associated with RCP starting circuitry.

Reactor Coolant Pump and Motor Interlocks, Trips and Alarms

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

Remote Start Logic (for RC-P-1A, Others are similar)



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 K6.14	
	Importance Rating	2.6	

Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Starting requirements.

Proposed Question: RO Question # 42

Technical Reference(s): OP-TM-226-000, p 12, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-5

Question Source: Bank #

Modified Bank # ID# 502404

New

Question History: Last NRC Exam: 1998

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:
The KA is matched because the operator must demonstrate knowledge of the effect that a malfunction of the #1 seal differential pressure will have on Reactor Coolant Pump starting interlocks.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall starting interlock setpoints associated with the Reactor Coolant Pumps.

What MUST be known:
1. The minimum value for the Total ICCW flow setpoint associated with RCP starting circuitry. 2. The minimum value for the Total Seal Injection flow setpoint associated with RCP starting circuitry. 3. The minimum value for the Bus Voltage setpoint associated with RCP starting circuitry. 4. The minimum value for the #1 Seal differential pressure setpoint associated with RCP starting circuitry.

Original Question:

Plant Conditions:

In preparation for initial Reactor Coolant Pump start following a refueling outage, interlocks are checked for RC-P-1A, resulting in the following information:
0 RCPs are already running.
Bus voltage is 6.75 KV.
Total Seal Injection flow is 30 gpm.
Seal #1 delta-P is 190 psid.
Total ICCW flow is 650 gpm.
All RCP oil levels are OK.
Backstop oil flow is OK.
Lift oil pressure is OK.
NS-P-1A is running.

Event:

The CRO attempts to start RC-P-1A but it fails to start.

Which of the following reasons is the cause of the Reactor Coolant Pump start failure?

A. ICCW is below 900 gpm.

B. Bus Voltage is below 6.8 kV.

C. Seal Injection is below 32 gpm.

D. #1 Seal dP is less than 210 psid.

Answer: D

43 **ID: 1142219** **Points: 1.00**

Plant Conditions:

- Reactor power is 100%, with ICS in full automatic.
- No surveillance testing in progress.

Given the above information, identify the ONE statement below that describes an RPS channel failure that will cause an RPS CHANNEL to trip.

A. Power imbalance fails to ZERO %.

B. Logic Relay Test Switch fails to TEST.

C. Loop 'A' RCS flow fails to ZERO lbm/hr.

D. RC-P-1B Pump Monitor Input to RPS fails OPEN.

Answer: C

Answer Explanation

A. Incorrect.
This failure mode will raise (rather than lower) the overpower trip setpoint based on total RCS flow and power imbalance, and a channel trip will not occur.

B. Incorrect.
One Logic Relay Test Switch failing to TEST will not cause a channel trip. It will cause two contacts in the 2 out of 4 logic circuit to OPEN.

C. Correct.
Zero (loop) flow with normal imbalance will generate an output (flux trip setpoint) signal that is less than 100% power, causing the RPS channel to trip on Flux/Flow/Imbalance.

D. Incorrect.
This failure mode will not result in a change to the high flux trip based on number of RCPs running, and therefore a channel trip will not occur. This high flux trip setpoint for 4 RCPs operating is the same as the setpoint for 3 RCPs operating.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 K6.11	
	Importance Rating	2.9	

Knowledge of the effect of a loss or malfunction on the following will have on the RPS: Trip setpoint calculators.

Proposed Question: RO Question # 43

Technical Reference(s): OPM F-02, p 27, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO-11

Question Source: Bank # ID# 356531

Modified Bank #

New

Question History:	Last NRC Exam:	2003

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

10 CFR Part 55 Content:	55.41	6
	55.43	

Comments:
The KA is matched because the operator must demonstrate knowledge of the effect that a malfunction of the RPS trip setpoint calcualtors will have on the Reactor Protection System..

The question is at the Comprehension/Analysis cognitive level because the operator must perform a cause-effect analysis associated with the Reactor Protection System.

What MUST be known:
1. The effect on RPS circuitry when power imbalance fails to 0%.
2. The effect on RPS circuitry when the logic relay test switch fails to the TEST position
3. The effect on RPS circuitry when the "A" Loop RCS flow fails to 0%.
4. The effect on RPS circuitry when RC-P-1B Pump Monitor input fails open.

44 **ID: 1142192** **Points: 1.00**

Plant Conditions (T=0 minutes):

- The plant is operating at 100% reactor power.

Sequence of Events:

- T= 30 minutes:
  - A Loss of off-site power (LOOP) occurs.
  - EG-Y-1A and EG-Y-1B failed to start.
  - Station Battery 1A load is 300 Amps and constant.
    - If battery load was maintained at 300 Amps, the battery would be discharged in 3.5 HOURS from now.
- T=60 minutes (**now**):
  - Station Battery 1A load shedding has just been completed in accordance with OP-TM-AOP-020, Loss of Station Power.
  - Load was reduced from 300 Amps down to 150 Amps in a short period of time.

Given the above information, identify the ONE selection below that describes how long Battery 1A will be able to supply the REMAINING loads **from now**.

A. 3.5 hours.

B. More than 3.5 hours, but less than 7 hours.

C. 7 hours.

D. More than 7 hours.

Answer: D

Answer Explanation

A. Incorrect.
Distracter is plausible because the original battery life was given as 3.5 hours and if the candidate does not recognize the reduced load, then the battery life would remain at 3.5 hours.

B. Incorrect.
Distracter is plausible because if the non-linear relationship between discharge rate and battery life where the extension of life is less than the reduction of discharge rate is assumed, then the life would fall some where between 3.5 and 7 hours.

C. Incorrect.
Distracter is plausible because with the discharge rate cut in half, if a linear relationship between change in discharge rate and change in battery life is assumed, then the battery life would be 7 hours.

D. Correct.
MAP A-1-7 gives the expected battery life for a 150 amp discharge rate as 9 hours. IAW Main Annunciator Panel A, A-1-7, Battery 1A Discharging:

NOTE: The following table shows expected battery time to discharge to 105VDC.

Amps	Total time to bank depletion
150	9 hours
200	6 hours
250	4 1/2 hours
300	3 1/2 hours
350	2 1/2 hours

Since the 300amp rate leads to bank depletion in 3.5 hours (210 minutes), and 30 miutes have passed, 1/7th of the battery has been discharged. Once 150amps is achieved as a battery discharge rate, the total time to bank depletion would be 9 hours (or 540 minutes). Since 1/7th of the battery has been discharged, the 9 hours is reduced in a less-than linear fashion. However, for ease of math, we will use a linear math equation to prove the correct answer. 1/7th of 540 is 462 minutes. COnverted into hours, that becomes 7.7 hours (which is greater than 7 hours). The non-linear factor would make the length of time even larger.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063 A1.01	
	Importance Rating	2.5	

Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including: Battery capacity, as it is affected by discharge rate.

Proposed Question: RO Question # 44

Technical Reference(s): MAP A, A-1-7, p 2, Rev 17

Proposed References to be provided to applicants during examination: None

Learning Objective: 734-GLO-6

Question Source:	Bank #	ID# 377067
	Modified Bank #	
	New	

Question History: Last NRC Exam: 2005

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

10 CFR Part 55 Content:	55.41	8
	55.43	

Comments:
The KA is matched because the operator must demonstrate the ability to predict the change in battery capacity based on a change in discharge rate.

The question is at the Comprehension/Analysis cognitive level because the operator must identify the variables that have changed and then perform a mathmatical computation to come to the proper conclusion.

What MUST be known:
1. Non-linear battery life available based on discharge rates. 2. The arithmetic to determine battery life left based on a value already discharged and a change in the discharge rate.

45 **ID: 1146835** **Points: 1.00**

Plant Conditions:

- The plant has experienced a loss of offsite power.

Event:

- PPC alarm A0372, Diesel Gen 1A Lube Oil Temp, is in alarm.
- The lube oil temperature has reached 225F and steady.

Given the above information, identify the ONE selection below that describes the correct action to take IAW OP-TM-861-901, Diesel Generator EG-Y-1A Emergency Operations?

A. Immediately trip EG-Y-1A.

B. Reduce load on EG-Y-1A.

C. Maintain current loading on EG-Y-1A.

D. Initiate OP-TM-861-910, Emergency Ventilation of EG-Y-1A Room.

Answer: B

Answer Explanation

A. Incorrect.
Distractor is plausible if the candidate believes that imminent damage is going to occur to the Emergency Diesel Generator. However, the lube oil temperature, although higher than the alarm setpoint, is stable, which indicates that loading is too high and the issue can be resolved by reducing load while maintaining EG-Y-1A running.

B. Correct.
IAW OP-TM-861-901, Diesel Generator EG-Y-1A Emergency Operations:

4.3 While EG-Y-1A is loaded (UNIT ops) on 1D 4160V bus

6. IAAT jacket coolant temperature (A0374) > 195°F or lube oil temperature (A0372) > 223°F, then REDUCE EDG load to maintain temperature within these limits.

C. Incorrect.
Distractor is plausible if the candidate believes that, since there is a Loss of Offsite Power event in progress, this is not an event that warrants changing the loading on the Emergency Diesel Generator.

D. Incorrect.
Distractor is plausible if the candidate recognizes the wording since that is a later step in the same section of the procedure. It is only used, however, if the Cooling Fan for the Diesel Generator room, AH-E-29A, is not operating. It does not provide any cooling effect to lube oil.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064 A1.01	
	Importance Rating	3.0	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ED/G system controls including: ED/G lube oil temperature and pressure.

Proposed Question: RO Question # 45

Technical Reference(s): OP-TM-861-901, p 4, Rev 15

Proposed References to be provided to applicants during examination: None

Learning Objective: 861-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	10
	55.43	

Comments:
The KA is matched because the operator must demonstrate the ability to monitor the change in Emergency Diesel Generator lube oil temperature and to take the correct action to prevent exceeding design limits.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must identify the correct action to take IAW procedures in order to not damage an Emergency Diesel Generator.

What MUST be known:
1. The procedural action to take based on out-of-spec parameters. 2. Procedural limits associated with Emergency Diesel Generator parameters.

46 **ID: 1149447** **Points: 1.00**

Plant Conditions (T=0 minutes):

- The plant is at Cold Shutdown conditions.
- "B" Decay Heat Removal train is in DHR Operating mode.
- Decay Closed cooling flow through the Decay Heat Removal coolers is throttled to maintain the RCS at 130°F.
- RM-L-2, "A" Decay Closed System Inline Radiation Monitor, is Out of Service for a power supply replacement.

Sequence of Events:

- T=1 minute:
  - o A Loss of Control Power occurs for DC-V-2B.
- T=2 minutes:
  - o Control Power is restored to DC-V-2B.

Given the above information and assuming no operator action has occurred, the RCS will ___(1)___ between T=1 minute and T=2 minutes, and the operators will regain control of RCS temperature IAW ___(2)___.

A. (1) heatup
(2) OP-TM-212-452, Control of DH Train B Flow and Temperatures

B. (1) heatup
(2) OP-TM-543-111, Shifting DC Train A From DHR Standby to DHR Operating Mode

C. (1) cooldown
(2) OP-TM-212-452, Control of DH Train B Flow and Temperatures

D. (1) cooldown
(2) OP-TM-543-111, Shifting DC Train A From DHR Standby to DHR Operating Mode

Answer: C

Answer Explanation

A. Incorrect.

Part 1 is incorrect. Between T=1 minute and T=2 minutes, DC-V-2B will fail open. This will allow maximum flow through DC-C-1B and will cause a cooldown of the RCS. Plausible if the student doesn't understand the failure modes of the valves. IAW OP-TM-543-000, Decay Heat Closed System:

2.1 Precautions

2.1.3 The Foxboro Spec 200 controllers are used in the control loops for DC-V-2A/B, DC-V-65A/B. On loss of control power, DC-V-2A/B fails Open, AND DC-V-65A/B fails Closed. (On power restoration, the position to which the valves travel is indeterminate.) Following control power restoration, verify position of DC-V-2A/B AND DC-V-65A/B for plant conditions.

Part 2 is correct. The crew would already be controlling DH flow and temperatures IAW OP-TM-212-452. Once control power is restored, there is no reason to do anything other than control DH flow and temperatures still. IAW OP-TM-212-452, Control of DH Train B Flow and Temperatures:

1.0 PURPOSE

1.1 The procedure provides direction for adjustments to RCS temperature, DH system flow, and control of DC Train B parameters while DH Train B is in DHR Operating Mode.

4.3 THROTTLE DC-V-2B and DC-V-65B as necessary to maintain the following (CC):

- Decay Closed System flow 3300 to 3400 gpm on DC-FI-27 (CC)
- Incore temperature below limits and as directed in applicable GOP (i.e. 1102-11, 1103-11 or 1102-1)
- Cooldown rate < 30F/HR or heatup rate < 50F/HR as measured using DH2TI2 if RCPs are off or Tcold if an RCP is operating.
- Minimum RCS temperature (as indicated by DH2TI2 (CC)) above limits for LTOP & Pzr Surge Line on Figure 1A of 1102-11 Plant Cooldown or 1102-1 Plant Heatup.

B. Incorrect.

Part 1 is incorrect. Between T=1 minute and T=2 minutes, DC-V-2B will fail open. This will allow maximum flow through DC-C-1B and will cause a cooldown of the RCS. Plausible if the student doesn't understand the failure modes of the valves. IAW OP-TM-543-000, Decay Heat Closed System:

2.1 Precautions

2.1.3 The Foxboro Spec 200 controllers are used in the control loops for DC-V-2A/B, DC-V-65A/B. On loss of control power, DC-V-2A/B fails Open, AND DC-V-65A/B fails Closed. (On power restoration, the position to which the valves travel is indeterminate.) Following control power restoration, verify position of DC-V-2A/B AND DC-V-65A/B for plant conditions.

Part 2 is incorrect. There is no reason to shift to the "A" Train. Control power to DC-V-2A has returned and therefore DH flows and temperatures, and ultimately, RCS temperature, can be controlled with the current procedure in use. Additionally, RM-L-2 is Out of Service and therefore flow cannot be verified through RM-L-2. IAW OP-TM-543-111, Shifting DC Train A From DHR Standby to DHR Operating Mode:

4.0 MAIN BODY

4.1 VERIFY all prerequisites have been met.

4.2 ENSURE the following:

DC-T-1A level 25 to 35_ as read on DC-LI-109 (CR).

_DC-V-2A/65A ENABLE/DEFEAT switch in ENABLE (PCR).

_DC-V-65A 55% open demand (CC).

_DC-V-2A 0% demand (CC).

4.3 START DC-P-1A (CC).

4.4 THROTTLE DC-V-65A to 3300 to 3400 gpm as read on DC-FI-26 (CC).

4.5 VERIFY flow through RM-L-2.

C. Correct.

Part 1 is correct. Between T=1 minute and T=2 minutes, DC-V-2B will fail open. This will allow maximum flow through DC-C-1B and will cause a cooldown of the RCS. IAW OP-TM-543-000, Decay Heat Closed System:

2.1 Precautions

2.1.3 The Foxboro Spec 200 controllers are used in the control loops for DC-V-2A/B, DC-V-65A/B. On loss of control power, DC-V-2A/B fails Open, AND DC-V-65A/B fails Closed. (On power restoration, the position to which the valves travel is indeterminate.) Following control power restoration, verify position of DC-V-2A/B AND DC-V-65A/B for plant conditions.

Additionally, IAW TQ-TM-104-533-C001, Decay Heat Service Water (Decay Heat River Water and Decay Heat Closed Cooling):

3) Power Supplies:

a) DC-V-2A/65A

(1) Solenoid power - 125 VDC Panel XCC (DC "A")

(2) Controller power – VBA

4) Loss of Foxboro Controller Power

a) If control power is lost, DC-V-2A/B will fail open and DC-V-65A/B will fail close to ensure full DC flow through the DH Removal cooler.

b) On power restoration, the position to which these valves travel is indeterminate so position must be verified based on plant conditions.

Part 2 is correct. The crew would already be controlling DH flow and temperatures IAW OP-TM-212-452. Once control power is restored, there is no reason to do anything other than control DH flow and temperatures still. IAW OP-TM-212-452, Control of DH Train B Flow and Temperatures:

1.0 PURPOSE

1.1 The procedure provides direction for adjustments to RCS temperature, DH system flow, and control of DC Train B parameters while DH Train B is in DHR Operating Mode.

4.3 THROTTLE DC-V-2B and DC-V-65B as necessary to maintain the following (CC):

- Decay Closed System flow 3300 to 3400 gpm on DC-FI-27 (CC)
- Incore temperature below limits and as directed in applicable GOP (i.e. 1102-11, 1103-11 or 1102-1)
- Cooldown rate < 30F/HR or heatup rate < 50F/HR as measured using DH2TI2 if RCPs are off or Tcold if an RCP is operating.
- Minimum RCS temperature (as indicated by DH2TI2 (CC)) above limits for LTOP & Pzr Surge Line on Figure 1A of 1102-11 Plant Cooldown or 1102-1 Plant Heatup.

D. Incorrect.

Part 1 is correct. Between T=1 minute and T=2 minutes, DC-V-2B will fail open. This will allow maximum flow through DC-C-1B and will cause a cooldown of the RCS. IAW OP-TM-543-000, Decay Heat Closed System:

2.1 Precautions

2.1.3 The Foxboro Spec 200 controllers are used in the control loops for DC-V-2A/B, DC-V-65A/B. On loss of control power, DC-V-2A/B fails Open, AND DC-V-65A/B fails Closed. (On power restoration, the position to which the valves travel is indeterminate.) Following control power restoration, verify position of DC-V-2A/B AND DC-V-65A/B for plant conditions.

Additionally, IAW TQ-TM-104-533-C001, Decay Heat Service Water (Decay Heat River Water and Decay Heat Closed Cooling):

3) Power Supplies:

a) DC-V-2A/65A

(1) Solenoid power - 125 VDC Panel XCC (DC "A")

(2) Controller power – VBA

4) Loss of Foxboro Controller Power

a) If control power is lost, DC-V-2A/B will fail open and DC-V-65A/B will fail close to ensure full DC flow through the DH Removal cooler.

b) On power restoration, the position to which these valves travel is indeterminate so position must be verified based on plant conditions.

Part 2 is incorrect. There is no reason to shift to the "A" Train. Control power to DC-V-2A has returned and therefore DH flows and temperatures, and ultimately, RCS temperature, can be controlled with the current procedure in use. Additionally, RM-L-2 is Out of Service and therefore flow cannot be verified through RM-L-2. IAW OP-TM-543-111, Shifting DC Train A From DHR Standby to DHR Operating Mode:

4.0 MAIN BODY
- 4.1 VERIFY all prerequisites have been met.
- 4.2 ENSURE the following:
  - _DC-T-1A level 25_ to 35_ as read on DC-LI-109 (CR).
  - _DC-V-2A/65A ENABLE/DEFEAT switch in ENABLE (PCR).
  - _DC-V-65A 55% open demand (CC).
  - _DC-V-2A 0% demand (CC).
- 4.3 START DC-P-1A (CC).
- 4.4 THROTTLE DC-V-65A to 3300 to 3400 gpm as read on DC-FI-26 (CC).
- 4.5 VERIFY flow through RM-L-2.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	005	A2.04	
	Importance Rating	2.9		

Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR valve malfunction.

Proposed Question: RO Question # 46

Technical Reference(s):
TQ-TM-104-533-C001, p 14,35-36, Rev 7
OP-TM-212-452, p1-2, Rev 5
OP-TM-543-000, p 2, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 533-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	5
	55.43	

Comments:

The KA is matched because the operator must demonstrate the ability to comprehend the consequence to the RCS upon a Loss of Instrument Air while on Decay Heat Removal and then identify the correct procedure to mitigate the consequence.

The question is at the Comprehension/Analysis cognitive level because the operator must make multiple jumps to understand the consequnce to the RCS upon a Loff of Instrument Air (since there is no direct cause-effect).

What MUST be known:
1. What the failure mode of Decay Closed Cooling Water valves are upon a Loss if Instrument Air. 2. How a change in Decay Closed Cooling Water valves affects the RCS. 3. What procedure is used to regain control of RCS temperature?

47 ID: 1146847 Points: 1.00

Plant Conditions:

- The plant is operating at 100% reactor power.
- AH-C-1 is in service.
- AH-C-1B is secured.

Event:

- Due to temperature concerns, AH-E-111A is started.
- AH-E-111B/C/D all remain shutdown.
- RB air temperatures are still slightly rising above the 320' elevation.

Given the above information, which one of the following is a correct statement?

A. AH-E-111A would experience a motor overload due to reverse air flow through AH-E-111B unless AH-E-111B is started IAW OP-TM-823-440, Controlling RB Air Temperature.

B. AH-E-111A would experience a motor overload due to reverse air flow through AH-E-111C unless AH-E-111C is started IAW OP-TM-823-440, Controlling RB Air Temperature.

C. AH-E-111A would NOT experience a motor overload because the discharge damper for AH-E-111B is normally closed and the associated Spray Pumps may be started IAW OP-TM-823-271, RB Ventilation System Lineup Verification.

D. AH-E-111A would NOT experience a motor overload because the discharge damper for AH-E-111C is normally closed and the associated Spray Pumps may be started IAW OP-TM-823-271, RB Ventilation System Lineup Verification.

Answer: A

Answer Explanation

A. Correct.

The fans associated with AH-C-1B are run in pairs, either AH-111A/B or AH-E-111C/D. IAW OP-TM-823-000, Reactor Building Heating and Ventilation System:

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

2.1.1 Reductions in purge flow during an outage can have significant impact on RB radiological environment. Radiological Control personnel should be notified in the event of any planned or unplanned flow reduction.

2.1.2 To avoid motor overload caused by reverse air flow through an idle fan, AH-C-1B fans should be run in pairs:

- AH-E-111A and AH-E-111B
- AH-E-111C and AH-E-111D

Additionally, IAW TQ-TM-104-824-C001, Reactor Building Ventilation:

Industrial Cooler, AH-C-1B (6.0 x 106 BTU/hr)

1) Four (4) fans, 25 hp each: AH-E-111A/111B/111C/111D

a) AH-D-111A/B/C/D fan discharge dampers.

These dampers are permanently mechanically blocked open to prevent an air supply failure from reducing the efficiency of the RB coolers (ECR 07-00582).

Lastly, AH-E-111B will be started IAW OP-TM-823-440, Controlling RB Air Temperature:

4.1 IAAT RB average air temperature above 320' elevation approaches an upper limit per the following:

4.1.2 PLACE control switches for the following in Normal After Start as desired to maintain temperature at or below upper limit:

3. If starting AH-E-111A/B:
   A. AH-E-111A
   B. AH-E-111B
4. If starting AH-E-111C/D:
   A. AH-E-111C
   B. AH-E-111D

B. Incorrect.

Plausible if the candidate is not familiar with which fans are paired together and use a odd/even logic. However, IAW OP-TM-823-000, Reactor Building Heating and Ventilation System:

2.0 PRECAUTIONS AND LIMITATIONS

2.1 Precautions

2.1.1 Reductions in purge flow during an outage can have significant impact on RB radiological environment. Radiological Control personnel should be notified in the event of any planned or unplanned flow reduction.

2.1.2 To avoid motor overload caused by reverse air flow through an idle fan, AH-C-1B fans should be run in pairs:

- AH-E-111A and AH-E-111B
- AH-E-111C and AH-E-111D

C. Incorrect.
Plausible if the candidate understands that AH-E-111B is associated with AH-E-111A but does not recall that the discharge fan dampers are mechanically blocked open. IAW TQ-TM-104-824-C001, Reactor Building Ventilation:

Industrial Cooler, AH-C-1B (6.0 x 106 BTU/hr)
1) Four (4) fans, 25 hp each: AH-E-111A/111B/111C/111D
a) AH-D-111A/B/C/D fan discharge dampers.
These dampers are permanently mechanically blocked open to prevent an air supply failure from reducing the efficiency of the RB coolers (ECR 07-00582).

D. Incorrect.
Plausible if the candidate is not familiar with which fans are paired together and use a odd/even logic and does not recall that the discharge fan dampers are mechanically blocked open. IAW OP-TM-823-000, Reactor Building Heating and Ventilation System:

2.0 PRECAUTIONS AND LIMITATIONS
2.1 Precautions
2.1.1 Reductions in purge flow during an outage can have significant impact on RB radiological environment. Radiological Control personnel should be notified in the event of any planned or unplanned flow reduction.
2.1.2 To avoid motor overload caused by reverse air flow through an idle fan, AH-C-1B fans should be run in pairs:
- AH-E-111A and AH-E-111B
- AH-E-111C and AH-E-111D

Additionally, IAW TQ-TM-104-824-C001, Reactor Building Ventilation:

Industrial Cooler, AH-C-1B (6.0 x 106 BTU/hr)
1) Four (4) fans, 25 hp each: AH-E-111A/111B/111C/111D
a) AH-D-111A/B/C/D fan discharge dampers.
These dampers are permanently mechanically blocked open to prevent an air supply failure from reducing the efficiency of the RB coolers (ECR 07-00582).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022	A2.01
	Importance Rating	2.5	

Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Fan motor over-current.

Proposed Question: RO Question # 47

Technical Reference(s):
TQ-TM-104-824-C001, p 14, Rev 6
OP-TM-823-000, p 3, Rev 7
OP-TM-823-440, p 2, Rev 5

Proposed References to be provided to applicants during examination: None

Learning Objective: 824-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 7

55.43

Comments:

The KA is matched because the operator must demonstrate the ability to predict whether a motor overload will occur to an RB Ventilation Fan upon a malfunction and to take the correct action to prevent exceeding design limits.

The question is at the Comprehension/Analysis cognitive level because the operator must analyze what will happen if only one of the RB Ventilation fans are started with the discharge dampers open.

What MUST be known:
1. What the consequence is to the non-running RB Ventilation fan if only one fan in a pair is started. 2. The correct procedure required to start the non-running RB Ventilation fan to prevent damage.

48 **ID: 1142190** **Points: 1.00**

Plant Conditions:

- EF-P-1, Steam Driven Emergency Feedwater Pump, is OOS.
- The plant is operating at 100% reactor power.

Event:

- Loss of Offsite Power (LOOP).
- OTSG "A" pressure 1010 psig.
- OTSG "B" pressure 900 psig.
- OTSG "A" level 8" lowering.
- OTSG "B" level 10" slowly increasing.
- "A" Emergency Diesel EG-Y-1A trips.

Given the above information, select the one choice below that correctly describes the status of EFW flow after the diesel trips, and required actions, if any.

A. Total EFW flow will be excessive and must be throttled to < 515 GPM total flow.

B. EFW flow will be limited by cavitating venturies and no additional action are required.

C. EFW flow will be sufficient to both OTSGs and must be throttled to < 185 GPM per OTSG.

D. EFW flow will be insufficient for "A" OTSG and must be throttled to "A" OTSG >215 GPM with total EFW flow < 515 GPM.

Answer: A

Answer Explanation

A. Correct.
IAW Rule 4 1st RNO if only 1 motor drive EFW maintain flow <515.

B. Incorrect.
Plausible if the examinee believes cavitating venturis will limit flow to prevent run-out, incorrect flow will be all into "B" with low pressure and only one motor driven pump.

C. Incorrect.
Plausible if the examinee believes Rule 4 requirements for No RPC and dry generator apply, incorrect because while RCPs are off OTSG dry criteria is not yet met.

D. Incorrect.
Plausible as flow to "A" OTSG may be reduced below the 215 GPM value. Incorrect since no requirement exists to stay above 215 GPM per OTSG unless LSCM exists.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	061	A2.08	
	Importance Rating	2.7		

Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Flow rates expected from various combinations of AFW pump discharge valves.

Proposed Question: RO Question # 48

Technical Reference(s): OP-TM-EOP-010, P 8, Rev 17

Proposed References to be provided to applicants during examination: None

Learning Objective: EOPR4-PCO-4

Question Source: Bank # ID# 893079

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		

Comments:

The KA is matched because the operator must demonstrate the ability to predict whether a EFW flow is excessive based on equipment malfunctions, specifically that the EFW Control Valves (which are downstream of the EFW Pumps) are open without sufficient EFW Pumps running and therefore the expected flow rates are higher than allowed. Finally, the operator will predict the correct action to take IAW procedures to prevent exceeding design limits.

The question is at the Comprehension/Analysis cognitive level because the operator must analyze given conditions and conclude that only one EFW pump is operating and determine that the expected flow rates are too high, and then must decide the procedurally guided action to take.

What MUST be known:
1. What the consequence is to the EFW flow rate upon EFW Pump Control Valves being opened with only one EFW Pump operating. 2. The correct procedure required to throttle EFW Flow by manipulating the EFW Control Valves to prevent damage.

49 **ID: 1149448** **Points: 1.00**

Plant Conditions (T = 0 seconds):

- The Reactor is at 75% power.
- Pressures are as follows:

Pressure Indication	Value (psig)
OTSG 1A Pressure	908
Turbine Header Pressure A	906
OTSG 1B Pressure	906
Turbine Header Pressure B	904

Sequence of Events:

- T = 1 second:
  - A steam leak occurs just upstream of MS-V-3D/E/F.
  - OP-TM-AOP-051, Secondary Side Steam Leak, is entered.

- T = 45 seconds:
  - The plant remains online.
  - Pressures are as follows:

Pressure Indication	Value (psig)
OTSG 1A Pressure	825
Turbine Header Pressure A	895
OTSG 1B Pressure	895
Turbine Header Pressure B	894

Based on 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 = 45 seconds?

A. MS-V-1A is open.
MS-V-1B is open.
MS-V-1C is seated closed.
MS-V-1D is seated closed.

B. MS-V-1A is open.
MS-V-1B is seated closed.
MS-V-1C is open.
MS-V-1D is seated closed.

C. MS-V-1A is seated closed.
MS-V-1B is open.
MS-V-1C is seated closed.
MS-V-1D is open.

D. MS-V-1A is seated closed.
MS-V-1B is seated closed.
MS-V-1C is open.
MS-V-1D is open.

Answer: D

Answer Explanation

A. 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 "B" OTSG as the affected OTSG.

B. Incorrect.

Plausible if the candidate incorrectly 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.

C. 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 correctly recognizes the "A" OTSG as the affected OTSG but does not know which of the four steam lines go to each OTSG.

D. 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. 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039	A3.02
	Importance Rating	3.1	

Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS

Proposed Question: RO Question # 49

Technical Reference(s): TQ-TM-104-411-C001, p 13-14, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-6

Question Source: Bank #

Modified Bank # ID# 950947

New

Question History: Last NRC Exam: 12-01 (2014)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

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?

Original Question:

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.
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

Based on 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

50 **ID: 1142193** **Points: 1.00**

Plant Conditions:

- The Reactor is at 100% power.

Event:

- An instrument air leak has lowered the instrument air header pressure such that it is stabilized at 77 psig.

Assuming no operator action, which ONE of the following identifies the current position of the following Instrument Air System valves?

(1) IA Backup from SA, IA-V-1
(2) Auto Isolation Valves, IA-V-2104A/B
(3) IA-Q-2 Bypass Valve, IA-V-2133

A. (1) Closed
(2) Closed
(3) Open

B. (1) Open
(2) Open
(3) Closed

C. (1) Closed
(2) Open
(3) Open

D. (1) Open
(2) Closed
(3) Closed

Answer: B

Answer Explanation

A. Incorrect.
This is plausible because the operator may incorrectly believe that the IA System is designed such that the IA-Q-2 Bypass Valve is the only valve that will be opened at this pressure, and that the other valves will be opened at a lower pressure.

B. Correct.
According to 1104-25, the Auto Isolation valves, IA-V-2104A/B, will open when the local Instrument Air header pressure (PT-1404) drops to 85 psig. Consequently this valve is open. According to TQ-TM-104-850-C001, IA-V-1,Instrument Air backup supply valve from Service Air, opens at 80 psig. Consequently, this valve is open. According to TQ-TM-104-850-C001, in Auto, IA-Q-2 Bypass Valve, IA-V-2133, will open (bypass IA-Q-2) when Instrument Air pressure at PS-1406 is <75 psig. Consequently, this valve is closed.

C. Incorrect.
This is plausible because the operator may incorrectly believe that the IA System is designed such that the IA-Q-2 Bypass Valve and the Auto Isolation Valves are the valves that will be opened at this pressure, and that the other valve will be opened at a lower pressure.

D. Incorrect.
This is plausible because the operator may incorrectly believe that the IA System is designed such that the IA Backup from SA Valve is the only valve that will be opened at this pressure, and that the other valves will be opened at a lower pressure.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039	A3.02
	Importance Rating	3.1	

Ability to monitor automatic operation of the IAS, including: Air Pressure

Proposed Question: RO Question # 50

Technical Reference(s): TQ-TM-104-850-C001, p 35,88, Rev 6
1104-25, p 17, Rev 149

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-GLO-10

Question Source: Bank # ID# 860265

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 4

55.43

Comments:
The KA is matched because the operator must demonstrate the ability to monitor automatic operation of the IAS, including air pressure. This is accomplished by presenting the operator with a set of conditions which includes a lowering IA pressure, and requiring the operator to conclude whether or not an automatic action has occurred.

The question is at the Comprehension/Analysis cognitive level because the operator must recall the automatic function setpoints of three valves, and then given an IA System pressure conclude whether or not an automatic action has occurred, to correctly answer the question.

What MUST be known:

1. At what pressure will the IA Backup from SA automatically open on lowering system air pressure?
2. At what pressure will the Auto Isolation Valves automatically open on lowering system air pressure?
3. At what pressure will the IA-Q-2 Bypass Valve automatically open on lowering system air pressure?
4. Under the present plant conditions, what position would these valves be in?

51 **ID: 1142217** **Points: 1.00**

From the list below, what is the result if the 69 Bypass switch for DC-P-1A at the 1P 480V ES Bus is placed in the **BYPASS** position?

A. Bypasses all DC-P-1A Interlocks.

B. Aligns the pump for operation during Remote Shutdown (RSD) conditions.

C. Enables the DC-P-1A breaker to be closed from either the local close pushbutton or from the Control Room.

D. Enables the DC-P-1A breaker to be closed from the local close push button and disables operation from the Control Room.

Answer: C

Answer Explanation

Old style 69 bypass (2 position - normal and bypass) is used to transfer control of certain components to allow local operation when control from the Control Room is not possible. When the key is inserted and turned the local operating controls are enabled. Override control is still available in the Control Room due to a parallel control circuit. This style bypass switch does not bypass interlocks.

A. Incorrect.

This will be chosen if the candidate confuses the non-keyed 2-position Pump breaker switches with the keyed 2-position Remote Shutdown switches. Plausible if the candidate does not recognize which switch is associated with DC-P-1A and if the candidate does not understand the positions available for the 69 Bypass switch for DC-P-1A. IAW TQ-TM-104-614-C001, Remote Shutdown System:

- 3 position 69 bypass (normal, bypass & emergency): Keyed (non-captured) brass switches.
  - Normal: As above
  - Bypass: Local control with interlock circuitry intact
  - Emergency: Relay Room and Control Room circuitry cut out, control of equipment at the breaker. Any interlocks in Control Room or Relay Room cabinets have no effect

- 2 position 69 bypass (normal and emergency): Not keyed.
  - Normal position: Control Room controls active.
  - Emergency position: Relay Room and Control Room circuitry cut out. Any interlocks in Control Room or Relay Room cabinets have no effect

- Important Note: do not confuse the RSD related 69 bypasses with the keyed brass 2 position (Normal/Bypass) switches located on many pump breakers that are used for local breaker control (for testing or when remote controls fail). RSD related switches have EMERGENCY positions that cutout CR controls.

B. Incorrect.

This will be chosen if the candidate does not understand the purpose of the non-keyed 2-position Pump breaker switches. Plausible if it is believed this type of 69 bypass transfers control to RSD and that DC-P-1A can be transferred to the RSD panel. IAW TQ-TM-104-614-C001, Remote Shutdown System:

- Important Note: do not confuse the RSD related 69 bypasses with the keyed brass 2 position (Normal/Bypass) switches located on many pump breakers that are used for local breaker control (for testing or when remote controls fail). RSD related switches have EMERGENCY positions that cutout CR controls.

C. Correct.

A keyed 2 position 69 bypass switch is used for DC-P-1A. IAW TQ-TM-104-533-C001, Decay Heat Service Water (Decay Heat River Water and Decay Heat Closed Cooling):

3) Key Operated 69 Selector Switch

a) DC-P-1A/1B breakers at 1P and 1S 480 V ES Switchgear are equipped with key operated 69 Switches. These switches have two positions: "NORMAL" AND "BYPASS".

b) Inserting the 69 key and turning the switch to the "BYPASS" position enables the local Close push button to close the breaker.

D. Incorrect.

This will be chosen if the candidate confuses the non-keyed 2-position Pump breaker switches with the keyed 2-position Remote Shutdown switches. Plausible if the candidate does not recognize which switch is associated with DC-P-1A and if the candidate does not understand the positions available for the 69 Bypass switch for DC-P-1A. IAW TQ-TM-104-614-C001, Remote Shutdown System:

- 2 position 69 bypass (normal and emergency): Not keyed.
    - Normal position: Control Room controls active.
    - Emergency position: Relay Room and Control Room circuitry cut out. Any interlocks in Control Room or Relay Room cabinets have no effect

- Important Note: do not confuse the RSD related 69 bypasses with the keyed brass 2 position (Normal/Bypass) switches located on many pump breakers that are used for local breaker control (for testing or when remote controls fail). RSD related switches have EMERGENCY positions that cutout CR controls.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005	A4.04
	Importance Rating	3.1	

Ability to manually operate and/or monitor in the control room: Controls and indication for closed cooling water pumps.

Proposed Question: RO Question # 51

Technical Reference(s): TQ-TM-104-533-C001, p36-37, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: 533-GLO-6

Question Source: Bank # ID# 1015059

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	

Comments:
The KA is matched because the operator must demonstrate the ability to manually operate the controls for a Decay Closed Cooling Water Pump from the Control Room given a set of indications.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall the locations for pump control when the 69 switch is placed in the bypass position.

What MUST be known:

1. What style 69 switch is associated with DC-P-1A?
2. What is the effect on local controls when the 69 switch associated with DC-P-1A is placed in the Bypass position?
3. What is the effect on Control Room controls when the 69 switch associated with DC-P-1A is placed in the Bypass position?

52 **ID: 1146967** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- NR-V-15A, IC-C-1A River Outlet Valve, is full open.
- NR-V-15B, IC-C-1B River Outlet Valve, is closed.
- NR-V-16A, NS-C-1A River Outlet Valve, is full open.
- NR-V-16B, NS-C-1B River Outlet Valve, is full open.
- NR-V-16C, NS-C-1C River Outlet Valve, is full open.
- NR-V-16D, NS-C-1D River Outlet Valve, is closed.
- NS-V-9D, NS-C-1D Outlet Valve, is full open.
- No evolutions are in progress.

Event:

- Nuclear River Water system temperature has risen.
- IC6-TI, ICCW COOLER OUTLET TEMPERATURE IND, is 93°F and rising slowly.
- PPC Point A0330, Nuclear Service Pump Discharge Temp, is 97°F and rising slowly.

Which of the following correctly identifies the procedural action that MUST be taken, if any, IAW OP-TM-541-461, IC & NS Temperature Control?

A. No action is required.

B. Throttle open NR-V-15B, ONLY.

C. Throttle open NR-V-16D, ONLY.

D. Throttle open NR-V-15B and NR-V-16D.

Answer: C

Answer Explanation

A. Incorrect.

Plausible if the candidate does not recognize that the NSCCW cooler outlet temperature is high out of band. NSCCW temperatures are not within the correct band and therefore action is required with regards to Nuclear Services Closed Cooling Water. IAW OP-TM-541-461, IC and NS Temperature Control:

4.2.6 If NS cooler outlet temperature (PPC Point A0330) is greater than 90F, then perform the following:

1. VERIFY the following procedures are not in progress:
    - OP-TM-232-557, Operation of the RC Evaporator
    - OP-TM-232-560, Operation of the Miscellaneous Waste Evaporator
2. OPEN NS-V-9D.
3. THROTTLE OPEN NR-V-16D to maintain NS cooler outlet temperature PPC Point A0330 between 70F and 95F.
4. When NR-V-16D is Closed and NS cooler outlet temperature PPC Point A0330 is less than 85F, then CLOSE NS-V-9D.

B. Incorrect.

Plausible if the candidate does not recognize that the NSCCW cooler outlet temperature is high out of band but believes that ICCW cooler outlet temperature is high out of band. ICCW temperatures are within the correct band and no therefore no action is required with regards to Intermediate Closed Cooling Water. IAW OP-TM-541-461, IC and NS Temperature Control:

4.1.3 IAAT NR-V-15A is full OPEN, or IC-C-1A is removed from service, then THROTTLE NR-V-15B to maintain IC cooler outlet temperature IC-6TI (CR) between 90F and 100F.

C. Correct.

ICCW temperatures are within the correct band and no therefore no action is required with regards to Intermediate Closed Cooling Water. IAW OP-TM-541-461, IC and NS Temperature Control:

4.1.3 IAAT NR-V-15A is full OPEN, or IC-C-1A is removed from service, then THROTTLE NR-V-15B to maintain IC cooler outlet temperature IC-6TI (CR) between 90F and 100F.

NSCCW temperatures are not within the correct band and therefore action is required with regards to Nuclear Services Closed Cooling Water. IAW OP-TM-541-461, IC and NS Temperature Control:

4.2.6 If NS cooler outlet temperature (PPC Point A0330) is greater than 90F, then perform the following:

1. VERIFY the following procedures are not in progress:
    - OP-TM-232-557, Operation of the RC Evaporator
    - OP-TM-232-560, Operation of the Miscellaneous Waste Evaporator
2. OPEN NS-V-9D.
3. THROTTLE OPEN NR-V-16D to maintain NS cooler outlet temperature PPC Point A0330 between 70F and 95F.
4. When NR-V-16D is Closed and NS cooler outlet temperature PPC Point A0330 is less than 85F, then CLOSE NS-V-9D.

D. Incorrect.
Plausible if the candidate recognizes that the NSCCW cooler outlet temperature is high out of band but believes that ICCW cooler outlet temperature is also high out of band.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008	A4.10
	Importance Rating	4.2	

Component Cooling Water System: Ability to manually operate and/or monitor in the control room: Conditions that require the operation of two CCW coolers.

Proposed Question: RO Question # 52

Technical Reference(s): OP-TM-541-461, p 3,5, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 531-GLO-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 10
55.43

Comments:

The KA is matched because the operator must demonstrate the ability to monitor CCW temperatures and determine the need for additional coolers to be placed in service. The K/A describes 2 CCW coolers, but since we have 4 NSCCW coolers, the question addresses the need for a 4th cooler.

The question is at the Comprehension/Analysis cognitive level because the operator must interpret data and then determine the correct action to take as result.

What MUST be known:
1. What is the allowable temperature band for the Intermediate Closed Cooling Water System?
2. What is the allowable temperature band for the Nuclear Services Closed Cooling Water System?
3. What is the appropriate action to take upon a high system temperature for the Nuclear Services Closed Cooling Water System?

53 **ID: 1147090** **Points: 1.00**

Given the following conditions:

- A refueling outage has just been completed.
- Preparations are being made to heat up the RCS and conduct a reactor startup.
- RCS temperature is 180°F.
- RCS pressure is 350 psig.
- RCS boron concentration is 2050 ppm.

Given the above information, and assuming that shutdown margin is maintained greater than 1% $\Delta k/k$, which ONE of the following conditions requires CONTAINMENT INTEGRITY to be established?

A. Raising RCS temperature by 30°F.

B. Raising RCS pressure by 100 psig.

C. Lowering boron concentration by 125 ppm.

D. Shifting from DHR cooling to RCP operation for SG cooling.

Answer: A

Answer Explanation

A. Correct.
RCS Pressure is already greater than 300 psig and nuclear fuel is already in the core (both from the given conditions in the question stem). Therefore, as soon as RCS temperature rises above 200°F, then containment integrity is required IAW Tech Specs. With a given initial RCS temperature of 180°F, a rise of 30°F will place RCS temperature at 210°F. IAW Tech Spec 3.6.1:

> T.S 3.6.1- Except as provided in 3.6.6, 3.6.8 and 3.6.12, CONTAINMENT INTEGRITY (section 1.7) shall be maintained whenever all three of the following conditions exist:
> a) Reactor Coolant pressure is 300 psig or greater,
> b) Reactor Coolant temperature is 200 degrees F or greater,
> c) Nuclear fuel is in the core.

B. Incorrect.
With an initial RCS pressure of 350 psig, the requirements for RCS pressure are already exceeded. And although there is nuclear fuel in the core, RCS temperature is NOT greater than 200°F. Therefore, since all three requirements (pressure, temperature, and fuel) must be met, Containment Integrity is NOT required. Plausible if the candidate is not familiar with the setpoints and beleives that, initially, RCS pressure is the only parameter under the required setpoint.

C. Incorrect.
Plausible if the candidate believes that boron concentration is a part of the Integrity Containment criteria, regardless of the shutdown margin. IAW Tech Spec 3.6.1:

> T.S 3.6.1- Except as provided in 3.6.6, 3.6.8 and 3.6.12, CONTAINMENT INTEGRITY (section 1.7) shall be maintained whenever all three of the following conditions exist:
> a) Reactor Coolant pressure is 300 psig or greater,
> b) Reactor Coolant temperature is 200 degrees F or greater,
> c) Nuclear fuel is in the core.

D. Incorrect.
Plausible if the candidate believes that shifting to S/G cooling would change containment from a method within the primary to a method that includes the secondary plant. IAW Tech Spec 3.6.3:

> T.S 3.6.3 Positive reactivity insertions which would result in a reduction in shutdown margin to less than 1% delta k/k shall not be made by control rod motion or boron dilution unless CONTAINMENT INTEGRITY is being maintained.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	022	A4.05	
	Importance Rating	3.8		

Ability to manually operate and/or monitor in the control room: Containment readings of temperature, pressure, and humidity system

Proposed Question: RO Question # 53

Technical Reference(s): Tech Spec 3.6, p 3-41, Amendment 278

Proposed References to be provided to applicants during examination: None

Learning Objective: 240-GLO-14

Question Source: Bank #

Modified Bank # ID# 374203

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

Comments:
The KA is matched because the operator must demonstrate the ability to monitor Containment parameters (RCS temperature, RCS pressure, and RCS boron concentration) and determine when containment integrity is required.

The question is at the Comprehension/Analysis cognitive level because the operator must interpret data and perform mathematical computations to determine which containment parameter will require containment integrity to be set.

What MUST be known:
1. What is the RCS pressure Tech Spec setpoint required for containment integrity to be set?
2. What is the RCS temperature Tech Spec setpoint required for containment integrity to be set?

Original Question:

Given the following conditions:

A refueling outage has just been completed. Preparations are being made to heat up the RCS and conduct a reactor startup.
RCS temperature is 180°F.
RCS pressure is 350 psig.
RCS boron concentration is 2050 ppm.

Which ONE of the following conditions requires CONTAINMENT INTEGRITY to be established?

A. Shifting from DHR cooling to RCP operation for SG cooling.

B. Reducing boron concentration to 1980 ppm.

C. Increasing RCS pressure to 400 psig.

D. Increasing RCS temperature to 205° F.

Answer: D

54 **ID: 1149449** **Points: 1.00**

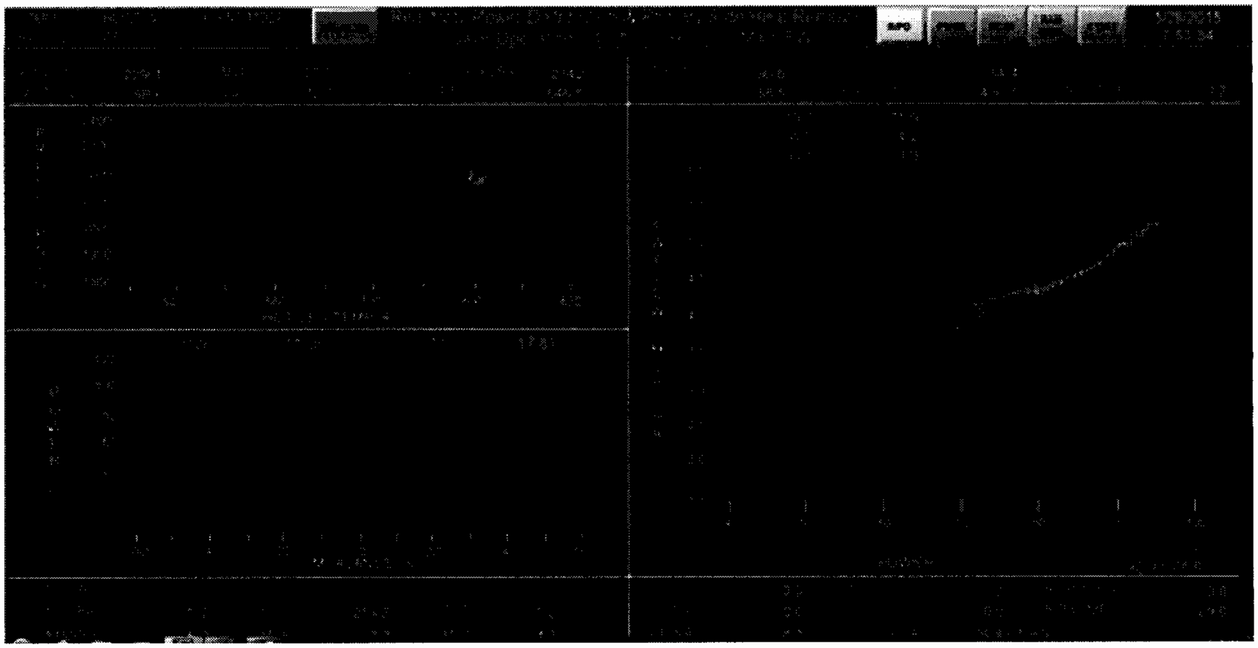
Plant Conditions:

- A plant power reduction is occurring, currently at 71% reactor power.

Event:

- A Reactor Coolant Pump has been secured.
- The ICS runback is complete.
- No manual operator actions have occurred.

Given the Plant Processing Computer screenshot below and assuming all equipment operates as designed, the URO has secured ___(1)___, and Main Feedwater ___(2)___ completely re-ratioed.



A. (1) RC-P-1B
(2) has

B. (1) RC-P-1B
(2) has NOT

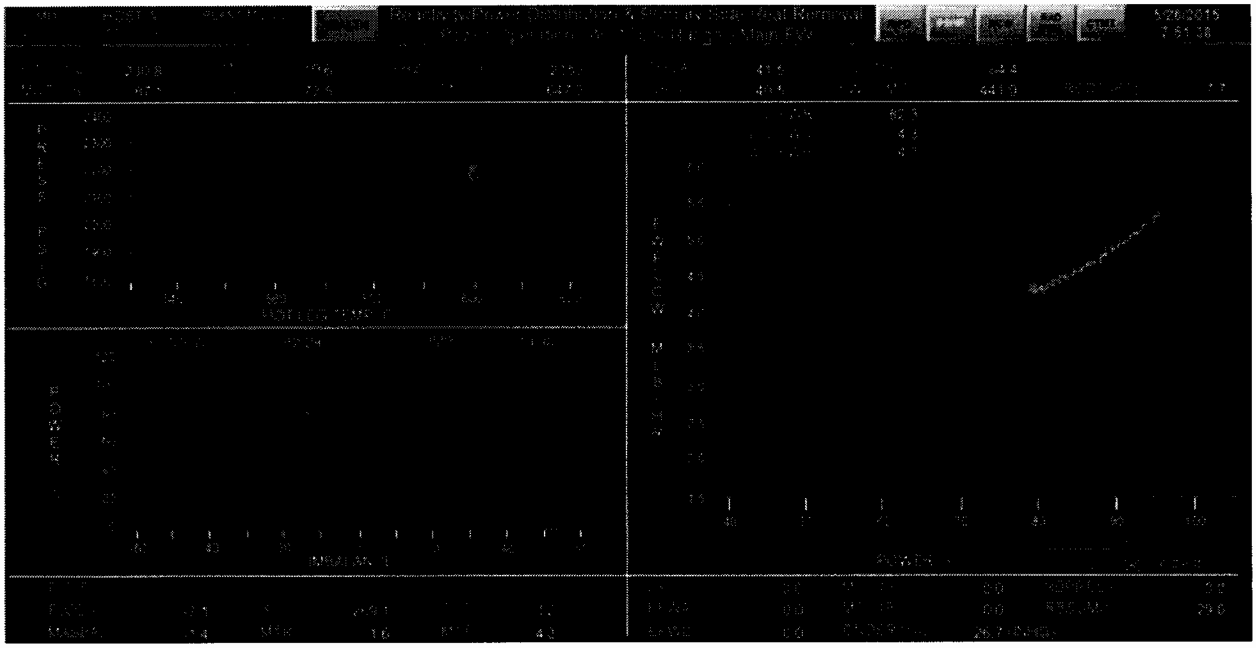
C. (1) RC-P-1C
(2) has

D. (1) RC-P-1C
(2) has NOT

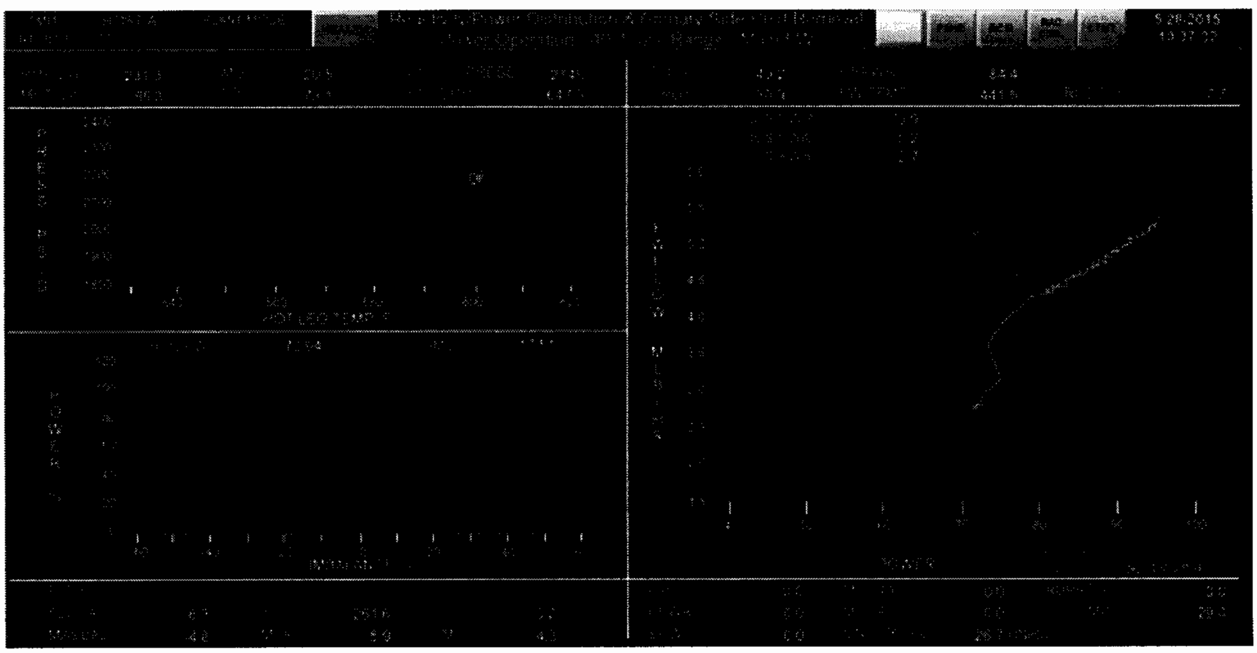
Answer: D

Answer Explanation

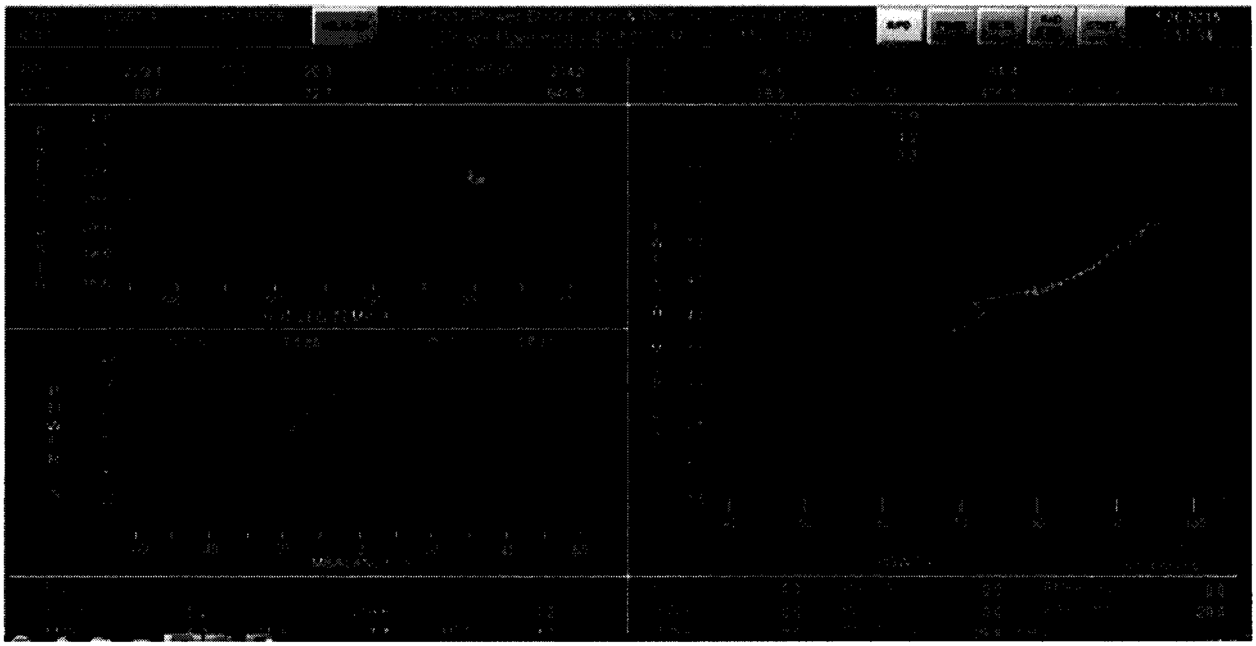
PPC Screenshot prior to the RCP trip:



PPC Screenshot after RCP trip, with ICS in AUTO:



PPC Screenshot after RCP trip, with ICS in HAND:



A. Incorrect.
Part 1 is incorrect. Plausible if the candidate believes that RC-P-1C is in the "A" RCS loop or misinterprets the screenshot from the Plant Processing Computer. IAW Operator Aid #107, The "B" Feedwater Loop will have less flow when there are two pumps operating in the "A" RCS loop and one pump operating in the "B" RCS Loop.

Part 2 is incorrect. Since reactor power is initially at approximately 80%, a small runback will occur. Additionally, when a Reactor Coolant Pump is secured, the feedwater flow to each OTSG should not be identical. The screenshot from the Plant Processing Computer shows relatively identical flow (slightly off due to the Tcold circuitry). Therefore, Main Feedwater has NOT reratioed. Plausible if the candidate does not recognize that, with a Reactor Coolant Pump secured, the feedwater flow to each OTSG should not be identical. The above pictures describe what the screenshot would look like if ICS Feedwater control had and had not reratioed.

B. Incorrect.
Part 1 is incorrect. Plausible if the candidate believes that RC-P-1C is in the "A" RCS loop or misinterprets the screenshot from the Plant Processing Computer. IAW Operator Aid #107, The "B" Feedwater Loop will have less flow when there are two pumps operating in the "A" RCS loop and one pump operating in the "B" RCS Loop.

Part 2 is correct. A proper interpretation of the Plant Processing Computer screenshot will reveal that feedwater flows are relatively even as compared to the expected response. However, they should not be matched upon a loss of a Reactor Coolant Pump. IAW TQ-TM-104-621-C001, Integrated Control System:

- LOSS OF ONE REACTOR COOLANT PUMP
  1. INITIAL PLANT CONDITIONS
     - Reactor at 75% power with all Integrated Control System stations in automatic.
  2. OVERVIEW OF TRANSIENT
     - A loss of one reactor coolant pump at greater than 75% power would actuate a plant runback to 75% at a rate of 50% per minute. However, with power already at 75%, this will not occur. The loss of reactor coolant flow will be sensed through the T control circuit, and will ratio feedwater flow accordingly. Ultimately feedwater will be ratioed 2.4:1, with approximately 2.0×10^6 lbm/hr., while flow in the non-affected loop rises to approximately 76×10^6 lbm/hr. feedwater flow to the affected OTSG.
  3. DETAILED DESCRIPTION

     A. Upon loss of the Reactor Coolant Pump, a reduction in Reactor Coolant Flow (RC Flow) occurs within the affected loop but rises in the non-affected loop. Within ten seconds RC Flow in the affected loop has lowered to approximately 32×10^6 lbm/hr, while flow in the non-affected loop rises to approximately 10^6 lbm/hr

     B. The differential flow developed between A and B loops will:
     1. Activate the ΔTc control circuit
     2. Swap Tavg control to the loop with the greatest flow
        - The Tc control has a circuit sensitive to RC Flow Differential, if greater than 10%. This circuit offers rate sensitive/promotional control. Thus, an immediate response to re-ratio feedwater demand proportionate with the rate of change to RC Flow occurs.

     C. An rise in feedwater flow to the non-affected loop is necessary to provide the heat removal required due to the rise in heat input (RC Flow).

     D. A drop in feedwater flow to the affected loop is necessary since the heat input (RC Flow) has been reduced.

     E. Within two minutes, the plant is stabilized with only fine tone requirements; these are provided by Tc integral/proportional action.

C. Incorrect.

Part 1 is correct. The Plant Processing Computer screenshot shows that the "B" RCS loop has a lower flow than the "A" RCS loop. The Reactor Coolant Pump physical layout is as follows:
IAW OP-TM-MAP-F0301, RC Loop A Flow LO:

2.0 Causes
- RC-P-1A and/or RC-P-1B malfunction or trip
- Loss of a 230KV feeder single phase to an Auxiliary Transformer

IAW OP-TM-MAP-F0302, RC Loop B Flow LO:

2.0 Causes
- RC-P-1C and/or RC-P-1D malfunction or trip
- Loss of a 230KV feeder single phase to an Auxiliary Transformer

Additionally, IAW Operator Aid #107, The "B" Feedwater Loop will have less flow when there are two pumps operating in the "A" RCS loop and one pump operating in the "B" RCS Loop. Although the Tcold circuitry has adjusted feedwater flow slightly, it shows the "B" Feedwater Loop as having less flow (The Blue line if printed in color, the "X" if printed in Black and White). Therefore, it indicates that a Reactor Coolant Pump in the "B" RCS Loop has tripped. Since RC-P-1C and RC-P-1D are in the "B" RCS Loop, it must be one of those two pumps. IAW Operator Aid #107:

Operator Aid # 107

	A Loop 2 pumps		B Loop 1 pump	
Power Level	Flow	Demand	Flow	Demand
75%	5.7	95%	2.4	41%
70%	5.3	88%	2.3	38%
65%	4.9	82%	2.1	35%
60%	4.5	76%	1.9	32%
55%	4.2	69%	1.8	30%
50%	3.8	63%	1.6	27%
45%	3.4	57%	1.5	24%
40%	3.0	50%	1.3	22%
35%	2.6	44%	1.1	19%
30%	2.3	38%	1.0	16%
25%	1.9	32%	0.8	14%
20%	1.5	25%	0.6	11%
15%	1.1	19%	0.5	8%
10%	0.8	13%	0.3	5%
5%	0.4	6%	0.2	3%

	A Loop 1 pump		B Loop 2 pumps	
Power Level	Flow	Demand	Flow	Demand
75%	2.4	41%	5.7	95%
70%	2.3	38%	5.3	88%
65%	2.1	35%	4.9	82%
60%	1.9	32%	4.5	76%
55%	1.8	30%	4.2	69%
50%	1.6	27%	3.8	63%
45%	1.5	24%	3.4	57%
40%	1.3	22%	3.0	50%
35%	1.1	19%	2.6	44%
30%	1.0	16%	2.3	38%
25%	0.8	14%	1.9	32%
20%	0.6	11%	1.5	25%
15%	0.5	8%	1.1	19%
10%	0.3	5%	0.8	13%
5%	0.2	3%	0.4	6%

Part 2 is incorrect. Since reactor power is initially at approximately 80%, a small runback will occur. Additionally, when a Reactor Coolant Pump is secured, the feedwater flow to each OTSG should not be identical. The screenshot from the Plant Processing Computer shows relatively identical flow (slightly off due to the Tcold circuitry). Therefore, Main Feedwater has NOT reratioed. Plausible if the candidate does not recognize that, with a Reactor Coolant Pump secured, the feedwater flow to each OTSG should not be identical. The above pictures describe what the screenshot would look like if ICS Feedwater control had and had not reratioed.

D. Correct.
Part 1 is correct. The Plant Processing Computer screenshot shows that the "B" RCS loop has a lower flow than the "A" RCS loop. The Reactor Coolant Pump physical layout is a s follows:
IAW OP-TM-MAP-F0301, RC Loop A Flow LO:

2.0 Causes
- RC-P-1A and/or RC-P-1B malfunction or trip
- Loss of a 230KV feeder single phase to an Auxiliary Transformer

IAW OP-TM-MAP-F0302, RC Loop B Flow LO:

2.0 Causes
- RC-P-1C and/or RC-P-1D malfunction or trip
- Loss of a 230KV feeder single phase to an Auxiliary Transformer

Additionally, IAW Operator Aid #107, The "B" Feedwater Loop will have less flow when there are two pumps operating in the "A" RCS loop and one pump operating in the "B" RCS Loop. Although the Tcold circuitry has adjusted feedwater flow slightly, it shows the "B" Feedwater Loop as having less flow (The Blue line if printed in color, the "X" if printed in Black and White). Therefore, it indicates that a Reactor Coolant Pump in the "B" RCS Loop has tripped. Since RC-P-1C and RC-P-1D are in the "B" RCS Loop, it must be one of those two pumps. IAW Operator Aid #107:

Operator Aid # 107

Power Level	A Loop 2 pumps Flow	A Loop 2 pumps Demand	B Loop 1 pump Flow	B Loop 1 pump Demand
75%	5.7	95%	2.4	41%
70%	5.3	88%	2.3	38%
65%	4.9	82%	2.1	35%
60%	4.5	76%	1.9	32%
55%	4.2	69%	1.8	30%
50%	3.8	63%	1.6	27%
45%	3.4	57%	1.5	24%
40%	3.0	50%	1.3	22%
35%	2.6	44%	1.1	19%
30%	2.3	38%	1.0	16%
25%	1.9	32%	0.8	14%
20%	1.5	25%	0.6	11%
15%	1.1	19%	0.5	8%
10%	0.8	13%	0.3	5%
5%	0.4	6%	0.2	3%

Power Level	A Loop 1 pump Flow	A Loop 1 pump Demand	B Loop 2 pumps Flow	B Loop 2 pumps Demand
75%	2.4	41%	5.7	95%
70%	2.3	38%	5.3	88%
65%	2.1	35%	4.9	82%
60%	1.9	32%	4.5	76%
55%	1.8	30%	4.2	69%
50%	1.6	27%	3.8	63%
45%	1.5	24%	3.4	57%
40%	1.3	22%	3.0	50%
35%	1.1	19%	2.6	44%
30%	1.0	16%	2.3	38%
25%	0.8	14%	1.9	32%
20%	0.6	11%	1.5	25%
15%	0.5	8%	1.1	19%
10%	0.3	5%	0.8	13%
5%	0.2	3%	0.4	6%

Part 2 is correct. A proper interpretation of the Plant Processing Computer screenshot will reveal that feedwater flows are relatively even as compared to the expected response. However, they should not be matched upon a loss of a Reactor Coolant Pump. IAW TQ-TM-104-621-C001, Integrated Control System:

- LOSS OF ONE REACTOR COOLANT PUMP
  1. INITIAL PLANT CONDITIONS
     - Reactor at 75% power with all Integrated Control System stations in automatic.
  2. OVERVIEW OF TRANSIENT
     - A loss of one reactor coolant pump at greater than 75% power would actuate a plant runback to 75% at a rate of 50% per minute. However, with power already at 75%, this will not occur. The loss of reactor coolant flow will be sensed through the ΔT control circuit, and will ratio feedwater flow accordingly. Ultimately feedwater will be ratioed 2.4:1, with approximately 2.0 x 10^6 lbm/hr., while flow in the non-affected loop rises to approximately 76 x 10^6 lbm/hr. feedwater flow to the affected OTSG.
  3. DETAILED DESCRIPTION

     A. Upon loss of the Reactor Coolant Pump, a reduction in Reactor Coolant Flow (RC Flow) occurs within the affected loop but rises in the non-affected loop. Within ten seconds RC Flow in the affected loop has lowered to approximately 32 x 10^6 lbm/hr, while flow in the non-affected loop rises to approximately 10^6 lbm/hr

     B. The differential flow developed between A and B loops will:
       1. Activate the ΔTc control circuit
       2. Swap Tavg control to the loop with the greatest flow
          - The ΔTc control has a circuit sensitive to RC Flow Differential, if greater than 10%. This circuit offers rate sensitive/promotional control. Thus, an immediate response to re-ratio feedwater demand proportionate with the rate of change to RC Flow occurs.

     C. An rise in feedwater flow to the non-affected loop is necessary to provide the heat removal required due to the rise in heat input (RC Flow).

     D. A drop in feedwater flow to the affected loop is necessary since the heat input (RC Flow) has been reduced.

     E. Within two minutes, the plant is stabilized with only fine tone requirements; these are provided by ΔTc integral/proportional action.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059	2.2.44
	Importance Rating	4.2	

Main Feedwater System: 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: RO Question # 54

Technical Reference(s): OP-TM-MAP-F0302, p 1, Rev 0A
TQ-TM-104-621-C001, p139, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 621-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 5

55.43

Comments:
The KA is matched because the operator must demonstrate the ability to interpret control room indications to verify the status and operation of the controls associated with the Mainfeedwater system, showing an understanding how initial operator actions (placing ICS in HAND) plant and system conditions.

The question is at the Comprehension/Analysis cognitive level because the operator must interpret Plant Processing Computer data and determine system status from the data.

What MUST be known:
1. What is proper Feedwater flow data on the Plant Processing Computer display screen upon a trip of the "C" Reactor Coolant Pump?
2. What is proper Feedwater reratio data on the Plant Processing Computer display screen upon a trip of a Reactor Coolant Pump?

55 **ID: 1147606** **Points: 1.00**

Plant Conditions:

- The Plant is in Hot Standby.
- A CRO is performing OP-TM-232-635, Draining the Reactor Building Sump.

Event:

- After draining the sump to the desired level, the CRO attempts to close WDL-V-535.
- WDL-V-535 fails to close from the Control Room.
- WDL-V-534 is still open.

Given the above information, what is the FIRST action **per Tech Specs,** if any, that is required concerning failure of this valve to close?

A. Manually close WDL-V-535.

B. Verify the operability of WDL-V-534.

C. No requirement during Hot Standby conditions.

D. Immediately ensure WDL-V-534 closed until WDL-V-535 is declared operable.

Answer: B

Answer Explanation

WDL-V-534 and WDL-V-535 are Reactor Building Sump Drain Isolation Valves. They are in series.

A. Incorrect.

Distractor is plausible because closing WDL-535 will satisfy the bases criteria to close one valve in the line within 48 hours. However, the question asks "what is the first thing that needs to be done", which would be verifying the operability of WDL-534. IAW Tech Spec 3.6.6 Bases:

1. When one of two CIVs in a line is inoperable, the capability to isolate the penetration using the other CIV in the line is promptly verified and at least one valve in the line must be closed within 48 hours or the plant must commence shut down.

B. Correct.

IAW Tech Spec 3.6:

T.S. 3.6.6: When CONTAINMENT INTEGRITY is required, if a CIV (other than a purge valve) is determined to be inoperable:

1. For lines isolable by two or more CIVs, the CIV(s) required to isolate the penetration shall be verified OPERABLE. If the inoperable valve is not restored within 48 hours, at least one CIV in the line will be closed or the reactor shall be brought to HOT SHUTDOWN within in the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours.

Additionally, IAW TS. 3.6.6 Bases:

1. When one of two CIVs in a line is inoperable, the capability to isolate the penetration using the other CIV in the line is promptly verified and at least one valve in the line must be closed within 48 hours or the plant must commence shut down.

C. Incorrect.

By definition, Containment Isolation is required during Hot Standby conditions. IAW Tech Spec 3.6.1:

3.6.1- Except as provided in 3.6.6, 3.6.8 and 3.6.12, CONTAINMENT INTEGRITY (section 1.7) shall be maintained whenever all three of the following conditions exist:

a) Reactor Coolant pressure is 300 psig or greater,
b) Reactor Coolant temperature is 200 degrees F or greater,
c) Nuclear fuel is in the core.

Additionally, IAW Tech Spec 1.2.4:

1.2.4 HOT STANDBY: The reactor is in the hot standby condition when all of the following conditions exist:

a. Tave is greater than 525F
b. The reactor is critical
c. Indicated neutron power on the power range channels is less than two percent of rated power

D. Incorrect.
The definition of CONTAINMENT INTEGRITY allows that "normally closed active CIVs (other than the purge valves) may be unisolated intermittently or manual control of the power-operated valves may be substituted for automatic control under administrative control.', thus allowing WDL-V-534 to be open under administrative control while WDL-V-535 is inoperable. Plausible if the candidate is not familiar with the conditions allowed for Containment Integrity to exist. IAW Tech Spec 1.7:

> 1.7 CONTAINMENT INTEGRITY: CONTAINMENT INTEGRITY exists when the following conditions are satisfied:
>
> > c. All active CIVs, including power-operated valves, check valves, and relief valves, are OPERABLE or locked closed. Normally closed active CIVs (other than the purge valves) may be unisolated intermittently or manual control of power-operated valves may be substituted for automatic control under administrative control.

Examination Outline Cross-reference:

Level	RO		SRO
Tier #	2		
Group #	1		
K/A #	103	2.2.37	
Importance Rating	3.6		

Containment System: Ability to determine operability and/or availability of safety related equipment.

Proposed Question: RO Question # 55

Technical Reference(s): Tech Spec 3.6, p 3-41, Amend 278
Tech Spec 3.6, p 3-41c, Amend 246

Proposed References to be provided to applicants during examination: None

Learning Objective: 240-GLO-14

Question Source: Bank # ID# 356587

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	7
	55.43	

Comments:
The KA is matched because the operator must demonstrate the ability to determine the operability of the Reactor Building Sump Drain Isolation Valves, which are Containment Isolation Valves, and are also safety related equipment.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall a Tech Spec action required to maintain the Containment System operable.

What MUST be known:
1. What is the Tech Spec action required to maintain the Containment System operable upon a Reactor Building Sump Drain Isolation Valve not operational?
2. What is the definition of Hot Standby?

56 **ID: 1147030** **Points: 1.00**

The Nuclear Services River Water System provides ___(1)___ and the Secondary Services River Water System provides ___(2)___.

A. (1) normal makeup to the Circulating Water System
(2) supplemental de-icing makeup to the Circulating Water System

B. (1) normal makeup to the Circulating Water System
(2) normal de-icing water to the Intake Screen and Pump House

C. (1) normal de-icing water to the Intake Screen and Pump House
(2) supplemental de-icing makeup to the Circulating Water System

D. (1) supplemental de-icing makeup to the Circulating Water System
(2) normal makeup to the Circulating Water System

Answer: D

Answer Explanation

A. Incorrect.

Part 1 is incorrect. Plausible if the candidate does not recognize that the Secondary Services River Water System provides for normal makeup to the Circulating Water System. This can be confusing since there are three systems discussed in this question that all interrelate with each other. IAW TQ-TM-511-C001, Circulating Water:

1. Circulating Water System Interfacing Systems
   i. Secondary Services River Water System (#532)
      1) A portion of the discharge from the Secondary Services heat exchangers flows through CW-V3 to provide make-up to the CW flume for evaporation losses from the NDCTs (CW-C1A/B).

Part 2 is incorrect. Plausible if the candidate does not recognize that the Nuclear Services River Water System provides for normal supplemental de-icing makeup to the Circulating Water System. This can be confusing since there are three systems discussed in this question that all interrelate with each other. IAW TQ-TM-511-C001, Circulating Water:

1. Circulating Water System Interfacing Systems
   h. Nuclear Services River Water System (#531)
      1) Nuclear Services River Water provides a supply of water for de-icing make up to the CW flume through motor operated valves NR-V-4A/B. This is used when the make-up capacity from the Secondary Services River Water System is exceeded.

B. Incorrect.

Part 1 is incorrect. Plausible if the candidate does not recognize that the Secondary Services River Water System provides for normal makeup to the Circulating Water System. This can be confusing since there are three systems discussed in this question that all interrelate with each other. IAW TQ-TM-511-C001, Circulating Water:

1. Circulating Water System Interfacing Systems
   i. Secondary Services River Water System (#532)
      1) A portion of the discharge from the Secondary Services heat exchangers flows through CW-V3 to provide make-up to the CW flume for evaporation losses from the NDCTs (CW-C1A/B).

Part 2 is incorrect. Plausible if the candidate does not recognize that it is the Circulating Water System that provides de-icing makeup to the Intake Screen and Pump House. This can be confusing since there are three systems discussed in this question that all interrelate with each other. IAW TQ-TM-511-C001, Circulating Water:

1. Circulating Water System Interfacing Systems
   a. Intake Screen and Pumphouse (#168)
      1) CW supplies the source of de-icing water to the ISPH.

C. Incorrect.
Part 1 is incorrect. Plausible if the candidate does not recognize that it is the Circulating Water System that provides de-icing makeup to the Intake Screen and Pump House. This can be confusing since there are three systems discussed in this question that all interrelate with each other. IAW TQ-TM-104-511-C001, Circulating Water:

1. Circulating Water System Interfacing Systems
   a. Intake Screen and Pumphouse (#168)
      1) CW supplies the source of de-icing water to the ISPH.

Part 2 is incorrect. Plausible if the candidate does not recognize that the Nuclear Services River Water System provides for normal supplemental de-icing makeup to the Circulating Water System. This can be confusing since there are three systems discussed in this question that all interrelate with each other. IAW TQ-TM-104-511-C001, Circulating Water:

1. Circulating Water System Interfacing Systems
   h. Nuclear Services River Water System (#531)
      1) Nuclear Services River Water provides a supply of water for de-icing make up to the CW flume through motor operated valves NR-V-4A/B. This is used when the make-up capacity from the Secondary Services River Water System is exceeded.

D. Correct.
Part 1 is correct. IAW TQ-TM-104-511-C001, Circulating Water:

1. Circulating Water System Interfacing Systems
   h. Nuclear Services River Water System (#531)
      1) Nuclear Services River Water provides a supply of water for de-icing make up to the CW flume through motor operated valves NR-V-4A/B. This is used when the make-up capacity from the Secondary Services River Water System is exceeded.

Part 2 is correct. IAW TQ-TM-104-511-C001, Circulating Water:

1. Circulating Water System Interfacing Systems
   i. Secondary Services River Water System (#532)
      1) A portion of the discharge from the Secondary Services heat exchangers flows through CW-V3 to provide make-up to the CW flume for evaporation losses from the NDCTs (CW-C1A/B).

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	075	K1.01	
	Importance Rating	4.2		

Knowledge of the physical connections and/or cause-effect relationship between the circulating water system and the following systems: SWS.

Proposed Question: RO Question # 56

Technical Reference(s): TQ-TM-104-511-C001, p 46-48, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 511-GLO-8

Question Source: Bank #

Modified Bank # ID# 355266

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

Comments:
The KA is matched because the operator must demonstrate the knowledge of the physical interrelationship between the Circulating Water System and the Nuclear Services River Water System as well as the Secondary Services River Water System.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall physical connections between three systems.

What MUST be known:
1. What is the physical interrelationship between the Circulating Water System and the Nuclear Services River Water System?
2. What is the physical interrelationship between the Circulating Water System and the Secondary Services River Water System?

Original Question:

Which of the following systems provides **supplemental** de-icing makeup to the Circulating Water flume?

A. Fire Service.

B. Nuclear River.

C. Secondary River.

D. Decay Heat River.

Answer: B

57 **ID: 1146881** **Points: 1.00**

Plant Conditions:

- The plant is in a Loss of Offsite Power event.
- Pressurizer heater control has been established IAW OP-TM-220-901, Emergency Power Supply for Pressurizer Heaters.

Sequence of Events:

- A Fire in the Relay Room occurs.
- Plant control is established at the RSD Panels.
- ESAS has spuriously actuated due to fire.
    - The SM has determined that ESAS unreliable and will not clear.
- Pressurizer heater level is 100 inches.

Given the above information, actions will be taken so that Pressurizer Heater Group ___(1)___ will be powered from ___(2)___ IAW OP-TM-220-901.

A. (1) 8
(2) 1P 480V Bus

B. (1) 8
(2) 1S 480V Bus

C. (1) 9
(2) 1P 480V Bus

D. (1) 9
(2) 1S 480V Bus

Answer: D

Answer Explanation

A. Incorrect.
Part 1 is incorrect. Plausible if the candidate believes that the Group 8 heaters can be energized while an ESAS signal is present. This choice is additionally plausible given the fact that Group 9 heaters were originally powered.

Part 2 is incorrect. This would be the correct power supply for Group 8 heaters but there is no 69 transfer switch on the 1P 480V Bus to establish emergency power to Group 8 heaters while an ESAS signal is present. Plausible since it is the emergency power supply for Group 8 heaters.

B. Incorrect.
Part 1 is incorrect. Plausible if the candidate believes that the Group 8 heaters can be energized while an ESAS signal is present. This choice is additionally plausible given the fact that Group 9 heaters were originally powered.

Part 2 is correct. The 1S 480V Bus is the only emergency power supply to the Pressurizer heaters that has a 69 transfer switch to establish emergency power to Pressurizer heaters. IAW OP-TM-220-901:

> 4.2.1. If ESAS is unreliable (e.g. fire in ESAS room, relay room or control room), then PLACE the "Press HTR Group 9" 69 transfer switch on the 1S 480 volt bus relay panel in EMERG to bypass the ES load shed interlock.

C. Incorrect.
Part 1 is correct. The Group 9 Pressurizer heaters are the only Pressurizer heaters associated with a 69 transfer switch to have the ability to receive power while an ESAS signal is present. IAW OP-TM-220-901, Emergency Power Supply for Pressurizer Heaters:

> 3.2.3. An ES signal will trip the pressurizer heaters off the bus but will not lock them out. The Group 9 ES signal may be bypassed using the "Press HTR Group 9" 69 transfer switch on the 1S 480 volt bus relay panel.

Part 2 is incorrect. This would be the correct power supply for Group 8 heaters but there is no 69 transfer switch on the 1P 480V Bus to establish emergency power to Group 8 heaters while an ESAS signal is present. Plausible since it is the emergency power supply for Group 8 heaters.

D. Correct.
Part 1 is correct. The Group 9 Pressurizer heaters are the only Pressurizer heaters associated with a 69 transfer switch to have the ability to receive power while an ESAS signal is present. IAW OP-TM-220-901, Emergency Power Supply for Pressurizer Heaters:

> 3.2.3. An ES signal will trip the pressurizer heaters off the bus but will not lock them out. The Group 9 ES signal may be bypassed using the "Press HTR Group 9" 69 transfer switch on the 1S 480 volt bus relay panel.

Part 2 is correct. The 1S 480V Bus is the only emergency power supply to the Pressurizer heaters that has a 69 transfer switch to establish emergency power to Pressurizer heaters. IAW OP-TM-220-901:

> 4.2.1. If ESAS is unreliable (e.g. fire in ESAS room, relay room or control room), then PLACE the "Press HTR Group 9" 69 transfer switch on the 1S 480 volt bus relay panel in EMERG to bypass the ES load shed interlock.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	011	K2.02	
	Importance Rating	4.2		

Pressurizer Level Control System: Knowledge of bus power supplies to the following: PZR heaters.

Proposed Question: RO Question # 57

Technical Reference(s): OP-TM-220-901, p 1,4, Rev 5

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO-4

Question Source: Bank #

Modified Bank # ID# 371774

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

Comments:

The KA is matched because the operator must demonstrate the knowledge of power supplies to PRessurizer Heaters (part of the Pressurizer Level Control System).

The question is at the Comprehension/Analysis cognitive level because the operator must interpret plant conditions and then determine the appropriate action to take.

What MUST be known:
1. What is the emergency power supply for Group 9 Pressurizer Heaters?
2. Which Emergency Pressurizer Heaters are available upon an ES signal?

Original Question:

Sequence of Events:

Loss of off-site power occurred.
Pressurizer heater control established by Group 8 heaters on 1P 480V Bus in accordance with OP-TM-220-901.
Fire in the Relay Room occurs.
Plant control is established at the RSD Panels.
ESAS has spuriously actuated and the SM has determined that it is unreliable and will not clear.

Which ONE of the following describes the operation of pressurizer heaters?

A. Group 8 heaters are no longer available and must be transferred back to the 1B Pressurizer Heater MCC.

B. ESAS actuation has locked in trips for both emergency pressurizer heater breakers and therefore both heater groups are unavailable.

C. Group 8 heaters are no longer available and Group 9 must be transferred to 1S Bus using the 69 bypass switch.

D. Group 8 heaters will remain available on 1P 480V Bus and coordination must be performed with the operator at the bus to energize heaters as necessary for pressure control.

Answer: C

58 **ID: 1146887** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- A Reactor Trip occurs.
- CRD Group 1 Rod 1 and Rod 2 are both stuck in the full withdrawn position.
- Pressurizer level control valve, MU-V-17, is closed in HAND.
- Pressurizer level is above the current automatic control setpoint.
- RCS Makeup Flow is 0 gpm.
- HPI Valves, MU-V-16A-D, are all closed.
- RCS letdown flow is 0 gpm.
- Total RCP Seal Injection Flow is 32 gpm, steady.
- Total RCP Seal #1 Leak-off flow is 20 gpm, steady.

Given the above conditions, identify the MINIMUM actions required to initiate emergency boration of the RCS.

A. Open MU-V-14A.

B. Open MU-V-14A and restore RCS letdown flow to establish proper total injection.

C. Open MU-V-14A and MU-V-14B.

D. Open MU-V-14A and MU-V-14B and restore RCS letdown flow to establish proper total injection.

Answer: B

Answer Explanation

A. Incorrect. Plausible if the candidate does not recognize that total injection is less than 50 gpm. Using this train of thought, Emergency Boration would be performed as follows IAW Rule 5, Emergency Boration:

3. Perform one of the following:
   - OPEN MU-V-14A
   - OPEN MU-V-14B
   - PERFORM Guide 1 "Emergency Boration Backup Methods".

B. Correct.

IAW OP-TM-EOP-010, Emergency Procedure Rules Guides and Graphs, Rule 5, Emergency Boration:

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:

3. Perform one of the following:
   - OPEN MU-V-14A
   - OPEN MU-V-14B
   - PERFORM Guide 1 "Emergency Boration Backup Methods".

4. VERIFY Total Injection (MU, SI and HPI) > 50 gpm.
RNO:
   1. INITIATE OP-TM-211-950, "Restoration of Letdown Flow".
   2. INITIATE OP-TM-211-441, "Increased Letdown Flowrates".

Additionally, IAW OP-TM-EOP-0101, Emergency Procedure Rules Guides and Graphs Basis Document:

Step 4 The step intent is to establish boration rate to satisfy Emergency Boration requirements. 50 gpm is established in Engineering calculation, as the minimum rate required to borate the Reactor Coolant system when using either the BWST or MUT boration flowpaths.

C. Incorrect.

Plausible if the candidate does not recognize that total injection is less than 50 gpm and believes that both MU-V-14 A and MU-V-14B must be opened. Using this train of thought, Emergency Boration would be performed as follows IAW Rule 5, Emergency Boration:

3. Perform one of the following:
   - OPEN MU-V-14A
   - OPEN MU-V-14B
   - PERFORM Guide 1 "Emergency Boration Backup Methods".

D. Incorrect.
Plausible if the candidate does recognize that total injection is less than 50 gpm but believes that both MU-V-14A and MU-V-14B must be opened. However, only one MU-V-14 valve needs to be open to accomplish the task. IAW Rule 5, Emergency Boration:

3. Perform one of the following:
    - OPEN MU-V-14A
    - OPEN MU-V-14B
    - PERFORM Guide 1 "Emergency Boration Backup Methods".

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	001	K3.01	
	Importance Rating	2.9		

Knowledge of the effect that a loss or malfunction of the CRDS will have on the following: CVCS.

Proposed Question: RO Question # 58

Technical Reference(s): OP-TM-EOP-010, p 10-11, Rev 17
OP-TM-EOP-0101, p 25, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: EOPR5-PCO-4

Question Source: Bank #

Modified Bank # ID# 363947

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

Comments:
The KA is matched because the operator must demonstrate the knowledge of how a malfunction of the Digital Control Rod Drive System will have on the Makeup System (CVCS).

The question is at the Comprehension/Analysis cognitive level because the operator must perform a mathematical calculation to determine if a procedural step is satisfied or not.

What MUST be known:
1. What is the action to take IAW emergency boration upon stuck rods?
2. What is the Rule 5 actions to take if total injection flow is less than 50gpm?
3. How to calculate total injection.flow.

Original Question:

Sequence of Events:

Reactor is tripped from 100% power.
CRD Group 1 Rod 1 and Rod 2 are stuck (fully withdrawn).
Pressurizer level control valve, MU-V-17, is closed in automatic.
Pressurizer level is above the current automatic control setpoint.
RCS Makeup Flow is 0 gpm.
HPI Valves, MU-V-16A-D, are all closed.
RCS letdown flow is 0 gpm.
Total RCP Seal Injection Flow is 38 gpm, steady.
Total RCP Seal #1 Leak-off flow is 10 gpm, steady.

Based on the above conditions, identify ONE of the following that describes MINIMUM actions required to initiate emergency boration of the RCS.

A. Open MU-V-14A and manually open MU-V-17 to raise RCS makeup flow to 13 gpm.

B. Open MU-V-14A and restore RCS letdown flow to raise RCS makeup flow to 13 gpm.

C. Open MU-V-14B and manually open MU-V-17 to raise RCS makeup flow to 23 gpm, ONLY.

D. Open MU-V-14A **AND** MU-V-14B and restore RCS letdown flow to raise RCS makeup flow to 23 gpm.

Answer: B

59 **ID: 1142257** **Points: 1.00**

Plant Conditions:

- An RC Evaporator Campaign is in progress.
- Currently filling the "B" Waste Evaporator Condensate Storage Tank, WDL-T-11B.
- WDL-T-11B was put on recirculation when the tank reached 8 feet at 11:00 hours with WDL-P-14B.
- WDL-P-14A is tagged out of service.

Sequence of Events:

- The tank was filled and isolated at 13:00 and the CRS completed the required sections of the Liquid Permit Pre-Release Report.
- After the paper work was submitted, the AO noted that a small pump packing leak on WDL-P-14B and has requested to secure the pump to minimize the volume being dumped to the auxiliary building sump via the floor drain.
- SM has determined the Leak on WDL-P-14B is NOT significant and has decided to release WDL-T-11B as scheduled.

Given the above information, the earliest that Chemistry can sample WDL-T-11B is ___(1)___, and the recirculation pump, WDL-P-14B, ___(2)___.

A. (1) 19:00
(2) may be secured after chemistry obtains sample

B. (1) 19:00
(2) must remain in service until the release is completed

C. (1) 21:00
(2) may be secured after chemistry obtains sample

D. (1) 21:00
(2) must remain in service until the release is completed

Answer: B

Answer Explanation

A. Incorrect.

Part 1 is correct. Although WDL-T-11B must recirculate for a minimum of eight hours, this includes the time that the tank was being filled. IAW CY-TM-170-2001, Releasing Radioactive Liquid Waste:

> NOTE: The time the tank is on recirculation while filling the tank may be considered in the 8-hour requirement.

4.1.1 RECIRCULATE these tanks a minimum of eight (8) hours prior to being sampled.
- This shall be done to ensure proper mixing in the tanks and to ensure that a representative sample of the tank can be drawn for release calculations
- Leave tank on recirculation until release is initiated

Therefore the recirculation started at 11:00 and with a minimum of 8 hours recirculation time the earliest chemistry can sample is 19:00, and tank must remain on recirc until release is initiated.

Part 2 is incorrect. CY-TM-170-2001 directs the use of OP-TM-223-554, Liquid Release of "B" WECST with WDL-P-14B, to release the contents of the tank. OP-TM-223-554 has WDL-P-14B running for the entirety of the procedure until the Return to Normal section, except if the release is stopped for a temporary period of time. IAW OP-TM-223-554:

3.3 Prerequisites

3.3.2 VERIFY WDL-T-11B is on recirculation IAW OP-TM-232-419, "Recirculation of "B" WECST with WDL-P-14B."

5.0 RETURN TO NORMAL

5.1 When WDL-T-11B level lowers to 0.5 feet, or the Liquid Release has been automatically or manually terminated, then CONTINUE.

5.2 SECURE WDL-T-11B recirculation IAW Section 5.0 of OP-TM-232-419.

B. Correct.

Part 1 is correct. Although WDL-T-11B must recirculate for a minimum of eight hours, this includes the time that the tank was being filled. IAW CY-TM-170-2001, Releasing Radioactive Liquid Waste:

> NOTE: The time the tank is on recirculation while filling the tank may be considered in the 8-hour requirement.

4.1.1 RECIRCULATE these tanks a minimum of eight (8) hours prior to being sampled.
- This shall be done to ensure proper mixing in the tanks and to ensure that a representative sample of the tank can be drawn for release calculations
- Leave tank on recirculation until release is initiated

Therefore the recirculation started at 11:00 and with a minimum of 8 hours recirculation time the earliest chemistry can sample is 19:00, and tank must remain on recirc until release is initiated.

Part 2 is correct. CY-TM-170-2001 directs the use of OP-TM-223-554, Liquid Release of "B" WECST with WDL-P-14B, to release the contents of the tank. OP-TM-223-554 has WDL-P-14B running for the entirety of the procedure until the Return to Normal section, except if the release is stopped for a temporary period of time. IAW OP-TM-223-554:

3.3 Prerequisites

3.3.2 VERIFY WDL-T-11B is on recirculation IAW OP-TM-232-419, "Recirculation of "B" WECST with WDL-P-14B."

5.0 RETURN TO NORMAL

5.1 When WDL-T-11B level lowers to 0.5 feet, or the Liquid Release has been automatically or manually terminated, then CONTINUE.

5.2 SECURE WDL-T-11B recirculation IAW Section 5.0 of OP-TM-232-419.

C. Incorrect.

Part 1 is incorrect. Plausible if the candidate does not include the filling time as part of the eight hours. Although WDL-T-11B must recirculate for a minimum of eight hours, this includes the time that the tank was being filled. IAW CY-TM-170-2001, Releasing Radioactive Liquid Waste:

NOTE: The time the tank is on recirculation while filling the tank may be considered in the 8-hour requirement.

4.1.1 RECIRCULATE these tanks a minimum of eight (8) hours prior to being sampled.

- This shall be done to ensure proper mixing in the tanks and to ensure that a representative sample of the tank can be drawn for release calculations
- Leave tank on recirculation until release is initiated

Therefore the recirculation started at 11:00 and with a minimum of 8 hours recirculation time the earliest chemistry can sample is 19:00, and tank must remain on recirc until release is initiated.

Part 2 is incorrect. Plausible if the candidate does not include the filling time as part of the eight hours.CY-TM-170-2001 directs the use of OP-TM-223-554, Liquid Release of "B" WECST with WDL-P-14B, to release the contents of the tank. OP-TM-223-554 has WDL-P-14B running for the entirety of the procedure until the Return to Normal section, except if the release is stopped for a temporary period of time. IAW OP-TM-223-554:

3.3 Prerequisites

3.3.2 VERIFY WDL-T-11B is on recirculation IAW OP-TM-232-419, "Recirculation of "B" WECST with WDL-P-14B."

5.0 RETURN TO NORMAL

5.1 When WDL-T-11B level lowers to 0.5 feet, or the Liquid Release has been automatically or manually terminated, then CONTINUE.

5.2 SECURE WDL-T-11B recirculation IAW Section 5.0 of OP-TM-232-419.

D. Incorrect.

Part 1 is incorrect. Plausible if the candidate does not include the filling time as part of the eight hours. Although WDL-T-11B must recirculate for a minimum of eight hours, this includes the time that the tank was being filled. IAW CY-TM-170-2001, Releasing Radioactive Liquid Waste:

> NOTE: The time the tank is on recirculation while filling the tank may be considered in the 8-hour requirement.

4.1.1 RECIRCULATE these tanks a minimum of eight (8) hours prior to being sampled.
- This shall be done to ensure proper mixing in the tanks and to ensure that a representative sample of the tank can be drawn for release calculations
- Leave tank on recirculation until release is initiated

Therefore the recirculation started at 11:00 and with a minimum of 8 hours recirculation time the earliest chemistry can sample is 19:00, and tank must remain on recirc until release is initiated.

Part 2 is correct. CY-TM-170-2001 directs the use of OP-TM-223-554, Liquid Release of "B" WECST with WDL-P-14B, to release the contents of the tank. OP-TM-223-554 has WDL-P-14B running for the entirety of the procedure until the Return to Normal section, except if the release is stopped for a temporary period of time. IAW OP-TM-223-554:

3.3 Prerequisites
3.3.2 VERIFY WDL-T-11B is on recirculation IAW OP-TM-232-419, "Recirculation of "B" WECST with WDL-P-14B."

5.0 RETURN TO NORMAL
5.1 When WDL-T-11B level lowers to 0.5 feet, or the Liquid Release has been automatically or manually terminated, then CONTINUE.
5.2 SECURE WDL-T-11B recirculation IAW Section 5.0 of OP-TM-232-419.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	068	K4.01	
	Importance Rating	3.4		

Liquid Radwaste System: Knowledge of design feature(s) and/or interlock(s) which provide for the following: Safety and environmental precautions for handling hot,acidic, and radioactive liquids.

Proposed Question: RO Question # 59

Technical Reference(s): CY-TM-170-2001, p 2,7, Rev 0
OP-TM-232-554, p 1,7, Rev 5

Proposed References to be provided to applicants during examination: None

Learning Objective:	RPT-APCO-1		
Question Source:	Bank #	ID# 773335	
	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		

Comments:
The KA is matched because the operator must demonstrate the knowledge of procedure adherence that shows the design feature that provides safety and environmental precautions for handling radioactive liquids.

The question is at the Comprehension/Analysis cognitive level because the operator must perform a mathematical calculation to answer the question.

What MUST be known:
1. What events may be considered when determining a recirculation of a Waste Evaporator Condensate Storage Tank for a minimum of eight hours?
2. What is the procedural requirement for WDL-P-14B to remain running?

60 **ID: 1146916** **Points: 1.00**

Plant Conditions:

- The Plant is operating at 100% reactor power.
- The following ICS stations are in HAND:
  - SG/Reactor Demand
  - SG A/B Load Ratio (Tc)
  - "A" Feedwater Demand
  - "B" Feedwater Demand

Event:

- The operator adjusts feedwater flow such that a lowering of Main Feedwater flow to the value equivalent to 60% power occurs.

Given the above information and assuming no operator actions, a ___(1)___ crosslimit signal will exist and ICS will ___(2)___ reactor power.

A. (1) Reactor to Feedwater
(2) raise

B. (1) Reactor to Feedwater
(2) reduce

C. (1) Feedwater to Reactor
(2) raise

D. (1) Feedwater to Reactor
(2) reduce

Answer: D

Answer Explanation

A. Incorrect.

Part 1 is incorrect. Plausible if the candidate confuses the direction of the crosslimit. However, Reactor to Feedwater crosslimits compare actual reactor power to demanded reactor power. IAW TQ-TM-104-621-C001, Integrated Control System:

a. Crosslimits – Reactor to Feedwater

- Positive neutron errors indicate a deficiency of power with respect to the demand and should put a limit on feedwater demand. This Neutron error is subtracted from the feedwater demand, thus a positive error greater than 5 percent will runback feedwater demand proportionally to the amount of error in excess of five percent.
- Negative errors, indicating excess neutron power, will run up feedwater proportionally to the amount of error less than (more negative) 5 percent. If either limiting action on feedwater does occur, it will transfer the ICS to the tracking mode as defined in the Unit Load Demand Subsystem. The occurrence of this limiting action indicates that the neutron power is not able to satisfy its demand. Therefore, the control will maintain the rest of the unit (the turbine and the feedwater flow) in the proper relation to neutron power to maintain stable operation. For such a condition the control is essentially in a turbine following mode in that theload can be no greater or less than 5 percent of the neutron power.

Part 2 is incorrect. The Integrated Control System (ICS) is a complex system which will ultimately lower reactor power to match feed flow. Plausible if the candidate is not familiar with the ICS circuitry.

B. Incorrect.

Part 1 is incorrect. Plausible if the candidate confuses the direction of the crosslimit. However, Reactor to Feedwater crosslimits compare actual reactor power to demanded reactor power. IAW TQ-TM-104-621-C001, Integrated Control System:

a. Crosslimits – Reactor to Feedwater

- Positive neutron errors indicate a deficiency of power with respect to the demand and should put a limit on feedwater demand. This Neutron error is subtracted from the feedwater demand, thus a positive error greater than 5 percent will runback feedwater demand proportionally to the amount of error in excess of five percent.
- Negative errors, indicating excess neutron power, will run up feedwater proportionally to the amount of error less than (more negative) 5 percent. If either limiting action on feedwater does occur, it will transfer the ICS to the tracking mode as defined in the Unit Load Demand Subsystem. The occurrence of this limiting action indicates that the neutron power is not able to satisfy its demand. Therefore, the control will maintain the rest of the unit (the turbine and the feedwater flow) in the proper relation to neutron power to maintain stable operation. For such a condition the control is essentially in a turbine following mode in that theload can be no greater or less than 5 percent of the neutron power.

Part 2 is correct. The Integrated Control System (ICS) is a complex system which will ultimately lower reactor power to match feed flow. IAW TQ-TM-104-621-C001, Integrated Control System:

- LOWERING IN LOAD WITH BOTH FEEDWATER LOOP DEMANDS IN MANUAL
  1. INITIAL PLANT CONDITIONS
     A. Reactor at 100% power, all ICS stations in automatic except feedwater loop demands which are in manual. ICS in track.

  4. INCORRECT OPERATOR ACTION
     A. The operator rapidly lowers feedwater flow to the value equivalent to 60% power. The plant response is much the same as previously discussed. However, the effect on reactor coolant temperature is much more severe. There is a large mismatch between the heat produced and the heat removal. Tavg rises and reactor coolant pressure rises rapidly. The Tavg error signal will modify the reactor demand to start lowering reactor power.
     B. With feedwater flow 5% less than the demanded flow (demand is established by generated megawatts when in track), the feedwater error signal will modify the reactor demand signal (crosslimits). This feedwater crosslimit signal to the reactor demand will act to lower reactor power. This helps to slow the rate of temperature rise in addition to the Tavg error modification to the reactor demand.

C. Incorrect.
Part 1 is correct. IAW TQ-TM-104-621-C001, Integrated Control System:

b. Crosslimits – Feedwater to Reactor
- o The reactor demand assumes the reactor and steam generator will respond in parallel. Any large feedwater errors indicate a failure of the Feedwater Subsystem to respond, thus necessitating that the system be switched from feed forward to actual conditions. As seen the previous section on Reactor to Feedwater Crosslimits, a complete set of crosslimits would either raise or lower the demand relative to the feed-forward signal. The reactor demand can only be lowered by cross limits, thus avoiding any reactor run ups.
- o The reactor demand signal is modified by a feedwater error signal in excess of 5 percent. This feedwater error signal is developed by comparing the temperature compensated feedwater demand signal to the total feedwater flow.
- o If feedwater demand exceeds feedwater flow by 5%, (Demand>Actual) the amount in excess of 5% reduces Neutron Demand to keep Reactor Power within 5% of Feedwater Flow.
- o This crosslimit also place the ICS into a Tracking Condition.

Part 2 is incorrect. The Integrated Control System (ICS) is a complex system which will ultimately lower reactor power to match feed flow. Plausible if the candidate is not familiar with the ICS circuitry.

D. Correct.
Part 1 is correct. IAW TQ-TM-104-621-C001, Integrated Control System:

b. Crosslimits – Feedwater to Reactor

- The reactor demand assumes the reactor and steam generator will respond in parallel. Any large feedwater errors indicate a failure of the Feedwater Subsystem to respond, thus necessitating that the system be switched from feed forward to actual conditions. As seen the previous section on Reactor to Feedwater Crosslimits, a complete set of crosslimits would either raise or lower the demand relative to the feed-forward signal. The reactor demand can only be lowered by cross limits, thus avoiding any reactor run ups.
- The reactor demand signal is modified by a feedwater error signal in excess of 5 percent. This feedwater error signal is developed by comparing the temperature compensated feedwater demand signal to the total feedwater flow.
- If feedwater demand exceeds feedwater flow by 5%, (Demand>Actual) the amount in excess of 5% reduces Neutron Demand to keep Reactor Power within 5% of Feedwater Flow.
- This crosslimit also place the ICS into a Tracking Condition.

Part 2 is correct. The Integrated Control System (ICS) is a complex system which will ultimately lower reactor power to match feed flow. IAW TQ-TM-104-621-C001, Integrated Control System:

- LOWERING IN LOAD WITH BOTH FEEDWATER LOOP DEMANDS IN MANUAL

  1. INITIAL PLANT CONDITIONS

     A. Reactor at 100% power, all ICS stations in automatic except feedwater loop demands which are in manual. ICS in track.

  4. INCORRECT OPERATOR ACTION

     A. The operator rapidly lowers feedwater flow to the value equivalent to 60% power. The plant response is much the same as previously discussed. However, the effect on reactor coolant temperature is much more severe. There is a large mismatch between the heat produced and the heat removal. Tavg rises and reactor coolant pressure rises rapidly. The Tavg error signal will modify the reactor demand to start lowering reactor power.

     B. With feedwater flow 5% less than the demanded flow (demand is established by generated megawatts when in track), the feedwater error signal will modify the reactor demand signal (crosslimits). This feedwater crosslimit signal to the reactor demand will act to lower reactor power. This helps to slow the rate of temperature rise in addition to the Tavg error modification to the reactor demand.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	035	K5.01	
	Importance Rating	3.4		

Knowledge of operational implications of the following concepts as they apply to the S/GS: Effect of secondary parameters, pressure, and temperature on reactivity.

Proposed Question: RO Question # 60

Technical Reference(s): TQ-TM-104-621-C001, p 57-58,127, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 621-GLO-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 6

55.43

Comments:
The KA is matched because the operator must demonstrate the knowledge of the effect of feedwater flow to the OTSG's (secondary parameters) on reactivity.

The question is at the Comprehension/Analysis cognitive level because the operator must analyze the conditions given and determine the cause-effect from that analysis.

What MUST be known:
1. What is the relationship between feedwater flow and reactor power during a Feedwater to Reactor Crosslimit?
2. How does ICS react to a Feedwater to Reactor Crosslimit with regards to reactor power?

61 **ID: 1142241** **Points: 1.00**

Plant Conditions:

- A refueling outage has commenced.
- The core is in the process of being off-loaded.

Event:

- RM-G-9, Fuel Handling Building Fuel Handling Bridge Radiation Monitor, loses power.

Given the above information, which ONE of the following describes the correct action IAW 1505-1, Fuel and Control Component Shuffles?

A. Spent Fuel Pool movements may continue as long as RM-A-4, FHB Vent Radiation Monitor, remains operable.

B. Cease all spent fuel pool fuel movement until proper portable survey instruments are provided to monitor radiation levels.

C. Spent Fuel Pool movements may continue as long as AH-E-10, FHB Supply Fan, is secured and the FHB isolation dampers are closed.

D. Cease all spent fuel pool fuel movement until FHB isolation dampers have been verified open and RM-A-4, FHB Vent Radiation Monitor, has been verified operable.

Answer: B

Answer Explanation

A. Incorrect.

RM-G-9 is specifically required IAW 1505-1 to comply with FSAR requirement. Plausible because RM-A-4 monitors FHB exhaust flow for radiation and would provide warning of off-normal airborne conditions. IAW 1505-1, Fuel and Control Component Shuffles:

> 5.1.2 If any RM-G unit becomes inoperable, portable survey instrumentation must be used having appropriate range and sensitivity to fully protect individuals involved in fuel handling operations until permanent instrumentation is returned to service.

B. Correct.

RM-G-9 is specifically required IAW 1505-1 to comply with FSAR requirement. Plausible because RM-A-4 monitors FHB exhaust flow for radiation and would provide warning of off-normal airborne conditions. IAW 1505-1, Fuel and Control Component Shuffles:

> 5.1.2 If any RM-G unit becomes inoperable, portable survey instrumentation must be used having appropriate range and sensitivity to fully protect individuals involved in fuel handling operations until permanent instrumentation is returned to service.

C. Incorrect.

Plausible if it is believed that the interlock function of RM-G-9 being satisfied would be satisfactory IAW 1505-1:

> **NOTE:** The interlock from RM-G9 and RM-A4 to isolate the spent fuel pool area from the FHB Normal Ventilation System must be functional for fuel handling ventilation to be operable.

However, portable survey instrumentation must also be used. IAW 1505-1, Fuel and Control Component Shuffles:

> 5.1.2 If any RM-G unit becomes inoperable, portable survey instrumentation must be used having appropriate range and sensitivity to fully protect individuals involved in fuel handling operations until permanent instrumentation is returned to service.

D. Incorrect.

Plausible if it was believed that ensuring RM-A-4 could provide replacement interlock function for RM-G-9. IAW 1505-1:

> **NOTE:** The interlock from RM-G9 and RM-A4 to isolate the spent fuel pool area from the FHB Normal Ventilation System must be functional for fuel handling ventilation to be operable.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	034	K6.02	
	Importance Rating	2.6		

Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling System: Radiation monitoring systems

Proposed Question: RO Question # 61

Technical Reference(s): 1505-1, p. 7, Rev 58

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-8

Question Source: Bank # ID# 862271

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

Comments:
The KA is matched because the operator must demonstrate the knowledge of the effect of a loss of Radiation Monitoring Equipment has on the Fuel Handling System.

The question is at the Memory/FundamentalKnowledge cognitive level because the operator must recall the procedural action to take upon a loss of a radiation monitor associated with Fuel Handling.

What MUST be known:
1. What is the procedural required action to take upon a loss of RM-G-9 during fuel handling?

62 **ID: 1142251** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- A Reactor Trip occurs due to a loss of a Reactor Coolant Pump without a proper ICS runback.

Given the above information, the Turbine Bypass Valves open to control Turbine Header Pressure at ___(1)___ and the Atmospheric Dump Valves will ___(2)___.

A. (1) 960 psig
(2) open fully at 1040 psig

B. (1) 960 psig
(2) begin to open at 1026 psig and be fully open at 1052 psig

C. (1) 1010 psig
(2) open fully at 1040 psig

D. (1) 1010 psig
(2) begin to open at 1026 psig and be fully open at 1052 psig

Answer: D

Answer Explanation

A. Incorrect.

Part 1 is incorrect. This is plausible because the Turbine Bypass valves will open at 960 psig while at 100% power, and the other two setpoints are associated with system operation. According to TQ-TM-104-411-C001:

- During normal operation the setting is 885 psig, corresponding to 47.5% on the dial. Setpoint biases applied by the ICS are 10 psig, 75 psig and 125 psig, resulting in normal automatic adjustable control setpoints of 895 psig, 960 psig and 1010 psig.
- 960 psig Automatic Control Setpoint
  - o In effect, when ICS ULD > 15%, this setpoint is intended to prevent inadvertent Turbine Bypass Valve operation during normal plant transients.

Part 2 is incorrect. This is plausible if the candidate does not recall the fact that the Atmospheric Dump Valves only transfer to the setpoint of 1040 psig upon a loss of condenser vacuum or on a loss of all Circulating water Pumps. Nothing in the stem would lead the candidate to believe that either of those conditions exist. Until either of those conditions do occur, the Atmospheric Dump Valves will control between 1026 psig and 1052 psig. IAW TQ-TM-104-411-C001, Main Steam:

- 1040 psig Automatic (Fixed) Control Setpoint
  - o This circuit provides an independent high-pressure relief that will open proportional to OTSG pressure.
  - o Transfers to MS-V-4A/4B controls on low vacuum or loss of CW Pumps.

B. Incorrect.

Part 1 is incorrect. This is plausible because the Turbine Bypass valves will open at 960 psig while at 100% power, and the other two setpoints are associated with system operation. IAW TQ-TM-104-411-C001, Main Steam:

- During normal operation the setting is 885 psig, corresponding to 47.5% on the dial. Setpoint biases applied by the ICS are 10 psig, 75 psig and 125 psig, resulting in normal automatic adjustable control setpoints of 895 psig, 960 psig and 1010 psig.
- 960 psig Automatic Control Setpoint
  - o In effect, when ICS ULD > 15%, this setpoint is intended to prevent inadvertent Turbine Bypass Valve operation during normal plant transients.

Part 2 is correct. IAW TQ-TM-104-411-C001, Main Steam:

- 1026-1052 psig Automatic (Fixed) Proportional Control Setpoint (ADV only)
  - o ADVs will modulate open proportionally to control steam flow as a function of pressure in the range 1026 to 1052 psig.
  - o This circuit is never used by the TBVs, and is blocked (not used) by the ADVs if Main Condenser vacuum is <23" Hg or less than 2 CW Pumps are running.

C. Incorrect.
Part 1 is correct. IAW TQ-TM-104-411-C001, Main Steam:

- TBV/ADV Pressure Control Schemes (MS-V-3A/B/C/D/E/F and MS-V-4A/B)
  - Automatic control schemes for the TBVs and ADVs use OTSG pressure signals.
    - Adjustable Automatic Control Setpoints
      - During normal operation the setting is 885 psig, corresponding to 47.5% on the dial. Setpoint biases applied by the ICS are 10, 75 and 125, resulting in normal automatic adjustable control setpoints of 895, 960 and 1010 psig.
    - 1010 psig Automatic Control Setpoint
      - This setpoint is to prevent excessive Pressurizer Level decrease on reactor trip.
      - Raising the OTSG pressure setpoint on a reactor trip limits RCS Cooldown and shrink, since this sets post trip OTSG saturated steam temperature at 555°F.

Part 2 is incorrect. This is plausible if the candidate does not recall the fact that the Atmospheric Dump Valves only transfer to the setpoint of 1040 psig upon a loss of condenser vacuum or on a loss of all Circulating water Pumps. Nothing in the stem would lead the candidate to believe that either of those conditions exist. Until either of those conditions do occur, the Atmospheric Dump Valves will control between 1026 psig and 1052 psig. IAW TQ-TM-104-411-C001, Main Steam:

- 1040 psig Automatic (Fixed) Control Setpoint
  - This circuit provides an independent high-pressure relief that will open proportional to OTSG pressure.
  - Transfers to MS-V-4A/4B controls on low vacuum or loss of CW Pumps.

D. Correct.
Part 1 is correct. IAW TQ-TM-104-411-C001, Main Steam:

- TBV/ADV Pressure Control Schemes (MS-V-3A/B/C/D/E/F and MS-V-4A/B)
  - Automatic control schemes for the TBVs and ADVs use OTSG pressure signals.
    - Adjustable Automatic Control Setpoints
      - During normal operation the setting is 885 psig, corresponding to 47.5% on the dial. Setpoint biases applied by the ICS are 10, 75 and 125, resulting in normal automatic adjustable control setpoints of 895, 960 and 1010 psig.
    - 1010 psig Automatic Control Setpoint
      - This setpoint is to prevent excessive Pressurizer Level decrease on reactor trip.
      - Raising the OTSG pressure setpoint on a reactor trip limits RCS Cooldown and shrink, since this sets post trip OTSG saturated steam temperature at 555°F.

Part 2 is correct. IAW TQ-TM-104-411-C001, Main Steam:

- 1026-1052 psig Automatic (Fixed) Proportional Control Setpoint (ADV only)
  - ADVs will modulate open proportionally to control steam flow as a function of pressure in the range 1026 to 1052 psig.
  - This circuit is never used by the TBVs, and is blocked (not used) by the ADVs if Main Condenser vacuum is <23" Hg or less than 2 CW Pumps are running.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	041	A1.02	
	Importance Rating	3.1		

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam pressure

Proposed Question: RO Question # 62

Technical Reference(s): TQ-TM-104-411-C001, p. 29-30, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-5

Question Source: Bank # ID# 719620

Modified Bank #

New

Question History: Last NRC Exam: 08-01 Retest

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

The KA is matched because the operator must demonstrate ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including steam pressure (i.e. pressures at which SDS valves open).

The question is at the Memory cognitive level because the operator must recall bits of information (i.e. what are the setpoints that the valves control at during a reactor trip).

What MUST be known:
1. What are the setpoints that the Turbine Bypass Valves control at during a reactor trip?
2. What are the setpoints that the Atmospheric Valves control at during a reactor trip?

63 **ID: 1147007** **Points: 1.00**

Note:

- FS-P-1: Circulating Water Diesel Driven Fire Pump.
- FS-P-2: Screen House Motor Driven Fire Pump.
- FS-P-3: Unit 1 River Diesel Fire Pump.

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- A Fire occurs on site.
- No safe shutdown equipment is affected.
- The Control Room operators are able to maintain power operation.
- The fire brigade has been dispatched.
- FS-PI-371, Fire Service Loop Pressure Indicator has failed in the Control Room.
- Local Fire System pressure has been reported as 74 psig and slowly lowering.
- No Fire Service Water Pumps are running.

Given the above information:
(1) Which Fire Service Pumps should have started automatically, and
(2) What action is required IAW OP-TM-AOP-001, Fire?

A. (1) FS-P-1 and FS-P-2.
(2) Initiate OP-TM-EOP-001, Reactor Trip.

B. (1) FS-P-1 and FS-P-2.
(2) Start fire pumps as necessary to raise FS pressure > 90 psig.

C. (1) FS-P-2 and FS-P-3.
(2) Initiate OP-TM-EOP-001, Reactor Trip.

D. (1) FS-P-2 and FS-P-3.
(2) Start fire pumps as necessary to raise FS pressure > 90 psig.

Answer: D

Answer Explanation

A. Incorrect.
Part 1 is incorrect. Plausible if the candidate believes that Fire Service Pumps start in numerical order. However, FS-P-2 should have started at 90 psig and FS-P-3 should have started at 80 psig. FS-P-1 will not have started until Fire Service System pressure reaches 70 psig.

Part 2 is incorrect. Plausible if the candidate believes that OP-TM-EOP-001 should be initiated based on Fire System pressure. However, the procedural requirement to initiate OP-TM-EOP-001 is based on criteria that are not met, as stated in the question stem. IAW OP-TM-AOP-001, Fire:

3.10 IAAT the fire causes any of the following conditions to exist:
- Serious damage to safe shutdown equipment
- Degraded ability to maintain power operation

then INITIATE EOP-001.

B. Incorrect.
Part 1 is incorrect. Plausible if the candidate believes that Fire Service Pumps start in numerical order. However, FS-P-2 should have started at 90 psig and FS-P-3 should have started at 80 psig. FS-P-1 will not have started until Fire Service System pressure reaches 70 psig.

Part 2 is correct. IAW OP-TM-AOP-001, Fire:

3.7 VERIFY FS pressure >90 psig on FS-PI-371.

RNO: START the following fire pumps as necessary to raise FS pressure > 90 psig:
- FS-P-2
- FS-P-3
- FS-P-1

C. Incorrect.
Part 1 is correct. IAW TQ-TM-104-810-C001, Fire Suppression:

2) Screen House Motor Driven Fire Pump FS-P-2
a) Located in the Unit 1 Screen House south side of middle wall
b) Vertical suction pump takes suction from the screen well at the Intake Screen House and discharges into the yard fire main
c) Auto-start at 90 psig

3) Unit 1 River Diesel Fire Pump FS-P-3
a) Located outside Unit 1 Screen House by north wall
b) Vertical suction pump takes suction from the separate well (fed from screen well) at the Intake Screen House and discharges into the yard fire main
c) Auto-start at 80 psig

4) Circulating Water Diesel Driven Fire Pump FS-P-1
a) Located in the east end of the Unit 1 Circulating Water Pump House
b) Horizontal suction pump takes water under positive head from the common intake flume and discharges into cooling tower loops
(1) Serves Yard Fire Main through FSV-5
c) Auto-start at 70 psig

Part 2 is incorrect. Plausible if the candidate believes that OP-TM-EOP-001 should be initiated based on Fire System pressure. However, the procedural requirement to initiate OP-TM-EOP-001 is based on criteria that are not met, as stated in the question stem. IAW OP-TM-AOP-001, Fire:

3.10 IAAT the fire causes any of the following conditions to exist:
- Serious damage to safe shutdown equipment
- Degraded ability to maintain power operation

then INITIATE EOP-001.

D. Correct.

Part 1 is correct. IAW TQ-TM-104-810-C001, Fire Suppression:

2) Screen House Motor Driven Fire Pump FS-P-2
   a) Located in the Unit 1 Screen House south side of middle wall
   b) Vertical suction pump takes suction from the screen well at the Intake Screen House and discharges into the yard fire main
   c) Auto-start at 90 psig

3) Unit 1 River Diesel Fire Pump FS-P-3
   a) Located outside Unit 1 Screen House by north wall
   b) Vertical suction pump takes suction from the separate well (fed from screen well) at the Intake Screen House and discharges into the yard fire main
   c) Auto-start at 80 psig

4) Circulating Water Diesel Driven Fire Pump FS-P-1
   a) Located in the east end of the Unit 1 Circulating Water Pump House
   b) Horizontal suction pump takes water under positive head from the common intake flume and discharges into cooling tower loops
      (1) Serves Yard Fire Main through FSV-5
   c) Auto-start at 70 psig

Part 2 is correct. IAW OP-TM-AOP-001, Fire:

3.7 VERIFY FS pressure >90 psig on FS-PI-371.

RNO: START the following fire pumps as necessary to raise FS pressure > 90 psig:

- FS-P-2
- FS-P-3
- FS-P-1

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	086	A2.02	
	Importance Rating	3.0		

Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low FPS header pressure.

Proposed Question: RO Question # 63

Technical Reference(s): TQ-TM-104-810-C001, p 12-13, Rev 9
OP-TM-AOP-001, p1, Rev 10

Proposed References to be provided to applicants during examination: None

Learning Objective: 810-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 10

55.43

Comments:
The KA is matched because the operator must demonstrate the ability to predict the impact on a malfunction of the Fire Service Sytem (a failure of FS-P-371 to send the appropriate signal) and to take procedural action to mitigate/compensate for the malfunction.

The question is at the Comprehension/Analysis cognitive level because the operator must interpret data and then determine the correct action to take as result.

What MUST be known:
1. What are the automatic start setpoints of Fire Service Pumps?
2. What is the procedural guidance given a low Fire Service header pressure?

64 **ID: 1142258** **Points: 1.00**

Plant Conditions (T = 0 minutes):

- The plant is operating at 55% reactor power.
- A damaged fuel pin has been identified.
- RB Purge in progress.

Sequence of Events:

- T = 5 minutes:
  - o The fuel pin damage has worsened.
- T=10 minutes:
  - o RM-G-20, Reactor Coolant Drain Tank Radiation Monitor, is in HIGH ALARM.
  - o RM-A-9, Reactor Building Purge Exhaust Duct Radiation Monitor, is in ALERT.

Given the above information, _________ will automatically close at T=10 minutes.

A. MU-V-2A/B, Letdown Cooler Outlet Valves

B. AH-V-1A/B/C/D, RB Purge Supply/Exhaust Valves

C. WDG-V-3/4, RB Vent Header Containment Isolation Valves

D. WDL-V-534/535, RB Sump Drain to Auxiliary Building Sump Valves

Answer: C

Answer Explanation

A. Incorrect.
MU-V-2A/B are interlocked with RM-L-1, which has not reached high alarm, and is not expected to due to a 30-60 minute time delay associated with the radiation monitor results. Distracter is plausible because a significant fuel pin leak will eventually lead to RM-L-1 alarming, shutting MU-V-2A/B to minimize rad levels in the Auxiliary Building. IAW OP-TM-MAP-C0101, Radiation Level Hi:

- RM-L-1 or RM-L-1 LO - Primary Coolant Letdown
  - CAUSES
    - High primary coolant activity
  - AUTOMATIC ACTIONS
    - RM-L-1: MU-V-2A and 2B close on Hi Alarm
    - RM-L-1 LO: None
  - NOTE: There is a **delay time** between changes in RCS activity and RM-L-1 response of 30 to 60 minutes depending on RM-L-1 flow rate.

B. Incorrect.
Although these valves are interlocked with RM-A-9, they will not close because RM-A-9 is only in the Alert range and not in the high Alarm range. Distracter is plausible because, with a RB purge in progress, automatic securing of the purge might seem like a logical action. IAW OP-TM-MAP-C0101, Radiation Level Hi:

- RM-A-9 - Reactor Building Purge Exhaust Duct
  - CAUSES
    - Filter Units AH-F-1 contamination
    - Improper Purge Flow
    - RCS Leak during purge
  - AUTOMATIC ACTIONS
    - The following occurs on a gaseous **high alarm**:
      - R.B. Purge Valves AH-V-1A, B, C, and D Close
      - R.B. Sump Isol. Valves WDL-V-534 and 535 Close
      - Remote sampler starts (MAP-5)

C. Correct.
WDG-V-3/4 are interlocked to close upon an RM-G-20 high alarm condition. IAW OP-TM-MAP-C0101, Radiation Level Hi:

- RM-G-20 - Reactor Coolant Drain Tank
  - CAUSES
    - High primary coolant activity coupled with RCS leakage to RC Drain Tank
  - AUTOMATIC ACTIONS
    - The following closes on a high alarm:
      - WDL-V-303
      - WDL-V-304
      - WDG-V-3
      - WDG-V-4

D. Incorrect.
Although these valves are interlocked with RM-A-9, they will not close because RM-A-9 is only in the Alert range and not in the high Alarm range. Distracter is plausible because WDL-534 provides a path outside containment if the RCDT rupture disk were to fail on high pressure. IAW OP-TM-MAP-C0101, Radiation Level Hi:

- RM-A-9 - Reactor Building Purge Exhaust Duct
  - CAUSES
    - Filter Units AH-F-1 contamination
    - Improper Purge Flow
    - RCS Leak during purge
  - AUTOMATIC ACTIONS
    - The following occurs on a gaseous **high alarm**:
      - R.B. Purge Valves AH-V-1A, B, C, and D Close
      - R.B. Sump Isol. Valves WDL-V-534 and 535 Close
      - Remote sampler starts (MAP-5)

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	071	A3.03	
	Importance Rating	3.6		

Ability to monitor automatic operation of the Waste Gas Disposal System including: Radiation monitoring system alarm and actuating signals

Proposed Question: RO Question # 64

Technical Reference(s): OP-TM-MAP-C0101, p 37, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-5

Question Source: Bank #

Modified Bank # ID# 356904

New

Question History: Last NRC Exam: 03-01

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	11
	55.43	

Comments:
The KA is matched because the operator must demonstrate the ability to predict (monitor) the automatic operation of valves located within the Waste Gas Disposal System upon an associated radiation monitor high alarm.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must identify interlock actions between the Radiation Monitor System and the Waste Gas Disposal System.

What MUST be known:
1. What are the automatic interlock actions associated with a RM-G-20 high alarm?
2. What is the time delay for RM-L-1 to display correct results?

Original Question:

Plant conditions:

Reactor power is 100%.
RB Purge in progress for Containment Building pressure control.

Event:

Significant FUEL PIN LEAK develops.
RM-G-20 (RCDT Discharge Monitor) HIGH ALARM actuates.

Which of the following statements describes components affected during this situation?

A. AH-V-1A/B/C/D, RB Purge Isolation Valves, automatically close.

B. MU-V-2A/B, Letdown Cooler Outlet Isolation Valves, automatically close.

C. WDG-V-3, RB Vent Header Containment Isolation Valve, automatically closes.

D. WDL-V-534, RB Sump Drain to Auxiliary Building Sump Valve, automatically closes.

Answer: C

65 **ID: 1142194** **Points: 1.00**

Plant Conditions:

- The plant is in COLD SHUTDOWN.
- Source Range indicates 20 CPS.

Given the above information, which ONE of the following correctly describes:

(1) The MINIMUM rise in Source Range Count Rate that will **AUTOMATICALLY** actuate the Reactor Building Evacuation Alarm, and

(2) Where the Reactor Building Evacuation Alarm can be **MANUALLY** initiated from the Control Room?

A. (1) 10 cps.
(2) Panel PL, ONLY.

B. (1) 10 cps.
(2) Panels PL and PRF.

C. (1) 20 cps.
(2) Panel PL, ONLY.

D. (1) 20 cps.
(2) Panels PL and PRF.

Answer: C

Answer Explanation

A. Incorrect.
Part 1 is incorrect. When 10 cps is added to the initial 20 cps, the result is 30 cps. Although 30 cps is associated with the Reactor Building evacuation alarm circuitry, it is the setpoint that the alarm will reset if already actuated. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

- Controls and Interlocks
  - Source Range (NI-11 and NI-12)
    - Reactor Building Evacuation Alarm
      - Actuates at 40 cps if enabled on PRF
      - Reset at 30 cps

Part 2 is correct. IAW 1105-12, Communication System:

- Reactor Building Evacuation Alarm - This is a steady tone of approximately 700 Hertz.
  1. Alarm is automatically initiated by source range detector channels at specified source range level.
  2. If there are potentially any personnel in the Reactor Building and the event has the potential to impact the personnel, then the Reactor Building Evacuation Alarm should be actuated.
  3. Alarm may be manually initiated by depressing and releasing pushbutton on Panel PL in the Control Room.
  4. Signal will sound throughout the plant via Main Plant Communication System page channel for about 30 seconds and stop automatically.

B. Incorrect.
Part 1 is incorrect. When 10 cps is added to the initial 20 cps, the result is 30 cps. Although 30 cps is associated with the Reactor Building evacuation alarm circuitry, it is the setpoint that the alarm will reset if already actuated. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

- Controls and Interlocks
  - Source Range (NI-11 and NI-12)
    - Reactor Building Evacuation Alarm
      - Actuates at 40 cps if enabled on PRF
      - Reset at 30 cps

Part 2 is incorrect. Panel PRF is only to enable the AUTOMATIC alarm actuation. Plausible if the candidate is not familiar with the location and purpose of the components associated with the Reactor Building evacuation alarm. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

- Controls and Interlocks
  - Source Range (NI-11 and NI-12)
    - Reactor Building Evacuation Alarm
      - Actuates at 40 cps if enabled on PRF
      - Reset at 30 cps

Additionally, IAW 1302-1.2, NI-11/12/11A/12A Source and Wide Range Neutron Flux Monitor Calibration:

- DESCRIPTION AND LOCATION OF SYSTEM/ASSEMBLY
  - NI-11 & NI-12 REFUELING ALARM switch (for RB Evacuation Alarm) located on panel PRF.

C. Correct.
Part 1 is correct. When 20 cps is added to the initial 20 cps, the result is 40 cps. At 40 cps, the Reactor Building evacuation alarm automatic circuitry will actuate. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

- Controls and Interlocks
  - Source Range (NI-11 and NI-12)
    - Reactor Building Evacuation Alarm
      - Actuates at 40 cps if enabled on PRF
      - Reset at 30 cps

Part 2 is correct. IAW 1105-12, Communication System:

- Reactor Building Evacuation Alarm - This is a steady tone of approximately 700 Hertz.
  1. Alarm is automatically initiated by source range detector channels at specified source range level.
  2. If there are potentially any personnel in the Reactor Building and the event has the potential to impact the personnel, then the Reactor Building Evacuation Alarm should be actuated.
  3. Alarm may be manually initiated by depressing and releasing pushbutton on Panel PL in the Control Room.
  4. Signal will sound throughout the plant via Main Plant Communication System page channel for about 30 seconds and stop automatically.

D. Incorrect.
Part 1 is correct. When 20 cps is added to the initial 20 cps, the result is 40 cps. At 40 cps, the Reactor Building evacuation alarm automatic circuitry will actuate. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

- Controls and Interlocks
  - Source Range (NI-11 and NI-12)
    - Reactor Building Evacuation Alarm
      - Actuates at 40 cps if enabled on PRF
      - Reset at 30 cps

Part 2 is incorrect. Panel PRF is only to enable the AUTOMATIC alarm actuation. Plausible if the candidate is not familiar with the location and purpose of the components associated with the Reactor Building evacuation alarm. IAW TQ-TM-104-623-C001, Nuclear Instrumentation System:

- Controls and Interlocks
  - Source Range (NI-11 and NI-12)
    - Reactor Building Evacuation Alarm
      - Actuates at 40 cps if enabled on PRF
      - Reset at 30 cps

Additionally, IAW 1302-1.2, NI-11/12/11A/12A Source and Wide Range Neutron Flux Monitor Calibration:

- DESCRIPTION AND LOCATION OF SYSTEM/ASSEMBLY
  - NI-11 & NI-12 REFUELING ALARM switch (for RB Evacuation Alarm) located on panel PRF.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	029	A4.04	
	Importance Rating	3.5		

Ability to manually operate and/or monitor in the control room: Containment evacuation signal

Proposed Question: RO Question # 65

Technical Reference(s): TQ-TM-104-623-C001, p 28, Rev 4
1105-12, p 14, Rev 27

Proposed References to be provided to applicants during examination: None

Learning Objective: 623-GLO-5

Question Source: Bank # ID# 719623
Modified Bank #
New

Question History: Last NRC Exam: 08-01 retest

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 operator must demonstrate the ability to manually operate and monitor the containment evacuation signal in the control room (i.e. be able to distinguish the tone from other site emergency alarms, and identify the location for manual operation within the Control Room).

The question is at the Comprehension cognitive level because the operator must calculate what rise in Source Rate Counts would actuate the alarm automatically. As well as recall where can the RB Evacuation Alarm be operated from.

What MUST be known:
1. At what Source Range counts will the Reactor Building evacuation alarm automatically actuate?
2. Basic arithmetic skills.
3. The Control Room location to manually actuate the Reactor Building evacuation alarm.

66 **ID: 1151709** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- The following computer alarms are observed for the "A" Reactor Coolant Pump, RC-P-1A:
  - OP-TM-PPC-A0704, RC-P-1A STATOR TEMP (DEG C).
  - OP-TM-PPC-A0699, RC-P-1A MOTOR AIR TEMPERATURE.
  - OP-TM-PPC-A0702, RC-P-1A MTR UPPER GUIDE BRG TEMP.
  - OP-TM-PPC-A0703, RC-P-1A MTR LOWER GUIDE BRG TEMP.

Given the above conditions, which of the following would cause the above indications for RC-P-1A?

A. RCP #1 Seal Failure.

B. Loss of Seal Injection.

C. Loss of Intermediate Component Cooling.

D. Loss of Nuclear Services Component Cooling.

Answer: D

Answer Explanation

A. Incorrect.
A #1 Seal Failure would not cause all of the given alarms simultaneously. Plausible because Seal Failure is a common fault associated with Reactor Coolant Pumps (OE31344, OE31249).

B. Incorrect.
A Loss of Seal Injection would not cause all of the given alarms simultaneously. Plausible because a loss of seal injection is a common fault associated with Reactor Coolant Pumps.

C. Incorrect.
A Loss of Intermediate Component Cooling would not cause all of the given alarms simultaneously. Plausible because a loss of Intermediate Component Cooling is a common fault associated with Reactor Coolant Pumps.

D. Correct.
A Loss of Nuclear Services Component Cooling is the common cause for all of the given alarms.

- OP-TM-PPC-A0704, RC-P-1A STATOR TEMP (DEG C): 2.0 CAUSES
  - **Loss of NS Cooling**
  - High ambient or NS temperature
  - ESAS Line Break Isolation or ESAS 30 psig RB Actuation

- OP-TM-PPC-A0699, RC-P-1A MOTOR AIR TEMPERATURE: 2.0 CAUSES
  - **Loss of NS Cooling**
  - High ambient or NS temperature
  - ESAS Line Break Isolation or ESAS 30 psig RB Actuation

- OP-TM-PPC-A0702, RC-P-1A MTR UPPER GUIDE BRG TEMP: 2.0 CAUSES
  - Bearing failure
  - Loss of Lift Oil flow
  - **Loss of NS Cooling**
  - High ambient or NS temperature
  - ESAS Line Break Isolation or ESAS 30 psig RB Actuation

- OP-TM-PPC-A0703, RC-P-1A MTR LOWER GUIDE BRG TEMP: 2.0 CAUSES
  - Bearing failure
  - Loss of Lift Oil flow
  - **Loss of NS Cooling**
  - High ambient or NS temperature
  - ESAS Line Break Isolation or ESAS 30 psig RB Actuation

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	2.1.19	
Importance Rating	3.9	

Ability to use plant computers to evaluate system or component status.

Proposed Question: RO Question # 66

Technical Reference(s):
OP-TM-PPC-A0699, p 1, Rev 1
OP-TM-PPC-A0702, p 1, Rev 1
OP-TM-PPC-A0703, p 1, Rev 1
OP-TM-PPC-A0704, p 1, Rev 1

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-8

Question Source: Bank # ID# 462960

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

Comments:
The KA is matched because the operator must demonstrate the ability to use plant computers to evaluate system or component status.

The question is at the Comprehension cognitive level because the operator must identify what interrelated system is affecting the Reactor Coolant Pumps given only Plant Computer Alarms.

What MUST be known:
1. What are the causes for OP-TM-PPC-A0704, RC-P-1A STATOR TEMP (DEG C), to alarm?
2. What are the causes for OP-TM-PPC-A0699, RC-P-1A MOTOR AIR TEMPERATURE, to alarm?
3. What are the causes for OP-TM-PPC-A0702, RC-P-1A MTR UPPER GUIDE BRG TEMP, to alarm?
4. What are the causes for OP-TM-PPC-A0703, RC-P-1A MTR LOWER GUIDE BRG TEMP, to alarm?

67 **ID: 1147049** **Points: 1.00**

Plant Conditions:

- A Plant Shutdown in progress.
- Tave = 567 degF.
- Maintaining pressurizer level IAW OP 1102-10, Plant Shutdown.

Based on the above information, what is the recommended pressurizer level to be maintained?

A. 48"

B. 151"

C. 182"

D. 191"

Answer: D

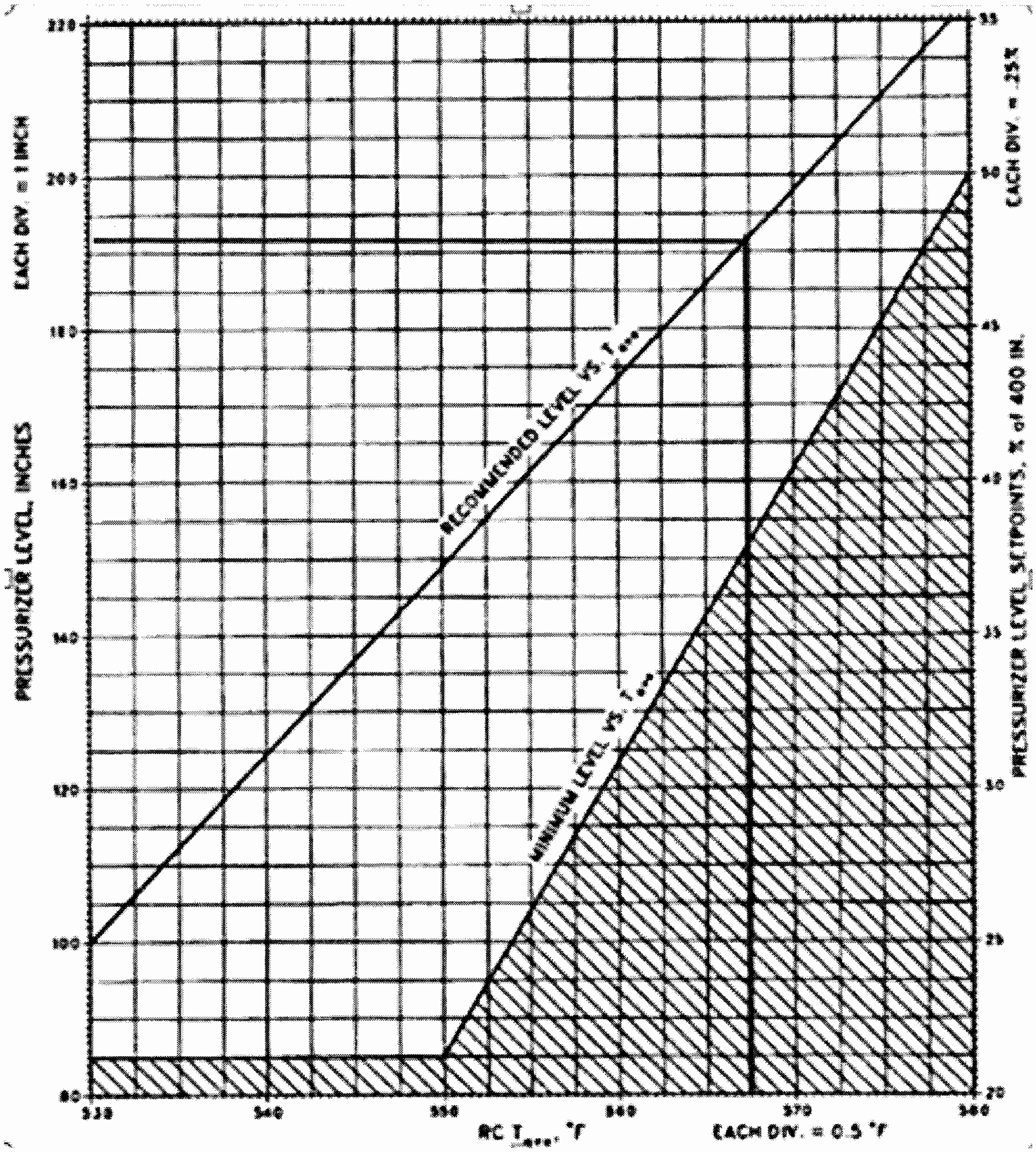
Answer Explanation

A. Incorrect.
Plausible if the candidate misinterprets the graph. If the candidate finds the correct temperature and the correct level line (the upper of the two slopes), but reads the Pressurizer Level Setpoints in % of 400 inches (the label on the right), then the candidate will determine that 48" is the correct answer.

B. Incorrect.
Plausible if the candidate misinterprets the graph. If the candidate finds the correct temperature but the minimum level line (the lower of the two slopes), then the candidate will determine that 151" is the correct answer.

C. Incorrect.
Plausible if the candidate misinterprets the graph. If the candidate does not find the correct temperature but does find the correct line (the upper of the two slopes), then the candidate will determine that 182" is the correct answer.

D. Correct.
567 deg-F is just to the left of the last line before 570 deg-F. Moving up from that point to where it intersects that recommended level line (the upper of the two slopes), the candidate will determine that 191" is the correct answer.



RECOMMENDED LEVEL VS. Tave
MINIMUM LEVEL VS. Tave
PRESSURIZER LEVEL, INCHES
EACH DIV. = 1 INCH
PRESSURIZER LEVEL SETPOINTS, % of 400 IN.
EACH DIV. = .25%
RC Tave, °F
EACH DIV. = 0.5 °F
220
200
180
160
140
120
100
80
530
540
550
560
570
580
55
50
45
40
35
30
25
20

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.25	
	Importance Rating	3.9	

Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: RO Question # 67

Technical Reference(s): OP-TM-211-472, p 6, Rev 4

Proposed References to be provided to applicants during examination: OP-TM-211-472 att 7.2

Learning Objective: GOP-010-PCO-1

Question Source: Bank #

Modified Bank # ID# 371611

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

Comments:
The KA is matched because the operator must interpret reference materials (a graph) associated with the job description of a Reactor Operator.

The question is at the Comprehension/Analysis cognitive level because the operator must interpret a graph and correctly plot a point.

What MUST be known:
1. What are the axis labels on the Pressurizer Level vs Temperature graph?
2. How to interpret a graph.

Original Question:

Plant conditions:

Plant shutdown in progress.
Tave = 575 deg-F.
Maintaining pressurizer level IAW OP 1102-10, Plant Shutdown.

Based on the above conditions, what is the **minimum** pressurizer level that must be maintained?

A. 100".

B. 185".

C. 205".

D. 220".

Answer: B

68 **ID: 1142260** **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

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	

Knowledge of tagging and clearance procedures.

Proposed Question: RO Question # 68

Technical Reference(s): OP-MA-109-101, pg 12, Rev 20

Proposed References to be provided to applicants during examination: None

Learning Objective: NOP-DBIG-PCO-2

Question Source: Bank # ID# 977950

Modified Bank #

New

Question History: Last NRC Exam: 2014

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.

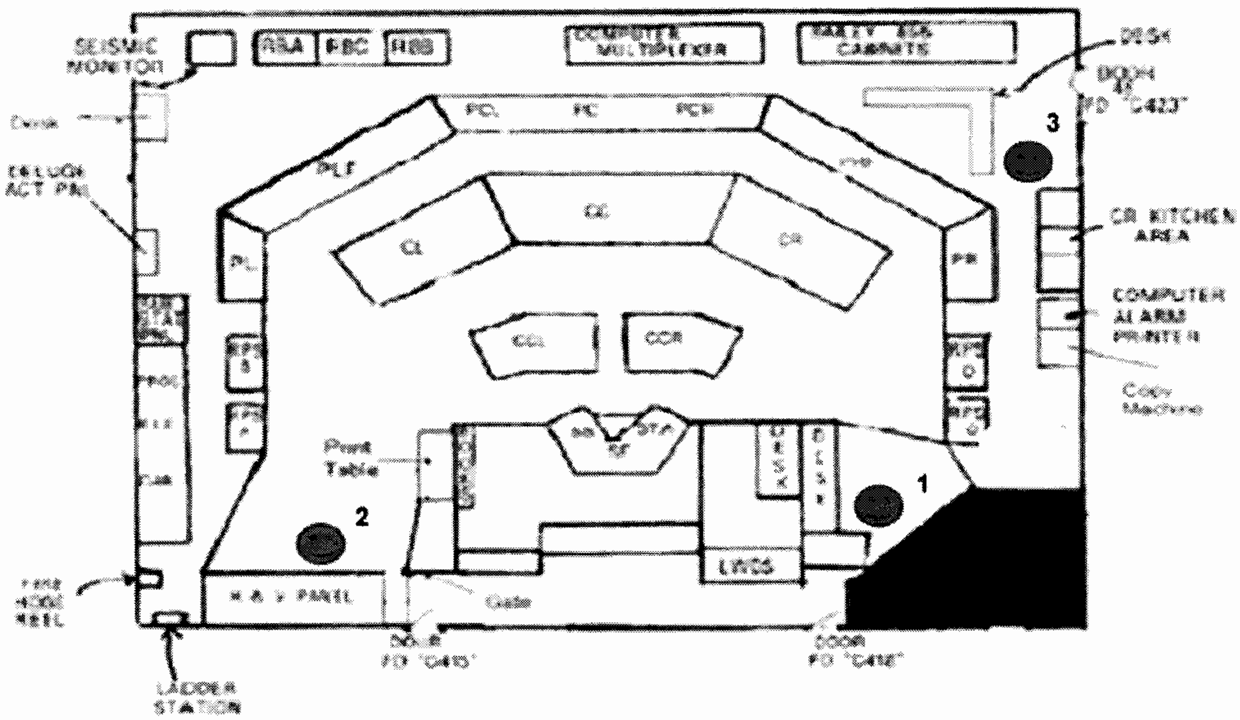
What MUST be known:
1. What are the procedural requirements associated with hanging a Danger Tag that has another tag applied already?

69 **ID: 1147068** **Points: 1.00**

Plant Conditions:

- Three Reactor Operators (1-3) are in the Control Room.
- There are **Non-Routine** Operations in progress.

IAW OP-TM-101-111-1002, Definition of the Operator at the Controls Area, which of the Reactor Operators can be considered the Operator at the Controls?



A. Operators 1 or 2.

B. Operators 2 or 3.

C. Operator 2, ONLY.

D. Operators 1,2, or 3.

Answer: A

Answer Explanation

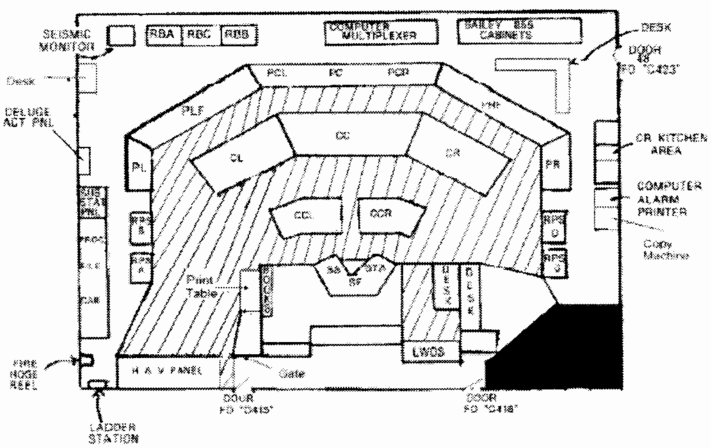
A. Correct.

IAW OP-TM-101-111-1002:

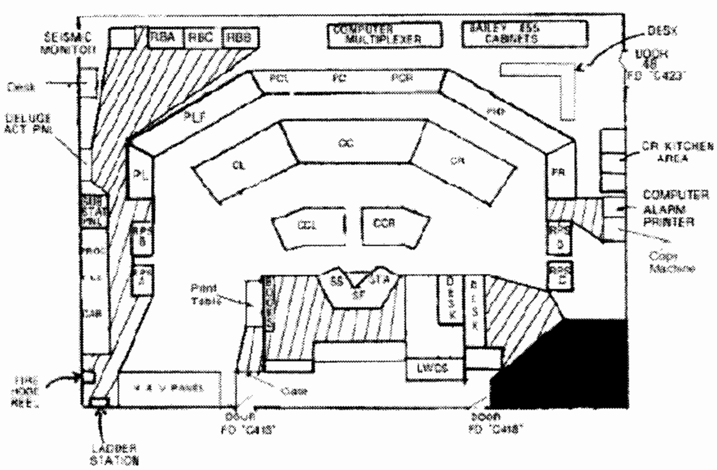
4.1. The Unit RO is to remain in the work areas defined by Attachment 1 and Attachment 2 as "Operator at the Controls" at all times when the Unit is fueled unless properly relieved.

4.1.1. The Unit RO will remain in the area defined by Attachment 1 unless required to enter the area defined by Attachment 2 to perform Non-Routine Operations.

ATTACHMENT 1
Routine Operations Area
Page 1 of 1



ATTACHMENT 2
Non-Routine Operations Area
Page 1 of 1



B. Incorrect.
Although Operator #2 can be considered At-the-Controls, Operator #3 cannot be considered At-the-Controls according to either Attachment 1 (Routine Operations) or Attachment 2 (Non-Routine Operations) of OP-TM-101-111-1002. See drawings above.

C. Incorrect.
Although Operator #2 can be considered At-the-Controls, Operator #1 can also be considered At-the-Controls since Non-Routine operations are in progress (as given in the stem). Plausible if the candidate does not recognize Non-Routine operations. See drawings above.

D. Incorrect.
Although Operators #1 and #2 can be considered At-the-Controls, Operator #3 cannot be considered At-the-Controls according to either Attachment 1 (Routine Operations) or Attachment 2 (Non-Routine Operations) of OP-TM-101-111-1002. See drawings above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.38	
	Importance Rating	3.6	

Knowledge of conditions and limitations in the facility license.

Proposed Question: RO Question # 69

Technical Reference(s): OP-TM-101-111-1002, pg 3-4, Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: 101-OAC-PCO-4

Question Source: Bank #
Modified Bank # ID# 884538
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	

Comments:

The KA is matched because the question requires the candidate to demonstrate a knowledge of conditions and limitations in the facility license. OP-TM-101-111-1002 was created IAW the following documentation:

NRC Regulatory Guide 1.114
10 CFR 50.54K
10 CFR 55.4
TMI 1 Technical Specifications Section 6.2.2

IAW NRC Regulatory Guide 1.114:

B. DISCUSSION
1. Operator at the Controls

Operating experience has shown that a need exists for guidance on acceptable methods of complying with the Commission's requirement that an operator must be present at the controls of a facility. The operator at the controls of a nuclear power unit has many responsibilities including the following:

- adhering to the unit's technical specifications, plant operating procedures, and NRC regulations;
- reviewing operating data, including data logging and review, to ensure that the unit is operating safely; and
- being able to manually initiate engineered safety features during various transient and accident conditions.

To carry out these and other responsibilities in a timely fashion, the operator at the controls of a nuclear power unit must pay attention to the condition of the unit at all times. The operator must be alert to ensure that the unit is operating safely and must be able to take action to prevent any progress toward an unsafe condition. This is facilitated by control room design and layout in which all controls, instrumentation displays, and alarms required for the safe operation, shutdown, and cooldown of the unit are readily available to the operator in the control room.

C. REGULATORY POSITION
1. Operator at the Controls

1.1 The operator at the controls of a nuclear power unit should have an unobstructed view of, and access to, the operational control panels, including instrumentation displays and alarms, to initiate prompt corrective action when necessary on receipt of any indication (i.e., instrument response or alarm) of a changing condition. Operational control panels are control panels that enable the operator at the controls to perform required manual safety functions and equipment surveillance and to monitor plant conditions under normal and accident conditions.

1.2 The operator at the controls should not normally leave the area where continuous attention, including visual surveillance of annunciators and instrumentation, can be given to reactor operating conditions and where the operator has access to the reactor controls. For example, the operator should not routinely enter areas behind control panels where he cannot monitor plant performance. If the control room design is such that an operator must enter areas behind control panels to monitor back panels, either a senior operator or reactor operator assigned to the current control room shift must be within view of the control panels during the time that the normally assigned operator is monitoring the back panels. The operator at the controls should not, under any circumstances, leave the surveillance area (i.e., defined by the administrative procedures described in response to Regulatory Position 1.3 below) for any nonemergency reason (e.g., to confer with others or for personal reasons) without ensuring that a qualified relief operator is at the controls. In an emergency that affects the safety of operations, the operator at the controls may momentarily be absent from the defined surveillance area to verify the receipt of an annunciator alarm or to initiate corrective action, provided that the operator remains within the confines of the control room.

1.3 Administrative procedures should be established that define and outline (preferably with sketches) the specific area within the control room designated as the “surveillance area” where the operator at the controls should remain. The procedures should define the surveillance area and other areas that the operator at the controls may enter to verify the receipt of an annunciator alarm or to initiate corrective action in an emergency that affects the safety of operations.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to view a drawing and determine facility license requirements with regards to At-the-Controls.

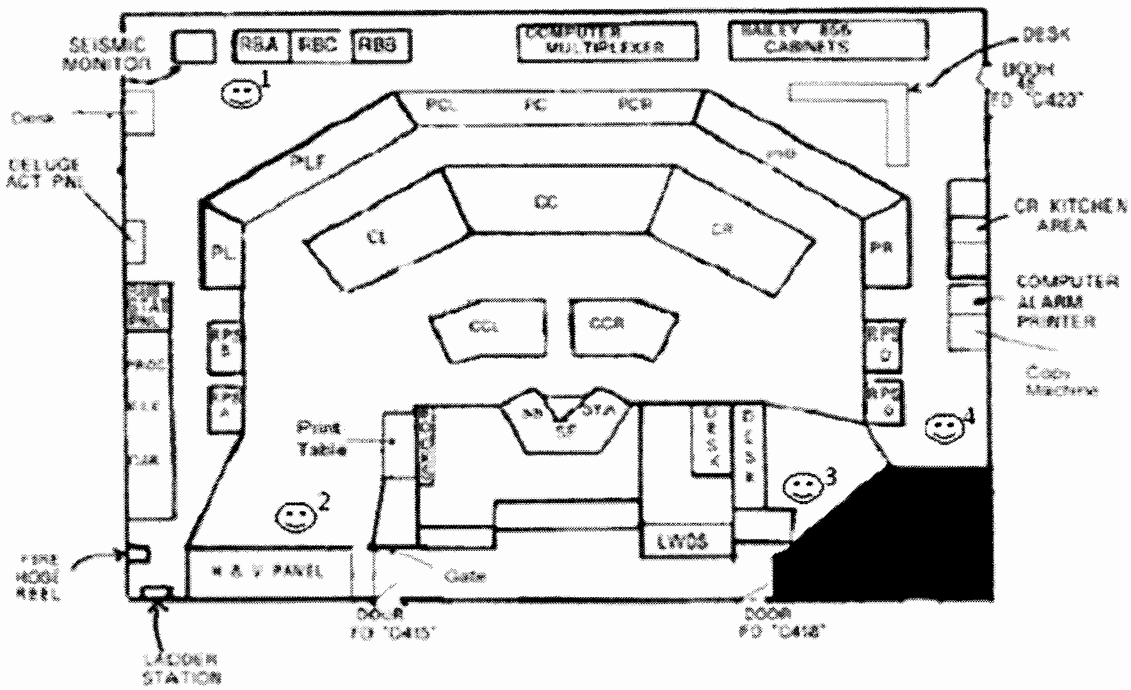
What MUST be known:
1. Where in the Control Room the "At the controls" Reactor Operator is allowed to physically be during Non-routine operations.

Original Question:

Plant Conditions:

Four Reactor Operators (1-4) are in the Control Room.
There are **Non-Routine** Operations in progress.

IAW OP-TM-101-111-1002, Definition of the Operator at the Controls Area, which of the Reactor Operators can be considered the Operator at the Controls?



A. 1, ONLY.

B. 1, 2, or 3.

C. 2, 3, or 4.

D. 4, ONLY.

Answer: B

70 **ID: 1142261** **Points: 1.00**

Plant Conditions:

- A Plant Start-up is in progress IAW 1102-2, Plant Startup.
- Reactor Power is 3%.
- TI-959A (A wide range Tcold) reads 534°F.
- TI-961A (B wide range Tcold) reads 534°F.
- The Attachment 2, Actions to Support a Power Escalation, steps that are required to be performed before raising reactor power above 3% have not been completed but are in progress.

Event:

- The "A" Turbine Bypass Valves, MS-V-3D/E/F, fail to 50% open.
- TI-959A (A wide range Tcold) reads 524°F.
- TI-961A (B wide range Tcold) reads 531°F.

From the list below choose the required operator action.

A. Stabilize plant and hold at 2% power.

B. Verify or make reactor at least 1% delta K/K shutdown.

C. Take manual control of MS-V-3D/E/F and continue the power escalation.

D. Ensure OTSG tube to shell delta T limits are <50°F for the affected OTSG and continue the power escalation.

Answer: B

Answer Explanation

A. Incorrect.
Plausible if the candidate does not recognize that RCS temperature has gone below an actionable setpoint and believes that reactor power must be on hold to complete the applicable steps of Attachment 2. IAW 1102-2, Plant Startup:

- In parallel with subsequent steps, perform actions when required per Enclosure 2. If conditions required by Enclosure 2 are not satisfied or any time as directed by the CRS, HOLD Reactor power level stable, using the reactor control mode applicable for that power level.

B. Correct.
IAW Tech Specs:

3.1.3.1 The reactor coolant temperature shall be above 525?F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.
3.1.3.2 Reactor coolant temperature shall be above DTT +10?F.
3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

Additionally, IAW 1102-2, Plant Startup:

2.0 LIMITS AND PRECAUTIONS
2.1 Reactivity & Reactor Power Control
2.1.1 Continuously monitor the nuclear instruments during any reactivity addition.
2.1.2 If at any time the most conservative valid wide range cold leg temperature indication shows a RCS temperature less than 525°F, ensure [insert control rods or borate] the reactor is at least 1 % _ K/K shutdown.

C. Incorrect.
Plausible if the candidate does not recognize that RCS temperature has gone below an actionable setpoint and believes that MS-V-3D/E/F can be controlled manually IAW OS-24, Conduct of Operations During Abnormal and Emergency Events or IAW 1102-2, Plant Startup, Enclosure 2, Actions to Support a Power Escalation. IAW OS-24:

- Actions Not Described in Procedures
  - Licensed operators may take action without procedural guidance, and without taking a variance under the following conditions.
    - Initiating a manual reactor TRIP when a licensed operator believes the reactor is not in a safe condition.
    - Action taken to directly compensate for the failure of an automatic system.

Additionally, IAW 1102-2, Plant Startup, Enclosure 2, Actions to Support a Power Escalation:

- Between 12% and 18% Reactor Power:
  - When Turbine Bypass Valve demand is > 30% and < 80%:
    - RAISE Turbine Load and ENSURE TBV control of Turbine Header Pressure between 885 and 905 psig.

D. Incorrect.
Plausible if the candidate does not recognize that RCS temperature has gone below an actionable setpoint and recognizes that 50F is within the allowable bands for OTSG tube to shell DT. Although it is prudent to minimize tube to shell delta T, 1102-2, Plant Startup, does not specifically discuss the limits like 1102-1, Plant Heatup to 525F does. Even then, the limit is +60F. The PPC alarm setpoint, C4015, OTSG A (B) Tube to Shell DT Hi/Lo Alarm, also states that +60F is the high 3 alarm setpoint. IAW OP-TM-PPC-4015:

- SETPOINTS
  - HI +40F
  - HI2 +50F
  - HI3 +60F
  - LO - 60F
  - LO2 - 70F

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.42	
	Importance Rating	3.9	

Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Proposed Question: RO Question # 70

Technical Reference(s): Tech Spec, p 3-6, Amend 278
1102-2, pg 3, Rev 157

Proposed References to be provided to applicants during examination: None

Learning Objective: GOP-002-PCO-1

Question Source: Bank # ID# 371534

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	

Comments:
The KA is matched because the question requires the candidate to recognize entry-level conditions for Technical Specifications.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to identify the entry level condition for Tech Specs from a given scenario.

What MUST be known:
1. What action is to be taken upon RCS temperature dropping below 525F upon a reactor startup?

71 **ID: 1147087** **Points: 1.00**

Plant Conditions:

- The plant is in a Refueling Outage.
- Fuel moves are in progress.
- Equipment Hatch Doors are open.

Event:

- MAP C-1-1 "Radiation Level High" in alarm.
- RM-A-9, Reactor Building Purge Exhaust Duct Radiation Monitor, High Alarm is lit.
  - The indication is on-scale.

Given the above information, which of the following describes the correct action to be taken?

A. Suspend Fuel Handling, and evacuate all personnel from the Reactor Building.

B. Ensure all Purge isolation valves (AH-V-1A, AH-V-1B, AH-V-1C, and AH-V-1D) are closed.

C. Make a plant page announcement to limit the use of environmental doors between Unit 1 and Unit 2.

D. Stop both Purge Supply fans (AH-E-6A and AH-E-6B) and continue to run both exhaust fans (AH-E-7A and AH-E-7B).

Answer: A

Answer Explanation

A. Correct.
IAW OP-TM-MAP-C0101, Radiation Level Hi:

- RM-A-9 - Reactor Building Purge Exhaust Duct

  4.6 IAAT high alarm is Lit, then perform the following:
    4. If fuel handling is in progress, then perform the following:
       - NOTIFY the fuel handling SRO to suspend fuel handling IAW 1505-1, "Fuel and Control Component Shuffles".

B. Incorrect.
The defeat switch is in defeat during refueling and alarm response specifically removes this action during refueling. Plausible as these valves would normally close on High Rad RM-A-9 Gas signal, however valves must remain open during re-fueling. IAW OP-TM-MAP-C0101, Radiation Level Hi:

- RM-A-9 - Reactor Building Purge Exhaust Duct
  4.6 IAAT high alarm is Lit, then perform the following:
    1. If fuel handling is not in progress, then ENSURE the following are Closed:
       - AH-V-1A
       - AH-V-1B
       - AH-V-1C
       - AH-V-1D

C. Incorrect.
This would be an action taken if the issue was in the Fuel Handling Building instead of the Reactor Building. Plausible since this action would be done for either an RM-A-4 Alert or High alarm. IAW OP-TM-MAP-C0101, Radiation Level Hi:

- RM-A-4 - Fuel Handling Bldg. Exhaust
  4.0 MANUAL ACTIONS REQUIRED
    4.1 ANNOUNCE the alarm and the following over plant page and radio: "Limit the use of all of the doors going through the Unit 1 and Unit 2 environmental barrier to emergency use only. This includes opening the Aux. Building elevator door on the 3rd floor (348' elevation)."

D. Incorrect.
This would be a similar action taken if the issue was in the Fuel Handling Building or the Auxiliary Building instead of the Reactor Building. This would ensure flow into the building to ensure filtered monitored flow (a standard practice for other buildings). Plausible if the candidate is not familiar with the difference in actions between the different Buildings associated with radiation monitors. IAW OP-TM-MAP-C0101, Radiation Level Hi:

- RM-A-4 - Fuel Handling Bldg. Exhaust
  - IAAT high alarm is Lit, then perform the following:
    1. ENSURE AH-E-10 (FHB Supply Fan) is Shutdown.
- RM-A-6 - Auxiliary Building Vent Exhaust
  4.5 IAAT high alarm is Lit, then perform the following: ___
    1. ENSURE AH-E-11 (Auxiliary Building Supply Fan) is shutdown.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.13	
	Importance Rating	3.4	

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: RO Question # 71

Technical Reference(s): OP-TM-MAP-C0101, p 17, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-6

Question Source: Bank # ID# 1110398

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 12

55.43

Comments:
The KA is matched because the question requires the candidate to demonstrate knowledge of radiological safety procedures (specifically, response to radiation monitor alarms).

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to recall bits of information (procedural step) when given a scenario.

What MUST be known:
1. What is the relationship between RM-A-9 and Fuel Handling?
2. What action is taken while RM-A-9 is in High Alarm and fuel movements are in progress?

72 **ID: 1142264** **Points: 1.00**

Plant Conditions:

- The plant is operating at 60% reactor 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, RM-A-2 Reactor Building Atmospheric Monitor ___(1)___ be used to determine whether the pressure increase is from 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

A. Incorrect.

Part 1 is incorrect. 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 not correct. 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:

- 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.

B. Incorrect.

Part 1 is incorrect. 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 not correct. Plausible as RM-A-2 has these components and they would saturate if the detector was not isolated. 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.

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.

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 not correct. Plausible as RM-A-2 has these components and they would saturate if the detector was not isolated. 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.15	
	Importance Rating	2.9	

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: RO Question # 72

Technical Reference(s): 1105-3, p 15, Rev 52
TQ-TM-104-661-C001, p 27, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-2

Question Source: Bank # ID# 719630
Modified Bank #
New

Question History: Last NRC Exam: 2014

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

10 CFR Part 55 Content:	55.41	11
	55.43	

Comments:

The KA is matched because the operator must demonstrate knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. (i.e. Fixed RM monitors associated with a release to the River, alarm functions for RM-L-12).

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 function of RM-A-2?
2. What components are actuated on an ES signal?
3. What interlocks are associated with RM-A-2?

73 **ID: 1149435** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- A severe Thunderstorm Warning is in effect for Middletown, PA.
- Outside air temperature is 96°F.

Event:

- Lightning has struck Three Mile Island, southwest of the plant.
- Outbuildings Operator calls in and reports the following:
  - He thinks he felt minor ground motion.
  - The lightning struck a fixed-structure tank and he smells toxic gas in the air.
- Currently:
  - Control Room ambient temperature is 84°F and relatively steady.
  - Panel PLF Wind Speed recorder is indicating sustained wind speed of 42 mph with occasional gusts up to 50 mph.
  - PRF-1-2, Threshold Seismic Condition, is in alarm.
  - PRF 1-3, Operating Basis Earthquake, is NOT in alarm.
  - All four (4) Nuclear Service Water Coolers are in service, the Cooler outlet temperature is constant at 96°F.
  - There is NO smell of toxic gas in the Control Room.

Given the above information, which one of the following procedures is currently required to be entered?

A. OP-TM-AOP-003, Earthquake.

B. OP-TM-AOP-058, Toxic Gas Release.

C. OP-TM-AOP-004, Tornado/High Winds.

D. OP-TM-AOP-034, Loss of Control Building Cooling.

Answer: A

Answer Explanation

To answer/understand this question an analysis of the given conditions balanced against the procedures that may be the answer is done. This analysis will show that although all of the given conditions are close to the entry conditions, only one, OP-TM-AOP-003 has been exceeded in the current situation.

A. Correct.
IAW OP-TM-AOP-003, Earthquake:

1.0 ENTRY CONDITIONS

- Any of the following:
    - Yellow EVENT indicator Lit on front panel of Strong Motion Accelerometer System (CR)
    - PRF-1-2, "Threshold Seismic Condition" actuated
    - Red OBE indicator Lit on front panel of Strong Motion Accelerometer System (CR)
    - PRF-1-3, "Operating Basis Earthquake" actuated
    - Ground motion felt by station personnel

B. Incorrect.
IAW OP-TM-AOP-058, Toxic Gas Release:

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.

C. Incorrect.
IAW OP-TM-AOP-004, Tornado/High Winds:

1.0 ENTRY CONDITIONS

- Any of the following:
    - A Tornado/High Wind Storm WATCH or WARNING issued by the National Weather Service (NWS) for areas surrounding TMI
    - Wind speed recorder NWS-R-501 indicates sustained winds 50 mph
    - Wind speed on Plant Process Computer Point A0992 indicates sustained winds 50 mph
    - Tornado/funnel cloud is sighted and reported by plant personnel.

D. Incorrect.
Nothing in the stem reaches the entry into AOP-034, however temperatures > 80F are listed as action points in AOP-034. IAW OP-TM-AOP-034, Loss of Control Building Cooling:

1.0 ENTRY CONDITIONS

1.1 Unplanned loss of forced circulation in any of the following rooms of the Control Building for >15 minutes or loss of both CB Chillers >15 minutes:

1. Control Room
2. ESAS Room
3. 1D or 1E 4160V SWGR Room
4. Relay Room
5. 1P or 1S 480V SWGR Room
6. Remote Shutdown area
7. A or B Inverter Room

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.4	
	Importance Rating	4.5	

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: RO Question # 73

Technical Reference(s): OP-TM-AOP-003, p 1, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-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	

Comments:

The KA is matched because the operator must demonstrate the ability to recognize entry-level criteria for abnormal operating procedures.

The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to identify entry criteria for multiple Abnormal Operating Procedures.

What MUST be known:

1. What is the entry criteria for OP-TM-AOP-003?
2. What is the entry criteria for OP-TM-AOP-004?
3. What is the entry criteria for OP-TM-AOP-034?
4. What is the entry criteria for OP-TM-AOP-058?

74 **ID: 1147088** **Points: 1.00**

Plant Conditions:

- The Reactor is at 100% power.

Sequence of Events:

- Loss of offsite power with EG-Y-1B start failure.
- OP-TM-EOP-001, Reactor Trip, immediate actions are complete.
- One control rod stuck at 100% withdrawn position.
- Channel A and B 1600 psig ES Actuation.
- Control room staffing is limited to the minimum required by Technical Specifications.
- Current Conditions:

Parameter	Value	Trend
RCS Pressure	1050 psig	Rising
Core exit T/C temperatures	556 °F	Rising
Loop A/B cold leg temperatures	547 °F	Lowering
Pressurizer level	85 inches	Steady
Total HPI Flow	520 gpm	Steady
RCP Seal Injection Flow	30 gpm	Steady
OTSG pressures	980 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 performed by the Control Room team.

A. Reduce HPI flow.

B. Minimize subcooled margin.

C. Start SBO Diesel Generator.

D. Initiate Guide 1, Emergency Boration Backup Methods.

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. IAW OP-TM-EOP-010, Rule 2, HPI Throttling:

- VERIFY MU PUMP FLOW </=515 gpm/pump.
    - RNO: THROTTLE HPI IAW OP-TM-211-901, "Emergency Injection (HPI / LPI)" to between 500 and 515 gpm/pump.

Also, 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. IAW OS-24:

- Rules are numbered according to priority. If multiple Rule based actions are required, the highest priority Rule is performed first.

B. 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.

C. 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 of actions 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.

D. 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 includes 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.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.5	
	Importance Rating	3.7	

Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

Proposed Question: RO Question # 74

Technical Reference(s): OP-TM-EOP-010, p 4, Rev 18
OS-24, p 7, Rev 25

Proposed References to be provided to applicants during examination: None

Learning Objective: HPI-PCO-4

Question Source: Bank # ID# 365673

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

Comments:

The KA is matched because the operator must demonstrate knowledge of the organization of the operating procedures network for evolutions.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to identify entry criteria for multiple Abnormal Operating Procedures and then must compare the procedures to decide which of the procedures takes precedence.

What MUST be known:
1. What are the procedural steps of Rule 2? 2. How are procedured prioritized at TMI?

75 **ID: 1142196** **Points: 1.00**

Sequence of Events:

- A LOCA has occurred.
- The crew has entered OP-TM-EOP-009, HPI Cooling, and is verifying ADEQUATE HPI.
- The following is observed:
  - RCS Pressure is 950 psig and steady.
  - Total HPI flow is 410 gpm and steady.
  - SCM is 28°F and steady.

Given the above information, which one of the following:
(1) Correctly assesses the above situation, and
(2) Identifies the proper actions to mitigate the event.

A. (1) HPI is adequate.
(2) Stop all but one Reactor Coolant Pump and open the PORV (RC-RV-2).

B. (1) HPI is adequate.
(2) Stop all Reactor Coolant Pumps and de-energize the Pressurizer Heaters.

C. (1) HPI is NOT adequate.
(2) Stop all but one Reactor Coolant Pump and open the PORV (RC-RV-2).

D. (1) HPI is NOT adequate.
(2) Stop all Reactor Coolant Pumps and de-energize the Pressurizer Heaters.

Answer: D

Answer Explanation

A. **Incorrect.**
1st part wrong, 2nd part wrong. See B and C.

B. **Incorrect.**
1st part wrong, 2nd part correct. This is plausible because the operator may interpret the Attachment incorrectly, and determine that adequate HPI exists; and the operator may confuse the strategies of EOP-009.

C. **Incorrect.**
1st part correct, 2nd part wrong. This is plausible because these actions would be taken if HPI was adequate. The operator may confuse the strategies of EOP-009.

D. **Correct.**
1st part correct, 2nd part correct. According to OP-TM-EOP-0091, Step 3.4 is accomplished to verify an adequate makeup supply is available for HPI COOLING. Adequate HPI is verified when HPI flow exceeds flow on Attachment 7.4 in OP-TM-211-901. Consequently, this attachment is provided for the operator to make this determination. Using the intersection of RCS pressure at 950 psig and Total HPI flow of 410 gpm, the operator will determine that HPI flow is NOT adequate. According to OP-TM-EOP-009, Step 3.4, if HPI flow is inadequate, the operator is directed to Section 4.0 of EOP-009 where the operator will shutdown all RCPs based on SCM ? 25°F, and de-energize all the Pressurizer Heaters.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.47	
	Importance Rating	4.2	

Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: RO Question # 75

Technical Reference(s):
OP-TM-EOP-009, p 3, Rev 8
OP-TM-EOP-0091, p 3, Rev 3
OP-TM-211-901, p 22, Rev 7

Proposed References to be provided to applicants during examination: Attachment 7.4 of OP-TM-211-901

Learning Objective:	EOP009-PCO-2		
Question Source:	Bank #	ID# 909256	
	Modified Bank #		
	New		
Question History:		Last NRC Exam:	2012 (10-02)
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments:
The KA is matched because the operator must demonstrate the ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. This is accomplished by providing the operator with a set of plant indications that would require the use of an Attachment available in the Control Room to make a determination on ECCS status; and then based on the status, take the appropriate action.

The question is at the Comprehension/Analysis cognitive level because the operator must recall bits of information, and then evaluate plant conditions properly to correctly answer the question.

What MUST be known:
1. Adequate HPI is determined by Attachment 7.4. 2. How is Attachment 7.4 used? 3. What procedure actions are taken within EOP-009 if it is determined that Inadequate HPI flow exists

76 **ID: 1147363** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Sequence of Events:

- The reactor is tripped due to large break LOCA.
- LPI, and HPI Pumps are tripped and recovery attempts failed.
- Incore Thermocouple temperature is 1610°F.
- RCS pressure is 800 psig.
- Both OTSG pressures are 400 psig.
- PORV, RC-RV-2, failed open early in the event.
- PORV Block Valve, RC-V-2, is closed.
- RCS High Point Vents are closed.
- Emergency Feedwater Pump, EF-P-1, is the only EFW Pump running.
- Auxiliary Steam is unavailable.
- The Technical Support Center is staffed.
- Reactor Coolant Pumps were not tripped within one minute of the loss of subcooling margin and are still running.

Given the above information, which ONE of the following correctly describes the:
(1) First action to be taken, and
(2) Procedural routing that the CRS will take?

A. (1) Secure all Reactor Coolant Pumps.
(2) Go to ER-TM-TSC-0010, Severe Accident Guidelines and exit OP-TM-EOP-008.

B. (1) Secure all Reactor Coolant Pumps.
(2) Initiate ER-TM-TSC-0010, Severe Accident Guidelines and remain in OP-TM-EOP-008.

C. (1) Open the PORV block valve and the PORV to reduce RCS pressure.
(2) Go to ER-TM-TSC-0010, Severe Accident Guidelines and exit OP-TM-EOP-008.

D. (1) Open the PORV block valve and the PORV to reduce RCS pressure.
(2) Initiate ER-TM-TSC-0010, Severe Accident Guidelines and remain in OP-TM-EOP-008.

Answer: A

Answer Explanation

A. Correct.

Part 1 is correct. By plotting an Incore Thermocouple temperature of 1610°F and an RCS pressure of 800 psig on Fidure 2, RCS Superheat, of OP-TM-EOP-010, Emergency Procedure Rules Guides and Graphs, the candidate will determine that the plant is beyond Curve C and is the Severe Accident Region. OP-TM-EOP-008, RCS Superheated, routing takes the Control Room Supervisor to Section 5.0, Clad Temperature Above 1800F. Section 5 directs securing of Reactor Coolant Pumps and does not mention operation of the PORV. IAW OP-TM-EOP-008:

> 5.0 CLAD TEMPERATURE ABOVE 1800F
> - Unit Status: Incore Thermocouple Temperature Beyond Curve C (Clad > 1800F).
>
> 5.1 TRIP all RC Pumps.

Part 2 is correct. OP-TM-EOP-008 Step 5.2 is met, as given in the question stem. Step 5.3 directs the Control Room Supervisor to Go To the Severe Accident Guidelines procedure. IAW OP-TM-EOP-008:

> 5.2 VERIFY TSC is manned.
> 5.3 GO TO ER-TM-TSC-0010, "Severe Accident Guidelines".

Additionally, the basis document gives further clarification by stating that OP-TM-EOP-008 is exited. IAW OP-TM-EOP-0081, RCS Superheated Basis Document:

> Step 5.3 The step intent is to Exit the Emergency Operating Procedures, and transfer control and management of the event to the Technical Support Center (TSC).

The Emergency Operating Procedures have failed to prevent core damage. Event management with a damaged core is outside the scope of the Emergency Operating Procedures. The Severe Accident Management Guidelines (SAMG) are specifically designed to manage these types of events.

B. Incorrect.

Part 1 is correct. By plotting an Incore Thermocouple temperature of 1610°F and an RCS pressure of 800 psig on Figure 2, RCS Superheat, of OP-TM-EOP-010, Emergency Procedure Rules Guides and Graphs, the candidate will determine that the plant is beyond Curve C and is the Severe Accident Region. OP-TM-EOP-008, RCS Superheated, routing takes the Control Room Supervisor to Section 5.0, Clad Temperature Above 1800F. Section 5 directs securing of Reactor Coolant Pumps and does not mention operation of the PORV. IAW OP-TM-EOP-008:

> 5.0 CLAD TEMPERATURE ABOVE 1800F
> - Unit Status: Incore Thermocouple Temperature Beyond Curve C (Clad > 180F).
>
> 5.1 TRIP all RC Pumps.

Part 2 is incorrect but plausible if the candidate does not recognize that the accident has progressed beyond the scope of the Emergency Operating Procedures. However, the basis document for OP-TM-EOP-008 gives clarification by stating that OP-TM-EOP-008 is exited. IAW OP-TM-EOP-0081, RCS Superheated Basis Document:

> Step 5.3 The step intent is to Exit the Emergency Operating Procedures, and transfer control and management of the event to the Technical Support Center (TSC).

The Emergency Operating Procedures have failed to prevent core damage. Event management with a damaged core is outside the scope of the Emergency Operating Procedures. The Severe Accident Management Guidelines (SAMG) are specifically designed to manage these types of events.

C. Incorrect.
Part 1 is incorrect. Plausible if the candidate reverses the axis on Figure 2, RCS Superheat, of OP-TM-EOP-010, Emergency Procedure Rules Guides and Graphs. By doing this, the candidate will determine that the cladding is in Region 2, which is above 25F Superheat, but not past either Curve B or Curve C. That would place the Control Room Supervisor in Section 3.0 of OP-TM-EOP-008. Step 3.8 gives guidance to lower RCS pressure. IAW OP-TM-EOP-008:

3.8 IAAT RCS pressure reaches the PORV setpoint, then perform the following:
1. ENSURE PORV block (RC-V-2) is Open.
2. OPEN the PORV (RC-RV-2).
3. When RCS pressure reduces to either condition:
   – 100 psig > highest OTSG pressure,
   – 100 psig > next superheat boundary (Curve B or C EOP-010 Figure 2),

then CLOSE the PORV.

Part 2 is correct. OP-TM-EOP-008 Step 5.2 is met, as given in the question stem. Step 5.3 directs the Control Room Supervisor to Go To the Severe Accident Guidelines procedure. IAW OP-TM-EOP-008:

5.2 VERIFY TSC is manned.
5.3 GO TO ER-TM-TSC-0010, "Severe Accident Guidelines".

Additionally, the basis document gives further clarification by stating that OP-TM-EOP-008 is exited. IAW OP-TM-EOP-0081, RCS Superheated Basis Document:

Step 5.3 The step intent is to Exit the Emergency Operating Procedures, and transfer control and management of the event to the Technical Support Center (TSC).

The Emergency Operating Procedures have failed to prevent core damage. Event management with a damaged core is outside the scope of the Emergency Operating Procedures. The Severe Accident Management Guidelines (SAMG) are specifically designed to manage these types of events.

D. Incorrect.
Part 1 is incorrect. Plausible if the candidate reverses the axis on Figure 2, RCS Superheat, of OP-TM-EOP-010, Emergency Procedure Rules Guides and Graphs. By doing this, the candidate will determine that the cladding is in Region 2, which is above 25F Superheat, but not past either Curve B or Curve C. That would place the Control Room Supervisor in Section 3.0 of OP-TM-EOP-008. Step 3.8 gives guidance to lower RCS pressure. IAW OP-TM-EOP-008:

3.8 IAAT RCS pressure reaches the PORV setpoint, then perform the following:
1. ENSURE PORV block (RC-V-2) is Open.
2. OPEN the PORV (RC-RV-2).
3. When RCS pressure reduces to either condition:
   – 100 psig > highest OTSG pressure,
   – 100 psig > next superheat boundary (Curve B or C EOP-010 Figure 2),

then CLOSE the PORV.

Part 2 is incorrect but plausible if the candidate does not recognize that the accident has progressed beyond the scope of the Emergency Operating Procedures. However, the basis document for OP-TM-EOP-008 gives clarification by stating that OP-TM-EOP-008 is exited. IAW OP-TM-EOP-0081, RCS Superheated Basis Document:

> Step 5.3 The step intent is to Exit the Emergency Operating Procedures, and transfer control and management of the event to the Technical Support Center (TSC).

The Emergency Operating Procedures have failed to prevent core damage. Event management with a damaged core is outside the scope of the Emergency Operating Procedures. The Severe Accident Management Guidelines (SAMG) are specifically designed to manage these types of events.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	011	EA2.01
	Importance Rating		4.7

Ability to determine or interpret the following as they apply to a Large Break LOCA: Actions to be taken based on RCS temperature and pressure - saturated and superheated.

Proposed Question: SRO Question # 1 (#76)

Technical Reference(s): OP-TM-EOP-008, p 11, Rev 8
OP-TM-EOP-010, p 40, Rev 17

Proposed References to be provided to applicants during examination: OP-TM-EOP-010 Fig 2

Learning Objective: EOP008-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

55.43 5

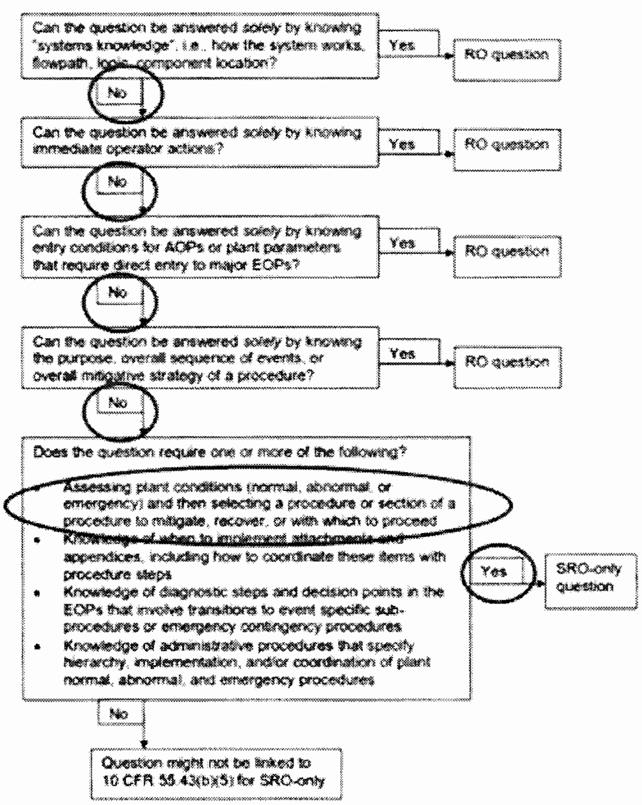
Comments:
The KA is matched because the question requires the candidates to demonstrate the ability to determine the correct action to take, based on RCS pressure and temperature (superheated conditions), during a Large Break Loss of Coolant Accident.

The question is at the Comprehension/Analysis cognitive level because the candidate must analyze the given information, interpret data on a graph. and then determine the correct course of action to take.

The question is at the SRO level because the candidates must assess plant emergency conditions and then select a section of a procedure with which to proceed. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 8). In this case, The Emergency Operating Procedures have failed to prevent core damage. Event management with a damaged core is outside the scope of the Emergency Operating Procedures. The Severe Accident Management Guidelines (SAMG) are specifically designed to manage these types of events.

What MUST be known:
1. How to interpret the graph labelled OP-TM-EOP-010 Fig 2.
2. What is the appropriate section of OP-TM-EOP-008 to enter based on plant conditions?
3. How is the transition from OP-TM-EOP-008 to ER-TM-TSC-0010 accomplished?

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



77 **ID: 1151747** **Points: 1.00**

Plant Conditions:

- The plant is operating at 45% reactor power.
- The "D" Reactor Coolant Pump, RC-P-1D, is secured.
- "A" and "C" Nuclear Services Closed Cooling Water Pumps, NS-P-1A and NS-P-1C, are operating.

Event:

- The Nuclear Services Cooling Water line to the "B" Reactor Coolant Pump, RC-P-1B, has developed a leak at the "D" ring wall.
- DW-P-1 has been started.
    - o NS surge tank level is stable at 6 feet.
    - o NS cooler outlet temperatures have not changed.
    - o RC-P-1B stator temperature has exceeded the HI-1 setpoint and is rising.

Given the above information, secure RC-P-1B ___(1)___ is the correct procedural action to take, and once secured, the required Tech Spec LCO action is to place the reactor in ___(2)___.

A. (1) at the HI-2 alarm setpoint IAW OP-TM-PPC-A0710, RC-P-1B Stator Temp (Deg C)
(2) Hot Standby within the next 24 hours

B. (1) at the HI-2 alarm setpoint IAW OP-TM-PPC-A0710, RC-P-1B Stator Temp (Deg C)
(2) Hot Shutdown within the next 36 hours

C. (1) immediately IAW OP-TM-AOP-031, Loss of Nuclear Services Component Cooling
(2) Hot Standby within the next 24 hours

D. (1) immediately IAW OP-TM-AOP-031, Loss of Nuclear Services Component Cooling
(2) Hot Shutdown within the next 36 hours

Answer: B

Answer Explanation

A. Incorrect.

Part 1 is correct. IAW OP-TM-PPC-A0710, RC-P-1B Stator Temp (Deg C):

1.0 SETPOINTS
- HI-2 Stator Temp (DegC): 150°C

4.0 MANUAL ACTIONS REQUIRED
- IAAT RC-P-1B Stator Temperature 150°C, then PERFORM OP-TM-226-152 to place RC-P-1B in Standby mode.

Part 2 is incorrect. Plausible if the candidate recognizes the 24 hour period associated with having one Reactor Coolant Pump secured in each loop. However, the 24 hour period is the time frame before Tech Spec action must be taken. IAW T.S. 3.1.1.1:

3.1.1.1 Reactor Coolant Pumps

b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

B. Correct.

Part 1 is correct. IAW OP-TM-PPC-A0710, RC-P-1B Stator Temp (Deg C):

1.0 SETPOINTS
- HI-2 Stator Temp (DegC): 150°C

4.0 MANUAL ACTIONS REQUIRED
- IAAT RC-P-1B Stator Temperature 150°C, then PERFORM OP-TM-226-152 to place RC-P-1B in Standby mode.

Part 2 is correct. With the "D" Reactor Coolant Pump secured, that leaves only the "C" Reactor Coolant Pump running in the "B" RCS loop. Since Reactor Power is already less than 49%, RC-P-1B can be secured. Once the "B" Reactor Coolant Pump is secured, that will leave only the "A" Reactor Coolant Pump running in the "A" RCS loop. Tech Specs address the action to take in this case. IAW T.S. 3.1.1.1:

3.1.1.1 Reactor Coolant Pumps

b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

C. Incorrect.
Part 1 is incorrect. Although there is an issue with the Nuclear Services Closed Cooling Water System, the entry criteria for OP-TM-AOP-031, Loss of Nuclear Services Component Cooling, is not met. Two NSCCWW pumps are operating, NS cooler outlet temperature has not changed, and the NSCCW surge tank is holding steady with the surge tank makeup valve open. Plausible if the candidate is not familiar with the entry criteria of OP-TM-AOP-031. IAW OP-TM-AOP-031:

- ENTRY CONDITIONS
  - The OTSGs are being used for RCS heat removal, and at least one secondary river or nuclear river pump is available, and any of the following:
  - NS cooler outlet temperature approaching 100 °F
  - Less than two NS pumps operating and RCP motor or bearing temperature approaching HI-2 alarm set point
  - NS surge tank level less than 1.6 ft. and lowering

Part 2 is incorrect. Plausible if the candidate recognizes the 24 hour period associated with having one Reactor Coolant Pump secured in each loop. However, the 24 hour period is the time frame before Tech Spec action must be taken. IAW T.S. 3.1.1.1:

3.1.1.1 Reactor Coolant Pumps
b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

D. Incorrect.
Part 1 is incorrect. Although there is an issue with the Nuclear Services Closed Cooling Water System, the entry criteria for OP-TM-AOP-031, Loss of Nuclear Services Component Cooling, is not met. Two NSCCWW pumps are operating, NS cooler outlet temperature has not changed, and the NSCCW surge tank is holding steady with the surge tank makeup valve open. Plausible if the candidate is not familiar with the entry criteria of OP-TM-AOP-031. IAW OP-TM-AOP-031:

- ENTRY CONDITIONS
  - The OTSGs are being used for RCS heat removal, and at least one secondary river or nuclear river pump is available, and any of the following:
  - NS cooler outlet temperature approaching 100 °F
  - Less than two NS pumps operating and RCP motor or bearing temperature approaching HI-2 alarm set point
  - NS surge tank level less than 1.6 ft. and lowering

Part 2 is correct. With the "D" Reactor Coolant Pump secured, that leaves only the "C" Reactor Coolant Pump running in the "B" RCS loop. Since Reactor Power is already less than 49%, RC-P-1B can be secured. Once the "B" Reactor Coolant Pump is secured, that will leave only the "A" Reactor Coolant Pump running in the "A" RCS loop. Tech Specs address the action to take in this case. IAW T.S. 3.1.1.1:

3.1.1.1 Reactor Coolant Pumps
b. Power operation with one idle reactor coolant pump in each loop shall be restricted to 24 hours. If the reactor is not returned to an acceptable RC pump operating combination at the end of the 24-hour period, the reactor shall be in a hot shutdown condition within the next 12 hours.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	015	AA2.09
	Importance Rating		3.5

Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on high stator temperatures.

Proposed Question: SRO Question # 2 (#77)

Technical Reference(s): OP-TM-PPC-0710, p 1, Rev 1
Tech Spec, p 3-1a, Amend 279

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-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

55.43

Comments:
The KA is matched because the question requires the candidates to demonstrate the ability to determine when to secure RCPs on a high stator temperature.

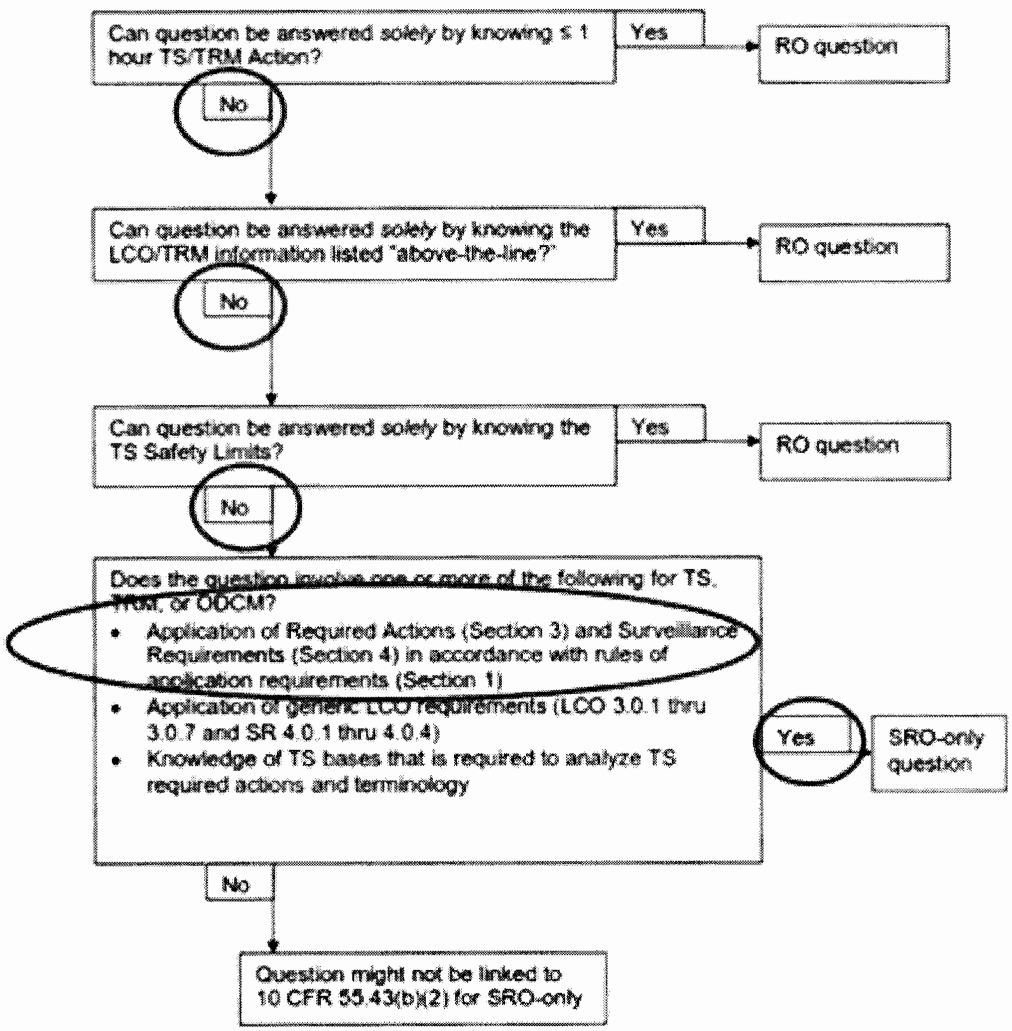
The question is at the Memory/Fundamental Knowledge cognitive level because the candidate must recognize procedural action to take based on conditions given and must also recall the correct tech spec action to take.

Although part 1 can be answered by an RO, the question is at the SRO level because the candidates must demonstrate the ability to apply generic Limiting Condition for Operation (LCO) requirements. RO's are expected to know the LCO statements and associated applicability information, but the question is specifically testing the correct action to take and the associated timeframe (Source: Clarification Guidance for SRO-only Questions, Rev 1, Pages 3-4).

What MUST be known:
1. What is the correct procedural guidance to take for a rising motor stator temperatures for a RCP? 2. What is the appropriate LCO action and timeframe given a RCP combination of 1 RCP secured in each loop?

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

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



78 **ID: 1151187** **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 the STA.
    - The STA verifies Subcooling Margin data on Panel Center Left (PCL) and on the Plant Processing Computer (PPC).

- Time = 90 seconds:
  - OP-TM-EOP-001, Reactor Trip, Immediate Manual Actions are the **ONLY** actions that have been completed.
  - The STA informs the CRS that there was a Loss of Subcooling Margin 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	535°F	Lowering
Loop B T-Cold	536°F	Lowering
Loop A T-Ave	541°F	Lowering
Loop B T-Ave	542°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 must 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, Conduct of Operations During Abnormal and Emergency Events:

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 is not the highest priority identified in the Symptom Check. 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.

- IAW OP-TM-EOP-0011, Reactor Trip Basis Document:
  - "Symptom Check": Plant conditions must be evaluated to make a proper SCM determination. RC pump status, RCS cooldown rate, and RB pressure must be assessed to determine the appropriate method and instruments to determine SCM. (Reference OS-24 Section 4.7).

  - If SCM is less than 25F, then RULE 1 is performed and EOP is initiated. Loss of SCM immediately after a reactor trip is indicative of a significant (several hundred GPM) LOCA.

  - An excessive primary to secondary heat transfer (XHT) event is one where the RCS cooldown is a significant challenge to plant control. XHT can be confirmed if ALL of the following conditions exist:
    - RCS average temperature below 540 °F
    - Uncontrolled lowering of RCS temperature
    - Tsat for OTSG pressure is less than Tcold on affected OTSG(s)

  - RCS average temperature following a reactor trip should be about 555F. Sluggish FW control or delays in reseating of MSSVs should not result in an XHT event (and require isolation of the affected OTSG).

- The definition is intended to be subjective and allow some judgment as to whether the reduction in RCS temperature is "uncontrolled".

- Lack of primary to secondary heat transfer (LOHT) can be confirmed if one of the following sets of conditions exists:
  - "Incore temperatures rising above 580°F and at least one RC Pump operating".
  - "Incore temperatures rising and no feedwater available".
    - ("No feedwater available" means the immediate attempts to establish main or emergency feedwater from the control room have not been successful.)
  - "Incore temperatures rising and RCS circulation can not be confirmed"

- Any time all RCPs are off, incore temperature is rising and natural circulation cannot be confirmed IAW Guide 10, entry into EOP-004 is appropriate.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	E02	EA2.1
Importance Rating		4.0

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 # 3 (#78)

Technical Reference(s):
OS-24. p 8, Rev 25
OP-TM-EOP-001, p 5, Rev 12
OP-TM-EOP-0011, p 9, Rev 5

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP001-PCO-5

Question Source: Bank # ID# 1147367

Modified Bank #

New

Question History: Last NRC Exam: 2014

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 candidate 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).

What MUST be known:
1. What are the required steps to take upon a Loss of Subcooling Margin, given that conditions have cleared? 2. Given multiple applicable procedures, determine highest priority.

79 **ID: 1148869** **Points: 1.00**

Plant Conditions (Time = 2025):

- The plant is operating at 100% reactor power.

Sequence of Events:

- T=2030:
  - A small break LOCA has occurred.
  - The leak has been estimated to be at or beyond the capacity of the High Pressure Injection Pumps.
  - OP-TM-EOP-001, Reactor Trip, has been entered.
  - OP-TM-EOP-002, Loss of 25 Degrees F Subcooling Margin, has been entered.

- T=2034:
  - OP-TM-EOP-008, RCS Superheated, has been entered.
  - Containment Pressure reaches it's maximum value of 35 psig.

- T=2040:
  - OP-TM-EOP-002, Loss of 25 Degrees F Subcooling Margin, has been re-entered.

- T=2045:
  - RM-G-22, High Range Containment Monitor, is reading 200 R/hr.
  - RM-G-23, High Range Containment Monitor, is reading 205 R/hr.
  - RCITS hot leg instruments have lowered to 0 inches.
  - In-core thermocouples are trending with Th instrumentation.

Based on the above information, which one of the following is the highest EAL that is in effect at T=2045?

A. FU1

B. FA1

C. FS1

D. FG1

Answer: C

Answer Explanation

A. Incorrect.

FU1 is declared when ANY Loss or ANY Potential Loss of–Containment occurs. This can only be declared upon one of the following conditions:

	Loss	Potential Loss
Containment Pressure	1. Rapid unexplained drop in Containment pressure following initial pressure rise. OR 2. Containment pressure or water level response not consistent with LOCA conditions.	3. Containment pressure > 55 psig and rising. OR 4. Hydrogen concentration in Containment >/=4% OR 5. a. Containment pressure > 30 psig. AND b. RB Emergency Cooling is less than any one of the following conditions: SPRAY COOLERS 2 0 0 3 1 1
CETC Reading		1. Tclad > 1800F AND 2. EOP restoration procedures are not effective within 15 minutes.
S/G Leakage / Rupture	1. Primary-to-Secondary leakrate > 10 gpm. AND 2. UNISOLABLE steam release from affected S/G to the environment.	
Containment Isolation Valve Status	1. Failure of isolation valves in any one line to close. AND 2. Direct downstream pathway to the environment exists after a containment isolation signal.	
Containment Rad Monitoring		Containment radiation (RM-G-22 or RM-G-23) reading > 4.40E+03 R/hr

None of the above conditions exist. Plausible if the candidate is not familiar with the Fission Product Barrier Matrix.

B. Incorrect.

FA1 is declared when ANY Loss or ANY Potential Loss of either Fuel Clad or RCS occurs. Although conditions are met for FA1, it is not the highest EAL met and therefore is incorrect. Plausible if the candidate does not recognize that FS1 is also applicable.

C. Correct.
FS1 is declared when a Loss or Potential Loss of any TWO barriers occurs. This EAL is met using the following criteria:

- Containment: None
- RCS: Loss (RCS Leakage results in <25F Sub-Cooled Margin) (OP-TM-EOP-002 was entered OR Containment radiation (RM-G-22 or RM-G-23) reading >25 R/hr)
- Fuel Clad: Potential (> 25F Super Heat)

D. Incorrect.
FG1 is declared when a Loss of ANY Two Barriers AND Loss or Potential Loss of third barrier. occurs. Since the Containment Barrier has not been affected, this EAL cannot be met. Plausible if the candidate recognizes the FS1 criteria and also believes that the Containment Barrier also has a Loss or Potential Loss occurrance.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	009	2.4.41
	Importance Rating		4.6

Knowledge of the emergency action level thresholds and classifications.

Proposed Question: SRO Question # 4 (#79)

Technical Reference(s): EP-AA-1009 Addendum 3, Page 2-3, Rev 0

Proposed References to be provided to applicants during examination: EP-AA-1009 Addendum 3, Page 2-3

Learning Objective: EOP001-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	6

Comments:
The KA is matched because the candidates must demonstrate knowledge of the proper emergency action level threshold and classification during a small break LOCA.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to analyze the plant conditions and interpret the given data table to determine the highest EAL in effect.

The question is at the SRO level because the candidates must evaluate core conditions and emergency classifications based on core conditions.(Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 9). Additionally, it is an SRO-only responsibility to declare an EAL.

What MUST be known:
1. How to interpret the table labeled EP-AA-1009 Addendum 3, Page 2-3. 2. What is the appropriate EAL to declare based on plant conditions?

80 **ID: 1150790** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- A Loss of Vital Bus D occurs.

Given the above information and IAW Technical Specification Table 3.5-3, Post Accident Monitoring Instrumentation, the loss of ___(1)___ will require that ___(2)___ if it is not operable within 7 days.

A. (1) RM-G-26, A OTSG Turbine Bypass Line Rad Monitor
(2) the plant be placed in Hot Shutdown within 12 hours

B. (1) RM-G-26, A OTSG Turbine Bypass Line Rad Monitor
(2) a Special Report submitted to the NRC within 30 days

C. (1) WDL-LT-807, Reactor Building Flood Level Transmitter
(2) the plant be placed in Hot Shutdown within 12 hours

D. (1) WDL-LT-807, Reactor Building Flood Level Transmitter
(2) a Special Report submitted to the NRC within 30 days

Answer: B

Answer Explanation

A. Incorrect.

Part 1 is correct. RM-G-26 is required per Tech Spec Table 3.5-3.

Part 2 is incorrect.
If the Minimum Number of Channels column was not satisfied, then the correct action statement is Action "A". Plausible if the candidate is not familiar with which action statement is associated with each component listed in Tech Spec Table 3.5-3. IAW Tech Spec Table 3.5-3:

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.

B. Correct.

Part 1 is correct. RM-G-26 is required per Tech Spec Table 3.5-3:

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

Part 2 is correct.
According to Tech Spec Table 3.5-3, both the number of channels and the minimum number of operable level transmitters for Main Steam Line Radiation Instruments is one per OTSG. Therefore, RM-G-26 being inoperable means that the requirement for the minimum number of channels is not met and Action Statement "A" is applicable. IAW Tech Spec Table 3.5-3:

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.

C. Incorrect.
Part 1 is incorrect.
Although listed in Tech Spec Table 3.5-3, this selection is incorrect. There are two RB Flood Level instruments (WDL-LT-806 and WDL-LT-807) and only 1 of those two are required to be operational. Since WDL-LT-806 is powered from Vital Bus "C" and nothing in the stem indicated that Vital Bus "C" has lost power, Tech Spec Table 3.5-3 is satisfied for Containment Flood Water Level instrumentation.

Part 2 is incorrect.
If the Minimum Number of Channels column was not satisfied, then the correct action statement is Action "A". However, Action "B" is associated with WDL-LT-807. Plausible if the candidate is familiar with the Tech Spec Table 3.5-3 action statement associated with WDL-LT-807. IAW Tech Spec Table 3.5-3:

- B.
  1. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Channels OPERABLE requirements, restore the inoperable channel(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.
  2. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements, restore the inoperable channels(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

D. Incorrect.
Part 1 is incorrect. Although listed in Tech Spec Table 3.5-3, this selection is incorrect. There are two RB Flood Level instruments (WDL-LT-806 and WDL-LT-807) and only 1 of those two are required to be operational. Since WDL-LT-806 is powered from Vital Bus "C" and nothing in the stem indicated that Vital Bus "C" has lost power, Tech Spec Table 3.5-3 is satisfied for Containment Flood Water Level instrumentation.

Part 2 is correct. Although correct, it is only the correct action for RM-G-26, not for WDL-LT-807. Plausible if the candidate is not familiar with which action statement is associated with each component listed in Tech Spec Table 3.5-3. IAW Tech Spec Table 3.5-3:

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.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	057	2.4.30
Importance Rating		4.1

Loss of Vital AC Inst. Bus: 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: SRO Question # 5 (#80)

Technical Reference(s): Tech Spec, p 3-40d, Amend 240
Tech Spec, p 3-40e, Amend 166

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-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	
	55.43	2

Comments:
The KA is matched because the candidates must demonstrate knowledge of an event associated with a loss of a vital AC Bus that must be reported to an external agency.

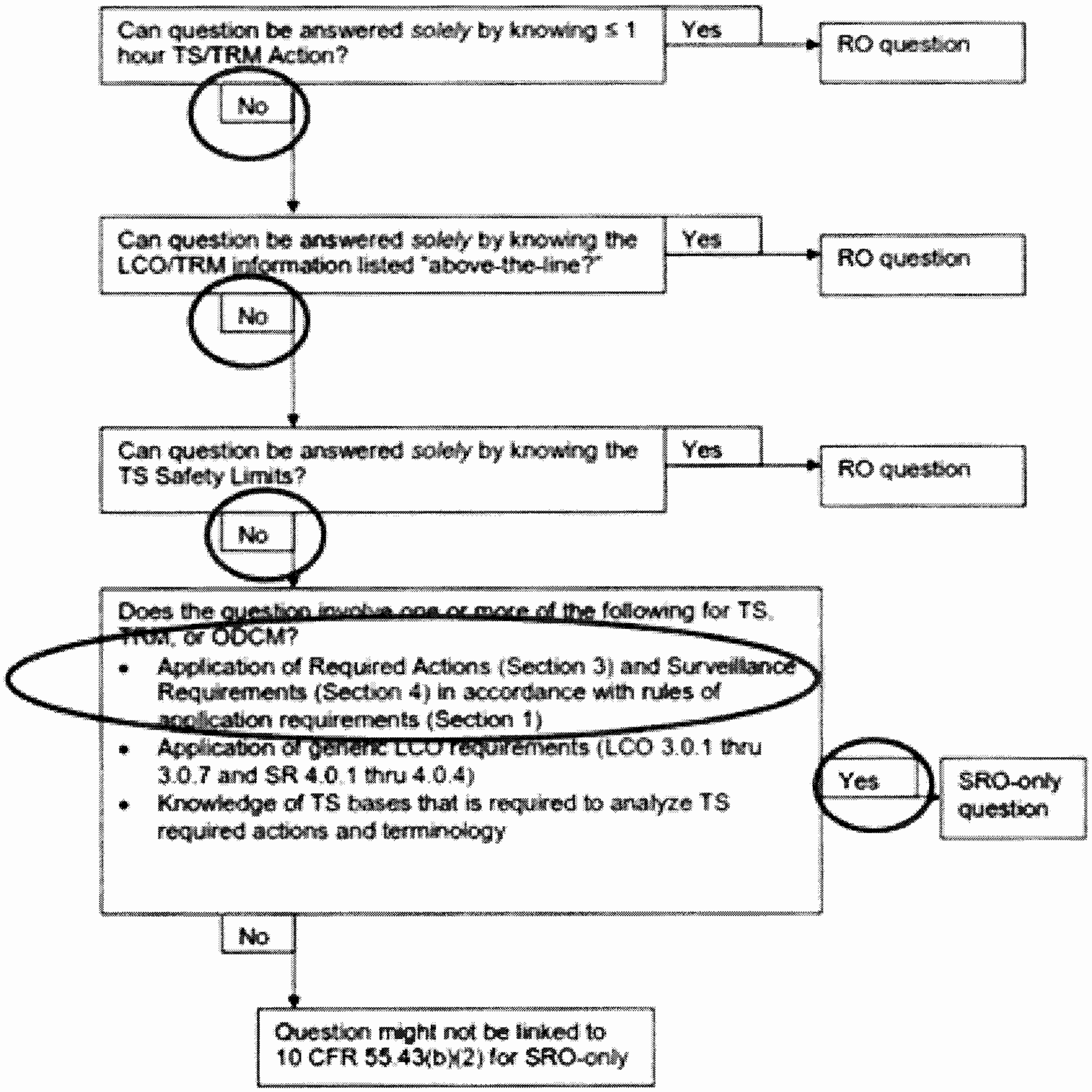
The question is at the Memory/Fundamental Knowledge cognitive level because the candidate is required to identify equipment lost on a loss of a vital AC Bus and then recall what the reporting criteria is.

Although part 1 can be answered by an RO, the question is at the SRO level because the candidates must demonstrate the ability to apply generic Limiting Condition for Operation (LCO) requirements. RO's are expected to know the LCO statements and associated applicability information, but the question is specifically testing the correct action to take and the associated timeframe (Source: Clarification Guidance for SRO-only Questions, Rev 1, Pages 3-4).

What MUST be known:
1. What components violate the minimum number of channels column of Tech Spec Table 3.5-3 upon a loss of VBD? 2. What is the reportability action statement associated with RM-G-26?

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

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



81 **ID: 1147417** **Points: 1.00**

Plant Conditions (Time = 0 minutes):

- The plant is operating at 100% reactor power.
- NS-P-1B is powered from the 1S 480V Bus.

Sequence of Events:

- Time = 10 minutes:
  - MAP A-2-8, BATT CHGR 1B/D/1F TROUBLE, actuates.
  - MAP A-1-8, BATTERY 1B DISCHARGING, actuates.
  - The AO at the battery chargers reports that there are no chargers are operating.

- Time = 17 minutes:
  - The crew enters AOP-024, "B" DC System Failure.

- Time = 25 minutes:
  - The AO at the battery chargers reports that 1B Battery Bus Voltage at the distribution panel is 119 VDC and slowly lowering.

Given the above information, the CRS must direct that ___(1)___ be initiated at Time = 25 minutes in order to ___(2)___.

A. (1) OP-TM-732-922, Transfer DCB to DC Diesel
(2) prevent a loss of protective relaying

B. (1) OP-TM-732-922, Transfer DCB to DC Diesel
(2) restore power to the "B" Train battery chargers

C. (1) OP-TM-732-902, Energize 1S 480V Bus Using ES Bus Cross Tie
(2) prevent a loss of protective relaying

D. (1) OP-TM-732-902, Energize 1S 480V Bus Using ES Bus Cross Tie
(2) restore power to the "B" Train battery chargers

Answer: A

Answer Explanation

A. **Correct.**

Part 1 is correct. According to OP-TM-AOP-024, "B" DC System Failure, the operator will be directed to Initiate OP-TM-734-922, Transfer DCB to DC Diesel. IAW OP-TM-AOP-024:

3.13 INITIATE OP-TM-734-922, "Transfer DCB to DC Diesel."

Part 2 is correct. IAW OP-TM-AOP-0241, "B" DC System Failure Basis Document:

- This step transfers DCB to the DC Diesel bus. This reduces the load on the "B" DC system. In addition, preemptively transferring DCB to a reliable power source prior to loss of 1B DC Distribution panel will prevent loss of protective relaying powered from DCB.

B. Incorrect.

Part 1 is correct. According to OP-TM-AOP-024, "B" DC System Failure, the operator will be directed to Initiate OP-TM-734-922, Transfer DCB to DC Diesel. IAW OP-TM-AOP-024:

3.13 INITIATE OP-TM-734-922, "Transfer DCB to DC Diesel."

Part 2 is incorrect. Plausible if the candidate confuses the reason for transferring DCB with that of energizing 1S 480V Bus. IAW OP-TM-AOP-0241, "B" DC System Failure Basis Document:

- This step transfers DCB to the DC Diesel bus. This reduces the load on the "B" DC system. In addition, preemptively transferring DCB to a reliable power source prior to loss of 1B DC Distribution panel will prevent loss of protective relaying powered from DCB.

Additionally:

If the reactor has been shut down and there is not a loss of the 1D 4160V bus, then the 1P and the 1S 480V busses can be cross tied to restore power to the battery chargers.

C. Incorrect.

Part 1 is incorrect. This is plausible because it the procedure step comes before the correct step, but this would only be correct if the reactor is shutdown. IAW OP-TM-AOP-024:

3.5 IAAT all of the following:
- o Reactor is shut down
- o 1E 4160V Bus is not energized
- o 1D 4160V Bus is energized

then INITIATE OP-TM-732-902, "Energize 1S 480V Bus Using ES Bus Cross Tie."

Part 2 is correct. IAW OP-TM-AOP-0241, "B" DC System Failure Basis Document:

- This step transfers DCB to the DC Diesel bus. This reduces the load on the "B" DC system. In addition, preemptively transferring DCB to a reliable power source prior to loss of 1B DC Distribution panel will prevent loss of protective relaying powered from DCB.

D. Incorrect.
Part 1 is incorrect. This is plausible because it the procedure step comes before the correct step, but this would only be correct if the reactor is shutdown. IAW OP-TM-AOP-024:

3.5 IAAT all of the following:
- o Reactor is shut down
- o 1E 4160V Bus is not energized
- o 1D 4160V Bus is energized

then INITIATE OP-TM-732-902, "Energize 1S 480V Bus Using ES Bus Cross Tie."

Part 2 is incorrect. Plausible if the candidate correctly remembers the reason for of energizing 1S 480V Bus. This would be the correct Part 2 if Part 1 was correct. IAW OP-TM-AOP-0241, "B" DC System Failure Basis Document:

- If the reactor has been shut down and there is not a loss of the 1D 4160V bus, then the 1P and the 1S 480V busses can be cross tied to restore power to the battery chargers.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	058	2.4.11
	Importance Rating		4.2

Loss of DC Power: Knowledge of abnormal condition procedures.

Proposed Question: SRO Question # 6 (#81)

Technical Reference(s): OP-TM-AOP-024, p 9, Rev 4
OP-TM-AOP-0241, p 8, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP024-PCO-2

Question Source: Bank #
Modified Bank # ID# 860029
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 operator must demonstrate the a ability to determine and interpret the 125V dc bus voltage, low/critical low, alarm as they apply to the Loss of DC Power (i.e. by requiring knowledge of the response associated with the only MAP alarm dealing with Vital DC voltage).

The question is at the Comprehension/Analysis cognitive level because the operator must evaluate the stated conditions, and determine the status of the DC Distribution System. Once determined the operator must 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 SRO-Only because the candidate 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). There are numerous setpoints, procedures, and IAAT statements in the AOP that must be understood and correctly implemented to deal with the degrading DC voltage situation, indicating that the candidate must be able to recall a strategy within a procedure. Additionally, the candidate must demonstrate the basis for the correct step (from the associated basis document).

What MUST be known:
1. What are the steps listed within OP-TM-AOP-024? 2. Which is the applicable actions within OP-TM-AOP-024 for a given battery voltage? 3. What is the basis for applicable action?

Original Question:

Plant conditions:

The plant is operating at 100% power.

Sequence of Events:

MAP A-2-8, BATT CHGR 1B/D/1F TROUBLE, actuated 15 minutes ago.
The AO dispatched reports no chargers are operating.
MAP A-1-8, BATTERY 1B DISCHARGING, actuated 8 minutes ago.
The AO dispatched reports 1B Battery Bus Voltage at the distribution panel is 119 VDC.
The crew has entered AOP-024, "B" DC System Failure.

Which ONE of the following describes the required action?

A. Initiate OP-TM-734-922, "Transfer DCB to DC Diesel" to reduce battery loading.

B. Isolate Battery 1B and start the 8 hour clock for one battery out-of-service in accordance with TS 3.7.2.g.

C. Initiate OP-TM-732-902, "Energize 1S 480V Bus Using ES Bus Cross Tie" to provide power for a charger restart.

D. Declare EG-Y-1B inoperable and perform the TS 3.7.2.c actions for one inoperable emergency diesel generator.

Answer: A

82 **ID: 1147528** **Points: 1.00**

Plant Conditions:

- The plant is in a refueling outage.
- Fuel offload is in progress in the Reactor Building.
- Fuel Transfer Canal Isolation Valves, FH-V-1A and FH-V-1B, are open.

Sequence of Events:

- RCS Water level is lowering due to a Spent Fuel Pool cooling valve being out of position.
- The Spent Fuel Pool valve lineup has been corrected.
    - Water level is steady at 22.5' above the Reactor Vessel Flange.

Given the above information, fuel handling:

A. May continue provided the Fuel Transfer Canal is isolated from the Spent Fuel Pool.

B. Must cease due to there being insufficient volume for cooling to maintain temperature less than 140F.

C. May continue as long as temperatures are maintained less than 130F on the outlet of the DHR cooler.

D. Must cease due to there being insufficient volume for iodine retention if a fuel handling accident were to occur.

Answer: D

Answer Explanation

IAW 1301-1, A Fuel Transfer Canal Level of 344 feet equates to 23 feet above the Reactor Vessel Flange:

- Is RCS / FTC level within the required level band? Y / N Y / N
  (FTC > 344' (23 feet above vessel flange) when handling irradiated fuel T.S. 3.8.11)

A. Incorrect.
The fuel handling operations may not continue. If FH-V-1A and FH-V-1B are not open, then minimum level allowed is 343' 6", which equates to 22.5 feet above the Reactor Vessel Flange. However, refueling activities are in progress, so a minimum level of 344', which equates to 22 feet above the Reactor Vessel Flange, is required. Plausible if the candidate is not familiar with the level requirements associated with fuel handling activities and also believes that isolating the Fuel Transfer Canal from the Spent Fuel Pool is sufficient to continue fuel handling activities. IAW 1505-1, Fuel and Control Component Shuffles:

- DATA SHEET 3 - DAILY Fuel Handling Building Checklist
  I. Water level between 343'-6" and 345'.
  a. If refueling (FH-V-1A and 1B open) water level < 344" (normally between 344' and 345') (TS 3.8.11).

B. Incorrect.
Although correct that refueling activities must cease, the reason is incorrect. Plausible if the candidate recalls the RCS temperature associated with OP-TM-AOP-060, Leakage While on Decay Heat Removal. Although this is a temperature of concern, it is not the reason for the minimum water level. IAW OP-TM-AOP-060:

3.9 VERIFY Incore temperature < 140F
RNO: INITIATE "Emergency RCS Cooldown" to < 140F IAW 1102-11. "Plant Cooldown".

Additionally, IAW OP-TM-AOP-0601, Leakage While on Decay Heat Removal Basis Document:

- MITIGATION STRATEGY
- Suspend ongoing activities (i.e. fuel handling, RCS water movements, notify Rad pro for determination on RB or AB evacuation), increase RCS makeup to maintain required water level, ensure RCS temperature is reduced below 140F and closely monitor decay heat system operation.

C. Incorrect.
The fuel handling operations may not continue. If FH-V-1A and FH-V-1B are not open, then minimum level allowed is 343' 6", which equates to 22.5 feet above the Reactor Vessel Flange. However, refueling activities are in progress, so a minimum level of 344', which equates to 22 feet above the Reactor Vessel Flange, is required. Plausible if the candidate is not familiar with the level requirements associated with fuel handling activities and also believes that maintaining DHR outlet temperature less than 130F is sufficient to continue fuel handling activities. IAW 1505-1, Fuel and Control Component Shuffles:

- DATA SHEET 3 - DAILY Fuel Handling Building Checklist
  I. Water level between 343'-6" and 345'.
  a. If refueling (FH-V-1A and 1B open) water level < 344" (normally between 344' and 345') (TS 3.8.11).

D. Correct.
IAW Tech Spec 3.8.11:

> 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 Basis Document:

- 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.

Lastly, IAW TMI-1 UFSAR:

14.2.2.1 Fuel Handling Accident

- The gases released from the fuel assembly pass upward through the spent fuel storage pool water prior to reaching the Fuel Handling Building atmosphere. Normally, the spent fuel assembly rests within the spent fuel storage rack, where it is covered with a minimum of 23 feet of water. Although there is experimental evidence that a portion of the noble gases will remain in the water, no retention of noble gases is assumed. Per Regulatory Guide 1.183, 99.5 percent (or a DCF of 200) of the iodine released from the fuel assembly is assumed to remain in the water.

- The refueling water decontamination factors are based on Regulatory Guide 1.183 assumptions, giving an effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. The depth of water above damaged fuel for a fuel handling accident in the containment is greater than 23 feet.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	036	AA2.02
	Importance Rating		4.1

Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Occurrence of a fuel handling incident.

Proposed Question: SRO Question # 7 (#82)

Technical Reference(s):	Tech Spec, p 3-45, Amend 260 Tech Spec, p 3-45a, Amend 257 TMI-1 UFSAR, p 14.2-1,2, Rev 22

Proposed References to be provided to applicants during examination: None

Learning Objective: 252-GLO-14

Question Source:	Bank #	ID# 375094
	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 operator must interpret the data given and determine that a fuel handling incident has occurred.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must evaluate the stated conditions, and the recall the correct action, and basis for the action, stated within Tech Specs and the FSAR.

The question is SRO-Only because the candidate must provide knowledge of Refuel floor SRO responsibilities. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 9). Additionally, the question separates from RO-only knowledge in the fact that the candidate must recall information from the Tech Spec basis section.

What MUST be known:
1. What is the minimum RCS level allowed during refuel activities?
2. What is the Tech Spec basis description associated with RCS water level?
3. What is the FSAR description associated with RCS water level?

83 **ID: 1151788** **Points: 1.00**

Plant Conditions:

- The plant is operating at 60% reactor power.
- ICS in full Manual control IAW the following procedures:
  - OP-TM-621-471, ICS Manual Control.
  - OP-TM-421-451, Manual Control of Feed Flow to "A" OTSG.
  - OP-TM-421-452, Manual Control of Feed Flow to "B" OTSG.
- You are the on-shift Control Room Supervisor.
- You were relieved by the Shift Manager and are returning to the Control Room from the Operations Office Building (OOB).

Event (Just as you pass through the Control Room entrance door):

- MAP H-1-8 ICS/NNI POWER LOST actuates.
- At Panel PCL:
  - ICS-HAND ICS/NNI Power indicator lamp is NOT lit.
  - SUBFEEDS AUTO/HAND Power indicator lamp is NOT lit.
- The following alarms actuate with no change in alarm related parameters:
  - C-2-7 DH PUMP SUCTION TEMP HI.
  - G-3-8 RC PRESS NARROW RNG HI/LO.
- OP-TM-AOP-026, Loss of ATB or ICS HAND Power, has been entered.
  - Immediate Manual Actions have been completed.

Given the above information, identify the ONE selection below that correctly describes:
(1) The requirements for you to assume a management role in response to the upset, and
(2) The correct CRS responsibility for routing through the applicable procedures.

A. (1) Obtain a short brief from the shift manager PRIOR to directing team activities.
(2) Perform all of the actions within OP-TM-EOP-001 and then Go To and perform all of the actions within OP-TM-AOP-026.

B. (1) Obtain a short brief from the shift manager PRIOR to directing team activities.
(2) Determine the sequence of actions simultaneously between OP-TM-AOP-026 and OP-TM-EOP-001 for overall event mitigation.

C. (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.
(2) Perform all of the actions within OP-TM-EOP-001 and then Go To and perform all of the actions within OP-TM-AOP-026.

D. (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.
(2) Determine the sequence of actions simultaneously between OP-TM-AOP-026 and OP-TM-EOP-001 for overall event mitigation.

Answer: B

Answer Explanation

A. Incorrect.
Part 1 is correct. IAW OS-24, Conduct of Operations During Abnormal And Emergency Events:

> 4.4.2 If a member of the Control Room team is absent when an transient procedure is entered, the team member receives a short briefing from any team member before taking any directed action.

Part 2 is incorrect. Plausible if the candidate believes that there is a hierarchy order regarding Emergency Operating Procedures and Abnormal Operating Procedures. However, the only hierarchy exists within Rules. IAW OS-24, Conduct of Operations During Abnormal And Emergency Events:

> 4.1.4 Rules and Guides
>
> D. Rules are numbered according to priority. If multiple Rule based actions are required, the highest priority Rule is performed first.
>
> 4.1.5 Performing Parallel Procedures
>
> 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

B. Correct.
Part 1 is correct. IAW OS-24, Conduct of Operations During Abnormal And Emergency Events:

> 4.4.2 If a member of the Control Room team is absent when a transient procedure is entered, the team member receives a short briefing from any team member before taking any directed action.

Part 2 is correct. OP-TM-AOP-026, Loss of ATB or ICS Hand Power, entry criteria has been met:

1.0 ENTRY CONDITIONS
- All of the following:
  - The OTSGs are being used for RCS heat removal,
  - ATB or ICS HAND is de-energized,
  - ICS AUTO power is energized,
  - 1D 4160V or 1E 4160V bus is energized.

NOTE: ICS "HAND" Power status is indicated by lights on panel PCL and MAP H-1-8. Loss of ATB is evident by loss of power to OWS, PPC monitors and MAP G-2-6 illuminated.

And then AOP-026 requires the crew to PERFORM EOP-001 IMAS, trip both feedwater pumps, INITIATE OP-TM-424-901 EFW and INITIATE EOP-001:

2.1 IAAT reactor power < 75% and either FW-V-17A or FW-V-17B is in HAND, then perform the following:

1. PERFORM EOP-001, "Reactor Trip" Immediate Actions.
2. TRIP both Main Feedwater pumps:
   - FW-P-1A
   - FW-P-1B
3. INITIATE OP-TM-424-901, "Emergency Feedwater".
4. INITIATE EOP-001, "Reactor Trip".

Once those actions are taken, the CRS will be in both procedures. This scenario is covered in OS-24. IAW OS-24, Conduct of Operations During Abnormal And Emergency Events:

4.1.5 Performing Parallel Procedures

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

C. Incorrect.
Part 1 is incorrect. Plausible if the candidate is not familiar with the responsibilities of IAW OS-24, Conduct of Operations During Abnormal And Emergency Events:

4.4.2 If a member of the Control Room team is absent when an transient procedure is entered, the team member receives a short briefing from any team member before taking any directed action.

Part 2 is incorrect. Plausible if the candidate believes that there is a hierarchy order regarding Emergency Operating Procedures and Abnormal Operating Procedures. However, the only hierarchy exists within Rules. IAW OS-24, Conduct of Operations During Abnormal And Emergency Events:

4.1.4 Rules and Guides

D. Rules are numbered according to priority. If multiple Rule based actions are required, the highest priority Rule is performed first.

4.1.5 Performing Parallel Procedures

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

D. Incorrect.
Part 1 is incorrect. Plausible if the candidate is not familiar with the responsibilities of IAW OS-24, Conduct of Operations During Abnormal And Emergency Events:

> 4.4.2 If a member of the Control Room team is absent when a transient procedure is entered, the team member receives a short briefing from any team member before taking any directed action.

Part 2 is correct. OP-TM-AOP-026, Loss of ATB or ICS Hand Power, entry criteria has been met:

1.0 ENTRY CONDITIONS
- All of the following:
  - The OTSGs are being used for RCS heat removal,
  - ATB or ICS HAND is de-energized,
  - ICS AUTO power is energized,
  - 1D 4160V or 1E 4160V bus is energized.

NOTE: ICS "HAND" Power status is indicated by lights on panel PCL and MAP H-1-8. Loss of ATB is evident by loss of power to OWS, PPC monitors and MAP G-2-6 illuminated.

And then AOP-026 requires the crew to PERFORM EOP-001 IMAS, trip both feedwater pumps, INITIATE OP-TM-424-901 EFW and INITIATE EOP-001:

2.1 IAAT reactor power < 75% and either FW-V-17A or FW-V-17B is in HAND, then perform the following:
1. PERFORM EOP-001, "Reactor Trip" Immediate Actions.
2. TRIP both Main Feedwater pumps:
   - FW-P-1A
   - FW-P-1B
3. INITIATE OP-TM-424-901, "Emergency Feedwater".
4. INITIATE EOP-001, "Reactor Trip".

Once those actions are taken, the CRS will be in both procedures. This scenario is covered in OS-24. IAW OS-24, Conduct of Operations During Abnormal And Emergency Events:

4.1.5 Performing Parallel Procedures
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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	A02	AA2.1
	Importance Rating		4.0

Ability to determine and interpret the following as they apply to the (NNI-X): Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Proposed Question: SRO Question # 8 (#83)

Technical Reference(s): OS-24, p 7, 18, Rev 25
OP-TM-AOP-026, p 1, Rev 5

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP026-PCO-1

Question Source: Bank #

Modified Bank # ID# 371264

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 operator must determine appropriate selection of procedures during abnormal operations associated with a Loss if ICS Hand Control (NNI-X).

The question is at the Comprehensive/Analysis cognitive level because the operator must evaluate the stated conditions, and then decide the appropriate order of procedure steps to take.

The question is SRO-Only because the candidate must demonstrate knowledge of administrative 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).

What MUST be known:
1. How is turnover accomplished if an event occurs while an operator is absent from the Control Room? 2. What are the appropriate actions for a loss of ICS Hand Power?

Original Question:

SRO ONLY
Plant Conditions:

Reactor is operating at 100% power, with ICS in full automatic.
You are the on-shift Control Room Supervisor.
You have been relieved by the Shift Manager and are returning to the Control Room from the Operations Office Building (OOB).

Event (Just as you pass through the Control Room entrance door):

One-half of the overhead lights go out, and do NOT come back on.
You observe the following MAPs actuate:
B-1-1, 4KV ES FDR BKR TRIP.
B-2-1, 4KV ES BUS UV/OV.
B-1-2, 4KV ES MOTOR TRIP.
F-1-5, RCP SEAL TOT INJECT FLOW HI/LO.
F-1-6, RCP PUMP LAB SEAL DP LO.

Based on these conditions, identify the ONE selection below that describes:
(1) The requirements for you to assume a management role in response to the upset, and
(2) The procedure that should be implemented to perform actions that are most critical to mitigation of the event.

A. (1) Obtain a brief update from the shift manager PRIOR to directing team activities.
(2) OP-TM-AOP-014, LOSS OF 1E 4160V BUS.

B. (1) Obtain a brief update from the shift manager PRIOR to directing team activities.
(2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP.

C. (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.
(2) OP-TM-AOP-014, LOSS OF1E 4160V BUS.

D. (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.
(2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP.

Answer: A

84 **ID: 1151188** **Points: 1.00**

Plant Conditions (T=0 minutes):

- The Reactor has been at 100% power for 150 days.

Sequence of Events:

- T=1 minute:
  - A large break LOCA occurs.
  - The reactor has been tripped.
  - One Control Rod (rod 5-3) indicates withdrawn to 50%.
  - Power Range Nuclear Instrumentation reads between ZERO and 1%.
  - Source Range counts are lowering.
- T=5 minutes:
  - Tave has lowered below 525F.
- T=10 minutes:
  - LPI is operating; current flow is 1285 gpm on each train.
  - Control Rod 5-3 has been fully inserted into the core.
  - 1% dk/k shutdown has **NOT** been confirmed.
  - The Unit Reactor Operator requests to terminate Emergency Boration.

Given the above information, Rule 5, Emergency Boration should have been initiated at ___(1)___ and because ___(2)___, Emergency Boration may be terminated at T=10 minutes.

A. (1) T=1 minute
(2) LPI injection flow is supplying adequate chemical boration from the BWST

B. (1) T=1 minute
(2) all control rods have been inserted, indicating a normal Reactor Trip response

C. (1) T=5 minutes
(2) LPI injection flow is supplying adequate chemical boration from the BWST

D. (1) T=5 minutes
(2) all control rods have been inserted, indicating a normal Reactor Trip response

Answer: A

Answer Explanation

A. Correct.

Part 1 is correct. IAW OP-TM-EOP-010, Emergency Procedure Rules, Guides, and Graphs:

- Rule 5: EB - Emergency Boration:
  - 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:

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

- Rule 5: EB - Emergency Boration:
  - Step 1 The step intent is to provide emergency boration termination criteria.
    - Whenever adequate shutdown margin (SDM) is restored, as measured by chemical means and compared to reactivity curves, then boration may be stopped.
    - LPI injection flow of 1250 gpm per line supplies adequate chemical boration from the BWST during a LOCA to satisfy termination of alternate chemical boration methods.
    - RCS Tavg > 525F with complete control rod insertion and proper neutron indication response indicates a normal Reactor trip response.

B. Incorrect.

Part 1 is correct. IAW OP-TM-EOP-010, Emergency Procedure Rules, Guides, and Graphs:

- Rule 5: EB - Emergency Boration:
  - 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:

Part 2 is incorrect. Plausible if the candidate recognizes that all Control Rods have been inserted. However, there are other criteria associated with control rods being inserted that must be met in order to terminate Emergency Boration. IAW OP-TM-EOP-0101, Emergency Procedure Rules, Guides, and Graphs Basis Document:

- Rule 5: EB - Emergency Boration:
  - Step 1 The step intent is to provide emergency boration termination criteria.
    - Whenever adequate shutdown margin (SDM) is restored, as measured by chemical means and compared to reactivity curves, then boration may be stopped.
    - LPI injection flow of 1250 gpm per line supplies adequate chemical boration from the BWST during a LOCA to satisfy termination of alternate chemical boration methods.
    - RCS Tavg > 525F with complete control rod insertion and proper neutron indication response indicates a normal Reactor trip response.

C. Incorrect.

Part 1 is incorrect. Plausible since 525F is a temperature associated with Emergency Boration in many procedures (such as OP-TM-EOP-005) and Shutdown Margin calculations take one stuck control rod into account. However, Rule 5 directs Emergency Boration commencement even if one control rod is stuck upon a reactor shutdown. IAW OP-TM-EOP-010, Emergency Procedure Rules, Guides, and Graphs:

- Rule 5: EB - Emergency Boration:
  - 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:

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

- Rule 5: EB - Emergency Boration:
  - Step 1 The step intent is to provide emergency boration termination criteria.
    - Whenever adequate shutdown margin (SDM) is restored, as measured by chemical means and compared to reactivity curves, then boration may be stopped.
    - LPI injection flow of 1250 gpm per line supplies adequate chemical boration from the BWST during a LOCA to satisfy termination of alternate chemical boration methods.
    - RCS Tavg > 525F with complete control rod insertion and proper neutron indication response indicates a normal Reactor trip response.

D. Incorrect.

Part 1 is incorrect. Plausible since 525F is a temperature associated with Emergency Boration in many procedures (such as OP-TM-EOP-005) and Shutdown Margin calculations take one stuck control rod into account. However, Rule 5 directs Emergency Boration commencement even if one control rod is stuck upon a reactor shutdown. IAW OP-TM-EOP-010, Emergency Procedure Rules, Guides, and Graphs:

- Rule 5: EB - Emergency Boration:
  - 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:

Part 2 is incorrect. Plausible if the candidate recognizes that all Control Rods have been inserted. However, there are other criteria associated with control rods being inserted that must be met in order to terminate Emergency Boration. IAW OP-TM-EOP-0101, Emergency Procedure Rules, Guides, and Graphs Basis Document:

- Rule 5: EB - Emergency Boration:
  - Step 1 The step intent is to provide emergency boration termination criteria.
    - Whenever adequate shutdown margin (SDM) is restored, as measured by chemical means and compared to reactivity curves, then boration may be stopped.
    - LPI injection flow of 1250 gpm per line supplies adequate chemical boration from the BWST during a LOCA to satisfy termination of alternate chemical boration methods.
    - RCS Tavg > 525F with complete control rod insertion and proper neutron indication response indicates a normal Reactor trip response.

Examination Outline Cross-reference:

Level	RO		SRO
Tier #			1
Group #			2
K/A #	024	2.4.6	
Importance Rating			4.7

Emergency Boration : Knowledge of EOP mitigation strategies.

Proposed Question: SRO Question # 9 (#84)

Technical Reference(s): OP-TM-EOP-010, p 10, Rev 17
OP-TM-EOP-0101, p 24, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: EOPR5-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	6

Comments:
The KA is matched because the operator must demonstrate knowledge of EOP mitigation strategy associated with Emergency Boration.

The question is at the Comprehensive/Analysis cognitive level because the operator must evaluate the stated conditions, and then decide when the appropriate procedure step is to be taken.

The question is SRO-Only because the candidate must demonstrate knowledge of Procedures and limitations involved in the determination of various internal and external effects on core reactivity. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 9). Additionally, it is SRO level of knowledge to understand the basis.

What MUST be known:
1. What are the requirements to initiate Emergency Boration? 2. What is the basis for allowing termination of Emergency Boration?

85 **ID: 1151189** **Points: 1.00**

OP-TM-EOP-006, LOCA Cooldown, Step 3.30 states:

> 3.30 VERIFY at least one train of LPI is operating,
> or both CF-V-1A and CF-V-1B are Closed.

Given the above information, which one of the following describes:
(1) The basis for the action within the above step, and
(2) The action to take if the step is not met?

A. (1) To prevent Nitrogen from being injected into the RCS.
(2) Perform a plant cooldown IAW Guide 11, Cooldown Rate Limits.

B. (1) To prevent Nitrogen from being injected into the RCS.
(2) Maintain incore thermocouples above 415°F until the step is met.

C. (1) To support Reactor Vessel Refill and Reflood volumes.
(2) Perform a plant cooldown IAW Guide 11, Cooldown Rate Limits.

D. (1) To support Reactor Vessel Refill and Reflood volumes.
(2) Maintain incore thermocouples above 415°F until the step is met.

Answer: B

Answer Explanation

A. Incorrect.

Part 1 is correct. IAW OP-TM-EOP-0061, LOCA Cooldown Basis Document:

Step 3.30 The step intent is to stop OTSG cooling component of RCS cooldown if low pressure injection sources are not available or a CFT isolation valve was not able to be closed.

- RCS pressure control is based on SCM, and maintaining RCS pressure may not be possible due to leak size. If OTSG cooling is used to augment RCS cooldown, then it is not desirable to intentionally lower RCS pressure below 202 psig, as Core Flood tank nitrogen may be injected into the RCS and interrupt primary to secondary heat transfer.
- Once Core Flood tanks are isolated, or at least one train of LPI is operating, the nitrogen cover gas can not be injected into the RCS interrupting primary to secondary heat transfer. OTSG cooling can be resumed until DHR conditions are established.
- For saturated conditions, the average incore temperature of 389F is a more accurate indication the RCS pressure is greater than 202 psig than direct indication of RCS pressure. 26F is added to 389F to provide the procedural limit of 415F to account conservatively for instrument error.

Part 2 is incorrect. Plausible if the candidate understands the purpose of OP-TM-EOP-006 but does not recognize that performing the cooldown will inject nitrogen into the RCS. IAW OP-TM-EOP-0061, LOCA Cooldown Basis Document:

Step 3.30: If OTSG cooling is used to augment RCS cooldown, then it is not desirable to intentionally lower RCS pressure below 202 psig, as Core Flood tank nitrogen may be injected into the RCS and interrupt primary to secondary heat transfer.

B. Correct.

Part 1 is correct. IAW OP-TM-EOP-0061, LOCA Cooldown Basis Document:

Step 3.30 The step intent is to stop OTSG cooling component of RCS cooldown if low pressure injection sources are not available or a CFT isolation valve was not able to be closed.

- RCS pressure control is based on SCM, and maintaining RCS pressure may not be possible due to leak size. If OTSG cooling is used to augment RCS cooldown, then it is not desirable to intentionally lower RCS pressure below 202 psig, as Core Flood tank nitrogen may be injected into the RCS and interrupt primary to secondary heat transfer.
- Once Core Flood tanks are isolated, or at least one train of LPI is operating, the nitrogen cover gas cannot be injected into the RCS interrupting primary to secondary heat transfer. OTSG cooling can be resumed until DHR conditions are established.
- For saturated conditions, the average incore temperature of 389F is a more accurate indication the RCS pressure is greater than 202 psig than direct indication of RCS pressure. 26F is added to 389F to provide the procedural limit of 415F to account conservatively for instrument error.

Part 2 is correct. IAW OP-TM-EOP-006, LOCA Cooldown:

Step 3.30 RNO:

- MAINTAIN incore thermocouples > 415F
- When at least one train of LPI is operating, or both CF-V-1A and CF-V-1B are Closed, then CONTINUE Plant Cooldown within the limits of Guide 11, "Cooldown Rate Limits".

C. Incorrect.
Part 1 is incorrect. Plausible if the candidate confuses the purpose for the given step with that of another associated with Core Flood Tanks. OP-TM-EOP-006 Step 3.4 states:

3.4 If Core Flood tank levels > 2 ft, then ENSURE both CF-V-1A and CF-V-1B are Open. (PCR)

And the basis for the step is:

Step 3.4 The step intent is to ensure Core Flood injection flow paths are available. The Core Flood system is an integral component of the Emergency Core Cooling System (ECCS). For this passive system to respond correctly, a flow path to the reactor vessel is necessary. The Core Flood tanks support reactor vessel refill and reflood volumes for a spectrum of Large Break Loss of Coolant Accidents (LBLOCA). For smaller breaks where adequate subcooled margin is not recovered, they provide additional volume with the HPI system, until the system has depressurized to the LPI injection phase. The 2 ft criteria accounts for previous injection where the CFT tank outlet valves may have been closed following injection to prevent CFT nitrogen from being injected into the RCS.

Part 2 is incorrect. Plausible if the candidate understands the purpose of OP-TM-EOP-006 but does not recognize that performing the cooldown will inject nitrogen into the RCS. IAW OP-TM-EOP-0061, LOCA Cooldown Basis Document:

Step 3.30: If OTSG cooling is used to augment RCS cooldown, then it is not desirable to intentionally lower RCS pressure below 202 psig, as Core Flood tank nitrogen may be injected into the RCS and interrupt primary to secondary heat transfer.

D. Incorrect.
Part 1 is incorrect. Plausible if the candidate confuses the purpose for the given step with that of another associated with Core Flood Tanks. OP-TM-EOP-006 Step 3.4 states:

3.4 If Core Flood tank levels > 2 ft, then ENSURE both CF-V-1A and CF-V-1B are Open. (PCR)

And the basis for the step is:

Step 3.4 The step intent is to ensure Core Flood injection flow paths are available. The Core Flood system is an integral component of the Emergency Core Cooling System (ECCS). For this passive system to respond correctly, a flow path to the reactor vessel is necessary. The Core Flood tanks support reactor vessel refill and reflood volumes for a spectrum of Large Break Loss of Coolant Accidents (LBLOCA). For smaller breaks where adequate subcooled margin is not recovered, they provide additional volume with the HPI system, until the system has depressurized to the LPI injection phase. The 2 ft criteria accounts for previous injection where the CFT tank outlet valves may have been closed following injection to prevent CFT nitrogen from being injected into the RCS.

Part 2 is correct. IAW OP-TM-EOP-006, LOCA Cooldown:

Step 3.30 RNO:
- MAINTAIN incore thermocouples > 415F
- When at least one train of LPI is operating, or both CF-V-1A and CF-V-1B are Closed, then CONTINUE Plant Cooldown within the limits of Guide 11, "Cooldown Rate Limits".

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	E08	2.4.18
Importance Rating		4.0

LOCA Cooldown - Depress. : Knowledge of the specific bases for EOPs.

Proposed Question: SRO Question # 10 (#85)

Technical Reference(s): OP-TM-EOP-006, p 11, Rev 12
OP-TM-EOP-0061, p 12, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP006-PCO-4

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 operator must demonstrate knowledge of specific bases for EOP's associated with LOCA Cooldown.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall the basis for a procedure step and also the RNO action in the event that the criteria within the step are not met.

The question is SRO-Only because the candidate must demonstrate knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific subprocedures or emergency contingency procedures (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 7). Specifically, the decision must be made in the RNO column to halt the cooldown, which would be in accordance with EOP Guides.

What MUST be known:

1. What is the basis associated with specific steps in OP-TM-EOP-006?
2. What is the RNO action to prevent CFT Nitrogen into the Reactor Vessel?

86 **ID: 1151797** **Points: 1.00**

Plant Conditions:

- The plant is in a Cold Shutdown condition to replace a CRDM stator.
- The plant is currently 50 EFPD away from the end of the fuel cycle.
- Plant Startup preparations are in progress.

Sequence of Events:

- The PORV block valve has just been discovered to be stuck in the OPEN position and has been declared inoperable.
- The PORV is operable.

Given the above information, which of the following describes the correct action to take IAW Tech Specs?

A. Maintain Cold Shutdown conditions until the PORV Block Valve is restored to an Operable status.

B. Maintain Cold Shutdown conditions until power is removed from the PORV to ensure it stays in the Open position.

C. The Plant Startup may commence but the PORV Block Valve must be returned to an Operable status within 8 hours of criticality.

D. The Plant Startup may commence and the PORV Block Valve must be restored to an Operable status prior to the first startup of the **next** fuel cycle.

Answer: D

Answer Explanation

A. Incorrect.

Would be correct if greater than 90 days out from the End of Core cycle. IAW Tech Specs:

3.1.12.4 The PORV Block Valve shall be OPERABLE during HOT STANDBY, STARTUP, and POWER OPERATION:

b. With the PORV block valve inoperable, restore the inoperable valve to OPERABLE status prior to startup from the next COLD SHUTDOWN unless the COLD SHUTDOWN occurs within 90 Effective Full Power Days (EFPD) of the end of the fuel cycle. If a COLD SHUTDOWN occurs within this 90 day period, restore the inoperable valve to OPERABLE status prior to startup for the next fuel cycle.

B. Incorrect.

The PORV cannot be taken out of service to take the reactor critical. The PORV must be kept operable while in the LTOP mode. If the plant was in a condition other than Cold Shutdown, this would be a correct answer. Plausible if the candidate is not familiar with the PORV and PORV Block Valve Tech Specs. IAW Tech Specs:

3.1.12.4 The PORV Block Valve shall be OPERABLE during HOT STANDBY, STARTUP, and POWER OPERATION:

a. With the PORV Block Valve inoperable, within 1 hour either:

1. restore the PORV Block Valve to OPERABLE status or
2. close the PORV (verify closed) and remove power from the PORV
3. otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

C. Incorrect.

There is no 8 hour time associated with the PORV Block Valve. There is, however, an 8 hour time frame associated with the PORV. Plausible if the candidate is not familiar with the PORV and PORV Block Valve Tech Specs. IAW Tech Specs:

3.1.12.2 The PORV settings shall be as follows:

a. Low Temperature Overpressure Protection Setpoint

1. When indicated RCS temperature is = 313°F, the LTOP system shall be operable as defined in Specification 3.1.12.1 and
2. The PORV will have a maximum lift setpoint of 592 psig. With the PORV setpoint above the maximum value, within 8 hours either:
    1. restore the setpoint below the maximum value, or
    2. verify pressurizer level is = 100 inches indicated and satisfy the requirements of Technical Specification 3.1.12.3 allowing the PORV to be taken out of service.

D. Correct.
IAW Tech Specs:

3.1.12.4 The PORV Block Valve shall be OPERABLE during HOT STANDBY, STARTUP, and POWER OPERATION:

b. With the PORV block valve inoperable, restore the inoperable valve to OPERABLE status prior to startup from the next COLD SHUTDOWN unless the COLD SHUTDOWN occurs within 90 Effective Full Power Days (EFPD) of the end of the fuel cycle. If a COLD SHUTDOWN occurs within this 90 day period, restore the inoperable valve to OPERABLE status prior to startup for the next fuel cycle.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	010	A2.03
Importance Rating		4.2

Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: PORV failures

Proposed Question: SRO Question # 11 (#86)

Technical Reference(s): Tech Spec, p 3-18d-3-18e, Amend 281

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-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	
	55.43	2

Comments:
The KA is matched because the operator must demonstrate the ability to predict the impacts of a PORV Block Valve failure on the PZR PCS and then use Tech Specs to determine the course of action with regards to a plant startup.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall the Tech Spec action to take associated with the PORV Block Valve.

The question is SRO-Only because the candidate must demonstrate knowledge of the application of Required Actions and the application of generic Limiting Condition for Operation (LCO) requirements (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3). Specifically, it is SRO LOK to understand the Tech Spec action statements.

What MUST be known:
1. What is the Tech Spec associated with the PORV Block Valve? 2. What is the Tech Spec action statement associated with the PORV Block Valve for Cold Shutdown conditions at end-of-life?

87 **ID: 1147554** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- RR-P-1A is OOS for repairs.

Event:

- Design Basis LOCA occurs.
- RB Pressure is currently 35 psig and rising.
- All "B" Trains of ESAS fail to automatically actuate.
- No operator actions have occurred.

Given the above information, Reactor Building pressure will ___(1)___ predicted limits, and manually initiating the "B" Train ___(2)___ is the appropriate procedural action to take based on the failures.

A. (1) exceed
(2) 30# ESAS, only

B. (1) exceed
(2) 4# and 30# ESAS

C. (1) be controlled within
(2) 30# ESAS, only

D. (1) be controlled within
(2) 4# and 30# ESAS

Answer: B

Answer Explanation

A. Incorrect.

Part 1 is correct. IAW Tech Spec 3.3 Basis:

- The post-accident reactor building emergency cooling may be accomplished by three emergency cooling units, by two spray systems, or by a combination of one emergency cooling unit and one spray system. The specified requirements assure that the required post-accident components are available.

Additionally, IAW TQ-TM-104-214-C001, Reactor Building Spray:

- Explanation
  - o In the event of a Large Break LOCA, the Building Spray System maintains Reactor Building pressure less than 55 psig by injecting spray water into the Reactor Building to reduce the pressure and temperature, thereby maintaining the integrity of the third and final boundary between fission products and the environment.

Part 2 is incorrect. Plausible if the candidate does not recognize that a block 4 signal must be present in order for Building Spray to operate on a 30# signal. A Block 4 signal will be in place if a 1600# RCS, 500# RCS, or a 4# RB signal is actuated. IAW TQ-TM-104-214-C001, Reactor Building Spray:

- The Reactor Building Spray pumps will start after the Engineered Safeguards Actuation System load sequence reaches block 4 (15 second delay) if RB pressure exceeds 30 psig. setpoint.

B. Correct.

Part 1 is correct. IAW Tech Spec 3.3 Basis:

- The post-accident reactor building emergency cooling may be accomplished by three emergency cooling units, by two spray systems, or by a combination of one emergency cooling unit and one spray system. The specified requirements assure that the required post-accident components are available.

Additionally, IAW TQ-TM-104-214-C001, Reactor Building Spray:

- Explanation
  - o In the event of a Large Break LOCA, the Building Spray System maintains Reactor Building pressure less than 55 psig by injecting spray water into the Reactor Building to reduce the pressure and temperature, thereby maintaining the integrity of the third and final boundary between fission products and the environment.

Part 2 is correct. A block 4 signal must be present in order for Building Spray to operate on a 30# signal. A Block 4 signal will be in place if a 1600# RCS, 500# RCS, or a 4# RB signal is actuated. IAW TQ-TM-104-214-C001, Reactor Building Spray:

- The Reactor Building Spray pumps will start after the Engineered Safeguards Actuation System load sequence reaches block 4 (15 second delay) if RB pressure exceeds 30 psig. setpoint.

Additionally, procedural guidance is provided within OP-TM-EOP-006, LOCA Cooldown:

3.1 ENSURE 4 psig ESAS IAW OP-TM-642-902, "4 psig ESAS Actuation".
3.5 If Containment pressure has exceeded 30 psig, then INITIATE OP-TM-642-903, "30 psig ESAS Actuation".

C. Incorrect.
Part 1 is incorrect. Plausible if the candidate is not familiar with the minimum equipment combinations for post-accident reactor building emergency cooling as stated in the Tech Spec Basis. The question leaves two emergency cooling units and no Building Spray pumps. This is inadequate as described in Tech Spec 3.3 Basis:

- The post-accident reactor building emergency cooling may be accomplished by three emergency cooling units, by two spray systems, or by a combination of one emergency cooling unit and one spray system. The specified requirements assure that the required post-accident components are available.

Part 2 is correct. A block 4 signal must be present in order for Building Spray to operate on a 30# signal. A Block 4 signal will be in place if a 1600# RCS, 500# RCS, or a 4# RB signal is actuated. IAW TQ-TM-104-214-C001, Reactor Building Spray:

- The Reactor Building Spray pumps will start after the Engineered Safeguards Actuation System load sequence reaches block 4 (15 second delay) if RB pressure exceeds 30 psig. setpoint.

Additionally, procedural guidance is provided within OP-TM-EOP-006, LOCA Cooldown:

3.1 ENSURE 4 psig ESAS IAW OP-TM-642-902, "4 psig ESAS Actuation".
3.5 If Containment pressure has exceeded 30 psig, then INITIATE OP-TM-642-903, "30 psig ESAS Actuation".

D. Incorrect.
Part 1 is incorrect. Plausible if the candidate is not familiar with the minimum equipment combinations for post-accident reactor building emergency cooling as stated in the Tech Spec Basis. The question leaves two emergency cooling units and no Building Spray pumps. This is inadequate as described in Tech Spec 3.3 Basis:

- The post-accident reactor building emergency cooling may be accomplished by three emergency cooling units, by two spray systems, or by a combination of one emergency cooling unit and one spray system. The specified requirements assure that the required post-accident components are available.

Part 2 is incorrect. Plausible if the candidate does not recognize that a block 4 signal must be present in order for Building Spray to operate on a 30# signal. A Block 4 signal will be in place if a 1600# RCS, 500# RCS, or a 4# RB signal is actuated. IAW TQ-TM-104-214-C001, Reactor Building Spray:

- The Reactor Building Spray pumps will start after the Engineered Safeguards Actuation System load sequence reaches block 4 (15 second delay) if RB pressure exceeds 30 psig. setpoint.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	013	A2.01
	Importance Rating		4.8

Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: LOCA

Proposed Question: SRO Question # 12 (#87)

Technical Reference(s):
Tech Spec, p 3-24, Amend ECR TM 09-00160
TQ-TM-104-214-C001, p 4,36, Rev 9
OP-TM-EOP-006, p 3, Rev 12

Proposed References to be provided to applicants during examination: None

Learning Objective: 214-GLO-8

Question Source:
Bank #
Modified Bank # ID# 857715
New

Question History: Last NRC Exam: 03-1

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 operator must demonstrate the ability to predict the impacts of a LOCA with failures on the ESAS system and then use procedural guidance to determine the proper course of action.

The question is at the Comprehension/Analysis cognitive level because the operator must analyze the conditions given and determine the operating equipment, and then must determine the consequences based on Tech Spec basis.

The question is SRO-Only because the candidate must demonstrate knowledge of TS bases that are required to analyze TS required actions and terminology. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3). Specifically, it is SRO LOK to understand the Tech Spec basis statements.

What MUST be known:
1. What is the Tech Spec basis associated with Building Spray Pumps/ RB Emergency Cooler combinations? 2. What is the effect of not meeting the proper Building Spray Pumps/ RB Emergency Cooler combinations? 3. What is the ESAS logic that will initiate Building Spray? 4. What is the procedural guidance for initiating Building Spray manually?

Original Question:

Plant Conditions:

Plant operating at 100% power.
RR-P-1A is OOS for repairs.

Event:

Design Basis LOCA occurs.
RB Pressure is currently 35 psig and rising slowly
B train ESAS block timing relays fail to actuate.

Identify the ONE selection that describes the effect on Reactor Building pressure.

Reactor Building pressure will:

A. Be controlled within predicted limits by TWO Building Spray pumps independent of containment recirculation fans.

B. Be controlled within predicted limits by ONE Building Spray pump and TWO AH-E-1 containment recirculation fans in slow.

C. Exceed predicted limits for this accident with only ONE Building Spray pump and TWO containment recirculation fans in slow.

D. Exceed predicted limits for this accident with only TWO Building Spray pumps and TWO containment recirculation fans in slow.

Answer: C

88 **ID: 1147550** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- Gas release from Waste Decay Tank, WDG-T-1C is in progress.
- Auxiliary/Fuel Handling Building Atmospheric Monitor, RM-A-8 P,I,G is operable.
- Auxiliary Building Atmospheric Monitor, RM-A-6 Gas is operable.
- Fuel Handling Building Atmospheric Monitor, RM-A-4 P and G are out-of-service and the monitor is in the DEFEAT position.
- Controlled Access Area and Machine Shop Flow Rate Transmitter, FT-150, is out-of-service.

Event:

- One hour into the release Waste Gas Disposal Effluent Atmospheric Monitor, RM-A-7 is reading 5E5 CPM but is not alarming.
- I & C has determined RM-A-7 to be out-of-service.

Given the above information, one possible cause for RM-A-7 to not alarm properly is because the alarm calibration setpoint has drifted ___(1)___, and the CRS must direct that the release ___(2)___.

A. (1) low
(2) be suspended by closing Waste Decay Tank discharge, WDG-V-47

B. (1) low
(2) continue as long as RM-A-6 and Fuel Handling Building Flow Rate Transmitter, FT-149 remain operable

C. (1) high
(2) be suspended by closing Waste Decay Tank discharge, WDG-V-47

D. (1) high
(2) continue as long as RM-A-6 and Fuel Handling Building Flow Rate Transmitter, FT-149 remain operable

Answer: C

Answer Explanation

A. Incorrect.

Part 1 is incorrect. Plausible if the candidate is not familiar with calibration drift and believes that a drift low will cause a larger distance of movement to reach the alarm setpoint.

Part 2 is correct. Since AH-FT-150 is inoperable, RM-A-7 must be operable. Additionally, a release may occur with RM-A-7 inoperable as long as the items listed in Action 25 of ODCM Table 2.1-2 are met PRIOR to the release. Since the release is already in progress, it must be supsended and the process of approval must begin anew IAW Action 25. IAW CY-TM-170-300, Offsite Dose Calculation Manual (ODCM):

Table 2.1-2

Radioactive Gaseous Process and Effluent Monitoring Instrumentation

INSTRUMENT	MINIMUM CHANNEL OPERABLE	APPLICABILITY	ACTION
1. Waste Gas Holdup System			
a. Noble Gas Activity Monitor (RM-A-7)	1	***	25

*** Operability is not required when discharges are positively controlled through the closure of WDG-V-47 or where RM-A-8, AH-FT-149 and AH-FT-150 are operable and RM-A-8 is capable of automatic closure of WDG-V-47

ACTION 25: With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank may be released to the environment provided that prior to initiating the release:

1. At least two independent samples of the tank's contents are analyzed in accordance with Table 3.2-2, Item A, and
2. At least two technically qualified members of the Unit staff independently verify the release rate calculations and verify the discharge valve lineup.
3. The TMI Plant Manager shall approve each release. Otherwise, suspend release of radioactive effluent via this pathway.

B. Incorrect.

Part 1 is incorrect. Plausible if the candidate is not familiar with calibration drift and believes that a drift low will cause a larger distance of movement to reach the alarm setpoint.

Part 2 is incorrect. Plausible if the candidate is not familiar with the triple asterisk Applicability statement of Table 2.1-2 of the ODCM and confuses RM-A-8 with RM-A-6. IAW CY-TM-170-300, Offsite Dose Calculation Manual (ODCM):

*** Operability is not required when discharges are positively controlled through the closure of WDG-V-47 or where RM-A-8, AH-FT-149 and AH-FT-150 are operable and RM-A-8 is capable of automatic closure of WDG-V-47

C. Correct.
Part 1 is correct. Drift is defined as "A change in an instrument's reading or set point value over extended periods due to factors such as time, line voltage, or ambient temperature effects."
(Source: www.omega.com/literature/transactions/volume1/glossary.html)

Additionally, IAW 1101-2.1, Radiation Monitoring System Setpoints:

- RM-A7 (Gas)
  - Sensitivity: 4E5 CPM/μCi/cc (based on Xe-133)
  - HIGH ALARM SETPOINT: 4E5 CPM
    - Basis: Equation 4.1.1, 4.1.2, in the Offsite Dose Calculation Manual (ODCM)
  - cTB ≤ 2.84E2 μCi/cc cskin ≤ 7.21E2 μCi/cc
    - CTB = smallest concentration allowable based on whole body exposure.
  - Parameters: Meter error factor = 75 percent
  - Calculation: (0.75) (2.84E2 μCi/cc) (4E5 CPM/μCi/cc) = 8.52E7 CPM
  - Therefore, the maximum allowable setpoint is 8.52E7 CPM.
  - Since this setpoint is higher than the range of the instrument read-out, a setpoint of 4E5 CPM will be used.

Therefore, if the alarm calibration setpoint has drifted high, the meter could read offscale and yet not alarm.

Part 2 is correct. Since AH-FT-150 is inoperable, RM-A-7 must be operable. Additionally, a release may occur with RM-A-7 inoperable as long as the items listed in Action 25 of ODCM Table 2.1-2 are met PRIOR to the release. Since the release is already in progress, it must be suspended and the process of approval must begin anew IAW Action 25. IAW CY-TM-170-300, Offsite Dose Calculation Manual (ODCM):

Table 2.1-2

Radioactive Gaseous Process and Effluent Monitoring Instrumentation

INSTRUMENT	MINIMUM CHANNEL OPERABLE	APPLICABILITY	ACTION
1. Waste Gas Holdup System			
a. Noble Gas Activity Monitor (RM-A-7)	1	***	25

*** Operability is not required when discharges are positively controlled through the closure of WDG-V-47 or where RM-A-8, AH-FT-149 and AH-FT-150 are operable and RM-A-8 is capable of automatic closure of WDG-V-47

ACTION 25: With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank may be released to the environment provided that prior to initiating the release:

1. At least two independent samples of the tank's contents are analyzed in accordance with Table 3.2-2, Item A, and
2. At least two technically qualified members of the Unit staff independently verify the release rate calculations and verify the discharge valve lineup.
3. The TMI Plant Manager shall approve each release. Otherwise, suspend release of radioactive effluent via this pathway.

D. Incorrect.
Part 1 is correct. Drift is defined as "A change in an instrument's reading or set point value over extended periods due to factors such as time, line voltage, or ambient temperature effects."
(Source: www.omega.com/literature/transactions/volume1/glossary.html)

Additionally, IAW 1102-2.1, Radiation Monitoring System Setpoints:

- RM-A7 (Gas)
    - Sensitivity: 4E5 CPM/µCi/cc (based on Xe-133)
    - HIGH ALARM SETPOINT: 4E5 CPM
        - Basis: Equation 4.1.1, 4.1.2, in the Offsite Dose Calculation Manual (ODCM)
    - cTB □ 2.84E2 µCi/cc cskin □ 7.21E2 µCi/cc
        - CTB = smallest concentration allowable based on whole body exposure.
    - Parameters: Meter error factor = 75 percent
    - Calculation: (0.75) (2.84E2 µCi/cc) (4E5 CPM/µCi/cc) = 8.52E7 CPM
    - Therefore, the maximum allowable setpoint is 8.52E7 CPM.
    - Since this setpoint is higher than the range of the instrument read-out, a setpoint of 4E5 CPM will be used.

Therefore, if the alarm calibration setpoint has drifted high, the meter could read offscale and yet not alarm.

Part 2 is incorrect. Plausible if the candidate is not familiar with the triple asterisk Applicability statement of Table 2.1-2 of the ODCM and confuses RM-A-8 with RM-A-6. IAW CY-TM-170-300, Offsite Dose Calculation Manual (ODCM):

*** Operability is not required when discharges are positively controlled through the closure of WDG-V-47 or where RM-A-8, AH-FT-149 and AH-FT-150 are operable and RM-A-8 is capable of automatic closure of WDG-V-47

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	073	A2.03
	Importance Rating		2.9

Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Calibration drift

Proposed Question: SRO Question # 13 (#88)

Technical Reference(s): 1101-2.1, p 51-52, Rev 84
CY-TM-170-300, p 24, 27, Rev 3

Proposed References to be provided to applicants during examination:	None

Learning Objective:	661-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	
	55.43	2

Comments:
The KA is matched because the operator must demonstrate the ability to predict the impacts of calibration drift with failures on the Process Radiation Monitoring system and then use procedural guidance to determine the proper course of action.

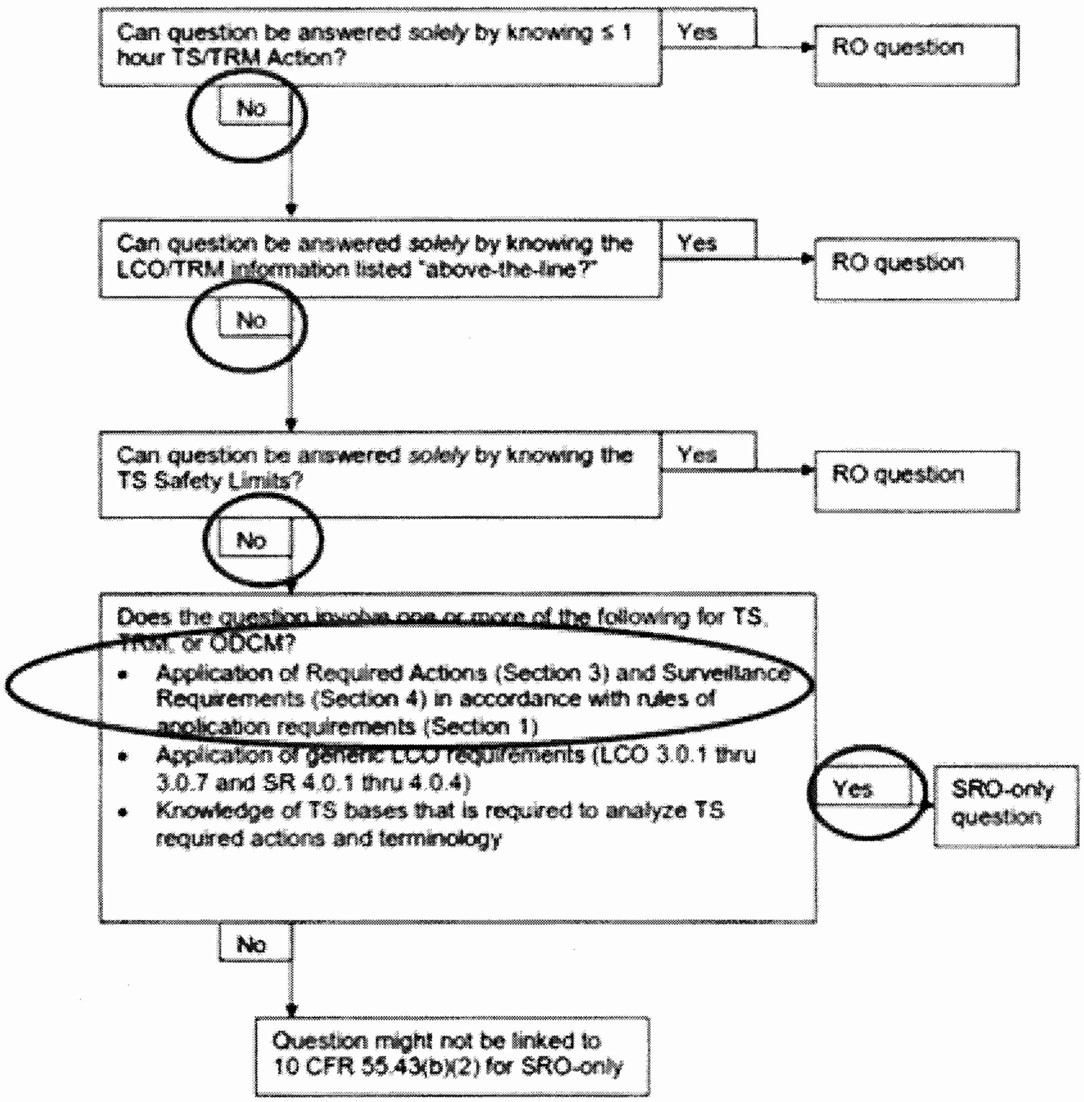
The question is at the Comprehension/Analysis cognitive level because the operator must analyze the conditions given and determine the operating equipment, and then must determine the consequences based on ODCM Action statements.

Although part 1 can be answered by an RO, the question is at the SRO level because the candidates must demonstrate the ability to apply generic Limiting Condition for Operation (LCO) requirements. RO's are expected to know the LCO statements and associated applicability information, but the question is specifically testing the correct action to take. (Source: Clarification Guidance for SRO-only Questions, Rev 1, Pages 3-4).

What MUST be known:
1. What is the ODCM action statement associated with RM-A-7? 2. What is the correct action to take with regards to a release in progress upon a loss of RM-A-7 with other failures? 3. What is the effect of drift on a meter?

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

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



89 **ID: 1147555** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.

Event:

- A complete loss of the "A" DC distribution system occurs.
- An Auxiliary Operator is dispatched to close Air Start Header isolation to EG-Y-1A Valve, EG-V-15A.
- Emergency Diesel Generator, EG-Y-1A, is running.

Given the above information, which one of the following correctly describes:
(1) The required action and the basis for the action, and
(2) The most limiting Tech Spec LCO Timeclock?

A. (1) The emergency diesel generator should be **immediately** tripped since it is not loaded.
(2) 12 Hours based on the low probability of an event occurring while equipment is out of service.

B. (1) The emergency diesel generator should be **immediately** tripped since it is not loaded.
(2) 8 Hours based on being able to perform three complete cycles of safeguard breaker operations.

C. (1) The emergency diesel generator should remain running **until** evaluated by the Shift Manager.
(2) 12 Hours based on the low probability of an event occurring while equipment is out of service.

D. (1) The emergency diesel generator should remain running **until** evaluated by the Shift Manager.
(2) 8 Hours based on being able to perform three complete cycles of safeguard breaker operations.

Answer: D

Answer Explanation

A. Incorrect.

Part 1 is incorrect. Plausible since the plant used to have an IMA to trip the EDG. IAW AOP-023, "A" DC System Failure:

NOTE: If EG-Y-1A started due to loss of DC, air start distributor damage is probable. EG-Y-1A will not be available once it has been shutdown, even if DC is restored.
4.14 WAAT SM has evaluated potential to restore DC and potential need for EG-Y-1A, and has determined EG-Y-1A should be secured, then TRIP EG-Y-1A fuel rack (S. side of EG-Y-1A).

Part 2 is incorrect but plausible since there are several Tech Spec Bases that state the reason for the time frame as being a low probability of anything occurring during that time frame. Examples include, 1 RCP secured in each loop, Checking a Diesel Generator for a common mode fault, and taking ES equipment out for maintenance.

B. Incorrect.

Part 1 is incorrect. Plausible since the plant used to have an IMA to trip the EDG. IAW AOP-023, "A" DC System Failure:

NOTE: If EG-Y-1A started due to loss of DC, air start distributor damage is probable. EG-Y-1A will not be available once it has been shutdown, even if DC is restored.
4.14 WAAT SM has evaluated potential to restore DC and potential need for EG-Y-1A, and has determined EG-Y-1A should be secured, then TRIP EG-Y-1A fuel rack (S. side of EG-Y-1A).

Part 2 is correct. IAW OP-TM-AOP-0231, "A" DC System Failure Basis Document:

1.2 Technical Specifications:
1.2.1 Tech Spec 3.7
A. 3.7.1.f Station batteries are charged and in service. Two battery chargers per battery are in service.
B. 3.7.2.g One station battery may be removed from service for not more than eight hours.
1.2.2 Standard Tech Spec 3.8.4.C
A. In modes 1,2,3 or 4 if the DC distribution system is inoperable for reasons other than inoperable battery charger or inoperable battery, operability must be restored within 2 hours or the unit must be placed in hot shutdown within 6 hours.

1.0 DESIGN OR LICENSING BASIS REQUIREMENTS
1.1 UFSAR: Section 8.2.2.6 describes the 125/250VDC system.
UFSAR: Section 8.2.2.6: C. The capacity of each of the two redundant batteries is sufficient to feed its connected essential load for 2 hours continuously and perform three complete cycles of safeguard breaker closures and subsequent tripping. The 2 hour rating is based on the time required to ensure that all nuclear and BOP emergency equipment can perform its intended function and on the criteria contained in Draft 3 of Reference 4, the IEEE criteria for Class 1E electrical systems.

C. Incorrect.
Part 1 is correct. IAW AOP-023, "A" DC System Failure:

NOTE: If EG-Y-1A started due to loss of DC, air start distributor damage is probable. EG-Y-1A will not be available once it has been shutdown, even if DC is restored.
4.14 WAAT SM has evaluated potential to restore DC and potential need for EG-Y-1A, and has determined EG-Y-1A should be secured, then TRIP EG-Y-1A fuel rack (S. side of EG-Y-1A).

Part 2 is incorrect but plausible since there are several Tech Spec Bases that state the reason for the time frame as being a low probability of anything occurring during that time frame. Examples include, 1 RCP secured in each loop, Checking a Diesel Generator for a common mode fault, and taking ES equipment out for maintenance.

D. Correct.
Part 1 is correct. IAW AOP-023, "A" DC System Failure:

NOTE: If EG-Y-1A started due to loss of DC, air start distributor damage is probable. EG-Y-1A will not be available once it has been shutdown, even if DC is restored.
4.14 WAAT SM has evaluated potential to restore DC and potential need for EG-Y-1A, and has determined EG-Y-1A should be secured, then TRIP EG-Y-1A fuel rack (S. side of EG-Y-1A).

Part 2 is correct. IAW OP-TM-AOP-0231, "A" DC System Failure Basis Document:

1.2 Technical Specifications:
1.2.1 Tech Spec 3.7
A. 3.7.1.f Station batteries are charged and in service. Two battery chargers per battery are in service.
B. 3.7.2.g One station battery may be removed from service for not more than eight hours.
1.2.2 Standard Tech Spec 3.8.4.C
A. In modes 1,2,3 or 4 if the DC distribution system is inoperable for reasons other than inoperable battery charger or inoperable battery, operability must be restored within 2 hours or the unit must be placed in hot shutdown within 6 hours.

1.0 DESIGN OR LICENSING BASIS REQUIREMENTS
1.1 UFSAR: Section 8.2.2.6 describes the 125/250VDC system.
UFSAR: Section 8.2.2.6: C. The capacity of each of the two redundant batteries is sufficient to feed its connected essential load for 2 hours continuously and perform three complete cycles of safeguard breaker closures and subsequent tripping. The 2 hour rating is based on the time required to ensure that all nuclear and BOP emergency equipment can perform its intended function and on the criteria contained in Draft 3 of Reference 4, the IEEE criteria for Class 1E electrical systems.

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #		2
Group #		1
K/A #	063	2.4.20
Importance Rating		4.3

DC Electrical Distribution: Knowledge of the operational implications of EOP warnings, cautions, and notes.

Proposed Question: SRO Question # 14 (#89)

Technical Reference(s):
OP-TM-AOP-023, p 19, Rev 5
OP-TM-AOP-0231, p 1, Rev 4
Tech Spec, p 3-43, Amend 278
Tech Spec, p 3-43a, Amend 224

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP023-PCO-4

Question Source: Bank #

Modified Bank # ID# 862920

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 2

Comments:
The KA is matched because the operator must demonstrate the knowledge of a note contained within an Abnormal Operating Procedure associated with the DC Electrical System. An Abnormal Operating Procedure is referenced in lieu of an Emergency Operating Procedure since there are no direct Emergency Operating Procedures associated with the DC Electrical Distribution System.

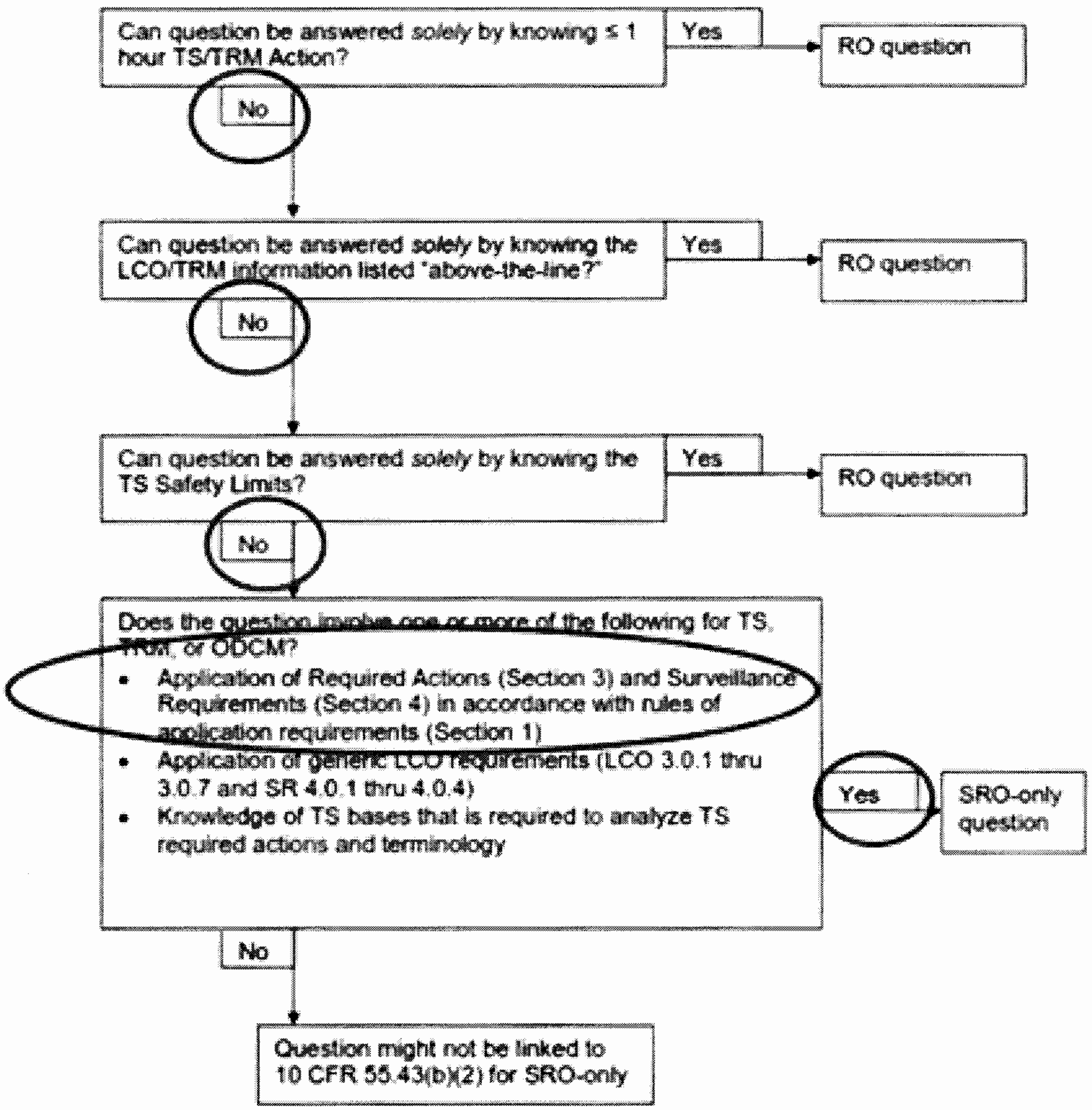
The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall the note contained within an Abnormal Operating Procedure and also the Tech Spec action associated with the DC electrical Distribution System.

The question is at the SRO level because the candidates must demonstrate the ability to apply generic Limiting Condition for Operation (LCO) requirements. RO's are expected to know the LCO statements and associated applicability information, but the question is specifically testing the correct action to take and the associated timeframe (Source: Clarification Guidance for SRO-only Questions, Rev 1, Pages 3-4).

What MUST be known:
1. What are the notes and steps within OP-TM-AOP-023? 2. What is the Tech Spec/Procedure/FSAR Basis for Battery life?

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

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



Original Question:

Plant Conditions:

The plant is operating at 100% power.

Event:

A complete loss of the "A" DC distribution system occurs.
After 5 minutes, the NLO dispatched to close Air Start Header isolation to EG-Y-1A Valve, EG-V-15A, reports Emergency Diesel Generator, EG-Y-1A, is running.

What is the required action and the basis for the action?

A. The emergency diesel generator should be **immediately** tripped since it is not loaded.

B. The emergency diesel generator should be **immediately** tripped since air distributor damage has likely already occurred.

C. The emergency diesel generator should be tripped **immediately** since there is no way to load the diesel to > 1MWe to burn unused fuel.

D. The emergency diesel generator should remain running **until** evaluated by the Shift Manager since it is likely that it will not start again until repaired.

Answer: D

Answer Explanation

Explanation (Optional):

A. **Incorrect.**

Part 1 is incorrect . Since the criteria are not met for the immediate manual action of OP-TM-EOP-030, the step is not applicable and the crew would move on to the follow up actions. Plausible if the candidate is not familiar with the steps contained within OP-TM-EOP-030. IAW OP-TM-EOP-030, Loss of Decay Heat Removal:

IMMEDIATE ACTIONS

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.

Part 2 is correct. IAW Tech Spec 3.4 Basis:

When the RCS temperature is below 250 degrees F, a single DHR String (Loop), or single OTSG with an EFW Pump and a flowpath capable of supporting natural circulation is sufficient to provide removal of decay heat at all times following the cooldown to 250 degrees F.

B. **Incorrect.**

Part 1 is incorrect. Since the criteria are not met for the immediate manual action of OP-TM-EOP-030, the step is not applicable and the crew would move on to the follow up actions. Plausible if the candidate is not familiar with the steps contained within OP-TM-EOP-030. IAW OP-TM-EOP-030, Loss of Decay Heat Removal:

IMMEDIATE ACTIONS

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.

Part 2 is incorrect. Although the statement is partially correct, it does not state that a second method be available. Plausible if the candidate is not familiar with the Decay Heat Removal Capability Bases. IAW Tech Spec 3.4 Bases:

The requirement to keep a DHR Loop in operation as necessary to maintain the RCS subcooled at the core outlet provides the guidance to ensure that steam conditions which could inhibit core cooling do not occur.

C. **Correct.**

Part 1 is correct. Since the Immediate Manual Action does not need to be taken, the follow-up actions are applicable. IAW OP-TM-EOP-030, Loss of Decay Heat Removal:

3.0 FOLLOW-UP ACTIONS

3.1 STOP any procedure in progress which is reducing RCS inventory.
3.2 If fuel handling operations are in progress in the reactor building, then NOTIFY the fuel handling SRO to suspend fuel handling IAW 1505-1, "Fuel And Control Component Shuffles".
3.3 ACTUATE Reactor Building evacuation alarm and ANNOUNCE "Decay Heat Cooling has been interrupted, all personnel shall exit the reactor building."

Part 2 is correct. IAW Tech Spec 3.4 Basis:

When the RCS temperature is below 250 degrees F, a single DHR String (Loop), or single OTSG with an EFW Pump and a flowpath capable of supporting natural circulation is sufficient to provide removal of decay heat at all times following the cooldown to 250 degrees F.

D. **Incorrect.**

Part 1 is correct. Since the Immediate Manual Action does not need to be taken, the follow-up actions are applicable. IAW OP-TM-EOP-030, Loss of Decay Heat Removal:

3.0 FOLLOW-UP ACTIONS

3.1 STOP any procedure in progress which is reducing RCS inventory.
3.2 If fuel handling operations are in progress in the reactor building, then NOTIFY the fuel handling SRO to suspend fuel handling IAW 1505-1, "Fuel And Control Component Shuffles".
3.3 ACTUATE Reactor Building evacuation alarm and ANNOUNCE "Decay Heat Cooling has been interrupted, all personnel shall exit the reactor building."

Part 2 is incorrect. Although the statement is partially correct, it does not state that a second method be available. Plausible if the candidate is not familiar with the Decay Heat Removal Capability Bases. IAW Tech Spec 3.4 Bases:

The requirement to keep a DHR Loop in operation as necessary to maintain the RCS subcooled at the core outlet provides the guidance to ensure that steam conditions which could inhibit core cooling do not occur.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	076	2.4.1
	Importance Rating		4.8

Service Water: Knowledge of EOP entry conditions and immediate action steps.

Proposed Question: SRO Question # 15 (#90)

Technical Reference(s): OP-TM-EOP-030, p 3, Rev 6
Tech Spec, p 3-26d, Amend 277

Proposed References to be provided to applicants during examination: None

Learning Objective: 533-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

55.43 2

Comments:
The KA is matched because the operator must demonstrate the knowledge of EOP entry conditions and immediate action steps associated with the Service Water system (specifically, a failure within the Decay River Water System)

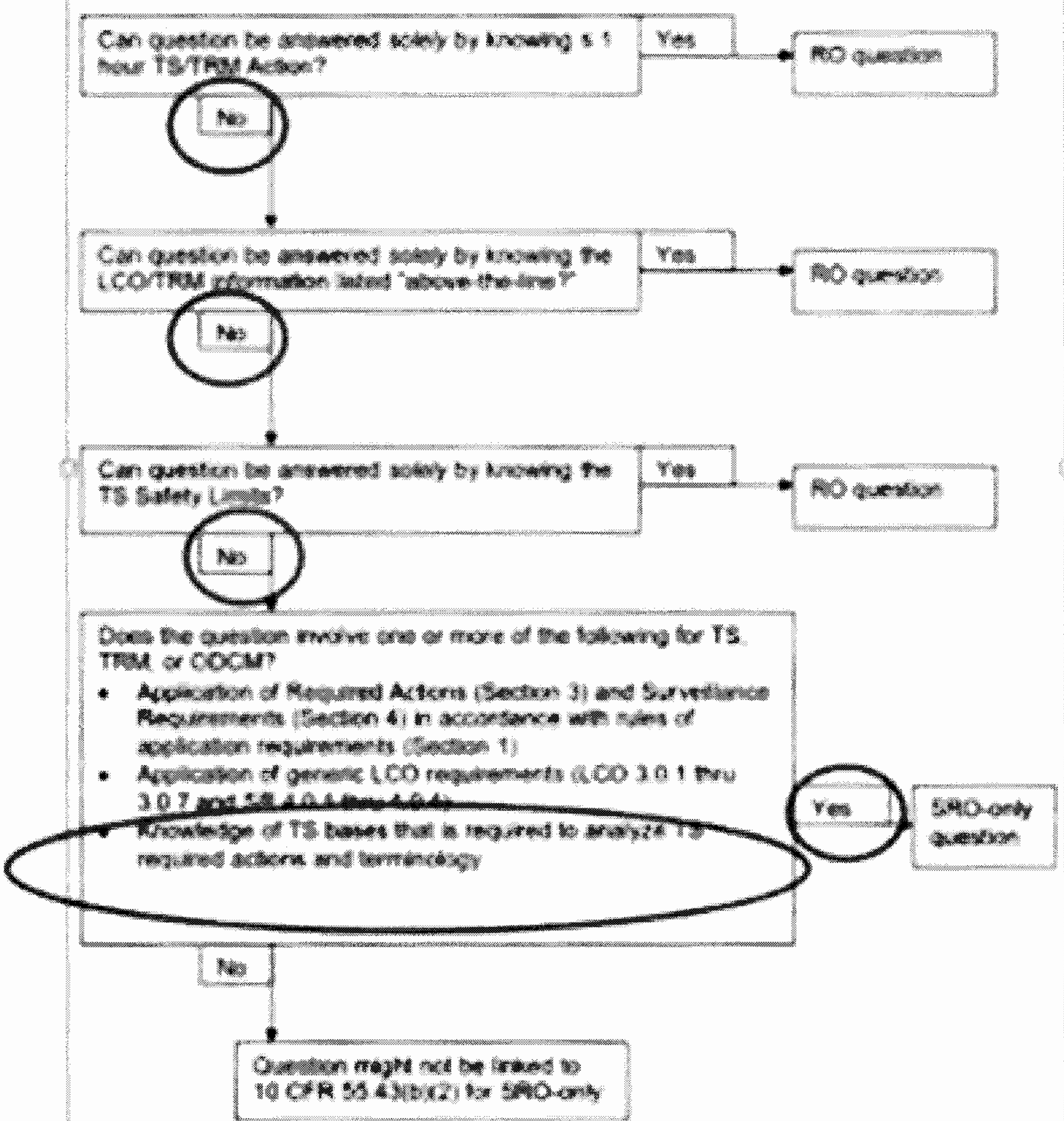
The question is at the Comprehension/Analysis cognitive level because the operator must recognize the required procedure routing and also must understand the Tech Spec basis based on the specific conditions given.

Although part 1 can be answered by an RO, the question is at the SRO level because the candidates must demonstrate the Knowledge of TS bases that are required to analyze TS required actions and terminology (Source: Clarification Guidance for SRO-only Questions, Rev 1, Pages 3-4).

What MUST be known:
1. What is the criteria for OP-TM-EOP-030 Immediate Manual Actions? 2. What is the Tech Spec Basis for Decay Heat Removal During Cold Shutdown conditions?

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

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



91 **ID: 1147557** **Points: 1.00**

Plant Conditions:

- The plant is operating at 80% reactor power.
- COLR Table 1, is shown below:

TABLE 1
Error-Adjusted Quadrant Tilt Limits

Full Incore System (FIS)	**6.83**	**4.53**	**16.8**
Minimum Incore System (MIS)[a]	**2.78**	**1.90**	**9.5**

Given the above information, the Full Incore System (FIS) Steady State Tilt Limit is ___(1)___ and if the Steady State Tilt Limit is exceeded by 9%, the plant has ___(2)___.

A. (1) 4.53
(2) 10 hours to reduce quadrant tilt to less than the limit to ensure that the initial LOCA maximum linear heat rate will not cause the maximum clad temperature to exceed 10 CFR 50 Appendix K criteria

B. (1) 4.53
(2) 2 hours to reduce quadrant tilt to less than the limit to ensure that the number of challenges to the Pressurizer safety valves and power operated relief valve are reduced during a design basis accident

C. (1) 6.83
(2) 10 hours to reduce quadrant tilt to less than the limit to ensure that the initial LOCA maximum linear heat rate will not cause the maximum clad temperature to exceed 10 CFR 50 Appendix K criteria

D. (1) 6.83
(2) 2 hours to reduce quadrant tilt to less than the limit to ensure that the number of challenges to the Pressurizer safety valves and power operated relief valve are reduced during a design basis accident

Answer: A

Answer Explanation

A. Correct.

Part 1 is correct. Since Reactor Power was given as 80% in the stem, the COLR steady state limit is the one associated with >60%, which is 4.53. IAW COLR, Table 1:

TABLE 1
Error-Adjusted Quadrant Tilt Limits

	Steady State Limit 15 < Power ≤ 60%	Steady State Limit Power > 60%	Maximum Limit Power > 15%
Full Incore System (FIS)	6.83	4.53	16.8
Minimum Incore System (MIS)[a]	2.78	1.90	9.5

Part 2 is correct. IAW Tech Spec 3.5.2.4:

e. If quadrant power tilt exceeds the tilt limit then within a period of 10 hours, the quadrant power tilt shall be reduced to less than the tilt limit except for physics tests, or the following verifications and/or adjustments in setpoints and limits shall be made:

Additionally, IAW Tech Spec 3.5.2 Bases:

- The axial power imbalance, quadrant power tilt, and control rod position limits are based on LOCA analyses which have defined the maximum linear heat rate. These limits are developed in a manner that ensures the initial condition LOCA maximum linear heat rate will not cause the maximum clad temperature to exceed 10 CFR 50 Appendix K. Operation outside of any one limit alone does not necessarily constitute a situation that would cause the Appendix K Criteria to be exceeded should a LOCA occur. Each limit represents the boundary of operation that will preserve the Acceptance Criteria even if all three limits are at their maximum allowable values simultaneously. Additional conservatism included in the limit development is introduced by application of:
    a. Nuclear uncertainty factors
    b. Thermal calibration uncertainty
    c. Fuel densification effects
    d. Hot rod manufacturing tolerance factors
    e. Postulated fuel rod bow effects
    f. Peaking limits based on initial condition for Loss of Coolant Flow transients.

B. Incorrect.
Part 1 is correct. Since Reactor Power was given as 80% in the stem, the COLR steady state limit is the one associated with >60%, which is 4.53.

Part 2 is incorrect. Plausible if the candidate either confuses the Basis for Quadrant Tilt Limits with that of the Basis for a loss of feedwater pumps or main turbine trip or confuses the 10 hour requirement with the 2 hour requirement to be less than 15%. IAW Tech Spec 3.5.2.4:

> f. Except for physics or diagnostic testing, if quadrant tilt is in excess of the maximum tilt limit defined in the CORE OPERATING LIMITS REPORT and using the applicable detector system defined in 3.5.2.4.a, b, and c above, reduce thermal power to <15% FP within 2 hours.

Additionally, IAW Tech Spec 3.5.1 Bases:

- Every reasonable effort will be made to maintain all safety instrumentation in operation. The reactor trip, on loss of feedwater may be bypassed below 7% reactor power. The bypass is automatically removed when reactor power is raised above 7%. The reactor trip, on turbine trip, may be bypassed below 45% reactor power (Reference 1). The safety feature actuation system must have two analog channels functioning correctly prior to startup. The anticipatory reactor trips on loss of feedwater pumps and turbine trip have been added to reduce the number of challenges to the safety valves and power operated relief valve but have not been credited in the safety analyses.

C. Incorrect.
Part 1 is incorrect. Plausible if the candidate is not familiar with the power levels associated with the Quadrant Tilt Limits and selects the limit that is applicable between 15% and 60% reactor power. IAW COLR, Table 1:

TABLE 1
Error-Adjusted Quadrant Tilt Limits

	Steady State Limit 15 < Power ≤ 60%	Steady State Limit Power > 60%	Maximum Limit Power > 15%
Full Incore System (FIS)	6.83	4.53	16.8
Minimum Incore System (MIS)[a]	2.78	1.90	9.5

Part 2 is correct. IAW Tech Spec 3.5.2.4:

> e. If quadrant power tilt exceeds the tilt limit then within a period of 10 hours, the quadrant power tilt shall be reduced to less than the tilt limit except for physics tests, or the following verifications and/or adjustments in setpoints and limits shall be made:

Additionally, IAW Tech Spec 3.5.2 Bases:

- The axial power imbalance, quadrant power tilt, and control rod position limits are based on LOCA analyses which have defined the maximum linear heat rate. These limits are developed in a manner that ensures the initial condition LOCA maximum linear heat rate will not cause the maximum clad temperature to exceed 10 CFR 50 Appendix K. Operation outside of any one limit alone does not necessarily constitute a situation that would cause the Appendix K Criteria to be exceeded should a LOCA occur. Each limit represents the boundary of operation that will preserve the Acceptance Criteria even if all three limits are at their maximum allowable values simultaneously. Additional conservatism included in the limit development is introduced by application of:
    a. Nuclear uncertainty factors
    b. Thermal calibration uncertainty
    c. Fuel densification effects
    d. Hot rod manufacturing tolerance factors
    e. Postulated fuel rod bow effects
    f. Peaking limits based on initial condition for Loss of Coolant Flow transients.

D. Incorrect.
Part 1 is incorrect. Plausible if the candidate is not familiar with the power levels associated with the Quadrant Tilt Limits and selects the limit that is applicable between 15% and 60% reactor power.

Part 2 is incorrect. Plausible if the candidate either confuses the Basis for Quadrant Tilt Limits with that of the Basis for a loss of feedwater pumps or main turbine trip or confuses the 10 hour requirement with the 2 hour requirement to be less than 15%. IAW Tech Spec 3.5.2.4:

> f. Except for physics or diagnostic testing, if quadrant tilt is in excess of the maximum tilt limit defined in the CORE OPERATING LIMITS REPORT and using the applicable detector system defined in 3.5.2.4.a, b, and c above, reduce thermal power to <15% FP within 2 hours.

Additionally, IAW Tech Spec 3.5.1 Bases:

- Every reasonable effort will be made to maintain all safety instrumentation in operation. The reactor trip, on loss of feedwater may be bypassed below 7% reactor power. The bypass is automatically removed when reactor power is raised above 7%. The reactor trip, on turbine trip, may be bypassed below 45% reactor power (Reference 1). The safety feature actuation system must have two analog channels functioning correctly prior to startup. The anticipatory reactor trips on loss of feedwater pumps and turbine trip have been added to reduce the number of challenges to the safety valves and power operated relief valve but have not been credited in the safety analyses.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #	001	A2.15
Importance Rating		4.2

Ability to (a) predict the impacts of the following malfunction or operations on the CRDS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Quadrant power tilt

Proposed Question: SRO Question # 16 (#91)

Technical Reference(s):
COLR, p 7, Rev 9
Tech Spec, p 3-34, Amend 273
Tech Spec, p 3-35a, Amend 274

Proposed References to be provided to applicants during examination: None

Learning Objective: 622-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
55.43 6

Comments:
The KA is matched because the operator must demonstrate the ability to predict the impacts of quadrant power tilt on the CRDS and, based on those predictions, use procedures to correct the consequences of the operation.

The question is at the Memory/Fundamental Knowledge cognitive level because the operator must recall bits of information found in the COLR and Tech Specs.

The question is at the SRO level because the candidates must demonstrate the ability to apply knowledge of TS bases for reactivity controls (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 9).

What MUST be known:
1. What is the COLR Maximum Quadrant Power tilt setpoint at 80% reactor power? 2. What is the Tech Spec Action and Basis for exceeding Quadrant Tilt?

92 **ID: 1147371** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% power 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.

Given the above information, identify the one selection below that describes:
(1) The impact, if any, on the operation of RC-RV-2, PORV, and
(2) The procedural actions required to mitigate the consequences of the failure.

A. (1) There is no impact to the PORV.
(2) Turn off Pressurizer heaters IAW OP-TM-AOP-043, Loss of Pressurizer.

B. (1) There is no impact to the PORV.
(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 PORV has failed.
(2) Turn off Pressurizer heaters IAW OP-TM-AOP-043, Loss of Pressurizer.

D. (1) Automatic operation of the PORV has 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 <329F 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:

- 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.
    - o 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:

- 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 <329F 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:

- 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:

- 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 # 17 (#92)

Technical Reference(s): TQ-TM-104-624-C001, pg 21, 24, Rev 3
OP-TM-MAP-G0308, pg 1, Rev 3
OS-24, pg 12, Rev 25

Proposed References to be provided to applicants during examination: None

Learning Objective: 624-GLO-14

Question Source:	Bank #	ID# 371278
	Modified Bank #	
	New	

Question History: Last NRC Exam: 10-02 (2012)

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).

What MUST be known:
1. What are the power supplied to the PORV? 2. What is the relationship between RPS and the PORV? 3. What are the procedural requirements for a failed PORV IAW G-3-8?

93 **ID: 1147556** **Points: 1.00**

Plant Conditions (Time = 0 minutes):

- The plant is operating at 98% reactor power.

Event (Time = 5 minutes):

- A Feed Line Rupture occurs.
- An ATWS occurs.
- HPI has been initiated.
- Reactor Power is 60% and lowering at 2%/minute.
- Subcooling Margin is 23F and lowering slowly.
- MAP G-1-1, Reactor Trip, is in alarm.
- MAP G-1-8, Tsat MArgin AB Lo, is in alarm.
- PPC Point C4015, OTSG A (B) Tube to Shell Delta T Hi/Lo, is in alarm.
    - OTSG Tube to Shell ΔT is 40F.
    - OTSG Tube to Shell ΔT is 80F.

Given the above information, which of the following correctly describes:
(1) The procedures that the crew have entered at Time = 5 minutes, and
(2) The procedures that the crew have entered at Time = 33 minutes?

A. (1) OP-TM-EOP-001, Reactor Trip, and
OP-TM-PPC-C4015, OTSG A (B) Tube to Shell Delta T Hi/Lo.
(2) OP-TM-EOP-010, Guide 14, Tube-to-Shell Delta-T Limit/Control, and
OP-TM-EOP-010, Rule 1, LSCM - Loss of Subcooling Margin (LSCM).

B. (1) OP-TM-EOP-001, Reactor Trip, and
OP-TM-EOP-010, Guide 14, Tube-to-Shell Delta-T Limit/Control.
(2) OP-TM-PPC-C4015, OTSG A (B) Tube to Shell Delta T Hi/Lo, and
OP-TM-EOP-010, Rule 1, LSCM - Loss of Subcooling Margin (LSCM).

C. (1) OP-TM-EOP-010, Rule 1, LSCM - Loss of Subcooling Margin (LSCM), and
OP-TM-EOP-010, Guide 14, Tube-to-Shell Delta-T Limit/Control.
(2) OP-TM-EOP-001, Reactor Trip, and
OP-TM-PPC-C4015, OTSG A (B) Tube to Shell Delta T Hi/Lo.

D. (1) OP-TM-EOP-010, Rule 1, LSCM - Loss of Subcooling Margin (LSCM), and
OP-TM-PPC-C4015, OTSG A (B) Tube to Shell Delta T Hi/Lo.
(2) OP-TM-EOP-001, Reactor Trip, and
OP-TM-EOP-010, Guide 14, Tube-to-Shell Delta-T Limit/Control.

Answer: A

Answer Explanation

A. Correct.

Time = 5 minutes: OP-TM-EOP-001, Reactor Trip, is entered to initiate HPI in order to shutdown the reactor. As this is a hold step, the CRS cannot move on to a symptom check. OP-TM-PPC-C4015, OTSG A (B) Tube to Shell Delta T Hi/Lo, is entered and cannot be exited until either the tube to shell delta-t is within limits or the reactor is shutdown.

Time = 33 minutes: The reactor has met shutdown conditions and therefore Rule 1, LSCM - Loss of Subcooling Margin (LSCM), which is not able to be entered until the reactor is shutdown, is now entered. Additionally, Guide 14, Tube-to-Shell Delta-T Limit/Control, can be entered now that the reactor is shutdown.

IAW OS-24, Conduct of Operations During Abnormal and Emergency Events:

> 3.15 REACTOR SHUTDOWN: The reactor is shutdown when the heat generation by fission has been stopped. This condition can be confirmed immediately following a reactor trip as follows:
> 1) Power Range Nuclear Instrumentation indicates less than 5%,
> 2) all control rods are inserted or
> 3) source range count rate is continuously lowering.

IAW OP-TM-EOP-001, Reactor Trip:

> 2.2 VERIFY REACTOR SHUTDOWN.
> RNO
> 1. TRIP both 1L-02 and 1G-02
> 2. If the REACTOR is SHUTDOWN, then GO TO step 2.3
>
> 3. If Main Feedwater is not available, then perform the following:
>   - ENSURE Main Turbine is tripped.
>   - INITIATE OP-TM-424-901, "Emergency Feedwater".
> 4. MAINTAIN primary-to-secondary heat transfer (PSHT).
> 5. When RCS pressure < 2500 psig, then INITIATE OP-TM-211-901, "Emergency Injection HPI/LPI".
> 6. When the REACTOR is SHUTDOWN, then CONTINUE.

IAW Rule 1, LSCM - Loss of Subcooling Margin (LSCM):

- IAAT SCM < 25F, and the REACTOR is SHUTDOWN, then perform the following:

IAW Guide 14, Tube-to-Shell Delta-T Limit/Control:

- IAAT the reactor is shutdown, then perform the following:

IAW OP-TM-PPC-C4015, OTSG A (B) Tube to Shell Delta T Hi/Lo:

> 4.0 MANUAL ACTIONS REQUIRED
> 4.1 If reactor is critical, then PERFORM the following...
> 4.2 If reactor is shutdown and EOP have been initiated, then GOTO Guide 14.

B. Incorrect.
Time = 5 minutes: Although it is correct that OP-TM-EOP-001, Reactor Trip, is entered to initiate HPI in order to shutdown the reactor, Guide 14, Tube-to-Shell Delta-T Limit/Control, cannot be entered until the reactor is shutdown, which will be at Time = 33 minutes. Plausible if the candidate is not familiar with the requirements to enter Guide 14, Tube-to-Shell Delta-T Limit/Control.

Time = 33 minutes: Although it is correct that Rule 1, LSCM - Loss of Subcooling Margin (LSCM), is entered when the reactor is shutdown (which occurs at Time = 33 minutes). Additionally, OP-TM-PPC-C4015, OTSG A (B) Tube to Shell Delta T Hi/Lo, is no longer in use because step 4.2 states:

4.2 If reactor is shutdown and EOP have been initiated, then GOTO Guide 14.

Plausible if the candidate is not familiar with the steps within OP-TM-PPC-C4015, OTSG A (B) Tube to Shell Delta T Hi/Lo.

C. Incorrect.
Time = 5 minutes: Neither Rule 1, LSCM - Loss of Subcooling Margin (LSCM), nor Guide 14, Tube-to-Shell Delta-T Limit/Control, can be entered until the reactor is shutdown, which will be at Time = 33 minutes. Plausible if the candidate is not familiar with the requirements to enter Rule 1, LSCM - Loss of Subcooling Margin (LSCM), or Guide 14, Tube-to-Shell Delta-T Limit/Control.

Time = 33 minutes: By this time, the reactor is shutdown and a symptom check has been performed. The symptom check will result in a routing out of OP-TM-EOP-001, Reactor Trip. Additionally, OP-TM-PPC-C4015, OTSG A (B) Tube to Shell Delta T Hi/Lo, is no longer in use because step 4.2 states:

4.2 If reactor is shutdown and EOP have been initiated, then GOTO Guide 14.

Plausible if the candidate is not familiar with the flowpath through OP-TM-EOP-001, Reactor Trip. or the steps within OP-TM-PPC-C4015,OTSG A (B) Tube to Shell Delta T Hi/Lo.

D. Incorrect.
Time = 5 minutes: Although OP-TM-PPC-C4015,OTSG A (B) Tube to Shell Delta T Hi/Lo, is entered and cannot be exited until either the tube to shell delta-t is within limits or the reactor is shutdown, Rule 1, LSCM - Loss of Subcooling Margin (LSCM), can not be entered until the reactor is shutdown, which will be at Time = 33 minutes. Plausible if the candidate is not familiar with the requirements to enter Rule 1, LSCM - Loss of Subcooling Margin (LSCM).

Time = 33 minutes: Although Guide 14, Tube-to-Shell Delta-T Limit/Control, can be entered now that the reactor is shutdown, but a symptom check has been performed. The symptom check will result in a routing out of OP-TM-EOP-001, Reactor Trip. Plausible if the candidate is not familiar with the flowpath through OP-TM-EOP-001, Reactor Trip.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	035	2.4.45
	Importance Rating		4.3

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

Proposed Question: SRO Question # 18 (#93)

Technical Reference(s):
OP-TM-PPC-C4015, pg 2, Rev 0
OP-TM-EOP-001, pg 1, Rev 12
OS-24, pg 5, Rev 25
OP-TM-EOP-010, p 3,26, Rev 17

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-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
55.43 5

Comments:
The KA is matched because the candidate must demonstrate the ability to prioritize and interpret the significance of each annunciator or alarm, including those associated with Steam Generators.

The question is at the Comprehension/Analysis cognitive level because the candidate is required to evaluate conditions given and then determine the correct path of procedure routing at two different time intervals.

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).

What MUST be known:
1. What are the entry and exit criteria for OP-TM-EOP-001, Reactor Trip?
2. What are the entry and exit criteria for OP-TM-PPC-C4015, OTSG A (B) Tube to Shell Delta T Hi/Lo?
3. What are the entry and exit criteria for Rule 1, LSCM - Loss of Subcooling Margin (LSCM)?
4. What are the entry and exit criteria for Guide 14, Tube-to-Shell Delta-T Limit/Control?
5. What are the reactor shutdown criteria IAW OS-24?

94 **ID: 1147373** **Points: 1.00**

Plant Conditions:

- The reactor is defueled during a refueling outage.
- No fuel handling is in progress.

Sequence of Events:

- Two hours into the shift, an Auxiliary Operator slips, hits her head, and becomes unconscious while travelling through the Reactor Building.
    - o She is 1 of 3 Auxiliary Operators (AO) on the shift.
- She is contaminated and is escorted to the hospital by the following personnel:
    - o 1 of the 2 available Rad Pro (RP) Technicians on shift.
    - o 1 of the 3 available Reactor Operators (RO) assigned to the shift.
- Fire Brigade requirements are met by separate onsite personnel.

Given the above information, which of the following is the action to take, if any, to ensure **Technical Specification** required minimum unit staffing?

A. Another AO should be called in to arrive within two hours.

B. Another RO should be called in to arrive within two hours.

C. Another RP technician should be called in to arrive within two hours.

D. No action is required because minimum staffing levels are still met for all positions.

Answer: D

Answer Explanation

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION[iii]

LICENSE CATEGORY QUALIFICATIONS	$T_{ave} > 200°$	$T_{ave} \leq 200°$
SRO[iv]	2	1[i]
RO[iv]	2	1
Non-Licensed Auxiliary Operator	2	1
Shift Technical Advisor	1[ii]	None Required

(i) Does not include the Licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising (a) irradiated fuel handling and transfer activities onsite, and (b) all unirradiated fuel handling and transfer activities to and from the Reactor Vessel.

(ii) May be on a different shift rotation than licensed personnel.

(iii) Except for the Shift Manager, shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an incoming shift crewman being late or absent.

OP-TM-112-101-1002 Shift Staffing Requirements requires 4 Auxiliary Operators. Tech Spec table 6.2-1 only requires 1 less than $\leq$ 200F

Correct Answer- No action is required as minimum staffing levels are still met.

ANSWER JUSTIFICATION:

A. Incorrect.
Since the AO taken to the hospital is 1 of 3, 2 will remain. Since the plant is less than 200F, the Tech Spec requirement is one AO. Therefore, the minimum Tech Spec requirement is satisfied. Plausible if the candidate confuses Tech Spec requirements with the additional requirements found in OP-TM-112-101-1002, Shift Staffing Requirements:

Plant > 200°F RCS Temperature	Plant < 200°F RCS Temperature
1 Shift Manager (SM)	1 Shift Manager (SM)
1 Control Room Supervisor (SRO)	1 Control Room Supervisor*
3 Control Room Operators*** (at least 2 RO)	2 Control Room Operators (at least 1 RO)
4 Auxiliary Operators**	4 Auxiliary Operators
1 Shift Technical Advisor (STA)	

B. Incorrect.
Since the RO accompanying to the hospital is 1 of 3, 2 will remain. Since the plant is less than 200F, the Tech Spec requirement is one RO. Therefore, the minimum Tech Spec requirement is satisfied. Plausible if the candidate confuses Tech Spec requirements with the additional requirements found in OP-TM-112-101-1002, Shift Staffing Requirements:

Plant > 200°F RCS Temperature	Plant < 200°F RCS Temperature
1 Shift Manager (SM)	1 Shift Manager (SM)
1 Control Room Supervisor (SRO)	1 Control Room Supervisor*
3 Control Room Operators*** (at least 2 RO)	2 Control Room Operators (at least 1 RO)
4 Auxiliary Operators**	4 Auxiliary Operators
1 Shift Technical Advisor (STA)	

C. Incorrect.
Since the RP Tech accompanying to the hospital is 1 of 2, 1 will remain. Since the plant is less than 200F, the Tech Spec requirement is no HP Techs. Therefore, the minimum Tech Spec requirement is satisfied. IAW Tech Spec 6.2.2.2:

- An individual qualified pursuant to 6.3.2 in radiation protection procedures shall be on site when fuel is in the reactor.

Plausible if the candidate confuses Tech Spec requirements with the additional requirements found in OP-TM-112-101-1002, Shift Staffing Requirements:

> 4.3.3: E-Plan minimum manning requirements for Chemistry/Rad Pro include 1 Chemistry Tech (CT) and 2 RP Techs (RPT), and a Shift Dose Assessor. The position of the Shift Dose Assessor may be filled by either a CT or a RPT provided they are qualified to fill the position. The CT and RPT at the Shift Briefing will inform Operations Management who is fulfilling these required roles. These individuals will be recorded with the Ops Manning in the attached checklist and in the Ops logs.

D. Correct.
Since, as described in the above distractor explanations, all of the positions meet the minimum Tech Spec requirements, the correct answer is: “no action is required".

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.1.5	
	Importance Rating		3.9

Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

Proposed Question: SRO Question # 19 (#94)

Technical Reference(s): Tech Spec, p. 6-1a-6-2, Amend 219

Proposed References to be provided to applicants during examination: None

Learning Objective: Prewatch DBIG APCO-1

Question Source: Bank # ID# 984144

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 1

Comments:
The KA is matched because the operator must demonstrate the ability to use procedures related to shift staffing such as minimum crew complement, overtime limitations.

The question is at the Analysis cognitive level because the candidate must be able to analyze and interpret plant conditions to determine shift priorities, procedural requirements, and tech spec limitations with crew minimum staffing.

The question is at the SRO level because the candidates must demonstrate the required actions for not meeting administrative controls listed in Technical Specification (TS) Section 5 or 6, depending on the facility (e.g., shift staffing requirements) (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3). Additionally, IAW OP-TM-112-101-1002, It is the responsibility of the Shift Manager on duty to ensure these manning requirements are met at all times.

What MUST be known:
1. What are the AO Tech Spec minimum manning requirements for conditions <200F? 2. What are the RO Tech Spec minimum manning requirements for conditions <200F? 3. What are the RP Technician Tech Spec minimum manning requirements for conditions <200F?

95 **ID: 1147374** **Points: 1.00**

Plant Conditions:

- Refueling operations are in progress.

Given the above information, which one of the following is required to be done by the Licensed Fuel Handling Supervisor (LFHS) in charge of refueling in accordance with 1505-1, Refueling Operations?

A. Give permission to lower Fuel Hoist onto an assembly.

B. Authorize turnover on the Main Bridge with an assembly engaged.

C. Give permission prior to unlatching a fuel assembly in the Reactor Vessel.

D. Be present in Containment OR the Spent Fuel Pool Room in constant communications with the refueling team prior to starting core unload or reload.

Answer: C

Answer Explanation

Explanation (Optional):

A. Incorrect.
Plausible as 1507-3 requires LFHS and spotter verification of correct location, not permission to lower Fuel Hoist.

B. Incorrect.
1505-1 step 5.5.12 Operation Director or designee permission required if turnover must occur with the mast in other than up disengaged.

C. Correct.
To give permission prior to unlatching a fuel assembly in the Reactor Vessel. IAW 1505-1, Fuel and Control Component Shuffles:

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.

D. Incorrect.
The SRO in charge of refueling must be in Containment (SFP is not allowed) prior to unload or reload per 1505-1 step 2.5.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		1
K/A #	2.1.35	
Importance Rating		3.9

Knowledge of the fuel-handling responsibilities of SROs.

Proposed Question: SRO Question # 20 (#95)

Technical Reference(s): 1505-1, p 9, Rev 58

Proposed References to be provided to applicants during examination: None

Learning Objective:	252-GLO-15.43		
Question Source:	Bank #	ID# 575119	
	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 operator must demonstrate the knowledge of the Fuel Handling Supervisor's responsibilities during fuel movement.

The question is at the Memory cognitive level because the operator must remember the SRO responsibilities for Fuel Handling.

The question is SRO-ONLY because it requires knowledge of fuel handling facilities and procedures, specifically Refuel floor SRO responsibilities (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3).

What MUST be known:
1. What are the SRO responsibilities associated with Fuel Handling?

96 **ID: 1147571** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- 10 EFPD after a refueling outage.

Event:

- While reviewing system records, the Decay Heat System engineer reports that the following surveillance was NOT completed during the last refueling outage:
  - OP-TM-212-211, LPI Test of DH Train A.
- The Three Mile Island Surveillance Frequency Control Program states that the frequency for the above surveillance is each refueling interval.

Based on the above information, "A" Decay Heat Pump, DH-P-1A, is ___(1)___ and the Technical Specifications require the plant to ___(2)___.

A. (1) Operable
(2) perform the surveillance within the next 24 hours

B. (1) Operable
(2) perform a risk evaluation, manage the risk impact, and perform the surveillance at the first reasonable opportunity

C. (1) Inoperable
(2) perform the surveillance within the next 24 hours

D. (1) Inoperable
(2) perform a risk evaluation, manage the risk impact, and perform the surveillance at the first reasonable opportunity

Answer: B

Answer Explanation

The need to perform surveillances on the Low Pressure Injection system (which is the same system as Decay Heat Removal at Three Mile Island) is covered by Tech Spec 4.5.2.2:

> 4.5.2.2 Low Pressure Injection
>
> a. At the frequency specified in the Surveillance Frequency Control Program and following maintenance or modification that affects system flow characteristics, system pumps and high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable. The auxiliaries required for low pressure injection are all included in the emergency loading sequence test specified in 4.5.1.
>
> b. The test will be considered satisfactory if the decay heat pumps have been successfully started and the decay heat injection valves and the decay heat supply valves have completed their travel as evidenced by the control board component operating lights. Flow shall be verified to be equal to or greater than the flow assumed in the Safety Analysis for the single corresponding RCS pressure used in the test.
>
> c. When the Decay Heat System is required to be operable, the correct position of DH-V-19A/B shall be verified by observation within four hours of each valve stroking operation or valve maintenance which affects the position indicator.

A. Incorrect.

Part 1 is correct. IAW Tech Spec 4.0.2 Bases:

- SR 4.0.2 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance has not been completed within the specified frequency.

Part 2 is incorrect. Plausible if the candidate is not familiar with Tech Spec 4.0.2 bases and believes that a 24 hour window is required. IAW Tech Spec 4.0.2:

- A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time that it is discovered that the required surveillance has not been performed in accordance with Surveillance Standard 4.0.2 and not at the time that the specified frequency was not met.

B. Correct.

Part 1 is correct. IAW Tech Spec 4.0.2 Bases:

- SR 4.0.2 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance has not been completed within the specified frequency.

Part 2 is correct. IAW Tech Spec 4.0.2 Bases:

- When a surveillance with a frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering power operation after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, Surveillance Standard 4.0.2 allows for the full delay period of up to the specified frequency to perform the surveillance. However, since there is not a time interval specified, the missed surveillance should be performed at the first reasonable opportunity.

C. Incorrect.
Part 1 is not correct. Plausible if the candidate is not familiar with Tech Spec 4.0.2. IAW Tech Spec 4.0.2 Bases:

- SR 4.0.2 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance has not been completed within the specified frequency.

Part 2 is incorrect. Plausible if the candidate is not familiar with Tech Spec 4.0.2 bases and believes that a 24 hour window is required. IAW Tech Spec 4.0.2:

- A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time that it is discovered that the required surveillance has not been performed in accordance with Surveillance Standard 4.0.2 and not at the time that the specified frequency was not met.

D. Incorrect.
Part 1 is not correct. Plausible if the candidate is not familiar with Tech Spec 4.0.2. IAW Tech Spec 4.0.2 Bases:

- SR 4.0.2 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance has not been completed within the specified frequency.

Part 2 is correct. IAW Tech Spec 4.0.2 Bases:

- When a surveillance with a frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering power operation after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, Surveillance Standard 4.0.2 allows for the full delay period of up to the specified frequency to perform the surveillance. However, since there is not a time interval specified, the missed surveillance should be performed at the first reasonable opportunity.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		2
K/A #	2.2.21	
Importance Rating		4.1

Knowledge of pre- and post-maintenance operability requirements.

Proposed Question: SRO Question # 21 (#96)

Technical Reference(s): Tech Spec, p 4-1a-4-1b, Amend 256

Proposed References to be provided to applicants during examination: None

Learning Objective: 212-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

55.43 2

Comments:
The KA is matched because the operator must demonstrate the knowledge of Post-maintenance operability requirements.

The question is at the Memory cognitive level because the operator must remember the Tech Spec bases requirements for a missed surveillance.

The question is SRO-ONLY because it requires the Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Source: Clarification Guidance for SRO-only Questions, Rev 1, Page 3). Additionally, it is an SRO responsibility to make an operability determination.

What MUST be known:
1. What are the operability determinations associated with a missed surveillance?
2. What is the time interval allowed to perform a missed surveillance?

97 **ID: 1147568** **Points: 1.00**

Plant Conditions:

- The plant is at the end of a Refueling Outage.

Event:

- During an outage, it is determined that one of the pressurizer code safety valve's lift pressure was determined to be 2605 psig.

Based on the above information, complete the following statement: The reactor can not be made critical until the pressurizer safety valve is restored to lift within ___(1)___ psig because two Pressurizer code safety valves are required ___(2)___.

A. (1) 2400-2600
(2) to prevent overpressure for some transients

B. (1) 2400-2600
(2) for redundancy to prevent the failure of one valve allowing an overpressure condition

C. (1) 2475-2525
(2) to prevent overpressure for some transients

D. (1) 2475-2525
(2) for redundancy to prevent the failure of one valve allowing an overpressure condition

Answer: C

Answer Explanation

A. Incorrect.

Part 1 is incorrect but plausible for two reasons. First, Tech Spec 3.1.1.3 basis acknowledges up to a 3% drift, which would be a larger band than the correct answer. Second, a band of +/- 100 psig is believable since it is a common number. IAW Tech Spec 3.1.1.3 Bases:

- Pressurizer code safety valve setpoint drift of up to 3% is acceptable in accordance with the assumptions of the TMI-1 safety analysis

Part 2 is correct. 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 but plausible for two reasons. First, Tech Spec 3.1.1.3 basis acknowledges up to a 3% drift, which would be a larger band than the correct answer. Second, a band of +/- 100 psig is believable since it is a common number. IAW Tech Spec 3.1.1.3 Bases:

- Pressurizer code safety valve setpoint drift of up to 3% is acceptable in accordance with the assumptions of the TMI-1 safety analysis

Part 2 is incorrect. Plausible if the candidate confuses the requirement when shutdown with that of when critical. 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. Correct.

Part 1 is correct. Adding 1% to 2500 equals 2525. Subtracting 1% from 2500 equals 2475. IAW Tech Spec 3.1.1.3:

- The reactor shall not remain critical unless both pressurizer code safety valves are operable with a lift setting of 2500 psig ± 1%.

Part 2 is correct. 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. Incorrect.
Part 1 is correct. Adding 1% to 2500 equals 2525. Subtracting 1% from 2500 equals 2475. IAW Tech Spec 3.1.1.3:

- The reactor shall not remain critical unless both pressurizer code safety valves are operable with a lift setting of 2500 psig ± 1%.

Part 2 is incorrect. Plausible if the candidate confuses the requirement when shutdown with that of when critical. 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 #		3
	Group #		2
	K/A #	2.2.25	
	Importance Rating		4.2

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Proposed Question: SRO Question # 22 (#97)

Technical Reference(s): Tech Spec, p 3-1b, Amend 279
Tech Spec, p 3-2, Amend 266

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO-14

Question Source: Bank # ID# 503814

Modified Bank #

New

Question History: Last NRC Exam: N/A

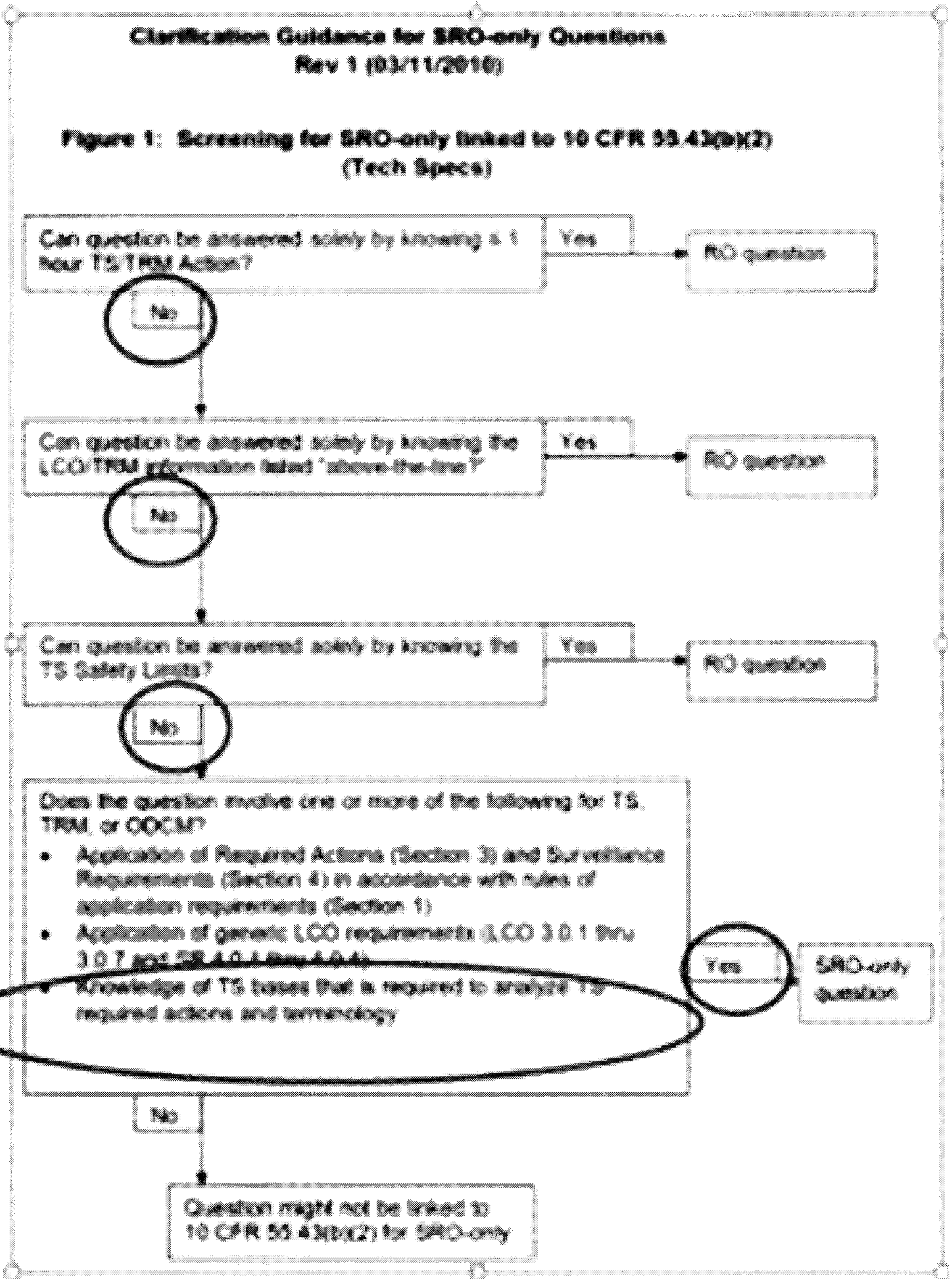
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	2

Comments:
The KA is matched because the operator must demonstrate the knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

The question is at the Memory cognitive level because the operator must recall the Tech Spec setpoint and bases associated with a limiting condition for operation and safety limits.

Although part 1 can be answered by an RO, the question is at the SRO level because the candidates must demonstrate the Knowledge of TS bases that are required to analyze TS required actions and terminology (Source: Clarification Guidance for SRO-only Questions, Rev 1, Pages 3-4).

What MUST be known:
1. What is the relief setpoint range for the pressurizer code safety valves while critical?   2. What is the bases for having two pressurizer code safety valves opearble while at power?



Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)
Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)
Can question be answered solely by knowing ≤ 1 hour TS/TRM Action?
Yes
RO question
No
Can question be answered solely by knowing the LCO/TRM information listed "above-the-line"?
Yes
RO question
No
Can question be answered solely by knowing the TS Safety Limits?
Yes
RO question
No
Does the question involve one or more of the following for TS, TRM, or ODCM?
• Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
• Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
• Knowledge of TS bases that is required to analyze TS required actions and terminology
Yes
SRO-only question
No
Question might not be linked to 10 CFR 55.43(b)(2) for SRO-only

98 **ID: 1147559** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor power.
- The "A" Waste Decay Tank, WDG-T-1A, was approved for release on the previous shift.

Sequence of Events:

- **Prior to initiation**, the release was **cancelled** due to cyclic problems with an inverter that feeds multiple radiation monitors.
- The cyclic inverter problem has been resolved.
    - o All radiation monitors are operable.

Given the above information, which one of the following is correct with regards to initiating the release of WDG-T-1A?

A. The existing permit and all attachments must be cancelled and a new permit must be issued.

B. Verify that WDG-T-1A has remained isolated and continue to use the existing permit as long as 12 hours has not passed from the approval time on the permit.

C. The exisitng permit must be cancelled but the dose calculation attachments from the existing permit can be re-used upon verifying that WDG-T-1A has remained isolated.

D. Obtain approval from the RP Supervisor and Shift Manager to continue to use the existing permit as long as 48 hours has not passed from the gas sample associated with the permit.

Answer: A

Answer Explanation

Explanation (Optional):

A. **Correct.**
IAW CY-TM-170-2011, Step 4.2:

> IAAT a gas release is canceled prior to release, then RECORD the reason on the permit and SEND the canceled permit to ODCM Specialist to be reviewed, updated and filed. UPDATE appropriate attachment in CY-TM-170-2004, Gaseous Radioactive Release Records to designate that particular release as canceled.

B. **Incorrect.**
Plausible if the candidate is not familiar with CY-TM-170-2011, Step 4.2. Additionally, there are other release which allow for up to 12 hours to resume a release IF it was started. Lastly, the previous procedure (which was replaced with CY-TM-170-2011) allowed restart within 12 hours if the release was pause while in progress.

C. **Incorrect.**
Plausible if the candidate is not familiar with CY-TM-170-2011, as there is a similar step within the procedure. IAW CY-TM-170-2011:

> 3.4.3 If Waste Gas Decay Tank release does not start within 48 hours of gas sample time, then CONFIRM that tank isolation valves have remained closed and tank pressure did not increase more than 1psig.

D. **Incorrect.**
Plausible if the candidate is not familiar with CY-TM-170-2011, but there is no caveat allowing for supervision permission to maintain the same permit.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	2.3.6	
	Importance Rating		3.8

Ability to approve release permits.

Proposed Question: SRO Question # 23 (#98)

Technical Reference(s): CY-TM-170-2011, p 4, Rev 0

Proposed References to be provided to applicants during examination: None

Learning Objective: NOP DBIG-APCO-1

Question Source: Bank # ID# 862349

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 4

Comments:
The KA is matched because the operator must demonstrate the ability to approve release permits (i.e. by requiring the application of knowledge regarding the continuing validity of approved radioactive release permits).

The question is at the Comprehension/Analysis cognitive level because the operator must analyze the given conditions and then determine the correct course of action based on the timeline of the permit approval/release.

Although part 1 can be answered by an RO, the question is at the SRO level because the candidate must demonstrate the Knowledge of the process for gaseous/liquid release approvals, i.e., release permits.(Source: Clarification Guidance for SRO-only Questions, Rev 1, Pages 6). Additionally, the question is SRO-Only because the responsibility for approval and application of release permits rests with the Shift Manager. Failure to comply with the facility SRO administrative responsibilities would lead to an unapproved release.

What MUST be known:
1. What are the actions with regards to a release permit, when the evolution is cancelled prior to the release?

99 **ID: 1147570** **Points: 1.00**

Plant Conditions:

- The plant is operating at 100% reactor 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, failed to 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.

Based on the above information:
(1) How will the CRS address Intermediate Closed Cooling Water, and
(2) What procedure addresses the 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. Starting IC-P-1B may cause damage to the RCP seals. Plausible if the candidate does not recognize that MU-V-32 having inadequate flow is a decision point within the procedure to not simply start IC-P-1B (which is the next step if MU-V-32 had adequate flow).

Part (2) is correct. OP-TM-EOP-020 Attachment 10, Step 1.4 addresses the situation from the RSD Panel. IAW OP-TM-EOP-020, Cooldown From Outside of Control Room:

- ATTACHMENT 10 - RSD INITIAL LINEUP
  1.4 VERIFY MU-V-32 INLET FLOW STATUS INADEQUATE light is not lit.
  RNO:
    1. ENSURE MU-V-20 is Closed.
    2. If 1E 4160V Bus was de-energized, then perform the following:
       A. CLOSE MU-V-189 (281' AB Behind Seal Return Coolers).
       B. STOP IC-P-1B.
       C. INITIATE OP-TM-226-901,"Loss of all RCP Seal Cooling".
       D. GO TO Step 1.7.

B. Incorrect.

Part 1 is incorrect. Starting IC-P-1B may cause damage to the RCP seals. Plausible if the candidate does not recognize that MU-V-32 having inadequate flow is a decision point within the procedure to not simply start IC-P-1B (which is the next step if MU-V-32 had adequate flow).

Part 2 is not correct. Plausible since OP-TM-AOP-032 deals with a Loss of ICCW. However, the procedure does not address actions to take from outside of the Control Room.

C. Correct.

Part 1 is correct. IAW OP-TM-EOP-020, Cooldown From Outside of Control Room:

3.2 INITIATE Attachment 10 "RSD Initial Lineup".

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

Step 3.2: After control is transferred to RSD, Attachment 10 is initiated to ensure equipment is properly lined up to mitigate worst-case fire effects. The fire may have affected control circuits before RSD transfer occurred, resulting in trip or spurious operation of equipment. This Attachment will ensure that required equipment is correctly aligned and operating. A redundant step to initiate

- The Attachment is designed to meet the timeline described in the FHAR for a worst-case Appendix R fire. The first step ensures that MU-V-3 is closed to help mitigate a potential overcooling if letdown was not isolated.

- Next, 1E 4160V bus power is verified. If unavailable, Attachment 9 is directed to start EG-Y-1B, since this is vital to performance of the rest of the RSD lineup.

- If no seal injection flow exists as indicated by the inadequate seal injection flow light on the RSP, MU-V-20 is ensured closed to isolate seal injection. All RCP seal cooling may or may not have been lost, depending on whether power was maintained to the 1E 4160V bus (which would maintain power to IC-P-1B).

- If EG-Y-1B was required to be stated 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.

- If 1E 4160V power was maintained, then ICCW will be ensured operating with IC-V-2,3,4 open.

Part (2) is correct. OP-TM-EOP-020 Attachment 10, Step 1.4 addresses the situation from the RSD Panel. IAW OP-TM-EOP-020, Cooldown From Outside of Control Room:

- ATTACHMENT 10 - RSD INITIAL LINEUP
  1.4 VERIFY MU-V-32 INLET FLOW STATUS INADEQUATE light is not lit.
  RNO:
    1. ENSURE MU-V-20 is Closed.
    2. If 1E 4160V Bus was de-energized, then perform the following:
       A. CLOSE MU-V-189 (281' AB Behind Seal Return Coolers).
       B. STOP IC-P-1B.
       C. INITIATE OP-TM-226-901, "Loss of all RCP Seal Cooling".
       D. GO TO Step 1.7.

D. Incorrect.
Part 1 is correct. IAW OP-TM-EOP-020, Cooldown From Outside of Control Room:

3.2 INITIATE Attachment 10 "RSD Initial Lineup".

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

Step 3.2: After control is transferred to RSD, Attachment 10 is initiated to ensure equipment is properly lined up to mitigate worst-case fire effects. The fire may have affected control circuits before RSD transfer occurred, resulting in trip or spurious operation of equipment. This Attachment will ensure that required equipment is correctly aligned and operating. A redundant step to initiate

- The Attachment is designed to meet the timeline described in the FHAR for a worst-case Appendix R fire. The first step ensures that MU-V-3 is closed to help mitigate a potential overcooling if letdown was not isolated.

- Next, 1E 4160V bus power is verified. If unavailable, Attachment 9 is directed to start EG-Y-1B, since this is vital to performance of the rest of the RSD lineup.

- If no seal injection flow exists as indicated by the inadequate seal injection flow light on the RSP, MU-V-20 is ensured closed to isolate seal injection. All RCP seal cooling may or may not have been lost, depending on whether power was maintained to the 1E 4160V bus (which would maintain power to IC-P-1B).

- If EG-Y-1B was required to be stated 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.
- If 1E 4160V power was maintained, then ICCW will be ensured operating with IC-V-2,3,4 open.

Part 2 is not correct. Plausible since OP-TM-AOP-032 deals with a Loss of ICCW. However, the procedure does not address actions to take from outside of the Control Room.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.18	
	Importance Rating		4.0

Knowledge of the specific bases for EOPs.

Proposed Question: SRO Question # 24 (#99)

Technical Reference(s): OP-TM-EOP-0201, p 18, Rev 11
OP-TM-EOP-020, p 7,81, Rev 18

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP020-PCO-2

Question Source: Bank # ID# 859996
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 operator must demonstrate knowledge of the specific bases for EOPs (specifically EOP-020).

The question is at the Comprehension/Analysis cognitive level because the operator must analyze the given conditions and then determine the correct course of action, including the reason according to procedure basis.

The question is at the SRO level because the candidate must demonstrate the knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps (Source: Clarification Guidance for SRO-only Questions, Rev 1, Pages 6).

What MUST be known:
1. What are the applicable actions of Attachments within OP-TM-EOP-020?
2. What is the basis for actions taken regarding ICCW upon a Cooldown for Outside of the Control Room?
3. What is the applicability of OP-TM-AOP-032 vice OP-TM-EOP-020?

100 **ID: 1147560** **Points: 1.00**

Plant Conditions:

- The plant is at 100% power.

Event:

- Due to an improper ventilation lineup, smoke from welding in an adjacent area causes all of the smoke detectors in the Auxiliary Building, Elevation 281', in the Pipe Penetration Area to go into ALARM.

Given the above information, complete the statement below:

Fire System water ___(1)___ flow onto the equipment in the area. When the system is isolated for restoration the operator must ensure that a ___(2)___ is established within one hour, until the system is returned to OPERABLE status.

A. (1) will
(2) firewatch patrol to inspect the zone

B. (1) will
(2) continuous firewatch with backup suppression equipment

C. (1) will not
(2) firewatch patrol to inspect the zone

D. (1) will not
(2) continuous firewatch with backup suppression equipment

Answer: D

Answer Explanation

A. **Incorrect.** 1st part wrong, 2nd part wrong. This is plausible because if the operator incorrectly believes that this area is protected by a Deluge system, and there are many such systems within the plant, then this answer would be correct (See B); and because the operator may incorrectly believe that this is the appropriate action for the system being out of service (See C).

B. **Incorrect.** 1st part wrong, 2nd part correct. This is plausible because according to TQ-TM-104-810-C001, a Deluge system consists a deluge valve which when opened automatically or manually will allow Fire Service Water flow through opened ended nozzles onto the protected equipment/area. If the operator incorrectly believes that this area is protected by a Deluge system, and there are many such systems within the plant, then this answer would be correct.

C. **Incorrect.** 1st part correct, 2nd part wrong. This is plausible because according to AP-1038, Exhibit 2, with the number of inoperable fire detections instruments less than required by Table 1, the operator must establish a Firewatch patrol to inspect the affected Zone at least once per hour. The operator may incorrectly believe that this is the appropriate action for the system being out of service as well.

D. **Correct.** 1st part correct, 2nd part correct. According to AP-1038, Exhibit 2, this area is protected by a Pre-Action System. According to TQ-TM-104-810-C001, a Preaction System requires two independent actions to take place. First of all, a multimatic valve or a 2 inch flooding valve must open and charge the previously dry water header, and then fusible link sprinkler heads must melt to allow water flow. According to OP-TM-1104-45E, one fire alarm in Group 1 and one fire alarm in Group 2 will cause the Preaction System to actuate. According to AP-1038, Table 1, there are five smoke detectors to protect this area, three in Group 1 and two in Group 2. Since all of them have alarmed, the system will actuate and pressurize the previously dry water header up to the fusible link sprinkler heads. For water to be placed onto the equipment in the area, there must be a fire to generate enough heat to melt the fusible link sprinkler heads. According to AP-1038, Exhibit 2, when a system of this nature is inoperable, as is the case when the system is isolated to be restored, the operator must establish a continuous firewatch with backup suppression equipment within one hour. This is also reflected in the system procedure. According to OP-TM-1104-45E, suppression equipment is required within one hour per AP 1038, Exhibit 2, Section 3.3.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.25	
	Importance Rating		3.7

Knowledge of fire protection procedures.

Proposed Question: SRO Question # 25 (#100)

Technical Reference(s): AP-1038, p 32,35-38, Rev 82
TQ-TM-104-810-C001, p14-16, Rev 9
OP-TM-1104-45E, p16, Rev 13

Proposed References to be provided to applicants during examination: None

Learning Objective: 252-15.43

Question Source: Bank # ID# 860161

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 1

Comments:
The KA is matched because the operator must demonstrate knowledge of fire protection procedures.

The question is at the Comprehension/Analysis cognitive level because the operator must demonstrate an understanding of how a Preaction System works.

The question is at the SRO level because the candidate must demonstrate the 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). There are no longer any specific Fire Suppression System related Tech Specs, except in the definition section. However, the NRC has required –through conditions specified in the Facility Operating License Docket # 50-289 - that the company shall "...maintain in effect all provisions of the Fire Protection Program as described in the Updated FSAR...". This requires that the operator treat the Fire System as a system similar to other SSCs. The requirements identified in the FSAR are identified in the Technical Requirements Manual (TRM), which is incorporated into AP-1038.

What MUST be known:
1. What does the Fire System Deluge system consist of?

2. What does the Fire System Preaction System consist of?
3. What are the 1038 requirements associated with a portion of the Fire Suppression System inoperable?