EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

1 **ID: 2087125** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Reactor Trip
- OP-TM-EOP-001, REACTOR TRIP IMA's are complete, VSSV's are in progress
- Currently:
  - OTSG A level is 98% and rising
  - OTSG B level is 50% and lowering

What action(s) must the crew take in accordance with OP-TM-EOP-001 and what is the basis of the action?

A. (1) Place FW-V-16A and FW-V-17A Feedwater Control Valves in Hand and close
(2) Minimize overcooling effects

B. (1) Place FW-V-16A and FW-V-17A Feedwater Control Valves in Hand and close
(2) Minimize possible water carryover or main steam line flooding

C. (1) Trip FW-P-1A and FW-P-1B, Main Feedwater Pumps
(2) Minimize possible water carryover or main steam line flooding

D. (1) Trip FW-P-1A and FW-P-1B, Main Feedwater Pumps
(2) Minimize overcooling effects

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-EOP-001, REACTOR TRIP (Rev 16, Page 5) Step 3.5 directs the crew to verify both OTSG levels are less than 97.5% in the operating range; (2) if they are not then ENSURE FW-P-1A and FW-P-1B are tripped; (3) The OP-TM EOP-0011 REACTOR TRIP BASIS DOCUMENT (Rev 9, Page 10) Step 3.5 states that if feedwater is not being controlled and exceeds 97.5% then both Main Feedwater Pumps are tripped to stop the overfeed and minimize the chance of water carryover to the main steam lines.

A.	(1) Place FW-V-16A and FW-V-17A Feedwater Control Valves in Hand and close (2) Minimize overcooling effects	INCORRECT: (1) Plausible because this will stop feedwater flow to the 'A' OTSG. Incorrect because OP-TM-EOP-001 directs tripping both Main Feedwater Pumps. (2) Plausible because overfeeding could lead to overcooling. Incorrect because at this level the basis for this action is to prevent water carryover.
B.	(1) Place FW-V-16A and FW-V-17A Feedwater Control Valves in Hand and close (2) Minimize possible water carryover or main steam line flooding	INCORRECT: (1) Plausible because this will stop feedwater flow to the 'A' OTSG. Incorrect because OP-TM-EOP-001 directs tripping both Main Feedwater Pumps. (2) Correct Answer.
C.	(1) Trip FW-P-1A and FW-P-1B, Main Feedwater Pumps (2) Minimize possible water carryover or main steam line flooding	CORRECT: See above.
D.	(1) Trip FW-P-1A and FW-P-1B, Main Feedwater Pumps (2) Minimize overcooling effects	INCORRECT: (1) Correct action. (2) Plausible because overfeeding could lead to overcooling. Incorrect because at this level the basis for this action is to prevent water carryover.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	007	K3.01
	Importance Rating	4.0	

K/A: Reactor Trip: Knowledge of the reasons for the following as they apply to a reactor trip: Actions contained in EOP for reactor trip.

Proposed Question:	Question #1	
Technical Reference(s):	OP-TM-EOP-001, Rev 16	OP-TM-EOP-0011, Rev 9

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EOP001-PCO-1	

Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	Last NRC Exam:	N/A	

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

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the reason for a step performed in the Reactor Trip EOP.

High Cog: This question is high cog because the examinee must identify an abnormal condition and then determine which of the actions must be taken in accordance with the procedure. The examinee must correlate a high level in one OTSG to tripping both MFW pumps. The examinee must know the basis behind that action.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

2 **ID: 2096966** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Parameters indicate an increase of 0.25 gpm leakage into the RC Drain Tank
- RCS pressure steady at 2155 psig (Tsat = 648F)
- Ambient temperature condition at RC-RV-2, PORV, tailpipe is 100 degrees F

Based on these conditions identify the ONE selection below that describes the operation of computer alarm A0517, RC-RV-2 TAILPIPE DELTA TEMP.

If hot fluid is flowing from the PORV, A0517 will alarm at the PPC sensed **MINIMUM** temperature of _____ .

A. 130° F

B. 140° F

C. 608° F

D. 618° F

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The setpoint of PPC A0517 is 30F above ambient temperature at the tailpipe (100F in this case), so it will alarm at 130F tailpipe ambient temperature. (OP-TM-PPC-A0517, Rev 4 Page 1)

A.	130F	CORRECT: See above
B.	140F	INCORRECT: Plausible because anticipated setpoint is 40F above ambient temperature. Incorrect because the PPC takes into account a 10F bias to be conservative so the alarm comes in at 30F above ambient.
C.	608F	INCORRECT: Plausible because this is 40F below the saturation temperature of the Pressurizer (~648F). The examinee could believe the delta temperature setpoint is 40F below the Pressurizer saturation temperature as this would be closer to the actual temperature through an open PORV. Incorrect because it is 30F above ambient temperature.
D.	618F	INCORRECT: Plausible because this is 30F below the saturation temperature of the Pressurizer (~648F). The examinee could believe the delta temperature setpoint is 30F (with the 10F bias) below the Pressurizer saturation temperature as this would be closer to the actual temperature through an open PORV. Incorrect because it is 30F above ambient temperature

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	008	AA2.25	
	Importance Rating	3.2		

K/A: Pressurizer Vapor Space Accident: Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: High-Temperature Computer alarm and alarm type

Proposed Question:	Question #2	
Technical Reference(s):	OP-TM-PPC-A0517, Rev 4	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	223-GLO-6	

Question Source:	Bank #			
	Modified Bank #	363650		
	New			
Question History:	Sim Exam 6	Last NRC Exam:	N/A	

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

Comments:

KA Match: This question matches the KA because the examinee must have the ability to determine if the computer alarm will sense a pressurizer steam space leak pas the code safety.

High Cog: This question is high cog because the examinee must know the location that the alarm is based on and what the alarm setpoint is. The examinee calculate to get the correct answer.

Plant Conditions:

- Reactor is operating at 100% power with ICS in full automatic.
- Parameters indicate an increase of 0.25 gpm leakage into the RC Drain Tank.
- RCS pressure steady at 2155 psig.
- Ambient temperature condition at RC-RV-2, PORV, tailpipe is 100 degrees F.

Based on these conditions identify the ONE selection below that describes the operation of computer alarm A0517, RC-RV-2 TAILPIPE DELTA TEMP set at 30 degrees F and the minimum 19kw PZR heater margin.

If hot fluid is flowing from the PORV, A0517 will alarm at the PPC sensed **MINIMUM** temperature of ___(1)___. A steam space leak of 0.25 gpm ___(2)___ exceed the minimum 19kw PZR heater margin.

A. (1) 130° F
(2) does

B. (1) 130° F
(2) does not

C. (1) 618° F
(2) does

D. (1) 618° F
(2) does not

Answer A

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

3 **ID: 2085111** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- 3 gpm RCS leak into containment
- OP-TM-AOP-050 REACTOR COOLANT LEAKAGE has been entered

Event:

- Reactor power is 88% and lowering
- RCS Pressure is 1950 psig and lowering
- RB Pressure is 1.1 psig and rising
- RCS leak rate continues to rise
- HPI is initiated

(1) Which procedure will provide the mitigation strategy after HPI is initiated?
(2) What is the basis for using this procedure?

A. (1) Go to OP-TM-EOP-001, REACTOR TRIP
(2) The amount of leakage has exceeded the limit requiring a Reactor Trip

B. (1) Go to OP-TM-EOP-001, REACTOR TRIP
(2) To limit fuel centerline temperature to prevent zircaloy-water reaction

C. (1) Continue in OP-TM-AOP-050, REACTOR COOLANT LEAKAGE
(2) The amount of leakage has NOT caused an RPS trip setpoint to be exceeded

D. (1) Continue in OP-TM-AOP-050, REACTOR COOLANT LEAKAGE
(2) To expedite shutting down the reactor

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The initial plant conditions include an RCS leak, so the crew is implementing actions in OP-TM-AOP-050, REACTOR COOLANT LEAKAGE. OP-TM-AOP-050 (Rev 7, Page 3) Step 3.3 directs that if HPI is required to maintain pressurizer level, to initiate HPI and then trip the reactor and go to OP-TM-EOP-001, REACTOR TRIP; (2) The intent for this step, in accordance with OP-TM-AOP-0501 RCS LEAKAGE BASIS DOCUMENT (Rev 7, Page 5) Step 3.3, is to cover the range of leakage larger than the Technical Specification LCO but less than the range of leakage requiring a reactor trip.

A.	(1) Go to OP-TM-EOP-001, REACTOR TRIP (2) The amount of leakage has exceeded the limit requiring a Reactor Trip	CORRECT: See above.
B.	(1) Go to OP-TM-EOP-001, REACTOR TRIP (2) To limit fuel centerline temperature to prevent zircaloy-water reaction	INCORRECT: (1) Correct Answer. (2) Plausible because a large break loss of coolant accident could lead to a zircaloy-water reaction if equipment failed or malfunctioned and the core became uncovered. Incorrect because there is no indication in the stem to indicate that any equipment failed and/or the leak is significant enough for that reaction.
C.	(1) Continue in OP-TM-AOP-050, REACTOR COOLANT LEAKAGE (2) The amount of leakage has NOT caused an RPS trip setpoint to be exceeded	INCORRECT: (1) Plausible because some AOP and EOP procedures contain mitigation steps after they direct a reactor trip. Incorrect because OP-TM-AOP-050 is not one of those procedures. (2) Plausible because the examinee could believe that the only reason to trip the reactor is because an RPS trip setpoint was exceeded. Incorrect because in accordance with OP-TM-AOP-050, the reactor is tripped when HPI after is initiated.
D.	(1) Continue in OP-TM-AOP-050, REACTOR COOLANT LEAKAGE (2) To expedite shutting down the reactor	INCORRECT: (1) Plausible because some AOP and EOP procedures contain mitigation steps after they direct a reactor trip. Incorrect because OP-TM-AOP-050 is not one of those procedures. (2) Plausible if the examinee believes that the OP-TM-AOP-050 mitigation strategy includes an expedited shutdown on with HPI.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	009	2.4.18	
	Importance Rating	3.3		

K/A: Small Break LOCA: Knowledge of specific bases for EOPs

Proposed Question:	Question #3	
Technical Reference(s):	OP-TM-AOP-050, Rev 7	OP-TM-AOP-0501, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective:	AOP-050-PCO-1			
Question Source:	Bank #	770855		
	Modified Bank #			
	New			
Question History:	N/A	Last NRC Exam:	N/A	

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

Comments:

KA Match: This question matches the KA because the examinee must know the specific bases for an EOP entry. The question requires knowledge of the transition step bases from the Small Break LOCA AOP into the EOP network. OP-TM-EOP-001, REACTOR TRIP is entered to minimize the possibility of entering OP-TM-EOP-002, LOSS OF 25F SUBCOOLING MARGIN. Small break LOCA at Three Mile Island is an AOP. The examinee must know when (and the basis) to initiate an EOP.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

4 **ID: 2085116** **Points: 1.00**

Due to a Large Break LOCA, OP-TM-EOP-002 LOSS OF 25F SUBCOOLING MARGIN, has been entered.

In accordance with Rule 4, FEEDWATER CONTROL, OTSG levels must be raised to ___(1)___ and the basis for this level is to prepare ___(2)___.

A. (1) 50% Operating Range
(2) the OTSGs for natural circulation

B. (1) 50% Operating Range
(2) for Boiler Condenser Cooling

C. (1) 75% - 85% Operating Range
(2) the OTSGs for natural circulation

D. (1) 75% - 85% Operating range
(2) for Boiler Condenser Cooling

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The crew has already entered the procedure for a loss of subcooling margin that the operators must already be performing OP-TM-EOP-010 EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS, Rule 4, FEEDWATER CONTROL (Rev 20, Page 8) as part of the OP-TM-EOP-001, REACTOR TRIP VSSV's; (2) Due to the nature of the entry into OP-TM-EOP-002, the operator must know that subcooling margin is less than 25F, and that OTSG levels must be raised to 75% to 85% in the operating range; (3) In accordance with OP-TM-EOP-0101, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS BASIS DOCUMENT, the reason for raising levels is to promote boiler condenser cooling as a method of heat transfer to cool the reactor.

A.	(1) 50% Operating Range (2) the OTSGs for natural circulation	INCORRECT: (1) Plausible because the HSPS setpoint for this question would be 50% in the operating range. Incorrect because the examinee must raise the level to 75% to 85% in the operating range. (2) Plausible because a goal would be to restore natural circulation. Incorrect because natural circulation cannot be occurring because subcooling margin is less than 25F (OP-TM-EOP-010, Guide 10, NATURAL CIRCULATION, Rev 20, Page 23).
B.	(1) 50% Operating Range (2) for Boiler Condenser Cooling	INCORRECT: (1) Plausible because the HSPS setpoint for this question would be 50% in the operating range. Incorrect because the examinee must raise the level to 75% to 85% in the operating range. (2) Plausible because a goal would be to have boiler condenser cooling for this casualty. Incorrect because OTSG levels must be raised to higher in the operating range.
C.	(1) 75% - 85% Operating Range (2) the OTSGs for natural circulation	INCORRECT: (1) See above. (2) Plausible because a goal would be to restore natural circulation. Incorrect because natural circulation cannot be occurring because subcooling margin is less than 25F (OP-TM-EOP-010, Guide 10, NATURAL CIRCULATION, Rev 20, Page 23).
D.	(1) 75% - 85% Operating range (2) for Boiler Condenser Cooling	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	011	EK1.01	
	Importance Rating	4.1		

K/A: Large Break LOCA: Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA: Natural circulation and cooling, including reflux boiling

Proposed Question:	Question #4	
Technical Reference(s):	OP-TM-EOP-010, Rev 20	OP-TM-EOP-0101, Rev 11

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EOP002-PCO-1			
Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the operational implication of raising OTSG level to high in the operating range. To answer the question correctly the examinee must know that their actions are to promote boiler condenser cooling during a Large Break LOCA.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

5 **ID: 2085147** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- All Reactor Coolant Pumps have tripped
- HSPS level setpoint to all EF-V-30 valves is 0%
- Both OTSGs are 80" start-up range and lowering

Which of the following is correct regarding OTSG level control in accordance with Rule 4, FEEDWATER CONTROL?

A. Feed to 50% in the operating range with MFW to establish and maintain natural circulation

B. Throttle MFW to maintain 25" in the startup range to maintain adequate primary to secondary heat transfer

C. Feed to 50% in the operating range with EFW to establish and maintain natural circulation

D. Throttle EFW to maintain 25" in the startup range to maintain adequate primary to secondary heat transfer

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Once the reactor coolant pumps have tripped, the reactor will trip; (2) This should also provide a signal for the Heat Sink Protection System to start the Emergency Feedwater Pumps and control OTSG level at 50% in the Operating Range (TQ-TM-104-644-C001, Rev 2, Page 13); (3) Once flow is lost, in addition to other things, the crew will perform OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES AND GRAPHS (Rev 20, Page 8) Rule 4, FEEDWATER CONTROL, step 4 to feed the OTSG with EFW to > 50% in the Operating Range; (6) For establishing natural circulation EFW is preferred due to the location in the OTSG that the water is introduced. EFW feeds directly on the OTSG tubes at a higher elevation than MFW.

A.	Feed to 50% in the operating range with MFW to establish and maintain natural circulation	INCORRECT: Plausible because MFW will be available to feed. Incorrect because EFW is the preferred feed source due to the location in the OTSG that the feedwater is introduced.
B.	Throttle MFW to maintain 25" in the startup range to maintain adequate primary to secondary heat transfer	INCORRECT: Plausible because MFW will be available to feed. Incorrect because EFW is the preferred feed source due to the location in the OTSG that the feedwater is introduced. In addition, 25" in the startup range is not the appropriate level
C.	Feed to 50% in the operating range with EFW to establish and maintain natural circulation	CORRECT: See Above
D.	Throttle EFW to maintain 25" in the startup range to maintain adequate primary to secondary heat transfer	INCORRECT: Plausible because EFW is the correct feed source. Incorrect because 25" in the startup range is the wrong level.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015	AK3.07
	Importance Rating	4.1	

K/A: Reactor Coolant Pump Malfunction: Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Ensuring that S/G levels are controlled properly for natural circulation enhancement.

Proposed Question:	Question #5	
Technical Reference(s):	OP-TM-EOP-010, Rev 20	TQ-TM-104-644-C001, Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-10

Question Source:	Bank #	2036514	
	Modified Bank #		
	New		

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

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the consequence of losing all Reactor Coolant pumps and the reason for the plant response. The examinee must identify a malfunction present in HSPS regarding the EFW level setpoint. The examinee must know that EFW is perferred over MFW to enhance natural circulation.

High Cog: The question is high cog because the examinee must determine the consequence of losing all Reactor Coolant pumps and have knowledge of the reason behind the plant response. The examinee must analyze the question to determine there is a malfunction with HSPS for feeding the OTSGs.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

6 **ID: 2088328** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Loss of Offsite Power

Which of the following must be performed by control room operators prior to restoring letdown flow?

A. Verify MU-P-1A auto starts

B. Verify MU-P-1B auto starts

C. Manually start MU-P-1A

D. Manually start MU-P-1B

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When a Loss of Offsite Power (LOOP) occurs, all BOP and ES busses lose power, this includes the 1E 4160V bus that the normally running makeup pump (MU-P-1B) is powered from. (2) To start any makeup pump, the pump 43 selector switch must be selected to that specific pump, which also selects the pump to start on an ES. Only the pumps that are selected for ES can be started from the Control Room. (3) When power is lost, MU-P-1B will trip and will NOT be automatically restarted when the 1E 4160V bus is powered from an Emergency Diesel Generator, whereas if MU-P-1A or MU-P-1C were running when the LOOP occurred they would automatically be restarted because they are selected for ES. (4) In accordance with OP-TM-AOP-020, LOSS OF STATION POWER (Rev 24, Page 3) on Step 3.6 the operators will verify Seal Injection flow is greater than 22 GPM, which since no Makeup Pump is running, it will be 0 gpm. (5) The operators will initiate OP-TM-AOP-041, LOSS OF SEAL INJECTION (Rev 8, Page 3) will lead the operators to starting MU-P-1A (Step 3.5 RNO step 5) on its current power supply, the 1D 4160V bus.

A.	Verify MU-P-1A auto starts	INCORRECT: Plausible because MU-P-1A will be started on the 1D 4160V bus. Incorrect because it will not auto start.
B.	Verify MU-P-1B auto starts	INCORRECT: Plausible because if MU-P-1B were ES selected (which it normally is NOT) it would have auto started on the 1E 4160V bus. Incorrect because it is not ES selected.
C.	Manually start MU-P-1A	CORRECT: See above.
D.	Manually start MU-P-1B	INCORRECT: Plausible because OP-TM-AOP-041 does have a section to manually start MU-P-1B on the 1E 4160V bus. Incorrect because that section would only be performed if MU-P-1A could not be started. In addition, this would require action outside the control room.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	022	AA2.02	
	Importance Rating	3.2		

K/A: Loss of Reactor Coolant Makeup: Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup Pumps: Charging pump problems

Proposed Question:	Question #6	
Technical Reference(s):	OP-TM-AOP-041, Rev 8	OP-TM-AOP-020, Rev 24

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-020-PCO-4

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

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

Comments:

KA Match: This question matches the KA because the examinee must know that a loss of reactor coolant makeup occurs during a loss of offsite power in this lineup. The examinee must know the correct makeup pump to start. A loss of makeup pump power is a charging pump problem.

High Cog: The examinee must understand the different power supplies for the makeup pumps and recognize the conditions required for a Makeup Pump to auto start.

7 **ID: 2109169** **Points: 1.00**

Plant Conditions:

- The plant is at 50% reactor power

Event:

- All 230 KV breakers on SS-1 from the switchyard to the grid OPEN
- Plant power is lowering in ICS Track mode
- FW-P-1B is NOT responding to demand changes (constant speed)
- RCS pressure is 2230 psig and rising

Which of the following actions must be taken in accordance with OP-TM-AOP-022, LOAD REJECTION?

A. Trip the reactor

B. Start and Load EG-Y-1B

C. Secure one Reactor Coolant Pump in each loop

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

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-TM-AOP-022 (Rev 9 Page 1), if RCS pressure is greater than 2200 psig, then manually open RC-V-1 (Spray Valve) and then place RC-V-1 to Auto. (2) This is below where it would automatically open to 40% (2205 psig) and below where alarm response OP-TM-MAP-G0308, RC PRESS NARROW RNG HI/LO directs this same action (2255 psig).

A.	Trip the reactor	INCORRECT: Plausible because the crew would trip the reactor if both GB1-02 and GB1-12 (Generator Breakers) were open or the examinee believed an RPS trip setpoint was exceeded. Incorrect because there is no indication that the Generator Breakers are open and an RPS trip setpoint has not been exceeded.
B.	Start and Load EG-Y-1B	INCORRECT: Plausible because a Diesel Generator is started in OP-TM-AOP-022. Incorrect because the Diesel Generator that is started is EG-Y-1A.
C.	Secure one Reactor Coolant Pump in each loop	INCORRECT: Plausible because securing one Reactor Coolant Pump in each loop would lower the load on the Turbine Generator. In addition the examinee could believe this action would help maintain the 7kV busses within their voltage band in step 3.7. Incorrect because this action is not required by OP-TM-AOP-022.
D.	Open the Spray Valve in Manual, then place in Auto	CORRECT: See above

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

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

Proposed Question:	Question #7	
Technical Reference(s):	OP-TM-AOP-022, Rev 9	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP-022-PCO-1	

Question Source:	Bank #	1142379	
	Modified Bank #		
	New		

Question History:	N/A	Last NRC Exam:	14-01	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the operator must demonstrate the ability to recognize plant conditions that require manipulation of reactor controls to prevent the reactor from tripping during a Load Rejection. Reactor controls includes components which protect from overpressure, which the spray valve does.

High Cog: This question is high cog because the examinee must analyze the stem and determine the correct action to take.

8 **ID: 2108961** **Points: 1.00**

Plant Conditions:
- 100% power with ICS in full auto.
- Main Instrument Air Compressor, IA-P-4, has been secured for repairs.
- Instrument Air Compressors, IA-P-1A and IA-P-1B, are running as required.

EVENTS:

- LOCA
- Loss of Off-Site Power
- 1600 PSI ES Actuation

10 minutes later:

- Due to low instrument air pressure, the CRO starts IA-P-1A

Cooling for IA-P-1A/B will be aligned to ___(1)___ due to the ___(2)___.

A. (1) Fire Service Water, (2) ES Actuation Signal

B. (1) Fire Service Water, (2) Loss of Offsite Power

C. (1) Secondary Closed Cooling, (2) ES Actuation Signal

D. (1) Secondary Closed Cooling, (2) Loss of Offsite Power

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The normal cooling water supply to the instrument air compressors is Secondary Services Closed Cooling Water system; (2) If none of the closed cooling water pumps are operating (as sensed by the SC pumps breaker position) the three way valves SC-V-57 (IA-P-1A) and SC-V-58 (IA-P-1B) will automatically position to allow Fire Service Water to cool the respective air compressor (1104-12, SECONDARY SERVICES CLOSED COOLING WATER SYSTEMS, Rev 3, Page 16); (3) Loss of Offsite Power trips the secondary closed cooling pumps, which aligns fire service water to IA-P-1A/B (4) The ES actuation signal effects the auto start of IA-P-1A/B but not the cooling water.

A.	(1) Fire Service Water, (2) ES Actuation Signal	INCORRECT: Plausible since Fire Service Water is the alternate cooling water. If the examinee believes an ES could affect the SCCW Pumps (Load Shed) then the backup cooling would be initiated.
B.	(1) Fire Service Water, (2) Loss of Offsite Power	CORRECT: See above
C.	(1) Secondary Closed Cooling, (2) ES Actuation Signal	INCORRECT: Plausible if the examinee thinks the backup SCCW pump would start once the emergency diesels started and loaded on their bus. Incorrect because the power supplies to the SCCW pumps are 1C, 1J, and 1N 480V busses which are not energized.
D.	(1) Secondary Closed Cooling, (2) Loss of Offsite Power.	INCORRECT: Plausible if the examinee thinks the backup SCCW Pump would start once the emergency diesel generators started and loaded on their bus.

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

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

Proposed Question:	Question #8	
Technical Reference(s):	1104-12, Rev 63A	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	851-GLO-10	

Question Source:	Bank #	2108961	
	Modified Bank #		
	New		

Question History:	N/A	Last NRC Exam:	16-01	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.2	
	55.43		

Comments:

Comments:

KA Match: This question matches the KA because the examinee must know that Fire Service Water will backup Secondary Closed Cooling Water in the event of a loss of offsite power.

High Cog: This question is High Cog because the examinee must analyze the events in the stem to determine the cooling water that IA-P-1A/B are utilizing.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

9 **ID: 2088420** **Points: 1.00**

Plant Conditions:

- Reactor power is 70% with ICS in auto

Event:

- Annunciator "ICS/NNI POWER LOST" H-1-8 is received
- ICS AUTO Power light on Panel Center is off
- Pressurizer pressure is 2145 psig and lowering
- Pressurizer level, as indicated on LI-777A, is 220 inches and relatively steady

Which one of the following identifies the action(s) that must be taken in accordance with OP-TM-AOP-27, LOSS OF ATA OR ICS AUTO POWER?

A. Bypass Lo-Lo Level interlock in ICS/NNI Power Monitor Cabinet and operate Pressurizer Heater Banks 1, 2 & 3 in Auto

B. Energize Pressurizer Heater Banks 4 & 5 using the "ON/OFF" switches on CR until ICS Auto Power is restored

C. Bypass the Lo-Lo Level interlock in ICS/NNI Power Monitor Cabinet and operate Pressurizer Heater Banks 1, 2 & 3 from the Bailey Station on CC

D. Take manual control of FW-V-16A/B & 17A/B using the HAND/AUTO Stations on CC and close as necessary to prevent the excessive RCS cooldown

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When MAP H-1-8 alarms, in conjunction with the knowledge that the 30 amp ICS Auto light is blown, the crew must enter OP-TM-AOP-027, LOSS OF ATA OR ICS AUTO POWER (Rev 10). (2) With ICS Auto power lost, the crew must place the PZR HTR LO-LO LEVEL CUTOUT BYPASS switch to BYPASS (OP-TM-AOP-027, Rev 5 Page 5) in step 3.13, and initiate OP-TM-220-503, MANUAL CONTROL OF PRESSURIZER PRESSURE in step 3.15 (OP-TM-AOP-027, Rev 5, Page 7). (3) The operator must know that with ICS Hand power still available that heater groups 1 through 3 are operated in with the toggle switch on the Bailey Station (OP-TM-220-503, Step 4.1, Rev 4, Page 2)

A.	Bypass Lo-Lo Level interlock in ICS/NNI Power Monitor Cabinet and operate Pressurizer Heater Banks 1, 2 & 3 in Auto	INCORRECT: Plausible because the Pressurizer Lo-Lo Level Interlock must be bypassed. Incorrect because the Pressurizer heaters are still not operable in automatic at this point.
B.	Energize Pressurizer Heater Banks 4 & 5 using the "ON/OFF" switches on CR until ICS Auto Power is restored	INCORRECT: Plausible if the examinee believes the ON/OFF function of the Group 4 and 5 heaters is still available. Incorrect because the Pressurizer Lo-Lo Level Interlock must be bypassed to operate those heaters.
C.	Bypass the Lo-Lo Level interlock in ICS/NNI Power Monitor Cabinet and operate Pressurizer Heater Banks 1, 2 & 3 from the Bailey Station on CC	CORRECT: See above.
D.	Take manual control of FW-V-16A/B & 17A/B using the HAND/AUTO Stations on CC and close as necessary to prevent the excessive RCS cooldown	INCORRECT: Plausible since OP-TM-AOP-027 directs manual control or FW-V-16A/B and 17A/B. Incorrect because at 100% power an excessive RCS cooldown will not occur and these FW valves will not need to be operated.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027	AK2.03
	Importance Rating	2.6	

K/A: Pressurizer Pressure Control System Malfunction: Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and Positioners

Proposed Question:	Question #9	
Technical Reference(s):	OP-AOP-027, Rev 10	OP-TM-220-503, Rev 4

Proposed References to be provided to applicants during examination:	None

Learning Objective:	624-GLO-9	

Question Source:	Bank #	299478		
	Modified Bank #			
	New			
Question History:	Sim Exam 7	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know that the loss of ICS Auto power malfunction causes the pressurizer heaters to shutoff due to loss of level signal. The operator must operate the bypass feature to regain control of the pressurizer heater banks controller.

High Cog: The examinee will have to analyze the given conditions to determine the procedure path. The examinee must also understand how the loss of ICS Auto Power impacts Pressurizer Heater Control for both the SCR and analog controlled heaters.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

10 **ID: 2085315** **Points: 1.00**

REFERENCE PROVIDED

Plant Conditions:

- Reactor power is 50% with ICS in auto

Sequence of Events:

- ICS placed in HAND due to a feedwater transient that caused RCS temperature to rise
- An RCS leak occurs
- Currently:
  - RCS pressure is 2000 psig
  - RCS Thot is 617F
  - RCS Tcold is 593F

Based on the above information, an ATWS has:

A. NOT occurred and Main Feedwater flow must be raised

B. NOT occurred and the reactor must be tripped if the RCS leak cannot be isolated

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

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

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Using the provided reference, Technical Specification Figure 2.3-1, that an ATWS has occurred due to exceeding the Variable Low Pressure Trip. For most of Figure 2.3-1, 618.8F is the highest temperature allowed and 1900 psig is the highest pressure allowed. However, the Variable Low Pressure Trip (VLPT) is in effect on the bottom right hand corner of the figure. (2) OP-TM-EOP-001, REACTOR TRIP (Rev 16, Page 1) must be entered and the IMA's performed. (3) Based the nature of the correct answer and the other ATWS distractor, it is implied that pressing the Reactor Trip and DSS pushbuttons fail since both mention 'if reactor power stays at 50%'. (4) The RNO of step 2.2 is entered and the and because the Reactor is still at 50% when RNO step 5 is reached, HPI is then initiated.

A.	NOT occurred and Main Feedwater flow must be raised	INCORRECT: Plausible if the examinee does not correctly take into consideration the VLPT. In addition, the crew would slowly raise Main Feedwater flow to lower the temperature while dealing with the RCS leak. Incorrect because the VLPT was exceeded.
B.	NOT occurred and the reactor must be tripped if the RCS leak cannot be isolated	INCORRECT: Plausible if the examinee does not correctly take into consideration the VLPT. Because the examinee does not believe an ATWS occurred, they will continue in OP-TM-AOP-050, REACTOR COOLANT LEAK, which will continue with a plant shutdown. The reactor would be tripped if an RCS trip setpoint is approached or exceeded. Incorrect because the VLPT was exceeded.
C.	occurred, the Immediate Manual Actions of OP-TM-EOP-001, REACTOR TRIP, must be performed, and HPI initiation is required if reactor power remains at 50%	CORRECT: See above.
D.	occurred, the Immediate Manual Actions of OP-TM-EOP-001, REACTOR TRIP, must be performed, and a manual turbine trip is required if reactor power remains at 50%	INCORRECT: Plausible because the third step of the RNO section could lead the examinee to believe that tripping the Turbine is the correct answer. Incorrect because the Turbine would only be tripped if Main Feedwater was not available.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	029	EK3.02	
	Importance Rating	3.1		

K/A: Anticipated Transient Without Scram: Knowledge of the reasons for the following responses as they apply to an ATWS: Starting a specific charging pump

Proposed Question:	Question #10	
Technical Reference(s):	OP-TM-EOP-001, Rev 16	Tech Spec Figure 2.3-1, AMD 262

Proposed References to be provided to applicants during examination:				Fig 2.3-1, AMD 262 Black out AXIS labels and Region Names
Learning Objective:	641-GLO-10			
Question Source:	Bank #	978945		
	Modified Bank #			
	New			
Question History:	N/A	Last NRC Exam:	12-01	

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

Comments:

KA Match: This question matches the KA because the examinee must know the reason the charging pumps are started in order to answer the question correctly. The examinee must identify there is an ATWS, then know that the charging pumps are started because the reactor did not shutdown (vice tripping the turbine). At Three Mile Island there is no reason a specific charging pump would be started for an ATWS. This question starts all of the charging pumps.

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

11 **ID: 2085318** **Points: 1.00**

Plant Conditions:

- A tube rupture has occurred in the “A” OTSG
- The reactor has been tripped
- Current 5-minute RCS cooldown rate is 100°F/ hr
- RCS pressure is 1210 psig
- “A” OTSG level is 82% and rising at 3% / minute
- “A” loop RCS cold leg temperature is 540°F
- “B” loop RCS cold leg temperature is 543°F
- “A” OTSG Turbine Bypass Valves, MS-V-3D/E/F, are open 30% in manual
- “B” OTSG Turbine Bypass Valves, MS-V-3A/B/C, are closed

Which of the following actions must the crew take to mitigate this casualty?

A. Immediately lower RCS pressure to < 1000 psig using the PORV, RC-RV-2

B. Immediately lower RCS pressure to < 1000 psig using the Spray Valve, RC-V-1

C. Maximize RCS cooldown rate, remaining less than 240°F/hr to 500°F, using MS-V-3A/B/C

D. Maximize RCS cooldown rate, remaining less than 240°F/hr to 500°F, using MS-V-3D/E/F

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Due to the 'A' OTSG tube rupture, the crew is in OP-TM-EOP-005, OTSG TUBE LEAKAGE (Rev 10). The reactor is tripped and a cooldown is commenced. Due to the "A" OTSG Turbine Bypass Valve position, it is evident that the crew is preferentially steaming the "A" OTSG to cooldown the plant at the 100F/hr limit. (2) At 85% in the operating range (Step 3.34, Page 17) the crew must isolate the "A" OTSG, but RCS pressure must be below 1000 psig. (3) Due to challenging isolation criteria, in accordance with Step 3.31 (Page 15), the crew must initiate an RCS cooldown to ≤ 500F Thot at a rate < 240F/hr using the "A" OTSG Turbine Bypass Valves. (4) Using the "A" OTSG Turbine Bypass Valves will allow for the plant to be cooldown and the "A" OTSG level to lower (or not rise as fast) to maximize time before the isolation criteria is met.

A.	Immediately lower RCS pressure to < 1000 psig using the PORV, RC-RV-2	INCORRECT: Plausible because in Step 3.31 the crew will open the PORV AFTER initiating the RCS cooldown. Incorrect because the crew does not immediately perform this step.
B.	Immediately lower RCS pressure to < 1000 psig using the Spray Valve, RC-V-1	INCORRECT: Plausible because in Step 3.31 the crew will open the Spray Valve AFTER initiating the RCS cooldown. Incorrect because the crew does not immediately perform this step.
C.	Maximize RCS cooldown rate, remaining less than 240°F/hr to 500°F, using MS-V-3A/B/C	INCORRECT: Plausible because the examinee must maximize cooldown rate. General OP-TM-EOP-005 strategy is to steam both OTSGs until on DHR. Examinee could believe the appropriate OTSG to steam is the unaffected OTSG, which is wrong for the severity of this leak. Incorrect because the crew must preferentially steam the "A" OTSG, not the "B" OTSG.
D.	Maximize RCS cooldown rate, remaining less than 240°F/hr to 500°F, using MS-V-3D/E/F	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	038	EA1.36
	Importance Rating	4.3	

K/A: Steam Generator Tube Rupture: Ability to operate and monitor the following as they apply to a SGTR: Cooldown to a specified temperature

Proposed Question:	Question #11
Technical Reference(s):	OP-TM-EOP-005, Rev 10
Proposed References to be provided to applicants during examination:	None
Learning Objective:	EOP005-PCO-5
Question Source:	Bank # 862872
	Modified Bank #

	New			
Question History:	Simulator Exam 9	Last NRC Exam:	N/A	

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

Comments:

KA Match: This question matches the KA because the examinee must know the cooldown rate to a specific temperature for an OTSG tube rupture.

High Cog: This question is high cog because the examinee analyzes the parameters (the act of preferentially steaming and OTSG isolation criteria) and know the procedure steps that apply. The examinee must identify the correct action to take.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

12 **ID: 2087712** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

A Steam Line Rupture occurs, the crew trips the reactor and enters OP-TM-EOP-003, EXCESSIVE PRIMARY TO SECONDARY HEAT TRANSFER.

Reactor Building pressure peaks at 12 psig.

Which phases of isolation must be complete and where must RCS pressure/temperature be stabilized in accordance with OP-TM-EOP-003?

A. (1) Phase 1 Only
(2) At the values existing when the XHT is terminated

B. (1) Phase 1 Only
(2) At Hot Shutdown values for pressure/temperature

C. (1) Phase 1 and 2
(2) At the values existing when the XHT is terminated

D. (1) Phase 1 and 2
(2) At Hot Shutdown values for pressure/temperature

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The first immediate manual action of EOP-003 (Rev 9, Page 1) is to perform Rule 3, EXCESSIVE HEAT TRANSFER of OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS (Rev 20, page 6). Rule 3 directs the operator to perform Phase 1 isolation (Steam outputs of the OTSG), then verify the leak is not in the Reactor or Intermediate Buildings. (2) Reactor Building pressure peaking at 12 psig indicates that the leak is in the Reactor Building. (3) This requires the operator to go to the RNO and perform Phase 2 isolation (EFW cooling to the OTSG). (4) When Phase 1 and 2 isolations are complete, there is no feed to the OTSG. After the remaining steam is released from the OTSG the cooldown will be terminated. (5) EOP-003 (Rev 9, Page 3) directs the crew to perform Guide 12, RCS STABILIZATION of EOP-010 (Rev 20, Page 25), which directs the crew to adjust the remaining OTSG Pressure to stabilize RCS temperature and throttle HPI at the current values post cooldown (pressure/temperature when the XHT is terminated). (6) The basis for Guide 12 in OP-TM-EOP-0101, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS BASIS DOCUMENT (Rev 11, Pages 50 and 51) is to stabilize plant (pressure and temperature) after the cooldown has subsided to prevent further stresses and pressurized thermal shock.

A.	(1) Phase 1 Only (2) At the values existing when the XHT is terminated	INCORRECT: (1) Plausible because the question does not explicitly state that the Steam Leak is in the Reactor Building. Incorrect because it is unisolable in the Reactor Building based on the elevated Reactor building pressure. (2) The RCS Pressure/Temperature after the XHT is terminated is the correct answer.
B.	(1) Phase 1 Only (2) At Hot Shutdown values for pressure/temperature	INCORRECT: (1) Plausible because the question does not explicitly state that the Steam Leak is in the Reactor Building. Incorrect because it is unisolable in the Reactor Building based on the elevated Reactor building pressure. (2) Plausible because the Hot Shutdown values are where the operators normally stabilize the plant after a reactor trip. Incorrect because the goal after an overcooling event is to keep temperature and pressure stable at the lowest value possible to minimize further stress on OTSG and Reactor components.
C.	(1) Phase 1 and 2 (2) At the values existing when the XHT is terminated	CORRECT: See above.
D.	(1) Phase 1 and 2 (2) At Hot Shutdown values for pressure/temperature	INCORRECT: (1) Phase 1 and 2 is the correct answer. (2) Plausible because the Hot Shutdown values are where the operators normally stabilize the plant after a reactor trip. Incorrect because the goal after an overcooling event is to keep temperature and pressure stable at the lowest value possible to minimize further stress on OTSG and Reactor components.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	040	AK1.04
	Importance Rating	3.2	

K/A: Steam Line Rupture - Excessive Heat Transfer: Knowledge of the operational implications of the following concepts as they apply to the Steam Line Rupture: Nil Ductility Temperature

Proposed Question:	Question #12			
Technical Reference(s):	OP-TM-EOP-003, Rev 9		OP-TM-EOP-010, Rev 20	
	OP-TM-EOP-001, Rev 11			
Proposed References to be provided to applicants during examination:			None	
Learning Objective:	EOP003-PCO-4			
Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	Last NRC Exam:	N/A	

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

Comments:

KA Match: This question matches the K/A because the examinee must know how to operationally control excessive RCS heat transfer. When the plant uncontrollably overcools the possibility of components reaching their Nil-ductility temperature rises, so when the cooldown has subsided the operators should maintain the RCS temperature/pressure at the post cooldown values.

High Cog: This question is high cog because the examinee must analze the stem and determine that Phase 2 isolation needs to be completed. In addition, the examinee must know where to stabilize the RCS parameters.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

13 **ID: 2096952** **Points: 1.00**

Plant conditions:

- Plant startup in progress
- Reactor Power is 12%
- Both Aux Boilers are secured
- Chest Warming is in progress on the main turbine
- FW-P-1A is on line
- FW-P-1B is secured

Events:

- A large steam leak occurs in the Intermediate Building
- "A" OTSG pressure lowers to 500 psig
- The CRO trips the reactor and performs Phase 1 isolation of "A" OTSG, only

Assuming no further operator actions, what is the current response of Main Feedwater flow to (1) "A" and (2) "B" OTSG's?

A. (1) Main Feedwater WILL be isolated to the "A" OTSG
(2) Main Feedwater WILL maintain the "B" OTSG level at the setpoint

B. (1) Main Feedwater WILL be isolated to the "A" OTSG
(2) Main Feedwater will NOT maintain the "B" OTSG at setpoint

C. (1) Main Feedwater will NOT be isolated to the "A" OTSG
(2) Main Feedwater WILL maintain the "B" OTSG level at setpoint

D. (1) Main Feedwater will NOT be isolated to the "A" OTSG
(2) Main Feedwater will NOT maintain the "B" OTSG level at setpoint

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The Heat Sink Protection System (HSPS) will isolate Main Feedwater to the "A" OTSG when pressure lowers to less than 600 psig and the system is not bypassed. (TQ-TM-104-644-C001, Rev 2 Page 29). (2) When MS-V-1A and MS-V-1B (during Phase 1 Isolation of the OTSG) are closed, that removes the only available source of steam to FW-P-1A, which secures all available MFW sources (i.e. all MFW flow drops to 0 lbm/hr). (3) When OTSG level lowers to "B" OTSG, then EFW will be initiated and feed to keep level at 25" in the startup range.

A.	(1) Main Feedwater WILL be isolated to the "A" OTSG (2) Main Feedwater WILL maintain the "B" OTSG level at the setpointt	INCORRECT: (1) Correct answer. (2) Plausible if the examinee does not recognize that the only running MFW pump Main Steam isolation valves are closed. Incorrect because these valves got closed during Phase 1 isolation.
B.	(1) Main Feedwater WILL be isolated to the "A" OTSG (2) Main Feedwater will NOT maintain the "B" OTSG at setpoint	CORRECT: See above.
C.	(1) Main Feedwater will NOT be isolated to the "A" OTSG (2) Main Feedwater WILL maintain the "B" OTSG level at setpoint	INCORRECT: (1) Plausible if the examinee believes HSPS did not isolate the "A" OTSG on low OTSG pressure. Incorrect because MFW is isolated to this OTSG. (2) Plausible if the examinee does not recognize that the only running MFW pump Main Steam isolation valves are closed. Incorrect because these valves got closed during Phase 1 isolation.
D.	(1) Main Feedwater will NOT be isolated to the "A" OTSG (2) Main Feedwater will NOT maintain the "B" OTSG level at setpoint	INCORRECT: (1) Plausible if the examinee believes HSPS did not isolate the "A" OTSG on low OTSG pressure. Incorrect because MFW is isolated to this OTSG. (2) Correct answer.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	054	AA2.02	
	Importance Rating	4.1		

K/A: Loss of Main Feedwater: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Differentiation between loss of all MFW and trip of one MFW pump

Proposed Question:	Question #13	
Technical Reference(s):	TQ-TM-104-644-C001, Rev 2	OP-TM-EOP-010, Rev 20

Proposed References to be provided to applicants during examination:	NONE

Learning Objective:	401-GLO-11			
Question Source:	Bank #	689340		
	Modified Bank #			
	New			
Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee will have to differentiate between a loss of one main feedwater pump and all main feedwater pumps. The examinee will have to determine that the isolation valves to the only running main feedwater pump are closed.

High Cog: This question is high cog because the examinee must analyze the plant response when Phase 1 isolation is complete. In addition, the examinee must identify that the "A" OTSG pressure is below 600 psig and that HSPS isolated MFW to that OTSG.

14 **ID: 2096960** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Loss of Offsite Power

Current Parameters:

- RCS SCM is 70°F
- OTSG 1A/1B pressures are 850 psig
- OTSG 1A/1B levels are 45% and rising at 5%/minute
- All EF-V-30's are 100% open in automatic control
- RCS natural circulation cooling has been verified
- Atmospheric dump valves (ADVs) MS-V-4A/B are both closed in automatic

Identify the selection below that describes the resulting operational implications on natural circulation flow after the OTSGs have reached their level setpoint.

Natural circulation flow rate will _____.

A. rise because OTSG temperature rises.

B. not be affected because OTSGs are at level setpoint.

C. initially lower as RCS temperature rises and then stabilize at a lower value.

D. not be affected as long as steam is being supplied to the Steam Driven EFW Pump, EF-P-1.

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) A differential temperature develops proportional to decay heat and heat. B & W plant experience has shown that the temperature difference can approach 50F following reactor trip and loss of forced flow. As decay heat lowers, this temperature difference lowers proportionally. If the differential temperature continues to rise that would be an indication either lack of heat transfer or lack of heat sink. (OP-TM-EOP-0101, Rev 11 Page 45) (2) At the point in which the question asks the status of the natural circulation, EFW flow (currently the only Feedwater available) stops flowing into the OTSG. This would lower the differential temperature on the RCS in that FW is not providing as much cooling for the OTSG tubes. (3) The steam indicates currently the ADVs are closed, so at the point at which EFW stops feeding and the ADV's are closed natural circulation will initially lower and stabilize at a lower value. (OP-TM-EOP-010, Rev 20 Page 41)

A.	rise because OTSG temperature rises	INCORRECT: Plausible because OTSG temperature will rise. Incorrect because the temperature rise would be in the cold leg, which reduce flow due to lower difference in RCS density.
B.	not be affected because OTSGs are at level setpoint	INCORRECT: Plausible because natural circulation is also effected by steaming the OTSG's (i.e. opening the ADVs0. Incorrect because feeding the OTSGs with EFW also promotes natural circulation.
C.	initially reduce as RCS temperature rises and then stabilize at a lower value	CORRECT: See above.
D.	not be affected as long as steam is being supplied to the Steam Driven EFW Pump, EF-P-1	INCORRECT: Plausible because the examinee could believe that as long as steam is being drawn off for EF-P-1 that natural circulation will not change (i.e. steaming the OTSG's). Incorrect because the question is strictly asking about natural circulation when the EFW setpoint is reached. Steam to EF-P-1 is negligible at that point.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	056	AK1.01
	Importance Rating	3.7	

K/A: Loss of Offsite Power: Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Principle of cooling by natural circulation

Proposed Question:	Question #14	
Technical Reference(s):	OP-TM-EOP-0101, Rev 11	OP-TM-EOP-010, Rev 20

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP-020-PCO-5	

Question Source:	Bank #	363884	
	Modified Bank #		

	New			
Question History:	Sim Exam 8	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.14	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the operational implication that reaching the OTSG level setpoint in automatic will have on natural circulation.

High Cog: The examinee has to analyze the conditions given in the question stem and determine that since the OTSG pressure is lower than the ADV setpoint, that when EFW reaches its level setpoint the amount of natural circulation will lower.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

15 **ID: 2085756** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- MAP A-1-6, INVERTER FAILED illuminates
- MAP A-3-8, INVERTER 1B/1D/1F TROUBLE illuminates
- D RPS cabinet de energized (no lights)
- ES Train B Channel trip indication on Panel PCR

What action must the crew take and what is the basis for this action?

LEGEND:
RM-A-1 - Control Tower Air Monitor
RM-A-8 - Aux and FH Building Air Monitor

A. PLACE RM-A-1G interlock switches in the DEFEAT position to enable restoration of CB ventilation

B. PLACE RM-A-8G interlock switch in the DEFEAT position to enable restoration of Aux and FH building ventilation

C. PLACE ES Status Light Power Supply Select switch (PCR) in the BUS-A position to align the ES status light indications to the “A” Vital Bus to restore status indications

D. PLACE ES Status Light Power Supply Select switch (PCR) in the BUS-B position to align the ES status light indications to the “B” Vital Bus to restore status indications

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must: (1) Analyze the conditions and determine that a loss of a loss of Vital Bus has occurred (the definitive bullets being the "D" RPS cabinet de-energized, and the ES train "B" Channel Trip). (2) OP-TM-AOP-018, LOSS OF VBD (Rev 9 Page 3) directs the crew to defeat RM-A-1G and restore Control Building Ventilation.

A.	PLACE RM-A-1G interlock switches in the DEFEAT position to enable restoration of CB ventilation	CORRECT: See above.:
B.	PLACE RM-A-8G interlock switch in the DEFEAT position to enable restoration of Aux and FH building ventilation	INCORRECT: Plausible because these actions would apply if Vital Bus "C" is lost. Incorrect because the loss of Vital Bus "D" has occurred.
C.	PLACE ES Status Light Power Supply Select switch (PCR) in the BUS-A position to align the ES status light indications to the “A” Vital Bus to restore status indications	INCORRECT: Plausible because these actions would apply if Vital Bus "A" is lost. Incorrect because the loss of Vital Bus "D" has occurred.
D.	PLACE ES Status Light Power Supply Select switch (PCR) in the BUS-B position to align the ES status light indications to the “B” Vital Bus to restore status indications	INCORRECT: Plausible because these actions would apply if Vital Bus "B" is lost. Incorrect because the loss of Vital Bus "D" has occurred.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	057	AA2.20
	Importance Rating	3.6	

K/A: Loss of Vital AC Instrument Bus: Ability to determine and interpret the following to the Loss of Vital AC Instrument Bus: Interlocks in effect on loss of ac vital electrical instrument bus that must be bypassed to restore normal equipment operation

Proposed Question:	Question #15	
Technical Reference(s):	OP-TM-AOP-018, Rev 9	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP18-PCO-5		
Question Source:	Bank #		

	Modified Bank #	1720293		
	New			
Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must determine the Vital Bus that was lost and identify the component which must be defeated to restore control building ventilation.

High Cog: Examinee must analyze plant conditions and indications to identify the Vital Bus that was loss and the proper actions to take to restore equipment operation of equipment.

Plant Event

- Numerous alarms occur simultaneously in the control room while operating at 100% power. The following indications are noted by the crew:
    - MAP A-1-6, INVERTER FAILED illuminates.
    - MAP A-3-8, INVERTER 1B/1D/1F TROUBLE illuminates.

Which ONE of the following describes the required action per procedure AND the reason?

A. PLACE RC-P-2B-2 in PTL to reduce load on the “B” DC system.

B. PLACE RM-A-1G interlock switches in the DEFEAT position to enable restoration of CB ventilation.

C. PLACE RM-A-8G interlock switch in the DEFEAT position to enable restoration of AB and FHB ventilation.

D. PLACE ES Status Light Power Supply Select switch (PCR) in the BUS-B position to align the ES status light indications to the “B” Vital Bus to restore status indications.

Answer B

16 **ID: 2085769** **Points: 1.00**

Plant conditions:

- Reactor power is 100% with ICS in full auto
- EG-Y-1B is OOS for Maintenance
- NR-P-1B is powered by the 1R 480V ES Bus and selected for ES in accordance with OP-TM-541-443, SWAP NR-P-1B TO ALTERNATE POWER SUPPLY
- NR-P-1A and NR-P-1C are running

Sequence of Events:

- Loss of Offsite Power
- Large Break LOCA occurs 30 seconds later
- 4 psig ES actuation

Which of the following is the correct response 10 seconds after the 4 psig ES actuation?

A. NR-P-1A will auto start

B. NR-P-1B will be off

C. NR-V-1A will be closing

D. NR-V-1B will be opening

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) For this stem, when offsite power is lost, the Nuclear River Water pump in standby will start. (2) When a Nuclear River Water pump starts, its outlet valve (NR-P-1B) will travel open. (3) The valve only goes closed when the manual CLOSE pushbutton is pushed (TQ-TM-104-531-C001, PRIMARY COOLING SYSTEMS (Rev 9 Pages 9 & 37) (4) The non-ES selected pump (NR-P-1A) will trip on an ES actuation.

A.	NR-P-1A will auto start	INCORRECT: Plausible because in a normal plant lineup NR-P-1A is ES selected and will auto start if an ES were to actuate. Incorrect because NR-P-1B is ES selected on the 1R 480V ES bus. In addition, after the LOOP, NR-P-1B will auto start because it is the standby pump, not NR-P-1A.
B.	NR-P-1B will be off	INCORRECT: Plausible because in a normal plant lineup NR-P-1B will have started when the LOOP occurred, then tripped when the ES occurred. Incorrect because NR-P-1B is ES selected on the 1R 480V ES bus.
C.	NR-V-1A will be closing	INCORRECT: Plausible because some ES valves (Decay River) go closed when the pump trips. Incorrect because the NR-V-1's do not have that interlock.
D.	NR-V-1B will be opening	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	062	AK3.01
	Importance Rating	3.2	

K/A: Loss of Nuclear Service Water: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The conditions that will initiate the automatic opening and closing of the SWS isolation valves to the Nuclear Service Water Coolers.

Proposed Question:	Question 16
Technical Reference(s):	TQ-TM-104-531-C001, Rev 9
Proposed References to be provided to applicants during examination:	None
Learning Objective:	531-GLO-5

Question Source:	Bank #	862968		
	Modified Bank #			
	New			
Question History:	System Exam 6	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the opening signal to the river water isolation valve to the Nuclear Service Heat Exchangers. These are the only isolation valves to in the Nuclear Service Water System that isolate part of the heat exchanger that have automatic features. These valves do not have an automatic closing feature.

High Cog: Examinee has to analyze the plant conditions, the power supply switch and the effect of the LOOP with an ES actuation signal has on the effected components.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

17 **ID: 2110232** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Annunciator PLB-1-7 INSTRUMENT AIR PRESS LOW TURBINE AREA actuates
- Secondary IA Pressure on PI-1403 is 55 PSIG and lowering

In accordance with OP-TM-AOP-028, LOSS OF INSTRUMENT AIR, which one of the following actions must be taken first by the control room operating crew?

A. Initiate a manual Reactor Trip

B. Start available air compressors

C. Place both FW pumps in manual on the motor speed changers

D. Direct an AO to open the Vacuum Pump suction valves (VA-V-5's)

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) With Secondary Instrument Air pressure less than 80 psig and lowering, the crew must enter OP-TM-AOP-028, LOSS OF INSTRUMENT AIR. When pressure on PI-1403 drops below 60 psig action is taken to ensure some primary cooling valves will remain open and then the reactor is tripped (OP-TM-AOP-028, Rev 9 Page 3). (2) The steps before tripping the reactor are to dispatch operators. The first action to take after 60 psig is to trip the reactor.

A.	Initiate a MANUAL Reactor Trip	CORRECT: See above.
B.	Start available air compressors	INCORRECT: Plausible because this action will be (and should be taken) when pressure gets below 85 psig. Incorrect because it is not the action that must be taken now. In addition, all air compressors have setpoints higher than 55 psig and should have started by 55 psig.
C.	Place both FW pumps in manual on the motor speed changers	INCORRECT: Plausible because the Feedwater pumps are being controlled on the air speed changer. Incorrect because the Feedwater pumps are tripped if feedwater control becomes unreliable.
D.	Direct an AO to OPEN the Vacuum Pump suction valves (VA-V-5's)	INCORRECT: Plausible because the VA-V-5s fail closed on a loss of air. Incorrect because this is not the first action that must be taken.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	065	AA1.05	
	Importance Rating	3.3		

K/A: Loss of Instrument Air: Ability to operate and / monitor the following as they apply to the Loss of Instrument Air: RPS

Proposed Question:	Question #17	
Technical Reference(s):	OP-TM-AOP-028, Rev 9	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP028-PCO-4	

Question Source:	Bank #	371798	
	Modified Bank #		
	New		

Question History:	Simulator Exam 7	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	X	

	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the relation between a loss of instrument air and when the reactor must be tripped (actuation of the RPS system).

18 **ID: 2103867** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Loss of both Main Feedwater Pumps

Subsequently:

- All 8 (eight) HSPS switches placed in DEFEAT.
- No other operator actions have been taken.

What is the HSPS level setpoint and where are each OTSG EFW valves setpoints indicated?

A. (1) 0" on the startup range
(2) OTSG 'A' EFW valves on CC and OTSG 'B' EFW Valves on CL

B. (1) 25" on the startup range
(2) OTSG 'A' EFW valves on CC and OTSG 'B' EFW Valves on CL

C. (1) 0" on the startup range
(2) OTSG 'A' EFW valves on CL and OTSG 'B' EFW Valves on CC

D. (1) 25" on the startup range
(2) OTSG 'A' EFW valves on CL and OTSG 'B' EFW Valves on CC

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The Heat Sink Protection System (HSPS) will actuate Emergency Feedwater (EFW) when both Main Feedwater Pumps are tripped. (2) The EF-V-30 A/B/C/D valves will control OTSG level at 25" in the SU range when both Main Feedwater Pumps are lost. (3) To clear a setpoint the operator must take the switches to defeat (for whichever actuation occurred) and place the EF-V-30 in manual (TQ-TM-104-644-C001, Rev 2 Pages 38 and 39).

A.	(1) 0" on the startup range (2) OTSG 'A' EFW valves on CC and OTSG 'B' EFW Valves on CL	INCORRECT: (1) Plausible of the examinee believes the setpoint is reset when all of the HSPS switches are in defeat. Incorrect because the respective EF-V-30s must be placed in hand to clear the seal-in relay. (2) Plausible if examinee does not remember the location of the EF-V-30 controllers.
B.	(1) 25" on the startup range (2) OTSG 'A' EFW valves on CC and OTSG 'B' EFW Valves on CL	INCORRECT: (1) See above. (2) Plausible if examinee does not remember the location of the EF-V-30 controllers.
C.	(1) 0" on the startup range (2) OTSG 'A' EFW valves on CL and OTSG 'B' EFW Valves on CC	INCORRECT: (1) Plausible of the examinee believes the setpoint is reset when all of the HSPS switches are in defeat. Incorrect because the respective EF-V-30s must be placed in hand to clear the seal-in relay. (2) Correct Location
D.	(1) 25" on the startup range (2) OTSG 'A' EFW valves on CL and OTSG 'B' EFW Valves on CC	CORRECT: (1) See above. (2) See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	E04	2.1.31	
	Importance Rating	4.6		

K/A: Inadequate Heat Transfer - Loss of Secondary Heat Sink: Ability to locate control room switches controls and indications, and to determine that they correctly reflect the desired plant lineup.

Proposed Question:	Question #18	
Technical Reference(s):	TQ-TM-104-644-C001, Rev 2	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	644-GLO-5	

Question Source:	Bank #		
	Modified Bank #	719815	

	New			
Question History:	System Exam 9	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know that the correct level setpoint for the EFW valves for a Lack of Heat Transfer condition. Also, they must know the location of the controllers for the valves.

High Cog: This question is high cog because the examinee must know the EFW setpoint for a loss of both MFW pumps and how to clear the memory for and EFW valve setpoint.

Plant Conditions:

- Loss of BOTH FW pumps
- Reactor and Turbine trip

Event:

- All 8 (eight) HSPS switches placed in DEFEAT.
- No other operator actions have been taken.

What is the setpoint, from HSPS, for OTSG level control?

A. 0" on the startup range.

B. 25" on the startup range.

C. 50% on the operating range.

D. 75-85% on the operating range.

Answer B

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

19 **ID: 2087129** **Points: 1.00**

Plant Conditions:

- Reactor power was lowered to 50% with ICS in auto

Event:

- ICS T-Ave setpoint signal fails ramping HIGHER at 5°F/minute

Which of the following describes the plant response to this malfunction?

Reactor power _____.

A. rises because of ICS automatic operation

B. lowers because of ICS automatic operation

C. does not change because ICS transfers to Track Mode

D. does not change because the Diamond transfers to Manual Control

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Tave setpoint failing high (as compared to the Tave actual) a positive error sign will be developed (called Neutron Error) in the Reactor Demand Subsystem of ICS (D553732, Rev P). (2) This positive Neutron Error will cause a rod withdrawal. (3) This malfunction makes it appear that Tave is below the setpoint. The ICS system will raise Tave by pulling the control rods further out, which causes power to rise.

A.	rises because of ICS automatic operation	CORRECT: See above.
B.	drops because of ICS automatic operation	INCORRECT: Plausible because this is the effect if the actual Tave signal (rather than the setpoint) began to fail high.
C.	does not change because ICS transfers to Track Mode	INCORRECT: Plausible if examinee thinks that a large Tave error signal would cause Cross Limits and initiate the Tracking Mode. This would not have an effect on Reactor Power increasing.
D.	does not change because the Diamond transfers to Manual Control	INCORRECT: Plausible if the examinee believes the Diamond would swap to manual control due to a large Neutron Error signal. Incorrect because a large Neutron Error signal does not transfer the Diamond to manual, but it does prevent transferring the Diamond to auto.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	001	AA2.04	
	Importance Rating	4.2		

K/A: Continuous Rod Withdrawal: Ability to determine and interpret the following as they apply to Continuous Rod Withdrawal: Reactor Power and its trend

Proposed Question:	Question #19	
Technical Reference(s):	D553732, Rev P	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	621-GLO-11	

Question Source:	Bank #		
	Modified Bank #	462750	
	New		

Question History:	System Exam 13	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the relation between the ICS Tave setpoint failing high and the effect on the control rods. The control rods will continuously pull to maintain Tave as the setpoint fails high, which raises reactor power.

High Cog: The examinee has to distinguish between the effect of a setpoint failure versus a Tave instrument failure.

Plant Conditions:

- Reactor power is 50% with ICS in automatic.

Sequence of Events:

- ICS T-Ave SETPOINT signal fails slowly, ramping HIGHER at 5°F/minute. due to setpoint potentiometer malfunction.
- ICS control mode does NOT transfer to Tracking during this sequence.

Based on these conditions, identify the ONE statement below that describes initial reactor power trend and control system response to this event.

A. Reactor power RISES due to rod withdrawal.

B. Reactor power is REDUCED due to rod insertion.

C. Reactor power DOES NOT CHANGE due to CRD logic circuit operation.

D. Reactor power DOES NOT CHANGE due to T-Ave control transfer to Feedwater.

Answer A

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

20 **ID: 2097107** **Points: 1.00**

Plant Conditions:

- Reactor power is 50% with ICS in auto
- All Group 5 rodes are fully withdrawn

Event:

- A Group 5 rod drops fully into the core
- MAP G-2-1 CRD PATTERN ASYMMETRIC and the associated PPC Alarm for asymetric rod fault both come in

Which of the following correctly describes the effect of the dropped rod on the Group 5 average position?

Group 5 average position will _____.

A. include the dropped rod and indicate lower after the rod is dropped

B. include the dropped rod upon pressing RPI RESET on the Diamond

C. NOT include the dropped rod until the rod is restored to within 7" delta

D. NOT include the dropped rod until the rod is restored to within 9" delta

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When the absolute position of any CRDM deviates by more than 9" from its group average, and the ASYMMETRIC ROD FAULT BYPASS is false (OP-TM-622-000, Rev 8 Page 48 and 49) a ASYMMETRIC ROD FAULT is issued. Any rod generating an ASYMMETRIC ROD FAULT is removed from the Group Average Calculation. (2) In OP-TM-AOP-062, INOPERABLE ROD (Rev 7 Page 20), the note before Step 1.3 states that at 9" delta the dropped rod height will be restored to the group average calculation causing a small downward step change in group average height.

A.	include the dropped rod and indicate lower after the rod is dropped	INCORRECT: Plausible because the examinee could believe that the Group 5 average is simply the average of all the control rods in that group. Incorrect because the group average does not include rods that are greater than 9" asymmetric.
B.	include the dropped rod upon pressing RPI RESET on the Diamond	INCORRECT: Plausible because the examinee could know that the group average does not include the dropped rod, but not know the correct method in which the rod gets included into the average. Incorrect because the rod gets included when it is less than 9" apart from the group average.
C.	NOT include the dropped rod until the rod is restored to within 7" delta	INCORRECT: Plausible because the examinee could know that the group average does not include the dropped rod, but not know the correct height in which the rod gets included into the average. Incorrect because it gets included in the group average at 9".
D.	NOT include the dropped rod until the rod is restored to within 9" delta	CORRECT: See above

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	003	AA2.04	
	Importance Rating			

K/A: Dropped Control Rod: Ability to operate and / or monitor the following as they apply to the Dropped Control Rod: Demand position counter and pulse/analog converter

Proposed Question:	Question 20	
Technical Reference(s):	OP-TM-AOP-062, Rev 7	OP-TM-622-000, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 622-GLO-10

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.6	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know how to the demand position counter and pulse/analog converter (Relative Position Indication at Three Mile Island) operates. They must know that when a rod is being recovered that it will automatically be included back in the group average when it is within 9" of the group average.

21 **ID: 2095202** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- A Control Rod in Group 7 partially drops and is currently 10" below the group average

In accordance with OP-TM-AOP-062, INOPERABLE ROD, which one of the following is correct regarding re-aligning the control rod back to its group average while maintaining reactor power stable?

This control rod must be aligned to the group average within ___(1)___.

This time limit restriction is to prevent ___(2)___.

A. (1) 1 hour
(2) exceeding quadrant power tilt limits

B. (1) 1 hour
(2) power peaking and potential fuel damage

C. (1) 2 hours
(2) exceeding quadrant power tilt limits

D. (1) 2 hours
(2) power peaking and potential fuel damage

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) If a rod were to partially drop into the core (as stated in the stem) the crew would enter OP-TM-AOP-062, INOPERABLE ROD (Rev 7 Page 1) based on the control rod being misaligned greater than 9" from its group. (2) In accordance with OP-TM-AOP-062, Step 3.8 (Rev 7, Page 7) if the rod is not at its in-limit, the rod is misaligned by more than 9 inches and the rod has been misaligned for less than an hour the crew must reset the relative position indication and align the control rod with its group average. (3) The basis for the one hour time limit in accordance with OP-TM-AOP-062, INOPERABLE ROD BASIS DOCUMENT (Rev 6 Page 7) is to prevent power peaking and potential fuel damage due to lower assembly Xenon levels resulting from power suppression in the assembly with the misaligned rod.

A.	(1) 1 hour (2) exceeding quadrant power tilt limits	INCORRECT: (1) Correct answer. (2) Plausible because OP-TM-AOP-062 does verify that tilt is in within the limits of the COLR. Incorrect because tilt is not the reason for the 1 hour limit to re-align the control rod.
B.	(1) 1 hour (2) power peaking and potential fuel damage	CORRECT: See above.
C.	(1) 2 hours (2) exceeding quadrant power tilt limits	INCORRECT: (1) Plausible because the crew must complete a power reduction to less than 60% if the control rod cannot be aligned to its group average. Incorrect because this time limit has to do with the power reduction and not the rod re-alignment. (2) Plausible because OP-TM-AOP-062 does verify that tilt is in within the limits of the COLR. Incorrect because tilt is not the reason for the 1 hour limit to re-align the control rod.
D.	(1) 2 hours (2) power peaking and potential fuel damage	INCORRECT: (1) Plausible because the crew must complete a power reduction to less than 60% if the control rod cannot be aligned to its group average. Incorrect because this time limit has to do with the power reduction and not the rod re-alignment. (2) Correct answer.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	005	AK1.06
	Importance Rating	2.9	

K/A: Inoperable/Stuck Control Rod: Knowledge of the operational implications of the following concepts as they apply to Inoperable/Stuck Control Rod: Bases for power limit for rod misalignment.

Proposed Question:	Question #21	
Technical Reference(s):	OP-TM-AOP-062, Rev 7	OP-TM-AOP-0621, Rev 6

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP062-PCO-1	

Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.6	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know that if a control rod cannot be re-aligned that power will have to be lowered. The examinee must know the bases for the 1 hour limit at the current power level. The operational implication is that the rod must be aligned within 1 hour or power must be lowered. The basis of the time/power limit is that power peaking and potential fuel damage could occur.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

22 **ID: 2089477** **Points: 1.00**

Plant Conditions:

- Plant Startup in progress
- NI-12 and NI-12A are out of service
- 1103-8, APPROACH TO CRITICAL is being performed
- Reactor is NOT critical

Event:

- Loss of VBA

Which one of the following correctly describes whether a plant startup can be performed in this condition?

The plant startup ___(1)___.

A. may continue using NI-11A computer indication

B. may continue after NI-11A is cross connected to NI-11

C. may NOT continue because NO source range instrument is available

D. may NOT continue because ONLY one source range instrument is available

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When Vital Bus A (VBA) is lost, the last remaining source range instrument is lost (OP-AOP-015, Rev 11 Page 21). (2) Since the crew is performing 1103-8, APPROACH TO CRITICAL (and the reactor is NOT critical) with one source range instrument that when VBA is lost the crew has no indication of reactor power. (3) Technical Specification 3.5.1.1 (AMD 189 Page 3-27, AMD 247 Page 3-29, and AMD 189 Page 3-30) state that the crew should restore 1 source range instrument to operable to place the unit in HOT SHUTDOWN within 6 additional hours.

A.	may continue using NI-11A computer indication	INCORRECT: Plausible if examinee thinks NI-11A is powered from a power source other than VBA. Incorrect because NI-11A is powered from VBA.
B.	may continue after NI-11A is cross connected to NI-11	INCORRECT: Plausible since NI-11A is the installed backup to NI-11 and has to be physically cross connected to perform this function. Incorrect because NI-11A is powered from VBA.
C.	may NOT continue because NO source range instrument is available	CORRECT: See above.
D.	may NOT continue because ONLY one source range instrument is available	INCORRECT: Plausible if the examinee believes there are more source range instruments available. Incorrect because neither source range instrument has power.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	032		AK2.01
	Importance Rating	2.7		

K/A: Loss of Source Range Nuclear Instrumentation: Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: Power supplies including proper switch positions.

Proposed Question:	Question #22	
Technical Reference(s):	OP-TM-AOP-015, Rev 11	Technical Specification 3.5.1 AMD 247 and 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:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: The question matches the KA because the examinee must know the power supply to the Source Range Instruments and the operational effects of losing power.

High Cog: This question is high cog because the examinee must know the effect of a loss of VBA will have on a reactor startup with NI-12 out of service.

23 **ID: 2097109** **Points: 1.00**

Regarding required instrument overlap, intermediate range indication must come on scale by the time source range indicates _____.

A. 1E5 CPS, and must track for at least 1 decade

B. 1E5 CPS, and must track for at least 2 decades

C. 1E6 CPS, and must track for at least 1 decade

D. 1E6 CPS, and must track for at least 2 decades

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The source range indication covers a reactor power from 10E-1 to 10E6 CPS. (2) In accordance with 1103-8, APPROACH TO CRITICALITY (Rev 55 Page 1) states that the overlap between the source range and intermediate range shall be greater than 1 decade. (3) On Step 4.11.7 (Page 6) requires the intermediate range to be at 10E-11 amps by the time the source range is 10E5 cps.

A.	1E5 CPS, and must track for at least 1 decade	CORRECT: See above.
B.	1E5 CPS, and must track for at least 2 decades	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes that 2 decades is the requirement. Incorrect because 1 decade is the requirement.
C.	1E6 CPS, and must track for at least 1 decade	INCORRECT: (1) Plausible because this is a reading on the source range instruments. Incorrect because this is the top of the scale and the proper overlap would not be observed if the intermediate range did not come on scale until then. (2) Correct answer.
D.	1E6 CPS, and must track for at least 2 decades	INCORRECT: (1) Plausible because this is a reading on the source range instruments. Incorrect because this is the top of the scale and the proper overlap would not be observed if the intermediate range did not come on scale until then. (2) Plausible if the examinee believes that 2 decades is the requirement. Incorrect because 1 decade is the requirement.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	033	AA2.12
	Importance Rating	2.5	

K/A: Loss of Intermediate Range Nuclear Instrumentation: Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Maximum allowable channel disagreement

Proposed Question:	Question #23
Technical Reference(s):	1103-8, Rev 55
Proposed References to be provided to applicants during examination:	None
Learning Objective:	GOP-003-PCO-5

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

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

10 CFR Part 55 Content:	55.41	b.6
	55.43	

Comments:

KA Match: This question matches the KA because proper overlap is one of the intermediate range characteristics that operators at Three Miles Island use to verify that an intermediate range instruments are working correctly. If the intermediate range instrument comes on scale too early, that would indicate to an operator that the channel is outside of its max allowable band, and thus not a good instrument to use. If the operators cannot verify overlap on the intermediate range instruments then all of the intermediate range instruments are lost.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

24 **ID: 2107617** **Points: 1.00**

Plant conditions:

- The reactor is at 100% power with ICS in full auto
- The inner door of the Containment Personnel Airlock fails and cannot be closed
- All required Technical Specification actions are complied with

Event:

- Reactor Building entry is required to repair the inner door

Which of the following identifies the action necessary to ensure the Containment Integrity requirements are satisified?

A. The reactor must be shutdown to hot standby prior to opening the outer door.

B. The outer door may be opened provided it is immediately closed after passage.

C. The RCS must be cooled to less than 200°F, with pressure reduced to less than 500 psig prior to opening the outer door.

D. A temporary containment must be set up at the Auxiliary Building entrance from the Reactor Building prior to opening the outer door.

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1101-3, CONTAINMENT INTEGRITY AND ACCESS LIMITS (Rev 94 Page 7), 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. (2) Since the Personnel Hatch inner door has failed, one door of the personnel or emergency air lock may be open for maintenance, repair or modification provided the other door of the air lock is verified closed within 1 hour, locked within 24 hours, and verified to be locked closed monthly. This action is assumed to have occurred since it is given that all Technical Specification action has been complied with. (2) With the personnel or emergency air lock door interlock mechanism inoperable, entry and exit is permissible, per Technical Specification 3.6.2.c, under the control of a dedicated individual. Therefore, if an emergency entry is required, the opening of the outer door is to be controlled by a dedicated individual.

A.	The reactor must be shutdown to hot standby prior to opening the outer door.	INCORRECT: Plausible because some of the Containment Technical Specifications, such as building internal pressure, are dependent upon the operating status of the reactor. The operator may incorrectly believe that this Technical Specification is also dependent on the same.
B.	The outer door may be opened provided it is immediately closed after passage.	CORRECT: See above.
C.	The RCS must be cooled to less than 200°F, with pressure reduced to less than 500 psig prior to opening the outer door.	INCORRECT: Plausible because the examinee could believe that if containment integrity is not required then the outer door can be open. Incorrect because the requirement to relax containment is below 300 psig, not 500 psig.
D.	A temporary containment must be set up at the Auxiliary Building entrance from the Reactor Building prior to opening the outer door.	INCORRECT: Plausible because the examinee could believe that a temporary containment could be setup to prevent the spread of any contamination which may be in the Reactor Building. Incorrect because no procedure directs the use of temporary containment outside of the Auxiliary Building.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	069	AA2.02
	Importance Rating	2.8	

K/A: Loss of Containment: Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: Verification of automatic and manual means of restoring integrity.

Proposed Question:	Question #24	
Technical Reference(s):	1101-3, Rev 94	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	240-GLO-14	

Question Source:	Bank #	881713		
	Modified Bank #			
	New			
Question History:	N/A	Last NRC Exam:	N/A	

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

Comments:

KA Match: This question matches the KA because the examinee must know how to restore containment integrity with a malfunctioning inner RB door.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

25 **ID: 2088596** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Loss of Offsite Power
- EG-Y-1A failed to automatically start and was successfully started manually from the control room
- The following indications are available in the main control room for EG-Y-1A:

Frequency	60.2 Hz
Voltage	4275 Volts
Load	3.3 MWe

In accordance with OP-TM-861-901, DIESEL GENERATOR EG-Y-1A EMERGENCY OPERATIONS, what actions must be taken to correct the out of specification reading on EG-Y-1A?

A. Place the exciter in manual and lower voltage

B. Adjust governor control switch in the lower direction

C. Lower unit voltage rheostat setting at the local alarm panel

D. Shutdown non-essential engineering safeguards bus components

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) That 3.3 MWe is above the rated load for EG-Y-1A. (2) The corrective action in accordance with OP-TM-861-901, DIESEL GENERATOR EG-Y-1A EMERGENCY OPERATIONS (Rev 19, Page 3) is to shutdown non-essential loads to get less than 3.0 MWe.

A.	Place the exciter in manual and lower voltage	INCORRECT: Plausible because EG-Y-1A is operating above normal voltage. Incorrect since the voltage is within the correct band of 4100V to 4300V per OP-TM-861-901. In addition, this action will not lower the load on the diesel generator.
B.	Adjust governor control switch in the lower direction	INCORRECT: Plausible because EG-Y-1A is operating above normal frequency. Incorrect because the frequency is within the band of 59-61 hertz per OP-TM-861-901. In addition adjusting the governor will lower frequency and not load.
C.	Lower unit voltage rheostat setting at the local alarm panel	INCORRECT: Plausible because EG-Y-1A is operating above normal voltage. Incorrect since the voltage is within the correct band of 4100V to 4300V per OP-TM-861-901. In addition, this action will not lower the load.
D.	Shutdown non-essential Engineering Safeguards bus components	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	BW A05	AA1.3	
	Importance Rating	3.7		

K/A: Emergency Diesel Actuation: Ability to operate an/or monitor the following as they apply to the (Emergency diesel Actuation): Desired operating results during abnormal and emergency situations.

Proposed Question:	Question #25	
Technical Reference(s):	OP-TM-861-901, Rev 19	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	861-GLO-10	

Question Source:	Bank #	805575	
	Modified Bank #		
	New		

Question History:	Simulator Exam 7	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know which parameters are out of specification and the appropriate action to take (or what to operate) for the out of specification. The examinee must identify that the diesel generator is operating overloaded and that the crew must shutdown (operate) non-essential loads

High Cog: This question is high cog because the examinee must analyze the parameters in the stem and identify the out of specification. In addition, the examinee must know the procedural action to correct the out of specification.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

26 **ID: 2098967** **Points: 1.00**

TS 3.14.2, FLOOD CONDITION FOR PLACING THE UNIT IN HOT STANDBY, states that action must be taken upon reaching which of the following river conditions?

A. 296.0 ft elevation

B. 302.0 ft elevation

C. 350,000 cfs river flow

D. 640,000 cfs river flow

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Technical Specification 3.14.2 FLOOD CONDITION FOR PLACING THE UNIT IN HOT STANDBY (AMD 157 Page 3-60) direct the unit to be brought to hot standby the river reaches 302 feet, which corresponds to 1,000,000 cfs river flow.

A.	296.0 ft elevation	INCORRECT: Plausible because OP-TM-AOP-002, FLOOD requires the until to be shutdown at this level. Incorrect because this is not the technical specification limit.
B.	302.0 ft elevation	CORRECT: See above.
C.	350,000 cfs river flow	INCORRECT: Plausible because this is an entry criteria into OP-TM-AOP-002. Incorrect because this is not the technical specification limit.
D.	640,000 cfs river flow	INCORRECT: Plausible because there is an action in OP-TM-AOP-002 at 640,000 CFS. Incorrect because this action is to return to normal when it is this CFS and lowering.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	BW A07	2.2.22	
	Importance Rating	4.0		

K/A: Flooding: Knowledge of limiting conditions for operations and safety limits

Proposed Question:	Question #26	
Technical Reference(s):	OP-TM-AOP-002, Rev 15	TS 3.14.2, Amd 157

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP-002-PCO-4

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

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

10 CFR Part 55 Content:	55.41	b.4
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the limiting condition for operation flooding.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

27 **ID: 2087792** **Points: 1.00**

In accordance with Rule 2, HPI THROTTLING, HPI is required to be throttled when Subcooling Margin is _____ .

A. ≥ 250F because full HPI flow could challenge reactor vessel integrity

B. > 25F because adequate control of core cooling has been demonstrated

C. ≥ 250F because adequate control of core cooling has been demonstrated

D. > 25F because full HPI flow could challenge reactor vessel integrity

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) in Rule 2, HPI THROTTLING of OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES AND GRAPHS (Rev 20 Page 2) Step 2 directs the operators to verify Subcooling Margin (SCM) is less than 250F. (2) If it is not the operators must throttle HPI to control SCM to less than 250F. (3) The basis in accordance with OP-TM-EOP-0101, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS BASIS DOCUMENT (Rev 11 Page 10), Step A2 is to prevent failure of the reactor vessel by violating the reactor vessel pressure-temperature limit.

A.	≥ 250F because full HPI flow could challenge reactor vessel integrity	CORRECT: See above.
B.	> 25F because adequate control of core cooling has been demonstratedn demonstrated	INCORRECT: Plausible because the examinee could believe that as along as SCM is >25F that HPI is required to be throttled. Incorrect because this criteria is not reason alone to throttle HPI in accordance with Rule 2.
C.	≥ 250F because adequate control of core cooling has been demonstrated	INCORRECT: Plausible because adequate core cooling has been demonstrated by this point. Incorrect because when SCM is this high, the crew must throttle due to the pressure stress on the reactor vessel.
D.	> 25F because full HPI flow could challenge reactor vessel integrity	INCORRECT: Plausible because the examinee could believe that as along as SCM is >25F that HPI is required to be throttled. Incorrect because this criteria is not reason alone to throttle HPI in accordance with Rule 2.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	2		
	K/A #	BW E13	EK3.02	
	Importance Rating	3.2		

K/A: EOP Rules and Enclosures: Knowledge of the reasons for the following responses as they apply to the (EOP Rules): Normal, abnormal and emergency operating procedures associated with (EOP Rules)

Proposed Question:	Question #27	
Technical Reference(s):	OP-TM-EOP-010, Rev 20	OP-TM-EOP-0101, Rev 11

Proposed References to be provided to applicants during examination:	None

Learning Objective:	HPI-PCO-1	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.2	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the basis behind the HPI throttling step in Rule 2 (EOP Rule). Rule 2 sends the operator to 900 series procedure (abnormal operating procedure) in order to throttle.

28 **ID: 2084731** **Points: 1.00**

Which of the following will prevent a Reactor Coolant Pump from starting?

A. 7KV Bus Voltage below 6.8KV

B. ICCW total flow below 900 gpm

C. Seal Injection flow below 32 gpm

D. Motor Oil Lift pressure below 1000 psig

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know (1) the interlocks which are required to be met to start are Reactor Coolant Pump. OP-TM-226-000, REACTOR COOLANT PUMPS (Rev 14 Page 14) gives a list of the interlocks that must be met. (2) All the distractors at or below their normal value.

A.	7KV Bus Voltage below 6.8KV	INCORRECT: Plausible because low 7kV bus voltage will prevent the crew from starting a Reactor Coolant Pump. Incorrect because this limit is 6.15 kV.
B.	ICCW total flow below 900 gpm	INCORRECT: Plausible because low ICCW flow will prevent the crew from starting a Reactor Coolant Pump. Incorrect because this limit is 550 gpm.
C.	Seal Injection flow below 32 gpm	INCORRECT: Plausible because low seal injection will prevent the crew from starting a Reactor Coolant Pump. Incorrect because this limit is 22 gpm.
D.	Motor Oil Lift pressure below 1000 psig	CORRECT: Correct answer. Motor Oil Lift System pressure must be above 1000 psig.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	003	K4.03	
	Importance Rating	2.5		

K/A: Reactor Coolant Pump: Knowledge of RCPS design feature(s) and/or interlocks which provide for the following: Adequate lubrication of the RCP

Proposed Question:	Question #28	
Technical Reference(s):	OP-TM-226-000, Rev 14	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	226-GLO-5	

Question Source:	Bank #	502404	
	Modified Bank #		
	New		
Question History:	Simulator Exam 5	Last NRC Exam:	N/A

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the motor oil starting interlock for a Reactor Coolant Pump, which is an interlock to ensure adequate lubrication of a Reactor Coolant Pump.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

29 **ID: 2103861** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Reactor trip due to high RCS pressure
- Failed open Pressurizer safety valve resulted in ES actuation
- 1P 480V Bus trip due to electrical fault at the time of the ES actuation

Identify the selection that completes the following statement with regard to the following valves:

- Borated Water Storage Tank Outlet Valves, MU-V-14A and MU-V-14B
- DH Pump Discharge to Makeup Pump Valves, DH-V-7A and DH-V-7B

_____ will OPEN due to the ES actuation signal.

A. ONLY DH-V-7B

B. ONLY MU-V-14B

C. BOTH DH-V-7A and DH-V-7B

D. BOTH MU-V-14A and MU-V-14B

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The power supply to the BWST suction valve MU-V-14B is 1B 480V ES Valves MCC, which is powered from the 1S 480V ES Bus (1107-4, ELECTRICAL DISTRIBUTION PANEL LISTING (Rev 240 Page 53)). (2) The power supply to the BWST suction valve MU-V-14A and the DH pump B to Makeup Pump valve DH-V-7A is 1A ES MCC, which is powered from 1P 480V ES MCC (1107-4, ELECTRICAL DISTRIBUTION PANEL LISTING (Rev 240 Page 46)). The 1P 480V ES MCC loses power in the stem, so neither of these valves move position.

A.	ONLY DH-V-7B	INCORRECT: Plausible misconception that the DH-V-7B valve would open on an ES. The DH-V-7 valves could offer an alternate flowpath for the Decay Heat Pumps. The DH-V-4 valves are the normal outlet valves of the Decay Heat Pumps. They recieve an open signal on an ES, but no flow until RCS pressure drops below 400 psig. The examinee could believe that both sets of outlet valves open. In addition, the DH-V-7 valves do have indication on the ES status board. In this case, the DH-V-7A would not open because the 1P 480V bus has tripped. Incorrect because the DH-V-7B does not open on ES.
B.	ONLY MU-V-14B	CORRECT: See above
C.	BOTH DH-V-7A and DH-V-7B	INCORRECT: Plausible misconception that the DH-V-7B valve would open on an ES. The DH-V-7 valves could offer an alternate flowpath for the Decay Heat Pumps. The DH-V-4 valves are the normal outlet valves of the Decay Heat Pumps. They recieve an open signal on an ES, but no flow until RCS pressure drops below 400 psig. The examinee could believe that both sets of outlet valves open. In addition, the DH-V-7 valves do have indication on the ES status board. Incorrect because these valves do not open on ES.
D.	BOTH MU-V-14A and MU-V-14B	INCORRECT: Plausible if the examinee believes both MU-V-14A and B have power. Incorrect because MU-V-14A does not have power.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	004	K2.05	
	Importance Rating	2.7		

K/A: Chemical and Volume Control: Knowledge of bus power supplies to the following: MOVs.

Proposed Question:	Question #29	
Technical Reference(s):	1107-4, Rev 240	

Proposed References to be provided to applicants during examination: None

Learning Objective:	642-GLO-8	

Question Source:	Bank #	363623		
	Modified Bank #			
	New			
Question History:	Simulator Exam 5	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.8	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know that MU-V-14A will not open due to the loss of the 1P 480V Bus.

High Cog: This question is high cog because the examinee must know the power supply to the each of the MU-V-14's and that on an ES actuation that the valves will open.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

30 **ID: 2084732** **Points: 1.00**

Plant conditions:

- Reactor is at cold shutdown condition
- DHR Train A is operating
- DC-V-2A (Cooler Inlet) and DC-V-65A (Cooler Bypass) are both in intermediate positions

Event:

- Total loss of Instrument Air (0 psig)

Which of the following completes the statement to describe the response of DC-V-2A and DC-V-65A to this event?

DC-V-2A will ___(1)___ and DC-V-65A will ___(2)___.

A. (1) open
(2) open

B. (1) open
(2) close

C. (1) close
(2) close

D. (1) close
(2) open

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Due to the loss if instrument air, that the both the inlet/bypass (DC-V-2A/65A) valves to the operating Decay Heat Closed Cooling Cooler would go to their fail safe position (full cooling) (2) The fail safe position for DC-V-2A is full open, and the fail safe position for DC-V-65A is closed. This is referenced in OP-TM-AOP-028, LOSS OF INSTRUMENT AIR (Rev 9 Page 27) Attachment 6.1.

A.	(1) open (2) open	INCORRECT: Plausible if the examinee believes that DC-V-65A fails closed. Incorrect because it fails open.
B.	(1) open (2) close	CORRECT: See above.
C.	(1) close (2) close	INCORRECT: Plausible if the examinee believes that both valves (DC-V-2A/65A) have the same failure method. Incorrect because DC-V-2A fails open and DC-V-65A fails closed.
D.	(1) close (2) open	INCORRECT: Plausible if the examinee believes that both valves (DC-V-2A/65A) have the same failure method. Incorrect because DC-V-2A fails open and DC-V-65A fails closed.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	005	K6.03	
	Importance Rating	2.5		

K/A: Residual Heat Removal: Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR Heat Exchanger

Proposed Question:	Question #30	
Technical Reference(s):	OP-TM-AOP-028, Rev 9	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	543-GLO-8	

Question Source:	Bank #	355400	
	Modified Bank #		
	New		

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

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

10 CFR Part 55 Content:	55.41	b.8

	55.43		

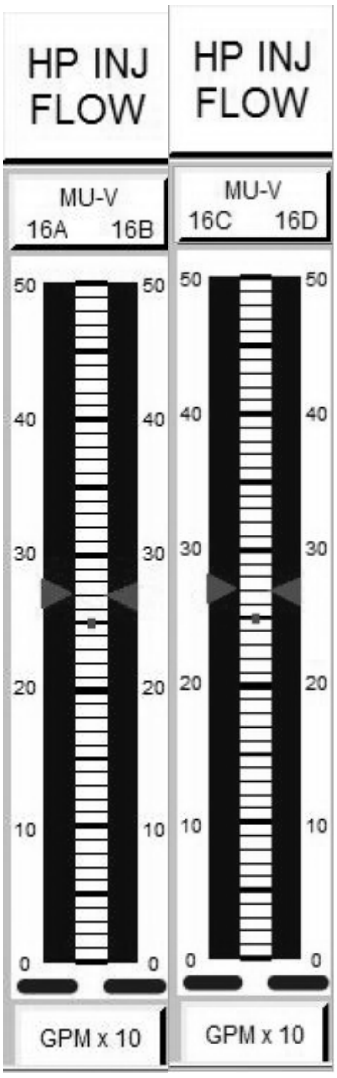
Comments:

KA Match: This question matches the KA because the examinee must have knowledge the effects that a malfunction on the decay closed heat exchanger inlet valve will have on the decay heat system.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

31 **ID: 2089320** **Points: 1.00**



Plant Conditions:

- Large Break LOCA
- Reactor is tripped

Based on the indications above, for the given conditions, which, if any, MU-V-16s (HPI Control Valves) must be throttled to maintain limits?

A. All of the MU-V-16 valves

B. None of the MU-V-16 valves

C. MU-V-16A and/or MU-V-16B

D. MU-V-16C and/or MU-V-16D

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The indication provided in the question measures flow at the in the HPI lines prior to the cross connects and MU-V-16s (Drawing 302661, Rev 63), so each set of indicators is measuring actual HPI flow from the Makeup Pumps running on that side. (2) In this lineup, MU-P-1A and B flow is measured by the HPI flows for MU-V-16A and MU-V-16B, and only MU-P-1C flow is measured by the HPI flows for MU-V-16C and MU-V-16D. (3) Rule2, HPI THROTTLING in OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS (Rev 20 Page 4) verifies MU PUMP FLOW is less than 515 gpm/pump, and if it is not then the operator must throttle flow to maintain the flow less than 515 gpm. (4) In this question, the flow through MU-V-16A and B is approximately 540 gpm total (270 gpm/pump) but the flow through MU-V-16C and D is 540 gpm total.

A.	All of the MU-V-16 valves	INCORRECT: Plausible if the examinee determines that flow through the MU-V-16C and D exceeds the limit but believes that all MU-V-16s must be throttled to balance flow in all of the HPI nozzles. Incorrect because the HPI lines are cross connected (A and C, B and D) downstream of the indicators so throttling all MU-V-16s is not required.
B.	None of the MU-V-16 valves	INCORRECT: Plausible if the examinee believes that the makeup pump flow is within the limits of Rule 2. Incorrect because the MU-P-1C is too high.
C.	MU-V-16A and/or MU-V-16B	INCORRECT: Plausible if the examinee believes that only the A side is above the limit or if MU-P-1B were aligned to the B ES train.
D.	MU-V-16C and/or MU-V-16D	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006	A1.09
	Importance Rating	2.8	

K/A: Emergency Core Cooling: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Pump Amperage, including start, normal and locked.

Proposed Question:	Question #31	
Technical Reference(s):	OP-TM-EOP-010, Rev 20	Flow Diagram 302-661, Rev 63

Proposed References to be provided to applicants during examination: None

Learning Objective:	EOP-010-PCO-1	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.8	
	55.43		

Comments:

This question matches the KA because the examinee must identify that MU-P-1C flow is too high. Flow is proportional to the pump amperage and is the primary indicator in the control room that a makeup pump may be overloaded. After the examinee identifies that MU-P-1C flow is too high, the examinee must know which MU-V-16s that must be throttled to maintain MU-P-1C within limits.

High Cog: This question is high cog because the examinee must determine the Makeup Pump flows from the given indications and determine which is over the limit.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

32 **ID: 2109100** **Points: 1.00**

Plant Conditions:

- Reactor Coolant System fill in progress.
- The Pressurizer is vented to the Reactor Coolant Drain Tank.
- All Reactor Coolant Drain Tank manual vent valves are closed.
- The Reactor Vessel is vented to the Reactor Building via the CRDM vents.
- The Hot Legs are vented to the Reactor Building via high point vents.

Event:

- WDG-V-4, Containment Isolation RB Vent Header, fails closed.

If the fill evolution were to continue, which of the following describes when Reactor Building Sump level will begin to rise?

A. After the Pressurizer has filled, when the RC Drain Tank rupture disk has failed.

B. Before Pressurizer level reaches 400 inches, when flow begins out the CRDM vents.

C. Before Pressurizer level reaches 400 inches, when flow begins out the Hot Leg vents.

D. After the Pressurizer has filled, when the RC Drain Tank overflows through the relief valve.

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1103-11, RCS WATER LEVEL CONTROL, components are vented to the RCDT while the RCS is being filled. When WDG-V-4 fails closed, the RCDT no longer will vent the gasses to the Waste Gas header and will begin to pressurize. The more gas that gets vented to the RCDT, the higher the pressure will get until the rupture disk relieves. When the rupture disk relieves, the RB sump level will begin ro rise. The Pressurizer and RCDT will be hydraulically locked which will cause the Reactor Vessel to overflow at a lower indicated pressurizer level than expected. The relief valve (WDG-V-1) setpoint will be exceeded, but because WDG-V-4 is closed it will not relieve any pressure.

A.	After the Pressurizer has filled, when the RC Drain Tank rupture disk has failed.	INCORRECT: Plausible because the examinee may believe that pressure will not begin to rise until the pressurizer is full. Incorrect because pressure in the RCDT will rise when WDG-V-4 goes closed.
B.	Before Pressurizer level reaches 400 inches, when flow begins out the CRDM vents.	CORRECT: See above.
C.	Before Pressurizer level reaches 400 inches, when flow begins out the Hot Leg vents.	INCORRECT: Plausible if the examinee believes that the Hot Leg vents are at a higher level than the CRDM vents. Incorrect because they are not.
D.	After the Pressurizer has filled, when the RC Drain Tank overflows through the relief valve.	INCORRECT: Plausible because the examinee may believe that pressure will not begin to rise until the pressurizer is full. Incorrect because pressure in the RCDT will rise when WDG-V-4 goes closed. In addition, the relief valve WDG-V-1 will not relieve any pressure due to WDG-V-4 failing closed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007	K3.01
	Importance Rating	3.3	

K/A: Pressurizer Relief/Quench Tank: Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment

Proposed Question:	Question #32	
Technical Reference(s):	308946, Rev 6	1103-11, Rev 77

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO-2

Question Source:	Bank #	575035	
	Modified Bank #		
	New		

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

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know what a failure of the PRTS (WDG-V-4) will due to containment. The failure will cause containment sump water level to rise. The examinee must analyze the what the closure of WDG-V-4 will have on the PRTS system that causes the containment level to rise.

High Cog: This question is high cog because the examinee must analyze the failure and determine effect on the Pressurizer water level and RB sump level.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

33 **ID: 2104650** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- Intermediate Closed Cooling Radiation Monitor, RM-L-9, indication is high at 1E5 CPM due to an earlier RCS leak into ICCW that has been isolated

Event:

- MAP C-3-2, IC SURGE TANK LEVEL HI/LO ACTUATED

Currently:

- Intermediate Cooling Surge Tank, IC-T-1, level is 7 inches, lowering at 1 inch per minute
- RM-L-7, Plant Effluent Rad Monitor, reading is slowly rising

Which one of the following describes the location of the leak and the procedure that must be entered for this condition?

A. (1) Inside one of the Letdown Coolers, MU-C-1A/B
(2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING

B. (1) Inside one of the Intermediate Service Coolers, IC-C-1A/B
(2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING

C. (1) Inside one of the Letdown Coolers, MU-C-1A/B
(2) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE

D. (1) Inside one of the Intermediate Service Coolers, IC-C-1A/B
(2) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The plant conditions that an RCS leak into the ICCW has already occurred (and been isolated) and the RM-L-9 reading is still high. (2) When MAP-C-3-2 comes in and RM-L-7 is rising, the examinee should determine that there is a leak from ICCW in the Intermediate Service Cooler IC-C-1A/B. (3) In addition, the entry criteria for OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING (Rev 5 Page 1) have been met.

A.	(1) Inside one of the Letdown Coolers, MU-C-1A/B (2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING	INCORRECT: (1) Plausible because these coolers are cooled by Intermediate Closed Cooling Water. Incorrect because counts are going up in the plant discharge monitor, so the leak must be in the Intermediate Service Cooler. (2) Correct Answer.
B.	(1) Inside one of the Intermediate Service Coolers, IC-C-1A/B (2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING	CORRECT: See above.
C.	(1) Inside one of the Letdown Coolers, MU-C-1A/B (2) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE	INCORRECT: (1) Plausible because these coolers are cooled by Intermediate Closed Cooling Water. Incorrect because counts are going up in the plant discharge monitor, so the leak must be in the Intermediate Service Cooler. (2) Plausible because OP-TM-AOP-050 covers RCS leaks in various components but would not be the correct procedures to enter in this time.
D.	(1) Inside one of the Intermediate Service Coolers, IC-C-1A/B (2) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE	INCORRECT: (1) Correct answer. (2) Plausible because OP-TM-AOP-050 covers RCS leaks in various components but would not be the correct procedures to enter in this time.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008	A2.04
	Importance Rating	3.3	

K/A: Component Cooling Water: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: PRMS alarm

Proposed Question:	Question #33	
Technical Reference(s):	OP-TM-AOP-033, Rev 5	OP-TM-MAP-C0302, Rev 4

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP-032-PCO-1	

Question Source:	Bank #			
	Modified Bank #	363656		
	New			
Question History:	unmod on Simulator Exam 6	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee will have to analyze the impact that the rising reading on the liquid radiation monitor has on the plant. The examinee will must identify the lead location from the ICCW surge tank level lowering and RM-L-7 reading rising. The examinee will have to choose the correct AOP to implement to mitigate the consequences.

High Cog: Examinee has to determine the proper location of the leak base on the given plant conditions and then determine the correct procedure flowpath.

Plant Conditions:

- Reactor is operating at 100% power with ICS in full automatic.
- Intermediate Cooling Pump, IC-P-1A, is operating.
- Both Intermediate Service Coolers, IC-C-1A and IC-C-1B, are in service.
- Both Letdown Coolers, MU-C-1A and MU-C-1B, are in service.
- Nuclear Services River Water Pumps, NR-P-1A and NR-P-1C, are operating.
- Intermediate Closed Cooling Radiation Monitor, RM-L-9, indication at 1E5 CPM due to an earlier RCS leak into ICCW that has been isolated.
- RM-L-7, Plant Effluent Rad Monitor, is in service.

Event:

- MAP C-3-2, IC Surge Tank Level Hi/Lo actuated.
- Currently:
  - Intermediate Cooling Surge Tank, IC-T-1, level is 7 inches, LOWERING at 1 inch per minute.
  - RM-L-7 reading is slowly rising.

Based on these conditions, identify the ONE selection below that describes the location of the leak.

A. Inside one of the Letdown Coolers.

B. CRD cooling outlet pipe inside the RB.

C. Intermediate Service Cooler, IC-C-1A/B.

D. Inside one of the Reactor Coolant Pump Thermal Barrier Heat Exchangers.

Answer C

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

34 **ID: 2089093** **Points: 1.00**

Plant Conditions:

- Reactor Trip from 100% power
- OP-TM-EOP-001, REACTOR TRIP IMA's and VSSV's are complete
- Pressurizer level is 98" and the setpoint is 100"

Event:

- Selected Pressurizer Level instrument starts to slowly fail low

Assuming no operator actions, MU-V-17, RCS Makeup Valve, will ___(1)___ and ___(2)___.

A. (1) open
(2) pressurizer heaters will deenergize

B. (1) open
(2) additional pressurizer heaters will energize

C. (1) close
(2) pressurizer heaters will deenergize

D. (1) close
(2) additional pressurizer heaters will energize

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When the pressurizer level instrument starts to fail low, that MU-V-17 will open to try to maintain level at the setpoint and open up. (TQ-TM-104-211-C001 (Rev 11 Page 43)). (2) There is an interlock which deenergizes the pressurizer heaters when the pressurizer level indicates less than 80". (TQ-TM-104-220-C001 (Rev 9 Page 38)).

A.	(1) open (2) pressurizer heaters will deenergize	CORRECT: See above.
B.	(1) open (2) additional pressurizer heaters will energize	INCORRECT: (1) Correct (2) Plausible because pressurizer level will lower. When pressurizer level lowers, pressure lowers as well. Incorrect because while pressurizer level will indicate lower, as MU-V-17 will open and pressurizer level will actually go up.
C.	(1) close (2) pressurizer heaters will deenergize	INCORRECT: (1) Plausible because it the setpoint signal were failing low the MU-V-17 would close. Incorrect because actual level is failing low, not the setpoint signal. (2) Correct Answer.
D.	(1) close (2) additional pressurizer heaters will energize	INCORRECT: (1) Plausible because it the setpoint signal were failing low the MU-V-17 would close. Incorrect because actual level is failing low, not the setpoint signal (2) Plausible because pressurizer level will lower. When pressurizer level lowers, pressure lowers as well. Incorrect because while pressurizer level will indicate lower, as MU-V-17 will open and pressurizer level will actually go up.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	010	K1.08	
	Importance Rating	3.2		

K/A: Pressurizer Pressure Control: Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: PZR LCS

Proposed Question:	Question #34	
Technical Reference(s):	OP-TM-211-472, Rev 4	

Proposed References to be provided to applicants during examination: None

Learning Objective:	211-GLO-11	

Question Source:	Bank #		
	Modified Bank #	354888	
	New		

Question History:	System Exam 11	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the relationship between pressurizer level, the setpoint, and the level that the pressurizer heaters cutout.

High Cog: This question is high cog because the examinee must analyze the malfunction and determine the plant response and identify an interlock that actuates.

Plant Conditions:

- 100% Power.

Event:

- The controlling pressurizer level channel fails high.

Assuming no operator action has occurred, which of the following statements describes the plant response?

The Pressurizer Level Control Valve, (MU-V-17), ___(1)___ and ___(2)___.

A. (1) opens
(2) actual level rises and spray valve opens

B. (1) closes
(2) actual pressurizer level and RCS pressure lower

C. (1) position remains the same
(2) actual pressurizer level and RCS pressure rise

D. (1) position remains the same
(2) actual pressurizer level and RCS pressure lower

Answer B

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

35 **ID: 2104682** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- NI-7 is failed high
- ATA is currently powered from TRA to support troubleshooting

Event:

- MAP A-1-6 INVERTER FAILED alarm comes in
- RPS Channel C cabinet lights are all dark

Which one of the following statements describes the required action(s)?

A. Enter OP-TM-AOP-017, LOSS OF VBC, ONLY

B. Ensure reactor power is reduced to less than 75%

C. Ensure the reactor trips and perform the IMA's of OP-TM-EOP-001, REACTOR TRIP

D. Enter OP-TM-AOP-017, LOSS OF VBC and OP-TM-AOP-027, LOSS OF ATA OR ICS AUTO POWER

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) With the combination of MAP A-1-6 and RPS Channel C lights becoming dark that Vital Bus C was lost. (2) NI-7, which is already failed high, will also lose power because its power supply is Vital Bus C. The plant will still have 3 operable NI's and also meet the required degree of redundancy.		
A.	Enter OP-TM-AOP-017, LOSS OF VBC, ONLY	CORRECT: See above.
B.	Ensure reactor power is reduced to less than 75%	INCORRECT: Plausible because when a vital bus is lost the RPS will only see three Reactor Coolant Pumps in operation. The examinee could believe that a runback occurs and/or the correct action is to lower power less than 75%. Incorrect because those actions are not required.
C.	Ensure the reactor trips and perform the IMA's of OP-TM-EOP-001, REACTOR TRIP	INCORRECT: Plausible because the initial plant conditions have NI-7 failed high, which would give one RPS channel trip. Losing any other Vital Bus would have resulted in an automatic reactor trip. Incorrect because NI-7 is powered from Vital Bus C, so when VBC was lost no further RPS channel trip was generated.
D.	Enter OP-TM-AOP-017, LOSS OF VBC and OP-TM-AOP-027, LOSS OF ATA OR ICS AUTO POWER	INCORRECT: Plausible if the examinee believes that VBC is the power source to TRA. Incorrect because VBC is the power source to TRB.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	012	A2.07	
	Importance Rating	3.6		

K/A: Reactor Protection: Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of instrument power

Proposed Question:	Question #35	
Technical Reference(s):	OP-TM-AOP-017, Rev 10	
	OP-TM-MAP-A0106, Rev 0	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	AOP-017-PCO-2	

Question Source:	Bank #	1738330		
	Modified Bank #			
	New			
Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know how the loss of Vital Bus C will affect the plant and the procedure used to mitigate the malfunction. The loss of the vital bus will cause the RPS channel to de-energize, the examinee must understand the impact and the procedure entry which is required.

High Cog: Examinee has to analyze the given plant indications and then determine the correct course of action based on these conditions.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

36 **ID: 2088359** **Points: 1.00**

In the event of a reactor coolant system rupture, what is the minimum equipment which will limit the peak clad temperature to less than 2,200°F and the metal-water reaction to that representing less than 1 percent of the clad?

A. 2 Decay Heat Removal Pumps, 1 Core Flood Tank

B. 2 Decay Heat Removal Pumps, 2 Core Flood Tanks

C. 1 Makeup Pump, 1 Decay Heat Removal Pump, 1 Core Flood Tank

D. 1 Makeup Pump, 1 Decay Heat Removal Pump, 2 Core Flood Tanks

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) With one train of ES available (Makeup Pump and a Decay Heat Removal Pump) and both Core Flood tanks will protect the core. (2) Each of these components has restrictive technical specifications which will require a plant shutdown. (3) TQ-TM-104-211-C001, MAKEUP SYSTEM (Rev 11 page 71) defines the ECCS acceptance criteria to limit peak cladding temperature to less than 2,200F and the metal-water reaction to that representing less than 1 percent of the clad. (4) The FSAR Chapter 6 (Page 6.1-1 Rev 22) describes how the makeup is required for a leak and the decay heat and core flood system are used for reflood phase. (5) TQ-TM-104-213-C001 CORE FLOOD SYSTEM (Rev 5 Page 52) says that both core flood tanks are required for LOCA analyses.

A.	2 Decay Heat Removal Pumps, 1 Core Flood Tanks	INCORRECT: Plausible since one of the ECCS criteria is to ensure long term cooling, having both Decay Heat Removal Pumps would help ensure this criteria is met. Incorrect because a Makeup Pump and both Core Flood Tanks are required.
B.	2 Decay Heat Removal Pumps, 2 Core Flood Tanks	INCORRECT: Plausible since one of the ECCS criteria is to ensure long term cooling, having both Decay Heat Removal Pumps would help ensure this criteria is met. Incorrect because a Makeup Pump and both Core Flood Tanks are required..
C.	1 Makeup Pump, 1 Decay Heat Removal Pump, 1 Core Flood Tanks	INCORRECT: Plausible because the examinee could believe that Core Flood is a redundant system (i.e. only one tank is required). Incorrect because both Core Flood Tanks are required.
D.	1 Makeup Pump, 1 Decay Heat Removal Pump, 2 Core Flood Tanks	CORRECT: See above

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	013	K3.01	
	Importance Rating	4.4		

K/A: Engineered Safety Features Actuation: Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Fuel

Proposed Question:	Question #36	
Technical Reference(s):	TQ-TM-104-211-C001, Rev 11	FSAR Chapter 14, Rev 21
	TQ-TM-104-213-C001, Rev 5	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	642-GLO-8	

Question Source:	Bank #		
	Modified Bank #		

	New	X		
Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must have knowledge that even with a loss of major components of the ECCS system, the minimum required components to show fuel protection are still available.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

37 **ID: 2084746** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- AH-E-1A/B/C, Reactor Building Cooling Fans are running in fast speed

Event:

- Manual reactor trip due to a Small Break LOCA inside Containment
- Operators manually initiate 'A' and 'B' 1600 psig ES

Which of the following describes the operation of the Reactor Building Emergency Cooling Fans (AH-E-1's)?

All AH-E-1's _____.

A. remain running in fast speed

B. immediately shift to slow speed

C. trip and then restart in fast speed

D. trip and then restart in slow speed

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When the operators initiate A and B 1600 psig, this will initate block loading.(2) When ES actuates, block loading initiates. The RB Cooling Fans trip on Block 1, then restart in slow on Block 2. (TQ-TM-104-740-C001, Rev 7 Page 23).

A.	remain running in fast speed	INCORRECT: Plausible because the examinee could believe that the speed that the AH-E-1's must run in on an ES is fast. Incorrect they run in slow speed on an ES.
B.	immediately shift to slow speed	INCORRECT: The examinee could believe that the fans are able to shift speeds without stopping. Incorrect because the AH-E-1's trip on Block 1 and restart in slow on Block 2.
C.	trip and then restart in fast speed	INCORRECT: Plausible if the examinee has the misconception that the AH-E-1's must be run in fast speed on an ES. The examinee could believe that they trip on Block 1 to ensure no electrical bus is overloaded, then start on Block 2 in fast speed.
D.	trip and then restart in slow speed	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022	A3.01
	Importance Rating	4.1	

K/A: Containment Cooling: Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation

Proposed Question:	Question #37	
Technical Reference(s):	TQ-TM-104-824-C001, Rev 8	TQ-TM-104-740-C001, Rev 7

Proposed References to be provided to applicants during examination:	None

Learning Objective:	824-GLO-10	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have the knowledge to know when the containment ventilation fans start after an ES actuation. The examinee must know that an ES signal will trip the pumps in fast speed and start them in slow speed.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

38 **ID: 2087162** **Points: 1.00**

Which of the following has the ability to provide backup power automatically to Building Spray Pump, BS-P-1A?

A. 4 Bus

B. EG-Y-1A

C. EG-Y-1B

D. EG-Y-4

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The normal power supply to BS-P-1A is 1D 4160V ES Bus (1107-4, ELECTRICAL DISTRIBUTION PANEL LISTING REV 221 Page 18). (2) The automatic backup to the 1D 4160V bus is EG-Y-1A.

A.	4 Bus	INCORRECT: Plausible if the examinee believes that 1D 4160V bus is powered from the 8 bus. Incorrect because the normal power supply is the 4 bus.
B.	EG-Y-1A	CORRECT: See above.
C.	EG-Y-1B	INCORRECT: Plausible if the examinee believes the back up to the 1D 4160V Bus is EG-Y-1B. Incorrect because it is not.
D.	EG-Y-4	INCORRECT: Plausible if the examinee believes that EG-Y-4 can automatically provide backup power to the 1D 4160V bus. Incorrect because EG-Y-4 can only be manually aligned to power the 1D 4160V bus.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	026	K2.01	
	Importance Rating	3.4		

K/A: Containment Spray: Knowledge of bus power supplies to the following: Containment spray pumps

Proposed Question:	Question #38	
Technical Reference(s):	1107-4, Rev 221	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	214-GLO-4	

Question Source:	Bank #		
	Modified Bank #	719780	
	New		

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

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the bus power supplies to the containment spray pumps.

719780
Which ONE of the following power supplies has the ability to provide backup power AUTOMATICALLY to the 1B Reactor Building Spray Pump (BS-P-1B)?

A. 4 Bus

B. 1A Emergency Diesel Generator (EG-Y-1A)

C. 1B Emergency Diesel Generator (EG-Y-1B)

D. Station Blackout Diesel Generator (EG-Y-4)

Answer C

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

39 **ID: 2104742** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Reactor Trip

What OTSG pressure setpoint will the Turbine Bypass Valves be maintaining and what is the basis of that pressure setpoint?

Turbine Bypass Valves open to control Turbine Header Pressure at ___(1)___ and the basis for this setpoint is to ___(2)___.

A. (1) 960 psig
(2) prevent emergency borating on a Reactor Trip

B. (1) 960 psig
(2) limit the Pressurizer level reduction due to the cooldown

C. (1) 1010 psig
(2) prevent emergency borating on a Reactor Trip

D. (1) 1010 psig
(2) limit the Pressurizer level reduction due to the cooldown

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) On a reactor trip, the ICS selects a +125 bias to adjust the turbine header pressure (TQ-TM-104-411-C001 MAIN STEAM, Rev 8 Page 30). (2) The basis of raising pressure is to limit the RCS cooldown and shrink (SDBD-TI-411, Rev 8, Page 3-15).

A.	(1) 960 psig (2) prevent emergency borating on a Reactor Trip	INCORRECT: (1) Plausible because the examinee could believe that the OTSG pressure stays the same for a reactor trip. Incorrect because the turbine bypass valves control at a higher pressure. (2) Plausible because in some procedures, if RCS Tcold is below 525F the crew must initiate Rule 5 to Emergency Borate.
B.	(1) 960 psig (2) limit the Pressurizer level reduction due to the cooldown	INCORRECT: (1) Plausible because the examinee could believe that the OTSG pressure stays the same for a reactor trip. Incorrect because the turbine bypass valves control at a higher pressure. (1) See above.
C.	(1) 1010 psig (2) prevent emergency borating on a Reactor Trip	INCORRECT: (1) Correct Setpoint (2) Plausible because in some procedures, if RCS Tcold is below 525F the crew must initiate Rule 5 to Emergency Borate.
D.	(1) 1010 psig (2) limit the Pressurizer level reduction due to the cooldown	CORRECT: See above

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	039	A1.05	
	Importance Rating	3.2		

K/A: Main and Reheat Steam: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: RCS T-ave

Proposed Question:	Question #39	
Technical Reference(s):	TQ-TM-104-411-C001, Rev 8	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	411-GLO-5	

Question Source:	Bank #		
	Modified Bank #	1142251	
	New		

Question History:	N/A	Last NRC Exam:	14-01	Unmodified

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the Main Steam Turbine Bypass setpoint for a reactor trip and the basis. The basis is to minimize the pressurizer level shrink, which can be directly tied to RCS T-ave. The turbine bypass valves maintain OTSG pressure at 1010 psig, which will maintain RCS T-ave at approximately 555F, which will limit the outsurge of the Pressurizer. The design limit this prevents exceeding is losing Pressurizer Level indication on a reactor trip.

High Cog: This question is high cog because the examinee must analyze what will happen to the turbine bypass valve setpoint on a reactor trip.

1142251
Plant Conditions:

- Plant is at 100% 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

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

40 **ID: 2084753** **Points: 1.00**

REFEENCE PROVIDED

Plant conditions:

- Reactor power is 90% with ICS in auto

Event:

- A high pressure feedwater heater has just been removed from service

Which of the following statements describes the IMMEDIATE ICS response to these conditions when FW temperature lowers?

A. The Feedwater Demand signal will be modified to RAISE feedwater flow

B. The Feedwater Demand signal will be modified to LOWER feedwater flow

C. Steam Generator BTU limits will be reached AND the unit will be placed in TRACK

D. Feedwater to Reactor Crosslimits will be reached AND the unit will be placed in TRACK

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When the heater string is removed from service, the actual temperature of the feedwater (as seen by the ICS Feedwater Subsystem) will be lower. (2) At this point, to maintain the heat balance the Feedwater Demand Signal will be modified lower as less is needed because the temperature is colder (to maintain the proper BTU exchange between the primary and secondary). (TQ-TM-104-621-C001, Rev 10 Page 39 and 40)

A.	The Feedwater Demand signal will be modified to RAISE feedwater flow	INCORRECT: Plausible if the examinee believes that more feedwater will be required because the plant is less efficient. Incorrect because less feedwater is needed.
B.	The Feedwater Demand signal will be modified to LOWER feedwater flow	CORRECT: See above
C.	Steam Generator BTU limits will be reached AND the unit will be placed in TRACK	INCORRECT: Plausible because FW Temperature is an input to the BTU Limit Circuit. Incorrect because the FW Temperature circuit would lower actual FW demand to maintain FW temperature above BTU Limits. Also, BTU limits does not place ICS in Track.
D.	Feedwater to Reactor Crosslimits will be reached AND the unit will be placed in TRACK	INCORRECT: Plausible because the lower FW temperature would effect the Total FW demand. Total FW demand is compared to actual FW Flow to cause a Feedwater to Reactor Crosslimit. Incorrect because FW demand has to be greater than actual FW flow to cause a Crosslimit. Cross Limits do cause the initiation of Tracking.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059	K1.07
	Importance Rating	3.2	

K/A: Main Feedwater: Knowledge of the physical connections and/or cause effect relationships between the MFW and the following systems: ICS

Proposed Question:	Question #40	
Technical Reference(s):	TQ-TM-104-621-C001, Rev 10	
	D553731, Rev Q	

Proposed References to be provided to applicants during examination:	D553731, Rev Q

Learning Objective:	621-GLO-5	

Question Source:	Bank #	357042	
	Modified Bank #		
	New		

Question History:	System Exam 13	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.5
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know how ICS controls Main Feedwater.

High Cog: This question is high cog because the examinee must analyze and understand how removing a high pressure feedwater string effects the plant and ICS.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

41 **ID: 2104765** **Points: 1.00**

Plant Conditions:

- Reactor Power is 100% with ICS in full auto
- Main Feedwater line leak identified inside “A” D-Ring
- Containment Building pressure is 1.5 psig and stable

Event:

- Containment Building pressure begins to rise
- Manual reactor trip due to RB pressure rise
- Current plant conditions:
  - OTSG 1A Startup Range level is 4 inches, stable
  - OTSG 1B Startup Range level is 95 inches, lowering at 1-inch per minute
  - Containment Building pressure is stable at 4.2 psig
  - RC-P-1A and RC-P-1B have been tripped

Which of the following identifies HSPS actuation response with regards to (1) Emergency Feedwater pumps, and (2) Emergency Feedwater valves?

A. (1) EF-P-1 & EF-P-2A start, ONLY.
(2) EF-V-30A & EF-V-30C level control setpoint is 25 inches in the STARTUP Range; EF-V-30B & EF-V-30D level control setpoint remains 0% in the OPERATE Range.

B. (1) EF-P-1 & EF-P-2A start, ONLY.
(2) EF-V-30A & EF-V-30C level control setpoint is 50% in the OPERATE Range. EF-V-30B & EF-V-30D level control setpoint remains 0% in the OPERATE Range.

C. (1) EF-P-1, EF-P-2A, & EF-P-2B start.
(2) EF-V-30A/B/C/D all control at 25 inches in the STARTUP Range.

D. (1) EF-P-1, EF-P-2A, & EF-P-2B start.
(2) EF-V-30A/B/C/D level control setpoint is 50% in the OPERATE Range.

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) One of the functions of the Heat Sink Protection System (HSPS) is to start Emergency Feedwater (EFW) on the following signals: Loss of both Main Feedwater Pumps, Loss of all Reactor Coolant Pumps, Reactor Building Pressure > 4 psig, and less than 10" in the Startup Range in either OTSG. The OTSG level setpoint 25" for all actuations with the exception of loss of all the Reactor Coolant Pumps which is 50% in the Operating Range. (MAP J Rev 49 Page 2). (2) HSPS will actuate on 4 psig in the Reactor Building and and Low 'A' OTSG level and start all the EFW pumps and control level at 25" in the Startup Range.

A.	(1) EF-P-1 & EF-P-2A start, ONLY. (2) EF-V-30A & EF-V-30C level control setpoint is 25 inches in the STARTUP Range, EF-V-30B & EF-V-30D level control setpoint remains 0% in the OPERATE Range.	INCORRECT: (1) Plausible if the examinee determines that the 'A' OTSG is the only OTSG which needs feedwater. With a leak in the 'A' D-ring the 'A' OTSG level indication would be errroneously low which the examinee could believe that 'A' HSPS train components actuate. Incorrect because even though the 'A' OTSGs level is low, all of the EFW pumps start. (2) Plausible because EF-V-30A and 30C are controlled by the 'A' train of HSPS. If the examinee believed only 'A' train would actuate its components then this would be correct. Incorrect because that is not how HSPS works.
B.	(1) EF-P-1 & EF-P-2A start, ONLY. (2) EF-V-30A & EF-V-30C level control setpoint is 50% in the OPERATE Range. EF-V-30B & EF-V-30D level control setpoint remains 0% in the OPERATE Range.	INCORRECT: (1) Plausible if the examinee determines that the 'A' OTSG is the only OTSG which needs feedwater. With a leak in the 'A' D-ring the 'A' OTSG level indication would be errroneously low which the examinee could believe that 'A' HSPS train components actuate. (2) Plausible because the Reactor Coolant Pumps in the 'A' loop are tripped. The examinee could believe that the setpoint for the 'A' train components is 50% in the Operating Range. Incorrect because that is not how HSPS works.
C.	(1) EF-P-1, EF-P-2A, & EF-P-2B start. (2) EF-V-30A/B/C/D all control at 25 inches in the STARTUP Range.	CORRECT: See above.
D.	(1) EF-P-1, EF-P-2A, & EF-P-2B start. (2) EF-V-30A/B/C/D level control setpoint is 50% in the OPERATE Range.	INCORRECT: (1) Correct answer. (2) Plausible because the examinee could believe that because the Reactor Coolant Pumps are secured in one loop that the setpoint is 50% in the Operating Range.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	061	K4.02	
	Importance Rating	4.5		

K/A: Auxiliary/Emergency Feedwater: Knowledge of AFW design feature(s) and/or interlock(s) which provide for the following: AFW automatic start upon loss of MFW pump, S/G level, blackout, or safety injection				
Proposed Question:	Question #41			
Technical Reference(s):	MAP J, Rev 49			
Proposed References to be provided to applicants during examination:			None	
Learning Objective:	424-GLO-5			
Question Source:	Bank #	371302		
	Modified Bank #			
	New			
Question History:	Sim Exam 5	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the automatic starting interlocks for the EFW pumps.

High Cog: This question is high cog because the examinee has to analyze conditions in the stem and determine which EFW pumps start and the level setpoint of the EFW control valves.

42 **ID: 2084797** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- 1B Auxiliary Transformer senses high pressure on 2 rate of rise detectors on the Load Tap Changer (LTC)

Which one of the following is the correct response of the plant electrical system?

A. All the 7KV and 4KV switchgear transfer to 1A Auxiliary Transformer

B. LTC is locked out, no further changes in electrical plant lineup occur

C. 86 relays will lockout busses associated with 1B Auxiliary Transformer

D. Only the 7KV and 4KV BOP switchgear transfer to 1A Auxiliary Transformer

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The Auxiliary Transformer will trip if the Load Tap Changers (LTC) pressure relays see both of the pressure relays exceed their limit (2 out of 2 logic) (TQ-TM-104-701-C001, Rev 9 Page 20). (2) When the 1B Auxiliary Transformer trips all of the BOP switchgear fast transfers to the 1A Auxiliary Transformer (MAP AA, Rev 41 Page 13).

A.	All the 7KV and 4KV switchgear transfer to 1A Auxiliary Transformer	INCORRECT: Plausible if the examinee believes that all equipment will transfer to the 1A Auxiliary Transformer. Incorrect because the ES equipment gets powered by its Emergency Diesel Generator.
B.	LTC is locked out, no further changes in electrical plant lineup occur	INCORRECT: Plausible if examinee does not know that the pressure switches associated with load tap changer will initiate the auto transfer. Incorrect because they will trip the Auxiliary Transformer.
C.	86 relays will lockout busses associated with 1B Auxiliary Transformer	INCORRECT: Plausible since 86 lockout relays will affect breakers associated with the 4 Bus and B Aux tranformer. Incorrect because they will not lock out the busses, the busses will be powered from the 1A Aux tranformer.
D.	Only the 7KV and 4KV BOP switchgear transfer to 1A Auxiliary Transformer	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062	K4.03
	Importance Rating	2.8	

K/A: AC Electrical Distribution: Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: Interlocks between automatic bus transfer and breakers

Proposed Question:	Question #42	
Technical Reference(s):	MAP AA, Rev 41	TQ-TM-104-701-C001, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: 731-GLO-11

Question Source:	Bank #	1006758	
	Modified Bank #		
	New		

Question History:	System Exam Final	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the actions which occur on an automatic bus transfer. The examinee must know which bus breakers open and then close to maintain power.

High Cog: This question is high cog because the examinee must determine that the malfunction will have an effect on the unit electrical distribution system. In addition, the examinee must know the lineup of the electrical busses after the malfunction.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

43 **ID: 2084804** **Points: 1.00**

Which selection below describes an occurrence that would indicate the existence of a problem in the “B” 125/250VDC Distribution System?

A. Loss of indication for RC-RV-2 (PORV)

B. Loss of control power for all Reactor Coolant Pumps

C. MAP K-3-4, MN TURB DC OIL PMP STRT/TRBL actuates

D. Loss of control power of LO-P-6 (Main Turbine Emergency Bearing Oil Pump)

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Under the current condition where the "B" 123/250VDC Distribution System is lost, the crew would enter OP-TM-AOP-024, "B" DC SYSTEM FAILURE. Attachment 7.1 (Rev 7 Page 35) describes the effects of a loss of "B" DC, one of which being loss of control power to all RCP breakers. This would be apparent in the control room by the breaker indications lights above each Reactor Coolant Pump being off.

A.	Loss of indication for RC-RV-2 (PORV)	INCORRECT: Plausible because the DC power to RC-RV-2 is downstream of the 'A' DC Distribution panel.
B.	Loss of control power for all Reactor Coolant Pumps	CORRECT: See above
C.	MAP K-3-4, MN TURB DC OIL PMP STRT/TRBL, actuates	INCORRECT: Plausible because this alarm comes in from a loss of DC. Incorrect because the loss of "A" DC brings this alarm in.
D.	Loss of control power of LO-P-6 (Main Turbine Emergency Bearing Oil Pump)	INCORRECT: Plausible because the power for LO-P-6 is 1C DC which is powered from the 'A' DC distribution center.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	063	A3.01	
	Importance Rating	2.7		

K/A: DC Electrical Distribution: Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights

Proposed Question:	Question #43	
Technical Reference(s):	OP-TM-AOP-0241, Rev 8	OP-TM-AOP-024, Rev 7

Proposed References to be provided to applicants during examination:	None

Learning Objective:	AOP-024-PCO-4	

Question Source:	Bank #	1738593	
	Modified Bank #		
	New		

Question History:	Comp 2	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know how the control power indicating lights on the Reactor Coolant Pumps could be used to monitor that the 'B' DC system has had a malfunction. If 'B' DC is lost, the control power indicating lights turn off.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

44 **ID: 2094874** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

EVENT:

- The relief valve for EG-Y-1A Air Reciever (EG-T-1A-1) fails open
- B-3-1, STARTING AIR PRESSURE LOW ALARM, on the Diesel Generator Alarm panel comes in
- EG-Y-1A Starting Air Pressure Indicator (EG-PI-535A), reads 35 psig

Which of the following identifies how EG-Y-1A will respond to a start signal?

EG-Y-1A will ___(1)___because ___(2)___.

A. (1) start
(2) EG-T-1A-2 will supply starting air

B. (1) start
(2) EG-P-1A, Motor Driven Compressor, will supply starting air

C. (1) NOT start
(2) both Air Receivers have blown down

D. (1) NOT start
(2) EG-V-16A and EG-V-16C Air Start valves will not open due to loss of air

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with drawing 302351 (Rev 28) the air start system for a diesel generator consists of 2 air recievers and an air compressor. (2) The tanks are protected from overpressure by relief valves. (3) The system is cross connected in such a fashion if one relief valve were to fail, that both the tanks will depressurize and the air compressor would discharge out of the failed relief valve. (4) EG-PI-535A reads the pressure at the common outlet of the tanks and compressor. The stem indicates that pressure is 35 psig, which is not sufficient enough to start the diesel generator in accordance with TQ-TM-104-861-C001, EMERGENCY DIESEL GENERATORS AND AUXILIARIES (Rev 12 Page 9)

A.	(1) start (2) EG-T-1A-2 will supply starting air	INCORRECT: (1) Plausible if the examinee believes that 35 psig will start the diesel, or that the other tank and/or air compressor have the required air to start the diesel. Incorrect because they are all mechanically connected. (2) Plausible if examinee does not understand the mechanical connection between the two tanks. Incorrect because there is not a check valve isolation to prevent blowing down both of the air tanks.
B.	(1) start (2) EG-P-1A, Motor Driven Compressor, will supply starting air	INCORRECT: (1) Plausible if the examinee believes that 35 psig will start the diesel, or that the other tank and/or air compressor have the required air to start the diesel. Incorrect because they are all mechanically connected. (2) Plausible if examinee does not understand the mechanical connection between the two tanks. Incorrect because there is not a check valve isolation to prevent blowing down both of the air tanks.
C.	(1) not start (2) both Air Receivers have blown down	CORRECT: See above
D.	(1) not start (2) EG-V-16A and EG-V-16C Air Start valves would not open due to loss of air	INCORRECT: (1) Correct answer. (2) Plausible if examinee thinks that Air Start Valves would fail closed when the pressure in the Receiver tanks has been depleted. Incorrect because the air system is used to start the diesel, not to supply air to valves which are used to start the diesel.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	064	K6.07	
	Importance Rating	2.7		

K/A: Emergency Diesel Generator: Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air Receivers

Proposed Question:	Question #44	
Technical Reference(s):	302351, Rev 28	
	TQ-TM-104-861-C001, Rev 12	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	861-GLO-5			
Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of what a failure of the air recievers would do to the diesel generators.

High Cog: This question is high cog because the examinee must relate the failure and the air pressure to whether the diesel generator will start.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

45 **ID: 2084805** **Points: 1.00**

Plant conditions:

- Reactor power is 100% with ICS in full auto
- Miscellaneous Waste Evaporator is in service

Event:

- RM-A-5 and RM-A-15, Condenser Vacuum Pump Exhaust Radiation Monitors, show rising trends
- MAP C-1-1, RADIATION LEVEL HIGH comes in due to RM-A-5 and RM-A-15

Which of the following actions, if performed within 15 minutes of the event, would initially differentiate between a fuel failure and OTSG tube leak in accordance with OP-TM-MAP-C0101?

A. Survey the letdown line to determine if a significant change in RCS activity has occurred.

B. Request Chemistry to obtain a condensate return sample of the evaporator and analyze for boron.

C. Check Letdown Radiation Monitor, RM-L-1, readings to determine if there is a sudden upward trend.

D. Check Letdown Radiation Monitor, RM-L-1, LO readings to determine if there is a sudden upward trend.

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) A note in OP-TM-MAP-C0101 (Rev 4, Page 8) notifies the crew that a change in RCS activity many not be observed on RM-L-1 (Letdown Line Radiation Monitor) until 30 to 60 minutes later due to transport time. (2) The procedure directs the crew to request Rad Con to perform a survey of the letdown line to determine if a significant change in RCS activity. (2) Due to the minute amount of OTSG leakage that may exist, an RCS leak may show up in RM-A-5/15 before RM-L-1.

A.	Survey the letdown line to determine if a significant change in RCS activity has occurred.	CORRECT: See above.
B.	Request Chemistry to obtain a condensate return sample of the evaporator and analyze for boron.	INCORRECT: Plausible because this is a step in OP-TM-MAP-C0101. Incorrect because Rad Con would be sampling for activity, not boron.
C.	Check Letdown Radiation Monitor, RM-L-1, readings to determine if there is a sudden upward trend.	INCORRECT: Plausible because this is a Letdown Monitor. Incorrect because this monitor would take 30 to 60 minutes for a change in reading to be observed.
D.	Check Letdown Radiation Monitor, RM-L-1, LO readings to determine if there is a sudden upward trend.	INCORRECT: Plausible because this is a Letdown Monitor. Incorrect because this monitor would take 30 to 60 minutes for a change in reading to be observed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073	2.1.7
	Importance Rating	4.4	

K/A: Process Radiation Monitoring: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question:	Question #45	
Technical Reference(s):	OP-TM-MAP-C0101, Rev 4	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	661-GLO-12	

Question Source:	Bank #	858179	
	Modified Bank #		
	New		

Question History:	Simulator Exam 9	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.11	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the operational implications that an elevated reading on the letdown line would have if RM-A-5/15 were to alarm. This information would differentiate between taking actions to shutdown via MAP-C0101 or OP-TM-EOP-005, OTSG TUBE LEAKAGE.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

46 **ID: 2106163** **Points: 1.00**

REFERENCE PROVIDED

In accordance with OP-TM-541-234, IST OF NR-P-1A AND NSRW VALVES DURING SINGLE PUMP OPERATIONS which areas of Attachment 7.2 correspond to the following limitations?

Area 1 is to ___(1)___.
Area 2 is to ___(2)___.
Area 3 is to ___(3)___.

A. 1) avoid pump wear or damage
2) avoid NR strainer clogging
3) prevent pump runout

B. 1) avoid NR strainer clogging
2) avoid pump wear or damage
3) prevent pump runout

C. 1) avoid pump wear or damage
2) prevent pump runout
3) avoid NR strainer clogging

D. 1) prevent pump runout
2) avoid pump wear or damage
3) avoid NR strainer clogging

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The limits and precautions of OP-TM-541-234 (Rev 7 Page 2) require Nuclear River Water Pumps be run in the operating band of OP-TM-541-000, PRIMARY COMPONENT COOLING (Rev 24A Page 37) Attachment 7.2. The examinee must know the limitations apply to the following areas in the provided reference:
Area 1 - To prevent excessive pump wear or damage, do not operate NR pumps for extended periods (> 4HRs) with NR-PI-217 pressure in the restricted region on Attachment 7.2.
Area 2 - To avoid clogging of the NR strainers (i.e., keep strainer pressure >20 psig), do not operate for extended periods (> 4HRs) with NR-PI-217 pressure in the restricted region on Attachment 7.2.
Area 3 - To prevent NR pump run-out if a NR pump trips when two NR pumps were operating, then maintain NSRW pressure above “two pump operation limit” on Attachment 7.2 on PI-217 (CC).

A.	1) avoid pump wear or damage 2) avoid NR strainer clogging 3) prevent pump runout	CORRECT: See above.
B.	1) avoid NR strainer clogging 2) avoid pump wear or damage 3) prevent pump runout	INCORRECT: Plausible if the examinee does not remember or cannot determine the right areas.
C.	1) avoid pump wear or damage 2) prevent pump runout 3) avoid NR strainer clogging	INCORRECT: Plausible if the examinee does not remember or cannot determine the right areas.
D.	1) prevent pump runout 2) avoid pump wear or damage 3) avoid NR strainer clogging	INCORRECT: Plausible if the examinee does not remember or cannot determine the right areas.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	076	2.2.12	
	Importance Rating	3.7		

K/A: Service Water System: Knowledge of surveillance procedures

Proposed Question:	Question #46	
Technical Reference(s):	OP-TM-541-000, Rev 24	OP-TM-541-234, Rev 7
Proposed References to be provided to applicants during examination:		Edited Attachment 7.2 of OP-TM-541-000

Learning Objective:	531-GLO-10			
Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of a surveillance procedure of a service water system.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

47 **ID: 2104876** **Points: 1.00**

Sequence of Events:

- A rupture occurs in the 1 1/2 to 3/4 inch reducer on the air header supplying MS-V-3D.
- IA-P-4 trips on overload.
- IA-P-1A/B and SA-P-1A/B are started successfully.
- Instrument air primary (PI-222) and secondary (PI-1403) pressure indicators on PL are tracking together and continue to lower.

Which of the following describes plant response?

IA-V-26 (Secondary Plant IA Supply Valve) will close at _____.

A. 60 psig, then PI-222 starts to RISE and PI-1403 continues to LOWER

B. 60 psig then PI-1403 starts to RISE and PI-222 continues to LOWER

C. 80 psig then PI-222 starts to RISE and PI-1403 continues to LOWER

D. 80 psig then PI-1403 starts to RISE and PI-222 continues to LOWER

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) MS-V-3D is in the Turbine Building, which will be isolated when IA-V-26 closes at 60 psig header pressure. (2) When IA-V-26 is closed, that will isolate the secondary instrument air from the primary instrument air. PI-1402 will continue to lower and PI-222 will being to rise.

A.	60 psig then PI-222 starts to RISE and PI-1403 continues to LOWER.	CORRECT: See above.
B.	60 psig then PI-1403 starts to RISE and PI-222 continues to LOWER	INCORRECT: Plausible if location of air leak is misunderstood. Incorrect instrument header responses would be opposite.
C.	80 psig then PI-222 starts to RISE and PI-1403 continues to LOWER	INCORRECT: Plausible because 80 psig is a common instrument air setpoint. Incorrect because it is not the correct setpoint for IA-V-26.
D.	80 psig then PI-1403 starts to RISE and PI-222 continues to LOWER.	INCORRECT: Plausible because 80 psig is a common instrument air setpoint. Incorrect because it is not the correct setpoint for IA-V-26.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	078	A4.01	
	Importance Rating	3.1		

K/A: Instrument Air: Ability to manually operate and/or monitor in the control room: Pressure gauges

Proposed Question:	Question #47	
Technical Reference(s):	302271, Rev 73	302268, Rev 18

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-028-PCO-5

Question Source:	Bank #	353834	
	Modified Bank #		
	New		

Question History:	Simulator Exam 8	Last NRC Exam:		

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know how the pressure gauges in the control room respond during a loss of instrument air.

High Cog: This question is high cog because the examinee must know the setpoint for IA-V-26 and analyze the plant response for the location of the leak.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

48 **ID: 2104885** **Points: 1.00**

Plant Conditions:

- A shutdown is in progess due to RCS leakage

Event:

- The Reactor trips on low pressure
- Rule 1, LOSS OF SUBCOOLING MARGIN is performed
- RB pressure is 31 psig and rising
- No ES signals have been defeated or bypassed

Which one of the following valves must be verified closed and what procedure will give the actions to take if the valve fails to close?

A. (1) MU-V-20, SEAL INJECTION ISOLATION VALVE
(2) OP-TM-642-903, 30 PSIG ESAS ACTUATION

B. (1) MU-V-25, RCP SEAL RETURN ISOLATION VALVE
(2) OP-TM-244-901, CONTAINMENT ISOLATION

C. (1) MU-V-20, SEAL INJECTION ISOLATION VALVE
(2) OP-TM-244-901, CONTAINMENT ISOLATION

D. (1) MU-V-25, RCP SEAL RETURN ISOLATION VALVE
(2) OP-TM-642-903, 30 PSIG ESAS ACTUATION

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When equipment operates properly, there are separate containment isolation valves that close on a 4# ES and 30# ES signal. (2) In addition, the operator must know that the 30# ES signal is not a redundant signal to the 4# ES, specifically that the 30# ES is the only signal that closes NS-V-4/15/35, IC-V-2/3/4/6, MU-V-25/26. The 30# ES will close none of the valves that the 4# ES closes and vice versa. (3) The step of the question sets up a situation where the examinee must identify that a 4# ES signal has occured and is still in because the 30# signal needs a 4# permissive. (4) The mitigation procedure for this failure is OP-TM-244-901, CONTAINMENT ISOLATION.

A.	(1) MU-V-20, SEAL INJECTION ISOLATION VALVE (2) OP-TM-642-903, 30 PSIG ESAS ACTUATION	INCORRECT: (1) Plausible because this is the Seal Injection valve. The examinee could have a misconception that this valve closes on ES. (2) Plausible due to the title of the procedure, and this procedure will initiate OP-TM-244-901, but has no actions to close the appropriate valve.
B.	(1) MU-V-25, RCP SEAL RETURN ISOLATION VALVE (2) OP-TM-244-901, CONTAINMENT ISOLATION	CORRECT: See above
C.	(1) MU-V-20, SEAL INJECTION ISOLATION VALVE (2) OP-TM-244-901, CONTAINMENT ISOLATION	INCORRECT: (1) Plausible because this is the Seal Injection valve. The examinee could have a misconception that this valve closes on ES. (2) Correct answer.
D.	(1) MU-V-25, RCP SEAL RETURN ISOLATION VALVE (2) OP-TM-642-903, 30 PSIG ESAS ACTUATION	INCORRECT: (1) Correct answer. (2) Plausible due to the title of the procedure, and this procedure will initiate OP-TM-244-901, but has no actions to close the appropriate valve.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	103	A2.03	
	Importance Rating	3.5		

K/A: Containment: Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions: Phase A and B isolation

Proposed Question:	Question #48	
Technical Reference(s):	OP-TM-244-901, Rev 4	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	EOP006-PCO-4	

Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This quesiton matches the KA because the examinee must know which valve must be closed on a 30# ES actuatation. The examinee must know that the 4# signal must still give a permissive for this valve to close (Phase A and B isolation). The examinee must know there is an contingency action in OP-TM-244-901.

High Cog: This question is high cog because the examinee must know that to completely isolate the Reactor Building that both the 4# and 30# signals are needed.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

49 **ID: 2084744** **Points: 1.00**

Plant Conditions:

- Reactor Coolant (RCS) pressure 2166 psig
- Reactor Coolant Drain Tank (RCDT) pressure is 10 psig
- The PORV is stuck partially open

Given the above information, what is the approximate temperature and phase of the fluid downstream of the PORV?

A. 193°F

B. 240°F

C. 300°F

D. 648°F

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When fluid leaks past the PORV that it maintains a constant enthalpy. (2) Due to this process and the pressure drop from the Pressurizer to the Reactor Coolant Drain Tank (RCDT) the temperature will lower to the saturation temperature of the pressure in the RCDT.

A.	193°F	INCORRECT: Plausible if the candidate does not convert to psia. This is temperature at 10 psia. Incorrect because the examinee must find the temperature at 25 psia.
B.	240°F	CORRECT: See above.
C.	300°F	INCORRECT: Plausible if the examinee believes that the RCDT pressure will rise to the rupture disk setpoint and maintain pressure there. The rupture disk relieves at 55 psig. Incorrect because there is no indication the the pressure rise was that high.
D.	648°F	INCORRECT: Plausible if the examinee believes this is a constant temperature process.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	010	K5.02	
	Importance Rating	2.6		

K/A: Pressurizer Pressure Control: Knowledge of the operational implications of the following concepts as they apply to the PZR PCS: Constant enthalpy expansion through a valve

Proposed Question:	Question #49	
Technical Reference(s):	Steam Tables / Mollier Diagram	

Proposed References to be provided to applicants during examination:	Steam Tables

Learning Objective:	220-PCO-5		

Question Source:	Bank #	909285	
	Modified Bank #		
	New		

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

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

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know that leakage past the PORV is a constant enthalpy expansion process.

High Cog: This question is high cog because the examinee must use the mollier diagram to get the correct temperature and quality of steam.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

50 **ID: 2087160** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- 1E 125/250V DC ES Distribution Panel is de-energized to support ground-busting

Which one of the following describes the impact on CRD breaker operation if an automatic RPS reactor trip signal is generated during these conditions?

A. Breakers CRD-CB-A and CRD-CB-C will NOT open
Breakers CRD-CB-B and CRD-CB-D will open

B. Breakers CRD-CB-A and CRD-CB-D will NOT open
Breakers CRD-CB-B and CRD-CB-C will open

C. Shunt Trips will NOT operate for Breakers CRD-CB-A and CRD-CB-D
All CRD breakers will open

D. Shunt Trips will NOT operate for Breakers CRD-CB-A and CRD-CB-C
All CRD breakers will open

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When an RPS generates two trip signals, all of the CRD breakers (CRD-CB-A/B/C/D) should open. (2) To ensure they open the breakers are designed with a shunt trip coil and an UV relay. (3) The shunt trips must be powered from 125 VDC to open the breaker. The shunt signal is generated from RPS. (4) CRD-CB-A and CRD-CB-C are powered from 1E 125 VDC. CRD-CB-B and CRD-CB-D are powered from 1F 125VDC (TQ-TM-014-622-C001, Rev 7 Page 29 and 30). (4) All breakers will open from the UV relays.

A.	Breakers CRD-CB-A and CRD-CB-C will NOT open; Breakers CRD-CB-B and CRD-CB-D will open	INCORRECT: Plausible because the shunt trip from CRD-CB-A and CRD-CB-B are not operable. The examinee could believe they will not open. Incorrect because the UV coils will open the breakers.
B.	Breakers CRD-CB-A and CRD-CB-D will NOT open; Breakers CRD-CB-B and CRD-CB-C will open	INCORRECT: Plausible if the examinee does not know which DC supplies power to each CRD breaker. Incorrect because all CRD breakers open.
C.	Shunt Trips will NOT operate for Breakers CRD-CB-A and CRD-CB-D; All CRD breakers will open	INCORRECT: Plausible if the examinee does not know which DC supplies power to each CRD breaker. Incorrect because the CRD-CB-A and CRD-CB-C shunt trips will not operate.
D.	Shunt Trips will NOT operate for Breakers CRD-CB-A and CRD-CB-C; All CRD breakers will open	CORRECT: See above

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	012	K1.02	
	Importance Rating	3.4		

K/A: Reactor Protection: Knowledge of the physical connections/or cause effect relationships between the RPS and the following systems: 125V dc system

Proposed Question:	Question #50	
Technical Reference(s):	OP-TM-AOP-024, Rev 7	OP-TM-AOP-0241, Rev 8
	TQ-TM-104-622-C001, Rev 7	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	641-GLO-4	

Question Source:	Bank #	371279	
	Modified Bank #		
	New		

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

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the physical connection between the RPS and 125V DC system.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

51 **ID: 2087140** **Points: 1.00**

Event:

- A Plant Cooldown in progress in preparation for Refueling
- MAP D-3-8, CF-V-1A/B POSITION ABNORMAL alarm is received
- CF-V-1A/1B breakers are closed
- RCS pressure is 645 psig and lowering slowly

What (1) Control Room position indicating light(s) is/are available to verify this alarm and, (2) which action(s) must be taken IAW MAP D-3-8?

A. (1) ES Status Panel indication, ONLY
(2) Stabilize RCS Pressure at 615 psig and close CF-V-1A and CF-V-1B

B. (1) Control Console indication, ONLY
(2) Stabilize RCS pressure or close CF-V-1A and CF-V-1B

C. (1) BOTH ES Status Panel AND Control Console indication
(2) Stabilize RCS pressure or close CF-V-1A and CF-V-1B

D. (1) BOTH ES Status Panel AND Control Console indication
(2) Stabilize RCS Pressure at 615 psig and close CF-V-1A and CF-V-1B

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1102-11, PLANT COOLDOWN (Rev 156 Page 22) the CF-V-1 (Core Flood Tank Outlet Valves) may be closed when the reactor is shutdown and RCS pressure is between 2200 - 750 psig. (2) In this case the MAP alarm D0308 will come in when RCS pressure is less than 650 psig and CF-V-1A and/or CF-V-1B are open. (3) OP-TM-MAP-D0308 (Rev 3, Page 1) directs RCS pressure to be stabilized or close CF-V-1A and CF-V-1B. (4) CF-F-1A and CF-V-1B have indicating lights on the ES status panel and Console Center.

A.	(1) ES Status Panel indication, ONLY (2) Stabilize RCS Pressure at 615 psig and close CF-V-1A and CF-V-1B	INCORRECT: (1) Plausible because the indicating lights for CF-V-1A/B on Console Center are off due to the breakers being open at power. Incorrect because at this point in the cooldown the breakers would have been closed and indicating light power available. (2) Plausible because the nominal pressure for the Core Flood tanks to empty into the RCS is 600 psig. 615 psig is above 600 psig and the minimum pressure per Tech Specs. Incorrect: This action is NOT per the Alarm Response procedure.
B.	(1) Control Console indication, ONLY (2) Stabilize RCS pressure or close CF-V-1A and CF-V-1B	INCORRECT: (1) Plausible if examinee does not believe these valves are on the ES Status Panel since they are not affected by any ES Actuation Signal. (2) Correct answer.
C.	(1) BOTH ES Status Panel AND Control Console indication (2) Stabilize RCS pressure or close CF-V-1A and CF-V-1B	CORRECT: See above.
D.	(1) BOTH ES Status Panel AND Control Console indication (2) Stabilize RCS Pressure at 615 psig and close CF-V-1A and CF-V-1B	INCORRECT: (1) Correct answer. (2) Plausible because the nominal pressure for the Core Flood tanks to empty into the RCS is 600 psig. 615 psig is above 600 psig and the minimum pressure per Tech Specs. Incorrect: This action is NOT per the Alarm Response procedure.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	006	A4.02	
	Importance Rating	4.0		

K/A: Emergency Core Cooling: Ability to manually operate and/or monitor in the control room: Valves

Proposed Question:	Question #51	
Technical Reference(s):	OP-TM-MAP-D0308, Rev 3	TQ-TM-104-213-C001, Rev 5

Proposed References to be provided to applicants during examination: None

Learning Objective:	213-GLO-10			
Question Source:	Bank #	371767		
	Modified Bank #			
	New			
Question History:	Simulator Exam 6	Last NRC Exam:	N/A	

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

Comments:

KA Match: This question matches the KA because the examinee must know the indication and manual operation of valves in an ES system (Core Flood) during a plant cooldown.

High Cog: This question is high cog because the examinee must know the reason the alarm D-3-8 came in and analyze the plant conditions to determine correct position of the valves.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

52 **ID: 2107217** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- A Small Break LOCA causes a Reactor Trip on Low Pressure
- OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER is entered due to a Main Feedwater upset that occurred after the Reactor Trip.
- EFW is manually initiated in accordance with OP-TM-424-901, EMERGENCY FEEDWATER

After the Foxboro controllers are back in automatic, which of the choices below correctly completes the following statements?

The EF-V-30 setpoint lever is placed in ___(1)___ and the thumbwheel is set to 25.
The cooldown rate limit is ___(2)___.

A. (1) L (Local)
(2) 100F/hr

B. (1) R (Remote)
(2) 100F/hr

C. (1) L (Local)
(2) 50F/hr

D. (1) R (Remote)
(2) 50F/hr

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Enclosure 4 of 1105-19, HEAT SINK PROTECTION SYSTEM (Rev 30 Page 45) describes the use of the EF-V-30 controllers. The R/L switch selects either remote (auto setpoint) or local (manual setpoint). The thumbwheel adjusts the local setpoint. (2) The question does not explicitly imply that any automatic EFW actuation setpoint was exceeded so using the controller is the only way to put the EF-V-30's in automatic. (3) In accordance with OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES AND GRAPHS, Guide 11, COOLDOWN RATE LIMITS (Rev 20 Page 24) the cooldown rate is 100F/hr.

A.	(1) L (local) (2) 100F/hr	CORRECT: (1) See above
B.	(1) R (Remote) (2) 100F/hr	INCORRECT: (1) Plausible because that is the other option on the switch. Incorrect because the Remote setpoint would be at 0 since no EFW actuation setpoint was exceeded. (2) Correct answer.
C.	(1) L (Local) (2) 50F/hr	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes that OP-TM-EOP-004 directs securing all Reactor Coolant Pumps. Incorrect because it only directs securing one in each loop.
D.	(1) R (Remote) (2) 50F/hr	INCORRECT: (1) Plausible because that is the other option on the switch. Incorrect because the Remote setpoint would be at 0 since no EFW actuation setpoint was exceeded. (2) Plausible if the examinee believes that OP-TM-EOP-004 directs securing all Reactor Coolant Pumps. Incorrect because it only directs securing one in each loop.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	061	A3.02	
	Importance Rating	4.0		

K/A: Auxiliary/Emergency Feedwater: Ability to monitor automatic operation of the AFW, including: RCS cooldown during AFW operations.

Proposed Question:	Question #52	
Technical Reference(s):	1105-19, Rev 30	OP-TM-EOP-010, Rev 20

Proposed References to be provided to applicants during examination:	None

Learning Objective:	644-GLO-5	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know how to place EFW control in automatic control and the cooldown limits based on plant conditions.

High Cog: This question is high cog because the examinee must know how the EFW controller works and how to correctly set the lever. Also, the examinee must analyze plant conditions to determine the cooldown rate limit.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

53 **ID: 2084785** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- A malfunction caused CO-V-9A (FW-P-1A SUCTION VALVE) to close

Which of the choices completes the statement regarding which feedwater pump(s) will trip and the initiating trip signal?

___(1)___ will trip, due to ___(2)

A. (1) ONLY FW-P-1A
(2) CO-V-9A position

B. (1) ONLY FW-P-1A
(2) FW pump suction pressure

C. (1) BOTH FW-P-1A and FW-P-1B
(2) CO-V-9A position

D. (1) BOTH FW-P-1A and FW-P-1B
(2) FW pump suction pressure

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The closure of CO-V-9A is an automatic trip for FW-P-1A (OP-TM-MAP-M0101, FWP 1A TRIP, Rev 2, Page 1)

A.	(1) ONLY FW-P-1A (2) CO-V-9A position	CORRECT: See above.
B.	(1) ONLY FW-P-1A (2) FW pump suction pressure	INCORRECT: Plausible if examinee recognizes CO-V-9 closing would cause low suction pressure. Incorrect because there is no low suction pressure trip for a Main Feedwater Pump (MFP)
C.	(1) BOTH FW-P-1A and FW-P-1B (2) CO-V-9A position	INCORRECT: Plausible because there is a common suction header for both MFPs and closing one CO-V-9 would effect both MFPs. Incorrect because CO-V-9A/B are after the common suction header.
D.	(1) BOTH FW-P-1A and FW-P-1B (2) FW pump suction pressure	INCORRECT: Plausible because there is a common suction header for both MFPs and closing one CO-V-9 would effect both MFPs. Incorrect because CO-V-9A/B are after the common suction header.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	059	A3.03	
	Importance Rating	2.5		

K/A: Main Feedwater: Ability to monitor automatic operation of MFW, including: Feedwater pump suction flow pressure

Proposed Question:	Question #53	
Technical Reference(s):	OP-TM-MAP-M0101, Rev 2	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	401-GLO-11	

Question Source:	Bank #	371788	
	Modified Bank #		
	New		

Question History:	System Exam 6	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know that a Main Feedwater Pump will trip if the suction valve closes. No indication exists on the control room panel solely dedicated to feedwater pump suction pressure, Condensate Booster Pump discharge pressure and the indication that the suction valve, CO-V-9A/B is open show that the Main Feedwater pumps have proper suction pressure. The examinee must know that a low feedwater suction pressure does not trip the Main Feedwater Pumps, but a closed suction valve will.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

54 **ID: 2087972** **Points: 1.00**

Plant Conditions:

- Reactor power is 50% with ICS in auto

Event:

- RC-P-1A trips

Which oil pump(s) automatically start and where are the pump indications and control switches located?

RC-P-2A = HP Lift Pump
RC-P-3A = Backstop Oil Pump

A. (1) ONLY RC-P-2A
(2) Console Right

B. (1) ONLY RC-P-2A
(2) Console Center

C. (1) BOTH RC-P-2A AND RC-P-3A
(2) Console Right

D. (1) BOTH RC-P-2A AND RC-P-3A
(2) Console Center

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-TM-MAP-F0101, RCP MOTOR TRIP (Rev 0 Page 1) when a Reactor Coolant Pump trips then the operators ensure one RC-P-2 is in service and start at least one RC-P-3. (2) All of the controls are on console center.

A.	(1) ONLY RC-P-2A (2) Console Right	INCORRECT: (1) Correct answer. (2) Plausible because the controls are on the right side of console center, close to Console Right.
B.	(1) ONLY RC-P-2A (2) Console Center	CORRECT: See above
C.	(1) BOTH RC-P-2A AND RC-P-3A (2) Console Right	INCORRECT: (1) Plausible because both pumps must be started in accordance with the alarm response. Incorrect because the RC-P-2 is the only one that auto starts. (2) Plausible because the controls are on the right side of console center, close to Console Right.
D.	(1) BOTH RC-P-2A AND RC-P-3A (2) Console Center	INCORRECT: (1) Plausible because both pumps must be started in accordance with the alarm response. Incorrect because the RC-P-2 is the only one that auto starts. 2. Correct answer.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003	2.1.31
	Importance Rating	4.6	

K/A: Reactor Coolant Pump: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup

Proposed Question:	Question #54	
Technical Reference(s):	OP-TM-MAP-F0101, Rev 0	

Proposed References to be provided to applicants during examination: None

Learning Objective: 226-GLO-10

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the location of the control room switches and the correct the oil pumps running for a Reactor Coolant Pump trip.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

55 **ID: 2105725** **Points: 1.00**

Plant conditions:

- The plant is at 14% power
- The Main Generator is being synchronized to the grid, IAW OP-TM-301-102, MAIN TURBINE GENERATOR STANDBY TO OPERATING MODE

In accordance with OP-TM-301-102, generator kilovolts must be ___(1)___ than system kilovolts, and the Synchroscope must be rotating in the ___(2)___ direction.

A. (1) greater
(2) counterclockwise

B. (1) greater
(2) clockwise

C. (1) less
(2) counterclockwise

D. (1) less
(2) clockwise

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-TM-301-102, MAIN TURBINE GENERATOR STANDBY TO OPERATING MODE (Rev 26 Page 32) in Step 4.19.9 the operators raise generator kilovolts 4KV higher than the system kilovolt. (2) In step 4.19.10 the operators adjust Turbine speed until the synchroscope is rotating in the fast direction at no more than 1 revolution every 3 seconds.

A.	(1) greater (2) counterclockwise	INCORRECT: (1) Correct answer. (2) Plausible of the examinee believes the synchroscope should be going in the slow (opposite) direction. Incorrect because the synchroscope should be going in the fast direction.
B.	(1) greater (2) clockwise	CORRECT: See above.
C.	(1) less (2) counterclockwise	INCORRECT: (1) Plausible if the examinee believes that the generator voltage should be system voltage. Incorrect because it must be greater to prevent motoring the generator. (2) Plausible of the examinee believes the synchroscope should be going in the slow (opposite) direction. Incorrect because the synchroscope should be going in the fast direction.
D.	(1) less (2) clockwise	INCORRECT: (1) Plausible if the examinee believes that the generator voltage should be system voltage. Incorrect because it must be greater to prevent motoring the generator. (2) Correct answer.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	062	A4.03	
	Importance Rating	2.8		

K/A: AC Electrical Distribution: Ability to manually operate and/or monitor in the control room: Synchroscope, including an understanding of running and incoming voltages.

Proposed Question:	Question #55	
Technical Reference(s):	OP-TM-301-102, Rev 26	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	711-GLO-6	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

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

Comments:

KA Match: This question matches the KA because the examinee must have the ability to monitor the correct parameters in the control room to synchronize the main generator to the grid.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

56 **ID: 2105895** **Points: 1.00**

Plant conditions:

- Reactor power is 100% with ICS in full auto

Event:

- MAP alarm A-1-6, INVERTER FAILED, illuminates
- MAP alarm A-3-8, INVERTER 1B/1D/1F TROUBLE, illuminates
- 4 psig RB Pressure Channel RB3B indicates actuated
- 500 psig ESAS Channel RC6B indicates actuated
- 1600 psig ESAS Channel RC3B indicates actuated
- 30 psig RB Pressure ESAS Channel RB6B indicates actuated
- The right side PIP screen is off

The plant has experienced a loss of Vital Bus ___(1)___, and to find controlling group rod position indication the operators can use ___(2)___.

A. (1) B
(2) PPC only

B. (1) B
(2) PPC and the left-side PIP panel

C. (1) D
(2) PPC only

D. (1) D
(2) PPC and the left-side PIP panel

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must recognize: (1) With the given indications that Vital Bus 'D' was lost and that OP-TM-AOP-018, LOSS OF VBD (Rev 9) was entered. Indications of loss of Vital Bus 'D' are in Attachment 1 on Page 13. (2) Step 3.8 allows the operators to select either groups 1-4 or 5-7 on CRD-FPM-A (left side PI Panel). (3) Rod position indication can still be found on the PPC (TQ-TM-104-622-C001, Rev 7 Page 34) when Vital Bus 'D' was lost.

A.	(1) B (2) PPC only	INCORRECT: (1) Plausible if the examinee identifies the loss of power as Vital Bus 'B'. Loss of Vital Bus 'B' would give similiar alarms and indications. Incorrect because the power loss is Vital Bus 'D'. (2) Plausible if the examinee believes that the only remaining indication for the controlling group rods is the PPC. Initially the left PI panel will show only group 1-4 rods. The left PI panel will not show the controlling group of rods (groups 5-7) until the operators reposition the switch by the PI panel. Incorrect because when the operators reposition the switch the controlling group of rods will be shown on the PI panel.
B.	(1) B (2) PPC and the left-side PIP panel	INCORRECT: (1) Plausible if the examinee identifies the loss of power as Vital Bus 'B'. Loss of Vital Bus 'B' would give similiar alarms and indications. Incorrect because the power loss is Vital Bus 'D'. (2) Correct answer.
C.	(1) D (2) PPC only	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes that the only remaining indication for the controlling group rods is the PPC. Initially the left PI panel will show only group 1-4 rods. The left PI panel will not show the controlling group of rods (groups 5-7) until the operators reposition the switch by the PI panel. Incorrect because when the operators reposition the switch the controlling group of rods will be shown on the PI panel.
D.	(1) D (2) PPC and the left-side PIP panel	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	014	K3.02	
	Importance Rating	2.5		

K/A: Rod Position Indication: Knowledge of the effect that a loss or malfunction of the RPIS will have on the following: Plant Computer

Proposed Question:	Question #56	
Technical Reference(s):	OP-TM-AOP-018, Rev 9	TQ-TM-104-622-C001, Rev 7
	OP-TM-AOP-0181, Rev 5	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	AOP-018-PCO-1	

Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the effect that a malfunction of the P2 processor (which calculates rod position for the PIP) has on the PPC. The examinee must know rod position can still be found on the plant computer.

High Cog: This question is high cog because the examinee must analyze the stem and determine the cause of the indications. In addition, the examinee must determine where Rod Position Indication is available.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

57 **ID: 2084852** **Points: 1.00**

Sequence of Events:

- At T=0, the reactor trips from 100% power. All rods fully insert.
- At T=7 minutes:
  - NI-3 indicates 8 x 10^{-10} amps
  - NI-4 indicates 6 x 10^{-8} amps
- At T=9 minutes
  - NI-3 indicates 1.3 x 10^{-10} amps
  - NI-4 indicates 5.8 x 10^{-8} amps

Which one of the following explains the reason for the response of the intermediate range nuclear instruments?

Compensating voltage on ___(1)___is set too ___(2)___.

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

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

C. (1) NI-4
(2) low

D. (1) NI-4
(2) high

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must recognize: (1) On a normal reactor trip that the intermediate range power levels should decay off at about 1/3 decade per minute (DPM) (TQ-TM-104-623-C001, Rev 5 Page 17). (2) NI-3 is only instrument that is displaying the correct behavior so the examinee must determine that NI-4 is malfunctioning. (3) In accordance with the TQ-TM-104-623-C001 (Page 17), if NI-4 were overcompensating then the neutron flux would result in a less than actual indication. This is not the case since with indication is higher than expected. NI-4 is undercompensated, which means its compensating voltaget is set too low.

A.	(1) NI-3 (2) low	INCORRECT: (1) Plausible if the examinee determines that NI-4 is exhibiting the correct behavior after a reactor trip. Incorrect because NI-4 power indication is too high. (2) Plausible if the examinee does not understand over/under compensation of an intermediate range instrument.
B.	(1) NI-3 (2) high	INCORRECT: (1) Plausible if the examinee determines that NI-4 is exhibiting the correct behavior after a reactor trip. Incorrect because NI-4 power indication is too high. (2) Plausible if the examinee does not understand over/under compensation of an intermediate range instrument.
C.	(1) NI-4 (2) low	CORRECT: See above.
D.	(1) NI-4 (2) high	INCORRECT: (1) Correct answer. (2) Plausible if the examinee does not understand over/under compensation of an intermediate range instrument.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	015	K6.01	
	Importance Rating	2.9		

K/A: Nuclear Instrumentation: Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Sensors, detectors, and indicators

Proposed Question:	Question #57	
Technical Reference(s):	TQ-TM-104-623-C001, Rev 5	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	623-GLO-11	

Question Source:	Bank #	371840	
	Modified Bank #		
	New		

Question History:	Sim Exam 1	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.2	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the characteristics to identify the loss of a nuclear instrumentation detector.

High Cog: This question is high cog because the examinee must analyze the detector characteristics in the stem and determine which detector is correct.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

58 **ID: 2084857** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- Tave in each loop is as follows:
  - RCS Loop A Tave 580F
  - RCS Loop B Tave 578F
  - RCS Loop A&B Average 579F

- RCS Loop A&B is the selected Tave

Event:

- The Reactor Operator observes the following indications for RCS flow:
  - RCS loop A LOWERS to 33 x 10E6 lbm/hr
  - RCS loop B RISES to 78 x 10E6 lbm/hr

Which of the following describes the operation of RC-12-TaS, Tave Auto/Manual Transfer Switch, and what effect will this have on control rods?

RC-12-TaS will be automatically selected to Loop ___(1)___, and ICS will ___(2)___ control rods.

A. (1) A
(2) insert

B. (1) B
(2) insert

C. (1) A
(2) withdraw

D. (1) B
(2) withdraw

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) RC-12-TaS is a transfer switch that selects the RCS loop with the highest flow (if one loop is greater than 95% flow and the other is less than 95%) for Tave control. This Tave goes to the console digital Tave to the ICS (TQ-TM-104-624-C001, Rev 5 Page 28). (2) The examinee must recognize that the Tave control will switch to the B loop. (3) Once the examinee has correctly determined the Tave that is controlling, they will have to look at the actual Tave itself and determine how will effect the ICS signal. Since the B loop Tave is lower than the RCS Loop A&B Average, this will create a positive neutron error which will withdraw control rod (TQ-TM-104-621-C001, Rev 10 Page 50).

A.	(1) A (2) insert	INCORRECT: (1) Plausible if the examinee believes that the loop with the lowest flow gets selected for Tave control. Incorrect because the loop with the highest flow gets selected. (2) Plausible if the examinee incorrectly determines the Tave effect on the Reactor control. On the console, the error will show up reverse as the ICS signal. Incorrect because this event will lead to a rod withdraw.
B.	(1) B (2) insert	INCORRECT: (1) Correct answer. (2) Plausible if the examinee incorrectly determines the Tave effect on the Reactor control. On the console, the error will show up reverse as the ICS signal. Incorrect because this event will lead to a rod withdraw.
C.	(1) A (2) withdraw	INCORRECT: (1) Plausible if the examinee believes that the loop with the lowest flow gets selected for Tave control. Incorrect because the loop with the highest flow gets selected (2) Correct answer.
D.	(1) B (2) withdraw	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	016	K4.03
	Importance Rating	2.8	

K/A: Nonnuclear Instrumentation: Knowledge of the NNIS design feature(s) and/or interlock(s) which provide for the following: Input to control system

Proposed Question:	Question #58	
Technical Reference(s):	TQ-TM-104-621-C001, Rev 10	TQ-TM-104-624-C001, Rev 5

Proposed References to be provided to applicants during examination:	None

Learning Objective:	624-GLO-5	

Question Source:	Bank #		
	Modified Bank #	371840	

	New			
Question History:	Unmod on Sys Exam 11	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know how the Tave NNI system inputs into the ICS control system.

High Cog: This question is high cog because the examinee must analyze the current Tave and flow conditions and then determine the plant response.

356776

Plant Conditions:

- Operating at 100% power.
- ICS in full AUTOMATIC.

Event:

- The Reactor Operator observes the following indications for RCS flow:
  - RCS loop A LOWERS to 33 x 10^6 lbm/hr.
  - RCS loop B RISES to 78 x 10^6 lbm/hr.

Which of the following describes the operation of RC-12-TaS, Tave Auto/Manual Transfer Switch?

A. Automatically selects loop A.

B. Automatically selects loop B.

C. Allows the operator to select only loop A.

D. Allows the operator to select only loop B.

Answer B

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

59 **ID: 2109431** **Points: 1.00**

Plant Conditions:

- Reactor Coolant System is in REFUELING SHUTDOWN condition
- Fuel has been offloaded
- The Fuel Transfer Canal has been filled
- Fuel Transfer Canal Isolation Valves FH-V-1A and FH-V-1B are closed

Event:

- An Operating Basis Earthquake has occurred
- Spent Fuel Pool Temperature is 135°F and rising slowly
- Alarm PLB-2-9, SPENT FUEL POOL A LEVEL LO, is illuminated
- Alarm PLB-2-10, SPENT FUEL POOL B LEVEL LO, is illuminated
- Spent Fuel Pool level is 340' and slowly lowering

In accordance with OP-TM-AOP-035, LOSS OF SPENT FUEL COOLING, which of the following actions is required?

A. Open FH-V-1A and FH-V-1B

B. Transfer water from an RCBT to the Spent Fuel Pool

C. Place both trains of Spent Fuel Pool Cooling in service

D. Commence makeup to the Spent Fuel Pool from the BWST

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The entry criteria that OP-TM-AOP-035 (Rev 9 Page 1) is entered for is Spent Fuel Level less than 23.167' (342'6"). (2) Step 3.2 (Rev 9 Page 1) is to initiate 1104-29W to transfer water from the RCBT to the Spent Fuel Pool.

A.	Open FH-V-1A and FH-V-1B	INCORRECT: Plausible because this action would raise level in the Spent Fuel Pool. Incorrect because OP-TM-AOP-035 directs these valves closed if fuel is in the Reactor Vessel.
B.	Transfer water from an RCBT to the Spent Fuel Pool	CORRECT: See above.
C.	Place both trains of Spent Fuel Pool Cooling in service	INCORRECT: Plausible because this is an action in OP-TM-AOP-035. Incorrect because the operators take this action if Spent Fuel Temperature exceeds 160F.
D.	Commence makeup to the Spent Fuel Pool from the BWST	INCORRECT: Plausible because there is a flowpath where water from the BWST can be used to filled the Spent Fuel pool. Incorrect because at this point in Refueling Outage the BWST will be nearly empty.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	033	A1.01	
	Importance Rating	2.7		

K/A: Spent Fuel Pool Cooling: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling operating controls: Spent fuel pool water level

Proposed Question:	Question #59	
Technical Reference(s):	OP-TM-AOP-035, Rev 9	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	A35-PCO-4	

Question Source:	Bank #	757545	
	Modified Bank #		
	New		

Question History:	Simulator Exam 7	Last NRC Exam:		

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must know the procedure which will fill the Spent Fuel Pool after a lowering level is observed. By knowing that OP-TM-AOP-025 is entered and that 1104-29W is initiated, the examinee will know that controls must be operated to prevent level from lowering past a point where the spent fuel pool cooling design is exceeded.

60 **ID: 2088045** **Points: 1.00**

How is the Reactor Building Purge Flow Rate controlled?

RB Purge Flow Rate is controlled by a combination of controlling air flow from the discharge of AH-E-6A/6B, Purge Supply Fans using ___(1)___ and controlling air flow from the Air Intake Tunnel by controlling the position of ___(2)___.

A. (1) AH-D-8B, RB Purge Manual Loader
(2) AH-D-82, Makeup Damper

B. (1) AH-D-8B, RB Purge Manual Loader
(2) AH-V-1A, Purge Exhaust Valve

C. (1) AH-D-82, Makeup damper
(2) AH-V-8B, RB Purge Manual Loader

D. (1) AH-D-82, Makeup damper
(2) AH-V-1A, Purge Exhaust Valve

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Air for the Reactor Building Purge is brought in by the AH-E-6A/B fans from outside into the reactor building via AH-V-1C and AH-V-1D. (2) Air is exhausted from the Reactor Building by the AH-E-7A/B fans via AH-V-1A and AH-V-1B. (302-831, Rev 58). (3) Between the AH-V-1A/B valves and the AH-E-7A/B fans is a filter (AH-F-1) which filters air from the Reactor Building and makeup from the Air Intake Tunnel via AH-D-82 (302-832 Rev 9). (4) When the operator adjusts the Manual Purge Loader in the control room, they are adjusting AH-D-8B. When they are setting AH-PC-1146 they are adjusting makeup air to the AH-E-7 suction via AH-D-82 (OP-TM-823-406, RB PURGE - CONTAINMENT CLOSED, Rev 12 Page 7).

A.	(1) AH-D-8B RB Purge Manual Loader (2) AH-D-82 Makeup Damper	CORRECT: See above.
B.	(1) AH-D-8B RB Purge Manual Loader (2) AH-V-1A Purge Exhaust Valve	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes that adjusting a purge exhaust valve will adjust air flow from the air intake tunnel.
C.	(1) AH-D-82 Makeup damper (2) AH-V-8B RB Purge Manual Loader	INCORRECT: (1) Plausible if he examinee believe makeup comes from outside via the AH-E-6. Incorrect because this is the Purge Manual Loader. (2) Plausible if the examinee
D.	(1) AH-D-82 Makeup damper (2) AH-D-1A Purge Exhaust Valve	INCORRECT: 1. Plausible if the examinee believes the Makeup Damper controller, controls the air flow form the Intermediate Building. 2. Plausible if the examinee believes that controlling AH-V-1A would adjust air flow form the Air Intake Tunnel

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	029	A4.01	
	Importance Rating	2.5		

K/A: Containment Purge: Ability to manually operate and/or monitor in the control room: Containment purge flow rate

Proposed Question:	Question #60	
Technical Reference(s):	OP-TM-823-406, Rev 12	302-831, Rev 58
	302-832, Rev 9	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	824-GLO-10	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know how to manually control the containment purge in the control room.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

61 **ID: 2109607** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- Loss of Offsite Power

The following conditions exist:

- The reactor has tripped and OP-TM-EOP-001 has been initiated
- The initial post-trip Symptom Check is in progress

In accordance with OS-24, CONDUCT OF OPERATIONS DURING ABNORMAL AND EMERGENCY EVENTS, which of the following is the correct method to determine Subcooling Margin (SCM) during the initial symptom check?

A. Incore subcooling margin point C4008

B. A SCM meter on PCL (TI-977 or TI-978)

C. Highest BIRO incore temperature reading and PI-949A plotted on OP-TM-EOP-010, Fig 1

D. PPC point C4006 INCORE TC 5 HIGHEST TEMPS -AVG and RC3-PR CH3 plotted on OP-TM-EOP-010 Fig 1

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OS-24 (Rev 31 Page 21) if the Reactor Coolant Pumps are off and natural circulation has not been verified then the operators must use incore subcooling margin C4008. (2) Due to the LOOP, no Reactor Coolant Pumps are running. The design feature of the normally used subcooling margin indication is that the crew must have verified flow to be able to call their indication correct. (3) The description of the initial symptom check should indicate that natural circulation has not been verified yet based on the Reactor Trip actions described in Section 4.3.4 of OS-24 (Rev 31 Page 15)

A.	Incore subcooling margin point C4008	CORRECT: See above.
B.	A SCM meter on PCL (TI-977 or TI-978)	INCORRECT: Plausible because this is the method the operators would use on a Reactor Trip with forced flow. Incorrect because the Reactor Coolant Pumps are not running.
C.	Highest BIRO incore temperature reading and PI-949A plotted on OP-TM-EOP-010, Fig 1	INCORRECT: Plausible because this is part of the OS-24 Attachment J (Rev 31 Page 41) calculation for Subcooling Margin. Incorrect because in accordance with OS-24 step 4.7.1.2.1, the criteria for using this method has not been met.
D.	PPC point C4006 INCORE TC 5 HIGHEST TEMPS -AVG and RC3-PR CH3 plotted on OP-TM-EOP-010 Fig 1	INCORRECT: Plausible because this is part of the OS-24 Attachment J (Rev 31 Page 41) calculation for Subcooling Margin. Incorrect because in accordance with OS-24 step 4.7.1.2.1, the criteria for using this method has not been met.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	017	K4.01
	Importance Rating	3.4	

K/A: In-Core Temperature Monitor: Knowledge of ITM system design feature(s) and/or interlock which provide for the following: Input to subcooling monitors

Proposed Question:	Question #61	
Technical Reference(s):	OS-24, Rev 31	

Proposed References to be provided to applicants during examination: None

Learning Objective: 625-GLO-8

Question Source:	Bank #	299774	
	Modified Bank #		
	New		

Question History:	Sim Exam 9	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know that the design of the incore subcooling margin PPC point is to provide an indication of SCM when all RCPs are shutdown and natural circulation has not been verified. The incore subcooling margin PPC point recieves its temperature input from the Incore System. The Incore system is designed to still provide an accurate temperature when there is no forced flow and RCS natural circulation has not been verified.

High Cog: This question is high cog because the examinee must know the effects of a LOOP and what instruments are used to measure Subcooling Margin.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

62 **ID: 2105461** **Points: 1.00**

Which of the following events will cause the P-T plot screen to have this shape and parameters?

A. Loss of Offsite Power

B. Loss of all Feedwater

C. Turbine Trip from 100% power

D. Unisolable Steam Line Rupture

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The image in the question is a P-T plot after a Turbine Trip from 100% power from the PPC. (2) The shape of the curve is due to the cooldown and shrinkage of RCS water due to the trip. Initially Pressurizer level goes down until the cooldown subsides then pressure will start to rise to the normal operating bank. (3) In addition there is MFW flow indication on the left and side which the examinee can determine that there is feedwater flow.

A.	Loss of Offsite Power	INCORRECT: Plausible because this will cause a Reactor and Turbine trip. Incorrect because the natural circulation box would come up and there would be no indication for MFW flow.
B.	Loss of all Feedwater	INCORRECT: Plausible because this would cause a Reactor and Turbine Trip. Incorrect because there is still MFW flow on the left hand of the plot.
C.	Turbine Trip from 100% Power	CORRECT: See above
D.	Unisolable Steam Line Rupture	INCORRECT: Plausible because an RPS setpoint would be reached and the crew would trip the Reactor. Incorrect because an Unisolable Steam Line Rupture would cause an excessive heat transfer which would drag the plot back further down past left side of the box.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	045	A1.05	
	Importance Rating	3.8		

K/A: Main Turbine Generator: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MT/G system controls including: Expected response of primary plant parameters (temperature and pressure) following a T/G trip

Proposed Question:	Question 62	
Technical Reference(s):	Plant Simulator	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EOP001-PCO-2	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

Question Cognitive Level:	Memory or Fundamental Knowledge		

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.5	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know (and be able to monitor) the correct primary parameters (pressure and temperature) on a trip of the plant. Specifically the examinee must identify that the it was the turbine that tripped and not any of the other malfunctions.

High Cog: This question is high cog because the examinee will have to analyze the PT plot and parameters and identify the cause.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

63 **ID: 2105465** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

The Condensate and Feedwater lineup is as follows:

- CO-P-1A and CO-P-1B switches are in the NORM AFT START position
- CO-P-1C switch is in PULL TO LOCK position
- CO-P-2A and CO-P-2B switches are in the NORM AFT START position
- CO-P-2C switch is in NORM AFT STOP position
- FW-P-1A is the last RESET feed pump

Which of the following describes plant response to a trip of Condensate Pump, CO-P-1B?

NOTE:
CO-P-1A/B/C - Condensate Pumps
CO-P-2A/B/C - Condensate Booster Pumps
FW-P-1A/B - Feedwater Pumps

A. Only CO-P-2A will trip

B. Only CO-P-2B will trip

C. CO-P-2A and FW-P-1A will trip

D. CO-P-2B and FW-P-1B will trip

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The Condensate, Condensate Booster Pumps, and Feedwater Pumps trip circuit involves using a counting circuit. (2) If the number of condensate pumps does not match number of booster pumps, the circuit will attempt to start a condensate pump. (3) CO-P-1C is in PTL and will not start. (3) After .5 seconds, and if the counting circuit is still not satisfied, the circuit will sequentially trip booster pumps starting with CO-P-2A, then CO-P-2B until the counting circuit is satisfied. When two booster pumps are no longer running, the last reset FW pump will trip.

A.	Only CO-P-2A will trip	INCORRECT: Plausible if the examinee believes that only a condensate booster pump will trip and the counting circuit will not trip a Main Feedwater Pump.
B.	Only CO-P-2B will trip	INCORRECT: Plausible if the examinee believes that only a condensate booster pump will trip and the counting circuit will not trip a Main Feedwater Pump.
C.	CO-P-2A and FW-P-1A will trip	CORRECT: See above
D.	CO-P-2B and FW-P-1B will trip	INCORRECT: Plausible if the examinee believes that CO-P-2B would be the first pump to trip because CO-P-1B trips (i.e the pumps are run in pairs). In addition the examiee could believe that the last first RESET FW pumps trips and not the last RESET FW pump.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	2		
	K/A #	056	K1.03	
	Importance Rating	2.6		

K/A: Condensate: Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following: MFW

Proposed Question:	Question 63	
Technical Reference(s):	OP-TM-MAP-M0101, Rev 2	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	421-GLO-4	

Question Source:	Bank #	879453	
	Modified Bank #		
	New		

Question History:	System Exam 5	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the interrelation (cause-effect relationship) between condensate pumps and the feedwater pumps.

High Cog: This question is high cog because the examinee must analyze the abnormal lineup and determine the correct plant response.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

64 **ID: 2105468** **Points: 1.00**

Plant Conditions:

- "B" OTSG tube leak of approximately 20 gpm has occurred
- Plant Cooldown is in progress per OP-TM-EOP-005, OTSG Tube Leakage
- The most recent sample of RCS activity is 1.5 microcuries / ml
- B OTSG TURB BYP LINE RAD MONITOR, RM-G-27, is reading 2,000 cpm
- MAIN CONDENSER OFFGAS RAD MONITOR, RM-A-5, is reading 200 cpm

Event:

- At approximately 450 F, a small fuel failure occurs, raising noble gas content in the RCS from 0.05 to 1.5 microcuries / ml

Given the above information and assuming only noble gasses are released from the fuel failure, RM-G-27 ___(1)___, and RM-A-5 ___(2)___.

A. (1) stays constant
(2) approximately doubles

B. (1) stays constant
(2) goes up by a factor of about 30

C. (1) approximately doubles
(2) approximately doubles

D. (1) approximately doubles
(2) goes up by a factor of about 30

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) RM-G-27 measures activity in the steam line (TQ-TM-104-661-C001, Rev 7 Page 23). RM-G-27 counts double because the RCS activity roughly double (Most recent sample at 1.5 microcuries / ml + 1.5 microcuries / ml from the event). (2) RM-A-5 is the condenser off gas sampler. RM-A-5 goes up by a factor of 30 because the noble gas goes up by a factor of about 30. RM-A-5 measures noble gasses which to stay in solution.

A.	(1) stays constant (2) approximately doubles	INCORRECT: Plausible if the examinee believes RM-G-27 stays constant because noble gasses aren't seen by RM-G-27, and RM-A-5 approximately doubles because RCS activity approximately doubles. Incorrect since RM-G-27 does measure all activity passing through the TBV steam lines.
B.	(1) stays constant (2) goes up by a factor of about 30	INCORRECT: Plausible if the examinee believes RM-G-27 stays constant because noble gasses aren't seen by RM-G-27, and RM-A-5 goes up by a factor of about 30 because the noble gas activity goes up by a factor of about 30. Incorrect since RM-G-27 does measure all activity passing through the TBV steam lines.
C.	(1) approximately doubles (2) approximately doubles	INCORRECT: Plausible if the examinee believes RM-G-27 approximately doubles because RCS activity approximately doubles, and RM-A-5 approximately doubles because RCS activity approximately doubles. Incorrect since RM-A-5 is mostly measuring noble gasses which do not stay in solution.
D.	(1) approximately doubles (2) goes up by a factor of about 30	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	073	K5.01
	Importance Rating	2.7	

K/A: Process Radiation Monitoring: Knowledge of the operational implications of the following concepts as they apply to the ARM system: Radiation theory, including sources, types, units, and effects

Proposed Question:	Question #64
Technical Reference(s):	TQ-TM-104-661-C001, Rev 7
Proposed References to be provided to applicants during examination:	None
Learning Objective:	661-GLO-7

Question Source:	Bank #	469338
	Modified Bank #	
	New	

Question History:	Simulator Exam 9	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.11	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the operational implications which may arise from the radiation monitor trend. The examinee must know radiation theory to understand the factors in which the radiation monitors rise due to a fuel failure. RM-G-27 and RM-A-5 are process radiation monitors.

High Cog: The question is high cog because the examinee must know and analyze the relationship between RCS activity and the radiation monitor readings.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

65 **ID: 2086971** **Points: 1.00**

Event:

- A malfunction has occurred such that all of the Ultraviolet (UV) Detectors associated with the Air Intake Tunnel Halon System are inoperable and incapable of automatically actuating the system.

Which of the following identifies how automatic operation of the Air Intake Tunnel Halon System has been affected by this malfunction?

The Air Intake Tunnel Halon System _____.

A. will NOT be automatically actuated by any of the other devices.

B. will be actuated automatically by the Accelerometer (AS40) actuating, ONLY

C. will be actuated automatically by any Pressure Wave (Explosion) Detector(s) actuating, ONLY

D. will be actuated automatically by either the Accelerometer (AS40) actuating or one Pressure Wave (Explosion) Detector actuating

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: Each Halon zone acts independently with its own ultraviolet (UV) and pressure wave detectors. There are 13 UV detectors, with a minimum of two detectors located in each zone. Additionally, there are 10 Pressure Wave Detectors, also with a minimum of two detectors in each zone. Any one ultraviolet detector or pressure wave detector will actuate the Air Intake Tunnel Halon System in the associated section. Consequently, the Air Intake Tunnel Halon System can be actuated automatically by one Pressure Wave (Explosion) Detector actuating. (TQ-TM-104-810-C001, Rev 11 Page 75)

A.	will NOT be automatically actuated by any of the other devices.	INCORRECT: Plausible because the examinee could believe that the automatic system is disabled and can be only actuated manually. Incorrect because the Pressure Wave (Explosion) Detector(s) can will still actuate the system.
B.	will be actuated automatically by the Accelerometer (AS40) actuating, ONLY.	CORRECT: Plausible because the system has an Accelerometer. The system has one Accelerometer to detect airplane crashes into intake structure. This device provides for alarms in the Control Room, and the typical ventilation system responses, but will NOT actuate the Halon System. The operator may incorrectly believe that it is this device, and not the pressure wave detectors that will automatically actuate the system.
C.	will be actuated automatically by any Pressure Wave (Explosion) Detector(s) actuating ONLY	CORRECT: See above
D.	will be actuated automatically by either the Accelerometer (AS40) actuating or one Pressure Wave (Explosion) Detector actuating	INCORRECT: Plausible because the system has an Accelerometer. The system has one Accelerometer to detect airplane crashes into intake structure. This device provides for alarms in the Control Room, and the typical ventilation system responses, but will NOT actuate the Halon System. The operator may incorrectly believe that it is both this device, and the pressure wave detectors that will automatically actuate the system.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	086	K6.04
	Importance Rating	2.6	

K/A: Fire Protection System: Knowledge of the effect of a loss or malfunction on the Fire Protection System following will have on the: Fire, smoke, and heat detectors

Proposed Question:	Question #65
Technical Reference(s):	TQ-TM-104-810-C001, Rev 11
Proposed References to be provided to applicants during examination:	None
Learning Objective:	811-GLO-8

Question Source:	Bank #	860305		
	Modified Bank #			
	New			
Question History:	System Exam 12	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.7
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have knowledge on how a malfunction of the fire protection system effects the Air Intake Tunnel Halon System.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

66 **ID: 2105575** **Points: 1.00**

Which of the following elements/isotopes are checked in accordance with 1301-1 SHIFT AND DAILY CHECKS for Primary and Secondary Activity/Chemistry?

A. (1) Primary: Oxygen, Chlorides, Fluorides only
(2) Secondary: Iodine

B. (1) Primary: Iodine, Xenon, Oxygen, Chlorides, Fluorides
(2) Secondary: Iodine

C. (1) Primary: Oxygen, Chlorides, Fluorides only
(2) Secondary: Xenon

D. (1) Primary: Iodine, Xenon, Oxygen, Chlorides, Fluorides
(2) Secondary: Xenon

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In 1301-1 SHIFT AND DAILY CHECKS (Rev 178 page 35) operators check that Reactor Coolant and Secondary Chemistry is within the technical specification limits. The operators check Xenon, Iodine, Oxygen, Chlorides and Fluorides in the RCS. They check Iodine in the Secondary.

A.	(1) Primary: Oxygen, Chlorides, Fluorides (2) Secondary: Iodine	INCORRECT: (1) Plausible because one section (D.3) checks for these elements. Incorrect because another section checks for Xenon and Iodine as well. (2) Correct answer.
B.	(1) Primary: Iodine, Xenon, Oxygen, Chlorides, Fluorides (2) Secondary: Iodine	CORRECT: See above
C.	(1) Primary: Oxygen, Chlorides, Fluorides (2) Secondary: Xenon	INCORRECT: (1) Plausible because one section (D.3) checks for these elements. Incorrect because another section checks for Xenon and Iodine as well (2) Plausible if the examinee believes we check for Xenon in the Secondary.
D.	(1) Primary: Iodine, Xenon, Oxygen, Chlorides, Fluorides (2) Secondary: Xenon	INCORRECT: (1) Correct answer. (2) Plausible if the examinee believes we check for Xenon in the Secondary.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.34	
	Importance Rating	2.7	

K/A: Knowledge of primary and secondary chemistry limits.

Proposed Question:	Question #66
Technical Reference(s):	1301-1, Rev 178

Proposed References to be provided to applicants during examination: None

Learning Objective: 551-GLO-6

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

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

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the elements/isotopes in the primary and secondary that have limits.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

67 **ID: 2105590** **Points: 1.00**

Plant conditions:

- A refueling outage has commenced
- The core is in the process of being off-loaded

Which ONE of the following describes the correct action if RM-G-9, FUEL HANDLING BUILDING FUEL HANDLING BRIDGE, radiation monitor fails high?

A. Cease all Spent Fuel Pool fuel movement until proper portable survey instruments are provided to monitor radiation levels

B. Spent Fuel Pool fuel movements may continue as long as RM-A-4, FHB VENT RADIATION MONITOR, remains operable

C. Spent Fuel Pool fuel 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 closed AND RM-A-4 FHB VENT RADIATION MONITOR has been verified operable

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1505-1, FUEL AND CONTROL COMPONENT SHUFFLES (Rev 62 Page 7) step 5.1.2 says if any RM-G unit becomes inoperable, that 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. (2) In accordance OP-TM-MAP-C0101, RADIATION LEVEL HI (Rev 4 Page 25 and 26), the crew would clear the fuel handling bridge, ensure the interlock worked, and ensure one train of ESF ventilation is in service.

A.	Cease all spent fuel pool fuel movement until proper portable survey instruments are provided to monitor radiation levels	CORRECT: See above
B.	Spent Fuel Pool movements may continue as long as RM-A-4, FHB VENT RADIATION MONITOR, remains operable	INCORRECT: RM-A-4 monitors FHB exhaust flow for radiation and would provide warning of off-normal airborne conditions. Incorrect because RM-G-9 is specifically required by 1505-1.
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	INCORRECT: Plausible if it is believed that satisfying the interlock to RM-G-9 would be correct. Incorrect because another portable instrument is still required.
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	INCORRECT: Plausible if it was believed that ensuring RM-A-4 could provide replacement interlock function for RM-G-9. Incorrect because another portable instrument would still be required.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.36	
	Importance Rating	3.0	

K/A: Knowledge of procedures and limitations involved in core alterations.

Proposed Question:	Question #67
Technical Reference(s):	1505-1, Rev 62

Proposed References to be provided to applicants during examination: None

Learning Objective:	661-GLO-10		
Question Source:	Bank #	895238	

	Modified Bank #			
	New			
Question History:	N/A	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.12
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the procedures and limitations with core refueling.

RO Only: This question is RO only because part of the alarm response (OP-TM-MAP-C0101) directs fuel handling to be ceased. In addition, they must know that ventilation must be operating to move fuel in the FHB.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

68 **ID: 2109436** **Points: 1.00**

Given the following plant conditions:

- Core age is 345 EFPD
- Plant startup is in progress
- Xenon Free Startup
- The reactor was made critical 6 hours ago
- Plant power ascension is on hold at 55% power and is expected to remain at this power level for the next 5 hours
- Control rod index is 273%
- Boron concentrations are as follows:
  - RCS boron concentration is 1334 ppm boron
  - RCBT 'A' boron concentration is 27 ppm boron
  - RCBT 'B' boron concentration is 1331 ppm boron
  - RCBT 'C' boron concentration is 2052 ppm boron
  - BWST boron concentration is 2455 ppm boron

In order to maintain the current control rod index over the next 5 hours, RCS boron must be controlled by making up from the _____.

A. 'A' RCBT

B. 'B' RCBT

C. 'C' RCBT

D. BWST

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Xenon concentration will be building towards equilibrium values during the 5 hour hold which adds negative reactivity. (2) In order to maintain the current rod position, the RCS boron concentration will need to be reduced in order to add positive reactivity. (3) Water must be added from a boron concentration less than the current boron concentration to dilute. (4) OP-TM-211-455, FEED FROM A RCBT (Rev 30 covers feeding from the bleed tanks and the effects on control rod position.

A.	'A' RCBT	CORRECT: See above
B.	'B' RCBT	INCORRECT: Plausible since 'B' RCBT is close to the current RCS Boron Concentration but still less than RCS Boron. The examinee may believe Xenon will have minimal affect and this concentration would maintain CR Index for the time of 5 hours.
C.	'C' RCBT	INCORRECT: Plausible if examinee does not understand the direction Xenon concentration will go in for this short period of time.
D.	BWST	INCORRECT: Plausible if examinee does not understand the magnitude and direction Xenon concentration will go and a need to raise boron concentration rapidly to maintain rod position.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.43	
	Importance Rating	4.1	

K/A: Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion

Proposed Question:	Question #68	
Technical Reference(s):	OP-TM-211-455, Rev 3	

Proposed References to be provided to applicants during examination:	NONE

Learning Objective:	SOER 94-2 (V.I.11.06)	

Question Source:	Bank #	1588528	
	Modified Bank #		
	New		

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

Question Cognitive Level:	Memory or Fundamental Knowledge		

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.5	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know how to select the correct tank in order to choose the correct procedure.

High Cog: The examinee has to analyze the plan conditions and understand the reactivity feedback from Xenon and the procedure response for maintaining the control rod position.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

69 **ID: 2105974** **Points: 1.00**

In accordance with 1102-4, POWER OPERATIONS, which of the following personnel must be notified prior to lowering power for Turbine Valve Testing?

A. Transmission System Operator, only

B. Transmission System Operator and the Nuclear Duty Officer, only

C. Transmission System Operator, Nuclear Duty Officer, and Power Team

D. Transmission System Operator, Nuclear Duty Officer, and Generation Dispatch

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance 1102-4, POWER OPERATIONS, (Rev 137 Page 9) if power change is greater than 10 MWe then perform the following notify the TSO, NDO and Power Team. (2) Reactor power must be lowered to less than 90% on the hightest reading NI, which is greater than 10 MWe.

A.	Transmission System Operator, only	INCORRECT: Plausible if examinee thinks this is the only person necessary to notify for a change in MW production.
B.	Transmission System Operator and the Nuclear Duty Officer, only	INCORRECT: Plausible if examinee thinks these are the only people necessary to notify.
C.	Transmission System Operator, Nuclear Duty Officer, and Power Team	CORRECT:
D.	Transmission System Operator, Nuclear Duty Officer, and Generation Dispatch	INCORRECT: Plausible if examinee believes Generation Dispatch must be notified.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	K/A #	2.2.17		
	Importance Rating	2.6		

K/A: Knowledge of the process for managing maintenance activities during power operation, such as risk assessments, work prioritization, and coordination with the transmission operator.

Proposed Question:	Question #69	
Technical Reference(s):	1102-4, Rev 137	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	GOP-004-PCO-2	

Question Source:	Bank #	N/A	
	Modified Bank #	N/A	
	New	X	

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

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

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the correct communication requirements with respect to reducing electrical load, and who must be notified with respect to grid operation.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

70 **ID: 2087168** **Points: 1.00**

In accordance with OP-AA-109-101, PERSONNEL AND EQUIPMENT TAGOUT PROCESS, which of the following is correct regarding double valve isolation of a mechanical system?

Double valve isolation should be provided when isolating a system from an energy source of temperature greater than 200F ___(1)___ pressure greater than 500 psig. If available, ___(2)___ between the two closed isolation valves must tagged open or designated exceptional.

A. (1) OR
(2) BOTH vent and drain valves

B. (1) AND
(2) EITHER vent or drain valve

C. (1) OR
(2) EITHER vent or drain valve

D. (1) AND
(2) BOTH vent and drain valves

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-AA-109-101 (Rev 13 Page 28) Section 7.3.1 that double valve isolation should be provided when isolating from an energy source of greater than 200F or pressures greater than 500 psig. (2) If available, a vent or drain valve between the two closed valves should be tagged open or the clearance should be designated as exceptional.

A.	(1) OR (2) BOTH vent and drain valves	INCORRECT: Plausible if the examinee believes both valves must be tagged.
B.	(1) AND (2) EITHER vent or drain valve	INCORRECT: Plausible if the examinee does not know the requirements for double valve isolation and that both valves must be tagged.
C.	(1) OR (2) EITHER vent or drain valve	CORRECT: See above.
D.	(1) AND (2) BOTH vent and drain valves	INCORRECT: Plausible if the examinee does not know the requirements for double valve isolation.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	K/A #	2.2.13		
	Importance Rating	4.1		

K/A: Knowledge of tagging and clearance procedures.

Proposed Question:	Question #70	
Technical Reference(s):	OP-AA-109-101, Rev 13	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	NOP-DBIG-PCO-2	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

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

	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the tagging and clearance procedure.

Tier 3: This question is Tier 3 because it is not tied to any system.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

71 **ID: 2087171** **Points: 1.00**

OP-AA-108-101, CONTROL OF EQUIPMENT AND SYSTEM STATUS, contains directions on the use of the Abnormal Component Position Sheet (ACPS).

The ACPS provides a controlled method for _____.

A. aligning equipment outside of routine operations.

B. changing component positions for tasks involving more than 1 system

C. temporarily changing positions of components that have an effect on the UFSAR

D. positioning components as part of a Temporary Configuration Change Package (TCCP)

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-AA-108-101, CONTROL OF EQUIPMENT AND SYSTEM STATUS (Rev 14 Page 4) Step 4.1.1.1 says to utilize an ACPS for aligning equipment outside of routine operations.

A.	aligning equipment outside of routine operations	CORRECT: See above.
B.	changing component positions for task involving more than 1 system	INCORRECT: Plausible if the examinee believes that an ACPS would aid in completing work in a timely manner. Incorrect because step 4.1.2.2 prohibits use of an ACPS in this manner.
C.	temporarily changing positions of components that have an effect on the UFSAR	INCORRECT: Plausible if the examinee believes this method is acceptable because it is temporary. Incorrect because step 4.1.2.5 prohibits use of an ACPS in this manner.
D.	positioning components as part of a Temporary Configuration Change Package (TCCP)	INCORRECT: Plausible if the examinee believes this method could be used to document the changes required by a TCCP. Incorrect because step 4.1.2.1 prohibits use of an ACPS in this manner.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	2		
	K/A #	2.2.14		
	Importance Rating	3.9		

K/A: Knowledge of the process for controlling equipment configuration change.

Proposed Question:	Question #71	
Technical Reference(s):	OP-AA-108-101, Rev 14	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EQC02018	

Question Source:	Bank #	375154	
	Modified Bank #		
	New		

Question History:	Simulator Exam 2	Last NRC Exam:	N/A	

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

10 CFR Part 55 Content:	55.41	b.10

	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know the procedural requirements for changing equipment status that require an ACPS for configuration control.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

72 **ID: 2106162** **Points: 1.00**

Using the picture above, when the monitor reaches the High Alarm Setpoint:
(1) What is the correct color indication for the bar graph?
(2) Does MAP C0101, RADIATION LEVEL HI have reflash capability?

	1 BAR GRAPH INDICATION	2 C0101 ALARM RESPONSE
A.	Orange	Yes
B.	Orange	No
C.	Red	Yes
D.	Red	No

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with TQ-TM-104-661-C001 (Rev 9 Slide 42) describes the orange as warning and red as high. (2) MAP C0101 will reflash when any radiation monitor comes into warning or alarm (Slide 29).

A.	Orange Yes	INCORRECT: (1) Plausible if examinee does not know the indication for a high alarm on a detector. (2) Correct answer.
B.	Orange No	INCORRECT: (1) Plausible if examinee does not know the indication for a high alarm on a detector. (2) Plausible because most alarms do not automatically reflash.
C.	Red Yes	CORRECT: See above.
D.	Red No	INCORRECT: (1) Correct answer. (2) Plausible because most alarms do not automatically reflash.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	2.3.15		
	Importance Rating	2.9		

K/A: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question:	Question #72	
Technical Reference(s):	TQ-TM-104-661-C001, Rev 9	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	661-GLO-5	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

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

10 CFR Part 55 Content:	55.41	b.12
	55.43	

Comments:

KA Match: This question matches the KA because the examinee must have knowlege of a fixed instrument will respond to a high radiation level.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

73 **ID: 2085099** **Points: 1.00**

Which one of the following identifies a group of Post Accident Monitoring Instruments that are required to be OPERABLE by Technical Specification Table 3.5-3, Post Accident Monitoring Instrumentation?

A. Containment Pressure,
PORV Position Monitor, and
Steam Generator Pressure.

B. Containment Pressure,
PORV Position Monitor, and
RCS Cold Leg Temperature.

C. Containment Pressure,
Steam Generator Pressure, and
RCS Cold Leg Temperature.

D. PORV Position Monitor,
Steam Generator Pressure, and
RCS Cold Leg Temperature.

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with Technical Specification Table 3.5-3 (AMD 240 Page 3-40d) Containment Pressure, Steam Generator Pressure, and RCS Cold Leg Temperature are Post Accident Monitoring Instrumentation.

A.	Containment Pressure, PORV Position Monitor, and Steam Generator Pressure.	INCORRECT: Plausible because these instruments could be used for accident mitigation. Incorrect because the PORV Position Monitor is not a Post Accident Monitoring Instrument.
B.	Containment Pressure, PORV Position Monitor, and RCS Cold Leg Temperature.	INCORRECT: Plausible because these instruments could be used for accident mitigation. Incorrect because the PORV Position Monitor is not a Post Accident Monitoring Instrument.
C.	Containment Pressure, Steam Generator Pressure, and RCS Cold Leg Temperature.	CORRECT: All instruments are listed in Table 3.5.3 of Tech Spec 3.5.5
D.	PORV Position Monitor, Steam Generator Pressure, and RCS Cold Leg Temperature.	INCORRECT: Plausible because these instruments could be used for accident mitigation. Incorrect because the PORV Position Monitor is not a Post Accident Monitoring Instrument.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.3	
	Importance Rating	3.7	

K/A: Ability to identify post-accident instrumentation

Proposed Question:	Question #73
Technical Reference(s):	T.S. 3.5.5 Table 3.5-3, Amd 240
Proposed References to be provided to applicants during examination:	None
Learning Objective:	624-GLO-14

Question Source:	Bank #	907843	
	Modified Bank #		
	New		
Question History:	N/A	Last NRC Exam:	12-01

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

10 CFR Part 55 Content:	55.41	b.7	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know what post accident instrumentation is required by technical specifications.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

74 **ID: 2106063** **Points: 1.00**

In accordance with OS-24, CONDUCT OF OPERATIONS DURING ABNORMAL AND EMERGENCY EVENTS, which of the following is NOT an approved two handed operation?

While ___(1)___, using two hands to ___(2)___.

A. (1) throttling HPI
(2) operate the OPEN and STOP pushbuttons

B. (1) adjusting Delta Tc
(2) adjust A & B FW flow

C. (1) reseating the Main Steam Safety Valves
(2) lower pressure on both OTSGs

D. (1) paralleling an Emergency Diesel Generator
(2) close the breaker and raise load

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OS-24 (Rev 31 Page 6), the following operations are approved by the SOS for multiple concurrent operations (i.e two handed ops): EDG breaker closure and simultaneous load adjustment, operation of a STOP type jog valve (e.g. MU-V-16), using the DTCS trackball and keyboard to execute DTCS commands, adjusting Delta Tc by simultaneously adjusting A & B FW flow, and depressing the Reactor Trip and DSS pushbuttons.

A.	(1) throttling HPI (2) operate the OPEN and STOP pushbuttons	INCORRECT: See above.
B.	(1) adjusting Delta Tc (2) adjust A & B FW flow	INCORRECT: See above.
C.	(1) reseating the Main Steam Safety Valves (2) lower pressure on both OTSGs	CORRECT: This is a plausible two handed operation because there is one controller for each OTSG which controls the Turbine Bypass Valves. OP-TM-EOP-010, Guide 6 directs the operators to reseat the MSSV's after a reactor trip. It is allowable to take both of the controllers to hand and lower pressure on each OTSG until the MSSV's are reseated. It is now allowable to use two hands and reseat them at one time.
D.	(1) paralleling an Emergency Diesel Generator (2) close the breaker and raise load	INCORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	1		
	K/A #	2.1.20		
	Importance Rating	3.8		

K/A: Ability to interpret and execute procedure steps

Proposed Question:	Question #74	
Technical Reference(s):	OS-24, Rev 31	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	OS-24-PCO-1	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

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

Comments:

KA Match: This question matches the KA because the correct answer and each of the distractors would be performed by a procedure. The examinee must have knowledge that each of the distractors allows for multiple concurrent operations to be able to show they have the ability to execute the procedure step. In addition the examinee must be able to interpret the correct answers action to know that it does NOT allow for concurrent operations and show that they have the ability execute the procedure step one controller at a time.

Tier 3: This is a tier 3 KA because the question is not specifically tied to any Tier 2 system. Also, the examinee must have plantwide knowledge to know how to execute the actions of OS-24.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

75 **ID: 2109263** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- MU-P-1B is out of service
- MU-P-1A is supplying normal makeup and seal injection

Event:

- IC-V-3, Intermediate Cooling from Reactor Building Isolation Valve, fails closed
- Immediately following the closure of IC-V-3, the 1D 4160V bus trips

Following the trip of 1D 4160V Bus, the first procedure to enter is _____.

A. OP-TM-EOP-001, REACTOR TRIP

B. OP-TM-AOP-041, LOSS OF SEAL INJECTION

C. OP-TM-AOP-013, LOSS OF 1D 4160V BUS

D. OP-TM-AOP-032, LOSS OF INTERMEDIATE CLOSED COOLING

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When isolation valve IC-V-3 fails closed this will stop flow to ICCW cooled components in the reactor building. (2) One of the components cooled is the Reactor Coolant Pump thermal barriers. (3) When the 1D 4KV Bus is lost, Seal Injection flow is lost. (4) With no ICCW flow and no seal injection the reactor will trip. (5) When the reactor trips, the crew must enter OP-TM-EOP-001, REACTOR TRIP. (6) All other procedures will be entered after the reactor trip IMA's are complete.

A.	OP-TM-EOP-001, REACTOR TRIP	CORRECT: See above
B.	OP-TM-AOP-041, LOSS OF SEAL INJECTION	INCORRECT: Plausible because this procedure will be entered due to the loss of seal injection. Incorrect because the reactor trip will have occurred and the EOP-001 actions must be performed first.
C.	OP-TM-AOP-013, LOSS OF 1D 4160V BUS	INCORRECT: Plausible because this procedure will be entered due to the loss of seal injection. Incorrect because the reactor trip will have occurred and the EOP-001 actions must be performed first.
D.	OP-TM-AOP-032, LOSS OF INTERMEDIATE CLOSED COOLING	INCORRECT: Plausible because this procedure will be entered due to the loss of seal injection. Incorrect because the reactor trip will have occurred and the EOP-001 actions must be performed first.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	2.4.2		
	Importance Rating	4.5		

K/A: Knowledge of system setpoints, interlocks, and automatic actions associated with EOP entry conditions

Proposed Question:	Question #75	
Technical Reference(s):	OP-TM-EOP-001, Rev 16	

Proposed References to be provided to applicants during examination:	NONE

Learning Objective:	EOP-001-PCO-1	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

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

Question Cognitive Level:	Memory or Fundamental Knowledge		

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.10	
	55.43		

Comments:

KA Match: This question matches the KA because the examinee must know and interlock which leads to the EOP-001 entry.

High Cog: This question is high cog because the examinee must identify that a reactor trip occurred and the procedure that must be entered.

Tier 3: This question is a Tier 3 K/A because it requires the examinee to have knowledge of the overall procedure heirachy, which is not specific to any Tier 2 type question.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

76 **ID: 2095501** **Points: 1.00**

REFERENCE PROVIDED

Plant conditions:

- Refueling shutdown in progress
- DHR Train “A” aligned for RCS cooling
- Total Decay Heat Flow is 3000 gpm
- Incore thermocouple temperatures steady at 100 degrees F
- RCS water level is 30" above centerline

Event:

- A leak in the "A" Decay Heat train causes RCS Water Level to lower to 21" above centerline before being isolated.
- Incore thermocouple temperatures are now 115 degrees F, continuing to rise

Which selection below describes the actions the operators must take and the appropriate procedure to be implemented?

A. Stop any procedure in progress which could be reducing RCS inventory and initiate EOP-010 Guide 9, RCS Inventory Control.

B. Place DH Train B in service IAW OP-TM-212-901, EMERGENCY DHR OPERATIONS, to lower incore temperatures.

C. Throttle DH Train A flow using OP-TM-212-451, CONTROL OF DH TRAIN A FLOW AND TEMPERATURES, to lower incore temperatures.

D. Place DH-P-1A in pull-to-lock and raise RCS water level in accordance with OP-TM-220-555, FILL RCS FROM WASTE TRANSFER PUMP.

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The entry criteria for OP-TM-EOP-030, LOSS OF DECAY HEAT REMOVAL (Page 1, Rev 11) is exceeded when incore temperature is greater than 10F; (2) The examinee will have to determine that the RCS water level is below the DH pump limit of OP-TM-212-000, DECAY HEAT REMOVAL SYSTEM, Attachment 7.2 (Rev 24, Page 35) with the axis and regions blackened out. (3) OP-TM-220-555, FILL RCS FROM WASTE TRANSFER PUMP (Rev 5 Page 1) is used to fill makeup to the RCS from the Reactor Coolant Bleed Tanks when the Makeup System is in the shutdown mode.

A.	Stop any procedure in progress which could be reducing RCS inventory and initiate EOP-010 Guide 9, RCS Inventory Control.	INCORRECT: Plausible because stopping evolutions which are reducing RCS inventory is required by OP-TM-EOP-030 (Rev 11, Page 3). Incorrect because using EOP-010, Guide 9 is not the proper way to fulfill this step when the Makeup System is in the shutdown mode.
B.	Place DH Train B in service IAW OP-TM-212-901, EMERGENCY DHR OPERATIONS, to lower incore temperatures.	INCORRECT: Plausible because this would lower temperatures, which is the primary concern. Incorrect because the plant is operating at an RCS water level that is too low for DH pump operation for the current DH flow condition.
C.	Throttle DH-P-1A flow using OP-TM-212-451, CONTROL OF DH TRAIN A FLOW AND TEMPERATURES, to lower incore temperatures.	INCORRECT: Plausible because the examinee could believe that throttling the DH-P-1A flow to a higher flowrate will lower incore temperatures. Incorrect because the Decay Heat System is operating below the vortex curve and raising flow will cavitate DH-P-1A.
D.	Place DH-P-1A in pull-to-lock and raise RCS water level in accordance with OP-TM-220-555, FILL RCS FROM WASTE TRANSFER PUMP.	CORRECT ANSWER: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	025	2.1.25
	Importance Rating		4.2

K/A: Loss of Residual Heat Removal System: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question:	Question 76	
Technical Reference(s):	OP-TM-EOP-030, Rev 011	OP-TM-220-555, Rev 5
	OP-TM-212-000, Rev 24	
Proposed References to be provided to applicants during examination:	OP-TM-212-000, Attachment 7.2 with axis and regions blackened	
Learning Objective:	EOP030 - PCO-4	
Question Source:	Bank #	

	Modified Bank #	363675		
	New			
Question History:	N/A	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	b.5

Comments:

KA Match: This question matches the KA because the examinee will have to correctly interpret the figure to come up with the correct answer, and the question involves a loss of decay heat removal.

High Cog: This question is high cog because the examinee must analyze the question and correctly use the reference material to identify that the pump is running below its vortex limit.

SRO only: The question is SRO only because the examinee must know the content of the procedures in order to proceed to put the plant in a safe condition.

Initial plant conditions:
- Maintenance outage in progress.
- Reactor vessel head is removed
- DHR Train “A” aligned for RCS cooling.
- Incore thermocouple temperatures steady at 100 degrees F.

Event:
- Decay Heat Removal Pump discharge pressure and DHR system flow rate begin to oscillate excessively.
- Incore thermocouple temperatures are now 115 degrees F, continuing to rise.

Based on these conditions identify the ONE selection below that describes required operator actions, and the appropriate procedure to be implemented.

A. Stop any procedure in progress which could be reducing RCS inventory and initiate EOP-010 Guide 9, RCS Inventory Control.

B. Reduce DH Train A flow using OP-TM-212-451, Control of DH Train A Flow and temperatures, until the flow and pressure oscillations stop.

C. Place DH Train B in service IAW OP-TM-212-901, Emergency DHR Operations, and then vent DH-P-1A using OP-TM-212-553, Vent of DH-P-1A.

D. Place DH-P-1A in pull-to-lock and evacuate all personnel from the Reactor Building (RB) by actuating the RB Evacuation alarm IAW EOP-030, Loss of Decay Heat Removal.

Answer D

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

77 **ID: 2110669** **Points: 1.00**

Plant Conditions:

- Plant is operating at 88% power with ICS in auto

Sequence of Events:

- NI-5 fails high
- An electrical transient causes a loss of Vital Bus 'D' power
- The URO reports Reactor Power is 90% and steady

Which choice correctly completes the following statement describing the impact of these events on the E-plan and the required actions?

An Emergency Action Level threshold ___(1)___ be met and the CRS must direct a ___(2)___.

A. (1) has
(2) reactor trip

B. (1) has NOT
(2) reactor trip

C. (1) has
(2) controlled plant shutdown

D. (1) has NOT
(2) controlled plant shutdown

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) When NI-5 fails high, there will be a locked in trip signal. (2) When the loss of Vital Bus 'D' occurs, RPS channel D will lose power causing a separate trip signal. (3) With two trip signals locked in the examinee must identify that a reactor trip should have occurred; (4) When the trip does not occur an ATWS condition exists and the reactor operator must trip the reactor trip (OS-24, Section 5.5, Step 4, Rev 30, Page 25); (5) In accordance with EP-AA-1009 Addendum 3 (Rev 2, Page 2-5) there are 2 separate criteria for an Unusual Event: An automatic trip did not work, and the subsequent manual trip did work; or the manual trip did not work and the subsequent automatic trip did work. (5) For the question, the examinee must identify that the automatic trip did not work, which meets an EAL threshold.

A.	(1) has (2) reactor trip	CORRECT ANSWER: See above.
B.	(1) has NOT (2) reactor trip	INCORRECT: (1) Correct Answer. (2) Plausible due to the separate criteria of ATWS. The examinee may believe that ONLY an automatic trip ATWS may not meet the EAL threshold. Incorrect because it does meet the EAL threshold.
C.	(1) has (2) controlled plant shutdown	INCORRECT: (1) Plausible because the examinee could believe that the required action for not meeting the Technical Specification Requirements is a plant shutdown. Incorrect because OS-24 directs the reactor trip. (2) Correct Answer.
D.	(1) has NOT (2) controlled plant shutdownt	INCORRECT: (1) Plausible because the examinee could believe that the required action for not meeting the Technical Specification Requirements is a plant shutdown. Incorrect because OS-24 directs the reactor trip. (2) Plausible due to the separate criteria of ATWS. The examinee may believe that ONLY an automatic trip ATWS may not meet the EAL threshold. Incorrect because it does meet the EAL threshold.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	029	2.4.41
	Importance Rating		4.6

K/A: Anticipated Transient Without Scram: Knowledge of emergency action level thresholds and classification.

Proposed Question:	Question 77	
Technical Reference(s):	OS-24, Rev 30	EP-AA-1009, Addendum 3, Rev 2
	OP-TM-MAP-G0302, Rev 0	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EOP001-PCO-4	

Question Source:	Bank #		

	Modified Bank #	986736		
	New			
Question History:		Last NRC Exam:		Unmodified on 12-01

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

10 CFR Part 55 Content:	55.41	
	55.43	b.5

Comments:

KA Match: This question matches the KA because the examinee must identify that an ATWS has occurred and know if an EAL threshold has been met.

High Cog: This question is high cog because the examinee must analyze the conditions and identify that an ATWS has occurred and then determine the correct plant response.

SRO only: EAL thresholds are unique to the SRO position. They must assess the plant conditions and then have knowledge of the EAL procedure to know that an EAL threshold has been met.

Plant Conditions:

- The plant is operating at 88% power.

Sequence of Events:

- ICS is placed in Manual Control IAW OP-TM-621-471, ICS Manual Control.
- The Shutdown Bypass switch in the "A" RPS cabinet is taken to the "Bypass" position by an I&C Technician for maintenance.
- The Shutdown Bypass switch in the "A" RPS cabinet is taken to the "Normal" position by an I&C Technician.
- A Manual Channel Reset was **NOT** performed on the "A" RPS cabinet.
- The Shutdown Bypass switch in the "B" RPS cabinet is taken to the "Bypass" position by an I&C Technician for maintenance.
- The URO reports that Reactor Power is 90% and steady.

Given the above information, and assuming no other manual actions occurred, the CRS will direct the crew to ___(1)__, and the Tech Spec basis is to ___(2)___.

A. (1) lower power IAW 1102-4, Power Operations, due to a Nuclear Overpower condition
(2) prevent damage to the fuel cladding from rapid reactivity excursions

B. (1) lower power IAW 1102-4, Power Operations, due to a Nuclear Overpower condition
(2) prevent normal operation with part of the reactor protection system bypassed

C. (1) trip the Reactor and Main Turbine IAW OP-TM-EOP-001, Reactor Trip, due to an ATWS
(2) prevent damage to the fuel cladding from rapid reactivity excursions

D. (1) trip the Reactor and Main Turbine IAW OP-TM-EOP-001, Reactor Trip, due to an ATWS
(2) prevent normal operation with part of the reactor protection system bypassed

Answer D

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

78 **ID: 2095585** **Points: 1.00**

OP-TM-EOP-003, XHT is entered due to a unisolable steam line rupture of the 'A' OTSG.

Current Plant Parameters:

- Tave is 520^{0}F and lowering
- 'A' OTSG pressure is 150 psig and lowering
- RCS pressure is 1900 psig and lowering
- Cooldown rate is 90 ^{0}F/HR and lowering
- Tube-to-Shell Delta Temp - A OTSG (PPC Point C4015) is -20^{0}F
- Rule 3, XHT has been completed and Guide 9, RCS INVENTORY CONTROL has been initiated

Which of the following actions must be taken?

A. Minimize Subcooling Margin

B. Secure a Reactor Coolant Pump

C. Feed the 'B' OTSG with EFW at < 435 gpm

D. Open a MU-V-14 (MU Pump Suction from BWST)

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the candidate will have to know/assess the following conditions: (1) An unisolable steam line rupture will cause an uncontrollable cooldown of the RCS; (2) The operating crew will perform Rule 3, to attempt to isolate the OTSG and remove all feedwater sources from the affected OTSG; (3) Due the cooldown, the stem states that Tave falls below 525^{0}F, which in accordance with OP-TM-EOP-003 the operating crew would initiate to other procedures of OP-TM-EOP-010: Rule 5 (Rev 9, Page 3) and Rule 6; (4) Rule 5 provides actions for Tave falling below 525^{0}F regarding emergency boration, Rule 6 deals with Tavg falling below 525^{0}F which requires the crew to emergency borate using MU-V-14A or MU-V-14B (OP-TM-EOP-010, Rev 20, Page 11)

A.	Minimize Subcooling Margin	INCORRECT ANSWER: Plausible because this procedure is initiated in the same step by Rule 6. Incorrect, although Rule 6 will be initiated, the rule only requires minimizing subcooling margin under specific criteria, none of which are met.
B.	Secure a Reactor Coolant Pump	INCORRECT ANSWER: Plausible because there is a temperature requirement to secure a reactor coolant pump in OP-TM-EOP-003. Incorrect because that is required when RCS Tcold 407F and all 4 reactor coolant pumps are on.
C.	Feed the 'B' OTSG with EFW at < 435 gpm	INCORRECT ANSWER: Plausible because there is a limit to feed an OTSG if it were dry AND primary to secondary heat transfer were not established. Incorrect because primary to secondary heat transfer still exists.
D.	Open the a MU-V-14 (MU Pump Suction from BWST)	CORRECT ANSWER: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	BWE05	2.4.21
	Importance Rating		4.7

K/A: Steam Line Rupture - Excessive Heat Transfer: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling, and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question:	Question 78, SRO Question 3	
Technical Reference(s):	OP-TM-EOP-003, Rev 9	OP-TM-EOP-0031, Rev 4
	OP-TM-EOP-010, Rev 20	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	EOP DBIG PCO-2	
Question Source:	Bank #	
	Modified Bank #	

	New	X		
Question History:	None	Last NRC Exam:	None	

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

Comments:

KA Match: This question matches the KA because the examinee must use parameters and logic to assess the stem to determine the crew must emergency borate (Reactivity Control Safety Function).

High Cog: This question is High Cog because the candidate must assess the plant conditions and determine temperature thresholds for performing Rule 5. The examinee must determine that opening MU-V-14A or MU-V-14B is the correct action to take.

SRO Only: The question requires the examinee to assess plant conditions and to know the content of the EOP and the supplemental procedure to know what action that must be taken.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

79 **ID: 2087714** **Points: 1.00**

Event:

- Station Blackout

Current Plant Conditions:

- Neither 1D or 1E 4160V busses have been restored
- EF-P-1 is running
- OTSG Pressures:

	PSIG
'A' OTSG	1010
'B' OTSG	1010

- RCS Temperatures:

	Thot (F)	Tcold (F)
Loop A	597	543
Loop B	596	542

- RCS Pressure is 1950 psig
- Incore temperature is 563F

Which of the following identifies the status of Primary-to-Secondary Heat Transfer (PSHT) and the method of controlling RCS temperature in accordance with OP-TM-EOP-012, STATION BLACKOUT?

Primary-to-Secondary Heat Transfer (PSHT) is ___(1)___; operators must ___(2)___.

A. (1) occurring
(2) take local-manual control of the MS-V-4's to stabilize RCS temperature

B. (1) occurring
(2) throttle open the MS-V-4's from the control room to stabilize RCS temperature

C. (1) NOT occurring
(2) take local-manual control of the MS-V-4's to cooldown RCS temperature

D. (1) NOT occurring
(2) throttle open the MS-V-4's from the control room to cooldown RCS temperature

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) During a Station Blackout, none of the 4kV busses are available; (2) The crew must enter OP-TM-EOP-012, STATION BLACKOUT (Rev 5), which will stabilize the plant using emergency feedwater and the atmospheric dump valves; (3) The crew must interpret plant conditions to ensure natural circulation is being established which is determined by using OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES AND GRAPHS, Guide 10, NATURAL CIRCULATION (Rev 10, Page 23); (4) The crew must determine that the requirements for natural circulation are NOT met (RCS Thot - Tcold is greater than 50F); (5) Based on the OS-24, CONDUCT OF OPERATIONS DURING ABNORMAL AND EMERGENCY EVENTS, definition of Primary to Secondary heat transfer cannot be confirmed without natural circulation; (6) To establish natural circulation, the crew must lower OTSG pressure using the available means, which would be using the Atmospheric Dump Valves (ADV's) which are air operated valves; (7) In accordance with Step 3.21 of OP-TM-EOP-012 (Rev 5, Page 11) the crew must feed with EFW and open the MS-V-4's (ADV's) to maximize the cooldown; (7) The MS-V-4's are still available because they have Two Hour Instrument Air as the motive force to move them (TQ-TQ-104-C001, Rev 6, Page 14).

A.	(1) occurring (2) take local-manual control of the MS-V-4's to stabilize RCS temperature	Incorrect Answer: (1) Plausible because the examinee will have to assess whether heat transfer is adequate. Incorrect because it is not adequate. (2) Plausible because the operators could local manual control, but they still have air so they are operable from the control room.
B.	(1) occurring (2) throttle open the MS-V-4's from the control room to stabilize RCS temperature	Incorrect Answer: (1) Plausible because the examinee will have to assess whether heat transfer is adequate. Incorrect because it is not adequate. (2) Plausible because if primary to secondary heat transfer were adequate, the MS-V-4's would be used to stabilize RCS temperature.
C.	(1) NOT occurring (2) take local-manual control of the MS-V-4's to cooldown RCS temperature	Incorrect Answer: (1) Part 1 is correct. (2) Plausible because the operators could local manual control, but they still have air so they are operable from the control room.
D.	(1) NOT occurring (2) throttle open the MS-V-4's from the control room to cooldown RCS temperature	Correct Answer: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	055	EA2.01
	Importance Rating		3.7

K/A: Station Blackout: Ability to determine or interpret the following as they apply to a Station Blackout: Existing valve positioning on a loss of instrument air system.

Proposed Question:	Question 79	
Technical Reference(s):	OP-TM-EOP-012, Rev 5	OS-24, Rev 30
	OP-TM-EOP-010, Rev 20	TQ-TM-104-850-C001, Rev 6

Proposed References to be provided to applicants during examination:			None	
Learning Objective:	EOP-012-PCO-4			
Question Source:	Bank #	N/A		
	Modified Bank #	N/A		
	New	X		
Question History:	N/A	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	b.5

Comments:

KA Match: This question matches the KA because the examinee must know the MS-V-4's (Atmospheric Dump Valves) availability still exists with a loss if the instrument air system during a Station Blackout. The MS-V-4's will still operate from the control room with power to the controllers being supplied from the Backup Loaders (powered from VBB), and the motive force to operate the valves being the Backup two hour air bottles. During a station blackout, no instrument air compressor is available to operate these valves.

High Cog: This question is high cog because the examinee must assess plant conditions and make a decision on how to establish primary to secondary heat transfer.

SRO Only: This question is SRO only because the examinee must assess the plant conditions and make a decision to enter an event-specific sub-procedure (i.e. Feedwater available and Primary Secondary Heat Transfer not being adequate) and determine to re-establish plant control.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

80 **ID: 2110416** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto

Event:

- OP-TM-AOP-023, "A" DC SYSTEM FAILURE has been entered due to a complete loss of 'A' DC Distribution System

What Technical Specification action times must be met if the loss of 'A' DC Distribution System cannot be restored within the next 2 hours?

Place the Unit in HOT SHUTDOWN within the next ___(1)___, and COLD SHUTDOWN within an additional ___(2)___.

A. (1) 6 hours
(2) 24 hours

B. (1) 6 hours
(2) 36 hours

C. (1) 8 hours
(2) 24 hours

D. (1) 8 hours
(2) 36 hours

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-TM-AOP-0231, "A" DC SYSTEM FAILURE BASIS DOCUMENT (Rev 10 Page 1), standard tech spec 3.8.4.c states that 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 shudown within the next 6 hours. (2) The RNO of Step 4.13 intiates this shutdown if DC power cannont be restored within 2 hours.

A.	(1) 6 hours (2) 24 hours	INCORRECT: Plausible because these are the times for HOT SHUTDOWN and COLD SHUTDOWN in Technical Specification 3.0.1, which could be entered if the EDM cannot determin battery operability within 8 hours. Incorrect because in this case the standard tech spec applies because "A" DC is completely de-energized.
B.	(1) 6 hours (2) 36 hours	CORRECT: See above.
C.	(1) 8 hours (2) 24 hours	INCORRECT: Plausible because many Technical Specifications will put the Unit in an 8 hour Tech Spec. Incorrect because OP-TM-AOP-0231 says the Standard Tech Spec applies.
D.	(1) 8 hours (2) 36 hours	INCORRECT: Plausible because many Technical Specifications will put the Unit in an 8 hour Tech Spec. Incorrect because OP-TM-AOP-0231 says the Standard Tech Spec applies.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	058	2.2.22
	Importance Rating		4.0

K/A: Loss of DC Power: Knowledge of limiting conditions for operations and safety limits.

Proposed Question:	Question #80	
Technical Reference(s):	OP-TM-AOP-023, Rev 7	OP-TM-AOP-0231, Rev 10

Proposed References to be provided to applicants during examination:	None

Learning Objective:	734-GLO-14	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	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	b.2	

Comments:

KA Match: This question matches the KA because the examinee must know the technical specification limiting condition for operation for a loss of DC.

SRO Only: This question is SRO only because they must know how to apply the standard technical specifiction to a loss of DC power.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

81 **ID: 2095587** **Points: 1.00**

Plant conditions:

- Reactor power is 100% with ICS in full auto

Sequence of Events:

- Grid voltage was fluctuating for two minutes and then stabilized at 226 KV
- The operators observe on SS-1 that ONLY the 1091 line is connected to TMI
- The Transmission System Operator has informed the CRS that loss of a major switchyard is causing the problem and there is no current estimate for restoration
- The TMI Main Generator is operating at the limits of the capability curve

In accordance with 1107-11, TMI GRID OPERATIONS, what is the minimum action(s), if any, the crew must take to satisfy Technical Specification 3.7, UNIT ELECTRIC POWER SYSTEM?

A. Declare both Auxiliary Transformers inoperable and enter TS 3.0.1

B. No action required, all operability requirements of offsite sources are met

C. Start and load ONLY EG-Y-1A onto its bus until all operability requirements for offsite sources are met

D. Start and load BOTH EG-Y-1A AND EG-Y-1B onto their busses until all operability requirements for offsite sources are met

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Technical Specification 3.7.2.a.i (Amd 224, Page 3-42) requires two 230 kV lines to be in service. (2) Or in accordance with the exception 3.7.2e (Amd 278, Page 3-43) for only having one 230kV line continued reactor operation is permissible if one emergency diesel generator is started and run continuously until two lines have been restored. (3) In accordance with 1107-11 TMI GRID OPERATIONS, Section 3.4.8.2 (Rev 42, Page 34), the ES bus amperage is recorded, then the Emergency Diesel is started and loaded to 1 MW.

A.	Declare both Auxiliary Transformers inoperable and enter TS 3.0.1	INCORRECT: Plausible because if grid voltage had stabilized at < 223 kV this would have been the correct action (1107-11, Rev 42, page 18). Incorrect because voltage is not below 223 kV.
B.	No action required, all operability requirements of offsite sources are met	INCORRECT: Plausible because the examinee may believe that the 1091 line is sufficient. The 1091 and 1092 lines are capable of carrying 100% of TMI-1 generator output each. The examinee could believe that having one in service meets the requirements. Incorrect because Technical Specifications require two 230 kV lines to be in service or compensatory actions to be implemented.
C.	Start and load ONLY EG-Y-1A onto its bus until all operability requirements for offsite sources are met	CORRECT: See above.
D.	Start and load BOTH EG-Y-1A AND EG-Y-1B onto their busses until all operability requirements for offsite sources are met	INCORRECT: Plausible because the examinee could believe both emergency diesel generators must be started and loaded for stability of power to ES equipment. Incorrect because in accordance with 1107-11, only one emergency diesel generator is required to be started and loaded.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	077	AA2.09
	Importance Rating		3.9

K/A: Generator Voltage and Electric Grid Disturbances: Ability to determine and interpret the following as they apply to Generator Grid Disturbances: Operational status of emergency diesel generators

Proposed Question:	Question 81	
Technical Reference(s):	T.S 3.7, AMD 224	1107-11, Rev 42

Proposed References to be provided to applicants during examination: None

Learning Objective: 740-GLO-14

Question Source: Bank # 2049246

	Modified Bank #			
	New			
Question History:	N/A	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	b.2

Comments:

KA Match: This question matches the KA because the examinee must know the operational status of an emergency diesel generator for a grid disturbance.

High Cog: This question is high cog because the examinee must assess the grid voltage and the amount of 230 kV lines and determine the required actions.

SRO Only: This question is SRO only because the examinee must assess grid conditions and apply the required actions to a technical specification.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

82 **ID: 2105862** **Points: 1.00**

Which ONE of the following events would require a 4 hour report to the NRC Operations Center, in accordance with LS-AA-1020, EXELON REPORTABILITY MANUAL?

NOTE: Exclude any offsite notification which may be required for any of the conditions.

A. A Technical Specification deviation

B. An ECCS Discharge to the RCS as a result of a valid signal

C. An unanalyzed condition that significantly degrades plant safety is identified

D. An injury that results in a contaminated person transported to a medical facility

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with LS-AA-1020 (Rev 28, Page 5) any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation requires an ENS notification within 4 hours to the NRC.

A.	A Technical Specification deviation	INCORRECT: Plausible because a technical specification deviation does require a notification to the NRC. Incorrect because it requires an ENS notification within 1 hour in accordance with LS-AA-1020 (Rev 28, Page 4).
B.	An ECCS Discharge to the RCS as a result of a valid signal	CORRECT: See above.
C.	An unanalyzed condition that significantly degrades plant safety is identified	INCORRECT: Plausible because an unanalyzed condition does require a notifcation to the NRC. Incorrect because it requires an ENS notification within 8 hours in accordance within LS-AA-1020 (Rev 28, Page 4).
D.	An injury that results in a contaminated person transported to a medical facility	INCORRECT: Plausible because an injury that results in a contaminated person to a medical facility does require a notification to the NRC. Incorrect because it requires an ENS notification within 8 hours in accordance with LS-AA-1020 (Rev 28, Page 24).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	024	2.4.30
	Importance Rating		2.7

K/A: Emergency Boration: 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 transmission operator

Proposed Question:	Question 82	
Technical Reference(s):	LS-AA-1020, Rev 20	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	ADM08005	

Question Source:	Bank #	691285	
	Modified Bank #		
	New		

Question History:	18-01 Comp 3	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	b.1	

Comments:

KA Match: This question matches the KA because the examinee must know the notification requirement when the ECCS system (borates) into the RCS.

SRO only: This question is SRO only because the examinee must know the reportability requirements to the NRC. This is special knowledge exclusive to the SRO position.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

83 **ID: 2095158** **Points: 1.00**

In accordance with 1507-7, FUEL TRANSFER SYSTEMS OPERATING INSTRUCTIONS, which ONE of the following describes a VIOLATION of refueling administrative requirements?

A. Fuel moves in the Spent Fuel Pool are being supervised by an individual with an inactive SRO license.

B. An irradiated fuel assembly is being moved between Spent Fuel Pool locations with ESF ventilation shutdown, but available.

C. The fuel grapple FULL DOWN position has been determined by ZZ tape reading rather than by the Digital Fuel Elevation reading.

D. The Main Fuel Bridge has been left unattended with the fuel grapple at GRAPPLE UP DISENGAGED and the bridge de-energized.

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1505-7 (Rev 33, Page 4), irradiated fuel movements shall not be permitted in the Fuel Handling Building (FHB) unless an ESF Filtration System is in operation; (2) This administrative pre-caution is there to prevent violating Technical Specification 3.15.4 (Amd 3.15.4, Page 3-62e) which requires one train of ESF ventilation to be in service or moving fuel in the FHB is prohibited.

A.	Fuel moves in the Spent Fuel Pool are being supervised by an individual with an inactive SRO license.	INCORRECT: Plausible because an active license applies for core geometry changes but not spent fuel movement in the FHB
B.	An irradiated fuel assembly is being moved between Spent Fuel Pool locations with ESF ventilation shutdown, but available.	CORRECT: See above.
C.	The fuel grapple FULL DOWN position has been determined by ZZ tape reading rather than by the Digital Fuel Elevation reading.	INCORRECT: Plausible because both are methods for determining grapple position but ZZ tape meets the administrative requirement.
D.	The Main Fuel Bridge has been left unattended with the fuel grapple at GRAPPLE UP DISENGAGED and the bridge de-energized.	INCORRECT: Plausible because properly securing the bridge is an administrative requirement. However, this is an acceptable method per 1507-3.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	036	AA2.02
	Importance Rating		4.1

K/A: Fuel-Handling Incident: Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Occurence of a fuel handling incident.

Proposed Question:	Question 83
Technical Reference(s):	T.S 3.15.4 Amd 278
	1505-7, Rev 33
Proposed References to be provided to applicants during examination:	None
Learning Objective:	EOP DBIG - PCO-5
Question Source:	Bank # 575096
	Modified Bank #

	New			
Question History:	N/A	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	b.7	

Comments:

KA Match: This question matches the KA because the examinee must analyze all of the question options and determine which one is a violation of procedure and technical specification 3.15.4.1.

SRO only: This question is SRO only because the Senior Operator is responsible to ensure the technical specifications are met prior to moving fuel.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

84 **ID: 2095594** **Points: 1.00**

Sequence of Events:

- A reactor trip from 100% power has occurred due to a loss of off-site power (LOOP)
- Emergency Feedwater CANNOT be established to either OTSG
- Incore thermocouples are indicating 592F and rising slowly
- OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, has been entered
- SCM is 72F and stable

Which ONE of the following identifies the actions that must be taken?

A. Go to OP-TM-EOP-009, HPI COOLING, and establish HPI/PORV cooling and perform Rule 1 if SCM is lost.

B. Go to OP-TM-EOP-009, HPI COOLING and go to OP-TM-EOP-002, LOSS OF 25F SUBCOOLING MARGIN, if SCM is lost.

C. Continue with OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, and open the PORV and reclose it when SCM approaches 30F.

D. Continue with OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, and reduce OTSG Pressure so that secondary Tsat is 552-572F.

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) There are two major paths within OP-TM-EOP-004 based on the availability of feedwater. (2) For this question there is no feedwater available, on step 3.8 (Rev 12, Page 5) the crew will transition to OP-TM-EOP-009, HPI COOLING based on incore thermocouple temperature of 592°F and a SCM of 72F correlating to a saturation pressure of 2445 psig. (3) Within OP-TM-EOP-009 (Rev 8, Page 3), if SCM is lost (< 25F) the crew will perform Rule 1 (which can be performed from memory) then continue in OP-TM-EOP-009 without entering OP-TM-EOP-002, LOSS OF 25 ^{o}F SUBCOOLING MARGIN.

A.	Go to OP-TM-EOP-009, HPI COOLING, and establish HPI/PORV cooling and perform Rule 1 if SCM is lost	CORRECT: See above.
B.	Go to OP-TM-EOP-009, HPI COOLING and go to OP-TM-EOP-002, LOSS OF 25F SUBCOOLING MARGIN, if SCM is lost	INCORRECT: Plausible because the entry criteria for OP-TM-EOP-002 is a subcooling margin less than 25F. Incorrect because that procedure is written to combat RCS leaks, which is not occurring in this question.
C.	Continue with OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, and open the PORV and reclose it when SCM approaches 30F	INCORRECT: Plausible because this is a step in OP-TM-EOP-004. Incorrect because no feedwater is available and the crew must go to OP-TM-EOP-009.
D.	Continue with OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, and reduce OTSG Pressure so that secondary Tsat is 552-572F	INCORRECT: Plausible because this is a step in OP-TM-EOP-004. Incorrect because no feedwater is available and the crew must go to OP-TM-EOP-009.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	074	2.2.49
	Importance Rating		4.6

K/A: Inadequate Core Cooling: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Proposed Question:	Question 84	
Technical Reference(s):	OP-TM-EOP-004, Rev 12	OP-TM-EOP-009, Rev 8

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EOP004-PCO-4	

Question Source:	Bank #	860137		
	Modified Bank #			
	New			
Question History:	System Exam 8	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	b.5

Comments:

KA Match: This question matches the KA because Rule 1 requires the Reactor Coolant Pumps to be secured within 1 minute of Subcooling Margin and the crew has the ability to perform the rule with no reference. Rule 1 secures all running Reactor Coolant Pumps, initiates ES, and EFW.

High Cog: This question is high cog because the examinee must analyze plant conditions and determine that OP-TM-EOP-009 must be entered to establish HPI/PORV cooling.

SRO Only: This question is SRO only because the examinee must assess plant conditions and know the content of procedures in order to select a required course of action.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

85 **ID: 2094444** **Points: 1.00**

Plant Conditions:

- The reactor is at 83% power

Event:

- ICS RUNBACK alarm actuates
- Loop A main feedwater flow rapidly RISING
- Loop B main feedwater flow rapidly LOWERING
- Condensate flow indication is LOWERING

Which of the following procedures provides the required actions that mitigate these plant conditions?

A. OP-TM-MAP-H0101, ICS RUNBACK and 1102-4, POWER OPERATIONS

B. OP-TM-MAP-H0101, ICS RUNBACK and OP-TM-AOP-062, INOPERABLE ROD

C. OP-TM-AOP-010, LOSS OF THE 1A 4160V BUS and 1102-4, POWER OPERATIONS

D. OP-TM-MAP-K0106, POWER LOAD UNBALANCE and OP-TM-AOP-022, LOAD REJECTION

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) There are multiple conditions which will bring in the ICS RUNBACK alarm: Loss of a Reactor Coolant Pump (RCP), Loss of a Main Feedwater Pump, Asymmetric Rod fault, or RC Flow limit (OP-TM-MAP-H0101, Rev 2, Page 1); (2) Each abnormal condition causes a distinctly different Integrated Control System (ICS) response, for this condition an RCP in the 'B' RCS loop trips which causes the ICS to re-ratio Main Feedwater and the plant to runback to 665 MWe. For a RCP trip the ICS will raise FW in the loop with two running pumps, and lower the FW flow in the loop with one running RCP to maintain the primary to secondary heat balance all while running the plant back to the appropriate setpoint(TQ-TM-104-621-C001, Rev 10, Page 139-140); (3) After the examinee determines the cause of the abnormal condition, they will have to select the appropriate procedure to enter and mitigate the malfunction. For this specific malfunction, the crew must enter the ICS RUNBACK alarm response (OP-TM-MAP-H0101) and then determine that 1102-4, POWER OPERATION procedure includes crucial steps on how to operate the plant while it runs back.

A.	OP-TM-MAP-H0101, ICS RUNBACK and 1102-4, POWER OPERATIONS	CORRECT ANSWER: See Above.
B.	OP-TM-MAP-H0101, ICS RUNBACK and OP-TM-AOP-062, INOPERABLE ROD	INCORRECT: Plausible because these would be the procedures entered if a control rod had dropped into the core. In addition, if a dropped rod were to occur then condensate flow would lower and the ICS RUNBACK alarm would come in as it does in the stem of the question.
C.	OP-TM-AOP-010, LOSS OF THE 1A 4160V BUS and 1102-4, POWER OPERATIONS	INCORRECT: Plausible because these procedures would be entered if we lost the 1A 4160V Bus. A loss of the 1A 4160V bus would cause a Main Feedwater pump and a string of condensate to trip, also causing the ICS RUNBACK alarm and condensate flow to lower while the plant was running back. Incorrect because FW would not re-ratio.
D.	OP-TM-MAP-K0106, POWER LOAD UNBALANCE and OP-TM-AOP-022, LOAD REJECTION	INCORRECT: Plausible because these procedures would be entered if the examinee believes the stem indicates a power load unbalance (PLU). A PLU would cause condensate flow to lower, but not the ICS RUNBACK Alarm or FW re-ratio.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	A01	AA2.1
	Importance Rating		3.7

K/A: Plant Runback: Ability to determine and interpret the following as they apply to the (Plant Runback): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question:	Question #85	
Technical Reference(s):	OP-TM-MAP-H0101, Rev 2	
	TQ-TM-104-621-C001, Rev 10	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	621-GLO-11			
Question Source:	Bank #			
	Modified Bank #	357047		
	New			
Question History:	N/A	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	b.5	

Comments:

KA Match: This question matches the KA because the examinee must interpret the plant condition that caused the runback and select the appropriate procedure to mitigate the abnormal condition.

High Cog: This question is high cog because the examinee must assess the plant conditions, determine what the failed component is, and then select the correct procedure to mitigate the abnormal condition.

SRO Only: The question is SRO only because it requires the examinee to assess plant conditions and to know the content of the procedures in order to select a required course of action.

Plant Conditions:

- The reactor is at 83% power.

Event:

- ICS Runback alarm actuates.
- Loop A main feedwater flow rapidly RISING.
- Loop B main feedwater flow rapidly LOWERING.
- Condensate flow indication is LOWERING.

Based on these conditions, which one of the following equipment/component faults has occurred?

A. RC-P-1A tripped.

B. RC-P-1C tripped.

C. CO-P-2A tripped.

D. FW-P-1A tripped.

Answer B

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

86 **ID: 2107501** **Points: 1.00**

Plant Conditions:

- The reactor is operating at 100% power with ICS in full auto
- The following maintenance activities are in progress:
  - Repair of the air operator for sample isolation valve, CA-V-5A
    - CA-V-5A (Steam Generator A Feed Water Sample Valve) is closed
    - CA-V-4A (Steam Generator A Feed Water Sample Valve) is closed
  - Replacement of MU-V-16B HPI flow instrument, MU-FI-1127

Given the above information, which one of the following statements identifies:
(1) The maintenance activity that requires entry into a Tech Spec LCO, and
(2) a Technical Specification action time requirement associated with that maintenance activity?

A. (1) CA-V-5A maintenance
(2) Restore within 48 hours, or bring the reactor to HOT SHUTDOWN within the next 6 hours

B. (1) CA-V-5A maintenance
(2) Restore within 72 hours, or bring the reactor to HOT SHUTDOWN within the next 6 hours

C. (1) MU-FI-1127 maintenance
(2) Restore within 48 hours, or bring the reactor to HOT SHUTDOWN within the next 6 hours

D. (1) MU-FI-1127 maintenance
(2) Restore within 72 hours, or bring the reactor to HOT SHUTDOWN within the next 6 hours

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-TM-211-000, MAKEUP AND PURIFICATION SYSTEM (Rev 36 Page 15), HPI flow indication (MU-FI-1126, 1127, 1128, & 1129) is required to be operable for the associated HPI train to be operable if that HPI train is also lined up to provide seal injection. When an HPI flow indicator is not operable, ensure compliance with Technical Specification 3.3.2 and initiate a 72 hr TS time clock. (2) The MU pump discharge cross connect valve lineup may be swapped to align an inoperable HPI flow indicator with the HPI train not lined up to seal injection. An inoperable HPI flow indicator associated with an HPI train not lined up to provide seal injection does not affect HPI operability per Tech Spec 3.3

A.	(1) CA-V-5A maintenance (2) Restore within 48 hours, or bring the reactor to HOT SHUTDOWN within the next 6 hours	INCORRECT: (1) Plausible because CA-V-5A is a containment isolation valve. Incorrect because CA-V-4A and CA-V-5A are closed which is the required position for containment isolation. (2) Plausible because this is the time clock that would be in effect if both CA-V-4A and CA-V-5A were open.
B.	(1) CA-V-5A maintenance (2) Restore within 72 hours, or bring the reactor to HOT SHUTDOWN within 6 hours	INCORRECT: (1) Plausible because CA-V-5A is a containment isolation valve. Incorrect because CA-V-4A and CA-V-5A are closed which is the required position for containment isolation. (2) Plausible because this is a common Tech Spec time clock for many failed components. Incorrect because it does not apply to those component in this position.
C.	(1) MU-FI-1127 maintenance (2) Restore within 48 hours, or bring the reactor to HOT SHUTDOWN within the next 6 hours	INCORRECT: (1) Correct answer. (2) Plausible because this is the tech spec time clock if CA-V-4A and CA-V-5A were open.
D.	(1) MU-FI-1127 maintenance (2) Restore within 72 hours, or bring the reactor to HOT SHUTDOWN within the next 6 hours	CORRECT: See above

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	004	2.2.40
	Importance Rating		4.7

K/A: Chemical and Volume Control: Ability to apply Technical Specifications for a system

Proposed Question:	Question #86	
Technical Reference(s):	OP-TM-211-000, Rev 36	Tech Spec 3.3.2, ECR 14-00208
	Tech Spec 3.6 AMD 278	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	211-GLO-4			
Question Source:	Bank #	1685038		
	Modified Bank #			
	New			
Question History:	System Exam 14	Last NRC Exam:		

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

10 CFR Part 55 Content:	55.41	
	55.43	b.2

Comments:

KA Match: This question matches the KA because the examinee must have the ability to apply a technical specification of the Makeup System.

SRO only: This question is SRO only because the examinee must identify a condition that places the plant in a 72 hour technical specificaton time clock.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

87 **ID: 2095067** **Points: 1.00**

Plant Conditions:

- Reactor is 100% with ICS in full auto
- A drought and heat wave have resulted in rising river water temperature and lowering river water level
- River water temperature is 91F and rising

The following alarms come in:

- MAP C-1-4 IC CRD FILTER DP HI
- MAP C-1-3 IC CRD CLG OUTLET TEMP HI

Current Intermediate Closed Cooling Water (ICCW) parameters are:

- IC-10 FI CRD FLOW is 105 GPM and steady
- IC-9 TI TEMP FROM CRD reached 160F and is now lowering

Identify (1) the actions the CRS must direct, and (2) the controlling plant procedure.

A. (1) Reduce seal injection flow to ~ 32 gpm
(2) OP-TM-MAP-C0103, IC CRD CLG OUTLET TEMP HI

B. (1) Reduce seal injection flow to ~ 32 gpm
(2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING

C. (1) Cross-tie secondary river with nuclear river water
(2) OP-TM-MAP-C0103, IC CRD CLG OUTLET TEMP HI

D. (1) Cross-tie secondary river with nuclear river water
(2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) Due to the elevated river water temperature, all closed cooling water systems that use the Susquehanna River as a heat sink will be operating at an elevated temperature; (2) The examinee should identify an Intermediate Closed flow issue to the control rod drives when the 2 alarms come in; (3) With the given parameters and the alarms, OP-TM-MAP-C0103 (Rev 3, Page 1) will be the only procedure in which entry conditions were met; (4) On IC-9, TEMP FROM CRD, temperature is lowering due to the MU-V-1A/B interlock that isolates the letdown coolers, which is a major heat load on the ICCW system. (5) Lowering seal injection flow will maximize the amount of time before filling the pressurizer and requiring a reactor trip.

A.	(1) Reduce seal injection flow to ~ 32 gpm (2) OP-TM-MAP-C0103, IC CRD CLG OUTLET TEMP HI	CORRECT: See above
B.	(1) Reduce seal injection flow to ~ 32 gpm (2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING	INCORRECT: (1) Correct Answer. (2) Plausible because the examinee could believe that OP-TM-AOP-032 entry criteria have been met and actions must be given from that procedure.
C.	(1) Cross-tie secondary river with nuclear river water (2) OP-TM-MAP-C0103, IC CRD CLG OUTLET TEMP HI	CORRECT: (1) Plausible because cross tying secondary river with nuclear river water would give more cooling to ICCW components, incorrect because the criteria are not met to perform that procedure. (2) Correct answer.
D.	(1) Cross-tie secondary river with nuclear river water (2) OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING	INCORRECT: (1) Plausible because cross tying secondary river with nuclear river water would give more cooling to ICCW components, incorrect because the criteria are not met to perform that procedure. (2) Plausible because the examinee could believe that OP-TM-AOP-032 entry criteria have been met and actions must be given from that procedure.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	008	A2.03
	Importance Rating		3.2

K/A: Component Cooling Water: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (2) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High/Low CCW temperature.

Proposed Question:	Question #87	
Technical Reference(s):	OP-TM-MAP-C0103, Rev 3	
	OP-TM-AOP-032, Rev 5	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	AOP-032-PCO-4	

Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	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	b.5

Comments:

KA Match: This question matches the KA because the examinee has to predict the impact of a high temperature on the Intermediate Closed Cooling Water system. Based on those predictions know the content of the procedure to mitigate the consequence of the high temperature.

High Cog: This question is high cog because the examinee must know why intermediate closed cooling water peaked and 160F and is now lowering, and that in response to that, the overall plant response.

SRO Only: This question is SRO only because it requires the examinee to assess the plant conditions and know the content of procedures in order to select a required course of action. This correct answer is not a major AOP, and more than entry criteria and overall mitigation strategy is required to be known to answer the question correctly.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

88 **ID: 2095203** **Points: 1.00**

Sequence of Events:

- An In-service Test (IST) performed on AH-E-1A, Reactor Building Air Handling Unit 1A, showed an air flow rate of 24530 cfm, which is less than the minimum flow rate allowed by the IST Acceptance Criteria.
- Subsequent testing performed under a complex troubleshooting plan had shown:
  - Fan performance appeared to have leveled out at the lower flow rate.
  - Engineering evaluations concluded that there will be no further degradation of flow rate.

(1) AH-E-1 must be declared __(1)__.
(2) What, if any, Technical Specification time clock must be entered?

A. (1) Degraded but Operable
(2) No Technical Specification time clock

B. (1) Degraded but Operable
(2) 30 day AP 1038 timeclock for Three Train Safe Shutdown Systems Degraded

C. (1) Inoperable
(2) No Technical Specification time clock

D. (1) Inoperable
(2) 7 day time clock

Answer: D

Answer Explanation

To answer this question correctly, the examinee must know: (1) From OP-AA-108-104, TECHNICAL SPECIFICATION COMPLIANCE, the definition of INOPERABLE is when an SSC is considered to be INOPERABLE when it is not capable of meeting all of the requirements of the Technical Specification, ATR, TRM, ISFSI, or ODCM definition for OPERABILITY; (2) Even though Engineering has determined that the fan will not degrade further, this does not make the pump anything other than INOPERABLE because it is still lower than the minimum flow rate allowed by technical specifications. (2) Technical Specification 3.3.2 allows one train of the following systems to be out of service for 72 hours: Makeup and Purification, Decay Heat, RB Emergency Cooling, RB Spray, BWST level instrumentation, or cooling water systems for 72 hours; one of the exceptions to this is that one reactor building cooling fan and associated cooling unit shall be permitted to be out of service for seven days (Technical Specification 3.3.3). (3) AH-E-1A is powered from 1A ES MCC, AH-E-1B is powered from 1B ES MCC, and AH-E-1C is powered from 1C ESV (1107-5, Rev 155 Page 13). 1C ESV is normally powered from the 1S 480V bus. (4) AH-E-1A is the 'A' side ES cooling fan, and AH-E-1B and AH-E-1C are 'B' side ES fans. (5) In accordance with OP-TM-823-250, AH-E-1A, AH-E-1B, AH-E-1C COOLING FAN FLOW TESTING, Attachment 7.1 Step 4, the miniumum flowrate is 25,000 cfm. (6) The UFSAR Section 6.3 (Rev 21, Page 6.3-2) assumes 25,000 cfm for each unit in accident analysis.

A.	(1) Degraded but Operable (2) No Technical Specification time clock	INCORRECT: (1) Plausible because the term degraded only applies when the equipment is OPERABLE. The examinee may believe that the equipment is not INOPERABLE because the SSC will not degrade any further, but since it is below the technical specification value that the SSC is degraded but operable. (2) Plausible because no technical specification would be entered if the equipment was degraded but operable.
B.	(1) Degraded but Operable (2) 30 day AP 1038 timeclock for Three Train Safe Shutdown Systems Degraded	INCORRECT: (1) Plausible because the term degraded only applies when the equipment is OPERABLE. The examinee may believe that the equipment is not INOPERABLE because the SSC will not degrade any further, but since it is below the technical specification value that the SSC is degraded but operable. (2) Plausible because some systems have a 30 day AP 1038 timeclock. AP 1038 is considered part of Technical Specifications. Incorrect because Reactor Building Ventilation is not one of those systems.
C.	(1) Inoperable (2) No Technical Specification time clock	INCORRECT: (1) Correct Answer. (2) Plausible because if AH-E-1B or AH-E-1C were inoperable, then no technical specification time clock would be entered. In accordance with OP-TM-823-000, REACTOR BUILDING HEATING AND VENTILATION SYSTEM (Rev 10 Page 7), with two of three RB air coolers operable, one from each ES power train, the requirements of TS 3.3.1.3 are satisfied.
D.	(1) Inoperable (2) 7 day time clock	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	022	2.2.12
	Importance Rating		4.1

K/A: Containment Cooling: Knowledge of surveillance procedures				
Proposed Question:	Question #88			
Technical Reference(s):	Technical Specification 3.3.2 and 3.3.3, ECR 14-00208		UFSAR Section 6.3, Rev 21	
	OP-AA-108-104, Rev 2			
Proposed References to be provided to applicants during examination:			None	
Learning Objective:	108104-APCO-1			
Question Source:	Bank #			
	Modified Bank #	1700778		
	New			
Question History:		Last NRC Exam:		Unmodified on 16-01

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	b.2

Comments:

KA Match: This question matches the KA because the examinee must make the determination that AH-E-1A is inoperable even though the flow rate will not degrade any further and the engineer has determined no further degradation will occur. In addition the examinee must know the AH-E-1 exception to Technical Specification 3.3.2 for when one train of the system is out.

High Cog: This question is high cog because the examinee must analyze the surveillance data and determine the status of AH-E-1A. In addition, the examinee must apply Technical Specification 3.3.2 and 3.3.3.

SRO only: This question is SRO only because the examinee must apply the Technical Specification required actions and understand the surveillance requirements of the system.

Sequence of Events:

- An In-service Test (IST) performed on Nuclear River (NR) system pump NR-P-1A showed a flow rate of 6234 gpm, which is less than the minimum flow rate allowed by Technical Specification.
- Subsequent testing performed under a complex troubleshooting plan had shown:
  - Pump performance appeared to have leveled out at the lower flow rate, but still greater than the design minimum ASME value.
  - Engineering evaluations concluded that there would be no further degradation of flow rate.

After the troubleshooting plan is complete, NR-P-1A must be declared ___(2)___.

A. Operable

B. Degraded

C. Inoperable

D. Unavailable

Answer: C

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

89 **ID: 2088339** **Points: 1.00**

Plant Conditions:

- OTSG Tube Leakage exists in both the “A” and “B” OTSGs.
- OP-TM-EOP-001, Reactor Trip IMA’s have been completed.
- The crew transitions to OP-TM-EOP-005, OTSG TUBE LEAKAGE.
- Radiation Monitors read as follows:

  - RM-A-5/15, Condenser Vacuum Pump Exhaust: 5.9 E3 CPM, in Hi Alarm, rising.
  - RM-A-5 HI, Condenser Vacuum Pump Exhaust: 3.7 E3 CPM, in Alert, rising.
  - RM-G-26, Main Steam Line From OTSG “A”: 4.4 E3 CPM, in Hi Alarm, rising.
  - RM-G-27, Main Steam Line From OTSG “B”: 5.6 E4 CPM, in Hi Alarm, rising.

- OTSG “A”: Pressure is 1010 psig and steady; Level is 35% and steady.
- OTSG “B” Pressure is 985 psig and steady; Level is 50% and steady.
- RCS Pressure is 980 psig and lowering.

Event:

- A Loss of Offsite Power (LOOP) occurs.
- Offsite dose assessor reports that projected offsite integrated dose is 500 mRem TEDE.

(1) Which OTSG(s) must be isolated?
(2) How will cooling be provided to the RCS?

A. (1) Isolate OTSG 'B', only
(2) Throttle open 'A' OTSG ADV, in accordance with Guide 12, RCS STABILIZATION

B. (1) Isolate OTSG 'B', only
(2) Throttle open 'A' OTSG TBVs, in accordance with Guide 12, RCS STABILIZATION

C. (1) Isolate both OTSGs
(2) Go to OP-TM-EOP-009, HPI COOLING

D. (1) Isolate both OTSGs
(2) Cycle the PORV to maintain SCM in accordance with OP-TM-EOP-005

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The 'B' OTSG tube leak is worse than the 'A' OTSG based on radiation monitor readings and OTSG levels; (2) When offsite dose approaches 500 mRem TEDE (OP-TM-EOP-005, Rev 9, page 17), the operating crew must isolate the most affected OTSG when RCS pressure is less than 1000 psig; (3) Due to the LOOP, all Circulating Water Pumps trip which will cause a Condenser Vacuum Interlock (CVI). When CVI actuates, the Turbine Bypass Valves (MS-V-3A-F) latch closed. (4) With the MS-V-3's closed, the only way to control OTSG pressure is with the Atmospheric Dump Valves (MS-V-4's) on the OTSG which is NOT isolated.

A.	(1) Isolate OTSG 'B', only (2) Throttle open 'A' OTSG ADV, in accordance with Guide 12, RCS STABILIZATION	CORRECT ANSWER: The examinee will have to diagnose that the 'B' OTSG has a worse tube leak and must be the OTSG that is isolated. In addition, the examinee must know that MS-V-4A is the only way to control steam pressure on the 'A' OTSG due to the Loss of Offsite Power.
B.	(1) Isolate OTSG 'B', only (2) Throttle open 'A' OTSG TBVs, in accordance with Guide 12, RCS STABILIZATION	INCORRECT: (1) Correct Answer. (2) Incorrect because the LOOP actuated CVI. Plausible if the examinee does not understand that on a loss of offsite power that the condenser is unavailable to steam to.
C.	(1) Isolate both OTSGs (2) Go to OP-TM-EOP-009, HPI COOLING.	INCORRECT: (1) Incorrect because the procedure step 3.35 only directs isolation of the most affected OTSG. Plausible if the examinee believes both OTSG's can be isolated with RCS pressure is < 1000 psig. (2) Incorrect because both OTSGs are not being isolated. Both OTSGs would be isolated if both were challenging isolation criteria of >85% in the operating range. Plausible because Step 3.34 directs going to OP-TM-EOP-009 if both OTSGs are isolated.
D.	(1) Isolate both OTSGs (2) Cycle the PORV to maintain SCM in accordance with OP-TM-EOP-005	CORRECT: (1) Incorrect because the procedure step 3.35 only directs isolation of the most affected OTSG. Plausible if the examinee believes both OTSG's can be isolated with RCS pressure is < 1000 psig. (2) Incorrect there is no need to cycle the PORV because pressure is already below 100 psig. Plausible if the examinee believes lowering RCS pressure is the man priority at this point.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	039	A2.03	
	Importance Rating	3.7		

K/A: Main and Reheat Steam: Ability to (a) predict the impacts of the following on the MRSS; and (b) based on the predictions, use procedures to correct, control, or mitigate the operations: Indications and alarms for main steam and area radiation monitors (during SGTR).

Proposed Question:	Question #89	
Technical Reference(s):	OP-TM-EOP-005, Rev 9	

Proposed References to be provided to applicants during examination:			None	
Learning Objective:	EOP005 PCO4			
Question Source:	Bank #	1110826		
	Modified Bank #			
	New			
Question History:	18-01 Comp 3	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	b.5

Comments:

KA Match: The question matches the KA because the examinee must predict the impact of the isolation of the 'B' OTSG will have on the plant. The examinee has to have knowledge of a decision point in OP-TM-EOP-005 to assess the radiation monitor readings and decide with OTSG must be isolated.

High Cog: The question is High Cog because the examinee must know the isolation criteria for an OTSG and that the condenser is not available to steam too, in addition to assessing which OTSG must be isolated.

SRO Only: This is an SRO only question because the Senior Operator examinee must have knowledge of a diagnostic step in OP-TM-EOP-005 involves a transition to an event-specific sub procedure (Attachment 1 of EOP-005). The examinee must identify that the 'B' OTSG has a worse tube leak than the 'A' OTSG and then assess how to isolate the 'B' OTSG with a LOOP.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

90 **ID: 2106629** **Points: 1.00**

Plant Conditions:

- The reactor is at 8% power and holding following isolation of an Instrument Air (IA) leak.
- The section of the IA header that feeds AH-E-29A, Diesel Generator Room “A” Fan Damper, has been isolated and completely depressurized to facilitate repair of a tubing failure.
- Operators have performed OP-TM-861-910, Emergency Ventilation of EG-Y-1A Room.

Given the above information, which one of the following identifies:
(1) The current operability status of EG-Y-1A, and
(2) The requirements with respect to Fire Door D107, EG-Y-1A Room Door?

A. (1) OPERABLE
(2) Compensatory actions for a blocked open Fire Door are required

B. (1) OPERABLE
(2) The Fire Door must remain closed and latched except for temporary passage

C. (1) INOPERABLE
(2) Compensatory actions for a blocked open Fire Door are required

D. (1) INOPERABLE
(2) The Fire Door must remain closed and latched except for temporary passage

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The dampers on AH-E-29A & B will fail closed on a loss of instrument air (OP-TM-AOP-028, LOSS OF INSTRUMENT AIR, Rev 9 Page 31). (2) The Emergency Diesel could still operate, but would be declared inoperable and Technical Specification 3.7.2 would be entered (OP-TM-AOP-0281, LOSS OF INSTRUMENT AIR BASIS DOCUMENT, Rev 7 Page 4). (3) In accordance with OP-TM-861-910, EMERGENCY VENTILATION OF EG-Y-1A ROOM (Rev 2 Page 2) the crew must notify the shift management to implement FSI's and applicable compensatory action due to fire doors being blocked open.

A.	(1) OPERABLE (2) Compensatory actions for a blocked open Fire Door are required	INCORRECT: (1) Plausible because compensatory actions are being taken to address the Emergency Diesel Ventilation. Incorrect because the Emergency Diesel is still inoperable. (2) Correct answer.
B.	(1) OPERABLE (2) The Fire Door must remain closed and latched except for temporary passage	INCORRECT: (1) Plausible if the examinee does not know that AH-E-29A is required for Emergency Diesel Operability. (2) Plausible because fire doors are normally operated like this. Incorrect because OP-TM-861-910 allows for them to be open with FSI's.
C.	(1) INOPERABLE (2) Compensatory actions for a blocked open Fire Door are required	CORRECT: See above.
D.	(1) INOPERABLE (2) The Fire Door must remain closed and latched except for temporary passage	INCORRECT: (1) Correct answer. (2) Plausible because fire doors are normally operated like this. Incorrect because OP-TM-861-910 allows for them to be open with FSI's.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	078	2.2.44
	Importance Rating		4.6

K/A: Instrument Air: Ability to determine operability and/or availability of safety related equipment.

Proposed Question:	Question #90	
Technical Reference(s):	OP-TM-AOP-028, Rev 9	OP-TM-AOP-0281, Rev 7
	OP-TM-861-910, Rev 2	
Proposed References to be provided to applicants during examination:	None	
Learning Objective:	861-GLO-14	

Question Source:	Bank #	897077	
	Modified Bank #		

	New			
Question History:	Simulator Exam 6	Last NRC Exam:	12-01	

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	b.1

Comments:

KA Match: This question matches the KA because the examinee must be able to determine the operability of an Emergency Diesel Generator when its associated instrument air is lost.

High Cog: This question is high cog because the examinee must determine that the loss of instrument air to AH-E-29A makes the damper fail closed. The examinee must use that knowledge to determine the Emergency Diesel Generator is inoperable.

SRO only: This question is SRO only because the examinee must know the administration of the fire protection program, specifically compensatory actions for a fire door that must be open.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

91 **ID: 2096110** **Points: 1.00**

Plant Conditions:

- Reactor power is 100% with ICS in full auto
- Pressurizer Level LT-1 is selected as the controlling channel

EVENT:

- A leak develops on the reference leg of LT-1 causing pressurizer level to change 60 inches over 10 seconds

Following the event, assuming no operator action, MU-V-17, Pressurizer Level Control Valve will ___(1)___, and ___(2)___ will be entered to mitigate the malfunction?

A. (1) open
(2) OP-TM-MAP-G0205, PZR LEVEL HI/LO

B. (1) close
(2) OP-TM-MAP-G0205, PZR LEVEL HI/LO

C. (1) open
(2) 1105-6, NON-NUCLEAR INSTRUMENTATION AND CONTROLS

D. (1) close
(2) 1105-6, NON-NUCLEAR INSTRUMENTATION AND CONTROLS

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The pressurizer uses a wet reference leg, which means a 0 DP when the pressurizer is full; (2) Normal Level setpoint is 220 inches; (3) As the reference leg loses water, the DP would lower which would be seen as a pressurizer level rise on the PZR Level indication; (4) Since indicated pressurizer level is rising, MU-V-17, Pressurizer Level Control valve, will close to maintain level at setpoint of 220 inches; (5) Actual pressurizer would lower due to letdown flow remaining constant and makeup flow lowering due to MU-V-17 closing; (6) The pressurizer level instruments do not SASS on any failure. (6) OP-TM-MAP-G0205 PZR LEVEL HI/LO (Rev 3 Page 1) directs the crew to place MU-V-17 in hand and control Pressurizer level. The alarm response also directs the crew to select a valid level signal.

A.	(1) open (2) OP-TM-MAP-G0205, PZR LEVEL HI/LO	INCORRECT: (1) Plausible if the examinee determines that indicated Pressurizer level will be lower. Incorrect because indicated pressurizer level will be higher and MU-V-17 will close. (2) Correct answer.
B.	(1) close (2) OP-TM-MAP-G0205, PZR LEVEL HI/LO	CORRECT: (1) Correct answer. (2) Correct answer.
C.	(1) open (2) 1105-6, NON-NUCLEAR INSTRUMENTATION AND CONTROLS	INCORRECT: (1) Correct answer. (2) Plausible because some systems malfunctions/abnormal conditions are covered by the systems operating procedure. Incorrect because this malfunction is covered by the alarm response.
D.	(1) close (2) 1105-6, NON-NUCLEAR INSTRUMENTATION AND CONTROLS	INCORRECT: (1) Since indicated level is rising, MU-V-17 will close. This is plausible if the examinee does not know which way indicated level will go. (2) Plausible because some systems malfunctions/abnormal conditions are covered by the systems operating procedure. Incorrect because this malfunction is covered by the alarm response.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #	011 A2.03	
Importance Rating		3.8

K/A: Pressurizer Level Control: Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of PZR level

Proposed Question: Question #91

Technical Reference(s): OP-TM-MAP-G0205, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: 624-GLO-5

Question Source: Bank #

Modified Bank # 862259

New

Question History: Unmod on System Exam 11 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 b.5

Comments:
KA Match: The K/A is matched because the examinee must demonstrate knowledge of the operational implications of PZR reference leg leak abnormalities as they apply to Pressurizer Level Control Malfunctions. In addition, the examinee must know the procedure which will be used to mitigate the malfunction.

High Cog: The question is at the Comprehension/Analysis cognitive level because the examinee must demonstrate understanding of how the Pressurizer Level control system detects level, and then determine how a failure of a reference leg effects actual pressurizer level to correctly answer the question. In addition, the examinee must take into account how an erroneous pressurizer level effects the pressurizer makeup valve which would close and cause actual level to lower.

SRO Only: The question requires the examinee to assess the plant conditions and know what procedure which must be used to mitigate the malfunction. The procedures are not major EOPs or AOPs

Plant conditions:

- 100% power.
- Pressurizer Level LT-1 is selected as the controlling channel.

Event:

- A leak develops on the reference leg of LT-1.

Following the event, the trend of indicated (LT-1) Pressurizer level will ___(1)___ and actual Pressurizer level will ___(2)___.

A. (1) rise
(2) lower

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

C. (1) lower
(2) rise

D. (1) lower
(2) remain the same

Answer A

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

92 **ID: 2105983** **Points: 1.00**

Plant Conditions:

- Reactor power is being lowered from 100% due to a steam leak in the reactor building
- The reactor is tripped when RB pressure exceeds 2 psig

Event:

- Loss of Offsite Power (LOOP)

Conditions one minute after the Loss of Offsite Power:

- 1D and 1E 4160V busses are being powered by their Emergency Diesel Generators
- MAP H-1-8 alarm ICS/NNI POWER LOST is illuminated
- Incore temperatures are 560F and rising with only EF-P-1 available
- RM-A-5 HI, Condenser Vacuum Pump Radiation Monitor is in ALARM
- Pressurizer Level is lowering at 1 inch per minute

In accordance with OS-24, CONDUCT OF OPERATIONS DURING ABNORMAL AND EMERGENCY EVENTS, which of the following procedures has the highest priority based on current conditions?

A. OP-TM-AOP-020, LOSS OF STATION POWER

B. OP-TM-AOP-027, LOSS OF ATA OR ICS AUTO POWER

C. OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER

D. OP-TM-EOP-005, OTSG TUBE LEAKAGE

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OS-24 (Rev 31 Page 5), OTSG Tube leakage can be confirmed if the plant is post trip and a valid unexpected alarm from offgas or the steam line radiation monitors come in. (2) Prioritization of EOP actions is in the order that follows: Station Blackout, Loss of Subcooling Margin, Excessive Primary-to-Secondary Heat Transfer, Lack of Primary-to-Secondary Heat Transfer.

A.	OP-TM-AOP-020, LOSS OF STATION POWER	INCORRECT: Plausible because the entry criteria for this procedure are met. Incorrect because it does not have the highest priority based on current plant conditions.
B.	OP-TM-AOP-027, LOSS OF ATA OR ICS AUTO POWER	INCORRECT: Plausible because the entry criteria for this procedure are met. Incorrect because it does not have the highest priority based on current plant conditions.
C.	OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER	INCORRECT: Plausible because the entry criteria for this procedure are met. Incorrect because it does not have the highest priority based on current plant conditions. The crew must wait for natural circulation to build in prior to meeting the entry criteria for OP-TM-EOP-004.
D.	OP-TM-EOP-005, OTSG TUBE LEAKAGE	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			2
	Group #			2
	K/A #	055	2.4.8	
	Importance Rating			4.5

K/A: Condenser Air Removal: Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Proposed Question:	Question 92	
Technical Reference(s):	OS-24, Rev 31	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EOP005-PCO-2	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	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	b.5	

Comments:

KA Match: This question matches the KA because the examinee must know how AOP's and EOP's are used in conjunction with each other. The examinee must make the determination that the EOP should be entered from the radiation monitor on the that monitors the Condenser Air Removal System.

High Cog: This question is high cog because the examinee must assess the plant conditions and then choose the correct procedure to enter.

SRO only: This question is SRO only because the examinee must assess the plant conditions and choose the highest priority (hierarchy) procedure to enter.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

93 **ID: 2096148** **Points: 1.00**

Plant conditions:

- An approved release is in progress from Waste Gas Decay Tank (WDG-T-1A)
- All radiation monitors are operable

Event:

- All power is lost to RM-A-7, Gaseous Waste Discharge Tank Monitor

Which one of the choices is correct regarding the automatic response of WDG-V-47 (Waste Gas Release Stop and Control Valve) and the subsequent actions required and/or allowed per the ODCM?

(Assume all compensatory actions are complete in accordance with the ODCM table 2.1-2)

A. WDG-V-47 remains open. The release must be manually terminated until RM-A-7 is restored to operable.

B. WDG-V-47 remains open. The release may continue with RM-A-7 inoperable.

C. WDG-V-47 automatically closes. The release may be resumed with RM-A-7 inoperable.

D. WDG-V-47 automatically closes. The release must be terminated until RM-A-7 is restored to operable.

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with the MAIN ANNUNCIATOR PANEL C (MAP C, Rev 102A, C-3-1 alarm response, Rev 17) will alarm due to a loss of voltage to RM-A-7; (2) When that occurs, the crew must ensure tech spec and/or ODCM requirements are met while the rad monitor is out of service and ensure radioactive releases to the environment are terminated and not resumed until appropriate monitors are operable; (3) In accordance with CY-TM-170-300, OFFSITE DOSE CALCULATION MANUAL (ODCM) (Rev 5, Pages 24 and 27), 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 automatically closing WDG-V-47.

A.	WDG-V-47 remains open. The release must be manually terminated until RM-A-7 is restored to operable.	INCORRECT: Plausible because the examinee could believe that only RM-A-8 (Aux and Fuel Handling Building Gas Are Rad Monitor) closes WDG-V-47. In addition, the examinee could believe when power is lost to RM-A-7 that the interlock will not actuate. Incorrect because when RM-A-7 loses power, WDG-V-47 will go closed.
B.	WDG-V-47 remains open. The release may continue with RM-A-7 inoperable.	INCORRECT: Plausible because the examinee could believe that only RM-A-8 (Aux and Fuel Handling Building Gas Are Rad Monitor) closes WDG-V-47. In addition, the examinee could believe when power is lost to RM-A-7 that the interlock will not actuate. Incorrect because when RM-A-7 loses power, WDG-V-47 will go closed.
C.	WDG-V-47 automatically closes. The release may be resumed with RM-A-7 inoperable.	CORRECT: See above.
D.	WDG-V-47 automatically closes. The release must be terminated until RM-A-7 is restored to operable.	INCORRECT: Plausible because WDG-V-47 does isolate the release. In addition the examinee could believe that RM-A-7 is required for a waste gas release. Incorrect because as long as other ODCM requirements are met, the release can continue as long as RM-A-8 is operable.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	071	A2.02
	Importance Rating		2.9

K/A: Waste Gas Disposal: Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System; and (2) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Use of waste gas release monitors, radiation, gas flow and totalizer.

Proposed Question:	Question #91	
Technical Reference(s):	MAP C, Rev 102A	CY-TM-170-300, Rev 5

Proposed References to be provided to applicants during examination:	None

Learning Objective:	231-GLO-8			
Question Source:	Bank #	537973		
	Modified Bank #			
	New			
Question History:	Comp 1	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	b.1

Comments:

KA Match: This question matches the KA because the examinee must be able to predict the status of the waste gas release when power to RM-A-7 is lost. The malfunction is the loss of a waste gas release radiation monitor.

SRO only: The question is SRO only because the examinee must the condition of the facility license (ODCM) which govern the required equipment for a gaseous waste release.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

94 **ID: 2087602** **Points: 1.00**

Plant Conditions:

- Reactor is in a refueling shutdown condition
- All fuel is in the spent fuel pool

Sequence of Events:

- Two hours into the shift, a licensed CRO slips and becomes incapacitated
- The incapacitated CRO is 1 of 3 licensed CROs assigned to the shift
- The following personnel help transport the CRO to the hospital:
  - 1 of the 2 (not including the Shift Manager) available SRO assigned to the shift (both qualified STA)
  - 1 of the 4 available AO assigned to the shift
- Maintenance personnel have assumed the duties of Fire Brigade Team Leader
- Other than the Shift Manager, no additional operators on site

In accordance with OP-TM-112-101-1002, SHIFT STAFFING REQUIREMENTS, and assuming all watchstanders are fully qualified, which of the following is the action to take, if any, to ensure required minimum unit staffing?

A. Another AO must be called in to arrive within two hours.

B. Another RO must be called in to arrive within two hours.

C. Another SRO must be called in to arrive within two hours.

D. No action is required because minimum staffing levels are still met for all positions.

Answer: A

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) The reactor is in a refueling shutdown conditions, which in accordance with Technical Specification 1.2.6 (Page 1-1, AMD 278) which means reactor coolant temperature is no more than 140F; (2) According to Technical Specification table 6.2-1 (Page 6-2, AMD 219), the minimum shift crew composition is: 1 SRO, 1 RO, and 1 Non-Licensed Auxiliary Operator; (3) But due to other regulatory obligations, site procedure OP-TM-112-101-1002, SHIFT STAFFING REQUIREMENTS (Rev 13, Page 2) four auxiliary operators are required to be assigned to shift.

A.	Another AO must be called in to arrive within two hours.	CORRECT: See above.
B.	Another RO must be called in to arrive within two hours.	INCORRECT: Plausible because above 200F three CRO's are required. Incorrect because below 200F two CRO's are required.
C.	Another SRO must be called in to arrive within two hours.	INCORRECT: Plausible because a normal shift consists of 2 SRO's and a Shift Manager. Incorrect because only 1 SRO is required less than 200F
D.	No action is required because minimum staffing levels are still met for all positions.	INCORRECT: Plausible because the examinee could believe that 4 AO's are not required when less than 200F. All other staffing requirements lower when less than 200F, but 4 AO's are required in both conditions.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.1.5	
	Importance Rating		3.9

K/A: Ability to use procedures relate to shift staffing, such as minimum crew complement, overtime limitations, etc.

Proposed Question:	Question 94	
Technical Reference(s):	T.S Table 6.2-1, AMD 217	
	OP-TM-112-101-1002, Rev 13	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	Prewatch DBIG APCO-1	

Question Source:	Bank #		
	Modified Bank #	1147373	
	New		

Question History:	N/A	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	b.1	

Comments:

KA Match: This question matches the KA because the examinee must know requirements in the operations procedure to shift staffing.

SRO-only: This question is SRO only because they are responsible to ensure proper shift staffing levels at all times.

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.
  - She is 1 of 3 Auxiliary Operators (AO) on the shift.
- She is contaminated and is escorted to the hospital by the following personnel:
  - 1 of the 2 available Rad Pro Technicians on shift.
  - 1 of the 3 available Reactor Operators (RO) assigned to the shift.

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

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

95 **ID: 2106085** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- A worker reported a fire in the Relay Room and evacuated the area.
- The operating crew has entered OP-TM-EOP-020, COOLDOWN OUTSIDE OF CONTROL ROOM.
- The crew has tripped both Main Feedwater Pumps, all running Condensate Booster Pumps, and all running Condensate Pumps.

(1) What attachment will the Secondary Safe Shutdown NLO perform?
(2) What event will this attachment prevent or terminate?

A. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Terminate uncontrolled HPI due to a spurious "A" train ES actuation.

B. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Prevent an overcooling event by preventing MS-V-2A/B (Isolations to EF-P-1, TBV's and ADVs) from spuriously opening.

C. (1) Attachment 13, "Tripping RCPs Locally"
(2) Prevent core damage if Subcooling Margin is lost

D. (1) Attachment 13, "Tripping RCPs Locally"
(2) Prevent the Reactor Coolant Pumps from spuriously starting.

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) OP-TM-EOP-020, COOLDOWN FROM OUTSIDE OF CONTROL ROOM is entered when there is a fire in the control room or relay room that has a potential to cause damage to safe shutdown required equipment; (2) Once the determination is made that the control room must be evacuated, this starts a sequence in which actions are taken to prevent or terminate undesirable actions which may be caused by the fire; (3) With the announcement at the end of the IMA's, control room operators and non-licensed operators (NLOs) are directed to perform actions within attachments to mitigate possible effects of the fire; (4) The NLOs respond in accordance with OP-TM-EOP-020, and OS-24, CONDUCT OF OPERATIONS DURING ABNORMAL AND EMERGENCY EVENTS, which allows to them to take actions prior to having CRS concurrence (Attachment F, of OS-24), which includes performing Attachment 5 and Attachment 13 of OP-TM-EOP-020; (5) The Secondary Safe Shutdown NLO will perform Attachment 13, which trips the reactor coolant pumps at the 7kV busses to prevent them from spuriously starting without seal cooling.

A.	(1) Attachment 5, "Preventing Spurious Operation of MOV's" (2) Terminate uncontrolled HPI due to a spurious "A" train ES actuation.	Incorrect Answer - Plausible because the MS-V-8's (Isolation valves to the Turbine Bypass Valves) are closed to prevent an overcooling event due to the turbine bypass valves failing midscale on some fire events. In addition the correct reason for performing Attachment 5 is to terminate uncontrolled HPI. Incorrect because the Primary Safe Shutdown NLO performs this attachment.
B.	(1) Attachment 5, "Preventing Spurious Operation of MOV's" (2) Prevent an overcooling event by preventing MS-V-2A/B (Isolations to EF-P-1, TBV's and ADVs) from spuriously opening.	Incorrect Answer - Plausible because the MS-V-8's (Isolation valves to the Turbine Bypass Valves) are closed to prevent an overcooling event due to the turbine bypass valves failing midscale on some fire events. The MS-V-2's could also be used to isolate the turbine bypass valves, but closing the MS-V-2 would also isolate a steam supply path to EF-P-1 (Steam Driven Emergency Feedwater Pump) and an MS-V-4 (Atmospheric Dump Valve). Attachment 5 does open the breaker for MS-V-2A/B, but that is to maintain the valve open for use of EF-P-1 and the MS-V-4's. Incorrect because the Primary Safe Shutdown NLO performs this attachment.
C.	(1) Attachment 13, "Tripping RCPs Locally" (2) Prevent core damage if Subcooling Margin is lost	Incorrect Answer - Plausible because RCPs must be tripped if Subcooling Margin is lost. Incorrect because nothing in the stem indicates that subcooling margin will be lost. In addition, this is not the reason for this action in accordance with OP-TM-EOP-020 basis document.
D.	(1) Attachment 13, "Tripping RCPs Locally" (2) Prevent Reactor Coolant pumps from operating with no seal injection.	Correct Answer - See above.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		1
K/A #	2.1.8	
Importance Rating		4.1

K/A: Ability to coordinate personnel activities outside the control room.

Proposed Question: Question #95

Technical Reference(s): OP-TM-EOP-020, Rev 22 OS-24, Rev 31

OP-TM-EOP-0201, Rev 16

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP020-PCO-1

Question Source: Bank #

Modified Bank # 1720770

New

Question History: unmod on 18-01 Cert Exam Last NRC Exam: unmod on 16-01 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 b.5

Comments:

KA Match: This question matches the KA because the examinee will have to know where various personnel are required to go when the control room must be evacuated.

SRO Only: This question is SRO only because the examinee will have to assess that the stem in a way that determines that the remote shutdown sequence has begun and understand which attachment the NLOs have to perform.

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- A worker reported a fire in the Relay Room and evacuated the area.
- The operating crew has entered OP-TM-EOP-020, COOLDOWN OUTSIDE OF CONTROL ROOM.
- The crew has tripped both Main Feedwater Pumps, all running Condensate Booster Pumps, and all running Condensate Pumps.

(1) What attachment will the Primary Safe Shutdown NLO perform?
(2) What event will this attachment prevent or terminate?

A. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Terminate uncontrolled HPI due to a spurious "A" train ES actuation.

B. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Prevent an overcooling event by preventing MS-V-2A/B (Isolations to EF-P-1, TBV's and ADVs) from spuriously opening.

C. (1) Attachment 13, "Tripping RCPs Locally"
(2) Prevent the Reactor Coolant Pumps from spuriously starting.

D. (1) Attachment 13, "Tripping RCPs Locally"
(2) Prevent Reactor Coolant pumps from operating with no seal injection.

Answer: A

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

96 **ID: 2088112** **Points: 1.00**

The station is processing a change to a site operating procedure that will be a non-permanent change which DOES change the original procedure's intent.

In accordance with AD-AA-101, PROCESSING OF PROCEDURES AND T&RMs, this is defined as a(n) ___(1)___ change and must be approved by a ___(2)___.

A. (1) Interim
(2) Station Qualified Reviewer, only

B. (1) Interim
(2) Station Qualified Reviewer and a Site Functional Area Manager

C. (1) Temporary
(2) Qualified SRO, only

D. (1) Temporary
(2) Qualified SRO and a Station Qualified Reviewer

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with AD-AA-101 (Rev 29) that an INTERIM CHANGE (Page 2) is a non-permanent document change that contains a change of intent. They must also know that TEMPORARY CHANGE (Page 4) is a non-permanent procedure change that does NOT contain a change of intent; (2) According to step 4.2.1 (page 15), if the procedure is an Interim Change then use form AD-AA-101-F-01, DOCUMENT SITE APPROVAL. AD-AA-101-F-01 requires an SQR (Rev 7 Page 1) and SFAM (Page 2) approval.

A.	(1) Interim (2) Station Qualified Reviewer, only	INCORRECT: (1) Correct answer. (2) Plausible because an SQR is required. Incorrect because the SFAM is required as well.
B.	(1) Interim (2) Station Qualified Reviewer and a Site Functional Area Manager	CORRECT: See above.
C.	(1) Temporary (2) Qualified SRO, only	INCORRECT: (1) Plausible because there are only two types of changes, INTERIM and TEMPORARY. Incorrect because the revision contains a change of intent. It must be processed as an INTERIM CHANGE. (2) Plausible because an SROs must approve procedure changes. Incorrect because they are not the only personnel who approve them.
D.	(1) Temporary (2) Qualified SRO and a Station Qualified Reviewer	INCORRECT: (1) Plausible because there are only two types of changes, INTERIM and TEMPORARY. Incorrect because the revision contains a change of intent. It must be processed as an INTERIM CHANGE. (2) Plausible because a Qualified SRO and Station Qualified Reviewer would be required to process a Temporary Change.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.2.6	
	Importance Rating		3.6

K/A: 2.2.6 - Knowledge of the process for making changes to procedures.

Proposed Question:	Question #96	
Technical Reference(s):	AD-AA-101, Rev 29	
	AD-AA-101-F-10	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EQC00016	

Question Source:	Bank #		
	Modified Bank #		

	New	X		
Question History:	N/A	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	b.3	

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the permissions required to process a temporary procedure change.

SRO only: This question is an SRO only question because it involves knowing the process for changing plant procedures. This is an SRO task.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

97 **ID: 2105938** **Points: 1.00**

WC-AA-101-1004, ONLINE MAINTENANCE FOR LIMITING CONDITIONS FOR OPERATION OF SYSTEMS OR COMPONENTS specifies shutdown LCO durations that require continuously working critical path.

Which of the choices below identifies the LONGEST time duration shutdown LCO time limit that requires continuous work critical path, per WC-AA-101-1004?

A. 1 day (24 hours)

B. 2 days (48 hours)

C. 3 days (72 hours)

D. 7 days (168 hours)

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with WC-AA-101, ON-LINE WORK CONTROL PROCESS (Rev 28, Page 17), Attachment 1, a 7 day LCO is required to be worked 24 hours a day.

A.	1 day (24 hours)	INCORRECT: Plausible because a 24 hour LCO must be continuously worked. Incorrect because 7 days is the maximum that this requirement applies.
B.	2 days (48 hours)	INCORRECT: Plausible because a 48 hour LCO must be continuously worked. Incorrect because 7 days is the maximum that this requirement applies
C.	3 days (72 hours)	INCORRECT: Plausible because a 72 hour LCO must be continuously worked. Incorrect because 7 days is the maximum that this requirement applies
D.	7 days (168 hours)	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			3
	K/A #		2.2.19	
	Importance Rating			3.4

K/A: Knowledge of maintenance work order requirements

Proposed Question:	Question #97	
Technical Reference(s):	WC-AA-101, Rev 28	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	NOP DPBIG-APCO-3	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	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	b.1

Comments:

KA Match: This question matches the KA because the examinee must know when maintenance must be continuously worked, which would be a maintenance work order requirement.

SRO only: This question is SRO only because the examinee must know that if the work being performed would cause the Unit to enter an LCO of 7 days or less that the planned work must be continuously worked.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

98 **ID: 2105798** **Points: 1.00**

Plant Conditions:

- Reactor power is 25% with ICS in auto

Event:

- Personnel must enter the 'A' D-ring to investigate RC-P-1A for an oil leak

Given the above information, which of the following describes the minimum personnel required to authorize this work to be performed?

A. Shift Manager

B. Shift Manager and Radiation Protection Manager

C. Shift Manager, Radiation Protection Manager, and Plant Manager

D. Shift Manager, Radiation Protection Manager, Plant Manager and Site Vice President

Answer: C

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with RP-TM-460-1007, ACCESS TO TMI-1 REACTOR BUILDING (Rev 8, Page 4) approval of the Shift Manager, Plant Manager, and Radiation Protection Manager are required for entries within the secondary shield.

A.	Shift Manager	INCORRECT: Plausible because his permission is required. Incorrect because the Plant Manager and Radiation Protection Manager are also required.
B.	Shift Manager and Radiation Protection Manager	INCORRECT: Plausible because their permission is required. Incorrect because the Plant Manager is also required to give permission.
C.	Shift Manager, Radiation Protection Manager, and Plant Manager	CORRECT: See above.
D.	Shift Manager, Radiation Protection Manager, Plant Manager and Site Vice President	INCORRECT: Plausible because the Shift Manager, Radiation Protection Manager and Plant Manager are required. Incorrect because the Site Vice President is not required.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	3		
	K/A #	2.3.12		
	Importance Rating	3.7		

K/A: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question:	Question #98	
Technical Reference(s):	RP-TM-460-1007, Rev 8	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	NOP-DBIG-APCO-1	

Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	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	b.4	

Comments:

KA Match: This question matches the KA because the examinee must know have knowledge of the procedure which allows personnel into the reactor building at power. The examinee must know the containment entry requirements to allow work in the 'A' D-ring.

High Cog: This question is high cog because the examinee must assess the plan conditions and recall whose permission is required for work in the 'A' D-ring.

SRO Only: This question is SRO only because the examinee must have knowledge of the radiation hazard posed by work on the polar crane at power. The examinee must know a special level of authorization is required to perform work in this area.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

99 **ID: 2105849** **Points: 1.00**

In accordance with EP-AA-112-F-01, COMMAND AND CONTROL TURNOVER BRIEFING FORM, which of the following duties CANNOT be delegated to the EOF?

A. NRC Notifications

B. Event Classification

C. PAR Decision-Making

D. State/Local Notifications

Answer: B

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: In accordance with EP-AA-112-F-01 (Rev G Page 1), Event Classification is not a delagatable event to the EOF. In accordance with EP-AA-112 Step 3.1.4, if the TSC is not functional the SED can turnover to the EOF.		
A.	NRC Notifications	INCORRECT: Plausible because this can be delegated to the TSC and EOF.
B.	Event Classification	CORRECT: See above.
C.	PAR Decision-Making	INCORRECT: Plausible because this can be delegated to the TSC and EOF.
D.	State/Local Notifications	INCORRECT: Plausible because this can be delegated to the TSC and EOF.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	3		
	Group #	4		
	K/A #	2.4.37		
	Importance Rating	4.1		

K/A: Knowledge of the lines of authority during implementation of the emergency plan

Proposed Question:	Question #99	
Technical Reference(s):	EP-AA-112-F-01, Rev G	

Proposed References to be provided to applicants during examination:	None

Learning Objective:	EOP-DBIG-PCO-6		
Question Source:	Bank #		
	Modified Bank #		
	New	X	

Question History:	N/A	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	b.5

Comments:

KA Match: This question matches the KA because the examinee must know the lines of authority of the command and control checklist during an Eplan event.

SRO Only: This question is SRO only because the examinee must have knowledge of organizational hierarchy during an Eplan event. The examinee must know that event classification can only be performed in the control room or TSC.

EXAMINATION ANSWER KEY

ILT 18-01 NRC EXAM - SRO

100 **ID: 2110406** **Points: 1.00**

In accordance with Technical Specification 3.3, EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY SYSTEMS, which of the following BWST parameters is out of specification and what is the basis for that specification?

- BWST Boron Concentration is 2490 ppm.
- BWST Water Temperature is 49°F.

A. (1) BWST Water Temperature
(2) To prevent Boron Crystallization

B. (1) BWST Water Temperature
(2) To ensure the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident

C. (1) BWST Boron Concentration
(2) To prevent Boron Crystallization

D. (1) BWST Boron Concentration
(2) To ensure the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident

Answer: D

Answer Explanation

Explanation: To answer this question correctly, the examinee must know: (1) According to technical specification 3.3.1.1.a (AMD 289, Page 3-21), the boron concentration of the BWST must have a minimum 2500 PPM at a temperature of not less than 40F; (2) If the temperature or concentration is not within limits, an 8 hour time limit to restore to OPERABLE, or technical specification 3.0.1 applies; (3) In accordance with the bases of Technical Specification 3.3 (ECR 14-00208 Page 3-23) this limit is to ensure a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA; (4) The limits of on BWST minimum volume and boron concentration ensures that sufficient water is available within containment to permit recirculation cooling flow to the core AND the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident (LOCA).

A.	(1) BWST Water Temperature (2) To prevent Boron Crystallization	INCORRECT: Plausible if the examinee believes that the temperature is below the temperature required by Technical Specification 3.3. Boron crystallization would be a concern due to the low temperature. Incorrect because the temperature limit is 40F.
B.	(1) BWST Water Temperature (2) The ability to ensure the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident	INCORRECT: Plausible if the examinee believes that the temperature is below the temperature required by Technical Specification 3.3. Injecting cold water could be an reactivty concern. Incorrect because the temperature limit is 40F.
C.	(1) BWST Boron Concentration (2) To prevent Boron Crystallization	INCORRECT: Plausible because there is an upper limit to BWST boron concentration in accordance with 1301-1, SHIFT AND DAILIES (Rev 178 Page 20). The limit is 2750 ppm. If the examinee believed that the listed boron concentration exceeded an upper limit then boron crystallization would be a concern. Incorrect because the BWST boron concentration exceeded its lower limit.
D.	(1) BWST Boron Concentration (2) To ensure the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #			3
	Group #			4
	K/A #		2.4.9	
	Importance Rating			4.2

K/A: Knowledge of low power/shutdown implications in accident (e.g. loss of coolant accident or loss of residual heat removal) mitigation strategies

Proposed Question:	Question #100	
Technical Reference(s):	Technical Specification 3.3	1301-1, Rev 178

Proposed References to be provided to applicants during examination:	None

Learning Objective:	212-GLO-14			
Question Source:	Bank #			
	Modified Bank #			
	New	X		
Question History:	N/A	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	b.2

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of a shutdown implication if the BWST boron concentration is out of specification. The specification exists to ensure shutdown margin during a loss of coolant accident.

Tier 3: This is a Tier 3 KA because the examinee must have generic knowledge of a shutdown implication during a loss of coolant accident.