

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

1

ID: 1685138

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Reactor Trip due to low condenser vacuum
- Condenser vacuum is 22 in Hg mercury and slowly lowering
- OTSG 'A' is at 98% in the operating range and slowly rising
- OTSG 'B' is at 25" in the startup range and steady

What actions will the operating crew take to stabilize the plant?

- A.
 - 1) Trip 'A' Main Feed Pump
 - 2) Stabilize OTSG pressure with the MS-V-3's (Turbine Bypass Valves)
- B.
 - 1) Trip both Main Feed Pumps
 - 2) Stabilize OTSG pressure with the MS-V-4's (Atmospheric Dump Valves)
- C.
 - 1) Trip 'A' Main Feed Pump
 - 2) Stabilize OTSG pressure with the MS-V-4's (Atmospheric Dump Valves)
- D.
 - 1) Trip both Main Feed Pumps
 - 2) Stabilize OTSG pressure with the MS-V-3's (Turbine Bypass Valves)

Answer: B

Answer Explanation

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Explanation: To answer this question correctly the examinee must know: (1) on a low condenser vacuum, the MS-V-3's close and latch shut (2) at that point the MS-V-4's become the controlling valves to stabilize OTSG pressure at the MS-V-3 setpoint (3) the step to stabilize pressure is Step 3.10, which sends the operator to OP-TM-EOP-010, Guide 6 which directs the operator to MAINTAIN OTSG pressures below 1020 psig (4) due to the 'A' OTSG being greater than 97.5%, both of the Main Feed Pumps must be tripped, IAW OP-TM-EOP-001, step 3.6 RNO verifying BOTH OTSG levels below 97.5 % in the operating range, the RNO directs the operator ENSURE FW-P-1A and FW-P-1B (the Main Feed Pumps) are tripped.

A.	1) Trip 'A' Main Feed Pump 2) Stabilize OTSG pressure with the MS-V-3's (Turbine Bypass Valves)	Plausible because the 'A' OTSG is the only OTSG with a high level, but due to the Main Feedwater System being cross-connected, both Main Feedwater Pumps must be tripped. In addition, if the examinee does not know that a main condenser vacuum < 23 in HG will close and latch the MS-V-3's (Turbine Bypass Valves) the examinee could choose this answer.
B.	1) Trip both Main Feed Pumps 2) Stabilize OTSG pressure with the MS-V-4's (Atmospheric Dump Valves)	Correct Answer
C.	1) Trip 'A' Main Feed Pump 2) Stabilize OTSG pressure with the MS-V-4's (Atmospheric Dump Valves)	Plausible because the 'A' OTSG is the only OTSG with a high level, but due to the Main Feedwater System being cross-connected, both Main Feedwater Pumps must be tripped.
D.	1) Trip both Main Feed Pumps 2) Stabilize OTSG pressure with the MS-V-3's (Turbine Bypass Valves)	If the examinee does not know that a main condenser vacuum < 23 in HG will close and latch the MS-V-3's (Turbine Bypass Valves) the examinee could choose this answer.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E02	2.1.20
	Importance Rating	4.6	4.6

K/A: Ability to interpret and execute procedure steps.

Proposed Question: RO Question 1

Technical Reference(s): OP-TM-EOP-001, Rev 16 OP-TM-411-000, Rev 21
OP-TM-EOP-010, Rev 19

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Proposed References to be provided to applicants during examination: None

Learning Objective: EOP DBIG 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 10
55.43

Comments:

K/A Match: This question matches the KA because the examinee will have to interpret that the 'A' OTSG level is high, and that the procedurally driven action to execute is to trip both Main Feedpumps.

In addition, Guide 6, which is a required for plant stabilization, is required. The examinee will have to diagnose that the MS-V-3's (Turbine Bypass Valves) are closed and latched, and the the MS-V-4's (Atmospheric Dump Valves) are required to be utilized to stabilize the plant.

High Cog: This question is High Cog because the examinee must understand that when condenser vacuum lowers to < 22 in Hg, the plant response is that of cooldown to atmosphere and not the condenser. The MS-V-3 valves will latch closed, and transfer their setpoint to the MS-V-4 valves.

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2

ID: 1737142

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- A Small Break LOCA has occurred inside containment.
- The leak is estimated at 25 gpm.
- Reactor Building pressure is 0.5 psig and rising at approximately 0.1 psig every two hours.
- Reactor Building temperature above 320' elevation is 120°F and rising at approximately 2°F every hour.

OTSG Startup level indicates ____ (1) ____ than actual based on elevated Reactor Building ____ (2) ____.

- A. (1) lower
(2) pressure
- B. (1) higher
(2) pressure
- C. (1) lower
(2) temperature
- D. (1) higher
(2) temperature

Answer: D

Answer Explanation

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<<Explanation: To answer this question correctly the examinee must know: (1) The Startup Range Reference legs for OTSG level control are closed cell, wet reference legs; (2) Because of this, Reactor Building pressure will have no effect on the Startup Level indication, but Reactor Building temperature will; (3) As Reactor Building temperature rises, the water in the reference leg will become less dense; (4) When the water becomes less dense, the pressure on the High Pressure side of the D/P cell will lower, which lowers the D/P felt by the system; (5) The lower D/P results in a HIGHER than actual level.

A.	(1) lower (2) pressure	INCORRECT: Reactor Building pressure will have no effect on the Startup Range level indication. Plausible because Reactor Building pressure is a parameter that is rising in the above scenario.
B.	(1) higher (2) pressure	INCORRECT: Reactor Building pressure will have no effect on the Startup Range level indication. Plausible because Reactor Building pressure is a parameter that is rising in the above scenario.
C.	(1) lower (2) temperature	INCORRECT: Indicated level will appear higher than actual level. Plausible because the D/P cell feels a lower pressure on the High Pressure side.
D.	(1) higher (2) temperature	CORRECT: See above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	009	EK2.03
	Importance Rating	3.0	3.3

K/A: Small Break LOCA: Knowledge of the interrelationship between the small break LOCA and the following: S/Gs

Proposed Question: RO Question #2

Technical Reference(s): TQ-TM-104-411-C001, Rev 8 OS-24, Rev 28

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-2

Question Source: Bank # 375108
Modified Bank #
New

Question History: Sim Exam 5 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

EXAMINATION ANSWER KEY

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	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 7	
	55.43	
Comments:		
KA Match; This question matches the KA because the examinee must determine the level error in an OTSG caused by a small break LOCA.		
High Cog: The examinee must analyze the effects of a temperature rise on the OTSG level detectors.		

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3

ID: 1685156

Points: 1.00

Plant Conditions:

- Reactor is at 70% power
- All ICS stations are in HAND

EVENT:

- RC-P-1B trips

Tcold on the 'A' RCS loop will _____ and the operator will have to _____ 'A' feedwater flow to maintain a $0^{\circ}\text{F } \Delta T_c$.

- A. rise
raise
- B. rise
lower
- C. lower
lower
- D. lower
raise

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

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<<Explanation: To answer this question correctly, the examinee will have to know: (1) RC-P-1B is in the 'A' RCS loop (2) because ALL ICS stations are in hand, the plant will respond to the loss of 'A' loop RCS flow, but the control system will keep the other parameters (i.e. Feedwater flow, Control Rod Position, etc) steady, (3) ΔT_c is calculated by taking $T_{cold A} - T_{cold B}$, therefore with the loss of flow in the 'A' loop, the ΔT_c will become negative, (4) to make the ΔT_c become less negative, the operator will have to lower on 'A' feedwater flow.

A.	rise raise	This is plausible if a reactor coolant pump in the 'B' loop were secured, and the question still asked how to make ΔT_c by controlling the 'A' feedwater loop.
B.	rise lower	This is plausible if the examinee misunderstands the direction T_{cold} will go when an 'A' loop Reactor Coolant Pump is secured.
C.	lower lower	Correct Answer
D.	lower raise	This is plausible if the student does not understand how to correct the ΔT_c situation by with 'A' feedwater flow.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015	AA1.09
	Importance Rating	3.1	3.2

K/A: Ability to operate and / or monitor the following as they apply to the Loss of Reactor Coolant Pump Malfunction (Loss of RC Flow): RCS temperature detection system.

Proposed Question: RO Question #3

Technical Reference(s): 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 # 858178

Modified Bank #

New

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

Question Cognitive Level: Memory or Fundamental Knowledge

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 7	
	55.43	
Comments: (KA Match and why high cog)		
<p>K/A Match: This matches the K/A because the examinee will have to describe what they expect RCS Tcold to do upon securing a reactor cooling pump.. In addition, the question requires the operator to know how to operate the FW control station associated with level in the OTSG in which the reactor coolant pump was secured. The RCS temperature detection subsystem in the question is the ΔT_c ICS circuit which monitors the 'A' and 'B' RCS cold leg temperatures. When in automatic, this ICS circuit uses FW demand and ΔT_c to maintain a near zero cold leg temperature. If the system were in automatic, when a reactor coolant pump is secured, the system will automatically re-ratio feedwater to maintain a near zero ΔT_c.</p>		
<p>High Cog: This question is high cog because the examinee will have understand the what the impact of securing a reactor coolant pump has to analyze how the RCS parameters will change, and how to manipulate controls to return those parameters to normal.</p>		

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4

ID: 1685199

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- MU-P-1B trips on overload.
- MAP G-2-5 PZR LEVEL HI-LO annunciator has alarmed.
- Pressurizer level is 196 inches and slowly lowering.

Which one of the following identifies actions that the operator must take while restoring a Makeup Pump to operation?

- A. De-energize all Pzr Heaters; AND
CLOSE MU-V-33A-D, Control Bleed Off Valves.
- B. CLOSE MU-V-3, Letdown Isolation Valve; AND
CLOSE MU-V-33A-D, Control Bleed Off Valves.
- C. De-energize all Pzr Heaters; AND
Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to
HAND and CLOSE the valves.
- D. CLOSE MU-V-3, Letdown Isolation Valve; AND
Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to
HAND and CLOSE the valves.

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

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<<Explanation: To answer this question correctly the examinee must know: (1) Since there is no makeup flow, the inventory will begin to lower. Pressurizer level will lower due to the normal letdown and CBO flow of the the reactor coolant pumps. (2) The crew will enter OP-TM-AOP-041 for a loss of Seal Injection (due to the MU-P-1B trip), which directs MU-V-32 to be closed in step 3.3 and MU-V-3 and MU-V-17 to be closed in step 3.5 RNO.

A.	De-energize all Pzr Heaters; AND CLOSE MU-V-33A-D, Control Bleed Off Valves.	This is plausible because the operator may incorrectly believe that the heaters will need to be deenergized due to low pressurizer level; and because the operator may incorrectly believe that procedures direct the closure of MU-V-33A-D to save RCS inventory from being lost.
B.	CLOSE MU-V-3, Letdown Isolation Valve; AND CLOSE MU-V-33A-D, Control Bleed Off Valves.	This is plausible because the operator may incorrectly believe that procedures direct the closure of MU-V-33A-D to save RCS inventory from being lost. Closing MU-V-3 would save inventory, but incorrect due to MU-V-33's are not required to be closed.
C.	De-energize all Pzr Heaters; AND Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to HAND and CLOSE the valves.	This is plausible because the operator may incorrectly believe that that the heaters will need to be deenergized due to low pressurizer level. According to OP-TM-MAP-G0305, at 80 inches in the Pzr the Pzr Heaters will automatically trip. However, this level has not been reached, and the need to de-energize the Pzr heaters is not identified in either EOP-10 Guide 9, or AOP-041.
D.	CLOSE MU-V-3, Letdown Isolation Valve; AND Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to HAND and CLOSE the valves.	Correct Answer

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022	AK1.03
	Importance Rating	3.0	3.4

K/A: Knowledge of the operational implications of the following concepts as they apply to the Loss of Reactor Coolant Makeup: Relationship between charging flow and PZR level.

Proposed Question: RO Question #4

Technical Reference(s): OP-TM-AOP-041, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-041-PCO-1

EXAMINATION ANSWER KEY

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Question Source:	Bank #	719461
	Modified Bank #	
	New	
Question History:	Sim Exam 7	Last NRC Exam:
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	8
	55.43	
Comments:		
<p>K/A Match: The KA is matched because the operator must demonstrate knowledge (i.e. what actions must be taken) of the operational implications (i.e. RCS Inventory is being lost, and action that are needed to start MU Pump restart) of the concept of the relationship between charging flow and Pzr level as it applies to the Loss of Reactor Coolant Pump Makeup. Due to the loss of the charging flow, pressurizer level will lower, so letdown is isolated by closing MU-V-3 to minimize the loss of inventory from the pressurizer. MU-V-17(Pressurizer level control valve) and MU-V-32 (Seal Injection Valve) are closed in hand for a controlled restoration of makeup and seal injection upon the restoration of a makeup pump.</p>		

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5

ID: 1737146

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- All available Nuclear River water pumps trip, and cannot be restarted.
- OP-TM-AOP-031, LOSS OF NUCLEAR SERVICES COMPONENT COOLING is entered.

Which one of the following describes the strategy and basis for Reactor Coolant Pump (RCP) operation?

- A. The RCPs must be secured due to the loss of labyrinth cooling to each RCP.
- B. The RCPs must be secured to eliminate heat load from the NSCCW System.
- C. The RCPs may remain in operation provided seal injection flow is greater than 22 gpm.
- D. The RCPs may remain in operation provided the SR-NR cross tie is performed prior to reaching the HI-2 alarm setpoint for the RCPs.

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

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<<Explanation: To answer this question correctly, the examinee must know: (1) After the Reactor is tripped in OP-TM-AOP-031, the next step the operating crew must do is trip all four Reactor Coolant Pumps; (2) According to the basis, this is to eliminate heat load from the Nuclear Services system.		
A.	The RCPs must be secured due to the loss of labyrinth cooling to each RCP.	Incorrect: Plausible because Intermediate Closed Cooling Water (which is cooled by Nuclear River Water) does cool the thermal barrier on the reactor coolant pumps. Incorrect because the seal injection cools the labyrinth seal.
B.	The RCPs must be secured to eliminate heat load from the NSCCW System.	Correct Answer: See above.
C.	The RCPs may remain in operation provided seal injection flow is greater than 22 gpm.	Incorrect: Plausible because Intermediate Closed Cooling does cool the thermal barrier of the Reactor Coolant Pump, and there is an interlock with ICCW and SI. Incorrect because the Reactor Coolant Pumps must be secured because of the lack of cooling to the RCP motors and the heat load on the NSCCW system.
D.	The RCPs may remain in operation provided the SR-NR cross tie is performed prior to reaching the HI-2 alarm setpoint for the RCPs.	INCORRECT: Plausible because a cross connect does exist between Secondary River and Nuclear River water. The systems are cross connected in OP-TM-AOP-031. Incorrect because the Reactor Coolant Pumps are a major heat load on the system and must be secured prior to cross connecting.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	026	AK3.03
	Importance Rating	4.0	4.2

K/A: Loss of Component Cooling Water: Knowledge of the reasons for the following responses as they apply to Loss of Component Cooling Water: Guidance actions contained in EOP for Loss of CCW.

Proposed Question: RO Question #5

Technical Reference(s): OP-TM-AOP-031, Rev 6
OP-TM-AOP-031, Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-031-PCO-4

Question Source: Bank # 634661
Modified Bank #
New

EXAMINATION ANSWER KEY

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Question History: Sim 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:

K/A Match: This question matches the KA because the examinee must have knowledge of an action contained in the AOP for Loss of Nuclear Services Component Cooling.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

6

ID: 1685231

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- The PORV fails partially open.
- Efforts to close the PORV Block valve have failed.

Current Plant Conditions:

- The plant is being shutdown.
- RCS Pressure is 2050 psig and slowly lowering.
- Pressurizer level is 250 inches and slowly rising.

In accordance with OP-TM-AOP-043, LOSS OF PRESSURIZER, what guidance will be given regarding Subcooling Margin?

Place MU-V-17 in hand and raise makeup flow to maintain _____.

- A. > 25°F SCM
- B. > 40°F SCM
- C. > 45°F SCM
- D. > 70°F SCM

Answer: B

Answer Explanation

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<<Explanation: To answer this question correctly, the examinee must know: (1) The plant is being shutdown because there is inadequate heater capacity to maintain RCS pressure; (2) To maintain RCS pressure high enough to complete the shutdown, the steam bubble in the pressurizer will have to be squeezed, and pressurizer level will go up; (3) With current plant conditions, there is adequate pressurizer level indication and there the plant is not solid; (4) To use a steam table to calculate SCM is < 40°F SCM, when the guidance is applicable.

A.	> 25F SCM.	Plausible because 25F SCM is a limit for entry into OP-TM-EOP-002 when the plant is shutdown.
B.	> 40F SCM.	Correct Answer.
C.	> 45F SCM.	Plausible because this guidance would be applicable if the plant had a loss of all pressurizer level indication.
D.	> 70F SCM.	Plausible if the operating crew was performing a solid plant shutdown and cooldown. Pressurizer level is at 250 inches so solid plant operations are not applicable at this time.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027	AA2.12
	Importance Rating	3.7	3.8

K/A: Ability to determine and interpret the following as they apply to Pressurizer Pressure Control Malfunctions: PZR Level

Proposed Question: RO Question #6

Technical Reference(s): OP-TM-AOP-043, Rev 6
Steam Tables

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

Learning Objective: AOP-043-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

EXAMINATION ANSWER KEY

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10 CFR Part 55 Content:

55.41

5

55.43

Comments:

K/A Match: This is a K/A match because the PORV is part of the Pressurizer Pressure Control system, in that it receives an input from a narrow range pressure detector and opens when a setpoint is reached. Due to the PORV being open far enough to effectively lower pressure beyond the capacity of the pressurizer heaters, a shutdown must commence. To effectively ensure we have time to shutdown before tripping on a low pressure signal, the Pressurizer Level Control Valve, MU-V-17 is taken to HAND, and flow is raised to squeeze the bubble in the pressurizer. The operator will control pressurizer level to maintain a subcooling margin > 40F which the examinee can determine with the steam tables.

High Cog: This question is high cog because the examinee will have to analyze the current plant parameters and determine which subcooling margin guidance is applicable.

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7

ID: 1685342

Points: 1.00

Event:

- Loss of offsite power.
- Emergency Diesel Generator, EG-Y-1A fails to start.
- Emergency Diesel Generator, EG-Y-1B trips after starting.

Based on these conditions, which of the following control room extension controls or instruments remain operable?

- A. RM-A-2 Sample Pump.
- B. Motor Driven Fire Pump FS-P-2.
- C. 1SB-D2, 1B Auxiliary Transformer feeder breaker to 1D 4KV Switchgear.
- D. RR-PI-225, RB Emergency Cooling Outlet Cooler 1B pressure instrument.

Answer: C

Answer Explanation

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Explanation: To answer this question correctly, the examinee must know: (1) the DC electrical system is used to supply ES breaker control power and protective relaying.

A.	RM-A-2 Sample Pump.	Plausible because this is safety related equipment, but RM-A-2 sample pump is powered from 1A ES 480V MCC and uses 120V control power transformer housed in the breaker cubicle.
B.	Motor Driven Fire Pump FS-P-2.	Plausible because it is safety related equipment, but employs AC control power.
C.	1SB-D2, 1B Auxiliary Transformer feeder breaker to 1D 4KV Switchgear.	Correct Answer.
D.	RR-PI-225, RB Emergency Cooling Outlet Cooler 1B pressure instrument.	Plausible because this is an ES pressure indicator, but due to lack of power to TRB none of the RB Emergency Cooler outlet pressure instruments are available.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	055	EA2.04
	Importance Rating	3.7	

K/A: Ability to determine or interpret the following as they apply to a Station Blackout:
Instruments and controls operable with only dc battery power available.

Proposed Question: RO Question #7

Technical Reference(s): 1107-5, Rev 151

TQ-TM-104-740-C001, REV 7

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-020-PCO-2

Question Source: Bank #
Modified Bank # 363906
New

Question History: Sim Exam 8 Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

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Comprehension or Analysis

10 CFR Part 55 Content: 55.41
55.43 5

Comments:

K/A Match: This matches the K/A because the examinee must know what controls are DC power (which would make them available during a station blackout) in the control room to answer the question correctly. The question choices are a mixture of instrument and controls.

Modified Bank Question: This question was modified by changing the failure of EG-Y-B to trip after starting, and replacing MU-V-16C with an instrument (RR-PI-225). This replacement was performed to better match the K/A.

Event:

- Loss of offsite power.
- Emergency Diesel Generators EG-Y-1A and EG-Y-1B both failed to start.

Based on these conditions, identify the ONE statement below that describes control room extension controls that remain operable.

- A. RM-A-2 Sample Pump.
- B. Motor Driven Fire Pump FS-P-2.
- C. High Pressure Injection valve MU-V-16C.
- D. 1SB-D2, 1B Auxiliary Transformer feeder breaker to 1D 4KV Switchgear.

Answer: D

EXAMINATION ANSWER KEY

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EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

8

ID: 1685400

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- Station Blackout Diesel Generator, EG-Y-4 is unavailable due to an outage.

EVENT:

- Loss of Offsite power.
- Emergency Diesel Generator, EG-Y-1B has started and is powering 'E' 4kV Bus.
- Emergency Diesel Generator, EG-Y-1A has tripped on overspeed.

What action per OP-TM-AOP-020, Loss of Station Power, will be taken to prevent the loss of plant control and provide additional time to restore AC power?

- A. Secure Emergency DC Seal Oil Pump GN-P-2.
- B. Initiate OP-TM-732-901, Energize 1P 480V Bus using the ES Bus Cross Tie.
- C. Initiate OP-TM-732-902, Energize 1S 480V Bus using the ES Bus Cross Tie.
- D. Ensure Turning Gear Oil Pump LO-P-5 is running, then secure Emergency Bearing Oil Pump LO-P-6.

Answer: B

Answer Explanation

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To answer this question, the examinee must know (1) EG-Y-1A powers the 'D' 4kV bus (2) since EG-Y-1A and EG-Y-4 are unavailable, the 'D' 4kV bus has no power supply (3) 1P 480V Bus is downstream of the 'D' 4kV bus and has no power (4) the ES busses can be crosstied with the reactor is not critical

A. "Secure Emergency DC Seal Oil Pump." - Incorrect - Plausible since this equipment is identified in step 3.15 as "verify operating" in this AOP procedure not securing the pump.

B. "Initiate OP-TM-732-901, Energize 1P 480V Bus using the ES Bus Cross Tie.." - Correct - This is step 3.21. This energizes the battery chargers for the "A" station battery. In accordance with OP-TM-AOP-0201, Loss of Station Power basis document, *if AC is only available to one train, the 480V ES busses will be crosstied and large DC loads will be minimized. This will allow for additional time to restore an AC power source. The procedures to crosstie the ES busses can be performed without jeopardizing the reliability of the available AC supply and will prevent the loss of plant control caused by loss of DC power and two vital busses.*

C. "Initiate OP-TM-732-902, Energize 1S 480V Bus using the ES Bus Cross Tie." - Correct Answer - Plausible because this is a step in the procedure, however it is for when the 1E 4kV is de-energized not the situation given in the stem.

D. "Ensure Turning Gear Oil Pump LO-P-5 is running, then secure Emergency Bearing Oil Pump LO-P-6." - Incorrect - Plausible, since LO-P-6 is powered from "A" DC and will be running. LO-P-5 is powered from 1P 480V and will not be running. Step 3.13 directs the examinee to ensure LO-P-5 or LO-P-6 is running. Once that is verified, the examinee would move to the next step with no action taken.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	056	AK3.02
	Importance Rating	4.4	

K/A: Knowledge of the reasons for the responses as they apply to the Loss of Offsite Power: Actions contained in the EOP for loss of offsite power.

Proposed Question: RO question #8

Technical Reference(s): OP-TM-AOP-020, Rev 24

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP020-PCO-4

Question Source: Bank #

Modified Bank # 1244855

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 7

55.43

Comments: This is a KA match because contains actions in our loss of offsite power AOP. The loss of offsite power AOP for TMI is essentially an EOP in broad terms. The loss of offsite power EOP for TMI deals the an extended loss of offsite power.

Original Question:

Plant Conditions:

- Plant is at 100% power
- EG-Y-4 is unavailable due to an outage.

Event:

- Loss of Offsite power.
- Emergency Diesel Generator, EG-Y-1A has started and is powering 'D' 4kV Bus
- EG-Y-1B has tripped on overspeed.

What action per OP-TM-AOP-020, Loss of Station Power, will be taken to prevent the loss of plant control and provide additional time to restore AC power?

- A. Secure Emergency DC Seal Oil Pump GN-P-2.
- B. Initiate OP-TM-732-901 Energize 1P 480V Bus using the ES Bus Cross Tie.
- C. Initiate OP-TM-732-902 Energize 1S 480V Bus using the ES Bus Cross Tie.
- D. Ensure Turning Gear Oil Pump LO-P-5 is running, then secure Emergency Bearing Oil Pump LO-P-6.

Answer: C

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

9

ID: 1685632

Points: 1.00

Plant Conditions:

- Plant Cooldown is in progress per 1102-11, Plant Cooldown.
- Current Cooldown rate is 85°F/HR.
- Current RCS temperature is 359°F and RCS Pressure is 600 psig.

EVENT:

- Channel 1 of B train ES indicates actuated.
- NI-5 loses indication on console center.
- MU-V-2A/2B Letdown Cooler Outlet Valves close.

(1) Which one of the following describes the action required if RCS temperature lowers below 313°F?

(2) What is the reason for the action taken?

- A. (1) Open RC-RV-2 (PORV) until RCS pressure is below 540 psig.
(2) The NDTT interlock has lost power.
- B. (1) Open RC-V-1 (RCS Spray Valve) until RCS pressure is below 540 psig.
(2) The PORV and the NDTT interlock have lost power.
- C. (1) Open RC-RV-2 (PORV) until RCS pressure is below 540 psig.
(2) RC-V-1 and the NDTT Interlock have lost power.
- D. (1) Open RC-V-1 (RCS Spray Valve) until RCS pressure is below 540 psig.
(2) The NDTT interlock has lost power.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) Channel 1 of B train ES appearing actuated, NI-5 losing indication on console center, and MU-V-2A/2B letdown cooler outlet valves closing are all indications that Vital Bus A is lost; (2) the reason the stem indicates that on Channel 1 of B train ES appears actuated vice Channel 1 of A and B train ES appears actuated is because the A train ES lights are powered from Vital Bus A, and this information would not be apparent to the operator until the ES Status Light Power Supply Select Switch is positioned to the BUS-B position; (3) The operating crew would enter OP-TM-AOP-015, LOSS OF VBA, which would direct the operating crew to open RC-RV-2 if RCS pressure were > 590 psig when RCS temperature is < 313°F due to NDTT interlock not being operable.

A.	(1) Open RC-RV-2 (PORV) until RCS pressure is below 540 psig. (2) The NDTT interlock has lost power.	Correct Answer
B.	(1) Open RC-V-1 (RCS Spray Valve) until RCS pressure is below 540 psig. (2) The PORV and the NDTT interlock have lost power.	INCORRECT: Plausible because RC-V-1 (RCS Spray Valve) is usually used to lower RCS pressure. At this temperature and pressure, the PORV is used due to NDTT concerns. In addition, the PORV did not lose power.
C.	(1) Open RC-RV-2 (PORV) until RCS pressure is below 540 psig. (2) RC-V-1 and the NDTT Interlock has lost power.	INCORRECT: Plausible because RC-RV-2 is what the operators would use to lower pressure in this case. Incorrect because RC-V-1 did not lose power (1A ES MCC).
D.	(1) Open RC-V-1 (RCS Spray Valve) until RCS pressure is below 540 psig. (2) The NDTT interlock has lost power.	Plausible because RC-V-1 (RCS Spray Valve) is usually used to lower RCS pressure. At this temperature and pressure, the PORV is used due to NDTT concerns.

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

K/A: Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus.

Proposed Question: RO Question #9

Technical Reference(s): OP-TM-AOP-015, Rev 10

OP-TM-AOP-0151, Rev 7

Proposed References to be provided to applicants during examination: None

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Learning Objective: AOP-015-PCO-4

Question Source: Bank #

Modified Bank # 606540

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 10

55.43

Comments: (KA Match and why high cog)

K/A Match: This matches the KA because the examinee can determine that the PORV is used to lower pressure due to NDTT concerns (vice the spray valve). Loss of a vital bus is an AOP at this facility. The examinee must determine which AOP is entered and the action within.

High Cog: This question is high cog because the examinee will have to analyze plant conditions and determine which vital bus was lost.

Plant Conditions:

- Plant Cooldown is in progress per 1102-11, Plant Cooldown.
- Current Cooldown rate is 85°F/HR.
- Current RCS temperature is 359°F and RCS Pressure is 600 psig.

Sequence of Events:

- "A" inverter fails and VBA is lost.
- OP-TM-AOP-015, Loss of VBA is entered.
- 15 minutes into the casualty the Unit Reactor Operator reports "RCS Temperature is 312°F and RCS Pressure is 595 psig.

Based on the report, ____ (1) ____ should be opened and should remain open until RCS pressure is less than the procedurally required value of ____ (2) ____ psig.

A. (1) RC-RV-2 (PORV)
(2) 540

B. (1) RC-RV-2 (PORV)
(2) 500

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

C. (1) RC-V-1 (RCS Spray Valve)
(2) 540

D. (1) RC-V-1 (RCS Spray Valve)
(2) 500

Answer: A

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

10

ID: 1699640

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- Operating crew is performing Emergency Diesel Generator, EG-Y-1A, 24 hour endurance run in accordance with 1107-3.
- Currently EG-Y-1A is operating in parallel with the grid.

EVENT:

The following alarms are received simultaneously:

- Battery 1A Discharging, A-1-7.
- Battery Charger 1A/1C/1E Trouble, A-2-7.
- Inverter 1A/1C/1E System Trouble, A-3-7.
- CRDM Breaker Test Trouble, PRF 1-1-1.
- 230 KVolt Substation Trouble, NN-3-1.
- 7 KVolt Bus Trouble, AA-3-2.
- 4 KVolt BOP Bus Trouble, AA-3-3.
- 480 Volt BOP Bus Trouble, AA-3-5.

In order to protect EG-Y-1A the operating crew must:

G1-02 = EG-Y-1A Diesel output breaker to 1D 4160V Bus

EG-V-15A = Air Start Header Isolation Valve to EG-Y-15A

- A. Close EG-V-15A, only.
- B. Close EG-V-15A and trip EG-Y-1A fuel rack.
- C. Open breaker G1-02 from the control room and close EG-V-15A.
- D. Open breaker G1-02 using the local OPEN pushbutton and close EG-V-15A.

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) With the alarms present, there is a complete loss of 'A' DC; (2) when DC power is lost, EG-Y-1A loses its excitation field, in addition the ability to operate breakers associated with the 1D 4160V bus is lost (3) when DC is lost to EG-Y-1A with the generator breaker closed (parallel with the grid) the diesel will act as an inductive load and damage will occur (TQ-TM-104-861-COO1); (4) in addition, the air start solenoid will open which results in the air start distributor staying engaged, resulting in damage to the distributor.			
A.	Close EG-V-15A, only.	Plausible because if the examinee determines that a complete loss of 'A' DC occurred without regard to the current status of the diesel (which is parallel to the grid), then this would be the correct answer.	
B.	Close EG-V-15A and trip EG-Y-1A fuel rack.	Plausible because these actions are in OP-TM-AOP-023, which would be entered when 'A' DC was lost. Tripping the fuel racks is to shutdown the diesel, not for diesel protection. In addition, performing these steps would not open the diesel output breaker G1-02 which is causing the diesel to become an inductive motor.	
C.	Open breaker G1-02 from the control room and close EG-V-15A.	Plausible because these actions would protect the diesel. Incorrect because G1-02 must be opened locally at the breaker because 'A' DC control power was lost, and the control room extension control will not work.	
D.	Open breaker G1-02 using the local OPEN pushbutton and close EG-V-15A.	Correct answer - OP-TM-AOP-023 (step 4.15) directs the crew to close EG-V-15A. 1107-3 (Limit and Precaution 2.1.2) directs the breaker to be opened locally with the manual plunger.	
Examination Outline Cross-reference:			
Level		RO	SRO
Tier #		1	
Group #		1	
K/A #		058	AK3.01
Importance Rating		3.4	3.7
K/A: Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Use of dc control power by the D/Gs.			
Proposed Question: RO Question #10			
Technical Reference(s): 1107-3, Rev 146			
OP-TM-AOP-023, Rev 7			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 861-GLO-11			
Question Source: Bank #			
Modified Bank #			
New X			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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:			
<p>K/A Match: The question matches the K/A because the examinee will have to know what the effect of a loss of DC has on a running/paralleled diesel generator. The examinee will have to know that it effects the control power to the diesel generator breaker, in addition to the effects on the starting air solenoid.</p>			
<p>High Cog: The examinee will have to analyze the alarms present, and the present condition of the diesel (paralleled with the grid) to determine the steps needed to protect the diesel generator.</p>			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

11

ID: 1699734

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- NR-P-1C is out of service for maintenance.
- NR-P-1A is ES selected and running on 1R 480v bus.
- NR-P-1B is ES selected and running on 1T 480v bus.

EVENT:

- Loss of Offsite Power occurs.
- RCS pressure lowered to 1575 psig and is now rising slowly.
- RB pressure is 1.5 psig and relatively steady.

EVENT + 10 seconds:

- The Diesel Generators have started and are powering the ES Buses.

Which one of the following identifies the Nuclear River Water pump(s) that are running, if any, 1 minute after the re-energization of the ES Buses?

- A. NR-P-1A ONLY.
- B. NR-P-1B ONLY.
- C. NR-P-1A and NR-P-1B.
- D. No NR pump is running.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) When an loss of of offsite power occurs, all nuclear river water pumps shut off; (2) At an RCS pressure less than 1600 psig, and ES signal is generated; (3) When the diesels start and power the ES busses, and with an ES signal present, the remaining two Nuclear River Water pumps will start on Block 3.			
A.	NR-P-1A ONLY.	Plausible if the student believes a pump will start on standby. NR-P-1A would receive a start signal first if NR-P-1C tripped.	
B.	NR-P-1B ONLY.	Plausible because had the plant been in a normal lineup with NR-P-1A and NR-P-1C ES selected and running, then NR-P-1B would be the only running Nuclear River Water Pump after a loss of offsite power event (with no ES).	
C.	NR-P-1A and NR-P-1B.	Correct Answer	
D.	No NR pump is running.	Plausible because in this lineup, if no ES signal were present, then no NR pumps would be running.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	062	AK302
	Importance Rating	3.6	3.9

K/A: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of ESFAS.

Proposed Question: RO Question #11

Technical Reference(s): TQ-TM-104-531-C001, Rev 9
TQ-TM-104-740-C001, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: 740-GLO-10

Question Source: Bank # 298156
Modified Bank #
New

Question History: System Exam 6 Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 4

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

55.43

Comments:

K/A Match: This matches the K/A because the examinee has to understand what happens to the Nuclear River Water pumps on an Loss of Offsite power in conjunction with an ES actuation signal. The Nuclear River Water pumps are part of the Nuclear Service Water system, they provide cooling to the Nuclear Service Water Coolers.

High Cog: The examinee will have to analyze the current plant conditions: Only 2 nuclear river water pumps are available, and they that they are ES selected, that we have an Loss of Offsite power and that RCS pressure is 1575 psig, and understand that when RCS pressure is below 1600 psig, that an ES signal is present. Both nuclear service river water pumps trip when offsite power is lost and, both start on Block 3 ES after power is restored.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

12

ID: 1700681

Points: 1.00

<<REFERENCE PROVIDED

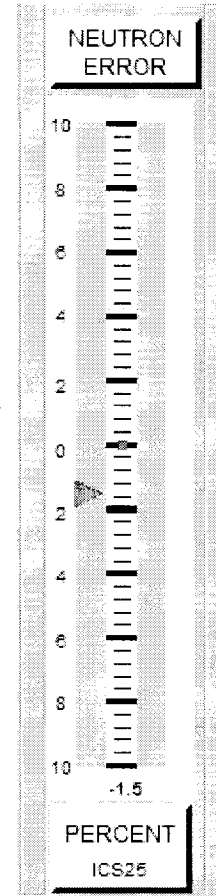
Plant Conditions

- Reactor power is 75% with ICS in auto.

EVENT:

- NI-5 power starts to slowly lower causing a SASS Mismatch.
- FW-V-17A, Main Feedwater Control Valve, starts to fail closed, causing a Feedwater to Reactor Cross Limit.
- Neutron Error is at -1.5% on console center.

Based on the above plant conditions, control rods will ____ (1) ____ due to ____ (2) ____.



- A. (1) insert
(2) NI-5 failing
- B. (1) insert
(2) Feedwater to Reactor Cross Limits
- C. (1) withdraw
(2) NI-5 failing
- D. (1) withdraw
(2) Feedwater to Reactor Cross Limits

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) NEGATIVE indicated neutron error results in control rod withdraw (Neutron Error = Reactor Demand - Actual Power); (2) NI-5 power drifting low causes a positive neutron error which is displayed on the console as a negative neutron error; (3) A SASS MISMATCH alarm prevents NI-5 from automatically swapping to the good instrument (NI-6); (4) NI-5 is the normally selected power range detector, with a SASS MISMATCH in, NI-5 will continue to be the controlling nuclear instrument, even though it is failing low; (5) The result is a situation where control rods will withdrawal to attempt to zero out neutron error, but as long as NI-5 continues to fail low, the negative neutron error will demand that control rod withdrawal; (6) Feedwater to Reactor Cross Limits cannot raise Reactor Power, therefore will not cause control rods to withdrawal.

A.	(1) insert (2) NI-5 failing	INCORRECT: NI-5 failure LOW would create a neutron error to the CRD System, while this error is positive (with respect to ICS actual power vs demanded power) the operator would see an inverted signal in which neutron error would appear negative and demand a control rod withdrawal. Rods inserting is plausible because this negative indication may cause the examiner to believe the control rods should be inserting.
B.	(1) insert (2) Feedwater to Reactor Cross Limits	INCORRECT: Control Rods would withdraw. The examinee may reason control rods would be inserting. Rods inserting is plausible because this negative indication may cause the examiner to believe the control rods should be inserting.
C.	(1) withdraw (2) NI-5 failing	CORRECT Answer: Indicated Neutron Error on console indication is reversed polarity to the actual neutron error going to the CRD System AND FW Cross Limits can only LOWER Reactor power, i.e. drive Control Rods IN.
D.	(1) withdrawal (2) Feedwater to Reactor Cross Limits	INCORRECT: Feedwater Cross Limits cannot raise Reactor Power, however with the indicated neutron error the operator may reason the Cross Limit is inducing this error.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	001	AA2.05
	Importance Rating	4.4	4.6

K/A: Continuous Rod Withdrawal Ability to determine and interpret the following as they apply to Continuous Rod Withdrawal: Uncontrolled rod withdrawal, from available indication

Proposed Question: RO Question #12

Technical Reference(s): ICS Analog Print: D553732

TQ-TM-104-621-C001, Rev 10

Proposed References to be provided to applicants during examination:

D553732

Learning Objective: 621-GLO-11

Question Source: Bank #

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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:

K/A Match: The conditions given in the question result in a Continuous Rod Withdrawal: indication of NI-5 failing low and Neutron Error. The operator must understand the indication and the final result of the multiple indications.

High Cog: The operator must understand the relationship between neutron error and the signal sent to the CRD System. While a negative error may imply Control Rods would insert, the ICS Signal would be positive and rods would withdraw. With a Feedwater to Reactor Cross Limit in effect the operator must understand the ICS does not allow this cross limit to raise reactor demand, only allows reactor demand to be lowered.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

13

ID: 1700738

Points: 1.00

Plant Conditions:

- 100% power 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.

Following the event, 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

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<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.

A.	(1) rise (2) lower	Correct. Pressurizer level instruments RC-1- LT-1 and LT-3, are manually selected on RC-1 LR recorder on CC. The level transmitters use a wet reference leg such that a full pressurizer has 0" DP, and an empty pressurizer has 400" DP. Consequently, a leak in the reference leg will reduce the differential pressure between the reference and the variable leg. As the differential pressure is reduced, indicated level will rise. Pressurizer level is maintained at 220" when the plant is operating at 100% full power. The level is sensed and applied to a level control circuit. The level control circuit when in the automatic mode will control the position of the normal makeup valve (MU-V-17). If the level drops below 220" the circuit will open the normal makeup valve until the levels returns to normal, and vice-versa. Consequently, when the reference leg leak occurs, the indicated level will start to rise, causing MU-V-17 to close. Since there is less makeup flow, pressurizer level will actually lower.
B.	(1) rise (2) remain the same	Incorrect. This is plausible because the operator may misunderstand the Pressurizer Level detector operation. If so, this is the exact opposite of what should happen. The examinee could believe that MU-V-17 would still maintain a pressurizer level of 220", which is not the case since it is receiving a signal to close due leaking reference leg.
C.	(1) lower (2) rise	Incorrect. This is plausible because the operator may incorrectly believe that when the leak is sensed that the level control will automatically shift to LT-3 similar to the SASS protected instrumentation. In addition, the examinee could believe that a leaking reference leg would make indicated pressurizer level to lower, which would cause MU-V-17 to open and pressurizer level to rise.
D.	(1) lower (2) remain the same	Incorrect. This is plausible because the operator may misunderstand the Pressurizer Level detector operation, and may incorrectly believe that when the leak is sensed that the level control will automatically shift to LT-3 similar to the SASS protected instrumentation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	028	AK1.01
	Importance Rating	2.8	3.1

K/A: Knowledge of the operational implications of the following concepts as they apply to Pressurizer Level Control Malfunctions: PZR reference leg leaks.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Proposed Question: RO Question #13

Technical Reference(s): TQ-TM-104-624-C001, Rev 4

TQ-TM-104-220-C001, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 624-GLO-5

Question Source: Bank # 862259

Modified Bank #

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.8

55.43

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.

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.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

14

ID: 1700748

Points: 1.00

Plant conditions:

- Plant startup is in progress.
- Source Range Channels indicate 100 cps.
- Group 5 rods are being withdrawn from 50% to 75%.

EVENT:

- Source Range Channel NI-11 fails low.

Based on this condition, the startup _____.

- A. may continue using Source Range Instrument NI-12.
- B. may continue using Source Range Instrument NI-11A.
- C. will be terminated AND the crew will initiate RCS Boration.
- D. will be terminated AND the crew will fully insert Group 5 and 4 control rods.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) There 2 independent source range channels that are part of the Reactor Protection System (NI-11 and NI-12); (2) These channels are required by Table 3.5-1 of Technical Specification 3.5.1.1; (3) At least one source range instrument is required to be operable until one intermediate range instrument is greater than 10 ⁻¹⁰ amps or 2 of 4 power range instruments are greater than 10% power; (4) If less than the required source range instruments are operable, the plant would be in a TS 3.0.1 situation.			
A.	may continue using Source Range Instrument NI-12.	CORRECT: Based on T.S. Requirements, Table 3.5-1	
B.	may continue using Source Range Instrument NI-11A.	INCORRECT:Plausible if the operator has the misconception that NI-11A can be used in place of NI-11. NI-11A is not part of the RPS system and is the installed spare that only provides indication.	
C.	will be terminated AND the crew will initiate RCS Boration.	INCORRECT: The startup is not required to be terminated. Plausible because if the operator did believe the startup must be terminated, the crew could choose to borate the RCS until they are shutdown to 1% dK/K.	
D.	will be terminated AND the crew will fully insert Group 5 and 4 control rods.	INCORRECT: The startup is not required to be terminated. Plausible since if a startup were aborted for unanticipated criticality, the crew would insert the group of control rods being pulled, and one additional group (which would be group 4 control rods in this scenario).	
Examination Outline Cross-reference:			
Level		RO	SRO
Tier #		1	
Group #		2	
K/A #		032	AK3.01
Importance Rating		3.2	3.6
K/A: Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: Startup termination on source-range loss			
Proposed Question: RO Question #14			
Technical Reference(s): TQ-TM-104-623-C001, Rev 5 Tech Specs Section 3.5.1.3			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 623-GLO-10			
Question Source: Bank # 355941			
Modified Bank #			
New			
Question History: NRC Exam 2000 Last NRC Exam: NO			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 K/A because the examinee must know the correlation Loss of SR NI during a startup and the required actions; continue the startup or terminate the startup.

High cog: The question is high cog because the examinee must know that the information in the answer explanation and analyze the event to determine the crew can still proceed with the startup.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

15

ID: 1737531

Points: 1.00

When required to verify a Radiation Monitoring System Setpoint, which is the PREFERRED METHOD for obtaining a Radiation Monitor's actual setpoints?

- A. Contacting the I&C Department to verify the setpoint.
- B. Performing a Source Check IAW 1301-4.1, Weekly Surveillance Checks.
- C. Reading Alarm setpoints from the meter face by pressing the alarm pushbutton.
- D. Refer to the setpoints section of alarm response OP-TM-MAP-C0101, RADIATION LEVEL HI.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) 1105-8, Radiation Monitoring System a Caution states **"Reading Alarm setpoints from the meter face by pressing the Alarm Pushbutton does not give an accurate indication of actual setpoint. This feature is not calibrated and should not be relied upon for accurate setpoint information"**. (2) Following that caution a procedure step directs setpoint verification by directing I&C department to refer to applicable sections of 1302-3.1 series procedures that govern RMS Monitors covered by Tech Specs or ODCM and IC-177, RMS Calibration for Non-Tech Monitors.

A.	Contacting the I&C Department to verify the setpoint.	CORRECT ANSWER: 1105-8, Radiation Monitoring System Caution and procedure step
B.	Performing a Source Check IAW 1301-4.1, Weekly Surveillance Checks.	INCORRECT ANSWER: Plausible because performing a source check will bring in the alarms. However, Tech Spec definition 1.13 defines a source check in the following way: A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source. It does not check the actual setpoints of a radiation monitor.
C.	Reading Alarm setpoints from the meter face by pressing the alarm pushbutton.	INCORRECT ANSWER: Plausible because setpoints can be observed when using the pushbutton, however 1105-8 cautions against relying upon it for accurate setpoint information.
D.	Refer to the setpoints section of alarm response OP-TM-MAP-C0101, RADIATION LEVEL HI.	INCORRECT ANSWER: Plausible because there is a setpoints section of this alarm response. Incorrect because this section has no setpoints listed and sends the operator to 1101-2.1, RADIATION MONITORING SYSTEM SETPOINTS.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	061	AA2.03
	Importance Rating	3.0	3.3

K/A: Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Setpoints for alert and high alarms

Proposed Question: RO Question #15

Technical Reference(s): 1105-8, Rev 91
1302-3.1 series procedures

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-9

Question Source: Bank # 887632
Modified Bank #

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

New

Question History: Simulator Exam 9 Last NRC Exam: None

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 matches the KA because the question requires the examinee to determine the method to correctly determine the high setpoint on a radiation monitor. The setpoint on the front of the radiation monitor is not calibrated, and cannot be used to correctly verify the high setpoint.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

16

ID: 1700780

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto..

EVENT:

- MAP C-1-1, RADIATION LEVEL HI alarm illuminates.
- RM-L-1-LO (RCS Letdown Rad Monitor) indicates a HIGH ALARM.
- RM-L-1-HI (RCS Letdown Rad Monitor) indicates an ALERT ALARM.

Based on the given conditions, letdown flow ____ (1) ____ isolated, because ____ (2) ____.

- A. (1) IS
(2) RM-L-1-LO HIGH alarm closes MU-V-1A and MU-V1B, Letdown Cooler Inlet valves
- B. (1) IS
(2) RM-L-1-LO HIGH alarm closes MU-V-2A and MU-V-2B, Letdown Cooler Outlet valves
- C. (1) IS NOT
(2) both RM-L-1-LO AND RM-L-1-HI have to be in HIGH Alarm
- D. (1) IS NOT
(2) RM-L1-LO HIGH alarm does NOT close MU-V-2A and MU-V-2B

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) RM-L-1-LO and RM-L-1-HI are not the same radiation monitor; (2) RM-L-1-LO has no interlocks; (3) RM-L-1-HI closes MU-V-2A/2B on a HIGH alarm, but not an ALERT alarm; (4) MU-V-1A and MU-V-1B do isolate letdown but there is no interlock between either of the RM-L-1 detectors.

A.	(1) IS (2) RM-L-1-LO HIGH closes MU-V-1A and MU-V1B, Letdown Cooler Inlet valves.	INCORRECT: Plausible since the HIGH alarm on some radiation monitors do initiate automatic interlocks, however RM-L-1-LO does NOT have any interlocks associated with its HIGH Alarm. Closing MU-V-1A&1B would isolate letdown flow. However these valves are closed by other interlocks independent of RM-L-1-LO or RM-L-1-HI.
B.	(1) IS (2) RM-L-1-LO HIGH alarm closes MU-V-2A and MU-V-2B, Letdown Cooler Outlet valves.	INCORRECT: Plausible since the High Alarm on some radiation monitors do initiate Automatic Interlocks, however RM-L-1-LO does NOT have any interlocks associated with its HIGH Alarm. RM-L-1-HI HIGH alarm closes MU-V-2A and MU-V-2B.
C.	(1) IS NOT (2) Both RM-L-1-LO AND RM-L-1-HI have to be in HIGH Alarm. MU-V-2A and MU-V-2B	INCORRECT: Letdown Flow is NOT isolated, BOTH RM-L-1-LO AND RM-L-1-HI do not need to be in HIGH alarm to isolate letdown, just RM-L-1-HI needs to be in HIGH. Plausible because the examinee could believe that a combination of alarms are required to isolate letdown.
D.	(1) IS NOT (2) RM-L1-LO "High Alarm" does NOT close MU-V-2A and MU-V-2B.	CORRECT: Letdown Flow IS NOT isolated because MU-V-2A and MU-V-2B remain open since they do NOT close from RM-L-1-LO being in HIGH Alarm.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	076	AA1.04
	Importance Rating	3.2	3.4

K/A: High Reactor Coolant Activity: Ability to operate and / or monitor the following as they apply to the High Reactor Coolant Activity: Failed fuel-monitoring equipment

Proposed Question: RO Question #16

Technical Reference(s): TQ-TM-104-661-C001
1105-8

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-10

Question Source: Bank #
Modified Bank #
New X

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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:

K/A Match: This question matches the KA because the examinee is required to recall the interlocks of RM-L-1-Hi which is the radiation monitor that is used to detected failed fuel. RM-L-1-HI takes a stream off the letdown line, which is RCS water going to the makeup tank.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

17

ID: 1720243

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- The ICS Selected Feedwater Loop Temperature transmitter slowly fails high.

The Total Feedwater demand signal will ____ (1) ____ due to ____ (2) ____.

- A. (1) rise
(2) BTU limits
- B. (1) rise
(2) Feedwater Temperature Modification
- C. (1) lower
(2) BTU limits
- D. (1) lower
(2) Feedwater Temperature Modification

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1)There is NO effect on BTU Limits since the transmitter failed high; (2) The operation of the FW temperature circuit compares actual FW temperature (Failed Temperature Transmitter) to a calculated temperature based on Total FW Demand; (3) In this question the circuit would raise FW Demand since the control system thinks Actual FW Temperature is hotter than calculated thus raise FW Flow to match FW Flow heat removal capacity to match Reactor heat output.

A.	(1) rise (2) BTU limits	INCORRECT: Plausible since FW Temperature is an input to the BTU Limit Circuit, however, with FW temperature failing high has no effect. If FW temperature had failed low, it would reduce FW Flow.
B.	(1) rise (2) Feedwater Temperature Modification	CORRECT: FW temperature Modification is used to correct FW demand for a given load such that there is balance of BTU exchange between the primary and secondary side of the steam generator. So, if FW temp is higher than the calculated expected average FW temp than FW demand signal will rise.
C.	(1) lower (2) BTU limits	INCORRECT: Plausible since FW temperature is an input to the BTU limit circuit and would lower FW flow on a low failure.
D.	(1) lower (2) Feedwater Temperature Modification	INCORRECT: Plausible if Examinee does NOT understand how the control system process compares actual FW temperature with calculated FW temperature.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	A02	AK2.1
	Importance Rating	3.8	4.0

K/A: Loss of NNI-X/Y: Knowledge of the interrelations between the (Loss of NNI-X) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: RO Question # 17

Technical Reference(s): OP-TM-104-621-C001, Rev 10 ICS FW Analog & Digital Prints

Proposed References to be provided to applicants during examination: NONE

Learning Objective: 621-GLO-11

Question Source: Bank # 357040
Modified Bank #
New

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 how a Feedwater Temperature failure affects the ICS system for automatic control.

High Cog: This question is High Cog because the examinee has to understand the control system interface, with respect to Main FW control. The examinee must know the control system compares Actual FW Temperature to Calculated Temperature and analyze the effects of a failure.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

18

ID: 1737211

Points: 1.00

To avoid damage due to gas in the Makeup Pumps, ____ (1) ____ must be closed if Makeup Tank level lowers to less than ____ (2) ____ inches with suction from LPI or the BWST.

- A. (1) MU-V-18, RCS Makeup Isolation Valve
(2) 18 inches
- B. (1) MU-V-12, Makeup Tank Outlet Isolation Valve
(2) 18 inches
- C. (1) MU-V-18, RCS Makeup Isolation Valve
(2) 40 inches
- D. (1) MU-V-12, Makeup Tank Outlet Isolation Valve
(2) 40 inches

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-211-000 Precautions & Limitations: To avoid pump damage due to gas entrainment in the pump, MU-V-12 must be closed if MU Tank level is less than 18 inches and LPI or BWST is supplying MU Pump Suction; (2) When a Makeup Pump is lined up for suction from LPI or the BWST, the only separation between those sources and the Makeup Tank is MU-V-112, which is a check valve; (3) The Makeup Pump source will be from the source with the highest pressure, which in some instances (based on BWST level or LPI flow) could the the Makeup Tank, which has a low inventory < 18 inches, could cause gas to be admitted to the Makeup Pump.			
A.	(1) MU-V-18, RCS Makeup Isolation Valve (2) 18 inches	INCORRECT: Plausible if the examinee thinks it is necessary to isolate other Makeup Pump flowpaths to prevent possible Makeup Pump Runout. Part 2 is correct.	
B.	(1) MU-V-12, Makeup Tank Outlet Isolation Valve (2) 18 inches	CORRECT: See above	
C.	(1) MU-V-18, RCS Makeup Isolation Valve (2) 40 inches	INCORRECT: Plausible because the minimum level to ensure the Makeup Pump has adequate NPSH when the suction source is the Makeup Tank exclusively is 40". If NPSH were a concern, the examinee could believe lowering the flow by closing MU-V-18 could fix the NPSH issue.	
D.	(1) MU-V-12, Makeup Tank Outlet Isolation Valve (2) 40 inches	INCORRECT: Plausible since Part 1 is correct and thinks that 40" is the correct setpoint.	
Examination Outline Cross-reference:			
Level		RO	SRO
Tier #		2	
Group #		1	
K/A #		004	A1.06
Importance Rating		3.0	3.2
K/A: Chemical and Volume Control System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CVCS controls including: VCT level			
Proposed Question: RO Question #18			
Technical Reference(s): OP-TM-211-000, Rev 33			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 211-GLO-14			
Question Source: Bank #			
Modified Bank #			
New		X	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 know that the lowering level in the Makeup Tank (VCT) could exceed a design limit for the CVCS system.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

19

ID: 1737168

Points: 1.00

REFERENCE PROVIDED

Plant Conditions:

- Plant Cooldown in Progress
- All 4 Reactor Coolant Pumps are OFF
- DH-P-1A is Operating.
- RCS Temperature = 170F
- RCS Pressure = 250 psig

Curve (1) prevents exceeding LTOP limits. The temperature instrument monitored to prevent exceeding these limits is (2).

- A. (1) Curve A
(2) DH6-TI-1 Decay Heat Suction Temperature
- B. (1) Curve B
(2) DH6-TI-1 Decay Heat Suction Temperature
- C. (1) Curve A
(2) DH2-TI-1 Decay Heat Cooler Outlet Temperature
- D. (1) Curve B
(2) DH2-TI-1 Decay Heat Cooler Outlet Temperature

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) The 1102-11 Limits and Precautions directs reactor coolant temperature to be maintained within limits specified in Enclosure 4, Figure 1 and 1A one of which is the LTOP and SURGE Line Limit (CURVE A); (2) The operators must ensure the pressure and temperature remain to the below and to the right of CURVE A; (3) The examinee must know the correct instruments to monitor to ensure the cooldown rate and LTOP limits are not violated, which for these plant conditions (RCPs secured, while on DHR) is DH2-TI1 or 2; (4) CURVE B is the Spray Restriction line.

A.	(1) Curve A (2) DH6-TI-1 Decay Heat Suction Temperature	INCORRECT: Plausible since the operator may assume DH Suction Temperature used during a cooldown. Incorrect because after RCPs are secured DH Cooler Outlet Temp is used. Curve A is correct.		
B.	(1) Curve B (2) DH6-TI-1 Decay Heat Suction Temperature	INCORRECT: Plausible since the operator may assume DH Suction Temperature used during a cooldown. Incorrect because after RCPs are secured DH Cooler Outlet Temp is used. Curve B is incorrect.		
C.	(1) Curve A (2) DH2-TI-1 Decay Heat Cooler Outlet Temperature	CORRECT: See above		
D.	(1) Curve B (2) DH2-TI-1 Decay Heat Cooler Outlet Temperature	INCORRECT: Plausible since the temperature indication is correct but CURVE B is the Spray Restriction Curve.		

Examination Outline Cross-reference:	Level	RO	SRO	
	Tier #	2		
	Group #	1		
	K/A #	005	K5.01	
	Importance Rating	2.6	2.9	

K/A: Residual Heat Removal System: Knowledge of the operational implications of the following concepts as they apply the RHRS: Nil ductility transition temperature (brittle fracture)

Proposed Question: RO Question #19

Technical Reference(s): TQ-TM-104-212-C001, Rev 17 Tech Spec 3.1.2
1102-11, Rev 153B

Proposed References to be provided to applicants during examination: 1102-11 Fig 1A

Learning Objective: 212-GLO-10

Question Source: Bank #
Modified Bank #
New X

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 reference curves in 1102-11 are developed in ensure that NDT and cyclic stress limits are not violated. The examinee must know which instrument is used to ensure the LTOP curve A is NOT violated.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

20

ID: 1700999

Points: 1.00

Plant conditions:

- Plant Cooldown is in progress in accordance with 1102-11.
- RCS pressure is 60 psig and steady prior to venting.
- OP-TM-220-552, VENTING THE PRESSURIZER TO THE RCDT (WDL-T-3), has been initiated to lower pressurizer pressure.

To prevent large pressure transients during the Pressurizer venting, the operator must _____

- A. open RC-V-18, Manual Pzr Vent to RCDT.
- B. throttle open RC-V-28, Pressurizer Vent Valve, for 1 second.
- C. throttle open RC-V-1, Spray Valve, to help lower Pressurizer pressure.
- D. ensure RCDT cooling pump (WDL-P-8) is in AUTO with both ICCW Pumps running.

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) During and plant cooldown, to lower RCS pressure in the latter stages, the pressurizer is vented to the RCDT; (2) To limit pressure and/or level transients, RC-V-28 is throttled open in small amounts (1 second per procedure); (3) At 60 psig in the cooldown, no RCPs are running so spray from RC-V-1 is not available.			
A.	open RC-V-18, Manual Pzr Vent to RCDT.	INCORRECT: Plausible since this is a vent valve to the RCDT; however it is not allowed to be open when the RCS is > 45 psig to prevent overpressurizing the RCDT.	
B.	throttle open RC-V-28, Pressurizer Vent Valve, for 1 second.	CORRECT: A CAUTION in OP-TM-220-552 says to use small taps on RC-V-28 open pushbutton to avoid large pressure or level transients. The procedure step is to throttle open for 1 second.	
C.	throttle open RC-V-1, Spray Valve, to help lower Pressurizer pressure.	INCORRECT: Plausible since RC-V-1 is the normal spray valve; however the last RCP was shut down prior to the venting operation being initiated.	
D.	ensure RCDT cooling pump (WDL-P-8) is in AUTO with both ICCW Pumps running.	INCORRECT: Plausible since two ICCW Pumps will be run if the RCS temperature was >400°F; however WDL-P-8 will be started and run continuously during the venting operation and the RCS temperature cannot be > 400°F based on the given condition of 60 psig in the RCS.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007	2.1.23
	Importance Rating	4.3	4.4
K/A: Ability to perform specific system and integrated plant procedures during all modes of plant operation.			
Proposed Question: RO Question # 20			
Technical Reference(s): OP-TM-220-000, Rev 23			
OP-TM-220-552, Rev 6			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 220-GLO-10			
Question Source: Bank # 8672881			
Modified Bank #			
New			
Question History: N/A Last NRC Exam: N/A			
Question Cognitive Level: Memory or Fundamental Knowledge X			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 b.10
55.43

Comments:

K/A Match: This question matches the KA because the examinee will have to demonstrate knowledge regarding the ability to mitigate large pressure transients during and RCS cooldown. The examinee will have to have knowledge of the procedure to vent the Pressurizer to the RCDT.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

21

ID: 1701024

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- EFW Area Cooling Fan, AH-E-24A, is operating.
- EFW Area Cooling Fan, AH-E-24B, is in standby.
- Emergency Feedwater Pump, EF-P-2A is running for an IST.

EVENT:

- Nuclear Service Closed Cooling Water Inlet Temperature Controller to AH-E-24A Coolers, TC-857, Fails LOW.

As a result of this controller failure, identify the resulting effect(s):

AH-E-24A trips due to high ____ (1) ____ temperature, and AH-E-24B ____ (2) ____.

- A. (1) outlet
(2) automatically starts
- B. (1) outlet
(2) can be started manually
- C. (1) motor
(2) automatically starts
- D. (1) motor
(2) can be started manually

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) AH-E-24A/24B have NO autostart features; (2) Fan outlet temperature is monitored by a temperature Switch TS-766A/766B; (3) The fan Motor does NOT have a Temperature Switch; (4) Since the NSCCW Temperature Controller failed LOW, the affect would reduce NSCCW Flow to the fan coolers resulting in fan outlet temperature rising (sensed by TS-766A); (5) At 125F the operating Fan would trip; (6) Alarm response HVA-5-8 would direct starting of the standby fan as long as there is no fire present (the stem of the question gives no indication of a fire).

A.	(1) outlet (2) automatically starts	INCORRECT: With the NSCCW Temperature Controller failed low, NSCCW inlet valve would close which would reduce cooling flow to the Fan Cooling Coils. This would result in fan discharge temperature rising. At 125F the fan would trip. AH-E-24B does not have an autostart feature. This is plausible because many components have autostart features on temperature and/or a trip.
B.	(1) outlet (2) can be started manually	CORRECT: With the NSCCW temperature controller failed low, NSCCW inlet valve would close which reduces cooling flow to the fan cooling coils resulting in fan discharge temperature rising. At 125F the fan would trip. AH-E-24B does not have an autostart feature and could be manually started.
C.	(1) motor (2) automatically starts	INCORRECT: Plausible since an operator may believe that the fan would trip on high motor temperature as opposed to fan discharge temperature. Since AH-E-24B is in standby, it would have to be manually started since there is no autostart feature on these fans.
D.	(1) motor (2) can be started manually	INCORRECT: Plausible since an operator may believe that the fan would trip on high motor temperature as opposed to fan discharge temperature. Since AH-E-24B is in standby, it could be manually started since there is no autostart feature on these fans.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	008	A4.09
	Importance Rating	3.0	2.9

K/A: Component Cooling Water System: Ability to manually operate and/or monitor in the control room: CCW temperature control valve

Proposed Question: RO Question # 21

Technical Reference(s): HVA-5-8 Rev 9
TQ-TM-104-821-C001, Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: 821-GLO-010

Question Source: Bank #

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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:	
<p>KA Match: This question matches the KA because the examinee will have to know that the operator has to manually operate AH-E-24B from the control room if the temperature control valve for the CCW to the AH-E-24A fan were to fail.</p>	
<p>High Cog: This question is high cog because the examinee has to identify the effect of the temperature controller failing low and the response of the associated control valve. This would result in the control valve closing, reducing cooling flow. The fan outlet temperature would rise, which is the signal monitored to for trip the operating fan.</p>	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

22

ID: 1701070

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Pressurizer heater pressure control SETPOINT (RC3-PIC) signal fails to zero (0) psig.

What action(s) are required?

- A. Place the Diamond Rod Control station in manual and insert control rods.
- B. Place the Pressurizer Spray Valve, RC-V-1 in Manual and CLOSE RC-V-1.
- C. Verify all pressurizer heaters are deenergized OR manually deenergize heaters.
- D. Adjust Heater Banks 1, 2 & 3 using RC3PIC, Pressurizer Pressure control in hand.

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) That the SETPOINT signal only affects PZR Heater Banks 1, 2, & 3 while the controller is in auto (2) the controller failing to 0 psig means heaters will not start at the desired setpoint (3) the crew can place the Pzr Controller in HAND per OS-24 and operate per OP-TM-220-503, "Manual Control of the Pzr" at that point the pressure lowering transient is concluded. Even though OP-TM-220-503 does NOT address instrument failures, control of RCS pressure is ensured more quickly and smoothly by entering and performing this procedure. (4) The mental process involved is that RCS pressure will lower if heaters are not on to account for losses.			
A.	Place the Diamond Rod Control station in manual and insert control rods.	INCORRECT - Plausible if the analysis of the transient were to mistakenly lead to the determination that ALL the requirement for entry into OP-TM-AOP-070 were met.	
B.	Place the Pressurizer Spray Valve, RC-V-1 in Manual and CLOSE RC-V-1.	INCORRECT - Plausible if the analysis of the failure causes the examinee to believe that the spray valve could erroneously open. This could happen on other pressure instrument/controller failures.	
C.	Verify all pressurizer heaters are deenergized OR manually deenergize heaters.	INCORRECT - Plausible if the analysis of the failure causes the examinee to believe that the setpoint failing low energizes the heaters. This could happen on other pressure instrument/controller failures.	
D.	Adjust Heater Banks 1, 2 & 3 using RC3PIC, Pressurizer Pressure control in hand.	CORRECT - See above.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027	AA1.01
	Importance Rating	4.0	3.9
K/A: Pressurizer Pressure Control System Malfunction: Ability to operate and / or monitor for the following as they apply to the Pressurizer Pressure Control Malfunctions: PZR heaters, sprays, and PORVs			
Technical Reference(s): OP-TM-220-503, Rev 4			
Proposed Question RO Question #22			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 220-GLO-11			
Question Source: Bank # 897217			
Modified Bank #			
New			
Question History: Last NRC Exam: N/A			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

KA MATCH: This is KA match because the operate will have to understand that heaters must be operated in manual with this controller failure.

HIGH COG: This is High Cog because they will have to realize the heater setpoint failing to zero will only effect the pressurizer heaters and then that manual operation is required to control the plant.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

23

ID: 1701075

Points: 1.00

Initial plant conditions:

- Reactor trip due to LOCA.
- Containment Building pressure 5 psig, steady.
- Manual ESAS actuation signals were NOT initiated.
- Automatic ES Actuation status:

ES Actuation	Train A	Train B
1600#	Actuated (NOT Bypassed)	Bypassed
500#	Not Actuated	Not Actuated
4#	Defeated	Actuated (NOT Defeated)

EVENT:

- Containment Building pressure rises rapidly to 35 psig.

Based on these conditions, identify the ONE statement below that describes the response of the Reactor Building Spray System

- A. Only BS-P-1A starts.
- B. Only BS-P-1B starts.
- C. Both BS-P-1A and BS-P-1B start.
- D. Neither BS-P-1A nor BS-P-1B starts.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) RB Pressure of 30 psi is ONE input to the Start Signal for a building spray pump; (2) The other requirement is a Block 4 permissive signal from any or the following ESAS Actuation signals: 1600 psig OR 500 psig OR 4 psig; (3) While the 4 psig signal is the only signal that opens the BS valves the other signals (1600 psig & 500 psig) input to the Block 4 permissive signal; (4) The BS Pump start signal does not require the BS valves to be open for the pump to start.

A.	Only BS-P-1A starts.	INCORRECT: Distracter is plausible because it acknowledges BS-P-1A starts due to ES Block 4 permit from 1600 psi. This distracter is based on the misconception that BS pumps require a Block 4 start permit (from 1600/500# ES actuation) in order to automatically start at 30 psig.
B.	Only BS-P-1B starts.	INCORRECT: Distracter is plausible because it acknowledges that BS-P-1B will start. This distracter is based upon the common misconception that BS pumps require a Block 4 start permit from 4# ES actuation (opens BS valves) in order to start at 30 psig.
C.	Both BS-P-1A and BS-P-1B start.	CORRECT: Both BS Pumps start since both the 1600 psi (Train A) and 4 psi (Train B) signals provide Block 4 permissive signals to enable the BS Pumps to start at 30 psi.
D.	Neither BS-P-1A and BS-P-1B start.	INCORRECT: Plausible since Part of Train A (4 psi) is in Defeat and part of Train B (1600 psi) is Bypassed preventing Block 4 BS Pump Permissive Signal.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013	K1.01
	Importance Rating	4.2	4.4

K/A: Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: Initiation signals for ESF circuit logic

Proposed Question: RO Question #23

Technical Reference(s): TQ-TM-104-642-C001, Rev 7 ESAS Electrical Prints

Proposed References to be provided to applicants during examination: None

Learning Objective: 642-GLO-10

Question Source: Bank # 371299
Modified Bank #
New

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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, why high cog, why SRO only)			
KA MATCH: This question matches the KA because the examinee must have knowledge of the initiation signals which start the Reactor Building Spray pumps, which is an ES system.			
HIGH COG: This question is High Cog because the examinee must analyze the reactor building conditions and the actuated/defeated ES signals to determine which (if any) reactor building spray pumps start.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

24

ID: 1701092

Points: 1.00

REFERENCE PROVIDED

Plant Conditions:

- 100% power with ICS in full auto.
- RB Cooling Fans AH-E-1A/B/C are all operating in FAST SPEED.
- Main Steam System leak inside the Containment Building.
- RB Temperature Elevation > 320 is at 132F and slowly rising.
- NN-2-7 RB AIR TEMP HI is in Alarm
- RB Pressure 2.3 psig

Event:

- AH-E-1A trips due to THERMAL OVERLOAD (49F).
- Thermal overload condition has NOT been reset at AH-E-1A breaker.

Based on these conditions, AH-E-1A ___(1)___ automatically start if ES actuates and IAW OP-TM-534-901, RB Emergency Cooling Operations, AH-E-1A must be started in ___(2)___ speed.

- A. (1) will
(2) slow
- B. (1) will not
(2) slow
- C. (1) will
(2) fast
- D. (1) will not
(2) fast

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) Separate thermal overload contacts exist for slow and fast speed operation (see reference 208-561, 49F/S contacts in blocks B6 and B5); (2) Operation at slow speed would not be affected by the fast speed thermal overload contact tripping; (3) ES Actuation signal will start the fan in slow speed and is not effected by the fast speed thermal overload contact; (4) Since RB Pressure is > 2 psig, the procedure requires the fan be manually started in Slow Speed (6) in addition, have the ability to interpret station electrical drawings.			
A.	(1) will (2) slow	CORRECT: Refer to 208-561. Two independent thermal overload contacts, one for slow speed operation and one for fast speed operation. ES Actuation starts fans in slow speed. Starting the fan in slow speed is required per the procedure.	
B.	(1) will not (2) slow	INCORRECT: Distracter is plausible because examinee may think the thermal overload blocks the ES Actuation signal. AH-E-1A fan will auto start in slow speed if ES actuates, since they are separate thermal overload protection for the fast and slow speed motor contactors. AH-E-1A fan can be manually started in slow speed.	
C.	(1) will (2) fast	INCORRECT: Plausible since the fan will auto start in slow speed if ES actuates. Plausible if the examinee does not understand that if RB Pressure is >2 psig the procedure requires starting the fan in slow speed.	
D.	(1) will not (2) fast	INCORRECT: AH-E-1A fan can be manually started in SLOW speed, and will auto start in slow speed if ES actuates. The examinee may think the thermal overload blocks the ES Actuation. Plausible if the examinee does not understand that if RB Pressure is >2 psig the procedure requires starting the fan in slow speed..	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022	A2.03
	Importance Rating	2.6	3.0
K/A: Containment Cooling System: Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Fan motor thermal overload/high-speed operation			
Proposed Question:		RO Question #24	
Technical Reference(s):		Electrical Print 208-561, 208-461	MAP NN-2-7 alarm response, Rev 2
		OP-TM-534-901 Rev 14	
Proposed References to be provided to applicants during examination:		Electrical Prints 208-561 208-461	
Learning Objective:		824-GLO-10	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question Source: Bank #

Modified Bank # 371313

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 be able to predict the impact of an actuated thermal overload on the normal and ES operation of the RB Emergency cooling fans and identify the procedure requirement for starting the fan based on RB Pressure.

HIGH COG: This question is high cog because the examinee must analyze the given station drawing to predict the impact of the actuated thermal overload and identify the procedure requirement for the given plant conditions.

Plant conditions:

- Reactor is operating at 100% power, with ICS in full automatic.
- RB Cooling Fans AH-E-1A/B/C are all operating in FAST SPEED.
- Main Steam System leak inside the Containment Building.

Event:

- AH-E-1A trips due to THERMAL OVERLOAD.
- Thermal overload condition has NOT been reset at AH-E-1A breaker.

Based on these conditions, identify the ONE selection below that describes impact on AH-E-1A operations:

- (1) Ability to manually start AH-E-1A in SLOW speed;
- (2) Automatic ES operation.
- (3) Reason for impact.

- A. (1) Can be manually started in slow speed.
(2) Will automatically start if ES actuates.
(3) Thermal overload protection only applies to FAST speed operation.
- B. (1) Can be manually started in slow speed.
(2) Will automatically start if ES actuates.
(3) SEPARATE thermal overload protection is provided for the high and slow speed motor contactors.
- C. (1) CANNOT be manually started in slow speed.
(2) Will automatically start if ES actuates.
(3) COMMON thermal overload protection (for fast AND slow speeds) is DEFEATED by ES actuation.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

- D. (1) CANNOT be manually started in slow speed.
(2) Will NOT automatically start if ES actuates.
(3) COMMON thermal overload protection (for fast AND slow speeds) is NOT defeated by ES actuation.

Answer: B

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

25

ID: 1718554

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- OP-TM-621-251 SASS Logic Test in Progress.
 - Test 'A' on NI-5 / NI-6 Rx Pwr was successful, however the SASS channel FAILED to reset.
- NI-5 is selected for control.

EVENT:

- NI-5 upper chamber power supply fails to zero volts.

The crew ____ (1) ____, because ____ (2) ____.

- A. (1) must immediately enter OP-TM-AOP-070, Primary to Secondary Heat Transfer Upset
(2) reactor power would be lowering
- B. (1) must immediately enter OP-TM-AOP-070, Primary to Secondary Heat Transfer Upset
(2) reactor power would be rising
- C. (1) must immediately enter OP-TM-EOP-001, Reactor Trip
(2) the NI-5 failure would cause a reactor trip
- D. (1) would have NO required actions
(2) the SASS would automatically transfer to NI-6 for reactor control

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) When the SASS channel fails to reset, that the SASS channel would not swap to a good instrument on a failure of the controlling instrument; (2) When the NI-5 upper chamber power supply fails to zero volts, the NI-5 indicated power drops to 50%; (3) Because the SASS is failed, when NI-5 fails low, the an ICS transient would occur that pulls control rods to maintain reactor power; (4) When control rods pull, reactor power would rise; (5) NI-5 could reach an RPS trip setpoint, but this would be the only immediate reactor trip signal; (6) To mitigate this plant transient, the crew would enter OP-TM-AOP-070 because a plant transient would be occurring requiring manual operation, and no valid ICS runback is present.

A.	(1) must immediately enter OP-TM-AOP-070, Primary to Secondary Heat Transfer Upset (2) reactor power would be lowering	INCORRECT: The entry criteria for OP-TM-AOP-070 will be met, but reactor power would be rising. This is plausible if the examinee believes NI-5 failing low would result in plant runback.
B.	(1) must immediately enter OP-TM-AOP-070, Primary to Secondary Heat Transfer Upset (2) reactor power would be rising	CORRECT: See above.
C.	(1) must immediately enter OP-TM-EOP-001, Reactor Trip (2) the NI-5 failure would cause a reactor trip	INCORRECT: Plausible if examinee believes if the examinee believes the NI-5 failure input to RPS would initiate a reactor trip which would require entry into OP-TM-EOP-001. If OP-TM-AOP-070 was not entered, a reactor trip could occur, but this failure in itself would not cause a reactor trip.
D.	(1) would have NO required actions (2) the SASS would automatically transfer to NI-6 for reactor control	INCORRECT: Plausible because this is how the system should work if the NI-5 upper chamber power supply failed low. Incorrect because the SASS channel failed to reset in the stem of the question.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	015	A2.01
	Importance Rating	3.5	3.9

K/A: Nuclear Instrumentation System: Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power supply loss or erratic operation

Proposed Question: RO Question # 25

Technical Reference(s): TQ-TM-104-621-C001 OP-TM-AOP-070, Rev 5
OP-TM-621-251 Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: 621-GLO-10

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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

Comments:

KA Match: This question matches the KA because the examinee must know the effects of a failing power supply on plant operations, and choose the procedure to mitigate the failure.

High Cog: This question is high cog because the examinee must analyze the effect of the failure and determine plant response and determine that the response will meet the threshold for the entry criteria of OP-TM-AOP-070.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

26

ID: 1720398

Points: 1.00

Which one of the following describes the operation of the Main Feedwater Pump Monitoring circuit that provides Emergency Feedwater with an input from the Heat Sink Protection System?

EACH PUMP uses:

- A. Two pressure bistables to sense Main Feedwater Pump discharge pressure.
- B. Three pressure bistables to sense Main Feedwater Pump discharge pressure.
- C. Two pressure bistables to sense hydraulic oil pressure at the Main Feedwater Pump turbine stop valves.
- D. Three pressure bistables to sense hydraulic oil pressure at the Main Feedwater Pump turbine stop valves.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question History:

Last NRC Exam: 10-02

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, why high cog, why SRO only)

KA Match: This question matches the KA because the operator must demonstrate knowledge of the physical connections and/or cause-effect relationships between the EFW and the MFW System. This is accomplished by requiring that the operator identify the number of bistables used in the MFW Pump monitoring circuit, and the parameter that is sensed to determine that the MFW Pumps are tripped to automatically start the EFW Pumps.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

27

ID: 1737222

Points: 1.00

Plant conditions:

- Reactor critical at 10^{-8} amps.
- Crew is performing a middle of cycle startup from a 3 week forced outage.
- ICS in HAND, EXCEPT for the Main & Startup Feedwater Valves and Turbine Bypass Valves.
- Power escalation is on hold; will recommence in one hour.

EVENT:

- MS-V-3D, Turbine Bypass Valve fails OPEN

Assume NO Operator actions.

Which statement below describes the plant response to this failure?

- A. RCS pressure lowers and the Reactor trips on low pressure.
- B. Reactor power increases to the 'Point of Adding Heat' and stabilizes at ~1%.
- C. Reactor power increases above the 'Point of Adding Heat' and then stabilizes at ~3%.
- D. Reactor power increases to the 'Point of Adding Heat' and then lowers to 10^{-8} Amps in the Intermediate Range.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) With the plant stable at 10E-8 AMPS on the IR and when TBV fails open, this would induce a positive reactivity feedback due to the Moderator Temperature Coefficient; (2) Reactor Power would rise to the Point of Adding Heat and continue to rise as T-ave lowers; (3) Eventually the power rise will stop and begin to lower as T-ave begins to rise; (3) Reactor power will stabilize at a lower power (~3%), based on the capacity of the TBV.

A.	RCS pressure lowers and the Reactor trips on low pressure.	INCORRECT: Plausible since RCS pressure will lower due to the overcooling, however, RCS pressure does NOT reach the low pressure trip setpoint with a single TBV failing open.
B.	Reactor Power increases to the 'Point of Adding Heat' and stabilizes at ~1% Reactor Power.	INCORRECT: Plausible since reactor power does rise to the Point of Adding Heat(~1%), however it does NOT stabilize nor return to 1%.
C.	Reactor power increases above the 'Point of Adding Heat' and then stabilizes at ~3%.	CORRECT: See above
D.	Reactor power increases to the 'Point of Adding Heat' and then lowers to 10 ⁻⁸ Amps in the Intermediate Range.	INCORRECT: Plausible since reactor power does rise to the Point of Adding Heat(~1%) and above, and the examinee may think that eventually Reactor Power will then start to lower and stabilize where the transient started.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039	K5.08
	Importance Rating	3.6	3.6

K/A: Main and Reheat Steam System: Knowledge of the operational implications of the following concepts as they apply to the MRSS: Effect of steam removal on reactivity

Proposed Question: RO Question # 27

Technical Reference(s): GFES: Reactor Operational Physics
TQ-TM-104-411-C001, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-10

Question Source: Bank #
Modified Bank #
New X

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	b.5	
	55.43		
Comments:			
<p>KA Match: This question matches the KA because the examinee will have to know that when a turbine bypass valve opens up, when the reactor is below the POAH, that the RCS temperature will drop. The RCS temperature drop, will result in positivity reactivity addition, indicated by a +SUR on the IR Instrument. Reactor Power will rise above the Point of Adding Heat and eventually stabilize at ~3% Power on the NI's. The Three Mile Island systems knowledge being tested is the capacity of the Turbine Bypass Valves.</p>			
<p>High Cog: This question is high cog because the examinee has to relate the component failure to the effect on RCS temperature and then the effect the change in RCS temperature has on reactivity feedback.</p>			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

28

ID: 1717325

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- EF-P-2B, Motor Driven Emergency Feedwater Pump, is out of service for maintenance.

EVENT:

- All four Reactor Coolant Pumps trip.
- EF-P-1, Steam Driven Emergency Feedwater Pump, tripped on overspeed.
- 1D 4160V ES bus tripped on overcurrent.
- Subcooling margin is 35F and slowly lowering.
- Incore temperatures are rising.
- 'A' OTSG Level is at 36% Operating Range
- 'B' OTSG Level is at 30% Operating Range

Based on these plant conditions, the operator is required to feed the OTSG's with ____ (1) ____ at a rate of ____ (2) ____ to promote ____ (3) ____.

- A. (1) Main Feedwater
(2) >1 Mlbm/hr
(3) Natural Circulation
- B. (1) Main Feedwater
(2) >1 Mlbm/hr
(3) Boiler Condenser Cooling
- C. (1) Emergency Feedwater
(2) > 215 GPM /OTSGs
(3) Boiler Condenser Cooling
- D. (1) Emergency Feedwater
(2) < 515 GPM
(3) Natural Circulation

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) When all four reactor coolant pumps trip, the reactor trips and enters the examinee into the EOP network; (2) Since EF-P-2B is out of service, EF-P-1 trips, and the 1D 4160V bus trips on overcurrent, there are no Emergency Feedwater pumps running, but the Main Feedwater Pumps are still running; (3) After the trip OP-TM-EOP-001 directs the ARO to feed the OTSG's in accordance with OP-TM-EOP-010, Rule 4; (4) From Rule 4, since subcooling margin is >25F, there is NO requirement to raise level to 75% to 85%; (5) But since there are no reactor coolant pumps operating and no emergency feedwater, feed with MFW at >1Mlb/hr.

A.	(1) Main Feedwater (2) >1Mlbm/hr (3) Natural Circulation	CORRECT: See above
B.	(1) Main Feedwater (2) >1Mlbm/hr (3) Boiler Condenser Cooling	INCORRECT: Plausible because the examinee could believe that since no reactor coolant pumps are running, that feeding the OTSGs at > 1Mlbm/hr is setting the plant up for boiler condenser cooling. Incorrect because the OTSGs are being setup to promote natural circulation.
C.	(1) Emergency Feedwater (2) >215 GPM /OTSGs (3) Boiler Condenser Cooling.	INCORRECT: Plausible because the examinee could believe that Emergency Feedwater is not lost. If Emergency Feedwater was available, and subcooling margin were less than 25F the feedrate would be > 215 GPM/OTSG.
D.	(1) Emergency Feedwater (2) < 515 GPM to both OTSGs (3) Natural Circulation	INCORRECT: Plausible because the examinee could believe that Emergency Feedwater is not lost. If Emergency Feedwater was available, and only EF-P-2A or EF-P-2B were running, there would be a limit of < 515 GPM.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061	K3.01
	Importance Rating	4.4	4.6

K/A: Auxiliary/Emergency Feedwater System: Knowledge of the effect that a loss or malfunction of the AFW will have on the following: RCS

Proposed Question: RO Question #28

Technical Reference(s): OP-TM-EOP-010, Rule 4, Rev 19
OP-TM-EOP-0101, Rule 4 basis, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: 424-GLO-11

Question Source: Bank #
Modified Bank #

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

	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, why high cog, why SRO only)			
KA MATCH: This question matches the KA because the AFW (EFW for TMI) malfunction results in the operator feeding the OTSGs at a different rate than if all of the EFW pumps were running.			
High Cog: This question is high cog because the examinee will have to analyze the event in the stem to realize that the reactor is tripped, and that there are no EFW pumps running.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

29

ID: 1717742

Points: 1.00

In accordance with 1107-2C Vital DC Electrical System, which of the following local indications will tell an operator that a battery charger is in the EQUALIZE mode?

- A. The AC Pilot Light is lit.
- B. The current flow is at least 50 amps.
- C. The output voltage is approximately 135 volts.
- D. Both the AC & DC input and output breakers are closed.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1107-2C, the operator observes each charger increases voltage to 135 ± 2 volts.

A.	The AC Pilot Light is lit	INCORRECT: Plausible if the examinee is NOT familiar with Battery Charger indication. The AC Pilot light is normally LIT.
B.	The current flow is at least 50 amps	INCORRECT: Plausible since current flow indication is available on the Battery Charger. Also, if inverters are on the DC power supply, current could be > 50 amps even in float mode.
C.	The output voltage is approximately 135 volts.	CORRECT: 1107-2C: Observe that the voltmeter on the front of each charger increases to 135 ± 2 volts.
D.	Both the AC & DC input and output breakers are closed.	INCORRECT: Plausible because both of these breakers are closed at all times of Battery Charger operation. Incorrect because these breakers being closed is not an exclusive indication that a battery charger is in the EQUALIZE mode.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063	A3.01
	Importance Rating	2.7	3.1

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

Proposed Question: RO Question #29

Technical Reference(s): 1107-2C, Rev 12

Proposed References to be provided to applicants during examination: None

Learning Objective: 734-GLO-2

Question Source: Bank # 356975

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

55.43

Comments: (KA Match, why high cog, why SRO only)

KA Match: This question matches the KA because the examinee must know which of the multiple indications listed at possible answer is correct for an equalizing charge (part of the DC distribution system).

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

30

ID: 1717423

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- 1M DC Bus is Powered from 1A DC Distribution Panel.

EVENT:

- Steam leak in the Reactor Building.
- The Reactor is tripped.
- RB Pressure increases to 4.2 psig.

10 Minutes Later:

- Loss of Offsite Power

Based on the above plant conditions:

___(1)___ transfer capability of 1M DC Bus to its alternate power supply is blocked.

This transfer capability is blocked due to the ___(2)___ signal.

- A. (1) Only the auto, (2) undervoltage
- B. (1) Only the auto, (2) Engineered Safeguards
- C. (1) Both the auto and manual, (2) under voltage
- D. (1) Both the auto and manual, (2) Engineered Safeguards

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<p>Explanation: To answer this question correctly, the examinee must know: (1) The design is for the 1M DC 480V Bus transfer to its alternate power supply when its normal power supply is lost on an undervoltage; (2) Any 1600 psig, 500 psig, or 4 psig ES signal blocks the auto and manual transfer if power were lost to 1M DC.</p>																		
A.	(1) Only the auto, (2) undervoltage	<p>INCORRECT: Plausible if the examinee has the misconception that the Loss of Offsite Power prevents the auto transfer of 1M DC. Since AC power is lost, the examinee could believe that 1M DC transfer is prohibited for reliability. 1M DC is still able to transfer to the alternate power supply when plant conditions/procedures allow.</p>																
B.	(1) Only the auto, (2) Engineered Safeguards	<p>INCORRECT: Plausible if the examinee has the misconception that the only the auto transfer is blocked with an ES Actuation signal. The examinee could believe that the operators could still transfer 1M manually. Incorrect because the auto and manual transfers are blocked on an ES signal.</p>																
C.	(1) Both the auto and manual, (2) under voltage	<p>INCORRECT: Plausible if the examinee has the misconception that the Loss of Offsite Power prevents auto transfer of 1M DC. Since AC power is lost, the examinee could believe that 1M DC transfer is prohibited for reliability.</p>																
D.	(1) Both the auto and manual, (2) Engineered Safeguards	<p>CORRECT: See above</p>																
<p>Examination Outline Cross-reference:</p> <table border="1"> <thead> <tr> <th>Level</th> <th>RO</th> <th>SRO</th> </tr> </thead> <tbody> <tr> <td>Tier #</td> <td>2</td> <td></td> </tr> <tr> <td>Group #</td> <td>1</td> <td></td> </tr> <tr> <td>K/A #</td> <td>063</td> <td>K4.02</td> </tr> <tr> <td>Importance Rating</td> <td>2.9</td> <td>3.2</td> </tr> </tbody> </table>				Level	RO	SRO	Tier #	2		Group #	1		K/A #	063	K4.02	Importance Rating	2.9	3.2
Level	RO	SRO																
Tier #	2																	
Group #	1																	
K/A #	063	K4.02																
Importance Rating	2.9	3.2																
<p>K/A: DC Electrical Distribution: Knowledge of DC electrical system design feature(s) and/ or interlock(s) which provide for the following: Breaker interlocks, permissives, bypasses and cross-ties.</p>																		
<p>Proposed Question: RO Question #30</p>																		
<p>Technical Reference(s): TQ-TM-104-740-C001, Rev 7 Electrical Print SS-209-050, Rev 3</p>																		
<p>Proposed References to be provided to applicants during examination: None</p>																		
<p>Learning Objective: 642-GLO-11</p>																		
<p>Question Source: Bank #</p> <p>Modified Bank # 356975</p> <p>New</p>																		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question History: System Exam 12 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

Comments:

KA Match: This question matches the KA because the examinee must know design feature of 1M DC that blocks transfer on an ES.

High Cog: This question is high cog because the examinee must recognize that plant conditions would initiate a 4 psi ES Actuation signal and this signal NOT only blocks auto transfer capabilities but prevents manual transfer as well.

1M DC Panel can be powered from 1A or 1B DC Distribution panel. It has both auto and manual transfer capabilities to one of these power sources.

Choose the correct letter below which makes the following statement TRUE.

On an _____ signal, 1M DC panel _____ transfer capabilities are prevented.

- A. under voltage, only auto
- B. Engineered Safeguards, only auto
- C. under voltage, both auto and manual
- D. Engineered Safeguards, both auto and manual

Answer: D

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

31

ID: 1717750

Points: 1.00

Which ONE of the following describes the electrical power supplies to the following pumps associate with Emergency Diesel Generator, EG-Y-1B:

- (1) the Fuel Oil Transfer Pumps, DF-P-1C/D, AND
(2) the Auxiliary Fuel Oil Pump, EG-P-10B

- A. (1) The two fuel oil transfer pumps are both AC powered.
(2) The Auxiliary Fuel Oil Pump is AC powered.
- B. (1) The two fuel oil transfer pumps are both AC powered.
(2) The Auxiliary Fuel Oil Pump is DC powered.
- C. (1) One fuel oil transfer pumps is AC powered, One fuel oil transfer pumps is DC powered.
(2) The Auxiliary Fuel Oil Pump is AC powered.
- D. (1) One fuel oil transfer pumps is AC powered, One fuel oil transfer pumps is DC powered.
(2) The Auxiliary Fuel Oil Pump is DC powered.

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) Each emergency diesel has two fuel transfer pumps, one is AC and one is DC powered; (2) In addition,each diesel has two fuel oil pumps: One is engine driven and the aux fuel oil pump is DC powered.			
A.	(1) The two fuel oil transfer pumps are both AC powered. (2) The Auxiliary Fuel Oil Pump is AC powered.	INCORRECT: Plausible since the examinee may believe that AC power is used for both fuel oil transfer pumps in addition to the Auxiliary Fuel Oil Pump.	
B.	(1) The two fuel oil transfer pumps are both AC powered. (2) The Auxiliary Fuel Oil Pump is DC powered.	INCORRECT: Plausible since the examinee may be believe that both fuel oil transfer pumps are both AC powered and the Auxiliary Fuel Oil Pump is DC powered.	
C.	(1) One fuel oil transfer pumps is AC powered, One fuel oil transfer pumps is DC powered. (2) The Auxiliary Fuel Oil Pump is AC powered.	INCORRECT: Plausible because the examinee may believe that Auxiliary Fuel Oil Pump is AC powered.	
D.	(1) One fuel oil transfer pumps is AC powered, One fuel oil transfer pumps is DC powered. (2) The Auxiliary Fuel Oil Pump is DC powered.	CORRECT: According to TQ-TM-104-861, 1A(1B) ES MCC provides power to Diesel Generator A(B) AC Fuel Pump, DF-P-1A (DF-P-1C); while 125/250 VDC Distribution Panel 1P (1Q) provide power to both the DC Fuel Transfer Pump, DF-P-1B (DF-P-1D), and the DC Auxiliary Fuel Pump (EG-P-10A/B).	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064	K2.02
	Importance Rating	2.8	3.1
K/A: Emergency Diesel Generators: Knowledge of bus power supplies to the following: Fuel Oil Pumps			
Proposed Question: RO Question # 31			
Technical Reference(s): 1107-5, Rev 151			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 861-GLO-4			
Question Source: Bank # 860277			
Modified Bank #			
New			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question History:	System 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.7	
	55.43		
Comments: (KA Match, why high cog, why SRO only)			
The KA is matched because the examinee must demonstrate knowledge of bus power supplies to the Fuel oil transfer pumps and the Auxiliary Fuel Oil Pump for a given EDG.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

32

ID: 1718191

Points: 1.00

Plant conditions:

- Loss of Station Power
- EG-Y-1A, Emergency Diesel Generator, had to be started from the Control Room using the Manual Start Pushbutton.
- EG-Y-1A 'Ready to Load' light is DE-ENERGIZED.
- EG-Y-1A voltage is 4050 V.
- EG-Y-1A frequency is 60.5 Hz.

All control room controls associated with EG-Y-1A were in the ES STANDBY line-up when the diesel was started.

From the list below, identify the ONE (1) action that would ENERGIZE the Ready to Load light under these conditions.

- A. Adjust the governor to lower frequency to 60.0 Hz.
- B. Energize the synchroscope for EG-Y-1A generator output breaker, G1-02.
- C. Adjust the local unit voltage rheostat to obtain an output voltage of 4100 V.
- D. Adjust the manual voltage controller on Console Right to obtain an output voltage of 4100 V.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) The 'Ready to Load' light energizes when the diesel is up to the correct operating bands: Voltage is 4100- 4300 volts, and frequency 60.24-61.0 Hz (OP-TM-642-231); (2) The examinee must realize that the voltage is low, but the frequency is in band; (3) The examinee must also recognize that there is only one method to return the voltage to specification, which is the local unit voltage rheostat.

A.	Adjust the governor to lower frequency to 60.0 Hz.	INCORRECT: Plausible if the examinee believes that a frequency of 60.5 Hz is out of tolerance for the 'Ready to Load' light. Incorrect because the 'Ready to Load' light will light at 60.24 Hz to 61.0 Hz.
B.	Energize the synchroscope for EG-Y-1A generator output breaker, G1-02.	INCORRECT: Plausible if examinee relates a manual start in the control room to a manual start when paralleling an emergency diesel with offsite power which requires the use of the synchroscope. When manually starting a diesel generator to parallel with offsite power, the synchroscope must be energized. For this situation, since there is no offsite power, the synchroscope does not need to be energized.
C.	Adjust the local unit voltage rheostat to obtain an output voltage of 4.1KV	CORRECT: See above
D.	Adjust the manual voltage controller on Console Right to obtain an output voltage of 4.1 KV.	INCORRECT: Plausible because there is a rheostat for voltage control on console right. Incorrect because this would adjust voltage if the voltage regulator was in manual, which it is not.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064	A3.03
	Importance Rating	3.4	3.3

K/A: Emergency Diesel Generators: Ability to monitor automatic operation of the ED/G system, including: Indicating Lights, meters, and recorders.

Proposed Question: RO Question # 32

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

OP-TM-861-901, Rev 18

Proposed References to be provided to applicants during examination: None

Learning Objective: 861-GLO-4

Question Source: Bank # 657377

Modified Bank #

New

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question History:	Sim 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.7	
	55.43		
Comments: (KA Match, why high cog, why SRO only)			
KA Match: This question matches the KA because the examinee must know which emergency diesel generator parameters are monitored by the 'Ready-to-Load' light.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

33

ID: 1718267

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- OP-TM-AOP-005, RIVER WATER SYSTEM FAILURES, is entered due to reduced river water level and rising river water temperature.

EVENT:

- Auxiliary Operator report:
 - ISPH Pump Bay Water Level is at 270 foot elevation.
 - River Water temperature is 92 degrees F.

Based on these conditions identify the ONE selection that describes required action(s).

- A. Initiate plant shutdown to be at CSD within the next 36 hours.
- B. Initiate plant shutdown to be at HSD IAW TS 3.0.1. requirements.
- C. Shutdown any river water pumps whose operation is not essential.
- D. Trip the reactor, initiate EOP-001, then trip all four Reactor Coolant Pumps.

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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, why high cog, why SRO only)

KA Match: This question matches the KA because the examinee must know the procedure actions associated with a low river water level.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

34

ID: 1718269

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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; (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 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.
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 #	2	
	Group #	1	
	K/A #	078	K1.04
	Importance Rating	2.6	2.9

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

Proposed Question: RO Question # 34

Technical Reference(s): 1104-12, Rev 63 TQ-TM-104-850-C001, Rev 6

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-GLO-10

Question Source: Bank #
Modified Bank # 978699
New

Question History: N/A Last NRC Exam: Unmodified on ILT 12-01

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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, why high cog, why SRO only)		
KA Match: This question matches the KA because the examinee must know the cooling water associated with each IA-P-1, and when it each cooling water is utilized.		
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.		
Plant Conditions:		
The plant is operating at 100% power.		
Main Instrument Air Compressor, IA-P-4, has been secured for repairs.		
"A" and "B" Instrument Air Compressors, IA-P-1A and IA-P-1B, are running as required.		
Event:		
A loss of all Secondary Closed Cooling Water Pumps has occurred.		
Given the above information and two minutes after the event, the Instrument Air Compressors will align to ____ (1) ____ as the source of cooling due to ____ (2) ____.		
A.	(1) Fire Service water (2) high Instrument Air compressor temperature	
B.	(1) Fire Service water (2) the tripping of all 3 Secondary Closed Cooling Water Pump breakers	
C.	(1) Nuclear Service Closed Cooling Water (2) high Instrument Air compressor temperature	
D.	(1) Nuclear Service Closed Cooling Water (2) the tripping of all 3 Secondary Closed Cooling Water Pump breakers	
Answer: B		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

35

ID: 1718430

Points: 1.00

Given the following plant conditions:

- Reactor is at 52% power.
- ICS SG/RX Hand/Auto Station is in HAND.
- All other ICS stations are in AUTOMATIC.

EVENT:

- Group 7 Rod 4 in Quadrant YZ drops fully into the core.

Which of the following indicates the effect on (1) Quadrant Power Tilt in Quadrant YZ and (2) T_{ave} parameter response?

ASSUME NO OPERATOR ACTIONS.

- A. (1) Negative Quadrant Power Tilt.
(2) T_{ave} lowers and remains low.
- B. (1) Negative Quadrant Power Tilt.
(2) T_{ave} lowers and returns to setpoint.
- C. (1) Positive Quadrant Power Tilt.
(2) T_{ave} lowers and remains low.
- D. (1) Positive Quadrant Power Tilt.
(2) T_{ave} lowers and returns to setpoint.

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) A dropped rod in any quadrant suppress power in that quadrant, thus creating a negative quadrant power tilt; (2) The Dropped Rod Runback is to 55% Reactor Power, so NO runback would be in effect for the given plant conditions; (3) On a Dropped Rod the CRD Out Inhibit clears when Reactor Power is <60% to allow for Automatic Tavg control; (4) In this case, the controlling ICS signal in which it bases the dropped rod runback for is Generated Megawatts, so when rods pull to maintain Tave, power will rise, but the runback will not come back in because Generated Megawatts remain constant; (4) The SG/RX Master in Hand would not effect Tavg Control.

A.	(1) Negative Quadrant Power Tilt. (2) Tave lowers and remains low.	INCORRECT: Plausible since part 1 is correct and Tave is low. The CRD Control System would allow Rods to Withdraw to correct for a low Tave, even with a Dropped Rod, as long as the reactor power remains <60%. The examinee may think 55% Reactor Power is the max power since this is the setpoint for a dropped rod runback. (i.e., that control rods will not withdraw); (3) The examinee may assume with the SG/RX Master in Hand it would require manual operation to raise Tave.
B.	(1) Negative Quadrant Power Tilt. (2) Tave lowers and returns to setpoint.	CORRECT: The dropped rod would suppress flux in this quadrant and this quadrant would be lower than the other quadrants. On a dropped rod the CRD Out Inhibit clears when Reactor Power is <60% to allow for Automatic Tave control.
C.	(1) Positive Quadrant Power Tilt. (2) Tave lowers and remains low.	INCORRECT: Plausible if the examinee is believes that a dropped control rod will give a positive power tilt, based on the calculations. Reversing the operation in the calculation gives the same numerical value but will reverse the sign. The examinee may think 55% Reactor Power is the max power since this is the setpoint for a dropped rod runback so control rods could not withdraw. The examinee may assume with the SG/RX Master in Hand it would require manual operation to raise Tave.
D.	(1) Positive Quadrant Power Tilt. (2) Tave lowers and returns to setpoint.	INCORRECT: Plausible if the examinee is believes that a dropped control rod will give a positive power tilt, based on the calculations. Reversing the operation in the calculation gives the same numerical value but will reverse the sign. On a Dropped Rod the CRD Out Inhibit clears when Reactor Power is <60% to allow for Automatic Tavg control

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	001	K3.02
	Importance Rating	3.4	3.5

K/A: Control Rod Drive System: Knowledge of the effect that a loss or malfunction of the CRDS will have on the following: RCS

Proposed Question: RO Question # 35

Technical Reference(s): OP-TM-AOP-062, Rev 7 TQ-TM-104-621-C001, Rev 10

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

TQ-TM-104-622-C001, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-062-PCO-5

Question Source: Bank #
Modified Bank # 887495
New

Question History: N/A Last NRC Exam: 2003 NRC

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, why high cog, why SRO only)

KA Match: This question matches the KA because the examinee will have to know which area of the core has a negative quadrant power tilt, in addition to what Tave will do.

High Cog: This question is high cog because the examinee will have to analyze the dropped rod to determine that quadrant power tilt in that region is negative. In addition, the examinee must know the ICS control on an dropped rod, and that there is no CRD out inhibit preventing ICS from maintaining Tave in the correct band.

Sequence of events:

- Reactor power is 100%, with ICS in full automatic.
- Group 7 Rod #4 drops into the core, Quadrant YZ.

Based on these conditions, identify the ONE statement below that describes initial plant response when the rod reaches full insertion.

- A. Pressurizer level rises.
- B. RCS pressure and temperature lower.
- C. Diamond rod control transfers to manual.
- D. Quadrant power tilt becomes more positive in Quadrant YZ.

Answer: B

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

36

ID: 1737551

Points: 1.00

Plant Conditions:

- 60% power with ICS in auto.

EVENT:

- Reactor Building pressure starts to rise.
- RCS pressure starts to lower.
- The reactor is tripped and the IMA's of OP-TM-EOP-001, REACTOR TRIP are performed.

POST TRIP CONDITIONS:

- Reactor Building pressure rises to 4.2 psig.
- RCS pressure lowers to 1675 psig.

Post Trip, RM-A-2, Reactor Building Atmosphere Monitor, (1) be used to determine whether the Reactor Building pressure increase is from the an reactor coolant leak because (2).

- A. (1) can
(2) there is NOT a 1600 psig ES Actuation Signal
- B. (1) can
(2) Reactor Building Atmospheric conditions will NOT inhibit proper air flow
not EQ for Post Acc
- C. (1) can NOT
(2) it is isolated from Containment
- D. (1) can NOT
(2) Reactor Building Atmospheric conditions may inhibit proper air flow
not EQ for Post Acc

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) ES Actuation signals, 1600 psig, 500 psig, and 4 psig RB pressure isolate RM-A-2 from the reactor building; (2) Since the stem never indicates that the 1600 psig was bypassed, when RCS pressure lowered below 1600 psig, RM-A-2 is isolated and cannot be used to differentiate between an RCS leak or steam leak; (3) In addition, RM-A-2 pump will de-energize when a combination of its' isolation valves close (CM-V-1, 2, 3, 4).			
A.	(1) can (2) there is NOT a 1600 psig ES Actuation Signal	INCORRECT: Plausible if the examinee believes that only a 1600 psig ES signal isolates RM-A-2. The 1600 psig signal is indicative of a LOCA, the examinee may believe that in the case of a LOCA that RM-A-2 gets isolated.	
B.	(1) can (2) Reactor Building Atmospheric conditions will NOT block flow	INCORRECT: Plausible if the examinee does not realize that a 1600 psig signal is present and/or believes that a 4 psig ES actuation is the only ES signal that isolates RM-A-2.	
C.	(1) can NOT (2) it is isolated from Containment	CORRECT: A 1600 psig ES isolates RM-A-2 via CM-V-1,2,3,4 closing. Also, there is an interlock which will automatically de-energize RM-A-2's pump when either CM-V-1 or CM-V-2 close.	
D.	(1) can NOT (2) Reactor Building Atmospheric conditions may prevent proper air flow	INCORRECT: Part 1 is correct but part 2 is incorrect. RM-A-2 does have charcoal and paper filters. Plausible if the candidate recognizes that they could saturate if the detector was not isolated but this would not happen since it is Isolated from RB Atmospheric conditions.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	002	A3.01
	Importance Rating	3.7	3.9
K/A: Reactor Coolant System: Ability to monitor automatic operation of the RCS, including: Reactor coolant leak detection system			
Proposed Question: RO Question # 36			
Technical Reference(s): 1105-3, Rev 053 TQ-TM-104-661-C001, Rev 7			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 661-GLO-2			
Question Source: Bank #			
Modified Bank #			
New X			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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, why high cog, why SRO only)			
KA Match: This question matches the KA because the examinee must know whether RM-A-2 is capable of being used for RCS leak detection.			
High Cog: This question is high cog because the examinee must determine that a 1600 psig ES signal is present. Then the examinee must determine that on a 1600 psig ES, the containment isolation valves for RM-A-2 go closed.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

37

ID: 1718596

Points: 1.00

Sequence of Events:

- Emergency Feedwater Pump, EF-P-2A, is OOS.
- Loss of Offsite Power (LOOP) occurred 10 minutes ago.
- A small break LOCA occurred following the loss of offsite power.
- Emergency Feedwater Pump, EF-P-2B, tripped when it started automatically.
- OP-TM-EOP-001, Reactor Trip, IMAs were performed.
- OP-TM-EOP-006, LOCA Cooldown and OP-TM-AOP-020, Loss of Station Power, actions are in progress.
- 4 psig ESAS was manually actuated.
- RCS temperature is 518°F and SCM has been maintained >25°F throughout the transient.
- Incore Thermocouples are slowly rising.
- Natural Circulation cannot be verified.

Which ONE of the following is the **NEXT** required action?

Reduce OTSG pressure _____.

- or MAX RATE*
- A. in both OTSGs to approximately 150 psig.
 - B. while maintaining >750 psig to avoid feedwater isolation.
 - C. in both OTSGs as low as possible (Atmospheric pressure or vacuum).
 - D. so that secondary T_{sat} is 40 to 60°F lower than incore thermocouple temperature, while maintaining >150 psig.

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-EOP-006, Section 4.0, INADEQUATE RCS COOLDOWN is entered under the following conditions: Primary to secondary heat transfer cannot be established, RCS temperature is > 300F, and core cooldown rate is less than 40F/hr; (2) Due to the loss of offsite power, there are no reactor coolant pumps. In addition the stem states that natural circulation is not verified. In order to have primary to secondary heat transfer, a reactor coolant pump must be running or natural circulation must be verified; (3) In section 4, the crew will lower OTSG pressures so Tsat is 40 to 60F lower than incore thermocouple temperature in order to maintain the OTSGs as a heat sink; (3) With only the steam driven Emergency Feedwater Pump available (EF-P-1), OTSG pressure must be maintained >150 psig to ensure feed flow to the OTSGs.

A.	in both OTSGs to approximately 150 psig..	INCORRECT: Plausible since this would strengthen the heat sink. Incorrect because lowering OTSG pressure rapidly to 150 psig could cause the OTSGs to become uncoupled from the primary and further disrupt the attempts as establishing primary to secondary heat transfer.
B.	while maintaining >750 psig to avoid feedwater isolation.	INCORRECT: Plausible since lowering OTSG is required and maintain OTSG Pressure > 750 psig would prevent main Feedwater Isolation, allowing it be available if power is restored. However, the procedure directs bypassing the MFW Isolation signal prior to OTSG pressure < 750 psig.
C.	in both OTSGs as low as possible (Atmospheric pressure or vacuum).	INCORRECT: Plausible because this is how the operating crew would perform a rapid RCS cooldown if SCM were lost and an RCP were still running. Incorrect because those symptoms to not exist.
D.	so that secondary Tsat is 40 to 60°F lower than incore thermocouple temperature, while maintaining >150 psig.	CORRECT: See above

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	035	A2.06
	Importance Rating	4.5	4.6

K/A: Steam Generator System: Ability to (a) predict the impacts of the following malfunctions or operations on the SG; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Small break LOCA

Proposed Question: RO Question # 37

Technical Reference(s): OP-TM-EOP-006, Rev 12

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP004-PCO-3

Question Source: Bank #

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Modified Bank # 572835

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.5
55.43

Comments: (KA Match, why high cog, why SRO only)

KA Match: This question matches the KA because the examinee must predict the impact that having only the steam drive emergency feedpump has on a LOCA Cooldown. In addition, the examinees must have knowledge of the LOCA cooldown procedure to understand the bands that the operating crew would set to establish the OTSG has a heat sink.

High Cog: This question is high cog because the examinee must ascertain all the requirements to enter section 4 of OP-TM-EOP-006 exist, and then understand steps within that section to strengthen the OTSG as a heat sink.

Sequence of Events:

- Emergency Feedwater Pump, EF-P-2A, is OOS.
- Loss of Offsite Power (LOOP) occurred 10 minutes ago.
- A small break LOCA occurred following the loss of offsite power.
- Emergency Feedwater Pump, EF-P-2B, tripped when it started automatically.
- OP-TM-EOP-001, Reactor Trip, IMAs were performed.
- OP-TM-EOP-006, LOCA Cooldown and OP-TM-AOP-020, Loss of Station Power, actions are in progress.
- RCS temperature is 518°F and SCM has been maintained >25°F throughout the transient.
- Cooldown rate is 5°F/hr.
- Conditions to verify Natural Circulation can not be met.

Which ONE of the following is the **NEXT** required action?

- A. Lower OTSG Pressure while maintaining >750 psig to avoid feedwater isolation in accordance with EOP-006, LOCA Cooldown.
- B. Lower OTSG pressure to establish >40°F/hr cooldown rate while maintaining >150 psig for Emergency Feedwater Pump, EF-P-1, operation in accordance with EOP-006, LOCA Cooldown.
- C. Lower OTSG Pressure in both OTSG's as low as possible (Atmospheric pressure or vacuum) to promote primary to secondary heat transfer in accordance with EOP-006, LOCA Cooldown.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

- D. Lower OTSG Pressure so secondary T_{sat} is 40-60°F higher than incore thermocouple temperature to promote primary to secondary heat transfer in accordance with EOP-004, Lack of Primary to Secondary Heat Transfer.

Answer: B

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

38

ID: 1718720

Points: 1.00

Plant conditions:

- 100% power with ICS in full auto.

EVENT:

- Overhead alarm MAP C-1-1, "RADIATION LEVEL HI", just actuated.
- The source of the alarm is RM-A-5, CONDENSER VACUUM PUMP EXHAUST.

The MAP-5 Sampler starts to sample the:

- A. steam lines for only iodine when RM-A-5 reaches the ALERT setpoint.
- B. steam lines for iodine and tritium when RM-A-5 reaches the ALERT setpoint.
- C. condenser offgas for only iodine when RM-A-5 reaches the HI-ALARM setpoint.
- D. condenser offgas for iodine and tritium when RM-A-5 reaches the HI-ALARM setpoint.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) The initiating setpoint for starting the MAP-5 is from the HI Alarm NOT the Alert Setting; (2) Physical location of the sample line is from Condenser Offgas, NOT from the Steam Lines; (3) The MAP-5 detects Iodine NOT tritium.			
A.	steam lines for only iodine when RM-A-5 reaches the ALERT setpoint.	INCORRECT: Plausible since both OTSGs have radiation monitors on the steam lines to detect for an OTSG Tube Leak. Incorrect because this is not the location of the interface for RM-A-5. In addition, the alert alarm does NOT start the MAP-5.	
B.	steam lines for iodine and tritium when RM-A-5 reaches the ALERT setpoint.	INCORRECT: Plausible since both OTSGs have radiation monitors on the steam lines and there is a parallel line for a tritium sample for the MAP-5. In addition, the Alert Alarm does NOT start the MAP-5.	
C.	condenser offgas for only iodine when RM-A-5 reaches the HI-ALARM setpoint	CORRECT: See above.	
D.	condenser offgas for iodine and tritium when RM-A-5 reaches the HI-ALARM setpoint.	INCORRECT: Plausible since it does sample for Iodine, however the sample line for Tritium is in parallel and a manual operation not associated with the MAP-5.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	055	K1.06
	Importance Rating	2.6	2.6
K/A: Condenser Air Removal System: Knowledge of the physical connections and/or cause/effect relationships between the CARS and the following systems: PRM System			
Proposed Question: RO Question # 38			
Technical Reference(s): TQ-TM-104-661-C001 OP-TM-MAP-C0101, Rev 3			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 661-GLO-9			
Question Source: Bank # 575075			
Modified Bank #			
New			
Question History: System Exam 11 Last NRC Exam: ILT 05-1			
Question Cognitive Level: Memory or Fundamental Knowledge X			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 b.7

55.43

Comments: (KA Match, why high cog, why SRO only)

KA Match: This question matches the KA because the examinee must have knowledge of where the radiation monitor connects to the condenser air removal system.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

39

ID: 1718734

Points: 1.00

Plant Conditions:

- OP-TM-232-551, "Liquid Release of 'A' WECST with WDL-P-14A" is in progress.
- MAP Alarm C-1-1 Radiation Level High is received.
- RM-L-7, Plant Discharge Rad Monitor is in high alarm.

What automatic action mitigates the potential radioactive release to the environment?

- A. WDL-V-124, WDL-P-14A Outlet to MDCT, closes and WDL-P-14A WECST Recirc Pump will remain running.
- B. WDL-V-124, WDL-P-14A Outlet to MDCT, closes and WDL-P-14A WECST Recirc Pump trips.
- C. WDL-V-257, WDL-P-14A/B Discharge to MDCT/River, closes and WDL-P-14A WECST Recirc Pump will remain running.
- D. WDL-V-257, WDL-P-14A/B Discharge to MDCT/River, closes and WDL-P-14A WECST Recirc Pump trips.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) When the high alarm is reached for RM-L-7, then WDL-V-257 will close; (2) WDL-P-14A remains running on recirculation.																							
A.	WDL-V-124, WDL-P-14A Outlet to MDCT, closes and WDL-P-14A WECST Recirc Pump will remain running.	INCORRECT: Plausible since WDL-V-124 is the release flow control valve however WDL-V-124 does not close on a high alarm from RM-L-7.																					
B.	WDL-V-124, WDL-P-14A Outlet to MDCT, closes and WDL-P-14A WECST Recirc Pump trips.	INCORRECT: Plausible since WDL-V-124 is the release flow control valve and WDL-P-14A is the discharge pump; however neither condition occurs on high radiation.																					
C.	WDL-V-257, WDL-P-14A/B Discharge to MDCT/River, closes and WDL-P-14A WECST Recirc Pump will remain running.	CORRECT: High alarm on RM-L-7 will close WDL-V-257 and the tank WDL-P-14A remains running.																					
D.	WDL-V-257, WDL-P-14A/B Discharge to MDCT/River, closes and WDL-P-14A WECST Recirc Pump trips.	INCORRECT: Plausible since WDL-V-257 will close on the high alarm signal from RM-L-7; however WDL-P-14A will not trip.																					
<table><tr><td>Examination Outline Cross-reference:</td><td>Level</td><td>RO</td><td>SRO</td></tr><tr><td></td><td>Tier #</td><td>2</td><td></td></tr><tr><td></td><td>Group #</td><td>2</td><td></td></tr><tr><td></td><td>K/A #</td><td>068</td><td>A3.02</td></tr><tr><td></td><td>Importance Rating</td><td>3.6</td><td>3.6</td></tr></table>				Examination Outline Cross-reference:	Level	RO	SRO		Tier #	2			Group #	2			K/A #	068	A3.02		Importance Rating	3.6	3.6
Examination Outline Cross-reference:	Level	RO	SRO																				
	Tier #	2																					
	Group #	2																					
	K/A #	068	A3.02																				
	Importance Rating	3.6	3.6																				
K/A: Liquid Radwaste System: Ability to monitor automatic operation of the Liquid Radwaste System including: Automatic Isolation																							
Proposed Question: RO Question: # 39																							
Technical Reference(s): OP-TM-MAP-C0101, Rev 3 TQ-TM-104-232-C001, Rev 7																							
Proposed References to be provided to applicants during examination: None																							
Learning Objective: 661-GLO-2																							
<table><tr><td>Question Source:</td><td>Bank #</td><td>679215</td></tr><tr><td></td><td>Modified Bank #</td><td></td></tr><tr><td></td><td>New</td><td></td></tr></table>				Question Source:	Bank #	679215		Modified Bank #			New												
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Question History:	N/A	Last NRC Exam:	08-01 NRC																				

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 radiation interlocks that cause automatic isolation on the liquid radwaste system.		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

40

ID: 1718755

Points: 1.00

REFERENCE PROVIDED

Plant conditions:

- Plant is shutting down to HOT SHUTDOWN.
- Tave is 575 deg-F.
- Pressurizer level is being maintained in accordance with 1102-10, PLANT SHUTDOWN.

Based on the above conditions, which indicated level below would be the **minimum** acceptable pressurizer level?

- A. 100".
- B. 185".
- C. 210".
- D. 220".

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) 1102-10 Plant Shutdown pressurizer must be maintained with OP-TM-211-472 Attachment 7.2; (2) Interpret and use attachment 7.2 to determine the minimum level based on the given plant conditions with a Tave of 575 deg-F is 185".			
A.	100"	INCORRECT: Plausible if the examinee believes that we should be maintaining Pressurizer level for hot shutdown.	
B.	185"	CORRECT: See above.	
C.	210"	INCORRECT: Plausible if the examinee uses the recommended vs Tave.	
D.	220"	INCORRECT: Plausible if the examinee believes that pressurizer level is to be maintained at 220", the normal operating pressurizer level.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.25	
	Importance Rating	3.9	4.2-
K/A: Ability to interpret reference materials, such as graphs, curves, tables, etc.			
Proposed Question:		RO Question # 40	
Technical Reference(s):		1102-10, Rev 99	OP-TM-211-472, Rev 4
Proposed References to be provided to applicants during examination:		OP-TM-211-472, Attachment 7.2 with Blacked out curve names.	
Learning Objective:		GOP-010-PCO-1	
Question Source:		Bank #	371611
		Modified Bank #	
		New	
Question History:		Sim Exam 2	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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

55.43

Comments: (KA Match, why high cog, why SRO only)

KA Match: This matches the KA because the examinee must be able to use a graph which operators use to cooldown the plant.

High Cog: This question is high cog because the examinee must understand how to read the graph given the plant parameters.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

41

ID: 1737589

Points: 1.00

Plant Conditions:

- The plant was at 100% power when a control rod in Group 7 dropped.
- During the runback two additional rods in Group 7 become stuck at 88% withdrawn.
- The ICS runback was completed in automatic.

With the above plant conditions the operating crew must _____ within one hour.

- A. trip the reactor
- B. be in HOT SHUTDOWN
- C. verify Axial Power Imbalance is within limits
- D. verify $SDM \geq 1\%$ delta k/k or initiate boration

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) A dropped rod in group 7 will cause a plant runback to less than 455 MWe (55% reactor power), this will drive in Group 7 control rods to ~ 25% rod index; (2) When the dropped rod runback is complete, the crew must enter OP-TM-AOP-062, INOPERABLE ROD due to the two stuck rods being inoperable (control rods misaligned with the group by more than 9 inches); (3) When two or more rods are inoperable, OP-TM-AOP-062, Step 3.2 requires that a shutdown margin (SDM) calculation be performed to verify $SDM \geq 1\% \text{ delta k/k}$ or initiate emergency boration.

A.	trip the reactor	INCORRECT: Plausible because operation with two or more rods inoperable is not permitted. Incorrect because a reactor trip is not required.
B.	be in HOT SHUTDOWN	INCORRECT: Plausible because the operating crew will initiate a plant shutdown to HOT SHUTDOWN. Incorrect because the crew has 6 hours to be in HOT SHUTDOWN.
C.	verify Axial Power Imbalance is within limits	INCORRECT: Plausible because there are limits on Axial Power which would require action within an hour, however there is nothing in the stem which would indicate that this limit is being violated.
D.	verify $SDM \geq 1\% \text{ delta k/k}$ or initiate boration	CORRECT: See Above.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.39	
	Importance Rating	3.9	4.5

K/A: Knowledge of less than or equal to one hour Technical Specification action statements for systems.

Proposed Question: RO Question #41

Technical Reference(s): OP-TM-AOP-062, Rev 7 OP-TM-MAP-H0101, Rev 2
OP-TM-AOP-0621, Rev 6

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-062-PCO-4

Question Source: Bank # 1110401
Modified Bank #
New

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 recognize a condition in which SDM must be verified within an hour. This is an operational 1 hour technical specification.

High Cog: This question is high cog because the examinee must analyze the stem to know that control rods have driven in further than 9 inches more than 88% index. I

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

42

ID: 1718778

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- The Unit has experienced several fuel pin failures.
- A leak must be repaired in the Aux. Bldg.
- The general area dose rate in the location of the repair is 600 mrem/hr.
- In order to reach the location of the repair the worker must transit through a 6 Rem/hr high radiation area for 2 minutes and return via the same path.
- The worker currently has an accumulated annual dose of 400 mrem.

The maximum allowable time that the worker can participate in the repairs and **NOT** exceed the TEDE Administrative Dose Control Limit is _____ minutes (Worker does **NOT** have a High Lifetime Exposure).

- A. 100
- B. 120
- C. 140
- D. 160

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) The TEDE administrative dose limit in RP-AA-203 is 2000 mrem routine cumulative TEDE/yr; (2) That the worker in question already has 400 mrem and must make a trip TO and FROM the work area without exceeding the TEDE limit.			
A.	100	INCORRECT: Based on 2000 mrem - 600 mrem (current) - 400 mrem (transient) = 1000 / 600 mrem/hr x 60 min = 100 minutes	
B.	120	B. CORRECT : The candidate should determine that the ADCL is 2000 mrem. Transient exposure is 400 mrem (6000mrem/hr x 4/60hr). (transit to and from the job). (Current) 400 mrem + (transit) 400 mrem = 800 mrem ADCL of 2000 mrem - 800 mrem = 1200 mrem allowable before reaching ADCL. 1200 mrem /600 mrem/hr = 2 hours	
C.	140	INCORRECT: Based on calculating using a one-way transit dose	
D.	160	INCORRECT: Based on using ADCL (2000) and NO transit dose.	
Examination Outline Cross-reference:			
		Level	RO
		Tier #	3
		Group #	3
		K/A #	2.3.4
		Importance Rating	3.2
			3.7
K/A: Knowledge of radiation exposure limits under normal or emergency conditions.			
Proposed Question: RO Question # 42			
Technical Reference(s): RP-AA-203, Rev 5			
Proposed References to be provided to applicants during examination: None			
Learning Objective: None			
Question Source: Bank # 375097			
Modified Bank #			
New			
Question History: Sim Exam 3 Last NRC Exam: N/A			
Question Cognitive Level: Memory or Fundamental Knowledge			
Comprehension or Analysis X			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 radiation exposure limits.

High Cog: This question is high cog because the examinee must use math to determine the correct allowable time the worker has in the area.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

43

ID: 1718893

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

Sequence of Events:

T = 1100:

A Reactor Building Purge has commenced.

T = 1200:

The Reactor Building purge is secured due to a minor equipment problem.

T = 1730:

The equipment problem has been fixed and the Reactor Building purge is ready to recommence.

Given the above information and IAW CY-TM-170-2012, RELEASING RADIOACTIVE GASEOUS EFFLUENTS:

The RB Purge may recommence_____.

- A. with the same release permit using original air sample data.
- B. with the same release permit using updated air sample data.
- C. when a new release permit is generated using a new release number with updated air sample data.
- D. when a new release permit is generated using the same release number with updated air sample data.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with CY-TM-170-2012, step 4.1.3: A Reactor Building Purge may be stopped and restarted within four hours, using the same release permit; (2) If the purge is secured for more than four hours, then INITIATE a new release permit (with new air samples).			
A.	with the same release permit using original air sample data.	INCORRECT: Plausible if the candidate is either not familiar with the time requirement allowed to use the same permit without a new air sample.	
B.	with the same release permit using updated air sample data.	INCORRECT: Plausible if the candidate is not familiar with the time requirement allowed to use the same permit even with a new air sample.	
C.	when a new release permit is generated using a new release number with updated air sample data.	CORRECT: See above.	
D.	when a new release permit is generated using the same release number with updated air sample data.	INCORRECT: Plausible if the candidate thinks the same Permit Number is permitted, since the old permit allowing the purge was not completed.	
Examination Outline Cross-reference:			
	Level	RO	SR O
	Tier #	3	
	Group #	3	
	K/A #	2.3.11	
	Importance Rating	3.8	4.3
K/A: Ability to control radiation releases.			
Proposed Question: RO Question # 43			
Technical Reference(s): CT-TM-170-2012, Rev 1			
Proposed References to be provided to applicants during examination: None			
Learning Objective: NOP-DBIG-APCO-1			
Question Source: Bank #			
Modified Bank # 1142262			
New			
Question History: N/A Last NRC Exam: Unmodified on 12-01 NRC			
Question Cognitive Level: Memory or Fundamental Knowledge			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 b.10 55.43	
Comments: (KA Match, why high cog, why SRO only)		
KA Match: The KA is matched because the question requires the candidates to recognize the requirements prior to reestablishing a Reactor Building Purge.		
High Cog: This question is at the Comprehension/Analysis level because the candidate is required to perform a math problem and, based on the results of the math problem, the candidate must recall procedural steps and requirements associated with Reactor Building Purge.		
Plant Conditions (T = 1600):		
<ul style="list-style-type: none">• The plant is operating at 100% power.• A Reactor Building purge has commenced.		
Sequence of Events:		
<ul style="list-style-type: none">• T = 1700:<ul style="list-style-type: none">• The Reactor Building purge is secured due to the Chemistry Supervisor questioning the calculations used.• T = 2030:<ul style="list-style-type: none">• The Chemistry Supervisor agrees that all calculations are correct and the Reactor Building purge is ready to recommence.		
Given the above information and IAW 6610-ADM-4250.12, Releasing Radioactive Gaseous Effluents, which ONE of the following statements is correct with regards to the Reactor Building purge?		
<p>A. The same release permit may be used.</p> <p>B. No release permit is needed for a Reactor Building purge.</p> <p>C. A new release permit needs to be generated with a new release number assigned.</p> <p>D. A new release permit needs to be generated but the same release number can be used.</p>		
Answer:	A	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

44

ID: 1719768

Points: 1.00

The plant has experienced a major transient with several component malfunctions.

Entry conditions are met for:

- OP-TM-EOP-001, REACTOR TRIP.
- Two EOP-010 Rules.
- Five EOP-010 Guides.
- Several AOP series procedures.
- Numerous alarm response procedures.

Following the Immediate Manual Actions of OP-TM-EOP-001, what is the correct sequence of performance when multiple procedure entries are required?

- A. Perform the initial symptom check. Following this, the CRS determines the sequence of actions required to mitigate the event successfully.
- B. Perform the initial symptom check, and the VSSVs. Following this, the CRS determines the sequence of actions required to mitigate the event successfully.
- C. Perform the initial symptom check. Remaining procedures must be implemented in the following order: EOP Rules, EOP Guides, AOPs and Alarm Response procedures.
- D. Perform the initial symptom check, and the VSSVs. Remaining procedures must be implemented in the following order: EOP Rules, EOP Guides, AOPs and Alarm Response procedures.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) OS-24, CONDUCT OF OPERATIONS DURING ABNORMAL AND EMERGENCY EVENTS, provides direction that any other procedure actions should be interrupted to perform the reactor trip IMA's and initial Symptom Check; (2) Once the IMAs and the initial symptom check have been accomplished, the Control Room Supervisor determines the sequence of action between parallel procedures. The CRS selects the action most significant to overall event mitigation.

A.	Perform the initial symptom check. Following this, the CRS determines the sequence of actions required to mitigate the event successfully.	CORRECT: See above
B.	Perform the initial symptom check, and the VSSVs. Following this, the CRS determines the sequence of actions required to mitigate the event successfully.	INCORRECT: Plausible if examinee believes the Vital System Status Verifications (VSSVs) of EOP-001 must be performed prior to moving to another procedure. While the VSSVs are important, the CRS has the leeway to perform other procedures prior to completion.
C.	Perform the initial symptom check. Remaining procedures must be implemented in the following order: EOP Rules, EOP Guides, AOPs and Alarm Response procedures.	INCORRECT: Plausible if examinee believes EOP Rules & Emergency Procedure must always take priority over AOP's or ARPs. In previous revisions of OS-24, this was the recommended course of action.
D.	Perform the initial symptom check, and the VSSVs. Remaining procedures must be implemented in the following order: EOP Rules, EOP Guides, AOPs and Alarm Response procedures	INCORRECT: Plausible if examinee believes Symptom Check and VSSV's are always done prior to moving to another procedure. In previous revisions of OS-24, this was the recommended course of action after the symptom check.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.16	
	Importance Rating	3.5	4.4

K/A: Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal | operating procedures, and severe accident management guidelines.

Proposed Question: RO Question # 44

Technical Reference(s): OS-24, Rev 28

Proposed References to be provided to applicants during examination: None

Learning Objective: None

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question Source: Bank # 503815

Modified Bank #

New

Question History: Sim 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.10

55.43

Comments:

KA MATCH: This question matches the KA because the examinees must have knowledge of the procedure hierarchy in OS-24.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

45

ID: 1719786

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- The reactor has tripped.
- Damage has occurred to the Control Tower, preventing plant control from either the Control Room or the Remote Shutdown Panels.
- On-shift personnel are not responding to any communications.

Based on the above conditions, the priority action to be taken by an Auxiliary Operator is to report to the:

- A. EFW area and perform OP-TM-AOP-009, Loss of Plant Control Facilities in order to establish EFW flow.
- B. Auxiliary Boilers and perform OP-TM-414-401 (402), to start AS-B-1A (1B) in order to establish Auxiliary Steam for necessary loads.
- C. Auxiliary Building and perform Attachment 5 of OP-TM-EOP-020, Cooldown From Outside of Control Room to prevent spurious operation of MOV's.
- D. Turbine Building and open AS-V-8 IAW Attachment E of OS-24, Conduct of Operations During Abnormal and Emergency Events, to make Auxiliary Steam available to Gland Steam when Auxiliary Steam pressure is available.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) According to OS-24, attachment E, in the event of damage to the Control Tower preventing plant control from either the Control Room or Remote Shutdown area all available NLO's should report to the EFW area and implement AOP-009; (2) AOP-009 should be performed immediately without CR concurrence; (3) The mitigation strategy for AOP-009 is to establish EFW flow to the OTSGs to remove decay heat.

A.	EFW area and perform OP-TM-AOP-009, Loss of Plant Control Facilities in order to establish EFW flow.	CORRECT: See above
B.	Auxiliary Boilers and perform OP-TM-414-401 (402), to start AS-B-1A (1B) in order to establish Auxiliary Steam for necessary loads.	INCORRECT: Plausible because this is an assigned post trip action, following a Reactor Trip. Incorrect because OS-24 directs all NLO's to the EFW area to establish feedwater.
C.	Auxiliary Building and perform Attachment 5 of OP-TM-EOP-020, Cooldown From Outside of Control Room to prevent spurious operation of MOV's.	INCORRECT: Plausible because the control room is not available. Incorrect because in this case, EOP-020 is not entered because the whole control tower is not available.
D.	Turbine Building and open AS-V-8 IAW Attachment E of OS-24, Conduct of Operations During Abnormal and Emergency Events, to make Auxiliary Steam available to Gland Steam when Auxiliary Steam pressure is available.	INCORRECT: Plausible because this is an assigned post trip action, following a Reactor Trip.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.35	
	Importance Rating	3.8	4.0

K/A: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Proposed Question: RO Question #45

Technical Reference(s): OS-24, Rev 28 OP-TM-AOP-009, Rev 10

Proposed References to be provided to applicants during examination: None

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Learning Objective: AOP-009-PCO-4

Question Source: Bank # 862353

Modified Bank #

New

Question History: Sim 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

55.43

Comments:

KA Match: This question matches the KA because the examinee must know where the location and actions taken by the AO's on a loss of plant facilities and resultant operational effects.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

46

ID: 1720267

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- A feedwater transient occurs.
- An automatic reactor trip on HIGH RCS temperature fails to occur.
- Main Feedwater remains available.
- The reactor trip and DSS Pushbuttons fail to trip the reactor.
- A CRO opens both the 1L-02 and 1G-02 breakers.
- Several groups of control rods failed to insert.
- Reactor power is 10% and steady.

Based on the above conditions, the CRO must ____ (1) ____ in order to ____ (2) ____ .

- A. (1) trip the Main Turbine,
(2) minimize peak RCS pressure
- B. (1) trip the Main Turbine,
(2) prevent an overcooling event
- C. (1) wait until RCS Pressure is less than 2500 psig, then initiate Emergency Injection HPI/LPI,
(2) inject borated water to the RCS
- D. (1) wait until RCS Pressure is less than 2500 psig, then initiate Emergency Injection HPI/LPI,
(2) prevent RCS pressure from exceeding 3000 psig

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) The reactor did not shutdown with an automatic or manual trip signal, this is an ATWS; (2) When this happens the operator opens 1L-02 and 1G-02, which should remove all power from the CRDMs; (3) After the breakers are open, since multiple groups of rods did not insert, and reactor power is greater than 5%, that indicates that the reactor is still NOT shutdown; (4) Since feedwater is available, the mitigation strategy is to maintain primary to secondary heat transfer and emergency borate with HPI (charging pumps).			
A.	(1) trip the Main Turbine, (2) minimize peak RCS pressure	INCORRECT: Plausible because if Main Feedwater were not available, then the operator would trip the turbine and initiate emergency feedwater.	
B.	(1) trip the Main Turbine, (2) prevent an overcooling event	INCORRECT: Plausible because if Main Feedwater were not available, then the operator would trip the turbine and initiate emergency feedwater.	
C.	(1) wait until RCS Pressure is less than 2500 psig, then initiate Emergency Injection HPI/LPI, (2) inject borated water to the RCS	CORRECT: See above.	
D.	(1) wait until RCS Pressure is less than 2500 psig, then initiate Emergency Injection HPI/LPI, (2) prevent RCS pressure from exceeding 3000 psig	INCORRECT: Plausible since HPI/LPI is initiated when RCS pressure is < 2500 psig, however preventing the RCS Pressure from exceeding 3000 psig is associated with maintaining the Main Turbine on-line.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	029	EA1.01
	Importance Rating	3.4	3.1
K/A: Anticipated Transient Without Scram (ATWS): Ability to operate and monitor the following as they apply to a ATWS: Charging Pumps			
Proposed Question: RO Question # 46			
Technical Reference(s): OP-TM-EOP-001, Rev 16 OP-TM-EOP-0011, Rev 8			
Proposed References to be provided to applicants during examination: None			
Learning Objective: EOP-001-PCO-5			
Question Source: Bank #			
Modified Bank #			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

	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: The examinee must know the the mitigation strategy for shutting down the reactor on an ATWS, which involves using the charging pumps to emergency borate.		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

47

ID: 1720289

Points: 1.00

A reactor trip from 100% power has occurred.

The following plant conditions exist:

- All four reactor coolant pumps are tripped.
- Emergency Feedwater CANNOT be established to either OTSG.
- Main Feedwater CANNOT be recovered.
- Incore thermocouples are indicating 592°F and rising slowly.
- OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, has been entered.
- SCM is 28°F and lowering.

The CRS must __ (1) __ based on __ (2) __.

- A. (1) go to OP-TM-EOP-009, HPI COOLING, and perform Rule 1 when Subcooling Margin is < 25°F,
(2) SCM approaching 25°F
- B. (1) go to OP-TM-EOP-009, HPI COOLING, and perform Rule 1 when Subcooling Margin is < 25°F,
(2) RCS pressure approaching 2450 psig
- C. (1) continue with OP-TM-EOP-004 to open the PORV then close it when RCS pressure is 1750 psig,
(2) RCS pressure approaching 2450 psig
- D. (1) continue with OP-TM-EOP-004 to open the PORV then close it when SCM approaches 30°F,
(2) SCM approaching 25°F

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-EOP-004, if SCM is approaching 25°F, the crew will go to OP-TM-EOP-009; (2) Additionally, the examinee must determine this is the correct course of action by using the incore temperature at 592°F and the SCM at 28°F to determine that RCS pressure is approximately 1800 psig (from the steam tables); (3) OP-TM-EOP-009 could be entered in two separate ways from OP-TM-EOP-004: The first was previously discussed when SCM is approaching 25°F, the second is if RCS pressure is approaching 2450 psig with no FEEDWATER available; (4) Due to SCM being 28°F and the examinee determining that pressure is not approaching 2450 psig, entry into OP-TM-EOP-009 is required via the approaching 25°F SCM method.

A.	(1) go to OP-TM-EOP-009, HPI COOLING, and perform Rule 1 when Subcooling Margin is < 25°F, (2) SCM approaching 25°F	CORRECT: See above.
B	(1) go to OP-TM-EOP-009, HPI COOLING, and perform Rule 1 when Subcooling Margin is < 25°F, (2) RCS pressure approaching 2450 psig	INCORRECT: Plausible because OP-TM-EOP-009 is required, incorrect because it is required because SCM is approaching 25°F, not due to RCS pressure approaching 2450 psig.
C.	(1) continue with OP-TM-EOP-004 to open the PORV then close it when RCS pressure is 1750 psig, (2) RCS pressure approaching 2450 psig	INCORRECT: Plausible because OP-TM-EOP-004 provides guidance on RCS pressure approaching 2450 psig, which would be to deep cycle the PORV. Incorrect because this is only done if FEEDWATER is available.
D.	(1) continue with OP-TM-EOP-004 to open the PORV then close it when SCM approaches 30°F, (2) SCM approaching 25°F	INCORRECT: Plausible because OP-TM-EOP-004 provides guidance on deep cycling the PORV when RCS pressure is approaching 2450 psig.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	E04	EK1.2
	Importance Rating	4.0	4.2

K/A: Inadequate Heat Transfer: Knowledge of the operational implications of the following concepts as they apply to the (Inadequate Heat Transfer): Normal, abnormal, and emergency operating procedures associated with (Inadequate Heat Transfer).

Proposed Question: RO Question # 47

Technical Reference(s): OP-TM-EOP-004, Rev 11 OP-TM-EOP-009, Rev 8

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Learning Objective: EOP-004-PCO-5

Question Source: Bank # 300014

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.10

Comments:

KA Match: This question matches the KA because the examinee must to the operational implication of approaching 25°F during a lack of heat transfer. They must have knowledge of the transition to OP-TM-EOP-009.

High Cog: This question is high cog because the examinee must analyze the conditions in the stem or in order to determine the correct step to implement in OP-TM-EOP-004.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

48

ID: 1720350

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Loss of #8 Bus

Based on the Loss of the #8 Bus, the CRS must enter __ (1) __ and the CRO is required to make a plant announcement using the __ (2) __.

- A. (1) OP-TM-AOP-013, LOSS OF 1D 4160V BUS, (2) "RED" plant page and radio
- B. (1) OP-TM-AOP-013, LOSS OF 1D 4160V BUS, (2) "GREY" plant page and radio
- C. (1) OP-TM-AOP-014, LOSS OF 1E 4160V BUS, (2) "RED" plant page and radio
- D. (1) OP-TM-AOP-014, LOSS OF 1E 4160V BUS, (2) "GREY" plant page and radio

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) When the plant loses the 8 bus, the 1A and 1C 4160V busses fast transfer to the B Auxiliary Transformer. Emergency Diesel Generator, EG-Y-1B starts to power the 1E 4160V bus; (2) This meets the entry criteria for entering OP-TM-AOP-014; (3) The power to the "Grey" plant page comes downstream of the 1D 4160V bus and is not effected, hence the plant page is made over the "Grey" phone; (4) OP-TM-AOP-013 directs the plant page to be made over the "Red" phone if the 1D 4160V bus lost power.			
A.	(1) OP-TM-AOP-013, LOSS OF 1D 4160V BUS, (2) "RED" plant page and radio	INCORRECT: Plausible if the examinee believes that the 1D 4160V bus was lost. OP-TM-AOP-013 would direct the announcement to be made over the "RED" plant page.	
B.	(1) OP-TM-AOP-013, LOSS OF 1D 4160V BUS, (2) "GREY" plant page and radio	INCORRECT: Plausible if the examinee believes that the 1D 4160V bus was lost. Although the procedure directs use of the "RED" plant page and radio, when the power was restored to the 1D 4160V bus by EG-Y-1A, the "GREY" plant page would work.	
C.	(1) OP-TM-AOP-014, LOSS OF 1E 4160V BUS, (2) "RED" plant page and radio	INCORRECT: Plausible if the examinee believes that the "GREY" plant page has lost power. Incorrect because the "GREY" plant page gets power from downstream of the 1D 4160V bus.	
D.	(1) OP-TM-AOP-014, LOSS OF 1E 4160V BUS, (2) "GREY" plant page and radio	CORRECT: See above	
Examination Outline Cross-reference:			
Level		RO	SRO
Tier #		1	
Group #		2	
K/A #		A05:	AA2.1
Importance Rating		3.5	4.2
K/A: Emergency Diesel Actuation: Facility conditions and selection of appropriate procedures during abnormal and emergency operations.			
Proposed Question: RO Question #48			
Technical Reference(s): OP-TM-AOP-014, Rev 9			
Proposed References to be provided to applicants during examination: None			
Learning Objective: AOP-014-PCO-6			
Question Source: Bank #			
Modified Bank #			
New X			
Question History: N/A Last NRC Exam: N/A			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 diesel actuates on undervoltage on a loss of the #8 bus. The examinee must recognize the entry criteria for the abnormal procedure which would govern this actuation.		
High Cog: This question is high cog because the examinee must analyze the plant status upon loss of the 8 bus, and understand that the 1E 4160V bus is powered by EG-Y-1B, and this is an entry criteria for OP-TM-AOP-014.		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

49

ID: 1720342

Points: 1.00

Plant Conditions:

- 40% power with ICS in auto.

EVENT:

- Reactor power is LOWERING.
- RCS pressure is RISING.
- Main Steam Safety Valves are OPEN.
- Turbine Bypass Valves, MS-V-3A-F, and Atmospheric Dump Valves, MS-V-4A/B, are OPEN.
- Indicating lights on Panel SS-1 are GREEN for the breakers for the Middletown 1092, Jackson 1051 and 500 kV tie lines.
- Indicating lights on Panel SS-1 for the Middletown 1091 breaker switches are GREEN and YELLOW.
- Indicating lights for the Main Generator Breakers are RED.
- Generator load is 49 megawatts.

Which of the following procedures must be entered to mitigate this event?

- A. MAP-H0101, ICS RUNBACK
- B. 1107-11, TMI GRID OPERATIONS
- C. OP-TM-EOP-001, REACTOR TRIP
- D. OP-TM-AOP-022, LOAD REJECTION

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) The stem conditions essentially leave the plant in a conditions where the unit is separated from the grid, where our generator breakers are shut powering our Auxiliary Transformers only; (2) This could be caused by multiple things, but the governing procedure in this case will be OP-TM-AOP-022, LOAD REJECTION.

A.	MAP-H0101, ICS Runback	INCORRECT: Plausible because both reactor power and MWe are lowering which is also an indication of a plant runback. Incorrect because no runback signal would exist for this situation.
B.	1107-11, TMI Grid Operations	INCORRECT: Plausible because the purpose of this procedure is to provide guidance on operation of components related to Auxiliary Transformers/Switchyard Equipment and to provide guidance for response to grid related abnormal situations. The given symptoms would indicate a potential problem associated with the grid, but the controlling procedure would be AOP-022.
C.	OP-TM-EOP-001, Reactor Trip	INCORRECT: Plausible since many of the symptoms are also associated with a reactor trip. Incorrect because there is indication of generator load, which would indicate the turbine and the reactor are NOT tripped.
D.	OP-TM-AOP-022, Load Rejection.	CORRECT: See above

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	077	2.4.11
	Importance Rating	4.0	4.2

K/A: Generator Voltage and Electrical Grid Disturbances: Knowledge of abnormal condition procedures

Proposed Question: RO Question # 49

Technical Reference(s): OP-TM-AOP-022, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-022-APCO-2

Question Source: Bank # 462877

Modified Bank #

New

Question History: Sim Exam 6 Last NRC Exam: N/A

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 abnormal operating procedure entered on a loss of the grid.

High Cog: This question is high cog because the examinee must analyze the stem indications to determine that a load rejection has occurred.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

50

ID: 1720492

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- RC3-PR Hot Leg A Narrow Range Channel 1 fails LOW.
- MAP alarm H-3-2, SASS Mismatch is illuminated.
- SASS selector switch for RC3A-PT1 Hot Leg A Narrow Range Channel 1 has the RED and WHITE lights illuminated.
- SASS selector switch for RC3B-PT1 Hot Leg B Narrow Range Channel 1 has only the RED light illuminated.

What is the system response of this failure?

- A. All Pressurizer Heater Banks energize.
- B. SASS transfers to an alternate channel.
- C. One channel of ES actuates on Train 'A' AND Train 'B'.
- D. Only SCR controlled banks 1, 2, and 3 Pressurizer Heaters energize.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) The range of the Narrow Range pressure instrument is 2500 to 1700 psig, and failing low would be 1700 psig; (2) The Narrow Range pressure instrument, RC3A-PT1, is the normally selected SASS instrument to RC3-PR for RCS pressure control; (3) This output of this SASS channel provides control for, PZR Heater Control (Low Pressure), Spray Valve control, and the PORV high pressure setpoint; (4) The effect of this failing low the Pressurizer Heater Banks energize at 2135 psig (SCR Controlled 1, 2 and 3)) and all heater banks (4 and 5) are energized by 2105.			
A.	All Pressurizer Heater Banks energize.	CORRECT: See above	
B.	SASS transfers to an alternate channel.	INCORRECT: Plausible if the examinee does not interpret the RED and WHITE light indications correctly, and determines that SASS has actuated. Incorrect because the WHITE light on RC3B-PT1 would be illuminated.	
C.	One channel of ES actuates on Train 'A' AND Train 'B'.	INCORRECT: Plausible if the examinee believes this pressure instrument inputs to ESAS in a way which would actuate one channel. Incorrect because these Narrow Range instruments only go as low as 1700 psig, and ESAS actuates on a 1600 psig pressure signal.	
D.	Only SCR Controlled Banks 1, 2, and 3 Pressurizer Heaters energize.	INCORRECT: Plausible if examinee thinks the instrument input only effects the SCR Controlled Heater Banks 1, 2, and 3. Incorrect because the bistable banks (4 and 5) are will also turn on for this failure.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010	K6.01
	Importance Rating	2.7	3.1
K/A: Pressurizer Pressure Control System: Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems			
Proposed Question: RO Question # 50			
Technical Reference(s): TQ-TM-104-624-C001, Rev 4			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 624-GLO-11			
Question Source: Bank #			
Modified Bank #			
New X			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 the controlling Narrow Range Pressure instrument effects plant control.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

51

ID: 1720498

Points: 1.00

Identify the conditions that promote Pool Boiler Condenser Cooling Mode (BCM) of Heat Transfer.

The OTSG thermal center remains __ (1) __ the RCP spillover elevation.
RCS liquid level is maintained __ (2) __ the secondary level.

- A. (1) above
(2) above
- B. (1) above
(2) below
- C. (1) below
(2) above
- D. (1) below
(2) below

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.5
	55.43	
Comments:	<p>KA Match: This question matches the KA because the examinee must have knowledge for the response of raising OTSG level on a loss of subcooling margin. The operating characteristic in the question is Boiler Condenser Cooling, and the effect in the question is what the temperature and level of OTSG water will do to the RCS parameters.</p>	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

52

ID: 1720537

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- RPS Channel A in MANUAL BYPASS.

EVENT:

- Vital bus "A" (VBA) losses power.

Given the information above, identify the selection below that describes (1) the number of remaining channels required to trip and (2) the degree of redundancy.

- A. (1) ONE (2) ONE
- B. (1) ONE (2) TWO
- C. (1) TWO (2) ONE
- D. (1) TWO (2) TWO

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) The MANUAL BYPASS switch on an RPS cabinet is used for testing or if an erroneous trip signal is present; (2) With an RPS channel in MANUAL BYPASS, the number of channels required to trip is TWO with a redundancy of TWO; (3) When VBA is lost, a trip signal is generated, regardless of the position of the MANUAL BYPASS switch; (4) This places the plant in a situation where if ONE additional channel generates a trip signal, the plant would trip; (5) Since RPS channels B,C, and D are still operable, there is a degree of redundancy of TWO additional channels which could provide the trip signal.

A.	(1) ONE (2) ONE	INCORRECT: Plausible because this possibility would exist if a power supply other than VBA was lost (i.e, VBB, VBC, or VBD). understood.
B.	(1) ONE (2) TWO	CORRECT: See above
C.	(1) TWO (2) ONE	INCORRECT: Plausible if the examinee believes that since RPS A is in MANUAL BYPASS, that this prohibits all trip signal to be generated. Incorrect because this trip signal is generated on a loss of power.
D.	(1) TWO (2) TWO	INCORRECT: Plausible if the examinee believes that since RPS A is in MANUAL BYPASS, that this prohibits all trip signal to be generated. Incorrect because this trip signal is generated on a loss of power..

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012	K6.02
	Importance Rating	2.9	3.1

K/A: Reactor Protection System: Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Redundant channels

Proposed Question: RO Question #52

Technical Reference(s): TQ-TM-104-641-C001, Rev 2 T.S Definition 1.4.2, Amd 278
OP-TM-AOP-015, Rev 10

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO-6

Question Source: Bank # 568246
Modified Bank #
New

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 loss of a vital bus will have on the redundant channels of RPS.	
High Cog:	This question is high cog because the examinee must analyze the effect of the loss of VBA on the RPS system.	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

53

ID: 1720550

Points: 1.00

Initial conditions:

- 100% Reactor power.
- AH-E-9A, PENETRATION COOLING FAN, is in Normal-after-start.
- AH-E-9B, PENETRATION COOLING FAN, is in Normal-after-stop.

Event:

- Reactor Trip occurs.
- 4# ES actuation occurs (both trains).
- HVA-2-1, PENETRATION COOLING AIR TEMP HI, alarms.
- Recorder TR-805, PENETR COOL AIR TEMPERATURE RECORDER, points 1-25, all read >205°F and rising.

What MINIMUM action(s) must be taken to lower penetration temperatures?

- A. Start AH-E-9B ONLY.
- B. Restart AH-E-9A ONLY.
- C. Bypass both 4# ES signals and then start AH-E-9B.
- D. Ensure AH-D-89, AH-E-9A/B OUTSIDE AIR INLET, and AH-D-90, AH-E-9A/B TURBINE BLDG INLET, are open.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) That AH-E-9A/9B trip on the following signals: Fire/Smoke detector alarm (TS 671), High Temperature at 195°F (TS 706A/706B), and AH-E-9A is load shed on any ES actuation; (2) When a component is load shed, it cannot be restarted until the ES Actuation signals are cleared; (3) Since AH-E-9B is not affected by load shed, it can be started immediately.			
A.	Start AH-E-9B ONLY.	CORRECT: See above. This is the MINIMUM requirement to start a penetration cooling fan.	
B.	Restart AH-E-9A ONLY.	INCORRECT: Plausible if the examinee believes that AH-E-9B is the load shed fan or does not recall the ES Signal must be cleared to restart fan.	
C.	Bypass both 4# ES signals and then start AH-E-9B.	INCORRECT: Plausible if the examinee believes that both fans are load shed and the correct course of action was to start the fan that was not running.	
D.	Ensure AH-D-89, AH-E-9A/B OUTSIDE AIR INLET, and AH-D-90, AH-E-9A/B TURBINE BLDG INLET, are open.	INCORRECT: Plausible if the examinee believes that BOTH the inlet dampers to the AH-E-9's must be open to start an AH-E-9.	
Examination Outline Cross-reference:			
Level		RO	SRO
Tier #		2	
Group #		1	
K/A #		022	K4.01
Importance Rating		2.5	3.0
K/A: Containment Cooling System: Knowledge of the CCS design feature(s) and/or interlock(s) which provide for the following: Cooling of containment penetrations.			
Proposed Question:		Question #53	
Technical Reference(s):		TQ-TM-104-240-C001, Rev 7 HVA-2-1 (HVA, Rev 58)	
		1104-16, Rev 24	
Proposed References to be provided to applicants during examination:		None	
Learning Objective:		240-GLO-10	
Question Source:		Bank # 880836	
		Modified Bank #	
		New	
Question History:		System Exam 9 Last NRC Exam: N/A	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 load shed interlock for AH-E-9A to answer the question correctly.

High Cog: This question is high cog because in addition to knowing the interlock with AH-E-9A, the examinee must know the actions to start another penetration cooling fan.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

54

ID: 1720590

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Due to a fault, the feeder breaker to 1D 4160V bus trips open.
- Emergency Diesel Generator, EG-Y-1A is powering the 1D 4160V bus.

The 1N 480V Bus is __ (1) __ because __ (2) __.

- A. (1) energized
(2) the 1N 480V bus automatically repowers from the 1L 480V bus crosstie
- B. (1) energized
(2) the 1N 480V bus is repowers when EG-Y-1A repowers the 1D 4160V bus
- C. (1) de-energized
(2) the UV results in a 1N 480V bus trip and lockout
- D. (1) de-energized
(2) the UV results in a 1N 480V bus trip with NO lockout

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 requires knowledge of the one line diagram of the TMI electrical distribution system.			
High Cog: This question is high cog because the examinee must know the status of the 1N 480V bus when the associated 4160V bus loses power and is then repowered.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

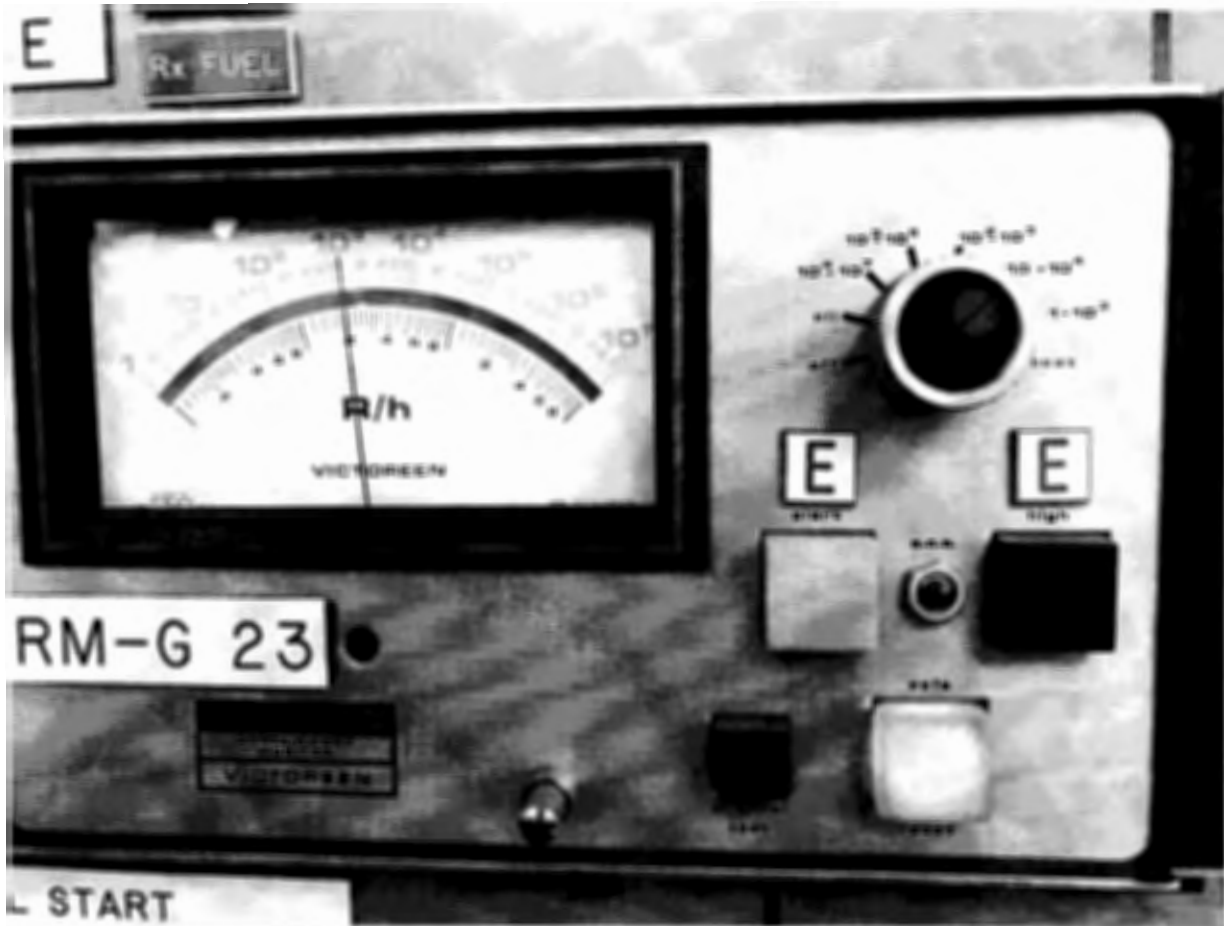
EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

55

ID: 1720616

Points: 1.00



Identify the correct reading for RM-G-23, RB High Range Radiation Monitor.

- A. 1×10^3 R/h
- B. 1×10^3 mr/h
- C. 2×10^2 R/h
- D. 2×10^2 mr/h

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1)The operation of the indication selector switch for RM-G-23, OFF/ALL/10E7/10E6/10E5/10E4/10E3:
1) Read top scale (RED) in ALL .
2) Read 3 decades in other positions.
Example: Scale Selected 10 - 10E4: Read lower black scale with first mark being 10E1 R/hr and the last mark being 10E4 R/hr
In addition, the black scale does NOT lower the Units to mr/hr.

A.	1X10 ³ R/h	INCORRECT: Plausible if the examinee does correctly interpret the position of the Scale Selector Switch
B.	1X10 ³ mr/h	INCORRECT: Plausible if the examinee does not properly read the scale units. Most Rad Monitors of this type are low range and read out in mr/hr.
C.	2X10 ² R/h	CORRECT: See above
D.	2X10 ² mr/h	INCORRECT: Plausible if the examinee misinterprets the black scale as mr/hr.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073	2.1.28
	Importance Rating	4.1	4.1

K/A: Process Radiation Monitoring System: Knowledge of the purpose and function of major system components and controls.

Proposed Question: RO Question #55

Technical Reference(s): TQ-TM-104-661

Proposed References to be provided to applicants during examination: RM-G-23 indication (in color)

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

	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 the function controls for a process radiation monitor.		
High Cog: This question is high cog because the examinee must use the picture to determine the correct reading by checking the switch position and the meter reading.		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

56

ID: 1720650

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Three 'A' OTSG Main Steam Safety Valves (MSSVs) are discovered to be inoperable.
- All 'B' OTSG Main Steam Safety Valves are operable.

- (1) What action, if any, is required?
(2) Why or why not?

- A. (1) No action is required.
(2) All 'B' OTSG MSSVs are operable.
- B. (1) No action is required.
(2) All 3 MSSVs are inoperable on the 'A' OTSG.
- C. (1) Reduce power and reduce the overpower trip setpoint.
(2) All 'B' OTSG MSSVs are operable.
- D. (1) Reduce power and reduce the overpower trip setpoint.
(2) Only 6 MSSVs are operable on the 'A' OTSG.

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) The number of MSSVs on each OTSG is nine; (2) If less than seven MSSVs are operable on either OTSG, power must be reduced and the overpower trip must be lowered as well.			
A.	(1) No action is required. (2) Only 'B' OTSG MSSVs are operable.	INCORRECT: Plausible because the examinee could believe that since the 'B' OTSG MSSVs are operable that no action is required.	
B.	(1) No action is required. (2) All 3 MSSVs are inoperable on the 'A' OTSG.	INCORRECT: Plausible because the examinee could believe that since only 3 MSSVs on the 'A' OTSG are inoperable that no action is required.	
C.	(1) Reduce power and reduce the overpower trip setpoint. (2) All 'B' OTSG MSSVs are operable.	CORRECT: Plausible because the examinee could believe that the reason for the power and setpoint reduction is that ONLY the 'B' OTSG MSSVs are operable. This is incorrect because the action is taken because 3 MSSVs are inoperable on the 'A' OTSG.	
D.	(1) Reduce power and reduce the overpower trip setpoint. (2) Only 3 MSSVs are inoperable on the 'A' OTSG.	INCORRECT: See above.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	041	A1.02
	Importance Rating	3.1	3.2
K/A: Steam Dump System (SDS)/Turbine Bypass Control: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam Pressure			
Proposed Question: RO Question #56			
Technical Reference(s): TQ-TM-104-411-C001, Rev 8 T.S. 3.4.1.2.3, Amd 277			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 411-GLO-7			
Question Source: Bank #			
Modified Bank #			
New X			
Question History: N/A Last NRC Exam: N/A			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 it tests the ability of the examinee to predict a change in parameters to prevent exceeding a design limit associated with the Steam Dump System (MSSVs). The setpoint is reduced to prevent exceeding the Main Steam System pressure rating by exceeding the capacity of the remaining operable MSSVs.		
High Cog: This question is high cog because the examinee must determine the correct course of action when 3 MSSVs are found to be inoperable.		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

57

ID: 1720668

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

Sequence of Events:

- The 'B' OTSG develops a steam leak in the Reactor Building.
- The crew manually trips the reactor based on rising Reactor Building pressure.
- A 4 PSIG ESAS actuation occurs.
- Current plant conditions:
 - RCS T_{ave} is 560°F and lowering slowly.
 - OTSG 1A level is 86 inches on the Startup Range and slowly lowering.
 - OTSG 1B level is 86 inches on the Startup Range and slowly lowering.
 - OTSG 1A pressure is 985 psig and steady.
 - OTSG 1B pressure is 950 psig and lowering very slowly.
 - Main Feedwater flow to OTSG 1A is 0.3×10^5 lbm/hr.
 - Main Feedwater flow to OTSG 1B is 1.9×10^5 lbm/hr

The 'B' OTSG must be isolated in accordance with which procedure?

- A. OP-TM-EOP-001, REACTOR TRIP
- B. OP-TM-EOP-010, Guide 12, RCS STABILIZATION
- C. OP-TM-AOP-051, SECONDARY SIDE HIGH ENERGY LEAK
- D. OP-TM-EOP-003, EXCESSIVE PRIMARY TO SECONDARY HEAT TRANSFER

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) If a steam leak in the reactor building occurs, OP-TM-AOP-051 would be entered; (2) When the Reactor Building pressure approaches 2 psig (which could happen on an OTSG steam leak), the crew will trip the reactor; (3) To trip the reactor the crew will perform the EOP-001 Immediate Manual Actions and then perform a symptom check; (4) In this case, the crew would be sensitive to symptoms of excessive heat transfer, which in accordance with the OS -24 definition, does NOT exist at this time; (5) The operating crew would proceed with EOP-001 actions until they reached step 3.16 which would isolate the affected OTSG per Attachment 1.

A.	OP-TM-EOP-001, REACTOR TRIP	CORRECT: See above.
B.	OP-TM-EOP-010, Guide 12, RCS STABILIZATION	INCORRECT: Plausible because Guide 12 is implemented from Rule 3 (which is performed in EOP-003). There are steps in Guide 12 to isolate the OTSG from the condenser, but that is not applicable in this question.
C.	OP-TM-AOP-051, Secondary Side High Energy Leak.	INCORRECT: Plausible because the crew does enter this procedure, but incorrect because this procedure doesn't isolate the OTSG for a leak in the Reactor Building, but does direct a reactor trip at 2 psig in the Reactor Building.
D.	OP-TM-EOP-003, Excessive Primary to Secondary Heat Transfer.	INCORRECT: Plausible since EOP-003 would isolate the OTSG if symptoms of excessive heat transfer existed. Incorrect because those symptoms do not exist.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103	2.4.6
	Importance Rating	3.7	4.7

K/A: Containment System: Knowledge of EOP mitigation strategies.

Proposed Question: RO Question # 57

Technical Reference(s): OP-TM-EOP-001, Rev 16

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-001-PCO-4

Question Source: Bank # 908527
Modified Bank #
New

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 mitigation strategy for an OTSG with a steam leak in Containment.

High Cog: This question is high cog because the examinee must analyze the conditions in the stem to determine that the symptoms of excessive heat transfer do NOT exist.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

58

ID: 1720680

Points: 1.00

Plant Conditions:

- 65% reactor power.
- ULD is in manual, all other ICS stations are in auto.
- Both SASS modules are de-energized.

EVENT:

- RC-P-1C breaker opens.

The output of RC-12-TAS Tave Auto/Manual Switch will __ (1) __ because __ (2) __.

- A. (1) remain at the Loop A&B Average
(2) SASS is inoperable
- B. (1) remain at the Loop A&B Average
(2) reactor power is below the ICS Runback setpoint
- C. (1) swap to Loop A Average
(2) the RC-P-1C breaker is open
- D. (1) swap to Loop A Average
(2) this loop has the highest RCS flow

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) RC-12-TAS Selector Switch is normally selected to Loop A&B Average; (2) The logic monitors RCS flow and if a RCS Loop Flow <9 6%, RC-12-TAS automatically swaps to the RCS Loop with the most flow; (3) which is in this case is the Loop A Average; (4) RCS Flow input to Tave selector switch is NOT an input to SASS, so the status of SASS has NO effect on the selector switch.																		
A.	(1) remain at the Loop A&B Average (2) SASS is inoperable	INCORRECT: Plausible if the examinee thinks RCS Flow is a SASS input to the logic for RC-12-TAS. RCS flow is a parameter that is SASED for input to ICS.																
B.	(1) remain at the Loop A&B Average (2) reactor power is below the ICS Runback setpoint	INCORRECT: Plausible because in most cases a RCP trip would cause a plant runback, but in this case the plant is below the runback setpoint. No runback would occur from a loss of this reactor coolant pump. If the examinee believes that since NO runback took place, that RC-12-TAS would not swap because reactor power is below the ICS runback setpoint.																
C.	(1) swap to Loop A Average (2) the RC-P-1C breaker is open	INCORRECT: Plausible since RC-12-TAS would swap to Loop A Average but NOT because the RCP breaker is open. The correct input is RCS Flow. An RCP Breaker trip is input to the ICS runback circuit.																
D.	(1) swap to Loop A Average (2) this loop has the highest RCS flow	CORRECT: See above																
<p>Examination Outline Cross-reference:</p> <table border="1"> <tr> <td>Level</td> <td>RO</td> <td>SRO</td> </tr> <tr> <td>Tier #</td> <td>2</td> <td></td> </tr> <tr> <td>Group #</td> <td>1</td> <td></td> </tr> <tr> <td>K/A #</td> <td>003</td> <td>K5.03</td> </tr> <tr> <td>Importance Rating</td> <td>3.1</td> <td>3.5</td> </tr> </table> <p>K/A: Reactor Coolant Pump System: Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP shutdown on T-ave., including the reason for the unreliability of T-ave. in the shutdown loop</p> <p>Proposed Question: RO Question #58</p> <p>Technical Reference(s): TQ-TM-104-624-C001, Rev 4 D554566 Analog Logic</p> <p>Proposed References to be provided to applicants during examination: None</p> <p>Learning Objective: 624-GLO-10</p> <p>Question Source: Bank #</p> <p>Modified Bank #</p> <p>New X</p>				Level	RO	SRO	Tier #	2		Group #	1		K/A #	003	K5.03	Importance Rating	3.1	3.5
Level	RO	SRO																
Tier #	2																	
Group #	1																	
K/A #	003	K5.03																
Importance Rating	3.1	3.5																

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 the operational implication of a loss of flow in one reactor coolant loop, and the effect on the Tave system.			
High Cog: This question is high cog because the examinee must analyze the conditions in the stem to understand the plant response for a reactor coolant pump trip.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

59

ID: 1720699

Points: 1.00

With the Enable/Defeat switch for DC-V-2A/2B Decay Closed Cooler Inlet and DC-V-65A/65B Decay Closed Cooler Bypass placed in the DEFEAT position, this would result in DC-V-2A/2B ____ (1) ____ AND DC-V-65A/65B ____ (2) ____ .

- A. (1) full closed
(2) full open
- B. (1) full closed
(2) full closed
- C. (1) full open
(2) full closed
- D. (1) full open
(2) full open

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) There are key switches in the control room which need to be in the ENABLE position to be able to control each respective set of valves (DC-V-2A/65A and DC-V-2B/65B); (2) When the switches are in DEFEAT a 3 way solenoid valve fails each set of valves to their full cooling positions (DC-V-2A/2B full OPEN, and DC-V-65A/65B full CLOSED); (3) This ensures full cooling to the Decay Heat Removal System on an ES actuation.			
A.	(1) full closed (2) full open	INCORRECT: Plausible if the examinee believes that the purpose of the key switch is to fail the valves to the NO COOLING position.	
B.	(1) full closed (2) full closed	INCORRECT: Plausible if the examinee believes the purpose of the key switch is to start with the valves fully closed and have a more controlled transition to using the DHR system for cooling. A flow path would still exist through the various pumps that the Decay Closed system provides cooling for.	
C.	(1) full open (2) full closed	CORRECT: See above.	
D.	(1) full open (2) full open	INCORRECT: Plausible if the examinee believes that the purpose of the switch in defeat is to fail the valves in a position where the pumps would have a maximum flow.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	005	A4.02
	Importance Rating	3.4	3.1
K/A: Residual Heat Removal System: Ability to manually operate and/or monitor in the control room: Heat exchanger bypass flow control			
Proposed Question: RO Question #59			
Technical Reference(s): TQ-TM-104-533-C001, Rev 8			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 533-GLO-4			
Question Source: Bank # 468123			
Modified Bank #			
New			
Question History: System Exam 10 Last NRC Exam: N/A			
Question Cognitive Level: Memory or Fundamental Knowledge X			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 operate the heat exchanger bypass flow valve from the control room.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

60

ID: 1720769

Points: 1.00

Which one of the following Alternate Emergency Boration source tanks is required in accordance with 1301-1, SHIFT AND DAILY CHECKS.

Boric Acid Mix Tank (BAMT)
Borated Water Storage Tank (BWST)
Reactor Coolant Bleed Tanks (RCBTs)
Reclaimed Boric Acid Tanks (RBATs)

- A. BWST OR RCBT
- B. BAMT OR RCBT
- C. BAMT OR RBAT "A" OR RBAT "B"
- D. BWST OR RBAT "A" OR RBAT "B"

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) When the TMI-1 is in a condition greater than HOT SHUTDOWN, a reactor operator performs Shift and Daily checks to ensure technical specifications are being met on key equipment to ensure compliance with LCOs. (2) One of the checks is to check what the alternate emergency boration source is and to ensure the tank meets the requirements to the alternate source; (3) The procedure identifies the BAMT, RBAT "A", or RBAT "B" as those sources.

A.	BWST OR RCBT	INCORRECT: Plausible since the BWST is a source of emergency boration but is not considered an alternate source. In addition, two of the three RCBTs contain a boron concentration of equal to or greater than the current RCS boron, so it is plausible that they could be considered a source as well.
B.	BAMT OR RCBT	INCORRECT: Plausible since the BAMT is one source and two of the three RCBTs contain a boron concentration of equal to or greater than the current RCS boron, so it is plausible that they could be considered a source as well.
C.	BAMT OR RBAT "A" or RBAT "B"	CORRECT: See above.
D.	BWST OR RBAT "A" or RBAT "B"	INCORRECT: Plausible since the RBATs are an alternate source but not the BWST.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	024	2.2.12
	Importance Rating	3.7	4.1

K/A: Emergency Boration: Knowledge of surveillance procedures.

Proposed Question: RO Question # 60

Technical Reference(s): 1301-1, Rev 176 TQ-TM-104-561-C001, Rev 4
OP-TM-EOP-010, Guide 1, Rev 19

Proposed References to be provided to applicants during examination: NONE

Learning Objective: 561-GLO-7

Question Source: Bank #
Modified Bank #
New X

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 procedure 1301-1, SHIFT AND DAILY CHECKS, is an surveillance procedure that checks the status of technical specification LCOs. Within that procedure, there is a section to ensure the alternate emergency boration status.		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

61

ID: 1720771

Points: 1.00

During normal 100% power operations, the steam supply to the Main Feedwater Pumps is the outlet of the __ (1) __, and shortly following a Reactor/Turbine trip, the source of steam to the Main Feedwater Pumps is __ (2) __.

- A. (1) "D" AND "F" Moisture Separators
(2) Main Steam
- B. (1) "D" AND "F" Moisture Separators
(2) Auxiliary Steam
- C. (1) HP Turbine prior to "C" AND "D" Moisture Separators
(2) Main Steam
- D. (1) HP Turbine prior to "C" AND "D" Moisture Separators
(2) Auxiliary Steam

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) At 100% power, due to the design of the Main Feedwater Pump poppets, they prefer the lower pressure steam from the outlet of the "D" and "F" Moisture Separators; (2) This steam goes away when the Reactor/Main Turbine are shutdown (or tripped as in this question) so the Main Feedwater Pumps transition to the Main Steam System (via MS-V-5A/B) for their source of steam (approximately 5% to 24% power); (3) Auxiliary Steam is used during shutdown conditions.

A.	(1) "D" AND "F" Moisture Separators (2) Main Steam	CORRECT: See above
B.	(1) "D" AND "F" Moisture Separators (2) Auxiliary Steam	INCORRECT: Plausible because Auxiliary Steam is used as a source of steam for the Main Feedwater Pumps. Incorrect because it isn't used until the Auxiliary Boilers are running after a reactor trip.
C.	(1) HP Turbine prior to "C" AND "D" Moisture Separators (2) Main Steam	INCORRECT: Plausible because there are steam line connections at that location. Incorrect because those steam lines to go feedwater heating.
D.	(1) HP Turbine prior to "C" AND "D" Moisture Separators (2) Auxiliary Steam	INCORRECT: Plausible because there are steam line connections at that location. Incorrect because those steam lines to go feedwater heating. In addition Auxiliary Steam is not used right after a reactor trip.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039	K1.08
	Importance Rating	2.7	2.9

K/A: Main and Reheat Steam System: Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: MFW

Proposed Question: Question #61

Technical Reference(s): 302-011 302-041
TQ-TM-104-411-C001, Rev 8 TQ-TM-104-431-C001, Rev 4

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-5

Question Source: Bank #
Modified Bank # 355620
New

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.7
	55.43	
Comments:		
<p>KA Match: This question matches the KA because the examinee must know the physical connection between the Main Steam System and the Main Feedwater Pumps, in addition the cause-effect of a reactor trip on both systems.</p> <p>Five minutes after a Reactor/Turbine Trip from 100% power, the source of steam to the main feedwater pumps is the:</p> <p>A. Main Steam System.</p> <p>B. Auxiliary Steam System.</p> <p>C. Outlet of the "C" Moisture Separator.</p> <p>D. 4th stage of the Extraction Steam System.</p> <p>Answer: A</p>		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

62

ID: 1720783

Points: 1.00

Plant Condition:

- 100% power with ICS in full auto.

EVENT

- Loss of Offsite Power

Subsequently:

- 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: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) On a Loss of Offsite Power, all Reactor Coolant Pumps trip; (2) When this happens, the HSPS setpoint sent to the EF-V-30s for OTSG level control is 50% in the operating range to promote natural circulation; (3) When the HSPS switches are taken to DEFEAT, the 50% operating range signal is still locked in, until the operator breaks the 'seal-in' relay by taking each EF-V-30 to manual; (4) Since the stem does not indicate that this happens, the setpoint remains 50% in the operating range.			
A.	0" on the startup range.	INCORRECT: Plausible because the examinee could believe that HSPS has lost power on the loss of offsite power, and that OTSG level control will have to be in manual. Incorrect because HSPS is powered from the Vital Busses, and they do not lose power on a loss of offsite power.	
B.	25" on the startup range.	INCORRECT: Plausible because on a loss of offsite power, both Main Feedwater Pumps would trip. When the Main Feedwater Pumps trip, HSPS sends a signal to control OTSG level at 25" in the startup range. Incorrect because the loss of reactor coolant pump signal has priority.	
C.	50% on the operating range	CORRECT: See above	
D.	75-85% on the operating range.	INCORRECT: Plausible because the operator would feed to this range in accordance with Rule 4 of OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS if subcooling margin were gone. Feeding this high would also promote natural circulation more. Incorrect because the operator has to take manual control to feed to this level.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	061	A3.03
	Importance Rating	3.9	3.9
K/A: Auxiliary/Emergency Feedwater (AFW) System: Ability to monitor automatic operation of the AFW, including: AFW S/G level control on automatic start			
Proposed Question:		RO Question #62	
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		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 the automatic operation of the EFW system that should occur on a loss of offsite power, which includes the signal that HSPS controls OTSG level.

High Cog: This question is high cog because the examinee must know the plant response on a loss of offsite power, and the effect on HSPS level control.

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

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

63

ID: 1724247

Points: 1.00

Plant Conditions:

- 40% power with ICS in auto.
- Main Condenser Tube Leak on the "A" side.
- Circulating Water Pumps, CW-P-1A, CW-P-1B, CW-P-1C are shutdown.
- Loop Cross Connect Valves, CW-V-4A/4B are closed.

EVENT:

- CW-P-1D Trips.
- Main Condenser vacuum starts to lower.
- CRS directs the CRO to trip the Main Turbine.
- Main Condenser vacuum lowers to 25" Hg and stabilizes.

Based on the conditions, the reactor has __ (1) __ and OTSG pressure control is being controlled by the __ (2) __.

- A. (1) tripped
(2) Turbine Bypass Valves
- B. (1) tripped
(2) Atmospheric Dump Valves
- C. (1) stabilized at ~18% power
(2) Turbine Bypass Valves
- D. (1) stabilized at ~18% power
(2) Atmospheric Dump Valves

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) OTSG pressure control transfers from the Turbine Bypass Valves (TBVs) to the Atmospheric Dump Valves (ADVs) when there are < 2 Circulating Water Pumps operating or when Main Condenser Vacuum is < 23" Hg; (2) In this question, when CW-P-1D trips, this still leaves 2 Circulating Water Pumps operating, and since Main Condenser Vacuum never lowers to less than 23" Hg, the TBVs retain OTSG pressure control; (3) Due to reactor power being below 45%, when the Main Turbine is tripped, the reactor will not automatically trip, but will runback to approximately 18% power.

A.	(1) tripped (2) Turbine Bypass Valves	INCORRECT: Plausible if the examinee does not recognize that at 40% power the Reactor will not trip on a Turbine Trip. Incorrect because the reactor does not trip.
B.	(1) tripped (2) Atmospheric Dump Valves	INCORRECT: Plausible if the examinee does not recognize that at 40% power the Reactor will not trip on a Turbine Trip. Incorrect because the reactor will not trip, and the ADVs will not have OTSG pressure control.
C.	(1) stabilized at ~18% power (2) Turbine Bypass Valves	CORRECT: See above
D.	(1) stabilized at ~18% power (2) Atmospheric Dump Valves	INCORRECT: Plausible because the examinee could believe that the ADVs have OTSG pressure control.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	075	2.4.21
	Importance Rating	4.0	4.6

K/A: Circulating Water System: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question: RO Question # 63

Technical Reference(s): ICS Digital Print D553854 TQ-TM-104-621-C001, Rev 8
TQ-TM-104-641-C001, Rev 2 OP-TM-411-000, Rev 21

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO-10

Question Source: Bank #
Modified Bank #
New X

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 have knowledge relative to the OTSG pressure control logic with respect to operation of the CW Pumps. In-addition requires knowledge related to heat production because of the runback and heat removal.			
High Cog: This question is high cog because the examinee analyzes plant conditions and has to integrate the relationship between the operation of several systems.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

64

ID: 1736357

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Electrical load reducing, with a constant reactor power.
- OTSG 1A steam pressure is 870 psig and decreasing.
- RB pressure is 0.3 psig and steady.
- Total FW flow is 10.2 E6 lbm/hr.
- PLF-1-9 and PLF-1-10 for Intermediate Bldg Fire AND Intermediate Bldg Trouble
- AH-E-73 (Intermediate Bldg.) discharge high temperature alarm.
- PPC Point A0331 TURB DRIV EF-P1 BRG COOL WTR OUT is in HI-2 Alarm
- RM-A-2G reading normal.

What action is required for the above event?

- A. Dispatch personnel to look for the steam leak.
- B. Manually trip the Reactor AND go to OP-TM-EOP-001.
- C. Reduce power to < 45% and manually trip the Main Turbine.
- D. Initiate a Plant Shutdown IAW 1102-4, POWER OPERATION.

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) Based on the symptoms there is steam leak in the Intermediated Building; (2) OP-TM-AOP-051, SECONDARY SIDE HIGH ENERGY LEAK, Section 5, directs if there is a leak in the Intermediate Building with a HI-2 alarm on safety related equipment to trip the Reactor and GO TO OP-TM-EOP-001, REACTOR TRIP; (3) EF-P-1 is safety related equipment.			
A.	Dispatch personnel to look for the steam leak.	INCORRECT: Plausible if the examinee is not sure of leak location. The procedure does address looking for the leak when it is in the Reactor Building.	
B.	Manually trip the Reactor AND go to OP-TM-EOP-001	CORRECT: See above.	
C.	Reduce power to < 45% and manually trip the Main Turbine	INCORRECT: Plausible if the examinee believes that since the steam leak is in the Intermediate Building they could initiate a plant shutdown and when below 45%, then the turbine could be tripped without an automatic reactor trip.	
D.	Initiate a Plant Shutdown IAW 1102-4, POWER OPERATION.	INCORRECT: Plausible since this step appears several times in OP-TM-AOP-051, however the HI-2 Alarm would require tripping the Reactor.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	040	AA2.02
	Importance Rating	4.6	4.7
K/A: Steam Line Rupture: Ability to determine and interpret the following as they apply to the Steam Line Rupture: Conditions requiring a reactor trip			
Proposed Question: RO Question #64			
Technical Reference(s): OP-TM-AOP-051, Rev 2			
Proposed References to be provided to applicants during examination: None			
Learning Objective: AOP-051-PCO-5			
Question Source: Bank #			
	Modified Bank # 462996		
	New		
Question History: N/A Last NRC Exam: N/A			
Question Cognitive Level: Memory or Fundamental Knowledge			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

	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 that the reactor must be tripped with a steam line rupture in the intermediate building if there is a HI-2 alarm on safety related equipment.		
High Cog: This question require analysis of plant conditions and based on these conditions, the priority of required actions.		
The plant has been operating at full power when the following indications are received:		
<ul style="list-style-type: none">• Electrical load reducing, with a constant reactor power.• OTSG 1A steam pressure is 870 psig and decreasing.• RB pressure is 0.3 psig and steady.• Total FW flow is 10.2 E6 lbm/hr.• AH-E-73 (Intermediate Bldg.) discharge high temperature alarm.• PPC Point A0331 TURB DRIV EF-P1 BRG COOL WTR OUT is in alarm and currently at 123 F and rising at 2 degrees every minute• AH-E-24A (EFW pump area) trips on high discharge temperature.• RM-A-2G reading normal.		
What action is required for the above event?		
A. Manually trip the Reactor AND go to OP-TM-EOP-001.		
B. Reduce power to < 45% and manually trip the Main Turbine.		
C. Reduce power at a rate specified by the CRS until an RPS trip setpoint is reached.		
D. Verify no personnel are in the Intermediate Building AND continue Power Operations.		
Answer:	A	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

65

ID: 1718268

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- Building Spray Pump, BS-P-1B, is out of service.

SEQUENCE OF EVENTS:

1. LOCA
2. RCS pressure lowers to 890 psig.
3. Reactor Building Pressure rises to 32 psig.
4. 1S 480V ES Bus trips.

Post Accident Reactor Building cooling is ____ (1) ____, because ____ (2) ____.

- A. (1) adequate
(2) one Building Spray Pump and one Reactor Building Emergency Cooling Fan (AH-E-1) are operating
- B. (1) adequate
(2) one Building Spray Pump and two Reactor Building Emergency Cooling Fans (AH-E-1) are operating
- C. (1) NOT adequate
(2) only one Building Spray Pump and one Reactor Building Emergency Cooling Fan (AH-E-1) are operating
- D. (1) NOT adequate
(2) only one Building Spray Pump and two Reactor Building Emergency Cooling Fans (AH-E-1) are operating

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) The post-accident reactor building emergency cooling is adequate with one Building Spray Pump and one Reactor Building Emergency Cooling Fan; (2) In this question, the examinee is starting with one Building Spray Pump out of service, the other Building Spray pump remains operable; (3) When the ES occurs (indicated by RCS pressure lowering to 890 psig), the transfer of 1C ES Valves 480V bus is inhibited and will remain powered from 1S 480V ES Bus; (4) When 1S 480V ES bus trips, AH-E-1B and AH-E-1C lose power, thus leaving only AH-E-1A operable.			
A.	(1) adequate (2) one Building Spray Pump and one Reactor Building Emergency Cooling Fan (AH-E-1) is operating	CORRECT: Correct Answer	
B.	(1) adequate (2) one Building Spray Pump and two Reactor Building Emergency Cooling Fans (AH-E-1) are operating	INCORRECT: Plausible because the examinee could believe that 1C ES Valves MCC swapped to 1P 480V ES Bus, which would have powered AH-E-1C. Incorrect because the ES blocked 1C ES Valves from swapping to 1P 480V ES bus, thus leaving AH-E-1C without power.	
C.	(1) NOT adequate (2) only one Building Spray Pump and one Reactor Building Emergency Cooling Fan (AH-E-1) is operating	INCORRECT: Plausible because the examinee could believe that having only one Building Spray pump and one RB Emergency Cooling fan operating is inadequate post accident cooling. Incorrect because having one of each running is adequate.	
D.	(1) NOT adequate (2) only one Building Spray Pump and two Reactor Building Emergency Cooling Fans (AH-E-1) are operating	INCORRECT: Plausible because the examinee could believe that having only one Building Spray pump and two RB Emergency Cooling fans is inadequate post accident cooling. Incorrect because only one of each running is adequate.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier # 2	2	
	Group # 1	1	
	K/A #	026	K3.01
	Importance Rating	3.9	4.1
K/A: Containment Spray System: Knowledge of the effect that a loss or malfunction of the CSS will have on the following: CCS			
Proposed Question:		RO Question #65	
Technical Reference(s):		TQ-TM-104-642-C001, Rev 7	
Proposed References to be provided to applicants during examination:			None
Learning Objective:		642-GLO-10	
Question Source:		Bank #	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Modified Bank #	
New	X
Question History:	N/A
Last NRC Exam:	NO
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 loss of one Building Spray Pump and two RB Emergency Cooling fans will have on the containment cooling system.	
High Cog: This question is High Cog because the examinee must analyze the stem and recognize that an ES signal has occurred and how the loss of power has affected the Building Spray and RB Emergency Cooling Systems.	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

66

ID: 1736429

Points: 1.00

Plant Conditions:

- 90% power with ICS in auto.

EVENT:

- PLB-1-7 INSTRUMENT AIR PRESS LOW TURBINE AREA comes in.
- PI-1403 Secondary IA pressure reads 59 psig.
- PI-222 Primary Instrument Air pressure reads 90 psig.

Based on the above plant conditions, the initial action the operating crew must perform is:

- A. Trip the Reactor.
- B. Initiate a plant shutdown.
- C. Dispatch a NLO to isolate Primary IA from Secondary IA.
- D. Start Instrument Air Compressors, IA-P-1A and IA-P-1B from the Control Room.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1)OP-TM-AOP-028, LOSS OF INSTRUMENT AIR is entered when pressure on PI-1403 < 80 psig; (2) When pressure on PI-1403 goes less than 60 psig, the next steps are to dispatch operators to block open MU-V-20 and IC-V-3 and 4, then trip the reactor. (3) Of the actions listed, the operating crew would trip the reactor.

A.	Trip the Reactor.	INCORRECT: See above
B.	Initiate a plant shutdown.	INCORRECT: Plausible if the examinee believes that due to the loss of instrument air a Technical Specification LCO must be entered, and a plant shutdown must occur. Incorrect the reactor must be tripped when PI-1403 reads less than 60 psig.Plausible since this would be done if Primary IA>80- and Secondary < 60 AND the Reactor was not tripped per procedure requirements. Then the procedure directs to restore Secondary Pressure.
C.	Dispatch a NLO to isolate Primary IA from Secondary IA.	INCORRECT: INCORRECT: Plausible since this could raise instrument air pressure. Incorrect because the operating crew must tripped when PI-1403 < 60 psig.
D.	Start Instrument Air Compressors, IA-P-1A and IA-P-1B from the Control Room.	INCORRECT: Plausible because this is an action in OP-TM-AOP-028 to take if IA-PI-491 < 85 psig. Incorrect because the reactor must be tripped when PI-1403 is less than 60 psig.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	065	AA2.06
	Importance Rating	3.6	4.2

K/A: Loss of Instrument Air: Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is de-creasing

Proposed Question: RO Question # 66

Technical Reference(s): OP-TM-AOP-028, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-PCO-5

Question Source: Bank #

Modified Bank #

New X

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 know when the reactor must be tripped on a loss of instrument air.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

67

ID: 1736434

Points: 1.00

Plant Conditions:

- The Plant has been shutdown for 24 days.
- Reactor Vessel level is being lowered in support of internal Reactor Vessel work.
- The "B" Train of Decay Heat Removal (DHR) is operating with a flow rate of 1,400 gpm.

EVENT:

- PLB-5-5 1B DECAY HEAT REMOVAL COMPARTMENT LEAK DETECT comes in.
- Decay Heat Pump, DH-P-1B, suction line temperature indicator has risen 10.5°F.
- RCS Cold Leg level is 15 inches above centerline and slowly lowering.
- DH-P-1B motor amperage and flow is significantly oscillating.

Based on the above conditions, which ONE of the following actions must be taken FIRST?

- A. Start "A" Train of Decay Heat Removal.
- B. Lower flowrate from DH-P-1B to less than 1,200 gpm.
- C. Immediately place DH-P-1B in the "Pull-to-Lock" position.
- D. Raise RCS level until Decay Heat removal Pump operating indications are normal.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) The OP-TM-EOP-030, LOSS OF DECAY HEAT REMOVAL, entry criteria are met: Incore temperature increases > 10F due to an unplanned condition; (2) In addition, since flow and motor amperage are oscillating, in accordance with Step 2.1, DH-P-1B must be placed in PTL.

A.	Start "A" Train of Decay Heat Removal.	INCORRECT: Plausible since the follow up actions direct this action, however the priority is to place DH-P-1B in PTL. In-addition, starting the alternate pump could cause a common mode failure with lowering level.
B.	Lower flowrate from DH-P-1B to less than 1,200 gpm.	INCORRECT: Plausible since lowering flowrate from DH-P-1B to less than 1,200 gpm would place parameters into a more acceptable region. This flow rate would still not be allowed.
C.	Immediately place DH-P-1B in the "pull-to-lock" position.	CORRECT: See above
D.	Raise RCS level until Decay Heat removal Pump operating indications are normal.	INCORRECT: Plausible since Raising RCS level until Decay Heat removal Pump operating indications are normal would maintain level in the acceptable region and possibly reduce oscillating pump parameters, but not allowed under these conditions, since EOP-030 requires stopping the pump given the initial conditions.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	025	AA2.01
	Importance Rating	2.7	2.9

K/A: Loss of Decay Heat Removal: Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Proper amperage of running LPI/decay heat removal/RHR pump(s)

Proposed Question: RO Question # 67

Technical Reference(s): OP-TM-EOP-030, Rev 9
OP-TM-212-000, Rev 23

Proposed References to be provided to applicants during examination:

OP-TM-212-000, Attachment
7.2 Min Height of Water to
Avoid Vortex Formation vs
DH Flow

Learning Objective: EOP-030-PCO-4

Question Source: Bank #

Modified Bank # 304375

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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

55.43

Comments:

KA MATCH: This question matches the KA because the examinee must interpret the oscillating running amperage of a DHR pump to determine that it is cavitating.

High Cog: The examinee must analyze the stem to determine the entry criteria for OP-TM-EOP-030 are met, and then determine that DH-P-1A must be placed in PTL. In addition, the examinee can check the answer to determine that the pump is below the vortex limit.

Plant Conditions:

- The Plant has been Shutdown for 24 days.
- Reactor Vessel level is being lowered in support of internal Reactor Vessel work.
- The "B" Train of Decay Heat Removal (DHR) is operating with a flow rate of 1,400 gpm.
- The "A" Train of DHR is out-of-service for minor maintenance and is expected to be returned to service within the next 4 hours.

Event:

- Decay Heat Pump, DH-P-1B, suction line temperature indicator has risen 10.5°F.
- RCS Cold Leg level is 10.5 inches above centerline.
- DH-P-1B Motor Amperage is significantly oscillating.

Based on the above conditions, which ONE of the following actions will be taken FIRST?

- A. Lower flowrate from DH-P-1B to less than 1,200 gpm.
- B. Immediately place DH-P-1B in the "pull-to-lock" position.
- C. Raise RCS level to at least 14 inches above Hot Leg centerline.
- D. Actuate the RB Evacuation alarm and evacuate ALL personnel from the RB.

Answer: B

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

68

ID: 1736443

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- MU-P-1A (Make-up Pump) is supplying normal make-up and Seal Injection.
- MU-P-1B is in Normal-After Stop.

SEQUENCE OF EVENTS:

- Loss of Coolant Accident.
- RCS pressure lowers to 1500 psig.
- RB pressure is 5 psig.
- Loss of Offsite Power.
- MU-P-1A Trips.

Based on the above conditions, if MU-P-1B is ES Selected using the 43 Selector Switch on the 1E 4160V Bus, MU-P-1B would __ (1) __ and MU-P-1C would __ (2) __.

- A. (1) start and continue to run, (2) trip.
- B. (1) start and continue to run, (2) continue to run.
- C. (1) start and immediately trip, (2) trip.
- D. (1) start and immediately trip, (2) continue to run.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) The LOCA occurs which causes an ES Actuation, both 1600 psig and 4 psig, this sends a start signal to MU-P-1A and MU-P-1C; (2) These are Block 1 ES loads, when the LOOP occurs, MU-P-1A and MU-P-1C stop and restart when the Emergency Diesel Generators start and load on the 1D and 1E 4160V busses; (3) When the 43 selector switch on the 1E 4160V Bus is rotated to ES select MU-P-1B, then MU-P-1B will start and MU-P-1C will trip.			
A.	(1) start and continue to run, (2) trip.	CORRECT: See above.	
B.	(1) start and immediately trip, (2) trip.	INCORRECT: Plausible if the examinee thinks that since MU-P-1B is green flagged that anti-pump would actuate when the pump is started and trip the pump. Incorrect because anti-pump would only actuate if a simultaneous trip and start signal.	
C.	(1) start and continue to run, (2) continue to run.	INCORRECT: Part 1 is correct. Part 2 is plausible if the examinee does not understand the operation of the 43 selector switch and its effect on the pump trip circuit. This interlock exists to prohibit from running two Makeup Pumps on a Emergency Diesel Generator.	
D.	(1) start and immediately trip, (2) continue to run.	INCORRECT: Plausible if the examinee thinks that since MU-P-1B is green flagged that anti-pump would actuate when the pump is started and trip the pump. Incorrect because anti-pump would only actuate if a simultaneous trip and start signal. Part 2 is plausible if the examinee does not understand the operation of the 43 selector switch and its effluent on the pump trip circuit.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	006	K2.01
	Importance Rating	3.6	3.9
K/A: Emergency Core Cooling System: Knowledge of bus power supplies to the following: ECCS Pumps			
Proposed Question: RO Question # 68			
Technical Reference(s): 1105-3, Rev 53			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 211-GLO-10			
Question Source: Bank #			
Modified Bank #			
New X			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 the power supply to the Makeup Pumps and knowledge of the interlock preventing from running two Makeup Pumps on an Emergency Diesel Generator.			
High Cog: This question is High Cog because the examinee must analyze the plant conditions to determine the status of MU-P-1B after a LOOP and ES actuation.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

69

ID: 1736454

Points: 1.00

Plant Conditions:

- 50% power with ICS in auto.

EVENT:

- MAP G-2-1 CRD PATTERN ASYMMETRIC alarms.
- Red Fault Indication lit on the Position Indication Panel (PIP).
- The Asymmetric Control Rod Fault indication is currently lit on the Diamond Control Panel.

These indications would be caused by a deviation of an individual rod position of ___(1)___ from the group average, which ___(2)___, and would require entry into OP-TM-AOP-062, INOPERABLE ROD.

- A. (1) 7 inches
(2) excludes the faulted rod
- B. (1) 7 inches
(2) includes the faulted rod
- C. (1) 9 inches
(2) excludes the faulted rod
- D. (1) 9 inches
(2) includes the faulted rod

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) Each rod in a group is compared with the group average; (2) If a rod in a group deviates by more than 7 inches (5%) from the group average, an individual amber fault light for that rod will light on the PI panel; (3) A rod that is 9 inches (6.5%) from its group average, the faulted rod is removed from the group average calculation; (4) An individual red fault light for that rod will light on the PIP and the asymmetric fault light on the Operator Diamond Control Panel will be lit; (5) Both the 7" and 9" Asymmetric fault initiate alarm MAP G-2-1. (6) The 7" Asymmetric Rod does NOT require entry into AOP-062.			
A.	(1) 7 inches (2) excluding the Faulted Rod.	INCORRECT: Plausible since this initiates the G-2-1 Alarm. Incorrect because the examinee may think that the 7" also excludes the faulted rod as does the 9' fault. In addition, a 7" rod does not require entry into OP-TM-AOP-062.	
B.	(1) 7 inches (2) including the Faulted Rod.	INCORRECT: Plausible since this initiates the G-2-1 Alarm and the 7" fault does include the faulted rod. Incorrect because a 7" misaligned rod does not require entry into OP-TM-AOP-062.	
C.	(1) 9 inches (2) excluding the Faulted Rod.	CORRECT: See above	
D.	(1) 9 inches (2) including the Faulted Rod.	INCORRECT" Plausible since the 9" fault would cause all the above indications, requires entry into OP-TM-AOP-062, however does not include the faulted rod in the calculation.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	014	2.4.4
	Importance Rating	4.5	4.7
K/A: Rod Position Indication: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.			
Proposed Question: RO Question # 69			
Technical Reference(s): OP-TM-AOP-062, Rev 007 TQ-TM-104-622-C001, Rev 007 OP-TM-622-000, Rev 007			
Proposed References to be provided to applicants during examination: None			
Learning Objective: 622-GLO-10			
Question Source: Bank # Modified Bank # 782160 New			
Question History: System Exam 14 Last NRC Exam: N/A			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 recognize the entry conditions to OP-TM-AOP-062 based on the misaligned rod indications.		
High Cog: This question is high cog because the examinee must analyze the indications in the stem to determine if an AOP must be entered.		
The Asymmetric control rod fault indication is currently lit on the Diamond control station.		
Select the ONE choice below that correctly describes the indications that constitute an asymmetric control rod fault indication on the Diamond control station.		
A.	Deviation of an individual rod's position indication of 9 inches, (6.5%), from the average position of the remaining 60 full length control rods.	
B.	Deviation of an individual rod's position indication of 7 inches, (5%), from the average position indication of all of the control rods in that group. Faulted rod remains in the group average.	
C.	Deviation of an individual rod's position indication of 9 inches, (6.5%), from the average position indication of the other control rods in that group. Faulted rod remains in the group average.	
D.	Deviation of an individual rod's position indication of 9 inches, (6.5%), from the average position indication of all of the control rods in that group. Faulted rod is removed from the group average.	
Answer:	D	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

70

ID: 1736470

Points: 1.00

What is minimum personnel that must be in the Control Room in accordance with OP-TM-112-101-1002, SHIFT STAFFING REQUIREMENTS, when the RCS temperature is greater than 200F?

- A. One SRO or one RO
- B. One SRO AND one RO
- C. One SRO AND two ROs
- D. One SRO OR one RO AND one STA

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) OP-TM-112-101-1002 requires that a minimum of 1 SRO and 1 RO be in the Control Room at all times when RCS temperature is > 200F.			
A.	One SRO or one RO	INCORRECT: Plausible since if RCS temperature were less than 200F this combination would be acceptable.	
B.	One SRO AND one RO	CORRECT: See Above	
C.	One SRO AND two ROs	INCORRECT: Plausible if the examinee interprets the table in step 4.1 to mean that the two RO's must be in the control room at all times. Incorrect because the table is referring to minimum shift staffing.	
D.	One SRO OR one RO AND one STA	INCORRECT: Plausible if the examinee interprets the table in step 4.1 to mean that the one RO and the STA must be in the control room at all times. Incorrect because the table is referring to minimum shift staffing.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.1	
	Importance Rating	3.8	4.2

K/A: Knowledge of conduct of operations requirements.

Proposed Question: RO Question #70

Technical Reference(s): OP-TM-112-101-1002, Rev 10 Tech Spec 6.2.2.2

Proposed References to be provided to applicants during examination: None

Learning Objective: Prewatch DBIG - APCO-1

Question Source: Bank # 372961
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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

55.43

Comments: (KA Match, why high cog, why SRO only)

KA MATCH: This question matches the KA because the examinee must know the minimum shift staffing in the control room.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

71

ID: 1736534

Points: 1.00

In accordance with Tech Specs, the Minimum Conditions for Criticality (Excluding Low Power Physics Testing) requires the RCS Temperature to be __ (1) __ and the Safety Rod Groups to be __ (2) __ prior to any other reduction in shutdown margin.

- A. (1) \geq 532F (2) fully withdrawn
- B. (1) \geq 532F (2) fully inserted
- C. (1) >525F (2) fully withdrawn
- D. (1) >525F (2) fully inserted

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) Tech Spec 3.1.3- The reactor coolant temperature shall be above 525°F except for portions of low power physics testing. (2) Safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deformation or regulating rod withdrawal during the approach to criticality			
A.	(1) ≥ 532F (2) fully withdrawn	INCORRECT: Plausible since the calculation for Estimated Critical Rod Position assumes RCS Temperature is 532F.	
B.	(1) ≥ 532F (2) fully inserted	INCORRECT: Plausible since the calculation for Estimated Critical Rod Position assumes RCS Temperature is 532F and if the examinee thinks the Safety Rods must be inserted to maintain the Tech Spec required 1% SDM prior to withdrawing rods with the intent to go critical.	
C.	(1) >525F (2) fully withdrawn.	CORRECT: See above	
D.	(1) >525F (2) fully inserted	INCORRECT: Plausible if the examinee thinks the Safety Rods must be inserted to maintain the Tech Spec required 1% SDM prior to withdrawing rods with the intent to go critical.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.38	
	Importance Rating	3.6	4.5

K/A: Equipment Control: Knowledge of conditions and limitations in the facility license.

Proposed Question: RO Question # 71

Technical Reference(s): 1103-8, Rev 55 T.S. 3.1.3, AMD 278

Proposed References to be provided to applicants during examination: None

Learning Objective: Normal Ops - 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 X
Comprehension or Analysis

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

10 CFR Part 55 Content: 55.41

55.43

Comments: (KA Match, why high cog, why SRO only)

KA Match: This question matches the KA because the examinee must have knowledge of the requirements to pull to criticality.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

72

ID: 1737637

Points: 1.00

A surveillance for an ES pump is in progress.

The analog suction pressure gauge, normally isolated except for this surveillance, is broken.

The suction pressure is used to calculate pump head, which is a required parameter for the surveillance test.

Another suitable analog gauge measuring pump suction pressure is available.

Based on the broken gauge, which one of the following identifies the required actions with respect to this surveillance?

- A. Continue performing the surveillance using the alternate means.
- B. Continue performing the surveillance and annotate in the procedure that the alternate means was used.
- C. Stop the surveillance. The surveillance may not be run until the gauge is fixed.
- D. Stop the surveillance and change the surveillance to allow the alternate means of measuring pump suction pressure.

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) ER-TM-321-1041, Attachment 1 addresses instrument requirements for an In-Service Test (IST); (2) If an instrument fails during and IST the correct actions are to : Stop the test, place the system in a safe configuration, locate a backup instrument or replace the existing instrument using Section 2 of the attachment, and resume or re-perform the test with the replaced instrument.			
A.	Continue performing the surveillance using the alternate instrument.	INCORRECT: Plausible because there is another suitable pressure gauge available. Incorrect because the operator must stop the surveillance prior to using the instrument.	
B.	Continue performing the surveillance and annotate in the procedure that the alternate instrument was used.	INCORRECT: Plausible because there is another suitable pressure gauge available. Incorrect because the operator must stop the surveillance prior to using the instrument.	
C.	Stop the surveillance. The surveillance may not be run until the gauge is fixed.	INCORRECT: Plausible because the examinee may believe that since the test involves an ES pump that all the instrumentation for the test must be functioning correctly.	
D.	Stop the surveillance and change the surveillance to allow the alternate means of measuring pump suction pressure.	CORRECT ANSWER: See above.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.12	
	Importance Rating	3.7	4.1
K/A: Knowledge of surveillance procedures.			
Proposed Question: Question 72			
Technical Reference(s): ER-TM-321-1041, Rev 7			
Proposed References to be provided to applicants during examination: None			
Learning Objective: EOP DBIG, APCO-2			
Question Source: Bank # 504102			
Modified Bank #			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

New

Question History: None

Last NRC Exam: None

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 IST program requirements at TMI.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

73

ID: 1736792

Points: 1.00

Plant Conditions:

- OP-TM-EOP-005, OTSG TUBE LEAKAGE has been entered for a 'A' OTSG tube rupture.
- Reactor is tripped.

'A' OTSG must be isolated when ____ (1) ____ AND ____ (2) ____.

- A. (1) RCS pressure is less than 1000 psig,
(2) BWST level < 22 feet
- B. (1) OTSG pressure is less than 1000 psig,
(2) BWST level < 22 feet
- C. (1) RCS pressure is less than 1000 psig,
(2) Offsite integrated dose approaches 500 mRem CDE (Thyroid)
- D. (1) OTSG Pressure is less than 1000 psig,
(2) Offsite integrated dose approaches 500 mRem CDE (Thyroid)

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-EOP-005, the isolation criteria is as follows: RCS pressure <1000 psig and OTSG Level > 85%, RCS pressure < 1000 psig and offsite dose projections approaches 500 mRem TEDE or 1500 mRem CDE Thyroid, RCS pressure < 1000 psig and BWST Level < 22 ft; (2) An OTSG is not isolated with RCS pressure above 1000 psig to preclude possibly lifting the MSSV.			
A.	(1) RCS pressure is less than 1000 psig, (2) BWST level < 22 feet	CORRECT: See above	
B.	(1) OTSG pressure is less than 1000 psig, (2) BWST level < 22 feet	INCORRECT: Plausible since OTSG pressure < 1000 psig would be below all setpoints for OSTG steam safety valves. Incorrect because OP-TM-EOP-005 directs isolation only after RCS pressure has lowered to less than 1000 psig.	
C.	(1) RCS pressure is less than 1000 psig, (2) Offsite integrated dose approaches 500 mRem CDE (Thyroid)	INCORRECT: Plausible since RCS Pressure < 1000 psig is correct and if the examinee believes that the limit for CDE is 500 mRem. Incorrect because the limit is 5000 mRem for TEDE.	
D.	(1) OTSG Pressure is less than 1000 psig, (2) Offsite integrated dose approaches 500 mRem CDE (Thyroid).	INCORRECT: Plausible if the examinee believes that the limit for CDE is 500 mRem. Incorrect because the limit is 5000 mRem for TEDE.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	038	EA2.01
	Importance Rating	4.1	4.7
K/A: Steam Generator Tube Rupture: Ability to determine or interpret the following as they apply to a SGTR: When to isolate one or more S/Gs			
Proposed Question: RO Question # 73			
Technical Reference(s): OP-TM-EOP-005, Rev 9 OP-TM-EOP-0051, Rev 6			
Proposed References to be provided to applicants during examination: None			
Learning Objective: EOP-005-PCO-4			
Question Source: Bank #			
Modified Bank #			
New X			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 when to isolate an OTSG based on RCS pressure and BWST level.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

74

ID: 1736795

Points: 1.00

Which ONE of the following describes the PREFERRED method of supplementing the passive mode of Post LOCA long term core circulation in the prevention of boron concentration effects 36 hours after the event?

- A. DH-P-1A suction is lined up to the RB Sump and then discharges to the Auxiliary Spray Line.
- B. DH-P-1B suction is lined up to the RB Sump and then discharges to the Auxiliary Spray Line.
- C. DH-P-1A suction is lined up to the RB Sump AND takes partial suction from DH drop line and then discharges through RV CF Nozzles.
- D. DH-P-1B suction is lined up to the RB Sump AND takes partial suction from DH drop line and then discharges through RV CF Nozzles.

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) The preferred method is to drain concentrated reactor coolant through the decay heat drop line and into the suction header for DH-P-1B which is lined up to the Reactor Building Sump. If this method is initiated within 24 hours of a large break LOCA, it alone will prevent boron precipitation without passive mixing; (2) A second active method is to inject Reactor Building Sump water into the RCS through the pressurizer auxiliary spray line. Flow from this method in combination with a small portion of the flow from the passive method will prevent boron precipitation. About 5 days after a LOCA, the auxiliary spray line method alone will prevent boron precipitation.

A.	DH-P-1A suction is lined up to the RB Sump and then discharges to the Auxiliary Spray Line.	INCORRECT: Plausible since this is a flowpath described in the procedure, however it is not effective until 5 days after the event.
B.	DH-P-1B suction is lined up to the RB Sump and then discharges to the Auxiliary Spray Line.	INCORRECT: Plausible since this is a flowpath described in the procedure, however it is not effective until 5 days after the event and in addition DH-P-1B requires verification of DH Pump discharge cross ties are open.
C.	DH-P-1A suction is lined up to the RB Sump AND takes partial suction from DH drop line and then discharges through RV CF Nozzles.	INCORRECT: Plausible if the examinee does not understand that the procedure directs using DH-P-1B since the Drop Line is lined up to DH-P-1B.
D.	DH-P-1B is lined to the RB Sump AND takes partial suction from DH drop line and then discharges through RV CF Nozzles.	CORRECT: See above

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.9	
	Importance Rating	3.8	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: RO Question # 74

Technical Reference(s): OP-TM-212-911, Rev 3 TQ-TM-104-212-C001, Rev 17

Proposed References to be provided to applicants during examination: None

Learning Objective: 212-GLO-10

Question Source: Bank #

Modified Bank # 356397

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

New

Question History: System 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.10
55.43

Comments:

KA Match: This question matches the KA because the examinee must know the mitigation strategy for a LOCA to prevent boron precipitation.

Which ONE of the following describes the preferred method of supplementing the passive mode of Post LOCA long term core circulation in the prevention of boron concentration effects 36 hours after the event?

DH-P-1A/1B takes a suction from RB Sump; _____.

- A. DH-P-1A/1B discharges through RV CF Nozzles
- B. DH-P-1A takes a partial suction from DH drop line; DH-P-1A provides Auxiliary Spray Flow
- C. DH-P-1A/1B provides MU pump suctions; MU Pumps inject water through RCS HPI Nozzles
- D. DH-P-1B takes partial suction from DH drop line; DH-P-1B discharges through RV CF Nozzles

Answer: D

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

75

ID: 1736802

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

Which one of the following identifies the status of the Reactor Building Hydrogen Monitor Recorders?

They are both __ (1) __ and they are located on Control Room __ (2) __.

- A. (1) ON (2) the H&V Panel.
- B. (1) ON (2) Panel Left.
- C. (1) OFF (2) the H&V Panel.
- D. (1) OFF (2) Panel Left.

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1105-18, CONTAINMENT HYDROGEN MONITOR, the H2 Monitors are normally maintained in the standby condition (isolated from containment but with the electronics kept warm), the control room chart recorder drive will remain off and will be turned on when the monitor is placed in service. (2) During normal plant operations, the Hydrogen Monitor System will be warmed up and maintained in a Standby mode. (3) Containment Hydrogen Monitoring are placed into service when required per procedure. (4) Recorder are Located on Control room Panel Left

A.	(1) ON (2) H&V Panel	INCORRECT: Plausible because the the Hydrogen Monitors are maintained in Standby and the examinee may think the Recorder are ON. H&V Panel has many Ventilation Controls and other monitors but not the H2 Monitor Controls.
B.	(1) ON (2) Panel Left.	INCORRECT: Plausible because the the Hydrogen Monitors are maintained in Standby and the examinee may think the Recorder are ON. Correct Location
C.	(1) OFF (2) H&V Panel	INCORRECT: Plausible since they are OFF but wrong location.
D.	(1) OFF (2) Panel Left	CORRECT: See above

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	028	A4.03
	Importance Rating	3.1	3.3

K/A: Hydrogen Recombiner and Purge Control System: Ability to manually operate and/or monitor in the control room: Location and operation of hydrogen sampling and analysis of containment atmosphere, including alarms and indications.

Proposed Question: RO Question # 75

Technical Reference(s): 1105-18, Rev 14

Proposed References to be provided to applicants during examination: None

Learning Objective: 240-GLO-10

Question Source: Bank #

Modified Bank #

New X

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

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	b.7
	55.43	
Comments:	<p>KA Match: This question matches the KA because the examinee must know the location of and operation of the Hydrogen Monitors and their indications. The Hydrogen Monitors are monitored in the Control Room by recorders, the examinee should know the status of the H2 Monitors including that their recorders are NOT on during normal operation. Operator action in the Control Room and in the plant is required to place the H2 Monitors in operation.</p>	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

76

ID: 1737249

Points: 1.00

Plant Conditions:

- Reactor has tripped on RCS low pressure.
- Pressurizer level is 370 inches and slowly increasing.
- RCS pressure is 1500 psig and slowly lowering.
- Core exit temperature is 556 degrees F and stable.
- Reactor Building sump level is 38 inches and increasing.

(1) What is the location of the RCS Leak?

(2) Which of the following combinations of procedures provide the correct guidance to cooldown the plant?

- A. (1) Leak in the RCS Cold Leg
(2) Enter OP-TM-EOP-002, LOSS OF 25F SUBCOOLING MARGIN, and use OP-TM-EOP-006, LOCA COOLDOWN to cooldown the plant.
- B. (1) Leak in the RCS Cold Leg
(2) Enter OP-TM-AOP-050, REACTOR COOLANT LEAKAGE and use 1102-11, PLANT COOLDOWN to cooldown the plant.
- C. (1) Leak in the Pressurizer Steam Space
(2) Enter OP-TM-AOP-043, LOSS OF PRESSURIZER, and use 1102-11, PLANT COOLDOWN to cooldown the plant.
- D. (1) Leak in the Pressurizer Steam Space
(2) Enter OP-TM-AOP-050, REACTOR COOLANT LEAKAGE and use 1102-11, PLANT COOLDOWN to cooldown the plant.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<p>Explanation: To answer this question correctly, the examinee must know: (1) To answer this question correctly, the examinee must determine that a leak in the Pressurizer Steam Space exists; (2) In the stem, it is clear that RCS leakage exists, but to differentiate this Pressurizer Steam Space leak from an leak in the RCS cold leg the examinee will have to deduce that RCS pressure is LOW for the given Pressurizer level and RCS Temperature; (3) If this were a RCS cold leg leak, for a pressure this low, pressurizer level would be much lower as well; (4) For the parameters given, the correct procedure the crew must implement is EOP-001, then enter AOP-043 and 1102-11 to cooldown the plant.</p>																							
A.	(1) Leak in the RCS cold leg (2) Enter OP-TM-EOP-002, LOSS OF 25F SUBCOOLING MARGIN, and use OP-TM-EOP-006, LOCA COOLDOWN to cooldown the plant.	INCORRECT: Plausible because the examinee could believe there is a leak other than an pressurizer steam space leak by RCS pressure lowering and RB Sump level rising. In addition, the examinee will have to check if conditions exist for the operating crew to enter OP-TM-EOP-002 for a Loss of SCM (They do not). Within that procedure, if HPI were not adequate to maintain pressurizer level the crew would enter OP-TM-EOP-006 to cooldown the plant.																					
B.	(1) Leak in the RCS cold leg (2) Enter OP-TM-AOP-050, REACTOR COOLANT LEAKAGE and use 1102-11, PLANT COOLDOWN to cooldown the plant.	INCORRECT: Plausible because the examinee could believe there is a leak other than an pressurizer steam space leak by RCS pressure lowering and RB Sump level rising. In addition, if the examinee does not recognize that the symptoms for a Loss of SCM exist, then AOP-050 could be entered. In AOP-050, 1102-11 is used to cooldown the plant.																					
C.	(1) Leak in the Pressurizer steam space. (2) Enter OP-TM-AOP-043, LOSS OF PRESSURIZER, and use 1102-11, PLANT COOLDOWN to cooldown the plant.	CORRECT ANSWER: OP-TM-AOP-043 must be entered, and then perform 1102-11 with guidance from an attachment in AOP-043.																					
D.	(1) Leak in the Pressurizer steam space. (2) Enter OP-TM-AOP-050, REACTOR COOLANT LEAKAGE and use 1102-11, PLANT COOLDOWN to cooldown the plant.	INCORRECT: Plausible because a leak in the Pressurizer Steam Space is correct. Incorrect because OP-TM-AOP-050 does not provide the additional guidance required to cooldown the plant with a leak in the pressurizer steam space. OP-TM-AOP-043 provides additional guidance on the low SCM, and solid plant operations.																					
<table border="1"> <tr> <td>Examination Outline Cross-reference:</td> <td>Level</td> <td>RO</td> <td>SRO</td> </tr> <tr> <td></td> <td>Tier #</td> <td></td> <td>1</td> </tr> <tr> <td></td> <td>Group #</td> <td></td> <td>1</td> </tr> <tr> <td></td> <td>K/A #</td> <td>008</td> <td>AA2.12</td> </tr> <tr> <td></td> <td>Importance Rating</td> <td>3.4</td> <td>3.7</td> </tr> </table>				Examination Outline Cross-reference:	Level	RO	SRO		Tier #		1		Group #		1		K/A #	008	AA2.12		Importance Rating	3.4	3.7
Examination Outline Cross-reference:	Level	RO	SRO																				
	Tier #		1																				
	Group #		1																				
	K/A #	008	AA2.12																				
	Importance Rating	3.4	3.7																				
<p>K/A: Pressurizer Vapor Space Accident: Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: PZR level indicators</p>																							
<p>Proposed Question: Question #76</p>																							
<p>Technical Reference(s): OP-TM-AOP-043, Rev 6</p>																							

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

OP-TM-AOP-0431, Rev 5

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

Learning Objective: AOP-043-PCO-4

Question Source: Bank #
Modified Bank # 462810
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 use the Pressurizer Level Indicator to determine that there is a leak in the Pressurizer Steam Space, vice a leak elsewhere in the RCS. The pressurizer level is rising, while pressure is lowering which is indicative of a pressurizer steam space leak.

High Cog: This question is high cog because the examinee must analyze the conditions in the stem to determine the location of the leak.

SRO ONLY: This question is SRO only because the examinee must assess the plant conditions, and then select and determine the required course of action.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Plant Conditions:

- Reactor has tripped on RCS low pressure.
- Pressurizer level is 240 inches and increasing.
- RCS pressure is constant at 1100 psig.
- Core exit temperature is 556 degrees F and stable.
- Reactor Building sump level is 38 inches and increasing.

From the list below, identify the ONE transient in progress.

- A. Leak in the RCS cold leg.
- B. Leak in the Pressurizer steam space.
- C. Rx vessel head vent valves failed open.
- D. Failure of the Pressurizer level control valve (MU-V-17) to close.

Answer: B

EXAMINATION ANSWER KEY

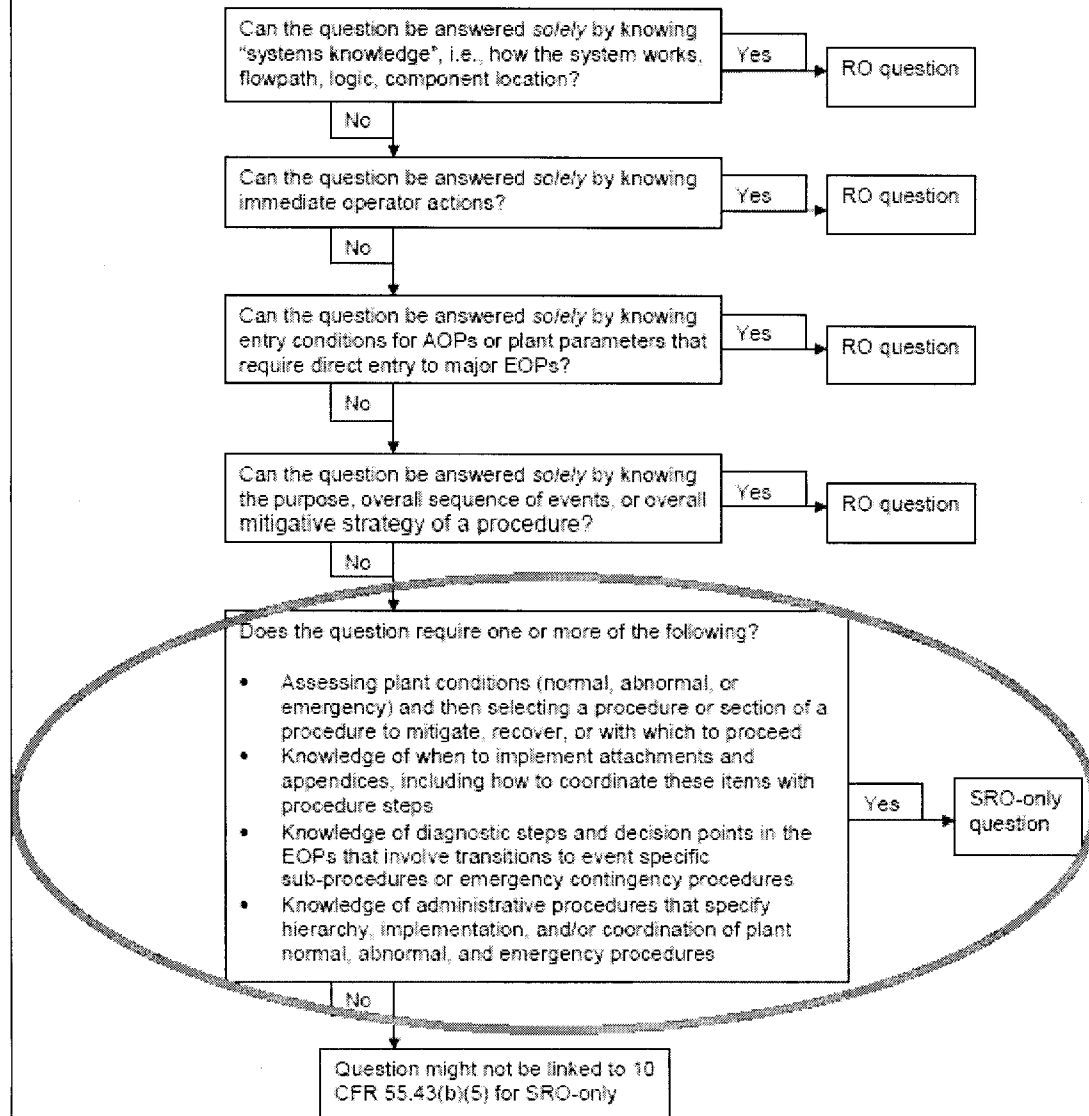
16-01 NRC EXAM SUBMITTAL

ES-401

8

Attachment 2

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

77

ID: 1700250

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- MAP C-3-2, IC SURGE TANK LEVEL HI/LO, alarm in.
- RM-L-9, Intermediate Closed Cooling Water Radiation Monitor, counts are rising.
- IC-T-1, Intermediate Closed Cooling Water Surge Tank, level is off-scale high, and an attempt to bring IC-T-1 back on scale has failed.
- OP-TM-AOP-050, REACTOR COOLANT LEAKAGE is entered.

Current plant parameters:

- Pressurizer level is 220 inches and steady.
- MU Tank level is 86 inches and slowly lowering.
- Reactor Building pressure is 0.2 psig and steady.

Which ONE of the following procedures provides the required actions that mitigate these plant conditions?

- A. OP-TM-AOP-050, REACTOR COOLANT LEAKAGE
- B. OP-TM-MAP-C0302, IC SURGE TANK LEVEL HI/LO
- C. OP-TM-EOP-010, Guide 9, RCS INVENTORY CONTROL
- D. OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) With the indication of the IC-T-1 level high, RM-L-9 rising, the MU Tank level lowering with pressurizer level remaining steady, there is a small break leak between the RCS and ICCW system (either the RCP seals, or Letdown coolers); (2) Because RM-L-9 is rising and IC-T-1 is already off-scale high, the actions to mitigate this event are to trip the reactor, trip all four reactor coolant pumps, close IC-V-2 and 3 and initiate OP-TM-AOP-050 "RCS Leakage", which is found in OP-TM-MAP-C0302.

A.	OP-TM-AOP-050, REACTOR COOLANT LEAKAGE	Plausible because the examinee should determine that and RCS leak is in progress. Actions to mitigate RCS leaks (including into ICCW) are found in OP-TM-AOP-050, but this answer is incorrect because this procedure routes the examinee to OP-TM-MAP-C0302 IC Surge Tank Level Hi/Lo. In addition, in normal procedure hierarchy the mitigation steps of this nature would be found in an AOP or an EOP.
B.	OP-TM-MAP-C0302, IC SURGE TANK LEVEL HI/LO	Correct Answer - The actions to mitigate an RCS to ICCW leak, where IC-T-1 cannot be maintained on scale is step 4.1.3 of this alarm response procedure. The examinee could arrive at this answer by directly entering the alarm response or OP-TM-AOP-050, step 3.5 which directs the examinee to OP-TM-MAP-C0302. Most primary mitigation steps are not found in alarm response procedures.
C.	OP-TM-EOP-010, Guide 9, RCS INVENTORY CONTROL	Plausible because this procedure is directed to be initiated from OP-TM-AOP-050, Reactor Coolant Leakage. This procedure provides guidance for a low pressurizer level or makeup tank level. The examinee could believe that either of those sections (low pressurizer level or makeup tank level) of Guide 9 are applicable at this point, but there are currently no actions from Guide 9 that the examinee would take to mitigate this event. In addition, in normal procedure hierarchy the mitigation steps of this nature would be found in an AOP or an EOP.
D.	OP-TM-AOP-032, LOSS OF INTERMEDIATE COMPONENT COOLING	Plausible because the examinee could believe that the mitigation steps are in OP-TM-AOP-032. This procedure would direct an operating crew to trip the reactor, but will not direct the RCPs to be tripped unless seal injection flow is < 22 gpm. An operating crew may enter OP-TM-AOP-032 at a later point to place IC-P-1A and IC-P-1B in PTL, but these actions would be taken after the actions of OP-TM-MAP-C0302. In addition, in normal procedure hierarchy the mitigation steps of this nature would be found in an AOP or an EOP.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	009	2.4.31
	Importance Rating		4.1

K/A: Small Break LOCA: Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: Question #77

Technical Reference(s): OP-TM-MAP-C0302, Rev 3 OP-TM-EOP-010, Rev 19

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

OP-TM-AOP-050, Rev 6		OP-TM-AOP-032, Rev 4	
Proposed References to be provided to applicants during examination:		None	
Learning Objective:		AOP-050-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	
		55.43	
		b.5	
Comments:			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

K/A MATCH: This matches the K/A because the examinee must know that the alarm response in this question provides the mitigating steps for the RCS leak into the ICCW system.

High Cog: This examinee must determine that a small RCS leak exists, and that the RCS is leaking into the ICCW stem.

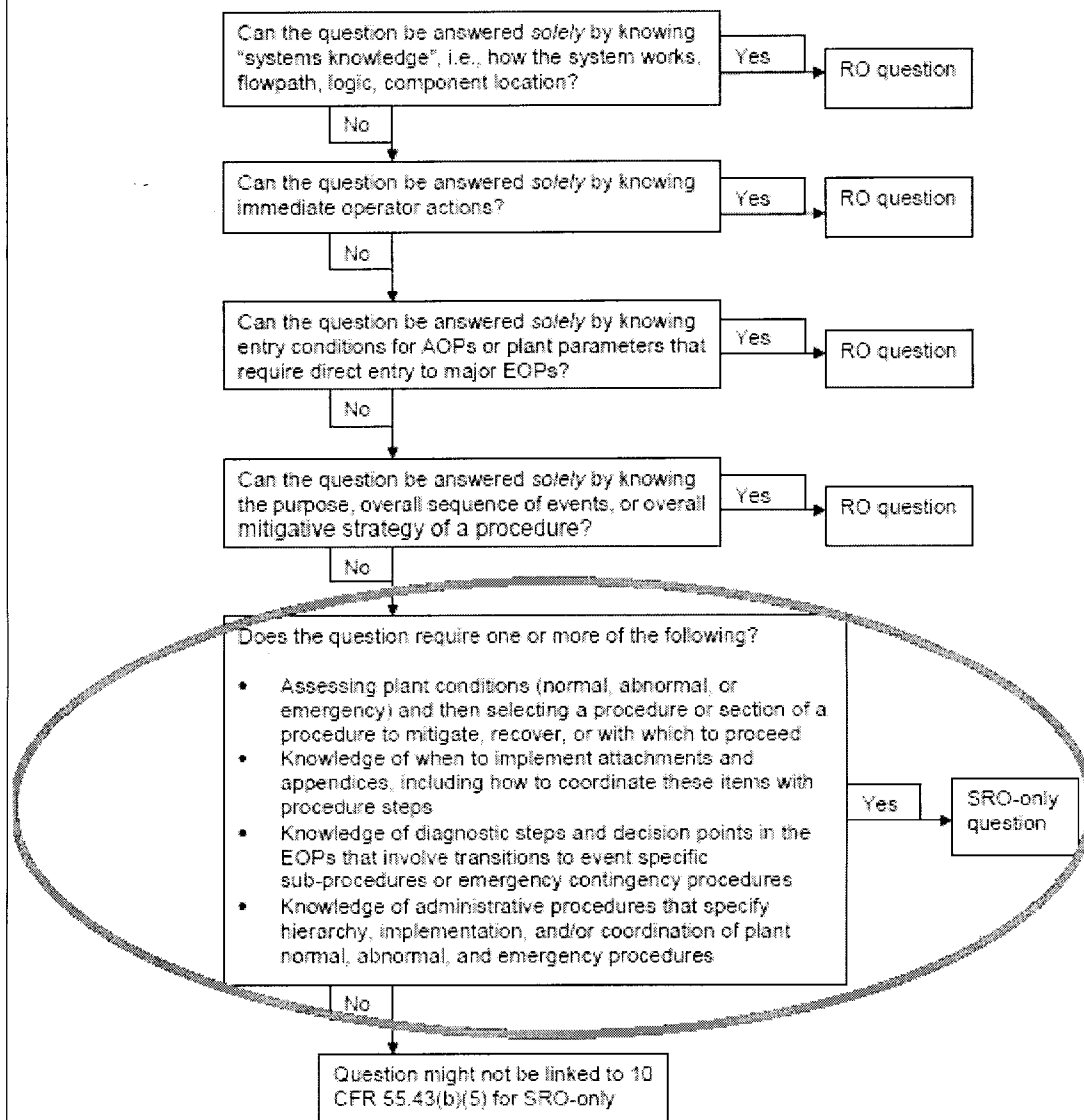
SRO Only: The question requires the examinee to assess plant conditions and to know the content of the listed procedures in order to select a required course of action. These procedures are not major EOPs.

ES-401

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Attachment 2

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)**



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

78

ID: 1685279

Points: 1.00

Plant conditions:

- Reactor is at 100% power
- DC-P-1A is tagged out for a bearing replacement.
- MU-P-1A and MU-P-1C are ES selected and operable.
- The Secondary NLO has just reported the fuel rack for the SBO Diesel was accidentally broken in the **tripped position** by a contract worker climbing over the lever.
- Maintenance says the repair will take 4 hours.

EVENT:

- One-half of the overhead lights go out, and do NOT come back on.
- You observe the following MAPs actuate:
 - B-1-1, 4KV ES FDR BKR TRIP.
 - B-2-1, 4KV ES BUS UV/OV.
 - B-1-2, 4KV ES MOTOR TRIP.
 - F-1-5, RCP SEAL TOT INJECT FLOW HI/LO.

Based on these conditions, which ONE of the following actions must be taken?

- A. Initiate Reactor shutdown within 1 hour.
- B. Swap MU-P-1A Cooling from DC to NS IAW OP-TM-543-439.
- C. Initiate OP-TM-864-901 SBO Diesel Generator (EG-Y-4) Operations.
- D. Ensure Emergency Diesel Generator EG-Y-1B restored to operable status within 7 days.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 5

Comments:

K/A Match: This matches the K/A because the examinee has to determine that both of the decay closed cooling water pumps are inoperable, hence Decay Closed Cooling is lost.

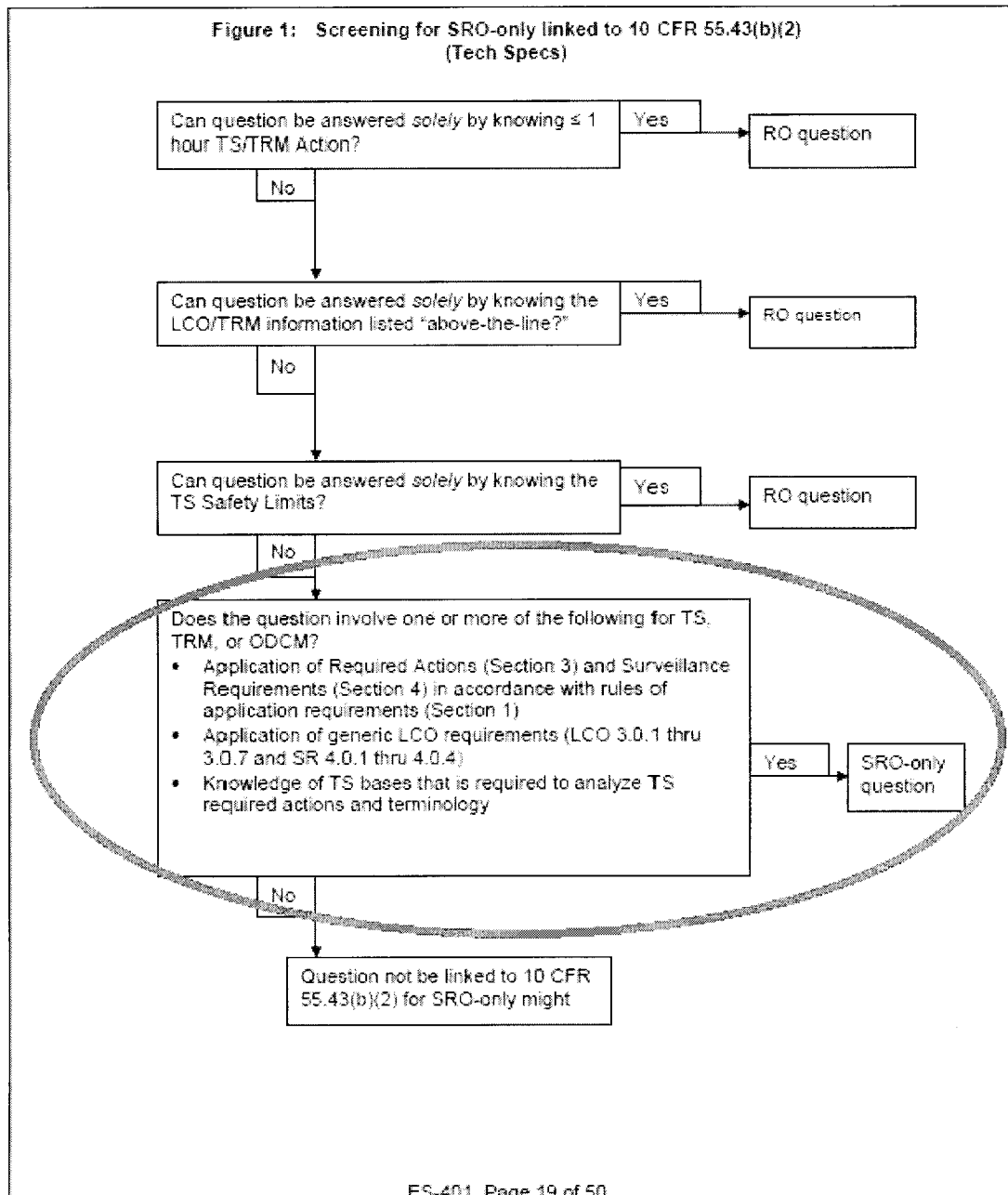
High Cog: This is high cog because the examinee must analyze the power loss and come to the understanding that both decay closed cooling water pumps are lost, then realize the plant does not meet the requirements of TS 3.4.1.1.b. The examinee must also know the action for no decay closed cooling water pumps is to initiate a reactor shutdown within 1 hour.

SRO Only: This is an SRO only question because it involves the application of T.S. 3.0.1. The SRO must analyze the stem, and realize that DC-P-1B is not operable or performing its ES function (due to EG-Y-1B not powering the 1E 4160V bus) and apply T.S. 3.0.1.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2)
(Tech Specs)



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

79

ID: 1685296

Points: 1.00

Plant conditions:

- Reactor power is 35%
- Turbine load is 300 MWe.
- Main Feedwater Pump 1A (FW-P-1A) is tagged out of service.
- All other equipment lineups are normal.

Event:

- MAP Alarm M-1-7, FWP-1B TRIP, actuates.
- Reactor power and turbine load are stable.
- OTSG "A" and "B" levels are lowering.

Given the above information, which ONE of the following actions is required?

- A. Initiate a MANUAL reactor trip and enter OP-TM-EOP-001, REACTOR TRIP.
- B. Ensure MAP H-1-1, ICS RUNBACK, actuates and implement 1102-4, POWER OPERATIONS.
- C. Rapidly reduce main turbine load until reactor power is less than 7% and implement OP-TM-424-901, Emergency Feedwater.
- D. Initiate a MANUAL turbine trip and reduce reactor power to within turbine bypass valve capability in accordance with MAP K-1-1 TURBINE TRIP.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) When reactor power is above 7%, the Main Feedwater Pump trip (where at least one MFW pump must be running) is active, meaning if reactor power is above 7% and no MFW pumps are running, the reactor and turbine should trip; (2) The presence of M-1-7, with OTSG levels lowering, but reactor power and turbine loads stable is indicative that the reactor did not trip when FW-P-1B tripped (FW-P-1A was tagged out of service in the stem, and reactor power is 35%); (3) The action for that is to trip the reactor and enter OP-TM-EOP-001, REACTOR TRIP.

A.	Initiate a MANUAL reactor trip and enter OP-TM-EOP-001, REACTOR TRIP.	Correct Answer
B.	Ensure MAP H-1-1, ICS RUNBACK, actuates and implement 1102-4, POWER OPERATIONS.	Plausible because at a higher power, when a MFW pump is lost a runback would occur (if both MFW pumps were running). In addition when both MFW pumps are lost, EFW actuates, which would provide a source of feedwater, although not one that is acceptable at 35% power to runback to less than 7% power.
C.	Rapidly reduce main turbine load until reactor power is less than 7% and implement OP-TM-424-901, Emergency Feedwater.	Plausible because at the point where the question ends, there is no other indication that any RCS parameter (i.e. RCS pressure, or reactor power) is exceeding an RPS trip setpoint. The examinee could determine since the reactor power and turbine load are stable that performing a plant runback is an acceptable option.
D.	Initiate a MANUAL turbine trip and reduce reactor power to within turbine bypass valve capability in accordance with MAP K-1-1 TURBINE TRIP.	Plausible because a turbine trip at less than 45% power does not warrant a reactor trip, but that is with the case where a MFW pump is running. At the point with where the question is, there are no MFW pumps running therefore the operators should trip the reactor and turbine in accordance with OS-24 responsibilities.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	054	AA2.01
	Importance Rating		4.4

K/A: Loss of Main Feedwater: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Occurrence of reactor and/or turbine trip.

Proposed Question: Question #79

Technical Reference(s): OP-TM-MAP-M0107, Rev 1
OP-TM-EOP-001

Proposed References to be provided to applicants during examination: None

Learning Objective: 401-GLO-11

Question Source: Bank # 1044555

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Modified Bank #

New

Question History: Systems Exam 14 Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments: (KA Match and why high cog)

K/A Match: This is a match because the examinee has to interpret that at reactor trip should have occurred when the only operating MFW pump tripped.

High Cog: This question is high cog because the examinee has to analyze the event to understand that a MFW pump did trip, but the reactor did not trip. The examinee has to determine the correct course of action for the operating crew to take.

SRO ONLY: This question is SRO only because it requires the SRO examinee to determine that an ATWS has occurred. The SRO examinee must choose between the various procedures which will shutdown the plant.

EXAMINATION ANSWER KEY

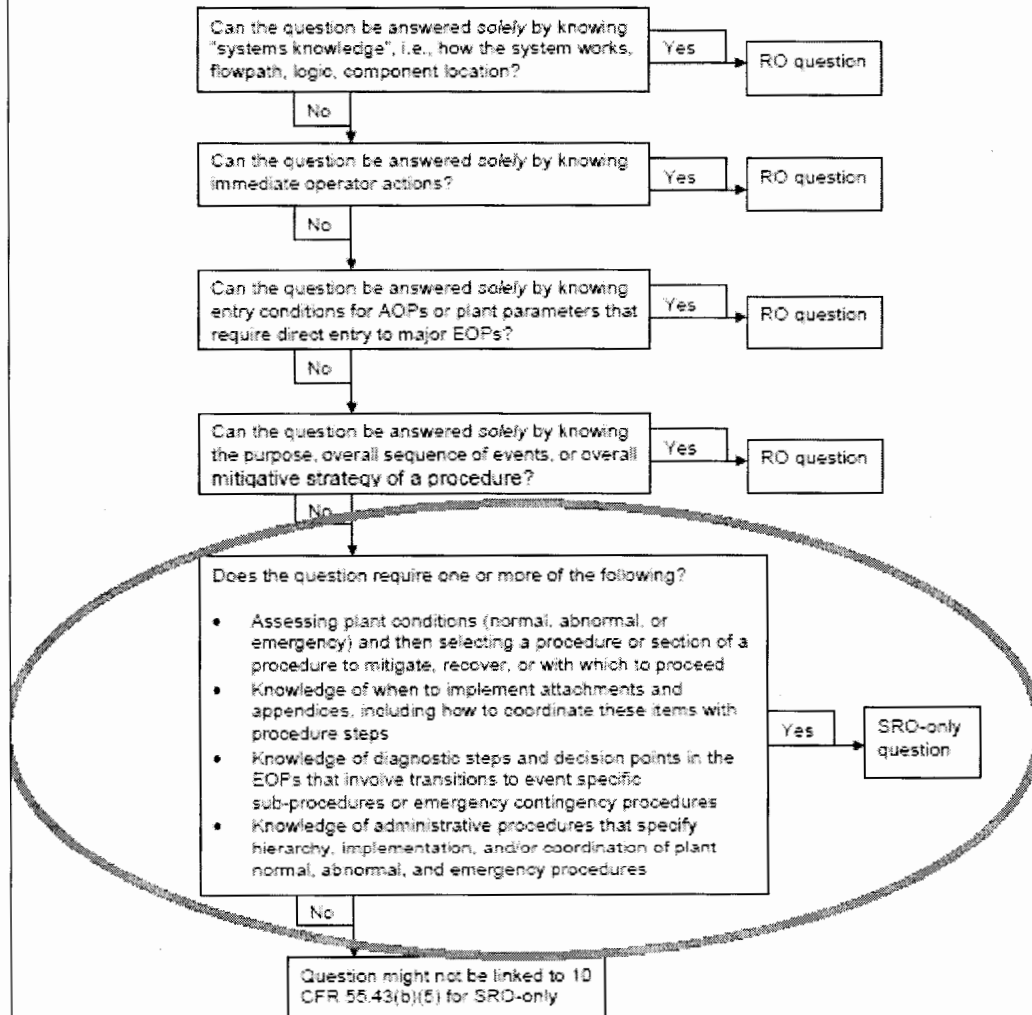
16-01 NRC EXAM SUBMITTAL

ES-401

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Attachment 2

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)**



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EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

80

ID: 1685310

Points: 1.00

Plant Conditions:

- RCS Heat up in progress.
- RCS Temperature is 290°F and rising.
- DC Distribution Panel 1M transfer switch is selected to "A" DC Distribution System and the switch on PCR is in "Auto".
- Makeup Pump 1B is supplying seal injection.

Event:

- The following alarms have actuated simultaneously:
 - AA-3-2 7 KV Bus Trouble.
 - AA-3-3 4 KV BOP Bus Trouble.
 - AA-3-5 480V BOP Bus Trouble.
 - A-1-7 Battery 1A Discharging.
 - A-2-7 Battery Charger 1A/1C/1E Trouble.
 - A-3-7 Inverter 1A/1C/1E Trouble.
 - B-3-1 4KV ES Bus Trouble.
 - PRF1-1-1 CRDM Brkr Test Trouble.
 - H&V A 4-2 Contr Bldg Bat Chgrs A Damper Tbl Fire-Smoke.
 - Loss of numerous breaker status lights at control switches.

Based on these conditions, identify the ONE selection below that describes the:

- (1) Applicable procedure to respond to these conditions, and
- (2) Impact on the ability to operate safety related equipment.

- A. (1) OP-TM-AOP-023, "A" DC System Failure.
(2) Makeup Pump 1B breaker must be racked out due to LTOP concerns.
- B. (1) OP-TM-AOP-023, "A" DC System Failure.
(2) Pressurizer Level must be verified \leq 100 inches because the PORV setpoint is greater than 592 psig.
- C. (1) OP-TM-AOP-024, "B" DC System Failure.
(2) Makeup Pump 1B breaker must be racked out due to LTOP concerns.
- D. (1) OP-TM-AOP-024, "B" DC System Failure.
(2) Pressurizer Level must be verified \leq 100 inches because the PORV setpoint is greater than 592 psig.

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) Indications from the event are representative to a Loss of 'A' DC; (2) that the PORV, RC-RV-2 is not operable when there is no 'A' DC; (3) Because there the plant is below 313F, then the requirements of Tech Spec 3.1.12 must met; (4) At the point before the loss of 'A' DC, and in accordance with 3.1.12.1, since there is an HPI breaker racked in (MU-P-1B), then MU-V-16A-D must be closed with breakers open, and MU-V-217 must be closed. In addition pressurizer level must be less than 100 inches; (5) When the PORV becomes inoperable after the loss of 'A' DC then the operating crew has 2 options to do within 8 hours: restore the setpoint below the maximum value (i.e. restore 'A' DC); or verify pressurizer level is ≤ 100 inches and satisfy the requirements of T.S 3.1.12.3. (6) Having MU-V-16A-D closed with their breakers open and MU-V-217 satisfies this requirement, and is already complete because MU-P-1B breaker is racked in supplying seal injection, hence no further action is required other than ensure pressurizer level is ≤ 100 inches.

A.	((1) OP-TM-AOP-023, "A" DC System Failure. (2) Makeup Pump 1B breaker must be racked out due to LTOP concerns.	Plausible because 'A' DC was lost, and racking out the makeup pump would be an option. Incorrect because the makeup pump does NOT have to be racked out. Since MU-V-16A-D are already closed with breakers racked out, and MU-V-217 is closed because the HPI breaker is racked in, the only action the crew would have to take is to ensure pressurizer level is ≤ 100 inches.
B.	(1) OP-TM-AOP-023, "A" DC System Failure. (2) Pressurizer Level must be verified ≤ 100 inches because the PORV setpoint is greater than 592 psig.	Correct Answer.
C.	(1) OP-TM-AOP-024, "B" DC System Failure. (2) Makeup Pump 1B breaker must be racked out due to LTOP concerns.	Plausible because the examinee could believe that 'B' DC was lost. In addition the reactor coolant system, reactor vessel, and pressurizer vents fail closed. The examinee could believe that these are LTOP valves because they could be opened to lower pressure. In addition, they could believe that racking the HPI breaker out is a compensatory action for T.S. 3.1.13 requiring a vent path from a Reactor Coolant System High Point vent (RC-40B and 41B are failed closed). This is not entered because RC-V-40A and 41A are still operable.
D.	(1) OP-TM-AOP-024, "B" DC System Failure. (2) Pressurizer Level must be verified ≤ 100 inches because the PORV setpoint is greater than 592 psig.	Plausible because the examinee could believe that 'B' DC was lost. Also, due to loss of some RCS, RV, and Pressurizer Vents, the examinee could believe that pressurizer level must be maintain less than 100 inches.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	058	AA2.03
	Importance Rating		5

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

K/A: Loss of DC Power: Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems.

Proposed Question: Question #80

Technical Reference(s): OP-TM-AOP-023, Rev 7

T.S. 3.1.12.3, AMD 281

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-023-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

55.43 b 5

Comments:

K/A Match: This question matches the K/A because the examinee must recognize that 'A' DC is lost, then choose the correct safety related equipment that is effected by the loss.

High Cog: This is high cog question because the student must analyze the alarms in the stem, then choose correct procedure and effected safety related equipment.

SRO Only: This question is SRO only because in the second part of this question the SRO examinee must determine the PORV operability based on the loss of DC power. The PORV is inoperable on a total loss of 'A' DC, but not necessarily inoperable on a partial loss of 'A' DC. The SRO examinee will have to determine applicable portion of OP-TM-AOP-023, and then determine since it is a complete loss of 'A' DC that the PORV is inoperable based on the fact that RCS temperature is <313°F and the maximum lift setpoint is greater than 592 psig.

EXAMINATION ANSWER KEY

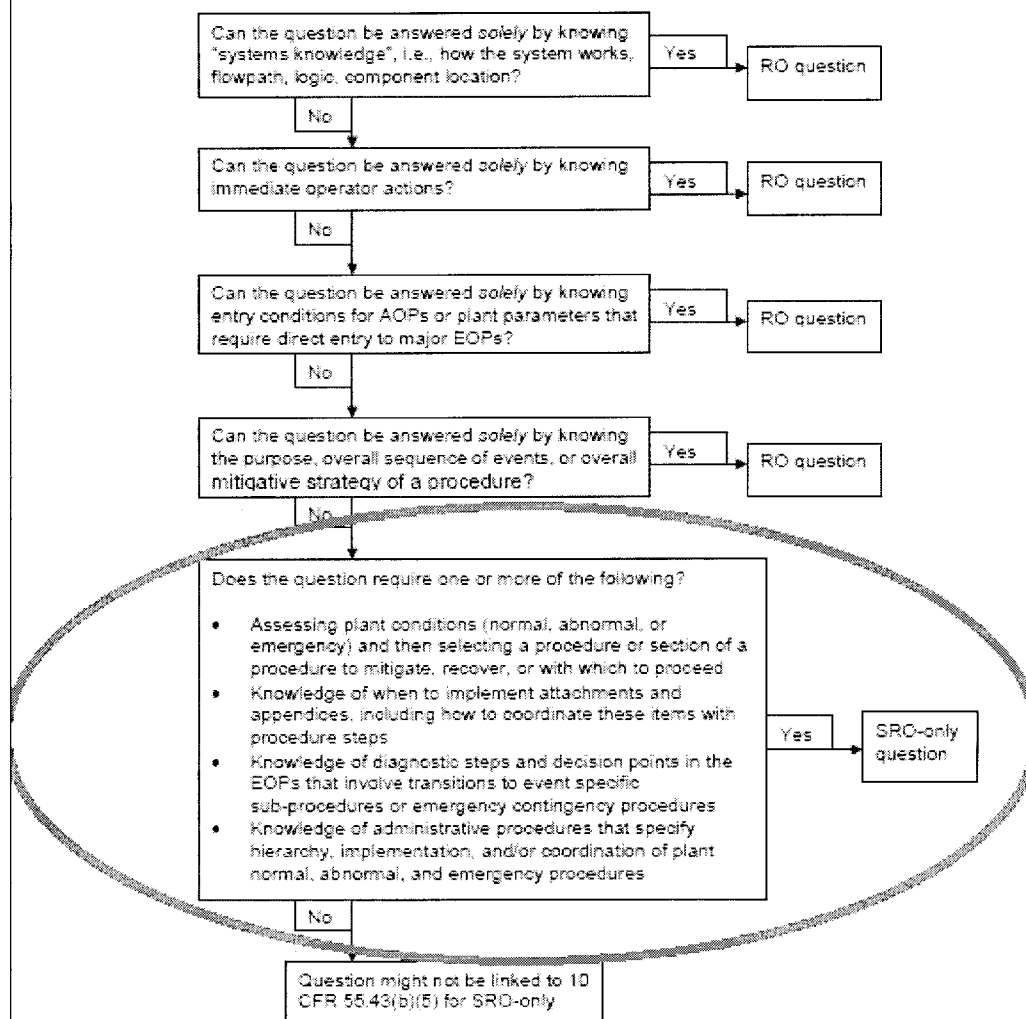
16-01 NRC EXAM SUBMITTAL

ES-401

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Attachment 2

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)**



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EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

81

ID: 1720494

Points: 1.00

Plant conditions:

- A loss of vacuum of causes both Main Feedwater pumps to trip.
- Due to equipment failures EF-P-2A, Motor Driven Emergency Feedwater Pump, is the only operating pump to supply FW to the OTSGs.
- RCS is at HSD conditions:
 - RCS T-Hot is 532 degrees F and rising.
 - RCS pressure is 2150 psig and rising.
 - OTSG pressures are 885 psig and steady.
 - OTSG levels are 25" and steady.
- FW-P-1A vacuum is being restored.

EVENT:

- EF-P-2A trip.
- Incore temperatures are rising.

Based on these conditions identify the ONE selection below that describes:

- (1) Method of core cooling to be established.
- (2) Applicable procedure.

- A.
 - (1) Rapid cooldown to LPI injection.
 - (2) OP-TM-EOP-002, LOSS OF 25 DEGREES F SUBCOOLING MARGIN.
- B.
 - (1) Condensate booster pump feed.
 - (2) OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER.
- C.
 - (1) Rapid cooldown to LPI injection.
 - (2) OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER.
- D.
 - (1) Condensate booster pump feed.
 - (2) OP-TM-EOP-002, LOSS OF 25 DEGREES F SUBCOOLING MARGIN.

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) One of the entry criteria for OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, is met (incore temperatures rising and NO FEEDWATER available); (2) Once the EOP is entered, the examinee must know the criteria for each know the routing step that applies to go begin cooling the Reactor, for this question, the examinee must realize that a condensate pump is on, a reactor coolant pump is on, therefore Attachment 1 of OP-TM-EOP-004 is the method of cooling which should be used next.

A.	(1) Rapid cooldown to LPI injection. (2) OP-TM-EOP-002, LOSS OF 25 DEGREES F SUBCOOLING MARGIN.	Incorrect - Plausible because OP-TM-EOP-004 mitigation strategy involves lowering OTSG pressure to strengthen the heat sink to establish heat transfer out of the OTSG's. In addition, if the examinee believes that if SCM is less than 25F, that OP-TM-EOP-002 is entered then it may be possible to perform a cooldown to LPI.
B.	(1) Condensate booster pump feed. (2) OP-TM-EOP-004, Lack of Primary to Secondary Heat Transfer.	Correct Answer - The stem only indicates that both Main Feedwater Pumps are tripped. All equipment is available to support Condensate Booster Pump feed.
C.	(1) Rapid cooldown to LPI injection. (2) OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER.	Incorrect- Plausible because in OP-TM-EOP-004, a cooldown is performed after Feedwater becomes available. Incorrect because the cooldown is not a rapid cooldown to LPI.
D.	(1) Condensate booster pump feed. (2) OP-TM-EOP-002, LOSS OF 25 DEGREES F SUBCOOLING MARGIN.	Incorrect- Plausible because the operators must feed in accordance with Rule 4, which includes raising OTSG levels to 75% to 85% in the operating range. Incorrect because condensate booster pump cooling is not an option in that guide.

Examination Outline Cross-reference:		Level	RO	SRO
		Tier #		1
		Group #		1
		K/A #	E04	2.2.44
		Importance Rating		4.4

K/A: Inadequate Heat Transfer: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question: Question #81

Technical Reference(s): OP-TM-EOP-004, Rev 11
OS-24, Rev 28

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP004-PCO-4

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question Source: Bank # 363689

Modified Bank #

New

Question History: Comp 2 Last NRC Exam:

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 stem indications to determine the correct path for establishing feedwater to the OTSGs. The operator actions will establish core cooling.

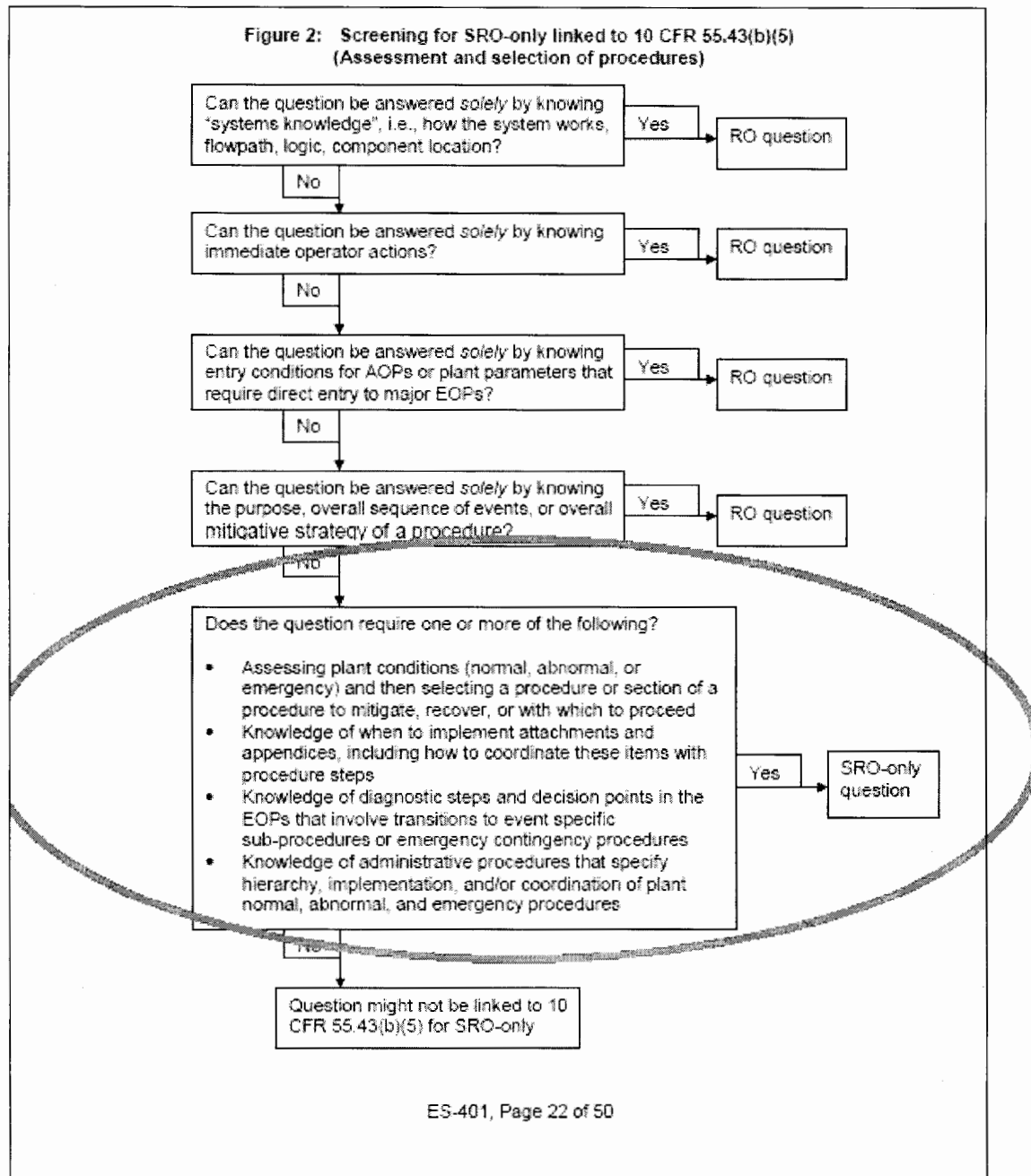
High Cog: This question is High Cog because the examinee will have analyze the stem conditions to determine there is no feedwater available to the OTSGs.

SRO ONLY: This question is SRO under b.5 because in addition to recognizing the entry conditions into OP-TM-EOP-004 the examinee must assess plant conditions then select the NEXT procedure which will provide feedwater and core cooling.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

82

ID: 1737141

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- LOCA requiring a plant trip.
- Pressurizer Level cannot be maintained without HPI.
- OP-TM-EOP-006 LOCA COOLDOWN is in progress.

(1) Which Reactor Coolant Pumps must get secured?

(2) What is the OP-TM-EOP-006 basis for securing them?

- A. (1) RC-P-1A and RC-P-1B
(2) Minimize the RCS inventory loss due to the LOCA.
- B. (1) RC-P-1A and RC-P-1B
(2) The subsequent cooldown could result in "core lift".
- C. (1) RC-P-1C and RC-P-1D
(2) Minimize the RCS inventory loss due to the LOCA.
- D. (1) RC-P-1C and RC-P-1D
(2) The subsequent cooldown could result in "core lift".

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-EOP-006, if all four Reactor Coolant Pumps are running, two of them get secured; (2) The basis for this to ensure no more than three Reactor Coolant Pumps are running when Tcold drops below 407F; (3) With four Reactor Coolant Pumps the differential pressure may result in core lift; (4) OP-TM-EOP-006 directs RC-P-1C and RC-P-1D to be shutdown.			
A.	(1) RC-P-1A and RC-P-1B (2) Minimize the RCS inventory lost due to the LOCA. (1) RC-P-1A and RC-P-1B (2) Minimize the RCS inventory lost due to the LOCA.	INCORRECT: Plausible but incorrect because the step indicates if RC-P-1A and RC-P-1B are running, THEN shutdown RC-P-1C and RC-P-1D. In addition the basis is to prevent core lift.	
B.	(1) RC-P-1A and RC-P-1B (2) Lower the RCS flow and differential pressure across the fuel bundles.	INCORRECT: Plausible but incorrect because the step indicates if RC-P-1A and RC-P-1B are running, THEN shutdown RC-P-1C and RC-P-1D.	
C.	(1) RC-P-1C and RC-P-1D (2) Minimize the RCS inventory lost due to the LOCA.	INCORRECT: Plausible because this action could lower the rate of RCS inventory loss on some LOCA. Incorrect because the Reactor Coolant Pumps are secured due to core lift concerns.	
D.	(1) RC-P-1C and RC-P-1D (2) The subsequent cooldown could result in "core lift".	Correct Answer: See above.	
Examination Outline Cross-reference:			
Level		RO	SRO
Tier #			1
Group #			2
K/A #		BW/E08	2.4.18
Importance Rating			4.0
K/A: LOCA Cooldown: Knowledge of the specific bases for EOPs.			
Proposed Question: Question #82			
Technical Reference(s): OP-TM-EOP-006, Rev 12 OP-TM-EOP-0061, Rev 7			
Proposed References to be provided to applicants during examination: None			
Learning Objective: EOP006-PCO-4			
Question Source: Bank #			
Modified Bank #			
New X			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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:

K/A Match: This question matches the K/A because the examinee must know the reason for securing two Reactor Coolant Pumps on a LOCA Cooldown.

SRO ONLY: The question is SRO only because the examinee must assess plant conditions and know the content of OP-TM-EOP-006 in order to select a required course of action. In addition, the examinee must know the basis for the action selected. This procedure is not one of the major EOPs, it is a supplementary EOP directed from within the major EOP.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

83

ID: 1737631

Points: 1.00

The specific activity of the primary and secondary coolant are as follows:

	Two days ago at 1200	Yesterday at 1200	Today at 1200
RCS Dose Equivalent I-131	0.50 microcuries/gram	0.53 microcuries/gram	0.52 microcuries/gram
OTSG Dose Equivalent I-131	0.05 microcuries/gram	0.053 microcuries/gram	0.052 microcuries/gram

The ____ (1) ____ is out of specification, and the basis of the specification is to maintain dose within the limits from ____ (2) ____.

- A. (1) RCS
(2) a tube rupture accident
- B. (1) OTSG
(2) a steam line break accident
- C. (1) RCS
(2) baseline OTSG tube leakage
- D. (1) OTSG
(2) normal condenser offgas

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) The limit for I-131 in the RCS is 0.35 microcuries/gram; (2) The limit for I-131 in the secondary fluid is .10 microcuries/gram, which is not exceeded in this question; (3) The basis for the RCS limit is that if activity levels are kept below this level then analysis shows that the calculated doses are within acceptable limits on a Steam Line Break (SLB) and Steam Generator Tube Rupture (SGTR) accident (T.S. 3.1.4.1 basis)

A.	(1) RCS (2) dose from a tube rupture accident	CORRECT ANSWER: See above.		
B.	(1) OTSG (2) dose from a steam line break accident	INCORRECT: Plausible if the examinee is not familiar with the secondary chemistry limits. In addition, the examinee could believe the secondary is the concern due to having one less barrier than the RCS to atmosphere.		
C.	(1) RCS (2) dose from baseline OTSG tube leakage	INCORRECT: Plausible because the RCS is violating the tech spec chemistry limits. Incorrect because the basis is not dose from a baseline OTSG tube leak. That is the basis for violating the secondary chemistry limits.		
D.	(1) OTSG (2) dose from normal condenser offgas	INCORRECT: Plausible if the examinee is not familiar with the secondary chemistry limits. In addition, dose from the normal condenser offgas could be a concern. Incorrect because the secondary chemistry is within limits.		

Examination Outline Cross-reference:		Level	RO	SRO
		Tier #		1
		Group #		2
		K/A #	076	2.2.38
		Importance Rating	3.6	4.5

K/A: High Reactor Coolant Activity: Knowledge of conditions and limitations in the facility license.

Proposed Question: Question #83

Technical Reference(s): T.S. 3.1.4
T.S. 3.1.13

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO-14

Question Source: Bank # 718419
Modified Bank #
New

Question History: System Exam 12 Last NRC Exam: 08-01

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 basis for the primary chemistry limits.

SRO ONLY: This question is SRO only because the examinee must identify the out of specification chemistry reading, and understand the basis behind the limit.

Correct answer. TS 3.1.4 states "With the specific activity of the primary coolant greater than 0.35 microcurie/gram DOSE I EQUIVALENT I-131 for more than 48 hours*" during one continuous time interval or exceeding the limit line shown on Figure 3.1-24 be in at least HOT SHUTDOWN within 6 hours. Power operation may continue when DOSE EQUIVALENT I-131 is below 0.35 microcurie/gram."

Basis "The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will be well within the Part 100 limit following a steam generator tube rupture accident or steam line break accident with postulated accident induced steam generator tube leakage in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM."

Plausible since this is the specific activity limit for the Secondary Coolant and it is based on a steam line rupture with tube leakage; however the specific activity limit is 0.35 microcurie/gram for the Primary Coolant and the primary-to-secondary steam generator leakage rate of 1.0 GPM is assumed.

Plausible since the limit and basis is correct; however it is correct for the Primary Coolant.

3.13 SECONDARY COOLANT SYSTEM ACTIVITY

3.13.1 The specific activity of the secondary coolant system shall be ≤ 0.10 - Ci/gram DOSE EQUIVALENT I-131.

3.13.2 With the specific activity of the secondary coolant system > 0.10 -Ci/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

Bases

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary-to-secondary tube leak in the steam generator of the affected steam line.

Plausible since the limit is correct for the Secondary Coolant; however the basis is for a coincident 1.0 GPM primary-to-secondary tube leak.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

84

ID: 1699987

Points: 1.00

The following plant conditions exist:

- A small break LOCA is in progress.
- The Reactor tripped, and ESAS actuated on low RCS pressure.
- A loss of off-site power (LOOP) occurred.
- Sub cooling margin was lost and is currently -1°F on the PT plot.
- Incore temperatures are rising slowly.
- RCITS shows a large steam void in each RCS hot leg.
- Current cooldown rate is $45^{\circ}\text{F} / \text{hr}$.

Which ONE of the following describes the correct actions the crew must take to ensure adequate core cooling?

- A. OTSG level must be maintained above 50%, with OTSG saturation temperature from 40°F to 60°F below RCS cold leg temperature to promote boiler-condenser cooling. No additional action is required due to the steam void.
- B. OTSG level must be maintained above 50%, with OTSG saturation temperature from 40°F to 60°F below RCS cold leg temperature to allow single-phase natural circulation. The steam void must be vented from the hot leg.
- C. OTSG level must be maintained 75% - 85%, with OTSG pressure at least 100 psig below RCS pressure to promote boiler-condenser cooling. No additional action is required due to the steam void.
- D. OTSG level must be maintained 75% - 85%, with OTSG pressure at least 100 psig below RCS pressure to allow single-phase natural circulation. The steam void must be vented from the hot leg.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) The controlling procedure is OP-TM-EOP-002, LOSS OF 25F SUBCOOLING MARGIN; (2) In addition to implementing actions OP-TM-EOP-002, the operating crew will be performing OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS, in which the entry criteria for Rule 4, FEEDWATER CONTROL, and Guide 6, OTSG PRESSURE CONTROL have ben met, which provide governance on how to promote heat transfer in loss of 25F subcooling event; (3) In addition steps are taken to remove non-condensible gasses from the RCS (if there were symptoms of them being present), NOT steam.			
A.	OTSG level must be maintained above 50%, with OTSG saturation temperature from 40°F to 60°F below RCS cold leg temperature to promote boiler-condenser cooling. No additional action is required due to the steam void.	INCORRECT ANSWER: Plausible because maintaining greater than 50% in the operating range is the level band if reactor coolant pumps are secured. Maintaining OTSG saturation temperature in the 40-60F band below RCS cold leg temperature is the band if the cooldown rate would be less than 40 F/hr. OP-TM-EOP-004 also addresses how to address a steam void of non-condensable gasses in the RCS. In this case, this would not be necessary due to the cooldown rate of > 40F/hr being present.	
B.	OTSG level must be maintained above 50%, with OTSG saturation temperature from 40°F to 60°F below RCS cold leg temperature to allow single-phase natural circulation. The steam void must be vented from the hot leg.	INCORRECT ANSWER: Plausible because maintaining greater than 50% in the operating range is the level band if reactor coolant pumps are secured. Maintaining OTSG saturation temperature in the 40-60F band below RCS cold leg temperature is the band if the cooldown rate would be less than 40 F/hr. OP-TM-EOP-004 also addresses how to address a steam void of non-condensable gasses in the RCS. In this case, this would not be necessary due to the cooldown rate of > 40F/hr being present.	
C.	OTSG level must be maintained 75% - 85%, with OTSG pressure at least 100 psig below RCS pressure to promote boiler-condenser cooling. No additional action is required due to the steam void.	CORRECT ANSWER. Both actions are required by OP-TM-EOP-010 (Rule 4 and Guide 6). Steps to vent are in OP-TM-EOP-004, which was not entered in this question. In addition, venting is only performed if it is believed non-condensable gasses are present in the RCS, not steam.	
D.	OTSG level must be maintained 75% - 85%, with OTSG pressure at least 100 psig below RCS pressure to allow single-phase natural circulation. The steam void must be vented from the hot leg.	INCORRECT ANSWER: Plausible because both actions are required by OP-TM-EOP-010 (Rule 4 and Guide 6). Steps to vent are in OP-TM-EOP-004, which was not entered in this question. In addition, venting is only performed if it is believed non-condensable gasses are present in the RCS, not steam.	
Examination Outline Cross-reference:			
Level		RO	SRO
Tier #			1
Group #			2
K/A #		E03	EA2.1
Importance Rating			4.0

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

K/A: Inadequate Subcooling Margin: Ability to determine and interpret the following as the apply to the (Inadequate Subcooling Margin): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: Question #84

Technical Reference(s): OP-TM-EOP-002

OP-TM-EOP-010, Rev 19

Proposed References to be provided to applicants during examination: None

Learning Objective: N-TM-TQ-104-EOP-DBIG-APCO -1

Question Source: Bank # 862875

Modified Bank #

New

Question History: Comp 1 Exam Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 b.5

Comments:

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

K/A Match: This matches the K/A because the examinee will have to know which procedures to use to promote cooling in the RCS. Although not explicitly stated, by asking steps in Rule 4 and Guide 6, the examinee will have selected which procedure is appropriate. In addition, to rule out the two of the distractors, the examinee will NOT have selected to choose another plausible EOP network path into OP-TM-EOP-004.

High Cog: This question is high cog because the examinee must analyze the given set of parameters and choose the correct procedure and steps in which to execute.

SRO Only: This question is an SRO only question because it the examinee must assess plant conditions and then select procedure steps to mitigate the casualty. This question tests coordination between how the operating crew will feed and steam the OTSGs.

EXAMINATION ANSWER KEY

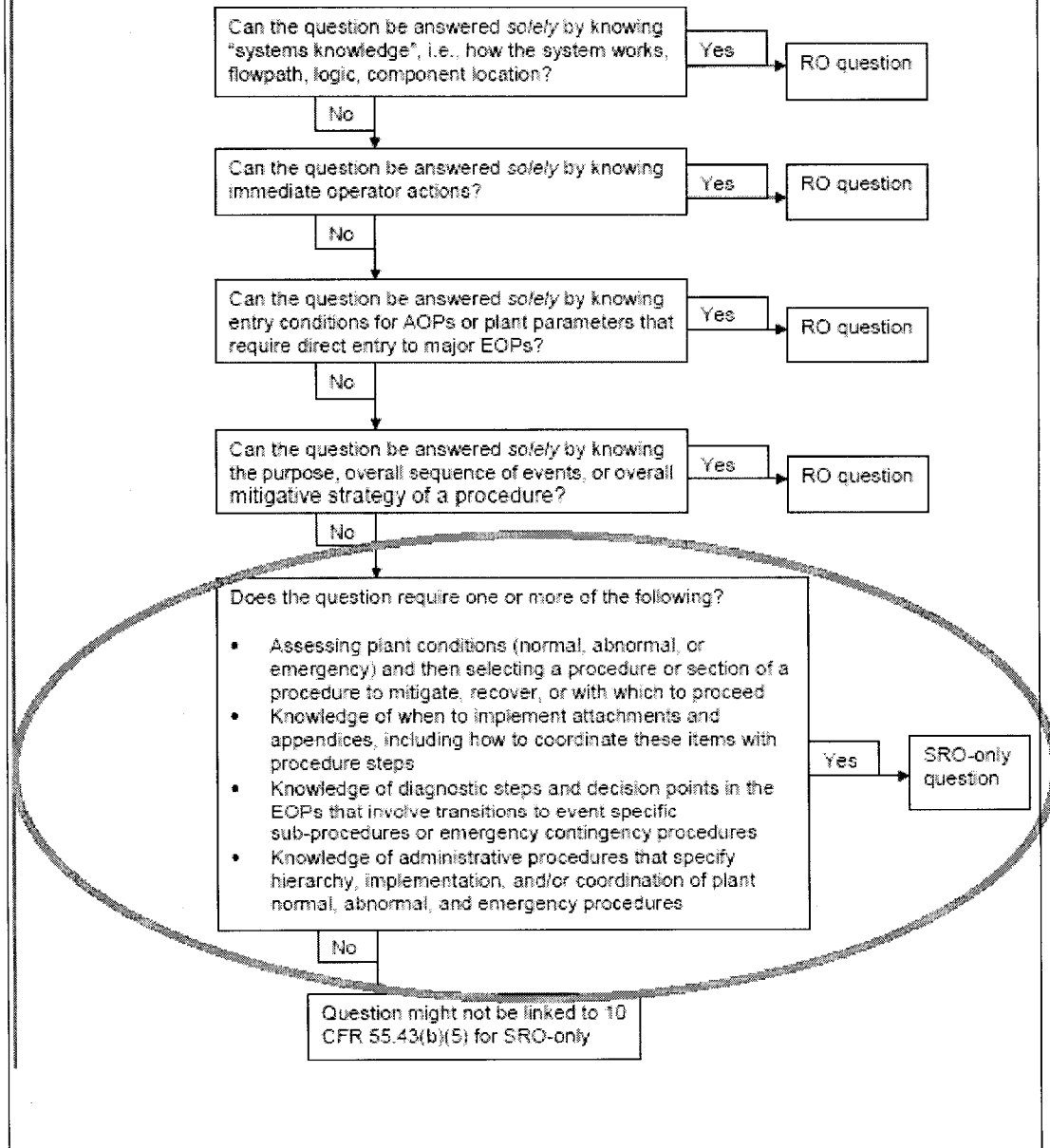
16-01 NRC EXAM SUBMITTAL

ES-401

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Attachment 2

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

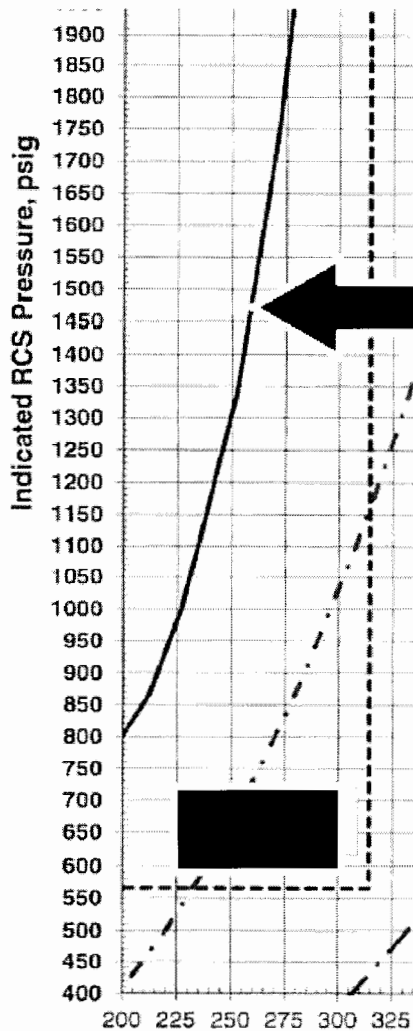
EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

85

ID: 1720678

Points: 1.00



<<

REFERENCE PROVIDED

Plant conditions:

- Cooldown in progress in accordance with OP-TM-EOP-006, LOCA COOLDOWN.

EVENT:

- Due to a plant upset, RCS temperature is approaching the curve on Figure 1, RCS PRESSURE-TEMPERATURE LIMITS.

The operating crew must ensure open the ____ (1) ____ to prevent exceeding the ____ (2) ____, in accordance with Guide 23, RCS PRESSURE AND TEMPERATURE LIMITS.

- A. (1) PORV
(2) TS NDT Curve
- B. (1) PORV
(2) PZR SURGE LIMIT Curve
- C. (1) Spray Valve
(2) TS NDT Curve
- D. (1) Spray Valve
(2) PZR SURGE LIMIT Curve

Answer: A

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<p><<Explanation: To answer this question correctly, the examinee must know: (1) During and LOCA Cooledown, OP-TM-EOP-010, Guide 23 and Figure 1 and 1a are used to maintain RCS pressure and temperature within limits; (2) The line that the arrow is point to is the TS NDT line, which is a composite curve to Tech Spec Figures 3.1-1 and 3.1-2 based on the reactor vessel individual components and welds; (3) The PORV is specified for RCS pressure control to move RCS pressure away from the limits described by the curve because RCS pressure reduction is the most effective method from moving away from the curve.</p>																		
A.	(1) Open the PORV (2) TS NDT Curve	Correct Answer - Guide 23, Step 1 : If RCS pressure is approaching "TS NDT limit", then OPEN RC-RV-2 (PORV).																
B.	(1) Open the PORV (2) PZR SURGE LIMIT Curve	Plausible because this curve is in the same general area, but on FIGURE 1A, LOW RANGE RCS PRESSURE-TEMPERATURE LIMITS. The action for exceeding that curve is to lower RCS pressure or raise RCS temperature. Incorrect because the curve referenced in the picture is the TS NDT Curve.																
C.	(1) Open the Spray Valve (2) TS NDT Curve	Plausible because opening the Spray Valve will lower pressure away from the TS NDT Curve. Incorrect because GUIDE 23 directs use of the PORV to lower pressure.																
D.	(1) Open the Spray Valve (2) PZR SURGE LIMIT Curve	Plausible because if the examinee believed that the PZR SURGE LIMIT Curve was being reference then opening the spray valve would be an acceptable method for lowering pressure. Incorrect because the TS NDT Curve is being referenced.																
<p>Examination Outline Cross-reference:</p> <table border="1"> <thead> <tr> <th>Level</th> <th>RO</th> <th>SRO</th> </tr> </thead> <tbody> <tr> <td>Tier #</td> <td></td> <td>1</td> </tr> <tr> <td>Group #</td> <td></td> <td>2</td> </tr> <tr> <td>K/A #</td> <td>BW/E13</td> <td>2.2.42</td> </tr> <tr> <td>Importance Rating</td> <td></td> <td>4.6</td> </tr> </tbody> </table>				Level	RO	SRO	Tier #		1	Group #		2	K/A #	BW/E13	2.2.42	Importance Rating		4.6
Level	RO	SRO																
Tier #		1																
Group #		2																
K/A #	BW/E13	2.2.42																
Importance Rating		4.6																
<p>K/A: EOP Rules and Guides: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.</p>																		
<p>Proposed Question: Question #85</p>																		
<p>Technical Reference(s): OP-TM-EOP-006, Rev 12 T.S Figure 3.1-1 OP-TM-EOP-0061, Rev 7 OP-TM-EOP-010, Rev 19</p>																		
<p>Proposed References to be provided to applicants during examination: OP-TM-EOP-010, Figure 1</p>																		
<p>Learning Objective: EOP006-PCO-4</p>																		
<p>Question Source: Bank # Modified Bank # New X</p>																		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 will have to recognize that RCS pressure-temperature is approaching the TS NDT Curve, which would be a violation of Technical Specifications if violated.			
High Cog: This question is high cog because the examinee will have to analyze the situation and choose the correct course of action.			
SRO Only: This question is SRO only because it involves recognizing that a TS NDT limit is going to be violated and choosing the correct action to prevent violation.			

EXAMINATION ANSWER KEY

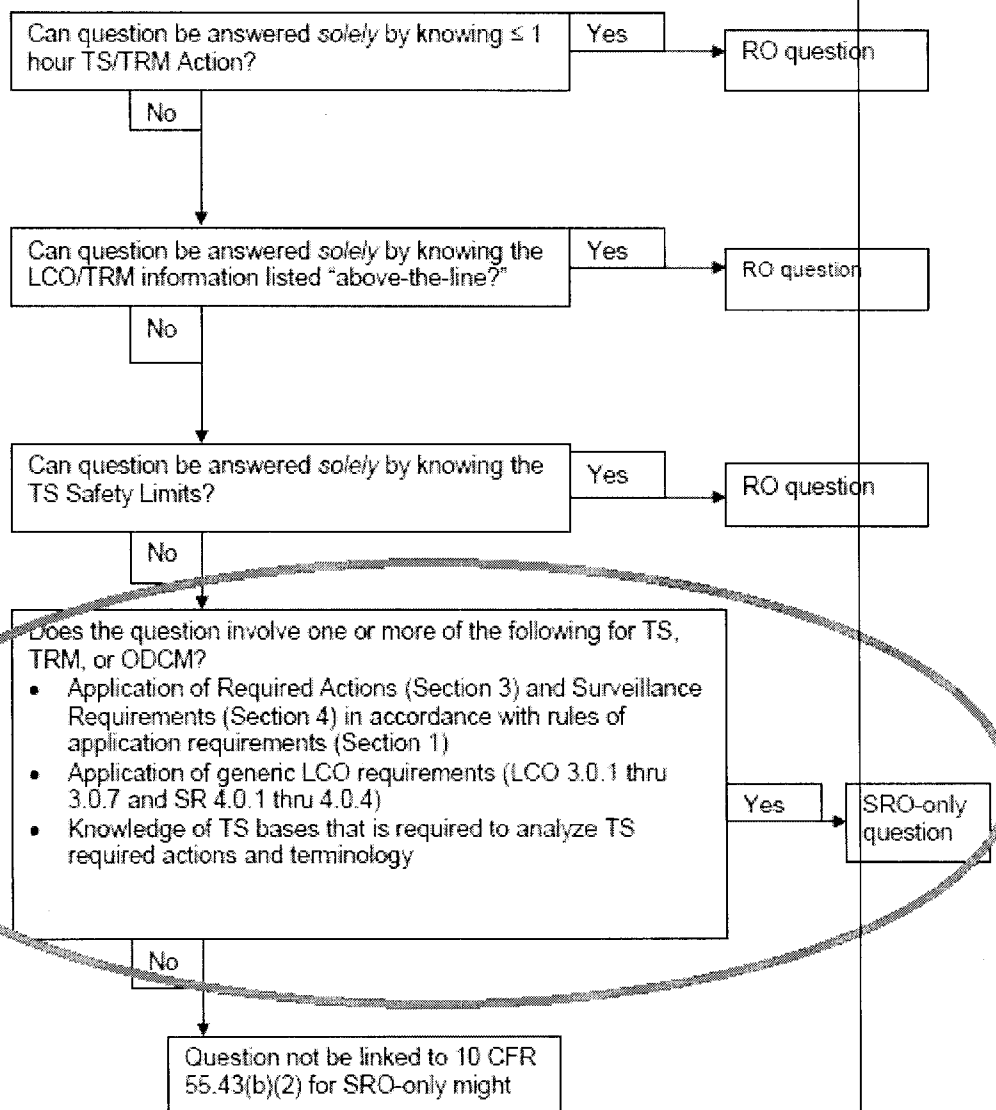
16-01 NRC EXAM SUBMITTAL

ES-401

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Attachment 2

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

86

ID: 1720700

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- BWST Boron Concentration is 2550 ppm.
- BWST Water Temperature is 52°F.

EVENT:

- Chemistry reports the latest BWST sample results as follows:
 - BWST Boron Concentration is 2490 ppm.
 - BWST Water Temperature is 45°F.

What is currently (1) the immediate concern, and (2) the action required?

- A. (1) Boron crystallization.
(2) BWST water temperature must be restored to limits within **8 hours IAW** OP-TM-212-457, BWST Heater Operations.
- B. (1) Boron crystallization.
(2) BWST water temperature must be restored to limits within **1 hour IAW** OP-TM-212-457, BWST Heater Operations.
- C. (1) The ability to ensure the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident.
(2) BWST boron concentration must be restored to limits within **8 hours IAW** OP-TM-212-527, Transfer of C RCBT to the BWST.
- D. (1) The ability to ensure the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident.
(2) BWST boron concentration must be restored to limits within **1 hour IAW** OP-TM-212-527, Transfer of C RCBT to the BWST.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) According to technical specification 3.3.1.1, 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) The basis for this technical specification is the 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) Boron crystallization. (2) BWST water temperature must be restored to limits within 8 hours IAW OP-TM-212-457, BWST Heater Operations.	Incorrect - Plausible because if the BWST temperature were below the 40F limit, then boron crystallization would be a concern. Incorrect because the boron concentration is below the technical specification limit.	
B.	(1) Boron crystallization. (2) BWST water temperature must be restored to limits within 1 hour IAW OP-TM-212-457, BWST Heater Operations.	Incorrect - Plausible because if the BWST temperature were below the 40F limit, then boron crystallization would be a concern. Incorrect because the boron concentration is below the technical specification limit.	
C.	(1) The ability to ensure the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident. (2) BWST boron concentration must be restored to limits within 8 hours IAW OP-TM-212-527, Transfer of C RCBT to the BWST.	Correct Answer - See above.	
D.	(1) The ability to ensure the reactor will remain at least one percent subcritical following a Loss-of-Coolant Accident. (2) BWST boron concentration must be restored to limits within 1 hour IAW OP-TM-212-527, Transfer of C RCBT to the BWST.	Incorrect - Plausible because the action statement if the boron concentration cannot be restored is to be in initiate a shutdown 1 hour to place the plant in a condition where the condition does not apply.	
Examination Outline Cross-reference:			
Level		RO	SRO
Tier #			2

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Group #	1	
K/A #	006	2.4.9
Importance Rating	4.2	

K/A: Emergency Core Cooling: 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 #86

Technical Reference(s): Tech Spec 3.3.1.1 and basis

Proposed References to be provided to applicants during examination: None

Learning Objective: 212-GLO-14

Question Source: Bank # 860008
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.2

Comments:

KA Match: The questions matches the KA because the examinee must know the implication of an ECCS system parameter not being at the technical specification minimum. This technical specification minimum ensures the reactor is shutdown in the event of a LOCA.

High Cog: This question is high cog because the examinee has to identify the out of specification and apply the LCO time.

SRO ONLY: This question is SRO only because the examinee must know the condition, required action, and completion time in addition to the basis.

EXAMINATION ANSWER KEY

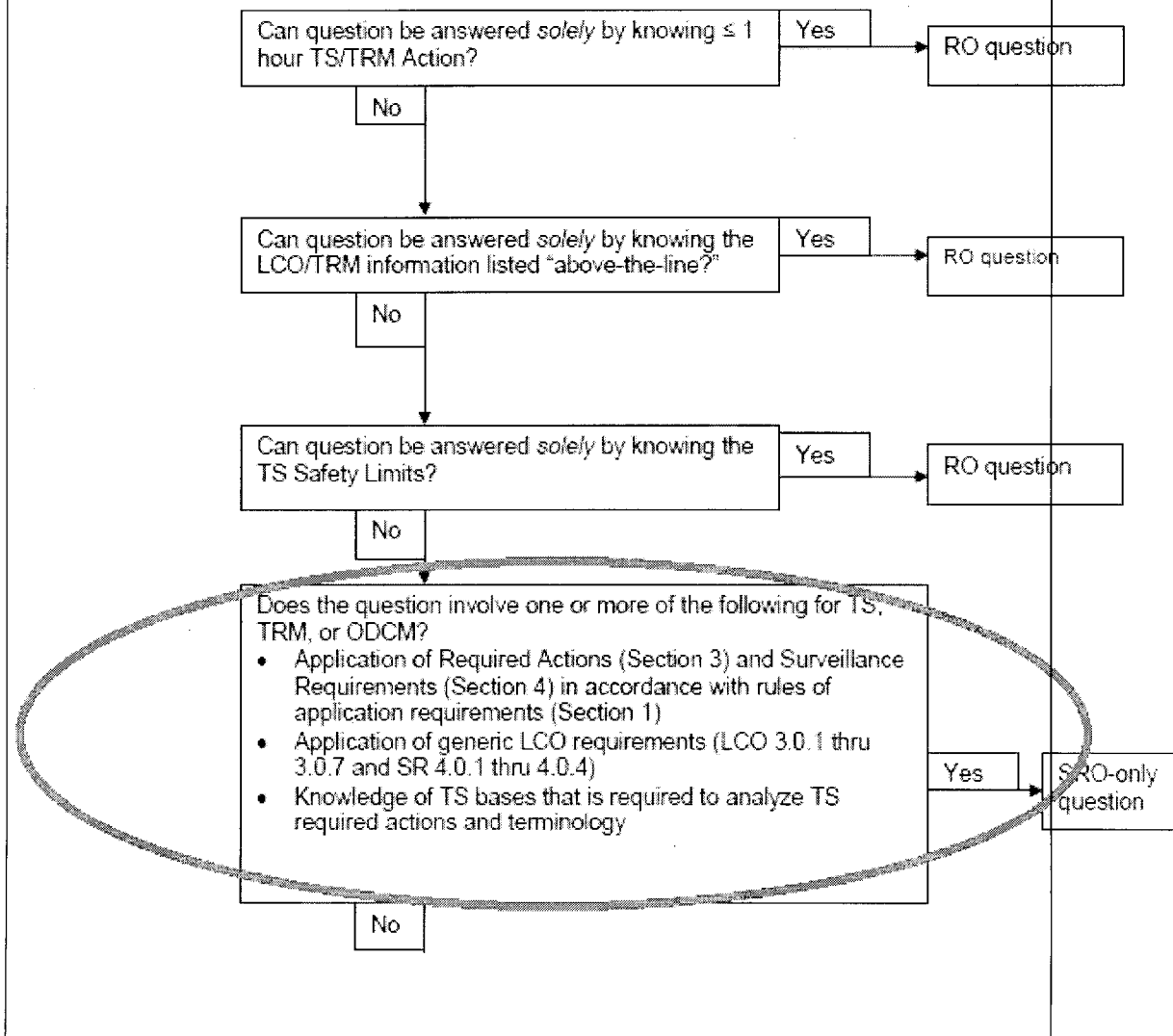
16-01 NRC EXAM SUBMITTAL

ES-401

5

Attachment 2

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

87

ID: 1720770

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 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

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<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 Primary Safe Shutdown NLO will perform Attachment 5, which among other things, will terminate uncontrolled HPI due to a spurious "A" train ES actuation by opening the breakers and locally closing MU-V-16A and B.

A.	(1) Attachment 5, "Preventing Spurious Operation of MOV's" (2) Terminate uncontrolled HPI due to a spurious "A" train ES actuation.	Correct Answer - The Primary Safe Shutdown NLO will perform attachment 5.
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 the turbine bypass valves failing midscale on some fire events. The MS-V-2's could also be used to isolated 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.
C.	(1) Attachment 13, "Tripping RCPs Locally" (2) Prevent the Reactor Coolant Pumps from spuriously starting.	Incorrect Answer - Plausible because Attachment 13 is performed by the Secondary Safe Shutdown NLO. Preventing Reactor Coolant Pumps from spuriously starting is one basis for performing Attachment 13.
D.	(1) Attachment 13, "Tripping RCPs Locally" (2) Prevent Reactor Coolant pumps from operating with no seal injection.	Incorrect Answer - Plausible because Attachment 13 is performed by the Secondary Safe Shutdown NLO. Preventing Reactor Coolant Pumps from operating without seal injection is one basis for performing Attachment 13.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	013	2.4.35
	Importance Rating		4.0

K/A: Engineered Safety Features Actuation: Knowledge of local auxiliary operator tasks during an emergency and their resultant operational effects.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Proposed Question: Question #87

Technical Reference(s): OP-TM-EOP-020, Rev 22 OS-24, Rev 28

OP-TM-EOP-0201, Rev 15

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP020-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

55.43 b.5

Comments:

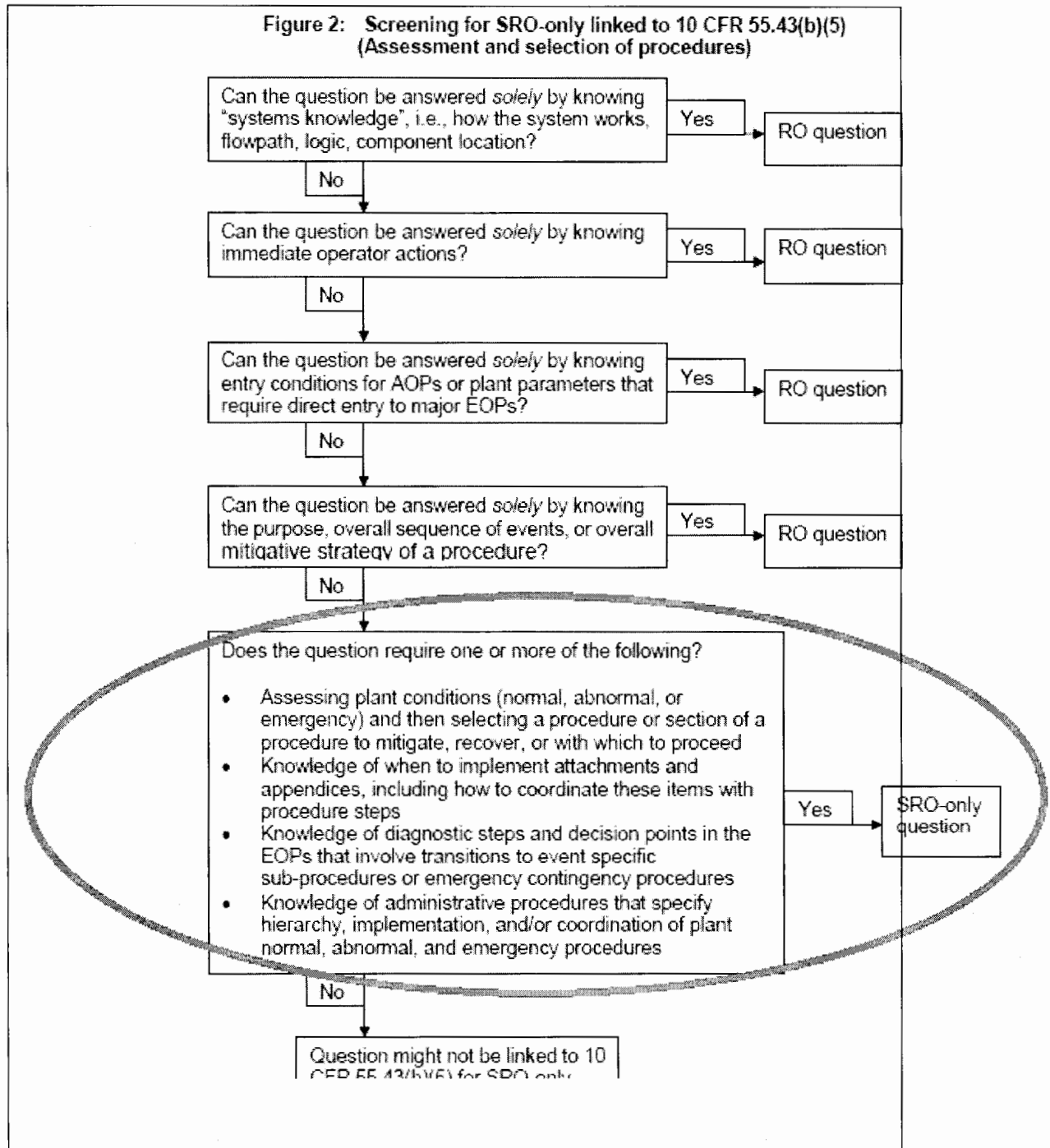
KA Match: This question matches the KA because the examinee will have to know that a fire in the relay room could cause an unwanted HPI actuation, but that the Attachment 5 will close MU-V-16A and B which would mitigate the consequence of such event.

High Cog: This question is high cog because the examinee will have to assess plant conditions to determine that a control room evacuation is required and that the remote shutdown sequence has been directed. Step 2.8 is a stopping point in which the operating crew would wait until conditions in the control room have deteriorated enough that crew must evacuate. This is denoted by the tripping of the feedwater, condensate booster, and condensate pumps.

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.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

88

ID: 1720789

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Main and Emergency Feedwater are both lost.
- HPI/PORV cooling is in progress.

POST EVENT:

- Both Main and Emergency Feedwater are restored.
- One Reactor Coolant Pump is running in each loop.
- Both OTSG's are intact and dry.
- PPC Points C4015 and C4016 (OTSG tube-to-shell differential temperatures) are at +70F.

In accordance with Guide 13, FEEDING A DRY OR DEPRESSURIZED OTSG, ____ (1) ____ is the preferred feed source because it will ____ (2) ____.

- A. (1) Main Feedwater
(2) cool down the OTSG tubes faster
- B. (1) Main Feedwater
(2) minimize the tensile loading on the OTSG tubes
- C. (1) Emergency Feedwater
(2) cool down the OTSG tubes faster
- D. (1) Emergency Feedwater
(2) minimize the tensile loading on the OTSG tubes

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) The stem sets up a scenario in which the plant is experiencing a lack of primary to secondary heat transfer (i.e. Reactor tripped, no feedwater available) which heats up the RCS, and thus the OTSG tubes heat; (2) While the tubes heat up, the OTSG shell temperature should remain constant or slightly cooldown due to the lack of feedwater; (3) This causes a compressive stress on the OTSG tubes (which is a positive TSDT); (4) This is also indicated by the (+) on computer points C4015 and C4016; (5) Being above the (+) limit of TSDT means there is a compressive stress on the OTSGs; (6) The actions for a (+) TSDT in Guide 13 with a reactor coolant pump on is to feed the OTSG with EFW; (7) In accordance with the Guide 13 basis document, EFW is chosen because it sprays directly on the OTSG tubes, and will lower the tube temperature immediately, thus lowering the TSDT and compressive stresses; (8) Since EFW sprays directly on the tubes it will cool down the tubes faster than MFW.

A.	(1) Main Feedwater (2) cool down the OTSG tubes faster	Incorrect - Plausible because in most cases when a reactor coolant pump is on, MFW is the preferred source of feedwater. This would be the case if the TSDT was negative. In addition, MFW would not cool down the OTSG tubes faster.
B.	(1) Main Feedwater (2) minimize the tensile loading on the OTSG tubes	Incorrect - Plausible because in some cases Main Feedwater would minimize the tensile loading on the OTSG tubes. Incorrect because EFW is the preferred source of feedwater.
C.	(1) Emergency Feedwater (2) cool down the OTSG tubes faster	Correct Answer: See above.
D.	(1) Emergency Feedwater (2) minimize the tensile load on the OTSG tube	Incorrect Answer: Plausible because EFW is the preferred source. Incorrect because the stress currently exhibited on the OTSG tubes is compressive. Spraying EFW on the tubes would minimize the compressive loading on the tubes, not the tensile loading.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	061	2.4.18
	Importance Rating		4.0

K/A: Auxiliary/Emergency Feedwater: Knowledge of the specific basis for EOPs.

Proposed Question: Question #88

Technical Reference(s): OP-TM-EOP-010, Guide 13, Rev 19
OP-TM-EOP-0101, Guide 13, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP010-PCO-4

Question Source: Bank #

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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, why high cog, why SRO only)			
KA Match: This question matches the KA because the examinee must know the basis for choosing EFW to feed the OTSG.			
High Cog: The question is high cog because the examinee must determine which step of Guide 13 which applies from the conditions in the stem.			
SRO Only: This question is SRO only because it requires the examinee to assess plant conditions (Restoring from lack of heat transfer, MFW and EFW available, RCP running, positive TSDT) and select a course of action (feed with EFW because a RCP is on and TSDT is positive). Guide 13 is not a major EOP, but is directed from other major EOPs.			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

89

ID: 1737969

Points: 1.00

Plant conditions:

- 100% power with ICS in full auto.
- Backup Instrument Air Compressor IA-P-2A is tagged out of service for maintenance.
- Main Vacuum Pump VA-P-1A is operating.

Sequence of events:

Time	Event
0100	Local fire (now extinguished) renders Service Air Compressor's SA-P-1A and SA- P-1B inoperable.
0200	MAP PLB-1-6 IA-P-4 IA-Q-2 TROUBLE actuates.
0210	Instrument Air Compressor IA-P-1A trips on motor overcurrent.
0220	Instrument Air Compressor IA-P-1B trips on motor overcurrent.
0230	The following valid alarms actuate: PLB-1-7 Instrument Air Press Low Turbine Area. PLB-1-8 Station Service Air Press Low. PLB-2-7 Instrument Air Press Low Aux Bldg Area.
0240	Current Conditions: <ul style="list-style-type: none">• All Instrument and Service Air System pressure indications are now at 75 psig, reducing at 1 psi per minute.• Main Condenser vacuum is 27 inches Hg, reducing at 0.2 inches Hg per minute.• Instrument Air Compressor IA-P-4 filter and dryer d/p at 22 psid

What action is required for these conditions that will prevent entry into Technical Specification LCO 3.4, Decay Heat Removal Capability?

- A. Trip the reactor.
- B. Take manual control of MU-V-20.
- C. Open VA-V-5A vacuum pump suction valve.
- D. Ensure open IA-Q-2 bypass valve IA-V-2133 and open IA-V-2124 (Pre-filter Bypass Valve).

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) When instrument air becomes less than 80 psig, OP-TM-AOP-028, LOSS OF INSTRUMENT AIR, is entered; (2) In OP-TM-AOP-028, there are various actions taken for the lowering levels for air pressures or high level of differential pressures; (3) The parameter exceeded for this scenario is high D/P for IA-P-4 filter and dryer (> 20 psid); (4) The Turbine Bypass Valves, (TBVs) MS-V-3A-F usually receive air from the instrument air header, and have backup air from the 'A' Backup Instrument Air Compressor, IA-P-2A; (5) Since IA-P-2A is OOS in the stem, and with the various events occurrence to the other compressors, when the IA-P-4 filter and dryer DP becomes > 22 psid, the MS-V-3's become inoperable; (6) One of the purposes of the TBVs is to remove decay heat when the reactor is shutdown, and 4 of 6 must be operable per technical specification 3.4.1.1.			
A.	Trip the reactor.	INCORRECT: Manual reactor trip at 60 psig is a required action to in order to avoid potential loss of control or erratic behavior of air operated components at significant power levels. Distracter is plausible because this answer represents a valid IAAT action at less than 60 psig, and it does result in reduction of core heat production by decay of fission products. In addition, the examinee could believe that tripping the reactor is an option to not enter a tech spec, as some tech specs only apply when the reactor is critical.	
B.	Take manual control of MU-V-20.	INCORRECT: The purpose of this action is to prevent loss of reactor coolant pump seal injection flow. Distracter is plausible because this answer represents a valid manual action included in AOP-028. In addition, MU-V-20 is a containment isolation valve, the examinee could believe that MU-V-20 fails open (it does not) which would represent a loss of containment and potentially an entry into the containment tech spec.	
C.	Open VA-V-5A vacuum pump suction valve.	INCORRECT: Plausible because the examinee could believe that vacuum is required for the TBV's to be operable. Distracter is plausible because VA-V-5A/B/C all fail closed on loss of instrument air.	
D.	Ensure open IA-Q-2 bypass valve IA-V-2133 and open IA-V-2124 (Pre-filter Bypass Valve).	Correct Answer: IA-P-4 is the last air compressor available that could provide air pressure to the turbine bypass valves. In addition to all of the other failures, if the dryer and filter were clogged, they could be bypassed to provide air to the air header.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	078	A2.01
	Importance Rating		2.9
K/A: Instrument Air: Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Air dryer and filter malfunctions			
Proposed Question: Question #89			
Technical Reference(s): OP-TM-AOP-028, Rev 9			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Tech Spec 3.4.1.1

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-028-PCO-4

Question Source: Bank # 371312

Modified Bank #

New

Question History: Comp 1 Last NRC Exam:

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 which parameter is out of specification due to an instrument air failure and determine the procedure step which will remedy the situation.

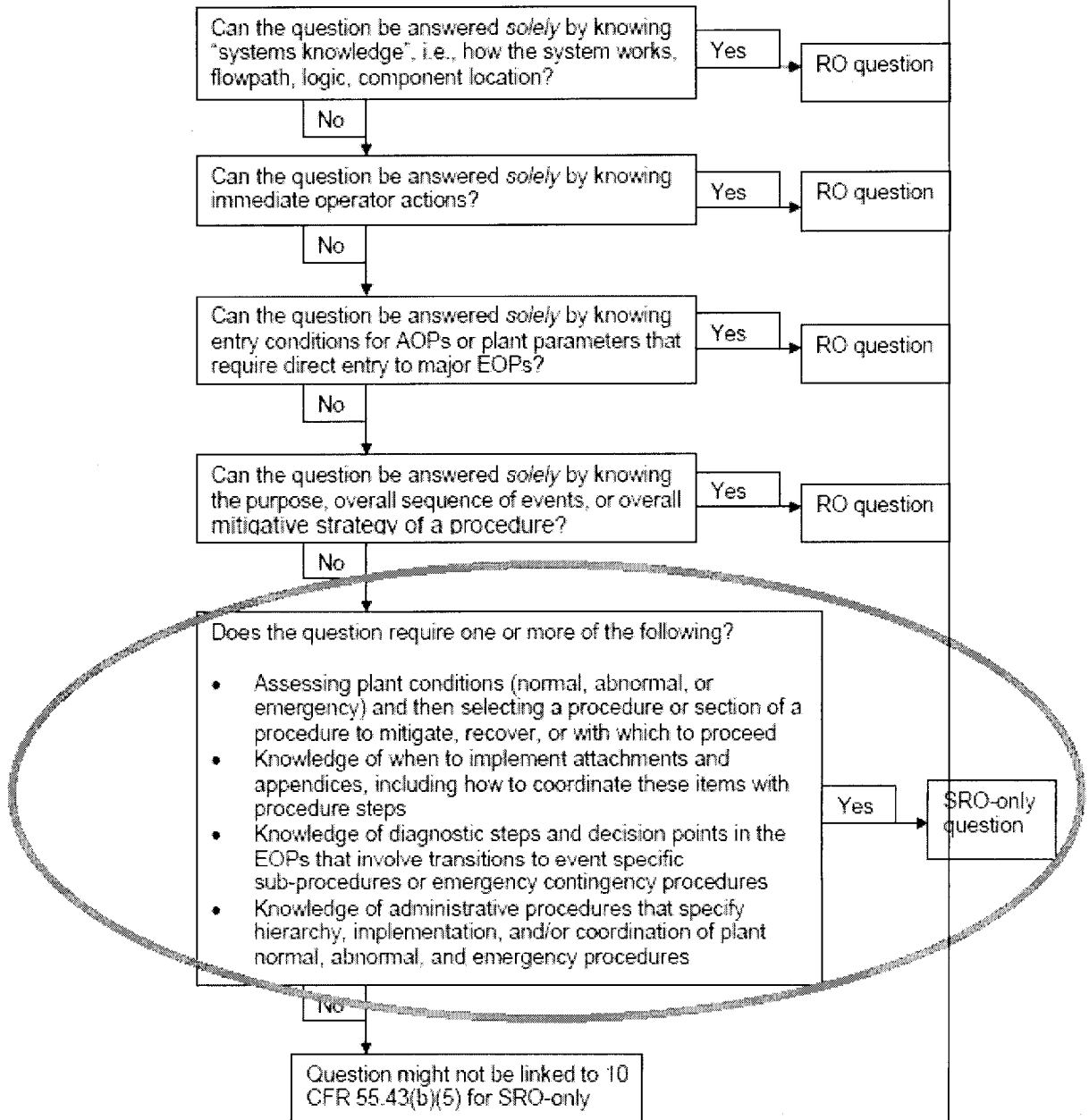
High Cog: This question is high cog because the examinee analyze the current parameters and trends of the instrument air system and make a decision on the correct course of action to ensure the plant does not enter a tech spec LCO.

SRO Only: This is an SRO only question because the examinee must assess plant conditions then select the appropriate procedure to mitigate the malfunctions. In addition, the SRO must know the content of the procedure in which to evaluate the stems parameters with, and choose the correct course of actions based on the assessment.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



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EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

90

ID: 1737999

Points: 1.00

Plant Conditions:

- The plant is shut down in a Refueling Outage.
- Core off-load in progress.
- Containment Integrity is NOT being maintained.
- RB Purge is in progress IAW OP-TM-823-408 RB Purge - RB Doors and/or Equipment Hatch Open.

Event:

- The Fuel Handling SRO reports that a Fuel Assembly has been dropped and damaged in the Fuel Transfer Canal.
- MAP C-1-1 RADIATION LEVEL HI, alarms.
- RB Purge Exhaust Duct Monitor RM-A-9 is off-scale high.
- RB Purge Exhaust Duct Monitor RM-A-9 Hi-Hi is 3000 CPM.
- NI-11 counts have gone up and steadied out at 30 cps.

In accordance with OP-TM-MAP-C0101, all personnel are ____ (1) ____ to be evacuated from the reactor building, and the RB Purge must be secured in accordance with ____ (2) ____.

- A. (1) required
(2) OP-TM-244-911, CONTAINMENT CLOSURE
- B. (1) required
(2) OP-TM-244-901, CONTAINMENT ISOLATION
- C. (1) NOT required
(2) OP-TM-244-911, CONTAINMENT CLOSURE
- D. (1) NOT required
(2) OP-TM-244-901, CONTAINMENT ISOLATION

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) During fuel handling, the containment is allowed to be open as long a reactor building purge is in service with the RB Purge Valve interlocks in defeat (in addition to other requirements for closing the reactor building which are not pertinent to the question); (2) When the fuel assembly is dropped in the fuel transfer canal gasses are released, which cause RM-A-9 to alarm, which would cause MAP C-1-1 alarm in the control room; (3) The alarm response for RM-A-9 directs the evacuation of ALL personnel from the containment building, and to initiate OP-TM-244-911, CONTAINMENT CLOSURE; (4) OP-TM-244-901 directs the closing of AH-V-1A, B, C, and D (RB purge valves) and to ensure containment closure devices outside of containment are set; (5) The NI-11 counts going up and steadying out at 30 cps is to enhance the plausibility of a distractor and for a better match of the KA as there would be an automatic RB evacuation alarm at 40 cps.			
A.	(1) required (2) OP-TM-244-911, CONTAINMENT CLOSURE	Correct Answer - See Above	
B.	(1) required (2) OP-TM-244-901, CONTAINMENT ISOLATION	INCORRECT - Plausible because OP-TM-823-408 and 1101-3, CONTAINMENT INTEGRITY AND ACCESS LIMITS directs the purge interlocks to close the purge valves to be defeated. The examinee must know that since the interlocks are defeated that the purge will not secure. If the examinee believes that the purge should have been secured, they examinee could believe that the CONTAINMENT ISOLATION procedure would direct closing the AH-V-1 valves. The CONTAINMENT ISOLATION procedure does include steps which close these valves, but it is used if a containment isolation valve fails to close from an ES signal (1600 psig, 500 psig, and/or 4 psig), which is not present in this question.	
C.	(1) NOT required (2) OP-TM-244-911, CONTAINMENT CLOSURE	INCORRECT - Plausible because NI-11 never gets over 40 cps, which would initiate an automatic RB Evacuation alarm. Not going over 40 cps is plausible because the fuel assembly was dropped in the fuel transfer canal, and not the reactor vessel.	
D.	(1) NOT required (2) OP-TM-244-901, CONTAINMENT ISOLATION	INCORRECT - Plausible because NI-11 never gets over 40 cps, which would initiate an automatic RB Evacuation alarm. Not going over 40 cps is plausible because the fuel assembly was dropped in the fuel transfer canal, and not the reactor vessel. In addition, plausible because OP-TM-823-408 and 1101-3, CONTAINMENT INTEGRITY AND ACCESS LIMITS directs the purge interlocks to close the purge valves to be defeated. The examinee must know that since the interlocks are defeated that the purge will not secure. If the examinee believes that the purge should have been secured, they examinee could believe that the CONTAINMENT ISOLATION procedure would direct closing the AH-V-1 valves. The CONTAINMENT ISOLATION procedure does include steps which close these valves, but it is used if a containment isolation valve fails to close from an ES signal (1600 psig, 500 psig, and/or 4 psig), which is not present in this question.	
Examination Outline Cross-reference:			
Level		RO	SRO
Tier #			2
Group #			1
K/A #		103	A2.04
Importance Rating			3.6

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 the procedures to correct, control, or mitigate the consequences of those malfunctions or operations

Proposed Question: Question #90

Technical Reference(s): 1101-3, Rev 93A OP-TM-MAP-C0101, Rev 3
OP-TM-244-911, Rev 4 OP-TM-823-408, Rev 12

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-5

Question Source: Bank #
Modified Bank # 1110308
New

Question History: Sim Exam 9 Last NRC Exam: Unmodified on 10-02

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 predict what the impact of a dropped fuel assembly will be on the operations going on in the reactor building during refueling. From those predictions, the examinee must know the content of the containment closure procedure (OP-TM-244-901) to know that the RB purge valves get closed.

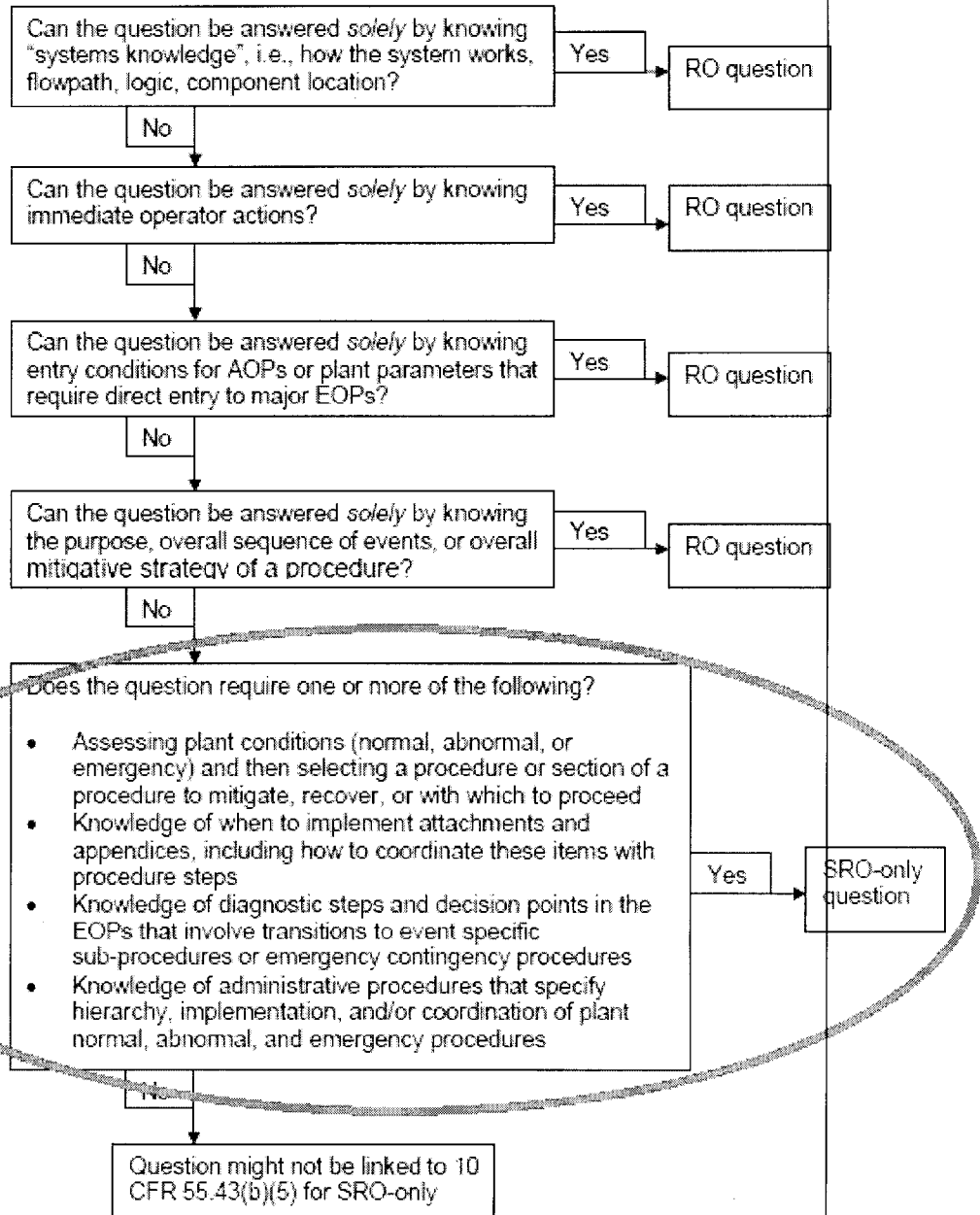
High Cog: This question is high cog because the examinee must know that an RB evacuation is required from the stem.

SRO Only: This question is SRO only because the examinee must assess plant conditions know the content of a procedure to ensure low pressure containment integrity is maintained.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



Plant Conditions:

- The plant is shut down in a Refueling Outage.
- Core off-load in progress.
- Containment Integrity is NOT being maintained.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

- RB Purge is in progress IAW OP-TM-823-408 RB Purge - RB Doors and/or Equipment Hatch Open.

Event:

- The Fuel Handling SRO reports that a Fuel Assembly has been dropped and damaged in the Refueling cavity.
- MAP C-1-1 RADIATION LEVEL HI , alarms.
- RB Purge Exhaust Duct Monitor RM-A-9 is off-scale high.
- RB Purge Exhaust Duct Monitor RM-A-9 Hi-Hi is 3000 CPM.

Assuming no action other than the initiation of a Containment evacuation has taken place, which ONE (1) of the following identifies the current positions of RB Purge Isolation Valves (AH-V-1A, 1B, 1C and 1D), AND the reason for their position?

- A. Open;
The RB Purge Line Isolation High Radiation Interlock is defeated.
- B. Closed;
The RB Purge Line Isolation High Radiation Interlock was operated by RM-A-9G when it went above the ALERT level.
- C. Closed;
The RB Purge Line Isolation High Radiation Interlock was operated by RM-A-9 Hi-Hi when it went above the ALERT level.
- D. Open;
The RB Purge Line Isolation High Radiation Interlock has not operated because RM-A-9 Hi-Hi has not exceeded the HIGH Alarm level.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

91

ID: 1700311

Points: 1.00

Plant conditions:

- The crew is performing a plant startup per 1102-02, Plant Startup.
- Deboration to the Target Critical Boron Concentration has just been completed.
- The crew is awaiting Plant Manager authorization to startup the reactor.

EVENT:

A check of the Backup Incore Thermocouple Readout (BIRO) reveals that the following Thermocouples have failed:

4-E
5-G
7-B
13-C

Given these current conditions, which ONE of the following identifies whether the LCO for Tech Spec 3.5.5, Accident Monitoring Instrumentation, is currently satisfied, and if not, also identifies the MINIMUM actions required to satisfy the LCO without reliance on any action statement?

- A. LCO 3.5.5 is met;
No action statements are required to be entered.
- B. LCO 3.5.5 is NOT met; AND
Performing maintenance and declaring Thermocouple 13-C OPERABLE will allow all actions statements to be exited.
- C. LCO 3.5.5 is NOT met; AND
Performing maintenance and declaring Thermocouple 4-E OPERABLE will allow all actions statements to be exited.
- D. LCO 3.5.5 is NOT met; AND
Performing maintenance and declaring Thermocouple 4-E and 7-B OPERABLE will allow all actions statements to be exited.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

****Needs 1105-5 Table 2 as a reference			
Explanation: To answer this question correctly, the examinee must know: (1) That 2 of the 4 instruments per quadrant must be operable or the action statement of Tech Spec 3.5.5 must be entered (2) The condition of the plant is one where the Tech Spec 3.5.5 is applicable (3) How to read Table 2 of 1105-5 to verify that 3 in a quadrant are failed.			
A.	LCO 3.5.5 is met; No action statements are required to be entered.	This is plausible because many LCOs are written to be applicable when the reactor is critical. Since it is not, the operator may incorrectly believe that thermocouples do NOT need to be OPERABLE to satisfy the LCO.	
B.	LCO 3.5.5 is NOT met; AND Performing maintenance and declaring Thermocouple 13-C OPERABLE will allow all actions statements to be exited.	This is plausible because the operator may incorrectly believe that thermocouple 13-C is part of the same Quadrant as two of the others. The provided reference allows the student to map the BIROs and determine that 3 out of four BIROs in one quadrant are not operable therefore the TS LCO is not met and that 13-C is not one of those three. Therefore, performing maintenance on this thermocouple would still leave the operator with three inoperable thermocouples in a quadrant.	
C.	LCO 3.5.5 is NOT met; AND Performing maintenance and declaring Thermocouple 4-E OPERABLE will allow all actions statements to be exited.	According to Technical Specification LCO 3.5.5 the instruments identified in Table 3.5-2 and Table 3.5-3 during STARTUP, POWER OPERATION and HOT STANDBY shall be OPERABLE. According to Technical Specification LCO 3.5.5, there are four Backup Incore Thermocouple Display (BIRO) channels/ per quadrant, and two of them must be OPERABLE. According to Technical Specification 1.0, the reactor shall be considered in the STARTUP mode when the shutdown margin is reduced with the intent of going critical. Since the plant has been heated up to beyond 250°F under the stated conditions, and the Boron concentration has been reduced to Target Critical Boron Concentration, the plant is in the STARTUP mode, and the LCO is applicable. This is supported by the System Operating Procedure. According to 1105-5 when RCS avg. temp is >250°F, the Backup Incore Thermocouple Display Channel (BIRO) must be operable with a minimum of 2 thermocouples per core quadrant in operation. Using the provided reference and generating a core map, the operator can determine that 4-E, 5-G, 7-B and 13-C are in two different quadrants and three (4-E, 5-G and 7-B) are in the same quadrant. Since a minimum of two are required to be OPERABLE in each quadrant, the operator can determine that one quadrant has only one OPERABLE thermocouple; 8-F. This means that LCO 3.5.5 is NOT being met. The operator can meet LCO 3.5.5 by performing maintenance and declaring Thermocouple 4-E OPERABLE.	
D.	LCO 3.5.5 is NOT met; AND Performing maintenance and declaring Thermocouple 4-E and 7-B OPERABLE will allow all actions statements to be exited.	This is plausible because the operator may incorrectly believe that only one thermocouple per quadrant is allowed to be inoperable. While performing maintenance on both thermocouples will render the LCO met, it can be met with less maintenance, by performing maintenance only on thermocouple 4-E; and therefore, it is NOT the minimum action necessary to meet the LCO.	
Examination Outline Cross-reference:			
Level		RO	SRO
Tier #			2

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Group #	2	
K/A #	017	2.4.3
Importance Rating	3.7	3.9

K/A: In-Core Temperature Monitor: Ability to identify post-accident instrumentation

Proposed Question: Question #91

Technical Reference(s): 1105-5, Rev 25
TS 3.5.5, Table 3.5-2

Proposed References to be provided to applicants during examination: 1105-5, Table 2

Learning Objective: 625-GLO-14

Question Source: Bank # 860160
Modified Bank #
New

Question History: Comp 2 Last NRC Exam:

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

10 CFR Part 55 Content: 55.41
55.43 b.2

Comments:

K/A Match: This question matches the K/A because the examinee must know how many thermocouples are required to be operable to meet the TS 3.5.5, which is for post accident monitoring.

The question is at the Comprehension/Analysis cognitive level because the operator must consider several pieces of information such as (1) the LCO governing the BIRO, (2) the condition of the plant and whether or not the LCO is applicable under the present conditions, and (3) the specific instrument failures; and then draw a conclusion regarding the status of the LCO (and the maintenance needed to satisfy it).

The question is SRO-Only because the operator must demonstrate the ability to comply with the plant Technical Specifications including knowing procedurally required minimum requirements for instrument operability .

EXAMINATION ANSWER KEY

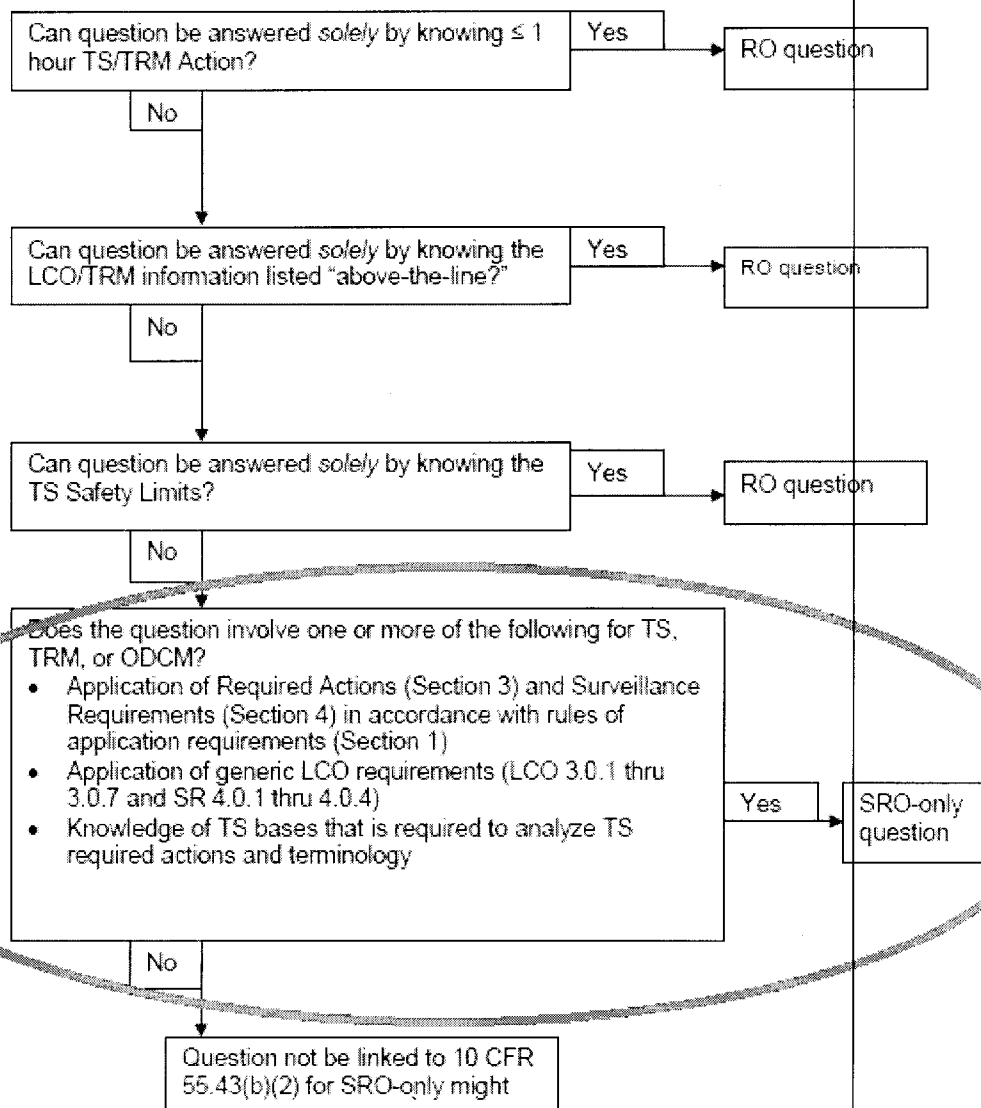
16-01 NRC EXAM SUBMITTAL

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Attachment 2

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

92

ID: 1736504

Points: 1.00

Plant conditions:

- 100% power with ICS in full Auto.
- Condensate Pump CO-P-1C is tagged out for maintenance.

EVENT:

- The Secondary NLO reports that vibrations on Condensate Pump CO-P-1A have increased since his last round and a high pitched noise seems to be coming from the lower motor bearing.
- Maintenance confirms that the lower bearing is failing and recommends CO-P-1A be stopped within the next hour to avoid shaft damage.

The operating crew must reduce power to ____ (1) ____, and secure CO-P-1A in accordance with ____ (2) ____.

- A. (1) < 665 MWe
(2) 1102-4, POWER OPERATIONS
- B. (1) < 665 MWe
(2) OP-TM-421-430, REMOVING CO-P-1A FROM SERVICE
- C. (1) < 50% power
(2) 1102-4, POWER OPERATIONS
- D. (1) < 50% power
(2) OP-TM-421-430, REMOVING CO-P-1A FROM SERVICE

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) Due to CO-P-1C being out of service, the operating crew must down power the plant in accordance with 1102-4; (2) Also, due to CO-P-1C being out of service, the crew must secure a Main Feedwater Pump and a Condensate Booster Pump prior to securing the Condensate Pump; (3) In accordance with 1102-4, the operating crew could secure the Main Feedpump, Condensate Booster Pump, and the Condensate Pump at less than < 560 MWe, and prior to a Condensate Booster Pump (CO-P-2) discharge pressure exceeding 675 psig; (4) 560 MWe is approximately 68% power; (5) A precaution in OP-TM-421-000 limits operations with only one Condensate Pump (with only one Condensate Booster Pump, and Main Feedwater Pump) to less than 50% power to avoid a Reactor Trip.			
A.	(1) < 665 MWe (2) 1102-4, POWER OPERATIONS	INCORRECT: Plausible if the examinee believes that the runback setpoint for loss of a Feedwater Pump is 665 MWe. This is incorrect because the plant would run back to less than 560 MWe, in addition this power level would be to high for sustained operation with a single condensate pump in operation.	
B.	(1) < 665 MWe (2) OP-TM-421-430, REMOVING CO-P-1A FROM SERVICE	INCORRECT: Plausible if the examinee believes that the runback setpoint for loss of a Feedwater Pump is 665 MWe. This is incorrect because the plant would run back to less than 560 MWe, in addition this power level would be to high for sustained operation with a single condensate pump in operation.	
C.	(1) < 50% power (2) 1102-4, POWER OPERATIONS	Correct Answer: See above.	
D.	(1) < 50% power (2) OP-TM-421-430, REMOVING CO-P-1A FROM SERVICE	INCORRECT: Plausible because the examinee could believe OP-TM-421-430 contains the correct guidance on securing CO-P-1A. Incorrect because this procedure would start the standby Condensate Pump first, which is out of service in the stem. The guidance to secure the Main Feedwater Pump, Condensate Booster Pump, and Condensate Pump are in 1102-4.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	056	A2.04
	Importance Rating		2.6
K/A: Condensate: Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System, and (b) based on those predictions, use procedures to correct, control, of mitigate the consequences of those malfunctions or operations: Loss of condensate pumps.			
Proposed Question: Question #92			
Technical Reference(s): OP-TM-421-000, Rev 16			
1102-4, Rev 133			
Proposed References to be provided to applicants during examination: None			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Learning Objective: 421-GLO-10

Question Source: Bank # 572934

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

55.43 b.5

Comments:

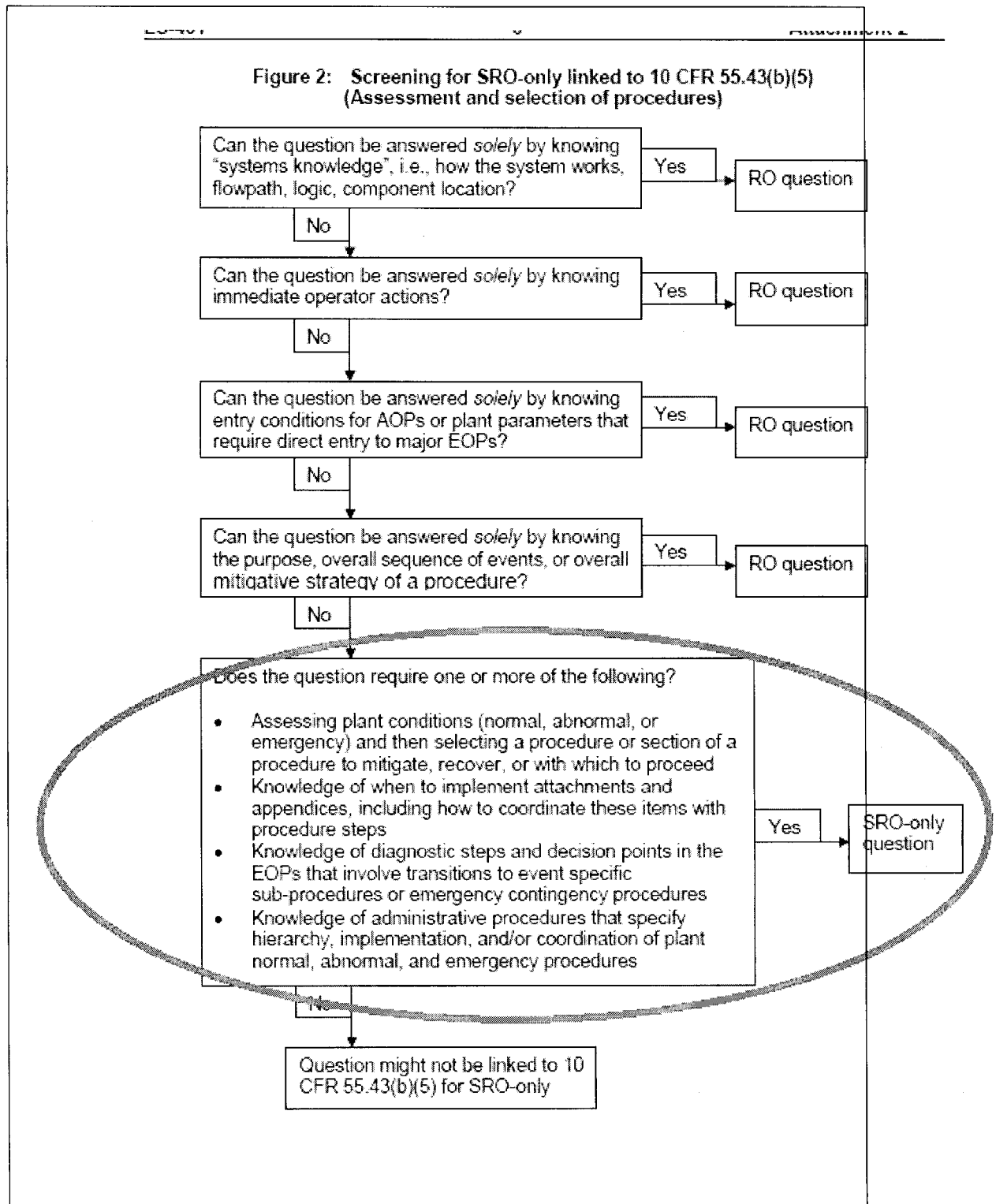
KA Match: This question matches the KA because the examinee has to understand the impact of losing a second condensate would be to the plant. The examinee must also determine the correct power and procedure that the operators would use to lower power.

High cog: This question is High Cog because the examinee must know that the plant power must be lowered if another condensate pump is lost. In addition, the examinee must pick the correct procedure to lower power.

SRO Only: This question requires the examinee to assess the plant conditions/equipment and to know the content of procedures in order to select the required course of action.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

93

ID: 1736533

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.
- An approved radioactive liquid release is in progress.

EVENT:

- Annunciator C-1-1 "RADIATION LEVEL HI" is in alarm.
- Radiation Monitor RM-L-6, RAD WASTE DISCHARGE, has failed (pegged high, off-scale).
- WDL-P-14A/B discharge valve to the MDCT, WDL-V-257, is stuck open and cannot be closed.

Identify the one selection below that describes:

(1) The correct action, and

(2) The timeclock associated with RM-L-6.

- A. (1) Terminate the release and notify Radiation Protection.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 14 days.
- B. (1) Terminate the release and notify Radiation Protection.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 30 days.
- C. (1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 14 days.
- D. (1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 30 days.

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<p><<Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-TM-MAP-C0101, when RM-L-6 alarm is received (in this case for RM-L-6 failing high), WDL-V-257 should close; (2) When the valve (WDL-V-257) does not close, and because RM-L-6 is failed high, the examinee should determine that RM-L-6 is inoperable; (3) OP-TM-MAP-C0101 directs the termination of the release and the notification of radiation protection; (4) The timeclock to return RM-L-6 is found in CY-TM-170-300, OFFSITE DOSE CALCULATION MANUAL (ODCM) step 2.1.1, where with less than the number OPERABLE best efforts to return the instrumentation to OPERABLE within 30 days, if not explain in the next Annual Effluent Release Report.</p>																							
A.	(1) Terminate the release and notify Radiation Protection. (2) Exert best efforts to return RM-L-6 to OPERABLE within 14 days.	INCORRECT - Plausible because many ODCM timeclocks are 14 days. Incorrect because the timeclock from RM-L-6 is 30 days.																					
B.	(1) Terminate the release and notify Radiation Protection. (2) Exert best efforts to return RM-L-6 to OPERABLE within 30 days.	Correct Answer - See above.																					
C.	(1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations. (2) Exert best efforts to return RM-L-6 to OPERABLE within 14 days.	INCORRECT - Plausible because this is the action from OP-TM-MAP-C0101 if RM-L-7 (Plant Discharge Rad Monitor) high alarm is in. In addition, the timeclock is 30 days, not 14 days.																					
D.	(1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations. (2) Exert best efforts to return RM-L-6 to OPERABLE within 30 days.	INCORRECT T - Plausible because this is the action from OP-TM-MAP-C0101 if RM-L-7 (Plant Discharge Rad Monitor) high alarm is in.																					
<table border="1"> <tr> <td>Examination Outline Cross-reference:</td> <td>Level</td> <td>RO</td> <td>SRO</td> </tr> <tr> <td></td> <td>Tier #</td> <td></td> <td>2</td> </tr> <tr> <td></td> <td>Group #</td> <td></td> <td>2</td> </tr> <tr> <td></td> <td>K/A #</td> <td>068</td> <td>A2.04</td> </tr> <tr> <td></td> <td>Importance Rating</td> <td></td> <td>3.3</td> </tr> </table>				Examination Outline Cross-reference:	Level	RO	SRO		Tier #		2		Group #		2		K/A #	068	A2.04		Importance Rating		3.3
Examination Outline Cross-reference:	Level	RO	SRO																				
	Tier #		2																				
	Group #		2																				
	K/A #	068	A2.04																				
	Importance Rating		3.3																				
<p>K/A: Liquid Radwaste: Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation.</p>																							
<p>Proposed Question: Question #93</p>																							

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Technical Reference(s): OP-TM-MAP-C0101, Rev 3

CY-TM-170-300, Rev 4

Proposed References to be provided to applicants during examination: None

Learning Objective: 232-GLO-14

Question Source: Bank # 1102535

Modified Bank #

New

Question History: System Exam 14 Last NRC Exam: 10-02
NRC

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 that if RM-L-6 fails high, that the release must be terminated and RP notified. To mitigate the failure, the examinee must know that the site has 30 days to repair the detector or the reason for the extended inoperability must be included in the next Annual Effluent Release Report.

SRO only: This question is SRO only because the examinee must know how to apply the action of the ODCM for RM-L-6 inoperability.

EXAMINATION ANSWER KEY

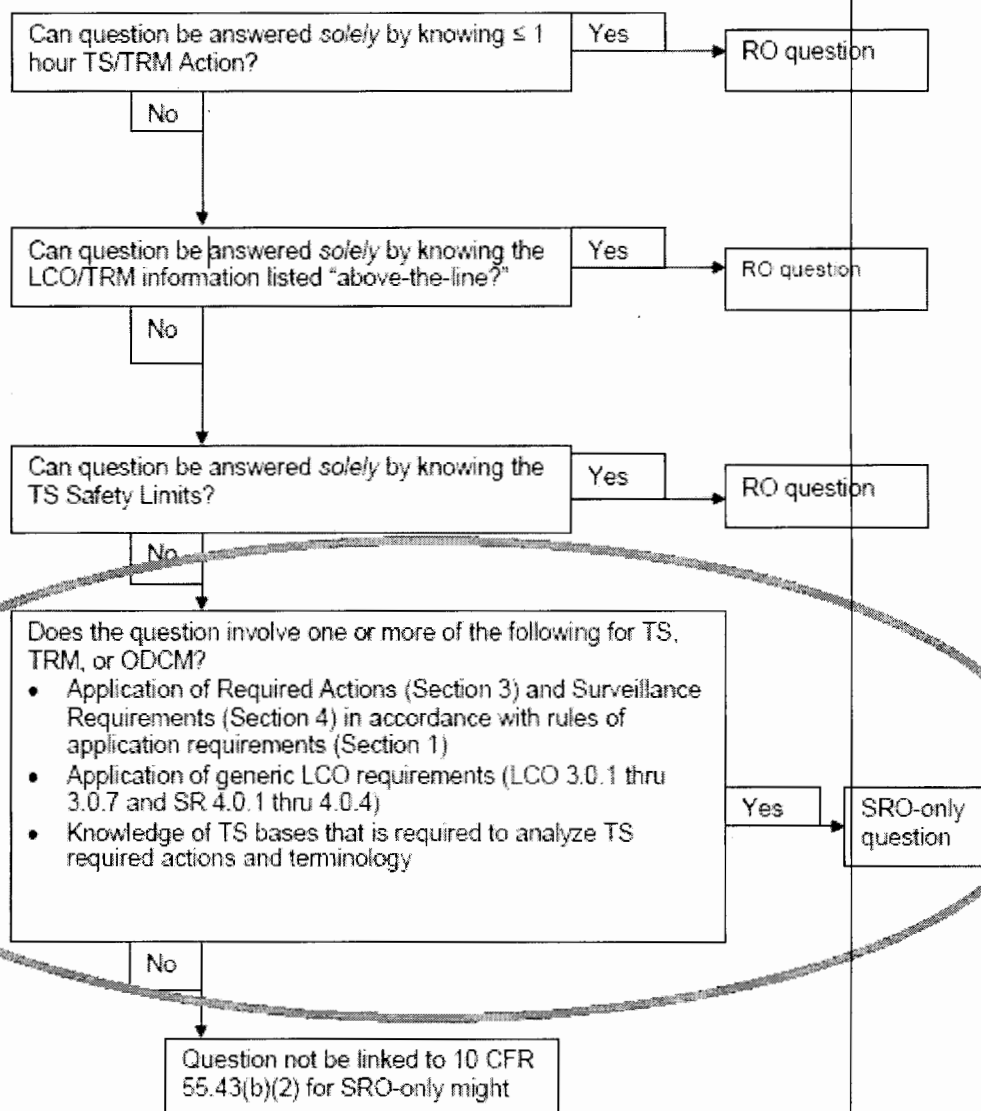
16-01 NRC EXAM SUBMITTAL

ES-401

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Attachment 2

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

94

ID: 1700144

Points: 1.00

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- MAP E-3-4, BWST TEMP/LVL HI/LO alarm is received due to a small leak at the base of BWST.
- ALL BWST inventory is lost through the leak.

Per Technical Specifications, which of the following is true regarding makeup tank level and pressure?

Extended operation inside the ____ (1) ____ is not permitted due to prevention of ____ (2) ____.

- A. (1) Restricted Region
(2) a loss of NPSH when 2 Makeup Pumps are running.
- B. (1) Low NPSH region
(2) a loss of NPSH when 2 Makeup Pumps are running.
- C. (1) Restricted Region
(2) gas entrainment into the makeup pumps in the event of an emergency injection on a Large Break LOCA.
- D. (1) Low NPSH region
(2) gas entrainment into the makeup pumps in the event of an emergency injection on a Large Break LOCA.

Answer: C

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) Operation in the LOW NPSH Region of Attachment 7.3 of OP-TM-211-000 is prohibited to ensure that adequate MU pump NPSH during the limiting design basis event; (2) Operation is not permitted in the "Prohibited" or "Restricted" Region to prevent gas entrainment into the makeup pumps in the event of emergency injection on a Large Break LOCA. The ensures that the MU tank gas bubble remains in the MU tank as long as the BWST level is above minimum.			
A.	(1) Restricted Region (2) a loss of NPSH when 2 makeup pumps are running.	Plausible because the restricted region is in a area where low makeup tank level and pressure are achievable (i.e 50" in the MU tank and 15 psig) hence the examinee could believe the restricted region area of concern is NPSH. Incorrect because with the BWST lowering, operation in the restricted and prohibited region is the primary concern. Due to a lower BWST level, the differential pressure between it and the makeup tank and is lower. The casualty which makes this become an issue is a Large Break LOCA, which requires inventory supplied by the makeup pumps. At a lower differential pressure, and with potentially all three makeup pumps running, the suction source could be either the makeup tank or the BWST (depending on which has the higher pressure). If the conditions existed where the makeup tank has the higher pressure than the BWST in conjunction with a low MU tank level, gas could enter the makeup pump suction line making all three makeup pumps inoperable.	
B.	(1) Low NPSH region (2) a loss of NPSH when 2 makeup pumps are running.	Plausible because this is the correct basis for prevention of operation in the low NPSH region. Incorrect because with a lowering BWST level, operation in the prohibited region is of concern.	
C.	(1) Restricted Region (2) gas entrainment into the makeup pumps in the event of an emergency injection on a Large Break LOCA.	Correct Answer.	
D.	(1) Low NPSH region (2) gas entrainment into the makeup pumps in the event of an emergency injection on a Large Break LOCA.	Plausible because the examinee could believe that with a Low NPSH, that gas entrainment is of concern during and emergency injection on a LOCA is a concern.	
Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #		2.1.32
	Importance Rating		4.0
K/A: Ability to explain and apply system limitations and precautions.			
Proposed Question: Question #94			

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Technical Reference(s): OP-TM-211-000, Rev 33

T.S. 3.3 basis

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO-6

Question Source: Bank #

Modified Bank #

New X

Question History: N/A

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:

K/A match: This is a K/A match because it is a precaution and limitation of the makeup system (OP-TM-211-000).

SRO Only: This is SRO only because the examinee must identify and recall the basis behind the MU tank pressure and level curves described in T.S. 3.3.1.1 g.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

95

ID: 1700178

Points: 1.00

Plant conditions:

- The plant is in refueling outage with fuel handling operations in the RB in progress
- The Main Bridge Operator has lowered an assembly that is NOT an anti-straddle type assembly into the core location with open water on one side only.
- The ZZ tape reading for the assembly is approximately 2" high with full down load cell indication

With the above indications the assembly is resting on the _____.

- A. grid of an adjacent assembly and can be shaken while moving through the grid region as the assembly is lowered
- B. grid of an adjacent assembly and the mast can be rotated to reposition the assembly prior to lowering
- C. reactor lower grid and can be moved slightly toward the open side and lowered
- D. reactor lower grid and the assembly can be raised and jog can be used to reposition the assembly prior to lowering

Answer: D

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: To answer this question correctly, the examinee must know: (1) the likely cause (and the only possible cause as determined by the stem and distractors) is that the assembly is resting on the reactor lower grid; (2) since the assembly is not boxed in and is NOT the anti-straddle design, the assembly should NOT be moved toward the open location; (3) The four allowable actions are to: Raise the F/A until full weight and try to re-lower, use jog to reposition the F/A, rotate mast or flow mast by pulling deten pin as required, shake the fuel hoist cable.			
A.	grid of an adjacent assembly and can be shaken while moving through the grid region as the assembly is lowered	Plausible since shaking a fuel assembly while moving is allowed; however not through the grid region.	
B.	grid of an adjacent assembly and the mast can be rotated to reposition the assembly prior to lowering	Plausible since rotating the mast is allowed; however the assembly is not resting on an adjacent assembly grid it is resting on the reactor lower grid.	
C.	reactor lower grid and can be moved slightly toward the open side and lowered	Plausible since the assembly is resting on the reactor lower grid; however moving the assembly toward the open area is not allowed.	
D.	reactor lower grid and the assembly can be raised and jog can be used to reposition the assembly prior to lowering	Correct answer. The assembly is on the reactor lower grid and can be raised and jog can be used to realign the assembly IAW section 7.4 of 1505-3.	
Examination Outline Cross-reference:			
Level		RO	SRO
Tier #			3
Group #			1
K/A #			2.1.42
Importance Rating			3.4
K/A: Knowledge of new and spent fuel movement procedures.			
Proposed Question: Question #95			
Technical Reference(s): 1505-3, Rev 23			
Proposed References to be provided to applicants during examination: None			
Learning Objective: N/A			
Question Source: Bank # 718417			
Modified Bank #			
New			

16-01 NRC EXAM SUBMITTAL

SRO Only: This question is SRO only because it is the responsibility of the Fuel Handling Supervisor to attempt to direct the reseal the fuel assembly by one of the approved methods in the procedure (1505-3).

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

96

ID: 1700737

Points: 1.00

Plant Conditions:

- Your shift has just assumed the watch in the Control Room.
- The Special Test Coordinator (a designee of the Senior Line Manager) has just commenced conducting a brief for Turbine Torsional Testing, which is scheduled to take place this shift.
- Due to one of the maintenance team members calling out sick, the testing group is short-handed.
- The Special Test Coordinator is briefing the following actions:
 - The Special Test Coordinator will assume the active role of the performer who called out sick.
 - Just in Time training is not required because the evolution was performed by the same team last time, so the members understand the actions required.
 - Station Management shall be present during the entire test.
 - The test is scheduled to take more than one shift and another brief will be required upon turnover.

In accordance with OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTIONS, which one of the following is a true statement with regards to the actions that were briefed?

- A. The Special Test Coordinator can not be an active performer of the test.
- B. Station Management does not need to be present during the special test.
- C. Only a Senior Line Manager can conduct the briefing for the special test.
- D. Just in Time training must occur to ensure the test is understood by those involved.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) A caution in section 4.3 of OP-AA-108-110 states the Test Coordinator is not an active performer of the test. They provide oversight for the test activities. They are coordinators and the interface with Operations during the test.

A.	The Special Test Coordinator can not be an active performer of the test.	Correct Answer
B.	Station Management does not need to be present during the special test.	Incorrect - Step 4.3.5 requires Senior Line Management and the Test Coordinator to be present prior to and during the special test or evolution performance. This distractor is plausible because with most complex evolutions, the Station Management only has to be present for the prejob brief.
C.	Only a Senior Line Manager can conduct the briefing for the special test.	Incorrect - Step 4.3.3 - The SLM shall ENSURE an IPA briefing is conducted in accordance with HU-AA-1211 prior to performing the special test or evolution. This does NOT mean the that SLM must conduct the brief. This is plausible because the examinee could believe the SLM must conduct the brief.
D.	Just in Time training must occur to ensure the test is understood by those involved.	Incorrect - Step 4.3 - Consider Just in Time training to ENSURE plant conditions and the activity are understood by those involved. In addition OP-AA-101-111-1001 states JIT must be considered only. JIT training is not required to be performed. Plausible because the examinee could believe it is required by this procedure of OP-AA-101-111-1001.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #		2.2.7
	Importance Rating		3.6

K/A: Knowledge of the process for conducting special or infrequent tests.

Proposed Question: Question #96

Technical Reference(s): OP-AA-108-110, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-DBIG-PCO-3

Question Source: Bank # 742715

Modified Bank #

New

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question History:	Sim Exam 5	Last 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 matches the KA because the examinee will have to have knowledge of OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTION procedure.		
SRO Only: This is an SRO only question because the question tests the examinees knowledge of a procedure that has the purpose of providing a mechanism for evaluation of infrequently performed, complex tasks to implement special administrative management controls. The examinee will have to understand the hierarchy and roles of personnel during the infrequently performed, complex task.		

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

97

ID: 1700778

Points: 1.00

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 a 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

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

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 pump will not degrade further, and that the flow is greater than the ASME value, this does not make the pump anything other than INOPERABLE because it is still lower than the minimum flow rate allowed by technical specifications.

A.	Operable	Incorrect but plausible because the definition of OPERABLE is the SSC is capable of performing its specified function. In addition, there is clarification that the deficiency which caused the equipment to be inoperable has been resolved. The stem states that evaluations have been completed that there will be no further degradation, this may imply, but does not explicitly say that the deficiency is resolved.
B.	Degraded	Incorrect, but 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 can still meet ASME value, and will not degrade further, but since is below that technical specification value that the SSC is degraded.
C.	Inoperable	Correct Answer
D.	Unavailable	Plausible because if this were a 2 pump system, this malfunction would be tracked under for unavailability. Incorrect because when NR-P-1A failed the T.S surveillance, NR-P-1B would have been ES selected on the 1T 480V bus.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #		2.2.21
	Importance Rating		4.1

K/A: Knowledge of pre- and post-maintenance operability requirements

Proposed Question: SRO Question #22

Technical Reference(s): OP-AA-108-104
ER-TM-310-1001

Proposed References to be provided to applicants during examination: None

Learning Objective: 108104-APCO-1

Question Source: Bank # 746855
Modified Bank #

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

New

Question History: Sim Exam 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.2

Comments: (KA Match, why high cog, why SRO only)

K/A Match: This question matches the KA because the examinee must make the determination that NR-P-1A is inoperable even though the flow rate is greater than the ASME code and engineer has determined no further degradation will occur.

SRO Only: This question is SRO only because the examinee must determine the operability of an ES pump based on the data given.

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

98

ID: 1700579

Points: 1.00

In accordance with CY-TM-170-2001, RELEASING RADIOACTIVE LIQUID WASTE, who approves the Liquid Permit Pre-Release for the release of a WECST if total radioactivity (excluding tritium and noble gases) is below $7E-7$ uCi/ml?

- A. Shift Manager
- B. Radwaste Supervisor
- C. Unit Supervisor
- D. Chemistry Management

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

WECST = Waste Evaporator Condensate Storage Tank (WDL-T-11A/11B). Procedures refer to the tanks as WECSTs, therefore WECST is not defined in the question to avoid confusion.

Explanation: To answer this question correctly, the examinee must know: (1) who approves the paperwork generated for a release of a WECST to the MDCT then to the Susquehanna River; (2) If the total chemistry is above 7E-7 uCi/ml, then Chemistry Management approval is required, but since it is not, so the Shift Manager is the only approval required.

A.	Shift Manager	Attachment 7.3 flow chart: After the peer review of dose calculations are performed, the permit is reviewed and Final Approval by Shift Manager is obtained.
B.	Radwaste Supervisor	Plausible because the radwaste supervisor is a position at TMI, which would review releases. The examinee could believe that level is low enough that shift management does not need to approve the release.
C.	Unit Supervisor	Plausible because the examinee could believe that all discharges may have any Control Room SRO approval. Incorrect because the shift manager approval is required.
D.	Chemistry Management	Plausible because the Chemistry Management approval is required if total radioactivity is above 7E-7 uCi/ml.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #		2.3.6
	Importance Rating		3.8

K/A: Ability to approve release permits.

Proposed Question: Question #98

Technical Reference(s): CY-TM-170-2001, Rev 0

Proposed References to be provided to applicants during examination: None

Learning Objective: 232-GLO-9

Question Source: Bank #
Modified Bank # 770815
New

Question History: Unmodified Sim Last NRC Exam: N/A
Exam 4

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	b.4
Comments:	(KA Match, why high cog, why SRO only)	
K/A match:	This matches the KA because the examinee must know who approves the radioactive release paperwork.	
SRO Only:	This is an SRO only question because the examinee must know the process for gaseous/liquid release approvals. The operations portion of a release is to initiate paperwork, approve the paperwork and then perform the release.	
IAW CY-TM-170-2001 Releasing Radioactive Liquid Waste, who is responsible for the Final Approval on Liquid Permit Pre-Release Report for the release of a WECST?		
A.	Shift Manager.	
B.	Chemistry Tech	
C.	Operations Director	
D.	Chemistry Management	
Answer:	A	

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

99

ID: 1685047

Points: 1.00

Plant Conditions:

- Reactor is at 100% power.
- A CODE WHITE has been declared by site security

EVENT:

- All communications between the ARO and control room have been lost
- CODE BLUE is declared and confirmed by site security
- CRS/URO continue with OP-TM-AOP-008, SECURITY THREAT / INTRUSION actions in the control room.

(1) When will the ARO start taking action after reporting to the 1E 4160V bus?

(2) What actions will the ARO take?

- A. (1) When EF-P-2B is running
(2) Trip MU-P-1B, start MU-P-1A, and ensure seal injection is not lost to the reactor coolant pumps
- B. (1) When EF-P-2B is running
(2) Take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to both OTSG's
- C. (1) When the control room is breached
(2) Trip MU-P-1B, start MU-P-1A, and ensure seal injection is not lost to the reactor coolant pumps
- D. (1) When the control room is breached
(2) Take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to both OTSG's

Answer: B

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Explanation: For the examinee to answer this question correctly, they will have to realize (1) on a CODE WHITE, the ARO is dispatched to the 1E 4160V bus to wait and monitor the EF-P-2B breaker (IAW Attachment 5) (2) when the CODE BLUE is declared, the control room will trip the reactor, trip all RCP's, trip the CW pumps, and open the feeder breaker to the 1G bus (3) when the RCP's are tripped, that will initiate EFW, thus closing the EF-P-2B breaker (4) since communications to the control room have been lost, and EF-P-2B breaker is closed, the ARO will continue Attachment 5 (5) the actions in attachment 5 will have the ARO take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to the OTSG's.

A.	(1) When EF-P-2B is running (2) Trip MU-P-1B, start MU-P-1A, and ensure seal injection is not lost to the reactor coolant pumps	Plausible because in the actions described in the second part of the question are actions the URO will take in the control room in ATTACHMENT 1.
B.	(1) When EF-P-2B is running (2) Take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to both OTSG's	Correct Answer
C.	(1) When the control room is breached (2) Trip MU-P-1B, start MU-P-1A, and ensure seal injection is not lost to the reactor coolant pumps	Plausible because one of the conditions for the ARO to take actions is a Control Room breach. While the CODE BLUE or CODE YELLOW could imply such action, since the CR communications have been lost, the ARO at the 1E 4160V bus would use the indication of EF-P-2B breaker being closed as their indication to proceed.
D.	(1) When the control room is breached (2) Take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to both OTSG's	Plausible because one of the conditions for the ARO to take actions is a Control Room breach. While the CODE BLUE or CODE YELLOW could imply such action, since the CR communications have been lost, the ARO at the 1E 4160V bus would use the indication of EF-P-2B breaker being closed as their indication to proceed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #		2.4.34
	Importance Rating		4.1

K/A: Knowledge of RO tasks performed outside the main control room during and emergency and the resultant operational effects

Proposed Question: Question #99

Technical Reference(s): OP-TM-AOP-008, Rev 16

OP-TM-AOP-0081, Rev 14

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-DBIG-PCO-2

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

8

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:

K/A match: This question matches the K/A because OP-TM-AOP-008 coordinates operators outside of the control room. The CRS/URO will have to know what the ARO will be doing and when at the 1E 4160V bus. The operational effect of procedure is to have control of a diesel generator, ensure RCS inventory is sufficient and have flow to both OTSG's.

High Cog: This question is High Cog because it requires various pieces of information to be analyzed to understand when, and what actions the ARO should be taking.

SRO Only: This is an SRO ONLY question because the examinee needs to know when the ARO will be implementing the attachment in OP-TM-AOP-008, and how it is coordinated with other aspects of the Security procedure.

EXAMINATION ANSWER KEY

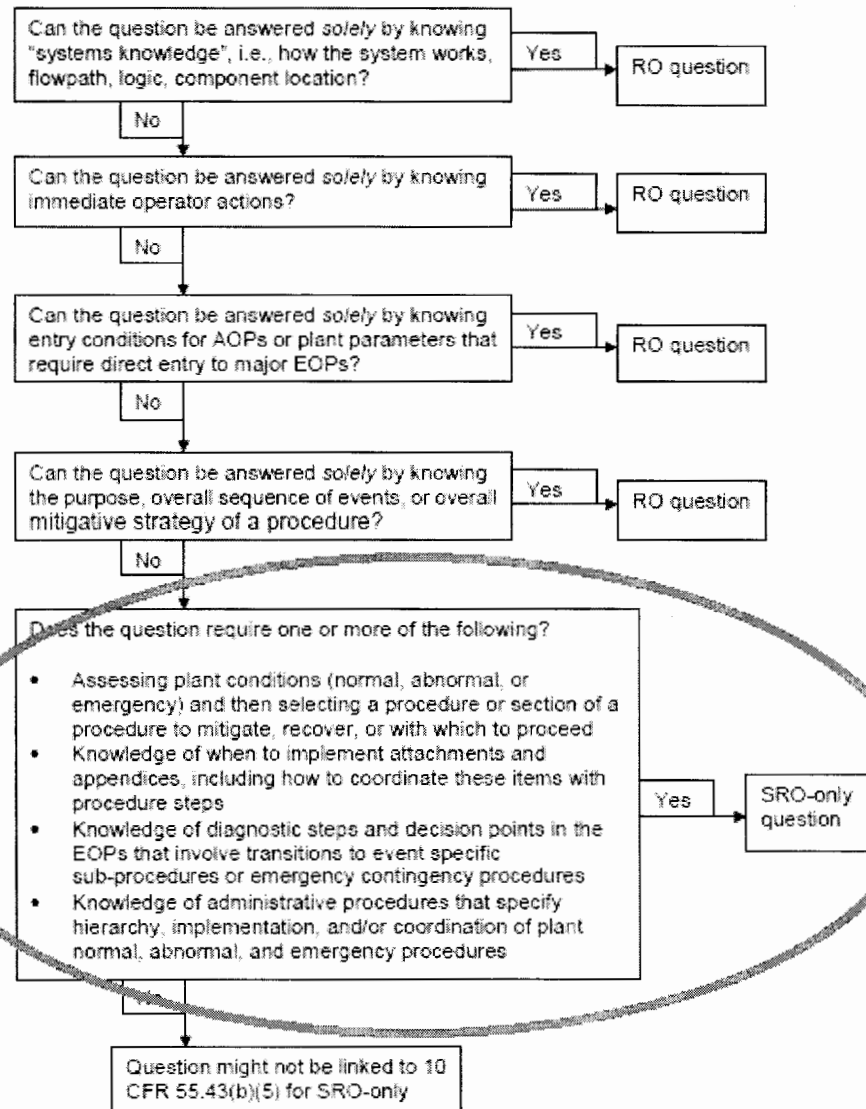
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Attachment 2

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

100

ID: 1700400

Points: 1.00

Plant Conditions:

- Reactor is operating at 100% power, with ICS in full automatic.
- You are the on-shift Control Room Supervisor.
- You have been relieved by the Shift Manager and are returning to the Control Room from the Operations Office Building (OOB).

EVENT:

Just as you pass through the Control Room entrance door:

- One-half of the overhead lights go out, and do NOT come back on.
- You observe the following MAPs actuate:
 - B-1-1, 4KV ES FDR BKR TRIP.
 - B-2-1, 4KV ES BUS UV/OV.
 - B-1-2, 4KV ES MOTOR TRIP.
 - F-1-5, RCP SEAL TOT INJECT FLOW HI/LO.

Based on these conditions, identify the ONE selection below that describes:

- (1) The requirements for you to assume a management role in response to the upset, and
 - (2) The procedure that must be implemented to perform actions that are most critical to mitigation of the event.
- A. (1) Obtain a brief update from the shift manager PRIOR to directing team activities.
(2) OP-TM-AOP-014, LOSS OF 1E 4160V BUS.
- B. (1) Obtain a brief update from the shift manager PRIOR to directing team activities.
(2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP.
- C. (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.
(2) OP-TM-AOP-014, LOSS OF 1E 4160V BUS.
- D. (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.
(2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP.

Answer: A

Answer Explanation

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

<<Explanation: To answer this question correctly, the examinee must know: (1) If a member of the control room team is absent when a transient procedure is entered, the team member receives a short briefing from ANY team member before taking ANY directed action; (2) With the indications present, the examinee must interpret that the loss of the 1E 4160V bus has occurred, (e.g., 1/2 control room lights off, Seal injection flow low indicates loss of the running makeup pump on the 1E 4160V bus), and (3) prioritize the loss of the 1E 4160V bus AOP over the listed MAP alarm responses; (4) In accordance with OS-24 section 4.1.5.B, the CRS selects the action significant to overall event mitigation, which would be the loss of the 1E 4160 V bus.

A.	(1) Obtain a brief update from the shift manager PRIOR to directing team activities. (2) OP-TM-AOP-014, LOSS OF 1E 4160V BUS.	Correct answer - Even though the on CRS on watch is returning to the control room, they should get an update before directing activities. Even though some of the alarms listed have actions in the alarm responses, the controlling document would be the AOP entered due to the loss of the 1E 4160V bus.
B.	(1) Obtain a brief update from the shift manager PRIOR to directing team activities. (2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP.	Plausible because MAP B-1-2 is a cherry alarm on the top row of its section, which indicates it is a priority alarm. The examinees are trained that the general order of priority are the cherry alarms, followed by alarms in the 1st row of every MAP section. In addition, the MAP B-1-2 alarm would NOT direct the examinee to the AOP for a loss of the 1E 4160V bus.
C.	(1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities. (2) OP-TM-AOP-014, LOSS OF 1E 4160V BUS.	Plausible because the returning CRS is a watchstander in command of the control room during abnormal events. Incorrect because OS-24 directs the CRS to receive an update prior to directing activities.
D.	(1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities. (2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP.	Plausible because the returning CRS is a watchstander in command of the control room during abnormal events. Incorrect because OS-24 directs the CRS to receive an update prior to directing activities. In addition, plausible because MAP B-1-2 is a cherry alarm on the top row of its section, which indicates it is a priority alarm. The examinees are trained that the general order of priority are the cherry alarms, followed by alarms in the 1st row of every MAP section. In addition, the MAP B-1-2 alarm would NOT direct the examinee to the AOP for a loss of the 1E 4160V bus.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #		2.4.45
	Importance Rating		4.3

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

Proposed Question: Question #100

EXAMINATION ANSWER KEY

16-01 NRC EXAM SUBMITTAL

Technical Reference(s): OS-24, Rev 28

OP-TM-AOP-014

Proposed References to be provided to applicants during examination: None

Learning Objective: 63501001

Question Source: Bank # 371264

Modified Bank #

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

55.43 b.5

Comments:

K/A Match: This matches the K/A because the examinee has to analyze the given conditions and alarms, and have knowledge of OS-24 to understand that it is the CRS positions job to select the procedure or alarm response to mitigate the event.

High Cog: This is high cog because the examinee must analyze the event and alarms, and choose the correct procedure to mitigate the event.

SRO Only: This is SRO only because the examinee must have knowledge of the administrative procedure OS-24 to understand how to prioritize the alarms and AOP entry.

EXAMINATION ANSWER KEY

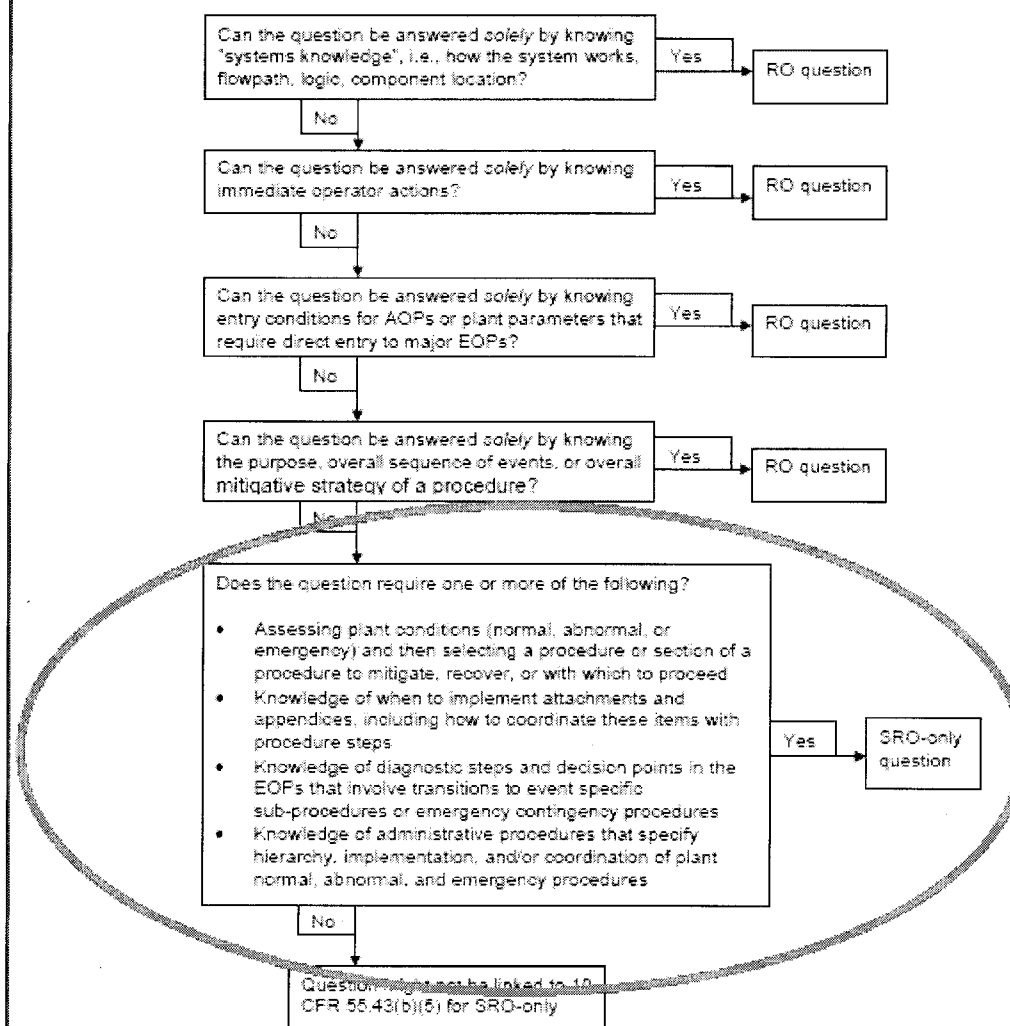
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Attachment 2

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)**



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