Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	211000 K1.01
Importance Rating	3.0

SLC

Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Core spray line break detection: Plant-Specific

Proposed Question: #1

Which one of the following describes the status of Core Plate dP indication and Core Spray line break indication when the Standby Liquid Control (SBLC) system is injected to the Reactor?

	Core Plate dP Indication	Core Spray Line Break Indication
A.	Unaffected	Unaffected
B.	Unaffected	Reads higher than actual
C.	Reads higher than actual	Unaffected
D.	Reads higher than actual	Reads higher than actual

Proposed Answer: C

Explanation: Both the SLC system, the Core Plate dP indication, and the Core Spray line break indication are physically related through use of a shared penetration for two pipes. The SLC system injects below the Core Plate through one pipe. The Core Spray line break indication is connected to the other pipe. The Core Plate dP indication is connected to both pipes. When SBLC injects to the Reactor, it raises pressure in its pipe, but does not affect the other pipe. Higher pressure in the one pipe causes Core Plate dP indication to read higher than actual. Since pressure is unchanged in the other pipe, Core Spray line break indication is unaffected.

Note: The question meets the K/A by testing knowledge of the physical connection of the SLC system and Core Spray line break detection piping arrangement, which includes many functions, such as core plate D/P indication, SLC injection sparger, and Core Spray line break detection system. To select the correct answer and eliminate distractors, the candidate must display knowledge of how SLC and the Core Spray line break detection system interact with this arrangement.

A. Incorrect – Core Plate dP indication reads higher than actual. Plausible if this indication used only one pipe, such as Core Spray line break indication, or if SBLC injection equally affected both pipes.
B. Incorrect – Core Plate dP indication reads higher than actual. Plausible if this indication used only one pipe, such as Core Spray line break indication, or if SBLC injection equally affected both pipes. Core Spray line break indication is unaffected. Plausible if this indication used the same pipe as SBLC used.
D. Incorrect – Core Spray line break indication is unaffected. Plausible if this indication used the same pipe as SBLC used.

Technical Reference(s): M-142, TM-OP-080 RBO-03 section B

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-053 RBO-5

Question Source: Modified Bank – LOC25 NRC #24

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross–reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	211000 K1.02		
	Importance Rating	2.7		

K1.02 – Knowledge of the physical connections and/or cause– effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Core plate differential pressure indication

Proposed Question: Common 24

Which of the following describes how the Standby Liquid Control (SBLC) System is physically connected to the Nuclear Instrumentation system?

The SBLC injection line piping arrangement allows for measuring pressure ___(1)___ the core plate.

When SBLC injects, Core Plate ΔP will indicate ___(2)___ than actual.

A. (1) above
(2) higher

B. (1) above
(2) lower

C. (1) below
(2) higher

D. (1) below
(2) lower

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 K1.03
	Importance Rating	3.8

Reactor Water Level Control

Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following: Reactor water level

Proposed Question: #2

Unit 1 is operating at 80% power with the following:

- The Reactor spuriously scrams.
- Reactor water level reaches a low of +7" before beginning to slowly rise.
- HPCI and RCIC remain in standby.
- NO operator actions have been taken.

Which one of the following describes how the Feedwater Level Control System (FWLC) controls Reactor water level ten (10) minutes later?

FWLC controls Reactor water level at...

A. +22" by controlling LV-10641, FW Low Load Valve, position.

B. +35" by controlling LV-10641, FW Low Load Valve, position.

C. +22" by controlling one RFPT speed in flow control mode.

D. +35" by controlling one RFPT speed in flow control mode.

Proposed Answer: A

Explanation: When Reactor water level lowers below +13", the Setpoint Setdown logic actuates. This causes one RFPT to shift to discharge pressure mode, two RFPT to shift to idle, and LV-10641 to control Reactor water level at +22".

B. Incorrect – Level is controlled at +22", not +35". Plausible because +35" is the normal Reactor water level control setpoint and would be correct if Setpoint Setdown had not actuated.
C. Incorrect – Level is controlled by LV-10641, not a RFPT in flow control mode. Plausible because Reactor water level was controlled by a RFPT in flow control mode prior to the scram, and this would be correct if Setpoint Setdown had not actuated.
D. Incorrect – Level is controlled by LV-10641, not a RFPT in flow control mode. Plausible because Reactor water level was controlled by a RFPT in flow control mode prior to the scram, and this would be correct if Setpoint Setdown had not actuated. Level is controlled at +22", not +35". Plausible because +35" is the normal Reactor water level control setpoint and would be correct if Setpoint Setdown had not actuated.

Technical Reference(s): OP-145-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-45 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 K2.02
	Importance Rating	2.5

Shutdown Cooling

Knowledge of electrical power supplies to the following: Motor operated valves

Proposed Question: #3

Which one of the following describes the power supplies to the Unit 1 RHR Shutdown Cooling Suction Isolation Valves?

	HV-151-F008, RHR Shutdown Cooling Suction Outboard Isolation Valve	HV-151-F009, RHR Shutdown Cooling Suction Inboard Isolation Valve
A.	480 VAC	480 VAC
B.	480 VAC	250 VDC
C.	250 VDC	480 VAC
D.	250 VDC	250 VDC

Proposed Answer: C

Explanation: HV-151-F008 is powered from 250 VDC 1D274. HV-151-F009 is powered from 480 VAC 1B236.

A. Incorrect – HV-151-F008 is powered from 250 VDC 1D274. Plausible because HV-151-F009 is powered from 480 VAC 1B236.
B. Incorrect – HV-151-F008 is powered from 250 VDC 1D274. Plausible because HV-151-F009 is powered from 480 VAC 1B236. HV-151-F009 is powered from 480 VAC 1B236. Plausible because HV-151-F008 is powered from 250 VDC 1D274.
D. Incorrect – HV-151-F009 is powered from 480 VAC 1B236. Plausible because HV-151-F008 is powered from 250 VDC 1D274.

Technical Reference(s): TM-OP-049

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 K2.02
	Importance Rating	2.5

LPCS

Knowledge of electrical power supplies to the following: Valve power

Proposed Question: #4

Unit 1 is operating at 100% power with the following:

- 480 VAC Bus 1B217 de-energizes due to a sustained electrical fault.

Which one of the following describes the effect of this fault on the Core Spray system?

A. Core Spray loop A CANNOT automatically inject to the Reactor.

B. Core Spray loop B CANNOT automatically inject to the Reactor.

C. Core Spray loop A minimum flow valve CANNOT automatically close.

D. Core Spray loop B minimum flow valve CANNOT automatically close.

Proposed Answer: A

Explanation: 480 VAC Bus 1B217 supplies power to HV-152-F005A, the Core Spray loop A inboard injection valve. This valve is normally closed and must open for Core Spray loop A to automatically inject to the Reactor. Without power from 1B217, Core Spray loop A CANNOT automatically inject to the Reactor.

B. Incorrect – Core Spray loop B injection valves can still open. Plausible because this would be the correct answer for failure of Bus 1B227.
C. Incorrect – Core Spray loop A min flow valve still has power. Plausible because this would be the correct answer for failure of Bus 1B216.
D. Incorrect – Core Spray loop B min flow valve still has power. Plausible because this would be the correct answer for failure of Bus 1B226.

Technical Reference(s): TM-OP-051 RBO-3

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-051 RBO-3

Question Source: Bank – JAF 9/14 NRC #3

Question History: JAF 9/14 NRC #3

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 K3.03
	Importance Rating	3.4

DC Electrical Distribution

Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Systems with D.C. components (i.e. valves, motors, solenoids, etc.)

Proposed Question: #5

Unit 2 is operating at 100% power with the following:

- 125 VDC Distribution Panel 2D624 de-energizes due to a sustained electrical fault.

Which one of the following describes the effect of this fault?

A. RCIC is inoperable.

B. HPCI is inoperable.

C. One division of RCIC steam leak detection is inoperable.

D. One division of HPCI steam leak detection is inoperable.

Proposed Answer: B

Explanation: 125 VDC Distribution Panel 2D624 results in HPCI becoming inoperable.

A. Incorrect – RCIC remains operable. Plausible because this would be correct if the fault were on 2D614.
C. Incorrect – RCIC steam leak detection is unaffected. Plausible because this would be correct if the fault were on 2D634 or 2D644.
D. Incorrect – HPCI steam leak detection is unaffected. Plausible because HPCI is affected in other ways and this would be correct for a loss of 2Y216 or 2Y226.

Technical Reference(s): ON-125VDC-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-052 RBO-5

Question Source: Modified Bank – LOC25 NRC #4

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross–reference:	Level	RO		SRO
	Tier #	2		
	Group #	1		
	K/A #	209001 K2.03		
	Importance Rating	2.9		

K2.03 – Knowledge of electrical power supplies to the following: Initiation Logic

Proposed Question:	Common 4

Unit 1 is operating at 80 percent power.

Annunciator 125V DC PANEL 1L610 SYSTEM TROUBLE (AR–106–A12) alarms.

It is determined that 125 VDC Bus 1D612 is de–energized.

Which of the following operational impacts is directly associated with this power loss?

A. RCIC Division 1 steam leak detection is inoperable

B. HPCI Division 2 isolation logic automatically initiates

C. Automatic initiation of Division 1 Core Spray logic will NOT actuate

D. Automatic initiation of Division 2 Core Spray logic will NOT actuate

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 K3.03
	Importance Rating	3.5

RCIC

Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: Decay heat removal

Proposed Question: #6

Unit 2 has scrammed and it is discovered that HV-149-F022, RCIC TEST LINE ISO TO CST, is stuck in the closed position.

Which one of the following describes the ability to use HPCI and RCIC in the pressure control mode with this failure?

A. Both HPCI and RCIC can be used in the pressure control mode.

B. HPCI can be used in the pressure control mode, but RCIC CANNOT.

C. RCIC can be used in the pressure control mode, but HPCI CANNOT.

D. NEITHER HPCI NOR RCIC can be used in the pressure control mode.

Proposed Answer: B

Explanation: HV-149-F022, RCIC TEST LINE ISO TO CST, is required to be opened to place RCIC in the pressure control mode, but not to place HPCI in the pressure control mode. Therefore with the valve stuck in the closed position, HPCI can be placed in the pressure control mode, but RCIC cannot.

A. Incorrect – RCIC cannot be placed in the pressure control mode. Plausible because the RCIC injection mode is still available, just not the pressure control mode.
C. Incorrect – RCIC cannot be placed in the pressure control mode. Plausible because the RCIC injection mode is still available, just not the pressure control mode. HPCI can be placed in the pressure control mode. Plausible because this would be correct for failure of related valve HV-155-F008, HPCI TEST LINE TO CST ISO.
D. Incorrect - HPCI can be placed in the pressure control mode. Plausible because this would be correct for failure of related valve HV-155-F011, HPCI TEST LINE TO CST.

Technical Reference(s): OP-150-001, OP-152-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-050 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 K4.04
	Importance Rating	2.7

SGTS

Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Radioactive particulate filtration

Proposed Question: #7

Both Units are operating at 100% power with the following:

- The Standby Gas Treatment System (SGTS) is in a normal standby lineup.
- A Reactor water level transient occurs on Unit 1.
- Unit 1 is manually scrammed.
- Unit 1 Reactor water level reaches a low of -45" during the transient.
- Unit 1 Reactor water level is now +35", steady.

Which one of the following describes the response of SGTS and the Reactor Building Ventilation System Zones?

SGTS...

A. remains in standby. NO Zones isolate.

B. automatically starts. Zone 1 isolates, only.

C. automatically starts. Zones 1 and 3 isolate, only.

D. automatically starts. Zones 1, 2, and 3 isolate.

Proposed Answer: C

Explanation: Low Reactor water level (<38") on either Unit causes both SGTS fans to automatically start (both are aligned in "Auto Lead" in a normal standby lineup). Low Reactor water level on a single Unit causes two of the RB Ventilation Zones to isolate (1 and 3 isolate on a Unit 1 signal, 2 and 3 isolate on a Unit 2 signal). This ensures SGTS is drawing suction from the Secondary Containment surrounding the affected Unit's Drywell to filter an radioactive leakage from the Primary Containment.

A. Incorrect – SGTS automatically starts and Zones 1 and 3 isolate due to Reactor water level <-38". Plausible no high Drywell pressure condition exists, no high radiation condition exists, and Reactor water level did not go much below setpoint before recovering to normal.
B. Incorrect – Zone 3 also isolates. Plausible because Zone 3 is common to both Units, there is no problem specifically in Zone 3 or on Unit 2.
D. Incorrect – Zone 2 does not isolate. Plausible that all Zones would isolate since both SGTS fans automatically start and the Units' Reactor Buildings are connected.

Technical Reference(s): ON-CONTISOL-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-070 RBO-4

Question Source: Modified Bank - LOC26R #23

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A	261000 K4.04 SGTS					Importance	2.7		
Statement	Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following: Radioactive particulate filtration								

QUESTION 23

Both Units are operating at 100% power with the following:

- The Standby Gas Treatment System (SGTS) is in a normal standby lineup.
- A steam leak develops inside the Unit 2 Drywell.
- Unit 2 is manually scrammed.
- Unit 2 Reactor water level reaches a low of -25" during the transient.
- Unit 2 Reactor water level is now +35", steady.
- Unit 2 Drywell pressure is 7.5 psig, up slow.

Which one of the following describes the response of SGTS and the Reactor Building Ventilation System Zones?

A.	Both SGTS fans start. Zones 1, 2, and 3 isolate.
B.	**Both SGTS fans start. Zones 2 and 3 isolate, only.**
C.	Only one SGTS fan starts. Zones 1, 2, and 3 isolate.
D.	Only one SGTS fan starts. Zones 2 and 3 isolate, only.

Proposed Answer	B	
Applicant References	None	
Explanation	High Drywell pressure (> 1.72 psig) on either Unit causes both SGTS fans to automatically start (both are aligned in "Auto Lead" in a normal standby lineup). High Drywell pressure on a single Unit only causes two of the RB Ventilation Zones to isolate (1 and 3 isolate on Unit 1 high Drywell pressure, 2 and 3 isolate on Unit 2 high Drywell pressure). This response ensures SGTS is drawing suction from the Secondary Containment surrounding the affected Drywell to filter any radioactive leakage from the Primary Containment.	
	A	Incorrect – No isolation signal is present for RB Ventilation Zone 1.
	B	Correct.
	C	Incorrect – Both SGTS fans start. No isolation signal is present for RB Ventilation Zone 1.
	D	Incorrect – Both SGTS fans start.
10CFR55	41.9	
Technical References	ON-259-002	
Learning Objectives	1991.a	
Question Source	Bank	Vision SYSD 33385

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 K4.02
	Importance Rating	2.9

AC Electrical Distribution

Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Circuit breaker automatic trips

Proposed Question: #8

Unit 1 is operating at 100% power with the following:

- Diesel Generator (DG) B is running synchronized to 4.16KV Bus 1B per SO-024-001B, Monthly Diesel Generator B Operability Test.
- Breaker 1A20204, DG B TO BUS 1B BKR, is closed.
- Breaker 1A20209, XFMR 211 (0X213) TO BUS 1B BKR, is closed.
- Breaker 1A20201, XFMR 111 (0X211) TO BUS 1B BKR, is open.

Then, a steam leak in the Drywell results in the following:

- The Reactor is manually scrammed.
- Drywell pressure is 1.72 psig, up slow.
- Reactor water level reaches a low of 0" before recovering to the normal band.

Which one of the following describes the status of breakers one (1) minute later?

	1A20204	1A20209	1A20201
A.	Closed	Closed	Open
B.	Closed	Open	Open
C.	Open	Closed	Open
D.	Open	Open	Closed

Proposed Answer: C

Explanation: The DG output breaker 1A20204 automatically trips upon receipt of a LOCA signal (in this case, Drywell pressure >1.72 psig) with either the normal or alternate ESS Bus supply breaker closed (in this case, 1A20209 is closed). With no undervoltage or lockout signals present, the ESS Bus supply breakers remain in their initial position (1A20209 is closed, 1A20201 is open).

A. Incorrect – 1A20204 automatically trips open. Plausible because the breaker is initially closed and the DG gets an automatic start signal due to high Drywell pressure.
B. Incorrect – 1A20204 automatically trips open. Plausible because the breaker is initially closed and the DG gets an automatic start signal due to high Drywell pressure. 1A20209 remains closed. Plausible because the DG gets an automatic start signal due to high Drywell pressure and because this breaker does have automatic trips on other conditions (undervoltage, lockout).
D. Incorrect – 1A20209 remains closed and 1A20201 remains open. Plausible because this breaker swap does occur under some conditions (bus undervoltage).

Technical Reference(s): TM-OP-024 RBO-4

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-024 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	203000 K5.01
	Importance Rating	2.7

RHR/LPCI: Injection Mode

Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: INJECTION MODE (PLANT SPECIFIC): Testable check valve operation

Proposed Question: #9

Unit 1 has experienced a coolant leak in the Drywell with the following:

- Reactor water level is -170", up slow.
- Reactor pressure is 390 psig, down slow.
- Drywell pressure is 10 psig, stable.

Which one of the following describes the status of HV-151-F015A, RHR INJECTION OUTBOARD ISOLATION, and HV-151-F050A, RHR LOOP TESTABLE CKV?

A. Both valves are open.

B. HV-151-F015A is open, but HV-151-F050A is closed.

C. HV-151-F015A is closed, but HV-151-F050A is open.

D. Both valves are closed.

Proposed Answer: B

Explanation: HV-151-F015A is open due to a combination of low Reactor water level (<-129"), high Drywell pressure (>1.72 psig), and low Reactor pressure (<420 psig). HV-151-F050A remains closed because Reactor pressure is higher than RHR pump capacity to inject to the Reactor (~300 psig) and the minimum flow line taps off upstream of HV-151-F050A.

A. Incorrect – HV-151-F050A remains closed because Reactor pressure is higher than RHR pump capacity to inject to the Reactor and the minimum flow line taps off upstream of HV-151-F050A. Plausible that this valve would be open to pass minimum flow or because the valve does have an opening mechanism (used for testing only). Also plausible because Reactor pressure is below 420 psig.
C. Incorrect – HV-151-F015A is open due to a combination of low Reactor water level (<-129"), high Drywell pressure (>1.72 psig), and low Reactor pressure (<420 psig). Plausible because the valve is normally closed and Reactor pressure is still above RHR injection capacity (~300 psig). HV-151-F050A remains closed because Reactor pressure is higher than RHR pump capacity to inject to the Reactor and the minimum flow line taps off upstream of HV-151-F050A. Plausible that this valve would be open to pass minimum flow or because the valve does have an opening mechanism (used for testing only). Also plausible because Reactor pressure is below 420 psig.
D. Incorrect – HV-151-F015A is open due to a combination of low Reactor water level (<-129"), high Drywell pressure (>1.72 psig), and low Reactor pressure (<420 psig). Plausible because the valve is normally closed and Reactor pressure is still above RHR injection capacity (~300 psig).

Technical Reference(s): TM-OP-049 RBO-2

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-2

Question Source: Bank – LOC24 NRC #28

Question History: LOC24 NRC #28

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 K5.02
	Importance Rating	3.3

RPS

Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements

Proposed Question: #10

The plant is operating at 40% power with the following:

- Turbine Stop Valve (TSV) 4 drifts 50% closed.
- Then, TSV 3 drifts 50% closed.

Considering only the Turbine Stop Valve Closure scram signal, which one of the following describes the resulting operation of the Reactor Protection System?

Based on the Turbine Stop Valve Closure scram signal, a full Reactor scram...

A. occurs while TSV 4 is drifting.

B. occurs while TSV 3 is drifting.

C. does NOT occur because only these two TSVs have drifted.

D. does NOT occur because these two TSVs have not tripped their scram position switches.

Proposed Answer: C

Explanation: Each Turbine Stop Valve (TSV) has two position switches that actuate when the valve is less than 94.5% open. These position switches input to the Reactor Protection System scram circuitry. The logic is arranged such that some combinations of 2 TSV (including TSV 3 & 4) cause a half scram, but 3 TSVs must close to cause an actual Reactor scram. In this case, both TSV 3 & 4 have drifted far enough to trip their scram position switches, which results in a half scram, but NOT an actual Reactor scram.

A. Incorrect – A scram does not occur. Plausible because one TSV <94.5% open trips two scram position switches, but this is not enough to cause RPS logic to enforce a Reactor scram.
B. Incorrect – A scram does not occur. Plausible because when TSV 3 drifts <94.5% open, a half scram is received, but not an actual Reactor scram.
D. Incorrect – These TSVs have both drifted far enough to trip their scram position switches (<94.5% open). Plausible because the valves have only partially drifted and stop valves have poor throttling characteristics, such that they would not be expected to significantly lower flow / raise pressure until drifted further.

Technical Reference(s): TM-OP-058 RBO-4

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-058 RBO-4

Question Source: Bank – JAF 9/14 NRC #10

Question History: JAF 9/14 NRC #10

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 K6.09
	Importance Rating	3.3

EDGs

Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET): D.C. power

Proposed Question: #11

Both Units are operating at 100% power when 125 VDC ESS Channel C distribution panel 1D634 de-energizes due to a sustained electrical fault.

Which one of the following describes the effect of this fault on Diesel Generator (DG) C?

DG C...

A. CANNOT be started until the panel is re-energized.

B. control power automatically swaps to an alternate source.

C. normal control power remains available, but the alternate source is lost.

D. control power is lost, but it can be manually transferred to an alternate source.

Proposed Answer: D

Explanation: DG C normally receives control power from 1D634. With 1D634 de-energized, DG C loses control power and automatic starting capability. Alternate control power is available from Unit 2 DC panel 2D634. Swap to this alternate control power is manual.

A. Incorrect – DG C control power can be manually transferred to 2D634, allowing start of the DG. Plausible because DG C will not start until manual action is taken to swap control power.
B. Incorrect – Control power must be manually swapped. Plausible because some other DC control power automatically swaps.
C. Incorrect – Normal control power is lost and alternate remains available. Plausible if the normal/alternate setup was swapped between the Units.

Technical Reference(s): ON-125VDC-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-024 RBO-4

Question Source: Modified Bank – LOC25 Cert #12

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 K6.09	
	Importance Rating	3.3	

K6.09 - Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET): D.C. Power	
Proposed Question:	Common 12

Unit 1 is at rated power.

A fault results in de-energization of 1D634.

How would Diesel Generator C respond to a LOCA start signal?

DG C would...

A.	NOT automatically start, but could be manually started from Control Room Panel 0C653, in the Droop Mode, using the "DG C Start" pushbutton.
B.	NOT automatically start, but could be manually started from Engine Control Panel 0C521, in the Local Mode.
C.	automatically start when the 125VDC Transfer switches are manually placed to alternate supply 2D614 and the DG Mode Switch is returned to REMOTE.
D.	**<u>automatically</u> start when the 125VDC Transfer switches are manually placed to alternate supply 2D634 and the DG Mode Switch is returned to REMOTE.**

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	300000 K6.12
Importance Rating	2.9

Instrument Air

Knowledge of the effect that a loss or malfunction of the following will have on the INSTRUMENT AIR SYSTEM: Breakers, relays and disconnects

Proposed Question: #12

Unit 1 is operating at 100% power with the following:

- Instrument Air Compressor (IAC) 1A is in service.
- IAC 1B is in standby.
- Instrument Air (IA) pressure is in the normal band.
- 1C140A switch HSS–12500, I–A COMPRESSOR A/B BASE UNIT SELECT, is selected to COMPRESSOR A.
- PCV-12560, Service Air Crosstie to Instrument Air, is closed and CANNOT be opened.
- Other Unit 1 IAC switches are aligned as follows:

Switch	IAC 1A	IAC 1B
Control Room Switch	AUTO	AUTO
1C140A Switch I-A COMPRESSOR CONTROL MODE SELECT	MAN	AUTO
1C140A Switch LOAD LIMIT SELECT	FULL	FULL

Then, the breaker for IAC 1A trips on overcurrent.

Which one of the following describes the response of IAC 1B?

IAC 1B first receives an automatic start signal __(1)__. IAC 1B will then unload when IA pressure rises to __(2)__.

	(1)	(2)
A.	due to an IAC 1A breaker position signal	107 psig
B.	due to an IAC 1A breaker position signal	98 psig
C.	when IA pressure lowers to 87 psig	107 psig
D.	when IA pressure lowers to 87 psig	98 psig

Proposed Answer: D

Explanation: IAC 1B automatically starts when IA pressure degrades to 87 psig. Since it is set up as the standby compressor, it will unload at 98 psig.

A. Incorrect – IAC 1B does not receive a start signal on IAC 1A breaker position. Plausible because many systems are arranged such that a standby component auto starts on trip of the corresponding components breaker. IAC 1B unloads at 98 psig. Plausible because it would unload at 107 psig if re-aligned as the lead compressor.
B. Incorrect – IAC 1B does not receive a start signal on IAC 1A breaker position. Plausible because many systems are arranged such that a standby component auto starts on trip of the corresponding components breaker.
C. Incorrect – IAC 1B unloads at 98 psig. Plausible because it would unload at 107 psig if re-aligned as the lead compressor.

Technical Reference(s): TM-OP-018

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-018 RBO-4

Question Source: Bank – JAF 9/14 NRC #20

Question History: JAF 9/14 NRC #20

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 K4.01
	Importance Rating	3.1

UPS (AC/DC)

Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following: Transfer from preferred power to alternate power supplies

Proposed Question: #13

Which one of the following identifies the effect on Vital UPS 1D666(2D666) of a loss of the Division 2 250 VDC bus 1D662(2D662)?

	Unit 1 - 1D666 Response to Loss of 1D662	Unit 2 - 2D666 Response to Loss of 2D662
A.	Transfers to alternate source	Transfers to alternate source
B.	Transfers to alternate source	Remains on preferred source
C.	Remains on preferred source	Transfers to alternate source
D.	Remains on preferred source	Remains on preferred source

Proposed Answer: B

Explanation: The Unit 1 Vital UPS, 1D666, is supplied from Class 1E 250 VDC bus 1D662. The Unit 2 Vital UPS, 2D666, is supplied from a separate non-Class 1E 250 VDC battery, 2D142. Therefore, 1D666 transfers on this loss, but 2D666 does not.

A. Incorrect – 2D666 remains on the preferred supply. Plausible because there is a design difference between the Units and Unit 1 does transfer to alternate.
C. Incorrect – 1D666 transfers to an alternate supply. Plausible because there is a design difference between the Units and Unit 2 does remain on preferred. 2D666 remains on the preferred supply. Plausible because there is a design difference between the Units and Unit 1 does transfer to alternate.
D. Incorrect – 1D666 transfers to an alternate supply. Plausible because there is a design difference between the Units and Unit 2 does remain on preferred. Also plausible that DC would be the alternate source for both UPSs.

Technical Reference(s): ON-250VDC-1(2)01, TM-OP-017

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-017 RBO-3

Question Source: Bank – LOC26 NRC #11

Question History: LOC26 NRC #11

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 A1.02
	Importance Rating	3.7

ADS

Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: ADS valve acoustical monitor noise: Plant-Specific

Proposed Question: #14

Unit 1 has experienced an accident with the following:

- Reactor pressure is 800 psig, down slow.
- Emergency RPV Depressurization is required per EO-100-112.
- Another operator has taken action in the Lower Relay Room to open the required ADS SRVs per EO-100-112.
- Acoustic monitors indicate as shown in the pictures on the following pages.

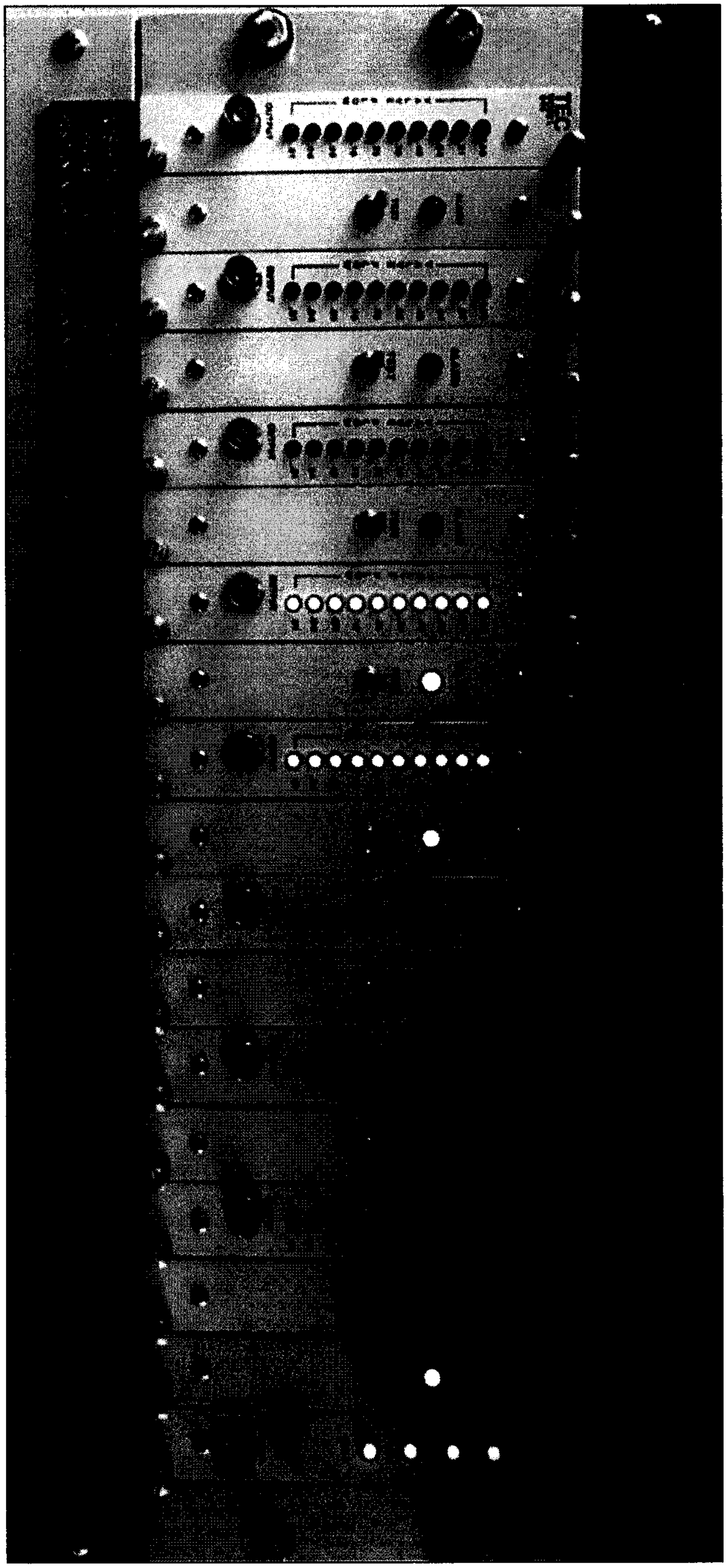
Which one of the following describes the status of the ADS SRVs and the need to open any additional SRVs, in accordance with EO-100-112?

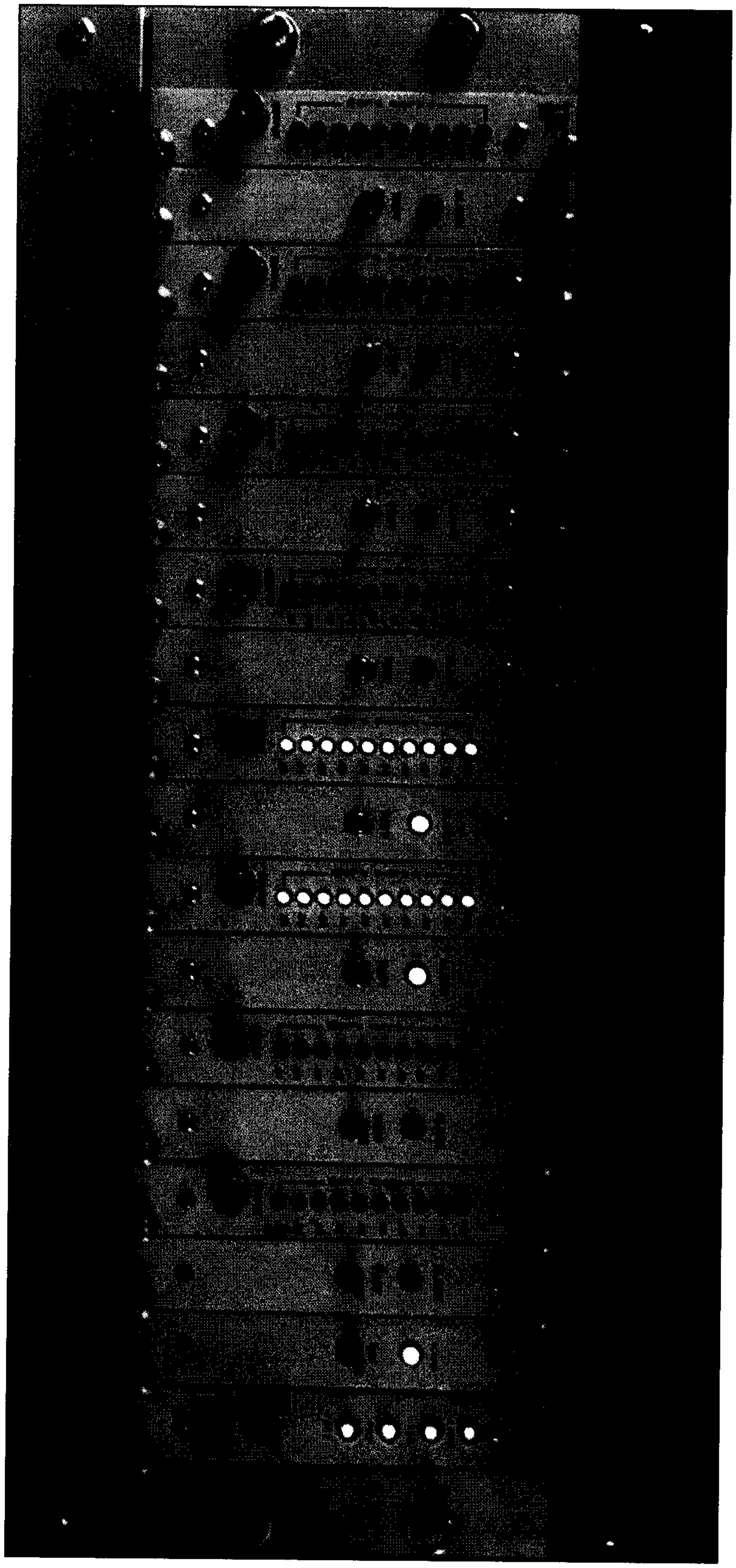
A. All ADS SRVs are open. NO action is required to open additional SRVs.

B. All ADS SRVs are open. Action is required to open one or more additional SRVs.

C. One or more ADS SRVs is NOT open. NO action is required to open additional SRVs.

D. One or more ADS SRVs is NOT open. Action is required to open one or more additional SRVs.





Proposed Answer: D

Explanation: EO-100-112 requires opening all 6 ADS SRVs. Only 4 ADS SRVs indicate open by the given acoustic monitor indications. Since 2 ADS SRVs cannot be opened, EO-100-112 then requires action to open additional SRVs until a total of 6 are open.

A. Incorrect – Only 4 SRVs indicate open, but there are 6 ADS SRVs and they are all required to be opened. Plausible if acoustic monitor indications are misinterpreted or EO-100-112 requirements are not understood.
B. Incorrect – Only 4 SRVs indicate open, but there are 6 ADS SRVs and they are all required to be opened. Plausible if acoustic monitor indications are misinterpreted or EO-100-112 requirements are not understood.
C. Incorrect – EO-100-112 requires opening additional SRVs. Plausible because if only 5 SRVs can be opened after all are attempted, then EO-100-112 does not require additional action to depressurize with other systems.

Technical Reference(s): EO-100-112

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-083E RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 A2.01
	Importance Rating	2.7

APRM / LPRM

Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply degraded

Proposed Question: #15

Unit 1 is operating at 100% power with the following:

- A circuit breaker trip results in a sustained loss of power from RPS bus A to the Power Range Neutron Monitoring (PRNM) system.
- All other power from RPS bus A remains available.

Which one of the following describes whether a half scram is received and corresponding operator action?

A half scram is...

A. received. The half scram can be reset after bypassing an APRM.

B. received. The half scram CANNOT be reset by bypassing an APRM.

C. NOT received. A half scram is required to be inserted by operators.

D. NOT received. NO half scram is required to be inserted by operators.

Proposed Answer: B

Explanation: RPS bus A supplies power to each APRM through the Quad Voltage Power Supplies. These supplies are auctioneered such that all APRMs continue to operate normally. This same RPS bus A circuit also gives power to two of the four Voters (1 & 3). With loss of power to Voters 1 & 3, each fails safe such that a half scram is received on RPS A. Since the effect is not based on failure of APRMs, it is not possible to reset the half scram until power is restored to the Voters.

A. Incorrect – Bypassing APRM(s) will not allow resetting the half scram because power is lost to two of the four Voters. Plausible because part of the auctioneered power supply to each APRM is lost, and if the half scram were just caused by APRM INOP condition, then bypassing APRMs would allow reset of half scram.
C. Incorrect – A half scram is received. Plausible because APRM power supplies are auctioneered such that APRMs continue to operate normally. Plausible that a half scram would be required to be inserted if candidate believed this left PRNM sufficiently inoperable that Tech Specs would require half scram.
D. Incorrect – A half scram is received. Plausible because APRM power supplies are auctioneered such that APRMs continue to operate normally. Plausible that no half scram would be required because APRM power supplies are auctioneered.

Technical Reference(s): TM-OP-058, TM-OP-078D

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-078D RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 A2.04
	Importance Rating	3.7

IRM

Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Up scale or down scale trips

Proposed Question: #16

A startup is in progress on Unit 2 with the following:

- All IRMs are on range 8.
- IRMs are indicating as follows:

IRM A	21
IRM B	8
IRM C	34
IRM D	40
IRM E	62
IRM F	89
IRM G	115
IRM H	71

Which one of the following identifies an action required to clear an IRM upscale or downscale alarm or trip?

Place IRM...

A. B on range 7.

B. B on range 9.

C. G on range 7.

D. G on range 9.

Proposed Answer: D

Explanation: IRM upscale trip is received at 120/125, IRM upscale alarm is received at 108/125, and IRM downscale alarm is received at 5/125. This results in IRM G being in upscale alarm and no other IRM having an upscale or downscale condition. To clear the IRM G upscale alarm, IRM G must be placed on range 9.

Note: The question meets both parts of the K/A by requiring the candidate to assess multiple IRMs to determine which one is causing an upscale or downscale alarm or trip, and then requires determining the correct mitigating action to clear the trip.

A. Incorrect – IRM B is not causing an upscale or downscale alarm or trip. Plausible because it is the lowest reading IRM and it is close to the downscale alarm setpoint.
B. Incorrect – IRM B is not causing an upscale or downscale alarm or trip. Plausible because it is the lowest reading IRM and it is close to the downscale alarm setpoint.
C. Incorrect – Placing IRM G on range 7 causes its indication to go higher, not lower. Plausible if candidate confuses direction of operation of the IRM range switches, which is a common and significant error.

Technical Reference(s): AR-104-A05, B05, and C05

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-078B RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 A3.03
	Importance Rating	3.9

HPCI

Ability to monitor automatic operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM including: System lineup: BWR-2,3,4

Proposed Question: #17

A transient on Unit 1 results in the following:

- HPCI automatically starts.
- Then, HPCI trips on high Reactor water level.
- Drywell pressure is 4 psig, up slow.
- Reactor water level is +45", down slow.

Which one of the following describes the response of HPCI if HS-E41-1S25, HPCI HI WTR LVL TRIP RESET, is depressed?

A. FV-15612, HPCI TURB STOP VLV, automatically re-opens to re-start HPCI.

B. HV-155-F001, HPCI TURB STEAM SUPPLY VLV, automatically re-opens to re-start HPCI.

C. HPCI remains secured. If Reactor water level reaches -38", FV-15612, HPCI TURB STOP VLV, automatically re-opens to re-start HPCI.

D. HPCI remains secured. If Reactor water level reaches -38", HV-155-F001, HPCI TURB STEAM SUPPLY VLV, automatically re-opens to re-start HPCI.

Proposed Answer: A

Explanation: The high level trip caused FV-15612 to close, tripping HPCI. With Reactor water level below 54" but still above -38", the high level trip is sealed-in. Additionally, with high Drywell pressure, a HPCI initiation signal is present. When HS-E41-1S25 is depressed, the high level trip clears, the initiation signal causes FV-15612 to automatically re-open, and HPCI re-starts.

B. Incorrect – FV-15612, not HV-155-F001, is what re-opens to re-start HPCI after a high level trip. Plausible because closure of HV-155-F001 would cause HPCI to stop and this valve opens on a normal HPCI start from standby.
C. Incorrect – HPCI automatically re-starts because Drywell pressure is above 1.72 psig. Plausible because Reactor water level is well above the -38" HPCI initiation setpoint.
D. Incorrect – HPCI automatically re-starts because Drywell pressure is above 1.72 psig. Plausible because Reactor water level is well above the -38" HPCI initiation setpoint. FV-15612, not HV-155-F001, is what re-opens to re-start HPCI after a high level trip. Plausible because closure of HV-155-F001 would cause HPCI to stop and this valve opens on a normal HPCI start from standby.

Technical Reference(s): OP-152-001, TM-OP-052

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-052 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 A3.01
	Importance Rating	3.4

PCIS/Nuclear Steam Supply Shutoff

Ability to monitor automatic operations of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF including: System indicating lights and alarms

Proposed Question: #18

Unit 1 is operating at 100% power with the following:

- A RBCCW temperature transient occurs.
- Reactor Water Cleanup (RWCU) filter demineralizer inlet temperature reaches a high of 147°F during the transient.
- The picture on the following page shows the status of RWCU once RBCCW temperature is restored to normal and plant conditions stabilize.

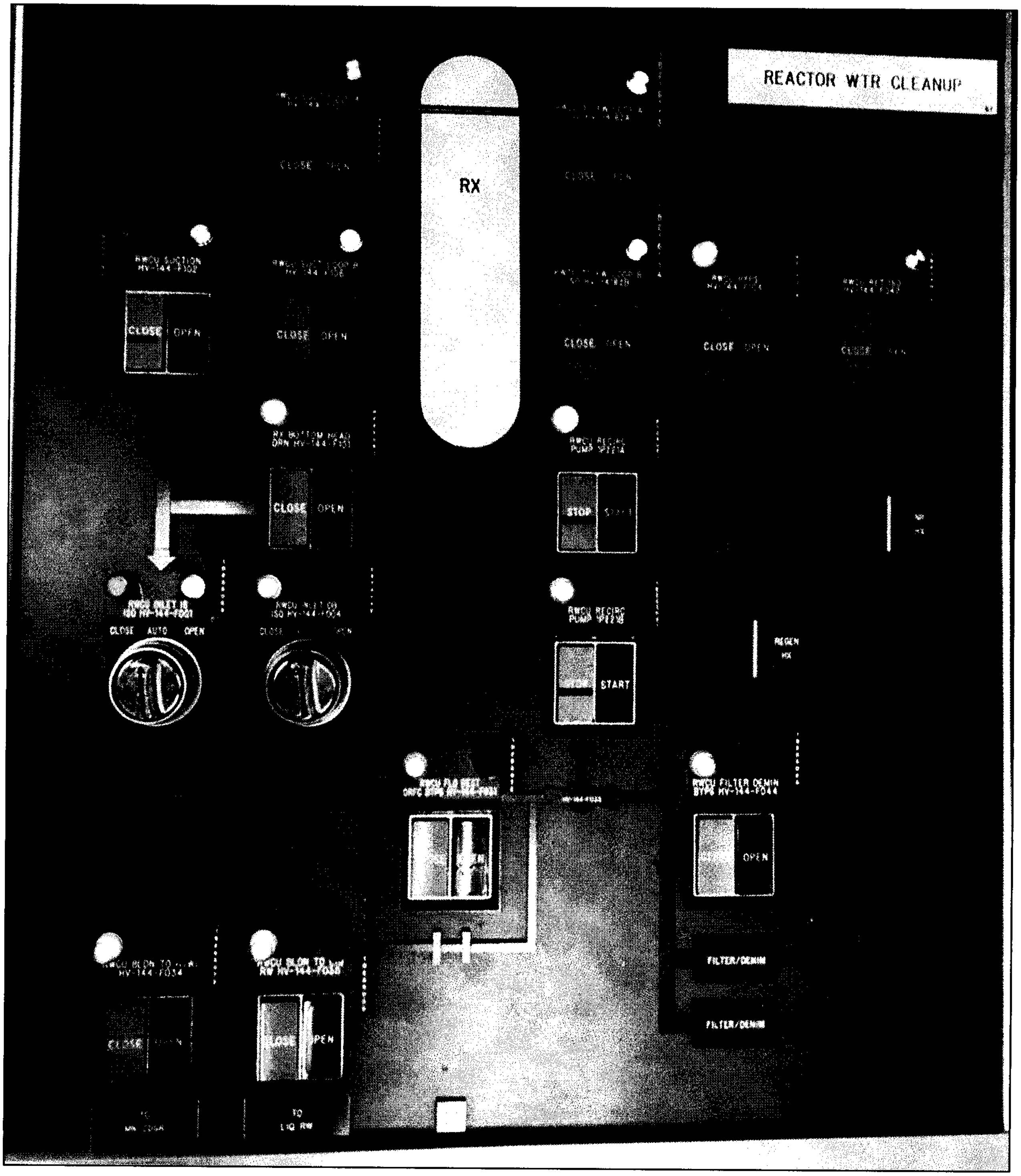
Which one of the following describes the status of HV-144-F001, RWCU INLET IB ISO, and HV-144-F004, RWCU INLET OB ISO?

A. Both valves have responded properly.

B. HV-144-F001 has responded properly, but HV-144-F004 has NOT.

C. HV-144-F004 has responded properly, but HV-144-F001 has NOT.

D. NEITHER valve has responded properly.


REACTOR WTR CLEANUP
RX
RWCU SUCTION HV-144-F102
CLOSE
OPEN
STOP
START
RWCU FILTER DEMIN BYPS HV-144-F044
FILTER/DEMIN
FILTER/DEMIN
TO LIQ RW

Proposed Answer: A

Explanation: RWCU filter demineralizer inlet temperature >145°F causes an isolation signal. This signal is different from many other RWCU isolation signals in that it closes HV-144-F004, but not HV-144-F001. The given indications show HV-144-F004 closed and HV-144-F001 open. Therefore, both valves have responded properly.

B. Incorrect – HV-144-F004 has responded properly. Plausible because this would be correct if temperature had not exceeded 145°F.
C. Incorrect – HV-144-F001 has responded properly. Plausible because this would be correct for most other RWCU isolation signals, which close both valves.
D. Incorrect – Both valves have responded properly. Plausible because this would be the answer chosen if the two valve isolation logics were mistakenly switched or if the indications were misinterpreted.

Technical Reference(s): AR-101-A01

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-061 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 A4.05
	Importance Rating	3.1

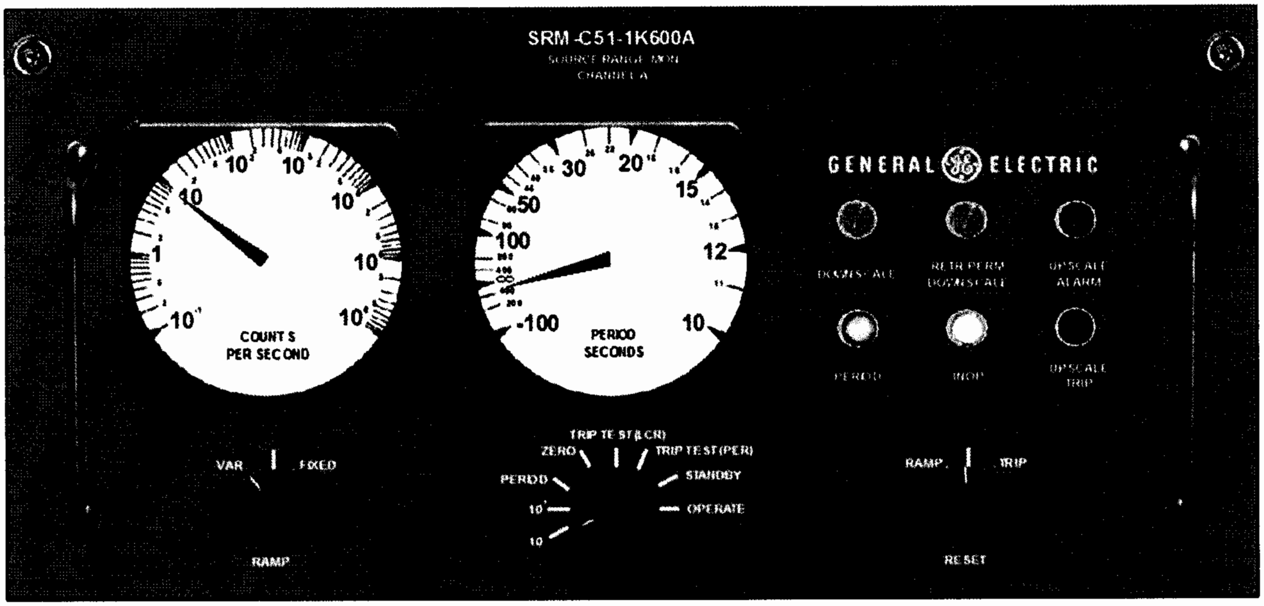
Source Range Monitor

Ability to manually operate and/or monitor in the control room: SRM back panel switches, meters, and indicating lights

Proposed Question: #19

Unit 1 startup is in Mode 4 with the following:

- Preparations are underway for a plant startup.
- The following indications are available for SRM A:



Which one of the following describes the status of SRM A?

SRM A...

A. is properly aligned and operating to support a plant startup.

B. count rate indication is acceptable, but the SRM mode switch is NOT properly positioned.

C. mode switch is properly positioned, but count rate indication is below the minimum acceptable for a startup.

D. mode switch is NOT properly positioned and count rate indication is below the minimum acceptable for a startup.

Proposed Answer: B

Explanation: The given indications show the SRM A mode switch in the "10" position, however it is required to be in the "OPERATE" position. The count rate indication is acceptable and all indicating lights match the given count rate indication (upscales and downscale clear).

A. Incorrect – The given indications show the SRM A mode switch in the "10" position, however it is required to be in the "OPERATE" position. Plausible because in Mode 4 all control rods are still in and candidate could believe the "10" position is the lowest scale required during initial startup (similar to need to place portable radiation monitor on lowest scale).
C. Incorrect – The given indications show the SRM A mode switch in the "10" position, however it is required to be in the "OPERATE" position. Plausible because in Mode 4 all control rods are still in and candidate could believe the "10" position is the lowest scale required during initial startup (similar to need to place portable radiation monitor on lowest scale). The count rate indications (counts and status of upscales/downscale lights) is proper for a startup. Plausible because counts are below the range used in GO-100-002 for withdrawing SRMs (<1E3).
D. Incorrect – The count rate indications (counts and status of upscales/downscale lights) is proper for a startup. Plausible because counts are below the range used in GO-100-002 for withdrawing SRMs (<1E3).

Technical Reference(s): GO-100-102, SI-178-215A, AR-104-C06, TM-OP-078A

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-078A RBO-4

Question Source: Modified Bank - NMP1 2015 Audit #20

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference: Level RO
Tier # 2
Group # 1
K/A # 215003 A4.04
Importance Rating 3.1

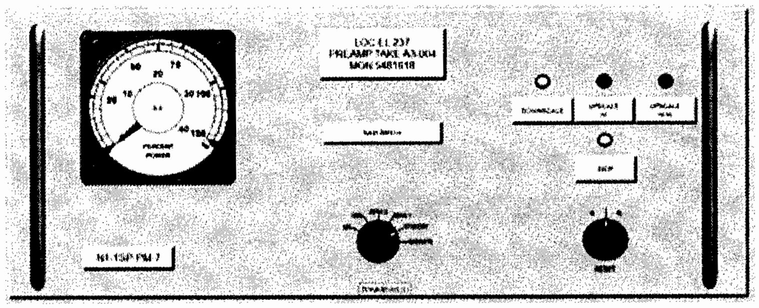
IRM

Ability to manually operate and/or monitor in the control room: IRM back panel switches, meters, and indicating lights

Proposed Question: #20

The plant is in cold shutdown with the following:

- Preparations are underway for a plant startup.
- The following indications are available on G panel for IRM 11:



Which one of the following describes the status of IRM 11?

IRM 11...

A. is properly aligned and operating to support a plant startup.

B. power indication is acceptable, but the IRM mode switch is NOT properly positioned.

C. mode switch is properly positioned, but power indication is below the minimum acceptable for a startup.

D. mode switch is NOT properly positioned and power indication is below the minimum acceptable for a startup.

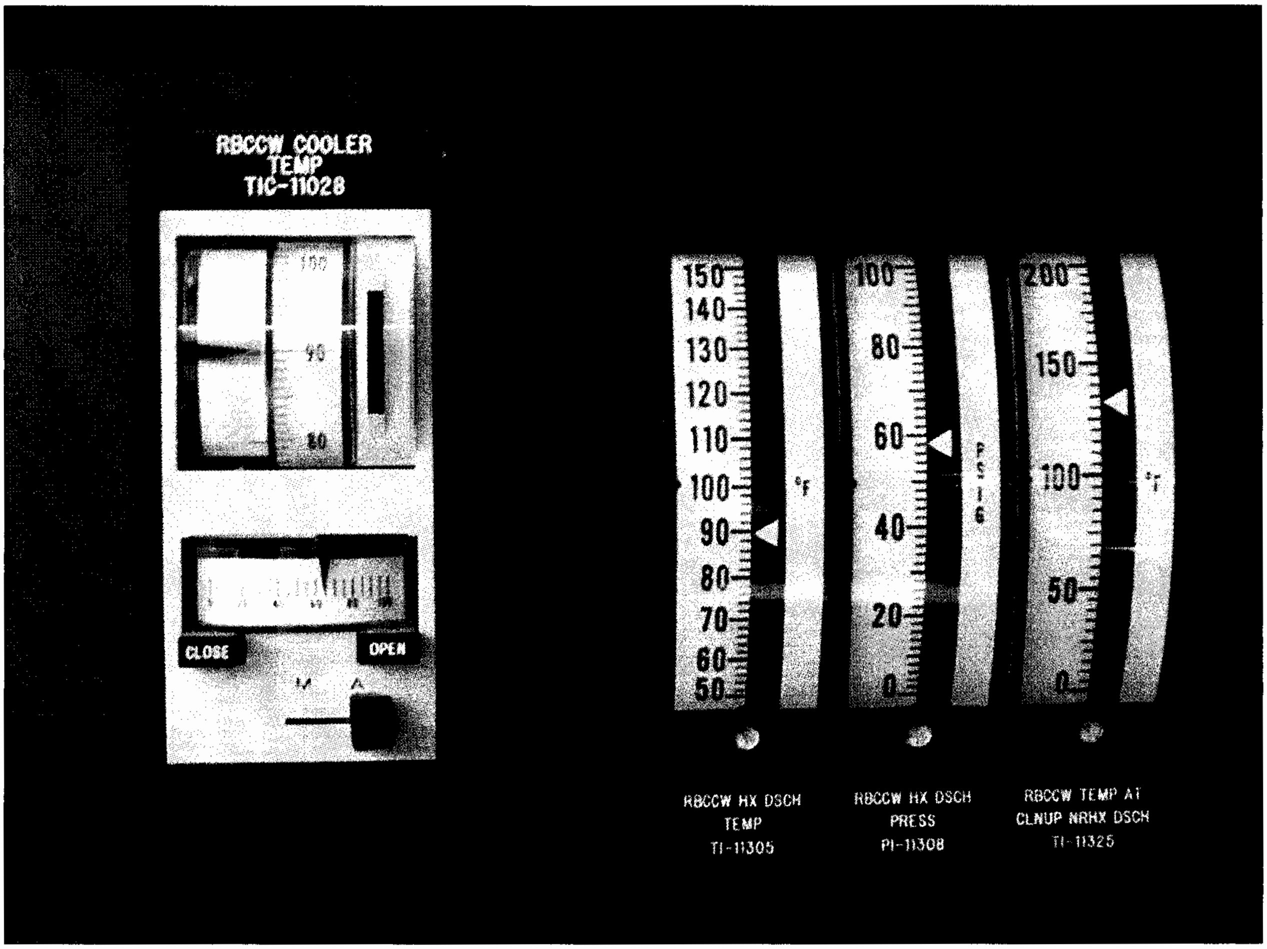
Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 A4.01
	Importance Rating	3.1

Component Cooling Water

Ability to manually operate and/or monitor in the control room: CCW indications and control

Proposed Question: #20

Unit 1 is operating at 100% power with the following RBCCW indications in the Control Room:



Which one of the following describes the status of the RBCCW heat exchanger discharge temperature and pressure indications?

A. Both of these indications are SAT.

B. Heat exchanger discharge pressure is too low, only.

C. Heat exchanger discharge temperature is too high, only.

D. Heat exchanger discharge pressure is too low and heat exchanger discharge temperature is too high.

Proposed Answer: B

Explanation: Normal RBCCW heat exchanger discharge pressure, as indicated in the Control Room, is ~72-82 psig. This given indication of ~59 psig is too low. Normal RBCCW heat exchanger discharge temperature is ~90°F, which matches with the given indication.

A. Incorrect – The given pressure indication of ~59 psig is too low. Plausible because pressure at this point in the system is always lower than the pump discharge pressure to 90-110 psig and the indication is in the upper half of scale.
C. Incorrect – The given pressure indication of ~59 psig is too low. Plausible because pressure at this point in the system is always lower than the pump discharge pressure to 90-110 psig and the indication is in the upper half of scale. The given temperature indication is SAT. Plausible because the third indicator given shows a temperature that would be too high for heat exchanger discharge temperature (even above the 105°F alarm setpoint).
D. Incorrect – Plausible because the third indicator given shows a temperature that would be too high for heat exchanger discharge temperature (even above the 105°F alarm setpoint).

Technical Reference(s): OP-114-001, AR-123-E03

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-014 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 2.4.9
	Importance Rating	3.8

SRVs

Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: #21

A cooldown was in progress on Unit 1 when a loss of coolant accident resulted in the following:

- Drywell pressure is 12 psig, steady.
- Suppression Chamber pressure is 10 psig, steady.
- Reactor pressure is 75 psig, steady.
- Reactor water level is -145", steady, with only Core Spray pump 1A running and injecting.
- NO other injection sources are running or available.
- NO SRVs are currently open.

Which one of the following describes the status and required control of ADS and SRVs, in accordance with EO-100-102, RPV Control?

ADS...

A. will open SRVs, but must be inhibited.

B. will open SRVs and must NOT be inhibited.

C. will NOT open SRVs, but six SRVs must be manually opened.

D. will NOT open SRVs and SRVs are NOT required to be manually opened.

Proposed Answer: D

Explanation: Reactor water level is <-129" and Drywell pressure is >1.72 psig, so the ADS timer has initiated. However, with only one Core Spray pump running, the ECCS pump permissive is not met, so ADS will not actually open SRVs when the timer expires. EO-100-102 does not require manually opening SRVs unless Reactor water level lowers further (<-161" and before -179").

A. Incorrect – The ADS timer has initiated, however ADS will not open SRVs because the ECCS pump permissive is not met with only one Core Spray pump running. Plausible because ADS would open SRVs if only one RHR pump was running or if another Core Spray pump was operating.
B. Incorrect – The ADS timer has initiated, however ADS will not open SRVs because the ECCS pump permissive is not met with only one Core Spray pump running. Plausible because ADS would open SRVs if only one RHR pump was running or if another Core Spray pump was operating.
C. Incorrect – EO-100-102 does not require manually opening SRVs unless Reactor water level lowers further (<-161" and before -179"). Plausible because for other failed actuations, manual backup would be required, however this would be against the requirements of EO-100-102 for ADS control.

Technical Reference(s): TM-OP-83E, EO-100-102

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-83E RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 2.2.12
	Importance Rating	3.7

RCIC

Knowledge of surveillance procedures.

Proposed Question: #22

Unit 1 is operating at 25% power with the following:

- SO-150-002, Quarterly RCIC Flow Verification Test, is being performed this shift.
- Initial Suppression Pool average temperature is 85°F.

Which one of the following identifies the Suppression Pool average temperature threshold that requires securing the test?

A. 90°F

B. 105°F

C. 110°F

D. 120°F

Proposed Answer: B

Explanation: SO-150-002 requires stopping the test if Suppression Pool average temperature reaches 105°F. This is based on the requirements of TS 3.6.2.1 while in Mode 1 with testing in progress that adds heat to the Suppression Pool.

A. Incorrect – 105°F is the correct threshold. Plausible because this is the normal limit per TS 3.6.2.1 when testing that adds hear to the Suppression Pool is not in progress and it is the EO-100-103 entry condition.
C. Incorrect – 105°F is the correct threshold. Plausible because this is the TS 3.6.2.1 LCO limit when shutdown, and this test can be performed in Mode 3. Also plausible because this is the applicable limit in TS 3.6.2.1 for requiring an immediate Reactor scram.
D. Incorrect – 105°F is the correct threshold. Plausible because this is the TS 3.6.2.1 limit for requiring Reactor depressurization.

Technical Reference(s): SO-150-002, TS 3.6.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-50 RBO-7

Question Source: Modified Bank – LOC25 Cert #68

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

LOC25 Cert

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

2.2.39 - Knowledge of less than one hour technical specification action statements for systems.

Proposed Question:	Common 68

Unit 1 is operating at rated power.

RCIC is in service for a scheduled surveillance test.

The following indications are noted:

Drywell average air temperature	130 °F
Suppression Pool level	24 ft 4 in
Suppression Pool average temperature	106 °F

Which of the above indications requires either a Technical Specification immediate action OR must be restored to within Technical Specifications limits within ONE hour to preclude a required action?

A.	**Suppression Pool average temperature, ONLY**
B.	Suppression Pool level, ONLY
C.	Suppression Pool average temperature AND Suppression Pool level, ONLY
D.	Drywell average air temperature, Suppression Pool average temperature AND Suppression Pool Water Level

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	215004 A1.03
Importance Rating	3.4

Source Range Monitor

Ability to predict and/or monitor changes in parameters associated with operating the SOURCE RANGE MONITOR (SRM) SYSTEM controls including: RPS status

Proposed Question: #23

A Unit 1 startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP/STBY.
- Shorting links are installed.
- Control rods are being withdrawn.
- SRMs are indicating as follows:

SRM	Count Rate (cps)	Position	Period (seconds)
A	6×10^4	Fully inserted	30
B	4×10^5	Fully inserted	100
C	3×10^2	Partially withdrawn	100
D	1×10^4	Partially withdrawn	100

Which one of the following describes the automatic response(s), if any, generated by the SRMs?

A. NEITHER a control rod block NOR a half scram occurs.

B. A control rod block occurs, but NO half scram occurs.

C. A control rod block and a half scram occur, but a full scram does NOT occur.

D. A control rod block and a full scram occur.

Proposed Answer: B

Explanation: SRM B >1x10^5 cps (>3.3x10^5 cps per TS) causes a rod block. This would also cause a full scram, but only if the shorting links were removed. With the shorting links installed, only a rod block is received.

A. Incorrect – A rod block is received due to SRM B. Plausible because all other SRMs are below setpoint and SRM B is fully inserted, which affects the downscale rod block, but not the upscale rod block.
C. Incorrect – A half scram is not received because the shorting links are installed. Plausible because if shorting links were removed, then a half scram would be received. Plausible that only a half scram would be received since most scrams have coincident logic and only one SRM is above the upscale setpoint.
D. Incorrect – A full scram is not received because the shorting links are installed. Plausible because if shorting links were removed, then a full scram would be received.

Technical Reference(s): AR-104-B06, AR-104-D06

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-78A RBO-4

Question Source: Modified Bank – LOC23 Cert #5

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

QUESTION	5

Unit 1 is preparing for a reactor startup. The following conditions exist:

- Mode Switch in STARTUP
- All control rods fully inserted
- Shorting links installed
- 24 VDC Load Center 1D672 de-energizes due to a ground fault

Considering ONLY the Source Range Monitors (SRM)...

Which one of the following are the CORRECT automatic responses (if any) generated by the SRMs?

A.	Control rod block, AND a HALF scram.
B.	Control rod block, AND a FULL scram.
C.	**Control rod block, but NO FULL OR HALF scram.**
D.	NO control rod block, AND NO FULL OR HALF scram.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 K3.02
	Importance Rating	4.5

ADS

Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: Ability to rapidly depressurize the reactor

Proposed Question: #24

Unit 1 was operating at 100% power when the following events occurred:

- An un-isolable Reactor Water Cleanup steam leak into the Reactor Building led to the need for an Emergency RPV Depressurization.
- Malfunctions have caused all ADS SRVs, plus 6 other SRVs, to fail closed.
- MSIVs closed due to an invalid isolation signal that has now cleared.
- Main Condenser vacuum is 25" Hgv.

Which one of the following describes the effect the malfunction of the ADS system and SRVs has on the ability to rapidly depressurize the Reactor, in accordance with EO-100-112, Emergency RPV Depressurization?

The "Minimum Number of SRVs Required for Emergency Depressurization" (the number required such that alternate pressure reduction systems are ***NOT required*** to be used by EO-100-112) __(1)__.

Turbine Bypass Valves __(2)__ be used to rapidly depressurize the Reactor.

	(1)	(2)
A.	is available	may
B.	is available	may NOT
C.	is NOT available	may
D.	is NOT available	may NOT

Proposed Answer: C

Explanation: The plant has 16 SRVs and the Minimum Number of SRVs Required for Emergency Depressurization is 5. With 12 SRVs failed closed (all 6 ADS valve plus 6 other SRVs), only 4 are available, which is less than the required 5. With less than 5 SRVs open, EO-100-112 directs use of Table P-2 Systems to rapidly depressurize the Reactor. Table P-2 Systems include Turbine Bypass Valves. Allowance is given to re-open the MSIVs if closed and to override isolation signals if present.

A. Incorrect – The Minimum Number of SRVs Required for Emergency Depressurization is 5. With 12 SRVs unavailable, only 4 remain available. Plausible because 4 SRVs is still enough to rapidly lower Reactor pressure.
B. Incorrect – The Minimum Number of SRVs Required for Emergency Depressurization is 5. With 12 SRVs unavailable, only 4 remain available. Plausible because 4 SRVs is still enough to rapidly lower Reactor pressure. With less than 5 SRVs open, EO-100-112 directs use of Table P-2 Systems to rapidly depressurize the Reactor. Table P-2 Systems include Turbine Bypass Valves. Allowance is given to re-open the MSIVs if closed and to override isolation signals if present. Plausible because MSIVs are closed, such that TBVs cannot be immediately used without taking extra action.
D. Incorrect – With less than 5 SRVs open, EO-100-112 directs use of Table P-2 Systems to rapidly depressurize the Reactor. Table P-2 Systems include Turbine Bypass Valves. Allowance is given to re-open the MSIVs if closed and to override isolation signals if present. Plausible because MSIVs are closed, such that TBVs cannot be immediately used without taking extra action.

Technical Reference(s): EO-100-112 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-83E RBO-7

Question Source: Bank – JAF 9/14 NRC #6

Question History: JAF 9/14 NRC #6

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 A4.01
	Importance Rating	2.6

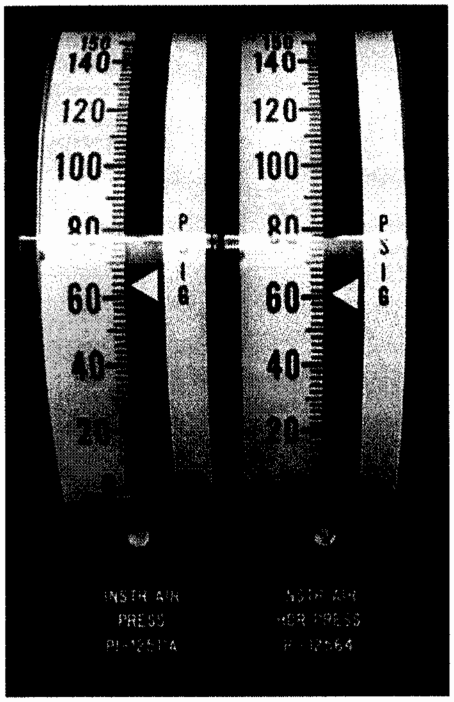
Instrument Air

Ability to manually operate and/or monitor in the control room: Pressure gauges

Proposed Question: #25

Unit 1 is operating at 100% power with the following:

- Instrument Air Compressor (IAC) 1A is operating as the lead compressor.
- IAC 1B is the standby compressor.
- Then, an air leak develops in the plant.
- Control Room air pressure indications are as shown in the following picture, and have been slowly lowering since the leak developed.



Which one of the following describes the status of PCV-12560, Service Air Crosstie to Instrument Air, and the need for a manual Reactor scram based on current plant parameters, in accordance with ON-INSTAIR-101, Loss of Instrument Air?

PCV-12560 is...

A. open. A manual Reactor scram is required.

B. open. A manual Reactor scram is NOT required.

C. closed. A manual Reactor scram is required.

D. closed. A manual Reactor scram is NOT required.

Proposed Answer: A

Explanation: Instrument air pressure indications are below the point at which PCV-12560 opens (~95 psig on PI-12511A, ~85 psig on PI-12564). Instrument air pressure indications are also below the threshold requiring a manual Reactor scram (65 psig).

B. Incorrect – A manual scram is required. Plausible because pressure is still relatively high and close to the threshold requiring a scram.
C. Incorrect – PCV-12560 is open at the given pressures. Plausible because this valve is normally closed and no indication is given that manual actions have been taken.
D. Incorrect – PCV-12560 is open at the given pressures. Plausible because this valve is normally closed and no indication is given that manual actions have been taken. A manual scram is required. Plausible because pressure is still relatively high and close to the threshold requiring a scram.

Technical Reference(s): ON-INSTAIR-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-018 RBO-7

Question Source: Modified Bank – LOC26R NRC #20

Question History:

Question Cognitive Level: Comprehension or Analysis

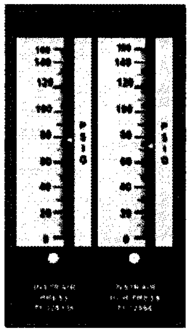
10 CFR Part 55 Content: 55.41(5)

Comments:

Exam	RO	Tier	2	Group	1	Cognitive Level	High	Level of Difficulty	3
K/A		300000 A4.01 Instrument Air				Importance		2.6	
Statement		Ability to manually operate and/or monitor in the control room: Pressure gauges							

QUESTION 20

Unit 1 is operating at 100% power with the following:



- Instrument Air Compressor (IAC) 1A is operating as the lead compressor.
- IAC 1B is the standby compressor.
- Then, an air leak develops in the plant.
- Control Room air pressure indications are as shown in the following picture, and have been slowly lowering since the leak developed.

Which one of the following describes the status of IAC 1B and PCV-12560, Service Air Crosstie to Instrument Air, and the need for a manual Reactor scram based on current plant parameters, in accordance with ON-118-001?

A.	IAC 1B is running. PCV-12560 is closed. A manual Reactor scram is NOT required.
B.	IAC 1B remains in standby. PCV-12560 is open. A manual Reactor scram is NOT required.
C.	IAC 1B is running. PCV-12560 is open. A manual Reactor scram is NOT required.
D.	IAC 1B is running. PCV-12560 is closed. A manual Reactor scram is required.

Proposed Answer	C

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	400000 A3.01
Importance Rating	3.0

Component Cooling Water

Ability to monitor automatic operations of the CCWS including: Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS

Proposed Question: #26

Unit 1 is operating at 100% power when a steam leak in the Drywell results in the following:

- Drywell pressure is 2.0 psig, up slow.
- Drywell temperature is 195°F, up slow.

Which one of the following identifies the implications of these conditions on cooling water to the Reactor Recirculation Pumps (RRPs)?

Cooling...

A. is lost to the motor winding, bearing, and seal coolers.

B. is lost to the motor winding coolers, but remains to the bearing and seal coolers.

C. is lost to the bearing and seal coolers, but remains to the motor winding coolers.

D. to the motor winding, bearing, and seal coolers automatically swaps from the normal to the alternate source.

Proposed Answer: A

Explanation: RBCCW and RBCW piping that penetrates the primary containment will isolate when DW pressure exceeds 1.72 psig, stopping flow to RRP motor winding coolers (RBCW) and bearing and seal coolers (RBCCW).

B. Incorrect – RBCCW isolates, not just RBCW, which causes loss of cooling water to the bearing and seal coolers also. Plausible because these loads are supplied by separate cooling water sources.
C. Incorrect – RBCW isolates, not just RBCCW, which causes loss of cooling water to the motor winding coolers also. Plausible because these loads are supplied by separate cooling water sources.
D. Incorrect – Both RBCW and RBCCW isolate, and no alternate cooling water source is available for automatic swap. Plausible because RBCW loads can be supplied automatically by RBCCW under other circumstances.

Technical Reference(s): ON-CONTISOL-101, ON-RBCCW-101, ON-RBCW-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-014

Question Source: Modified Bank – LOC23 NRC #18

Question History: LOC23 NRC #18

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

QUESTION	18

Unit 1 was operating at full power with all systems in a normal full power lineup when a steam leak occurs inside the drywell resulting in:

- Drywell pressure 2.0 psig, up slow
- Drywell temperature 195°F, up slow

In this condition, operators should closely monitor __(1)__ because __(2)__.

A.	(1) Reactor Recirc Pump (RRP) motor winding temperatures (2) Reactor Building Closed Cooling Water flow to the RRP motor winding coolers automatically isolated
B.	(1) Reactor Recirc Pump (RRP) motor winding temperatures (2) RRP motor winding cooling automatically shifted from Reactor Building Chilled Water to Reactor Building Closed Cooling Water
C.	**(1) Reactor Recirc Pump (RRP) motor bearing and seal temperatures** **(2) Reactor Building Closed Cooling Water flow to the Recirc pump bearing and seal coolers automatically isolated**
D.	(1) Reactor Recirc Pump (RRP) motor bearing and seal temperatures (2) RRP bearing and seal cooling automatically shifted from Reactor Building Chilled Water to Reactor Building Closed Cooling Water

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	214000 K1.05
	Importance Rating	3.3

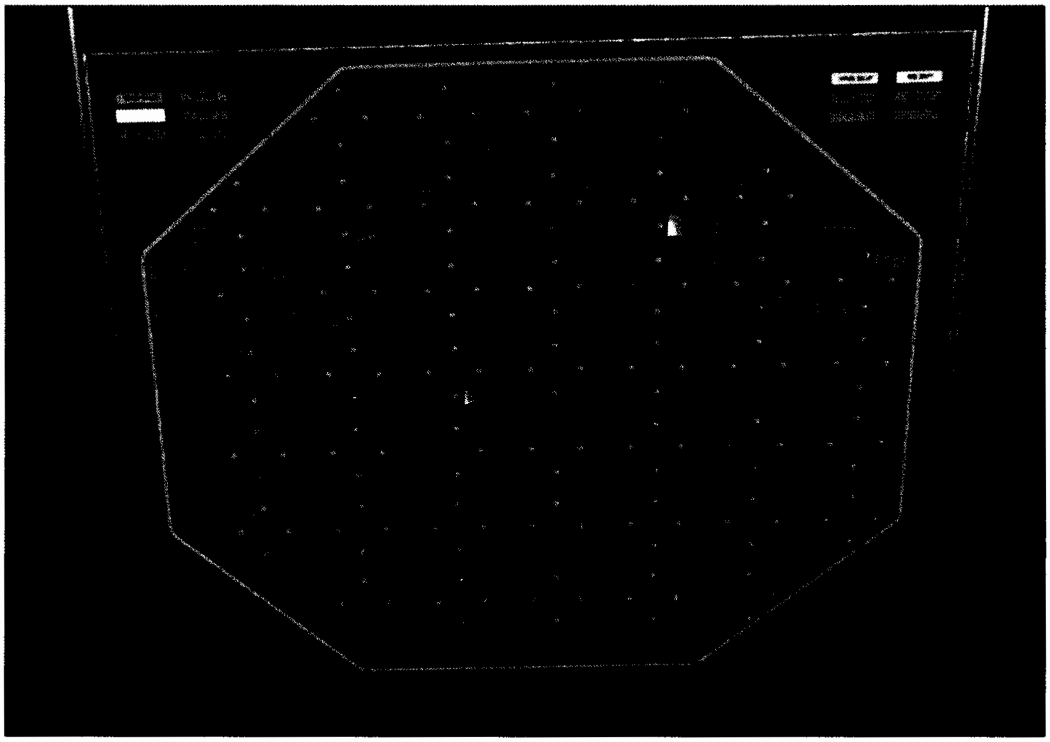
RPIS

Knowledge of the physical connections and/or cause-effect relationships between ROD POSITION INFORMATION SYSTEM and the following: Full core display: Plant-Specific

Proposed Question: #27

Unit 1 has experienced a Reactor scram with the following:

- All required ON-SCRAM-101, Reactor Scram, Immediate Operator Actions have been taken.
- EO-100-102, RPV Control, has just been entered.
- The full core display is shown below:



Which one of the following describes the status of the scram and the need for entry into EO-100-113, Level/Power Control?

A. All rods are full in. EO-100-113 entry is NOT required.

B. All rods are NOT full in. EO-100-113 entry is required.

C. All rods are NOT full in. EO-100-113 entry is NOT required.

D. All rods are NOT full in. More information is required to determine the need for EO-100-113 entry.

Proposed Answer: B

Explanation: The given picture shows the Full Core Display in FULL IN/OUT mode. The FULL IN/OUT display shows that all control rods, except 3, are fully inserted (green lights on). The other 3 rods have neither a green nor red light on, indicating that they are at some position other than 00 or 48. Since more than one rod is beyond 00, EO-100-102 requires entry into EO-100-113.

A. Incorrect – 3 rods are not full in. Plausible because most rods have green indication and the 3 other rods lack red indication.
C. Incorrect – EO-100-113 entry is required. Plausible because this would be correct if only 1 rod were not full in.
D. Incorrect – EO-100-113 entry is required. Plausible because this would be correct if the Maximum Subcritical Banked Withdrawal Position (MSBWP) was greater than 00.

Technical Reference(s): EO-100-102

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-056 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201001 K2.03
	Importance Rating	3.5

CRD Hydraulic

Knowledge of electrical power supplies to the following: Backup SCRAM valve solenoids

Proposed Question: #28

Which one of the following describes the electrical power supply and function of the backup scram valve solenoids?

A. 120 VAC, energize to cause a scram

B. 120 VAC, de-energize to cause a scram

C. 125 VDC, energize to cause a scram

D. 125 VDC, de-energize to cause a scram

Proposed Answer: C

Explanation: The backup scram valve solenoids are powered from 125 VDC sources (1D614 and 1D624). The solenoids are normally de-energized. The solenoids are energized when required to cause a scram.

A. Incorrect – The power supply is 125 VDC. Plausible because the normal scram solenoids are AC powered.
B. Incorrect – The power supply is 125 VDC. Plausible because the normal scram solenoids are AC powered. The solenoids are energized to cause a scram. Plausible because the normal scram solenoids are de-energized to cause a scram.
D. Incorrect – The solenoids are energized to cause a scram. Plausible because the normal scram solenoids are de-energized to cause a scram.

Technical Reference(s): TM-OP-055H

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-055H RBO-3

Question Source: Bank – JAF 4/14 NRC #28

Question History: JAF 4/14 NRC #28

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201002 K3.01
	Importance Rating	3.4

RMCS

Knowledge of the effect that a loss or malfunction of the REACTOR MANUAL CONTROL SYSTEM will have on following: Ability to move control rods

Proposed Question: #29

Unit 1 is operating at 100% power with the following:

- Annunciator AR-103-H04, RDCS INOP ROD BLOCK, alarms.
- I&C reports that the control rod 18-41 Transponder Card has failed.
- Control rod 18-41 is at position 12.

Which one of the following describes the effect on the ability to move control rods with RDCS?

A. Control rod 18-41 rod withdrawal is blocked, but not rod insertion. The ability to move all other control rods is unaffected.

B. Control rod 18-41 rod withdrawal and insertion is blocked. The ability to move all other control rods is unaffected.

C. Rod withdrawal is blocked for all control rods, but not rod insertion.

D. Rod withdrawal and insertion are blocked for all control rods.

Proposed Answer: D

Explanation: The RDCS inop rod block can be caused by failure of a single control rod transponder card, but it enforces both withdraw and insert rod blocks for all control rods.

A. Incorrect – Rod withdrawal and insertion are blocked for all control rods. Plausible that only control rod 18-41 would be affected because only its transponder card has failed. Plausible that only withdrawal would be blocked because most rod blocks do not block insertion.
B. Incorrect – Rod withdrawal and insertion are blocked for all control rods. Plausible that only control rod 18-41 would be affected because only its transponder card has failed.
C. Incorrect – Rod withdrawal and insertion are blocked for all control rods. Plausible that only withdrawal would be blocked because most rod blocks do not block insertion.

Technical Reference(s): AR-103-H04

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-056 RBO-6

Question Source: Modified Bank – LOC24 Cert #34

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

QUESTION	34

Unit 1 is at full power when annunciator AR-103-001 (H04) "RDCS INOP ROD BLOCK" is received.

I&C reports that control rod 18-41 Transponder Card has failed.

Which one of the following is CORRECT?

Rod motion is blocked for __(1)__, AND __(2)__.

A.	(1) ONLY control rod 18-41 (2) rod motion can be restored by replacing the transponder card AND resetting the Rod Drive Control System
B.	(1) ONLY control rod 18-41 (2) rod motion can be restored by bypassing rod 18-41 at the Rod Drive Control Cabinet, replacing the transponder card, THEN restoring the rod bypass switch to normal
C.	(1) ALL control rods (2) motion for ALL rods EXCEPT rod 18-41 can be restored by bypassing rod 18-41 at the Rod Drive Control Cabinet ONLY
D.	**(1) ALL control rods** **(2) motion for ALL rods EXCEPT rod 18-41 can ONLY be restored by bypassing rod 18-41 at the Rod Drive Control Cabinet AND resetting the Rod Drive Control System**

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201006 K4.05
	Importance Rating	2.8

RWM

Knowledge of ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: Substitute rod position data: P-Spec(Not-BWR6)

Proposed Question: #30

Unit 1 is operating at 5% power during a startup with the following:

- Control rod 30-35 was withdrawn to its withdraw limit of position 12.
- The control rod 30-35 position 12 reed switch then failed.
- A substitute rod position was entered into the Rod Worth Minimizer (RWM) in accordance with ON-CRD-101, Control Rod Malfunction.
- On a subsequent rod group, control rod 30-35 is withdrawn to position 14, which has a good reed switch.
- NO other operator actions have been taken.

Which one of the following describes the control rod 30-35 rod position indication on the RWM and RTime displays (including OD-7) when the control rod is at position 14 indication on the four rod display?

The RWM and RTime...

A. show "UNK".

B. show a blank rod position.

C. continue to display the substitute value of 12.

D. update to display the actual current position of 24.

Proposed Answer: C

Explanation: Once substitute rod position information is input to the RWM, this information is displayed until deleted. Therefore, even once the control rod is moved to position 14, it will still display as position 12 on the RWM and RTime displays.

A. Incorrect – The control rod will continue to display as position 12 on the RWM and RTime displays. Plausible because UNK is the indication received when the rod position is initially unknown, and a discrepancy now does exist between the actual indication and the substitute value.
B. Incorrect – The control rod will continue to display as position 12 on the RWM and RTime displays. Plausible because blank is the indication received on the 4-rod display when the rod position is initially unknown, and a discrepancy now does exist between the actual indication and the substitute value.
D. Incorrect – The control rod will continue to display as position 12 on the RWM and RTime displays. Plausible because the rod is now at a position with good indication and RPIS will provide the proper position signal.

Technical Reference(s): TM-OP-031D RBO-4

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-031D

Question Source: Bank – Vision SYSID 33573

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	234000 K5.01
	Importance Rating	2.9

Fuel Handling Equipment

Knowledge of the operational implications of the following concepts as they apply to FUEL HANDLING EQUIPMENT: Crane/hoist operation

Proposed Question: #31

Unit 2 is in MODE 5 with the following conditions present:

- Reactor Mode Switch is in REFUEL.
- Unit 2 Refueling platform Reactor Select Switch selected to NORM.
- Unit 2 Refueling platform is positioned over the Unit 2 Reactor.
- Refuel switch #1 is activated.
- Fuel grapple is UNLOADED.
- Monorail hoist is UNLOADED.
- Frame mounted hoist is UNLOADED.
- Control rod 10-23 is at position 48.

Which one of the following changes would prevent reverse refueling platform motion?

A. Refuel switch #2 is activated.

B. Refuel switch #1 is DE-activated.

C. The Frame mounted hoist is loaded with 750 lbs.

D. Control rod 10-27 is selected while at position 00.

Proposed Answer: C

Explanation: Reverse movement is blocked if selected to NORM and a control rod is withdrawn, refuel switch #1 is activated, and EITHER fuel grapple loaded >550lbs, OR frame hoist >500lbs; OR monorail hoist >500lbs. Raising the frame hoist load above this limit completes the REVERSE movement block circuit. It also prevents raising the hoist any further.

A. Incorrect – Refuel switch #2 is not in this circuit and conditions to enable it to prevent reverse motion are not present. Plausible if candidate incorrectly believes that refuel switch #2 provides input to this circuit.
B. Incorrect – De-activation of refuel switch #1 would indicate the bridge is no longer above the Reactor and would permit reverse motion. Plausible if candidate does not correctly recall the purpose and function of refuel switch #1, and because this would be an adverse condition.
D. Incorrect – An additional selected rod will not affect the circuit. Plausible due to similarity to one-rod out permissive interlock in RMCS.

Technical Reference(s): TM-OP-081A

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-081

Question Source: Bank – LOC23 NRC #31

Question History: LOC23 NRC #31

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(13)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	239001 K6.08
	Importance Rating	3.3

Main and Reheat Steam

Knowledge of the effect that a loss or malfunction of the following will have on the MAIN AND REHEAT STEAM SYSTEM: Main condenser vacuum

Proposed Question: #32

Unit 1 is operating at 30% power with the following:

- Main Condenser pressure begins to rise due to air in-leakage.
- The Reactor mode switch is placed in SHUTDOWN.
- No other operator actions are performed.
- Reactor water level reaches a low value of +10" following the scram.
- Main Condenser pressure is 21.2" Hga, stable.

Which one of the following describes the availability of Feedwater pumps for injection and Turbine Bypass Valves (TBVs) for pressure control?

Feedwater pumps are...

A. available for injection. TBVs are available for pressure control.

B. available for injection. TBVs are NOT available for pressure control.

C. NOT available for injection. TBVs are available for pressure control.

D. NOT available for injection. TBVs are NOT available for pressure control.

Proposed Answer: D

Explanation: Multiple actuations occur based on rising Main Condenser pressure. At 7.5" Hga, the Main Turbine trips. At 11.8" Hga, the Feed pump turbines trip, therefore Feedwater pumps are unavailable for injection. At 19.0" Hga, the MSIVs trip unless the low vacuum trip is bypassed. This trip is bypassed if the Mode Switch is in SHUTDOWN, TSVs are closed, and Main Condenser Low Vacuum Bypass switches are in BYPASS. This last condition is not met, so MSIVs are closed. This makes TBVs unavailable for pressure control, even though vacuum has not yet degraded to the TBV trip setpoint of 22.2" Hga.

A. Incorrect – Feedwater pumps are not available for injection because Main Condenser pressure is >11.8" Hga. Plausible because Reactor water level remained above the level that would automatically close MSIVs and Condensate remains available for injection through Feedwater valves. TBVs are not available for pressure control because MSIVs automatically closed when Main Condenser pressure rose >19.0" Hga. Plausible because Reactor water level remained above the level that would automatically close MSIVs and Main Condenser pressure is still below the trip setpoint for the TBVs.
B. Incorrect – Feedwater pumps are not available for injection because Main Condenser pressure is >11.8" Hga. Plausible because Reactor water level remained above the level that would automatically close MSIVs and Condensate remains available for injection through Feedwater valves.
C. Incorrect – TBVs are not available for pressure control because MSIVs automatically closed when Main Condenser pressure rose >19.0" Hga. Plausible because Reactor water level remained above the level that would automatically close MSIVs and Main Condenser pressure is still below the trip setpoint for the TBVs.

Technical Reference(s): ON-VACUUM-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-083 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290003 A1.03
	Importance Rating	2.6

Control Room HVAC

Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROOM HVAC controls including: Area temperatures

Proposed Question: #33

CREOASS Train A is operating in the Pressurization/Filtration mode, when the following occurs:

- AR-029-D01, EMERG OA CHARC TEMP HI/HI-HI, alarms.
- CREOASS Train A Outlet Air Temp indicates 455°F, up slow.

Which one of the following describes the response and/or required control of CREOASS Train A, in accordance with AR-029-D01?

CREOASS Train A...

A. remains in service and is required to be isolated.

B. remains in service and is NOT required to be isolated.

C. automatically isolates and deluge automatically initiates.

D. automatically isolates and deluge must be manually initiated.

Proposed Answer: C

Explanation: AR-029-D01 alarms on high temperature at 190°F and high-high temperature at 450°F. Since temperature is above 450°F, the high-high condition is present. This causes CREOASS Train A to automatically isolate and deluge to automatically initiate.

A. Incorrect – CREOASS automatically isolates on high-high temperature of 450°F. Plausible because this alarm is also received at a lower high temperature that does not result in automatic isolation but requires manual isolation.
B. Incorrect – CREOASS automatically isolates on high-high temperature of 450°F. Plausible because this alarm is also received at a lower high temperature that does not result in automatic isolation.
D. Incorrect – Water deluge automatically initiates. Plausible that manual action would be required to initiate water deluge since adding water to a charcoal filter unnecessarily is an adverse occurrence and manual initiation capability is provided.

Technical Reference(s): AR-029-D01

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-030 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201003 A2.07
	Importance Rating	3.1

Control Rod and Drive Mechanism

Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of CRD drive water flow

Proposed Question: #34

Unit 1 is operating at 100% power with the following CRD system parameters:

- CRD system flow is 63 gpm, steady.
- CRD charging header pressure is 1475 psig, steady.
- CRD drive water flow is 0 gpm.
- CRD drive water differential pressure is 200 psid, steady.
- CRD cooling water flow is 63 gpm, steady.
- CRD cooling water differential pressure is 15 psid, steady.

Which one of the following describes the negative impact of these conditions and the required action to correct these conditions?

	Impact of These Conditions	Throttle PV-146-F003, CRD WTR PRESS CTL STATION VLV, further...
A.	Improper CRDM cooling	open.
B.	Improper CRDM cooling	closed.
C.	Improper control rod drive speed	open.
D.	Improper control rod drive speed	closed.

Proposed Answer: D

Explanation: All conditions are normal except drive water differential pressure is low (below normal 250 psid). This would cause slower than normal control rod drive speed. This is corrected by further closing PV-146-F003, which raises upstream pressure to the drive water header.

Note: The question meets the K/A because low drive water differential pressure during standby conditions directly results in low drive water flow during control rod movement.

A. Incorrect – Cooling water header flow and pressure are normal. Plausible if candidate does not remember normal values for cooling water header flow and pressure. Also plausible because cooling water header is downstream of PV-146-F003.
B. Incorrect – Cooling water header flow and pressure are normal. Plausible if candidate does not remember normal values for cooling water header flow and pressure. Also plausible because cooling water header is downstream of PV-146-F003.
C. Incorrect – PV-146-F003 must be throttled further closed, not open, to raise drive water pressure. Plausible because if this valve were upstream of the drive water header (as are most pressure control valves upstream of the pressure they are intended to control), then it would need to be opened to raise pressure.

Technical Reference(s): OP-155-001, M-146

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-055H RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	241000 A3.09
	Importance Rating	3.3

Reactor/Turbine Pressure Regulator

Ability to monitor automatic operations of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM including: Control/governor valve operation

Proposed Question: #35

Unit 1 is operating at 95% power with the following:

- EHC pressure regulator channel A is in control.
- EHC pressure regulator channel B is set as the backup pressure regulator.

Then, the Pressure Averaging Manifold (PAM) pressure signal to just EHC pressure regulator channel A drifts to 0 psig.

Which one of the following describes the response of Turbine Control Valves (TCVs) and Reactor pressure?

TCVs...

A. close further until Reactor pressure causes a scram.

B. open further until Reactor pressure causes an MSIV isolation.

C. close further and stabilize Reactor pressure without causing a scram.

D. open further and stabilize Reactor pressure without causing an MSIV isolation.

Proposed Answer: C

Explanation: With EHC pressure regulator A initially in control, pressure setpoint bias is set such that EHC pressure regulator B will control Reactor pressure approximately 3 psig higher than A. When the PAM pressure signal fails low to EHC pressure regulator A, it begins to close TCVs to stop the sensed drop in pressure. This causes actual Reactor pressure and PAM pressure to rise. As EHC pressure regulator B senses this rise in pressure, its input summer error signal grows and overcomes the error signal from EHC pressure regulator A. EHC logic is setup so that the pressure regulator with the higher error signal controls pressure. Therefore, EHC pressure regulator B comes into control and stabilizes Reactor pressure approximately 3 psig higher than the initial value. This prevents a Reactor scram on high pressure, which would otherwise occur.

A. Incorrect – EHC pressure regulator B limits the pressure rise to approximately 3 psig. Plausible because on the opposite failure, the Reactor pressure change would cause a scram on MSIV closure.
B. Incorrect – TCVs close, not open. Plausible because this would be the response if pressure set drifted low. EHC pressure regulator B limits the pressure rise to approximately 3 psig. Plausible because on the opposite failure, the Reactor pressure change would cause MSIV closure.
D. Incorrect – TCVs close, not open. Plausible because this would be the response if pressure set drifted low.

Technical Reference(s): TM-OP-093L

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-093L RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	288000 A4.01
	Importance Rating	3.1

Plant Ventilation

Ability to manually operate and/or monitor in the control room: Start and stop fans

Proposed Question: #36

Unit 1 is operating at 100% power with the following:

- Reactor Building Zone 1 Exhaust Fan 1V205A is in standby.
- Reactor Building Zone 1 Exhaust Fan 1V205B is running.

Then, Fan 1V205B Dsch Dmp PDD-17578B fails closed.

Which one of the following describes the resulting status of Fans 1V205A and 1V205B one (1) minute later?

	Fan 1V205A	Fan 1V205B
A.	Standby	Running
B.	Standby	Tripped
C.	Running	Running
D.	Running	Tripped

Proposed Answer: D

Explanation: When Fan 1V205B Dsch Dmp PDD-17578B fails closed, a low flow condition occurs (<8,100 cfm). After a 30 second time delay, this trips the running fan (1V205B) and automatically starts the standby fan (1V205A).

A. Incorrect – 1V205A automatically starts. Plausible because this fan is initially in standby, no manual action is taken, and there is a time delay before it starts. 1V205B is tripped. Plausible because the fan is initially running, there is no direct problem with the fan, and there is a time delay in its trip.
B. Incorrect – 1V205A automatically starts. Plausible because this fan is initially in standby, no manual action is taken, and there is a time delay before it starts.
C. Incorrect – 1V205B is tripped. Plausible because the fan is initially running, there is no direct problem with the fan, and there is a time delay in its trip.

Technical Reference(s): LA-1275-D02, TM-OP-034

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-034 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202002 2.1.28
	Importance Rating	4.1

Recirculation Flow Control

Knowledge of the purpose and function of major system components and controls.

Proposed Question: #37

Unit 1 is operating at 90% power with the following:

- Condensate pump 1A trips.
- Reactor water level reaches a low of +21" during the transient before recovering.

Which one of the following describes the response of Recirculation pump flow control?

Recirculation pumps...

A. remain at the pre-transient speed.

B. runback to 48% speed.

C. runback to 30% speed.

D. rundown of 10% speed.

Proposed Answer: B

Explanation: Speed Limiter #2 initiates due to the Condensate pump trip. This runs Recirc pump speeds back to 48%.

A. Incorrect – Condensate pump trip is one of the conditions that causes Speed Limiter #2 to initiate, which runs Recirc pumps back to 48% speed. Initial Recirc pump speed is >48% to be at 90% power. Plausible because if initial Reactor power was low enough, Recirc pump speeds would be <48% and would remain at the pre-transient value. Also plausible because numerous other runback conditions are not met, including Reactor water level staying above +13".
C. Incorrect – Speed Limiter #2 initiates due to the Condensate pump trip. This runs Recirc pump speeds back to 48%. Plausible because Speed Limiter #1 causes a 30% runback and one condition that initiates Speed Limiter #1 is low Reactor water level.
D. Incorrect – Speed Limiter #2 initiates due to the Condensate pump trip. This runs Recirc pump speeds back to 48%. Plausible because other conditions do cause a rundown of 10% speed (excessive Feed demand, low RFP suction pressure).

Technical Reference(s): AR-102-C04

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-064 RBO-4

Question Source: Modified Bank – JAF 3/12 NRC #56

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO/SRO
	Tier #	2
	Group #	2
	K/A #	202001 A1.01
	Importance Rating	3.6/3.5

Recirculation: Ability to predict and/or monitor changes in parameters associated with operating the RECIRCULATION SYSTEM controls including: Recirculation pump flow.

Proposed Question: 56

The Plant is operating at 75% power with the following conditions:

- 'A' Recirculation Pump (RWR) Scoop Tube "locked".
- 'A' RWR Scoop Tube Auto Unlock: On

The following events occur:

- 'B' Reactor Feedwater Pump Turbine (RFT) Bearing Oil pressure: 3 psig and steady.
- RPV level: 192 inches and steady

Which one of the following choices correctly lists the status of the 'A' RWR MG set?

A.	locked
B.	**44% speed**
C.	30% speed
D.	tripped

Examination Outline Cross-Reference:	Level	RO/SRO
	Tier #	2
	Group #	2
	K/A #	202001 A1.01
	Importance Rating	3.6/3.5

Recirculation: Ability to predict and/or monitor changes in parameters associated with operating the RECIRCULATION SYSTEM controls including: Recirculation pump flow.

Proposed Question: 56

The Plant is operating at 75% power with the following conditions:

- 'A' Recirculation Pump (RWR) Scoop Tube "locked".
- 'A' RWR Scoop Tube Auto Unlock: On

The following events occur:

- 'B' Reactor Feedwater Pump Turbine (RFT) Bearing Oil pressure: 3 psig and steady.
- RPV level: 192 inches and steady

Which one of the following choices correctly lists the status of the 'A' RWR MG set?

A.	locked
B.	**44% speed**
C.	30% speed
D.	tripped

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	290002 2.4.20
	Importance Rating	3.8

Reactor Vessel Internals

Knowledge of the operational implications of EOP warnings, cautions, and notes.

Proposed Question: #38

Unit 1 has experienced a transient with the following:

- Reactor injection has been lost.
- EO-100-102, RPV Control, is being executed.
- Reactor injection is being re-established per step RC/L-3:

RESTORE and **MAINTAIN** RPV water lvl between **+13 in.** and **+54 in.** using one or more Preferred Injection Systems (**TABLE L-1**)
Supply 125 VDC loads as necessary IAW ES-002-001

RC/L-3

Which one of the following identifies an operational concern with using HPCI to inject to the Reactor vessel under these conditions, in accordance with EO-100-102?

A. Thermal shock of the Recirculation loops

B. Thermal shock at Feedwater penetrations

C. Power excursions due to injecting cold water

D. Power excursions due to rapidly raising water level

Proposed Answer: B

Explanation: EO-100-102 Table L-1 includes HPCI as a preferred injection system and references caution 11 for HPCI. Caution 11 states, "HPCI injections cause thermal shock to RPV at Feedwater penetrations."

A. Incorrect – HPCI injection enters the Reactor through Feedwater penetration, not in Recirculation loops. Plausible because other injection sources (RHR pumps) inject to the Recirculation loops.
C. Incorrect – Since EO-100-102 step RC/L-3 is being executed, all control rods are in. Therefore, power excursions are not the concern. Plausible because power excursions from injection of HPCI are a concern in portions of EO-100-113.
D. Incorrect – Since EO-100-102 step RC/L-3 is being executed, all control rods are in. Therefore, power excursions are not the concern. Plausible because power excursions from injection of HPCI are a concern while operating at power.

Technical Reference(s): EO-100-102

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-052 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EK1.01
	Importance Rating	4.1

High Drywell Pressure

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: Drywell integrity: Plant-Specific

Proposed Question: #39

Unit 1 has experienced a loss of coolant accident with the following:

- Reactor water level is -160", up slow, with Condensate pump B injecting.
- Core Spray pump A and HPCI have just been made available for injection.
- HPCI suction is aligned to the CSTs and CANNOT be swapped to the Suppression Pool.
- NO other injection sources are available.
- Drywell pressure is 63 psig and slowly rising.

Which one of the following identifies which injection source(s) is(are) preferred due to the high Drywell pressure, in accordance with EO-100-102, RPV Control?

A. HPCI is the preferred injection source

B. Core Spray pump A is the preferred injection source

C. Condensate pump B is the preferred injection source

D. HPCI and Condensate pump B are the preferred injection sources

Proposed Answer: B

Explanation: EO-100-102 step RC/L-2 states, "If Primary Containment pressure cannot be maintained less than 65 psig, then stop injection in the RPV from sources external to the Primary Containment not required for adequate core cooling." Core Spray injects from a source internal to the Primary Containment (Suppression Pool), whereas Condensate and HPCI inject from sources external to the Primary Containment (Hotwell / CSTs). Therefore, with pressure approaching 65 psig, Core Spray is the preferred injection source.

A. Incorrect – Core Spray injects from a source internal to the Primary Containment (Suppression Pool), whereas HPCI injects from a source external to the Primary Containment (CSTs). Therefore, with pressure approaching 65 psig, Core Spray is the preferred injection source. Plausible because HPCI can injection at higher Reactor pressures, has finer control, and is from a relatively cleaner source. Also plausible because in some situations, HPCI is preferred over Core Spray (ATWS).
C. Incorrect – Core Spray injects from a source internal to the Primary Containment (Suppression Pool), whereas Condensate injects from a source external to the Primary Containment (Hotwell / CSTs). Therefore, with pressure approaching 65 psig, Core Spray is the preferred injection source. Plausible because Condensate is already injecting, has finer control, and is from a relatively cleaner source. Also plausible because in some situations, Condensate is preferred over Core Spray (ATWS).
D. Incorrect – Core Spray injects from a source internal to the Primary Containment (Suppression Pool), whereas Condensate injects from a source external to the Primary Containment (Hotwell / CSTs). Therefore, with pressure approaching 65 psig, Core Spray is the preferred injection source. Plausible because Condensate is already injecting, has finer control, and is from a relatively cleaner source. Plausible because HPCI can injection at higher Reactor pressures, has finer control, and is from a relatively cleaner source. Also plausible because in some situations, Condensate and HPCI are preferred over Core Spray (ATWS).

Technical Reference(s): EO-100-102

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EK1.02
	Importance Rating	3.5

Suppression Pool High Water Temperature

Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Steam condensation

Proposed Question: #40

Unit 1 has experienced a Reactor scram with the following:

- MSIVs are closed.
- Reactor water level is +20", up slow, with RCIC injecting.
- Reactor pressure is 975 psig, up slow.
- SRV A has been cycled once and NO other SRVs have yet been opened.
- An Operator is preparing to open an SRV.

Which one of the following describes the next SRV to be opened, in accordance with EO-100-102, RPV Control, and the associated reason?

Open SRV...

A. A again to avoid inducing thermal shock to additional SRV piping.

B. A again to minimize the number of SRVs that sustain steam cutting damage to the valve seat.

C. B to more evenly distribute the heat from steam condensation throughout the Suppression Pool.

D. B to prevent steam release to the Suppression Chamber air space in case the SRV A vacuum breaker failed to fully re-seat.

Proposed Answer: C

Explanation: EO-100-102 requires opening SRVs according to an alphabetical sequence (A B C). The bases explain that this is required since it distributes heat uniformly throughout the suppression pool to avoid high local pool temperatures which may result in inefficient pool cooling. The opening sequence also uniformly distributes the total number of SRV actuations among the total number of SRVs.

Note: One of the operational implications of steam condensation, with respect to high Suppression Pool water temperatures, is how SRVs are manually controlled. This affects the distribution of steam condensation and Suppression Pool heatup. The question meets the K/A by testing this concept.

A. Incorrect – SRV B is required to be used next. Plausible because using SRV A again would minimize thermal shock/stress since it's discharge piping is already warm from the first actuation and less thermal shock/stress is generally desirable.
B. Incorrect – SRV B is required to be used next. Plausible because using SRV A again would limit the number of valves experiencing steam cutting, which would reduce the need for future maintenance and maintain other SRVs in better condition.
D. Incorrect – The reason is to more evenly distribute the heat from steam condensation throughout the Suppression Pool to avoid localize high temperatures. Plausible because the SRV A vacuum breaker is expected to cycle following SRV A actuation, and if it did fail to re-seat, then steam would be discharged directly to the Suppression Chamber air space, which would result in an adverse pressure response. This just is not the reason provided by the EOPs.

Technical Reference(s): EO-100-102 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-083 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK1.03
	Importance Rating	3.7

Reactor Low Water Level

Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Water level effects on reactor power

Proposed Question: #41

Unit 1 has experienced a failure to scram with the following:

- Multiple control rods are still withdrawn and CANNOT be inserted.
- Initial Reactor power level after the failure to scram was 15%.
- Reactor water level was intentionally lowered to below -60".
- Reactor power is now downscale on all APRMs and mid-scale on IRM range 6.
- Reactor water level is -65", up slow, with Feedwater injecting.
- Reactor pressure is 920 psig, down slow.
- Standby Liquid Control (SLC) is injecting.
- Initial SLC tank level was 1850 gallons.
- Current SLC tank level is 1050 gallons.
- Suppression Pool water temperature is 105°F, steady.

Given the following portion of EO-100-113, Level/Power Control:

TABLE 19, HSBW Injected	
Initial Tank (Gal.)	Final Tank (Gal.)
2000	1150
1900	1060
1800	975
1700	891
1600	806
1500	722
1400	637

Which one of the following describes the required Reactor water level control strategy, in accordance with EO-100-113?

A. Maintain Reactor water level below -60". Restoring Reactor water level between +13" and +54" inches must wait until the proper amount of SLC is injected.

B. Maintain Reactor water level below -60". Restoring Reactor water level between +13" inches and +54" inches must wait until the proper IRM indication is reached.

C. Restore and maintain Reactor water level between +13" and +54", based on the current IRM indications.

D. Restore and maintain Reactor water level between +13" and +54", based on the current amount of SLC injected.

Proposed Answer: A

Explanation: With an initial power level of >5%, Reactor water level was intentionally lowered to less than -60" per EO-100-113. Reactor water level must be maintained below this level until either a rod pattern is achieved that ensures the Reactor will remain shutdown under all conditions without boron or hot shutdown boron weight is injected. Currently, multiple control rods are still withdrawn and hot shutdown boron weight has not been injected, so the conditions for restoring Reactor water level above -60" are not met. Therefore Feedwater injection must be throttled to prevent level from exceeding -60".

B. Incorrect – IRM indications can be used to determine the Reactor is currently shutdown and begin an RPV cooldown, but they cannot be used to raise Reactor water level above -60".
C. Incorrect – Since the rod pattern does not ensures the Reactor will remain shutdown under all conditions without boron and hot shutdown boron weight has not been injected, Reactor water level must be maintained below -60". Plausible because Reactor water level can be restored under some conditions with control rods still out and because IRM indications are used to determine when to initiate a cooldown with rods still out.
D. Incorrect – Since the rod pattern does not ensures the Reactor will remain shutdown under all conditions without boron and hot shutdown boron weight has not been injected, Reactor water level must be maintained below -60". Plausible because when enough SLC has been injected, level can be restored. Also plausible because current SLC tank level would be enough if the starting level had been higher.

Technical Reference(s): EO-100-113

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-053 RBO-7

Question Source: Bank – JAF 4/14 NRC #41

Question History: JAF 4/14 NRC #41

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295028 EK2.01
	Importance Rating	3.7

High Drywell Temperature

Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell spray: Mark-I&II

Proposed Question: #42

Unit 1 has experienced an extended loss of Drywell cooling and a loss of coolant accident with the following:

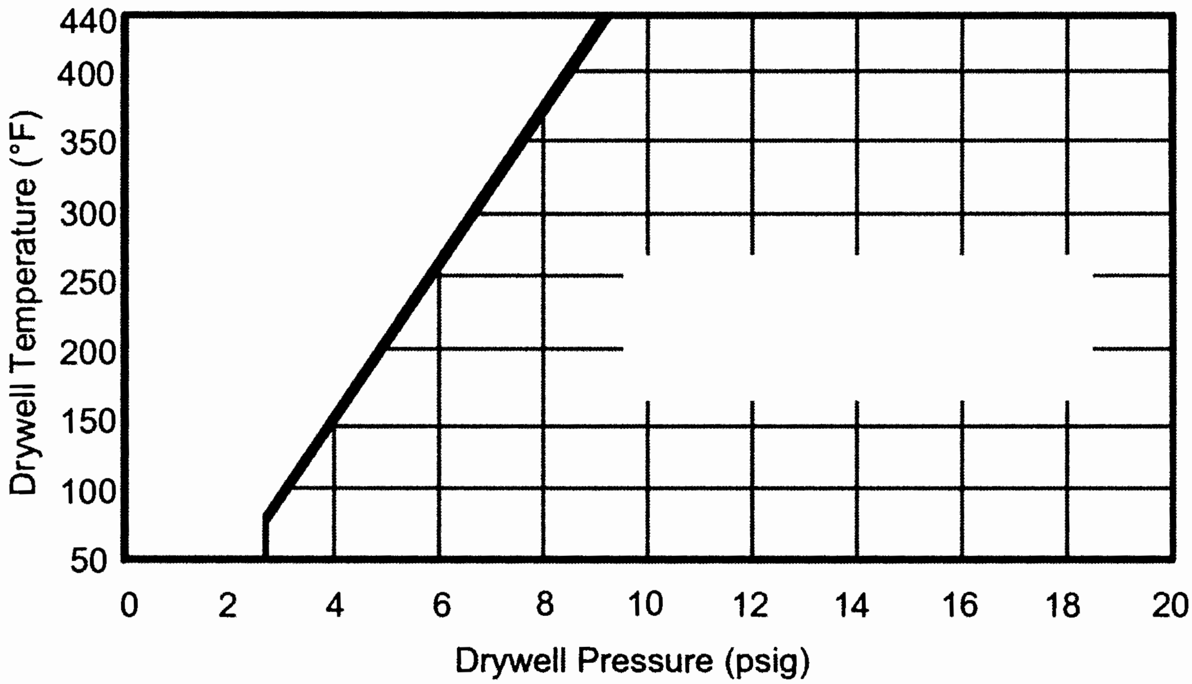
- Reactor pressure is 725 psig, down slow.
- Drywell pressure is 6 psig, up slow.
- Suppression Chamber pressure is 4 psig, up slow.
- Drywell temperature is 335°F, up slow.

Note: A portion of EO-100-103, Primary Containment Control, is provided on the following page.

Which one of the following identifies if Drywell Spray is allowed and the associated reason, in accordance with EO-100-103?

	Is Drywell Spray Allowed?	Associated Reason
A.	Yes	Drywell pressure is above 1.72 psig
B.	Yes	Drywell temperature is approaching 340°F
C.	No	Suppression Chamber pressure is below 13 psig
D.	No	Prevent a rapid pressure drop due to evaporative cooling

FIG. 6 DSIL
DRYWELL SPRAY INITIATION LIMIT



Proposed Answer: D

Explanation: Drywell Spray is not allowed because the combination of Drywell pressure and temperature result in operation on the bad side of the Drywell Spray Initiation Limit (DSIL) curve. Initiating Drywell Spray in this condition could result in a rapid evaporative pressure drop in the Drywell, which would cause an excessive D/P between the Suppression Chamber and Drywell and risk violating the integrity of the containment structure.

A. Incorrect – Drywell Spray is not allowed because the combination of Drywell pressure and temperature result in operation on the bad side of the Drywell Spray Initiation Limit (DSIL) curve. Plausible because Suppression Chamber Spray is currently allowed because Drywell pressure is above 1.72 psig.
B. Incorrect – Drywell Spray is not allowed because the combination of Drywell pressure and temperature result in operation on the bad side of the Drywell Spray Initiation Limit (DSIL) curve. Plausible because if DSIL were not violated, Drywell Spray would be allowed because Drywell temperature is approaching 340°F.
C. Incorrect – The reason is based on excessive D/P between Suppression Chamber and Drywell, not Suppression Chamber pressure being below 13 psig. Plausible because without such abnormally high Drywell temperature, the decision to initiate Drywell Spray would wait until Suppression Chamber pressure exceeded 13 psig.

Technical Reference(s): EO-100-103 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-7

Question Source: Bank – JAF 9/12 NRC #11

Question History: JAF 9/12 NRC #11

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 EK2.07
	Importance Rating	3.5

Low Suppression Pool Water Level

Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: Downcomer / horizontal vent submergence

Proposed Question: #43

Unit 1 Suppression Pool water level has lowered to 12' 2" and stabilized due to an un-isolable leak.

Which one of the following is correct for this Suppression Pool water level?

A. Downcomer flow can be quenched.

B. HPCI turbine exhaust flow can be condensed.

C. SRV steam flow will NOT be completely condensed.

D. Suppression Pool water temperature CANNOT be determined.

Proposed Answer: A

Explanation: Downcomer openings are at 12', therefore they are still covered and can be quenched.

B. Incorrect – HPCI exhaust begins to uncover at approximately 17'. Plausible if candidate confuses HPCI exhaust elevation with one of the other SP levels.
C. Incorrect – SRV tailpipes begin to uncover at 5'. Plausible if candidate confuses SRV tailpipe opening elevation with one of the other SP levels.
D. Incorrect – The four lower SP temperature detectors are located at 3'. Plausible because average SP temperature indications can only be used with level above 20.5'.

Technical Reference(s): EO-100-103 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-3

Question Source: Bank - LOC24 NRC #15

Question History: LOC24 NRC #15

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 AK2.08
	Importance Rating	3.2

Main Turbine Generator Trip

Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: A.C. electrical distribution

Proposed Question: #44

Unit 1 was operating at 100% power when a failure to scram resulted in the following:

- Reactor water level is being controlled at -135".
- Then, a Main Turbine trip occurs.

Which one of the following identifies the loads that will be shed for electrical distribution protection due to the resulting Main Generator lockout?

A. TBCCW Pumps

B. Service Water Pumps

C. Turbine Building Chillers

D. Instrument Air Compressors

Proposed Answer: B

Explanation: A Main Turbine trip will result in a Main Generator lockout. When the Main Generator lockouts trip with a LOCA initiation signal on low reactor level (-129") sealed-in, the Aux Buses undergo a Plant Aux load shed. Major 13.8 KV loads on the Aux Buses receive a momentary trip signal to ensure the Startup Buses are not overloaded when the Aux Buses fast transfer to the Tie Bus. This includes Service Water pumps.

A. Incorrect – The power supplies to the TBCCW pumps are not shed on a Plant Aux Load Shed. Plausible because the TBCCW pumps are supplied by 480V MCCs from the Aux Buses.
C. Incorrect – The TB Chillers are powered from the ESS Buses. Plausible because these chillers are shed on the LOCA signal, but are not affected by the status of the main generator.
D. Incorrect – IACs are not shed on a Plant Aux Load Shed. Plausible because IACs are locked out for 10 minutes on a -129" signal, but only if a Loss of Offsite Power has occurred.

Technical Reference(s): E-102 Sht 31, E-145 Sht 1

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-003 RBO-4

Question Source: Bank – LOC26 NRC #44

Question History: LOC26 NRC #44

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AK3.03
	Importance Rating	3.1

Partial or Complete Loss of CCW

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Securing individual components (prevent equipment damage)

Proposed Question: #45

Unit 1 is operating at 100% power with the following:

- TBCCW system performance is degraded.
- TBCCW supply temperature is 120°F, up slow.
- TBCCW-cooled component temperatures are also rising.
- ON-TBCCW-101, Loss of Turbine Building Closed Cooling Water, is being executed.

Which one of the following identifies components that will need to be secured per the Critical Condition in ON-TBCCW-101 if temperatures continue to rise and the associated reason, in accordance with ON-TBCCW-101?

	Components to Be Secured per ON-TBCCW-101 Critical Condition	Reason
A.	Condensate pumps	Lower heat load on TBCCW
B.	Condensate pumps	Prevent Condensate pump damage
C.	Instrument Air compressors	Lower heat load on TBCCW
D.	Instrument Air compressors	Prevent Instrument Air compressor damage

Proposed Answer: B

Explanation: The Critical Conditions of ON-TBCCW-101 require securing Condensate pumps if they alarm on high bearing temperature. This is based on preventing damage to the Condensate pumps.

A. Incorrect – The reason is to prevent damage to the Condensate pumps. Plausible because on a loss of RBCCW, RWCU is proactively removed from service to lower heat load on RBCCW.
C. Incorrect – The Critical Conditions of ON-TBCCW-101 secure Condensate pumps, not Instrument Air compressors. Plausible because Instrument Air compressors are also cooled by TBCCW and are discussed in Subsequent Operator Actions of ON-TBCCW-101. The reason is to prevent damage to the Condensate pumps. Plausible because on a loss of RBCCW, RWCU is proactively removed from service to lower heat load on RBCCW.
D. Incorrect – The Critical Conditions of ON-TBCCW-101 secure Condensate pumps, not Instrument Air compressors. Plausible because Instrument Air compressors are also cooled by TBCCW and are discussed in Subsequent Operator Actions of ON-TBCCW-101.

Technical Reference(s): ON-TBCCW-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-015 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK3.07
	Importance Rating	4.2

SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Various alternate methods of control rod insertion: Plant-Specific

Proposed Question: #46

Unit 1 has experienced a failure to scram with the following:

- Reactor power is 15% and stable.
- The Reactor Mode Switch is in SHUTDOWN.
- The manual scram pushbuttons have been depressed.
- ARI pushbuttons have been depressed.
- No control rods have inserted.
- All of the white RPS Scram Solenoid Group lights are extinguished.
- The red SCRAM VALVES lights on the full core display are extinguished.
- All of the red ACCUMULATOR lights on the full core display are extinguished.
- Reactor pressure is 425 psig and stable.
- No CRD pumps are available.
- Valve 246016, CRD Unit 1-Unit 2 Cross-Connect, is mechanically bound closed.

Which one of the following methods is available to insert control rods?

A. De-energize scram solenoids per ES-158-001.

B. Vent the scram air header per Posted Instructions.

C. Maximize CRD to drift control rods per OP-155-001.

D. Reset the scram and insert repeat scram signals per ES-158-002.

Proposed Answer: B

Explanation: The given indications show that the RPS scram groups de-energized (all white lights extinguished), but the scram valves did not open (all red lights extinguished) and the accumulators did not discharge (all red lights extinguished). This is indicative of a failure of the scram air header to depressurize. Venting the scram air header per OP-155-001 is available to insert control rods.

Note: The question meets the K/A by requiring the candidate to interpret diverse indications and understand the reasons why various alternate rod insertion methods will or will not work to insert rods.

A. Incorrect – The given indications show that the RPS scram groups are already de-energized (all white lights extinguished) without causing control rod insertion. Pulling RPS fuses per ES-158-001 will not result in any change since RPS scram groups are already de-energized.
C. Incorrect – Raising cooling water D/P per OP-155-001 would work to insert rods, however with no Unit 1 CRD pumps available and the cross-tie to Unit 2 bound closed, this method cannot be accomplished.
D. Incorrect – The given indications show that the RPS scram groups are already de-energized (all white lights extinguished) without causing control rod insertion. Repeating manual scrams per ES-158-002 will not do anything to correct the failure that prevented the scram air header from depressurizing on the first scram.

Technical Reference(s): EO-100-113, ES-158-001, ES-158-002, OP-155-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-055H RBO-7, TM-OP-056 RBO-7

Question Source: Bank – NMP1 2015 NRC #44

Question History: NMP1 2015 NRC #44

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295038 EK3.03
Importance Rating	3.7

High Off-site Release Rate

Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: Control room ventilation isolation: Plant-Specific

Proposed Question: #47

Which one of the following identifies a condition that results in CREOASS automatically initiating in the Pressurization/Filtration mode, and the associated reason for this CREOASS response per the FSAR?

	Condition	Associated Reason
A.	Reactor water level of -32"	Minimize offsite release of radioactive contamination.
B.	Reactor water level of -32"	Ensure continued habitability for Control Room operators.
C.	Control Structure outside air intake radiation level of 7 mR/hr	Minimize offsite release of radioactive contamination.
D.	Control Structure outside air intake radiation level of 7 mR/hr	Ensure continued habitability for Control Room operators.

Proposed Answer: D

Explanation: Control Structure outside air intake radiation level of 7 mR/hr results in CREOASS automatically initiating in the Pressurization/Filtration mode, which also isolates normal Control Structure HVAC air intake. One of the reasons for this response is to ensure continued habitability for Control Room operators.

A. Incorrect – Reactor water level of -32" does not automatically initiate CREOASS in the Pressurization/Filtration mode. Plausible because Reactor water level <-38" does cause this automatic response. Minimizing offsite radioactive release is not a reason. Plausible because this automatic response occurs during situations where an offsite release is a concern, and other ventilation automatic responses are based on minimizing offsite release (eg. Reactor Building).
B. Incorrect – Reactor water level of -32" does not automatically initiate CREOASS in the Pressurization/Filtration mode. Plausible because Reactor water level <-38" does cause this automatic response.
C. Incorrect – Minimizing offsite radioactive release is not a reason. Plausible because this automatic response occurs during situations where an offsite release is a concern, and other ventilation automatic responses are based on minimizing offsite release (eg. Reactor Building).

Technical Reference(s): AR-016-E09, ON-CONTISOL-101, FSAR

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-030 RBO-7

Question Source: Modified Bank – NMP1 2013 NRC #18

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference: Level RO
Tier # 1
Group # 1
K/A # 295038 EK3.03
Importance Rating 3.7

High Off-site Release Rate

Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: Control room ventilation isolation: Plant-Specific

Proposed Question: #18

Which one of the following describes a condition requiring manual isolation of the normal Control Room Ventilation System (CRVS) and manual initiation of the Control Room Emergency Ventilation System (CREVS), and the associated reason per FSAR?

	Condition Requiring Manual CRVS Isolation and CREVS Initiation	Associated Reason per FSAR
A.	Unit 2 enters N2-EOP-RR, Radioactivity Release Control.	Maintain Control Room equipment operability.
B.	**Unit 2 enters N2-EOP-RR, Radioactivity Release Control.**	**Ensure continued habitability for Control Room operators.**
C.	Unit 1 enters N1-EOP-5, Secondary Containment Control.	Maintain Control Room equipment operability.
D.	Unit 1 enters N1-EOP-5, Secondary Containment Control.	Ensure continued habitability for Control Room operators.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 AA1.05
	Importance Rating	3.9

Generator Voltage and Electric Grid Disturbances

Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Engineered safety features

Proposed Question: #48

Both Units are operating at 100% power with the following:

- Grid disturbances cause a loss of ALL offsite power.
- All Diesel Generators start and load their respective buses.
- DG A develops a high jacket water temperature of 250°F.
- DB B develops a low lube oil pressure of 10 psig.

Which one of the following describes the resulting status of DG A and B?

	DG A	DG B
A.	Running	Running
B.	Running	Tripped
C.	Tripped	Running
D.	Tripped	Tripped

Proposed Answer: B

Explanation: The loss of offsite power causes the Diesel Generators to start in Emergency Mode. In this mode, high jacket water temperature does not result in a DG trip, but low lube oil pressure (<30 psig) does. Therefore, DG A continues to run and DG B trips.

A. Incorrect – DG B trips due to low lube oil pressure. Plausible because there is still 10 psig of lube oil pressure and DG B is running in Emergency Mode, so many trips are not enabled.
C. Incorrect – DG A does not trip because it is operating in Emergency Mode. Plausible because in test mode, jacket water temperature >205°F causes a trip. DG B trips due to low lube oil pressure. Plausible because there is still 10 psig of lube oil pressure and DG B is running in Emergency Mode, so many trips are not enabled.
D. Incorrect – DG A does not trip because it is operating in Emergency Mode. Plausible because in test mode, jacket water temperature >205°F causes a trip.

Technical Reference(s): OP-024-001, AR-015-A10, TM-OP-024

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-024 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AA1.06
	Importance Rating	3.0

Plant Fire On-site

Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Fire alarm

Proposed Question: #49

The following Simplex fire alarm is received:

FIRE SUP X222_Z3 ALM
TIME: 0300 DATE: (Today's Date)
02–656 WPS111 CNDNSR

Which of the following describes the type of fire suppression for this area and its status?

This area is protected by a…

A. CO_2 suppression system which has actuated.

B. CO_2 suppression system which has NOT yet actuated.

C. water suppression system which has actuated.

D. water suppression system which has NOT yet actuated.

Proposed Answer: C

Explanation: This alarm indicates that fire suppression has actuated to this area based on the "SUP". "WPS" indicates this is a water protection system.

A. Incorrect – This is a water protection system, not CO_2. Plausible because many other areas that alarm on the same panel are CO_2 protected.
B. Incorrect – This is a water protection system, not CO_2. Plausible because many other areas that alarm on the same panel are CO_2 protected. The "SUP" indicates that fire suppression has actuated to this area. Plausible because there are other Simplex alarms that occur without actuation of the suppression system.
D. Incorrect – The "SUP" indicates that fire suppression has actuated to this area. Plausible because there are other Simplex alarms that occur without actuation of the suppression system.

Technical Reference(s): AR-SP-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-013E RBO-4

Question Source: Modified Bank - LOC25 NRC #43

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross–reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	600000 AK2.01		
	Importance Rating	2.6		

AK2.01 – Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Sensors, detectors and valves

Proposed Question:	Common 43

The following Priority One Simplex Alarm is received:

FIRE SUP X222_Z3 ALM
TIME: 0300 DATE: 08/14/13
02–656 WPS111 CNDNSR

Which of the following would be the plant response for the given Simplex Alarm?

A. Alarms for the Motor–Driven and Diesel–Driven Fire Pumps running will be received
HV16150 Condenser Area Transfer Sump Isolation Valve closes

B. High flow from FSH12201A (FSH FOR WPS–111 UNIT 1 TB CDSR AREA) and WPS–111 OS&Y SUPPLY VALVE via ZS–12201A NOT Full open
Input to Radwaste Collection Tanks will rise

C. Alarms for the Motor–Driven and Diesel–Driven Fire Pumps running will be received
Input to Radwaste Collection Tanks will rise

D. High flow from FSH12201A (FSH FOR WPS–111 UNIT 1 TB CDSR AREA) and WPS–111 OS&Y SUPPLY VALVE via ZS–12201A NOT Full open
HV16150 Condenser Area Transfer Sump Isolation Valve closes

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 AA1.01
	Importance Rating	3.3

Refueling Accidents

Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Secondary containment ventilation

Proposed Question: #50

Unit 1 is in a refueling outage with the following sequence of events:

Time (mm:ss)	Condition
00:00	An irradiated fuel bundle is dropped in the Spent Fuel Pool.
00:45	Annunciator AR-101-E05, SPENT FUEL POOL AREA HI RADIATION, alarms. Area Radiation Monitors (ARMs) 14 and 47, SPENT FUEL CRIT MON, are both in alarm high.
01:30	Annunciator AR-101-D05, REFUELING FLOOR AREA HI RADIATION, alarms. Area Radiation Monitors (ARMs) 15 and 42, SPENT FUEL CRIT MON, are both in alarm high.
03:00	Annunciator AR-101-A05, REFUEL FLOOR WALL EXHAUST HI-HI RADIATION, alarms. Refuel Floor Wall Exhaust Radiation Monitors are both in hi-hi alarm.

Which one of the following identifies the status of Reactor Building Zone III Exhaust Fans during this time period?

Reactor Building Zone III Exhaust Fans...

A. remain running throughout this time period.

B. trip at time 00:45.

C. trip at time 01:30.

D. trip at time 03:00.

Proposed Answer: D

Explanation: When Refuel Floor Wall Exhaust Radiation Monitors are both in hi-hi alarm at time 03:00, SBGT automatically initiates and Zone III isolates. This trips the Zone III Exhaust Fans.

A. Incorrect – The Zone III Exhaust Fans trip at time 03:00. Plausible because the other high radiation alarms do not result in trip of these fans.
B. Incorrect – Spent Fuel Pool ARMs in high alarm do not trip the Zone III Exhaust Fans. Plausible because other related high radiation conditions do trip these fans.
C. Incorrect – Refueling Floor ARMs in high alarm do not trip the Zone III Exhaust Fans. Plausible because other related high radiation conditions do trip these fans.

Technical Reference(s): AR-101-E05, AR-101-D05, AR-101-A05

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-079E RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 EA2.04
	Importance Rating	3.9

High Reactor Pressure

Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Suppression pool level

Proposed Question: #51

Unit 1 has experienced a failure to scram with the following:

- Reactor pressure is 1050 psig, up slow.
- Suppression Pool water temperature indications are:
  - Average SPOTMOS: 155°F, up slow.
  - Bottom SPOTMOS: 172°F, up slow.
- Suppression Pool water level is 16', down slow.

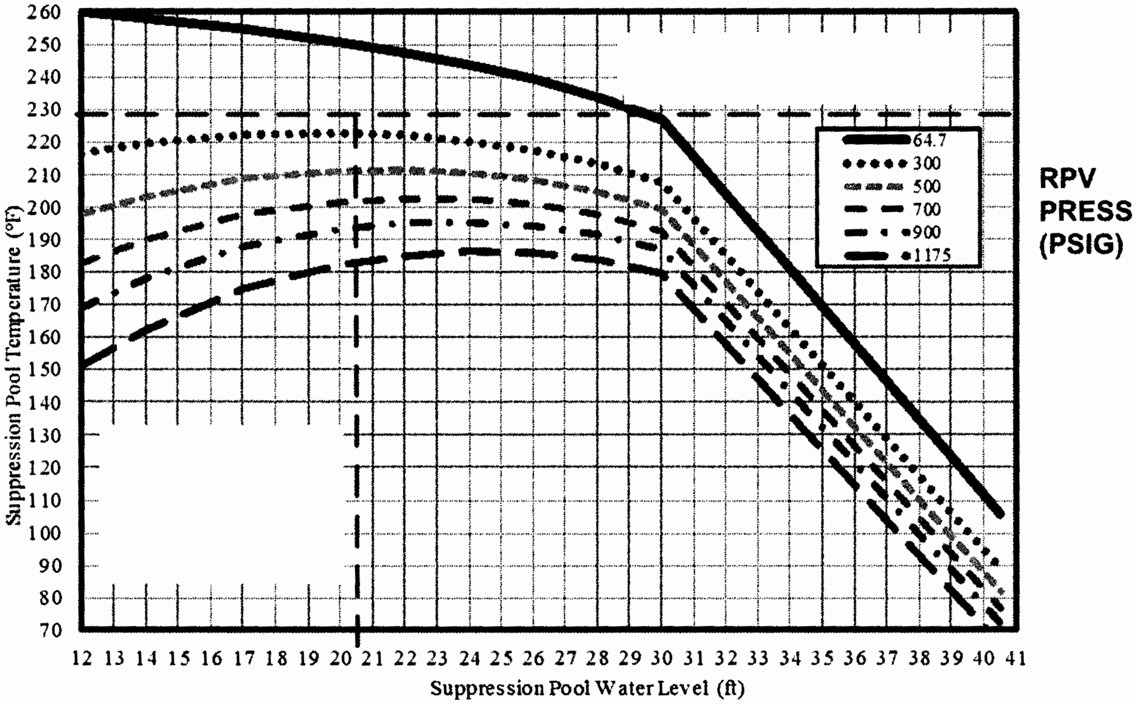
Note: Heat Capacity Temperature Limit (HCTL) is provided on the following page.

Which one of the following describes the status of HCTL and whether SRVs can currently be used for Reactor pressure control, in accordance with the Emergency Operating Procedures?

HCTL is __(1)__ and SRVs __(2)__ be used for Reactor pressure control

	(1)	(2)
A.	exceeded	can
B.	exceeded	CANNOT
C.	NOT exceeded	can
D.	NOT exceeded	CANNOT

FIG 2 HCTL
HEAT CAPACITY TEMPERATURE LIMIT



Proposed Answer: A

Explanation: With Reactor pressure at 1050 psig, the highest and most restrictive Reactor pressure HCTL curve must be used (1175 psig). With Suppression Pool level below 20.5', Average SPOTMOS indication cannot be used and Bottom SPOTMOS must be used. The combination of 16' and 172°F is above the applicable HCTL curve, therefore it is exceeded. SRVs can be used for Reactor pressure control because Suppression Pool level remains above 5'.

B. Incorrect – SRVs can be used for Reactor pressure control because Suppression Pool level remains above 5'. Plausible because Suppression Pool level is low and HPCI cannot be used for pressure control because of how low level is.
C. Incorrect – HCTL is exceeded. Plausible because either using Average SPOTMOS or the wrong Reactor pressure curve would result in HCTL being SAT.
D. Incorrect – HCTL is exceeded. Plausible because either using Average SPOTMOS or the wrong Reactor pressure curve would result in HCTL being SAT. SRVs can be used for Reactor pressure control because Suppression Pool level remains above 5'. Plausible because Suppression Pool level is low and HPCI cannot be used for pressure control because of how low level is.

Technical Reference(s): EO-100-103, EO-100-113

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AA2.05
	Importance Rating	3.1

Partial or Complete Loss of Forced Core Flow Circulation

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Jet pump operability: Not-BWR-1&2

Proposed Question: #52

Unit 1 is operating at 95% power with the following:

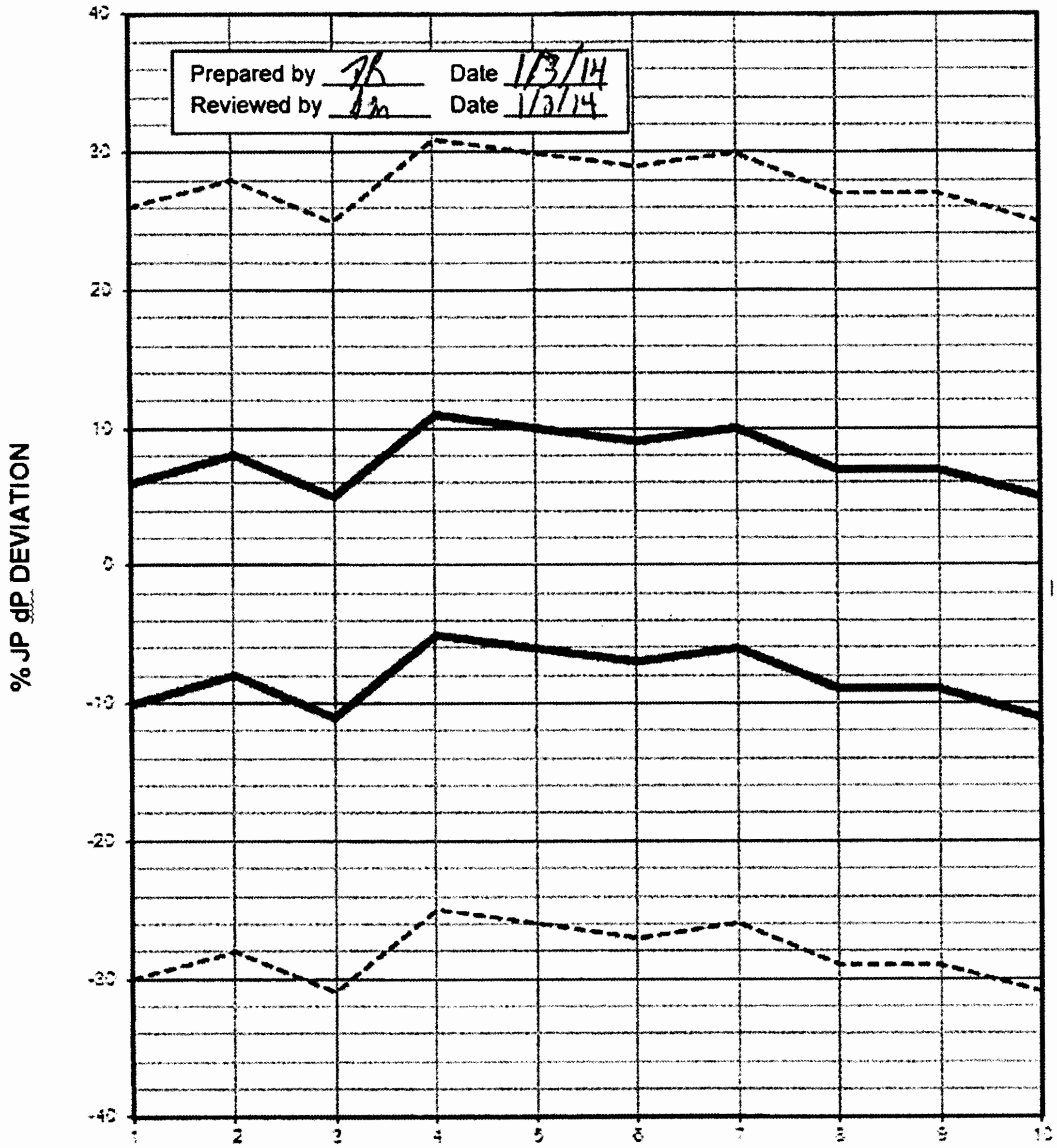
- Earlier in the shift, a drop in core flow was observed.
- SO-100-007, Daily Surveillance Operating Log, is being performed to verify Jet Pump operability.
- Both Recirculation pumps are operating.
- Jet Pump 3 % JP dP deviation has been calculated as +27%.
- Jet Pump 4 % JP dP deviation has been calculated as +29%.

Note: A portion of SO-100-007 Attachment I, Two Loop Jet Pump Distribution Curves, is provided on the following page.

Which one of the following describes the status of Jet Pumps 3 and 4, in accordance with SO-100-007 Attachment I?

	Jet Pump 3	Jet Pump 4
A.	Acceptable	Acceptable
B.	Acceptable	NOT acceptable
C.	NOT acceptable	Acceptable
D.	NOT acceptable	NOT acceptable

FIGURE 2
TWO LOOP JET PUMP DISTRIBUTION LOOP B



Proposed Answer: C

Explanation: Both JP 3 and 4 % dP deviation are higher than normal. However, it is acceptable for a JP % dP deviation to be up to 20% higher than the established baseline. JP 4 has a much higher established baseline than JP 3. JP 4 is within its acceptable range (<+31%) and JP 3 is NOT within its acceptable range (>+25%).

A. Incorrect – JP 3 is not acceptable because it is more than 20% above its established baseline (>+25%). Plausible because JP 4 is acceptable at an even higher % dP.
B. Incorrect – JP 4 is acceptable because is less than 20% above its established baseline (<+31%). Plausible because JP 3 is NOT acceptable at a lower % dP. JP 3 is not acceptable because it is more than 20% above its established baseline (>+25%). Plausible because JP 4 is acceptable at an even higher % dP. Also plausible because JP 4 is above the "normal" range.
D. Incorrect – JP 4 is acceptable because is less than 20% above its established baseline (<+31%). Plausible because JP 3 is NOT acceptable at a lower % dP. Also plausible because JP 4 is above the "normal" range.

Technical Reference(s): SO-100-007

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-064 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AA2.02
	Importance Rating	3.6

Partial or Complete Loss of Instrument Air

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Status of safety-related instrument air system loads (see AK2.1 - AK2.19)

Proposed Question: #53

Unit 1 has experienced a scram with the following:

- Reactor water level is +38", up slow.
- Reactor water level is being controlled with Feedwater.
- Reactor pressure is 925 psig, stable.
- Reactor pressure is being controlled with Turbine Bypass Valves.

Then, a complete loss of Instrument Air occurs. Instrument Air pressure lowers to 0 psig.

Which one of the following describes the continued control of Reactor water level and pressure?

	Reactor Water Level Control	Reactor Pressure Control
A.	Continues on Feedwater	Continues with Turbine Bypass Valves
B.	Continues on Feedwater	Must switch to another system
C.	Must switch to another system	Continues with Turbine Bypass Valves
D.	Must switch to another system	Must switch to another system

Proposed Answer: D

Explanation: The complete loss of Instrument Air causes outboard MSIVs to fail closed. Feedwater is lost because the motive steam is lost to the Feedwater pump turbines. Turbine Bypass Valves still function, however they are unavailable for pressure control because MSIVs are closed.

A. Incorrect – Feedwater is lost because the motive steam is lost to the Feedwater pump turbines. Plausible because the Feedwater pumps themselves do not require Instrument Air to function. TBVs are unavailable for pressure control because MSIVs are closed. Plausible because the TBVs themselves still function properly.
B. Incorrect – Feedwater is lost because the motive steam is lost to the Feedwater pump turbines. Plausible because the Feedwater pumps themselves do not require Instrument Air to function.
C. Incorrect – TBVs are unavailable for pressure control because MSIVs are closed. Plausible because the TBVs themselves still function properly.

Technical Reference(s): ON-INSTAIR-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-018 RBO-6

Question Source: Modified Bank – LOC25 NRC #81

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross–reference:	Level	RO		SRO
	Tier #			1
	Group #			1
	K/A #	295019 2.4.6		
	Importance Rating			4.7

2.4.6 – Emergency Procedures / Plan: Knowledge of EOP mitigation strategies (Partial or Total Loss of Inst. Air)

Proposed Question:	SRO: 81

A reactor startup is in progress on Unit 1. Reactor power is 20 percent.

A seismic event occurs.

The reactor automatically scrams. All control rods fully insert.

A loss of Instrument Air occurs.

Current reactor conditions are as follows:

Reactor level	+5 inches, down slow
Reactor pressure	900 psig, up slow
Drywell pressure	1.21 psig, up slow
Drywell temperature	125 °F, up slow

The Seismic Monitor has been determined to be Inoperable

If level continues on its present trend, which of the following systems may be used to restore and maintain reactor level, AND how would this event be classified for the given conditions?

	RPV Level Control Systems	Classification
A.	Condensate, Feedwater, CRD, RCIC, HPCI	Alert
B.	Condensate, Feedwater, CRD, RCIC, HPCI	Unusual Event
C.	**CRD, RCIC, HPCI, SBLC**	**Alert**
D.	CRD, RCIC, HPCI, SBLC	Unusual Event

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 2.2.39
	Importance Rating	3.9

SCRAM

Knowledge of less than one hour technical specification action statements for systems.

Proposed Question: #54

Unit 1 is operating at 100% power.

Which one of the following conditions requires an immediate Reactor scram, in accordance with Technical Specifications?

A. MAPRAT at 1.1.

B. Drywell pressure at 1.1 psig.

C. Suppression Pool average water temperature at 111°F.

D. CRD charging water header pressure at 0 psig with two scram accumulators inoperable.

Proposed Answer: C

Explanation: TS 3.6.2.1 requires an immediate Reactor scram because Suppression Pool average water temperature is >110°F.

A. Incorrect – With MAPRAT >1.0, one or more APLHGRs are above the limits in the COLR. TS 3.2.1 allows 2 hours to correct this condition. Plausible because this condition does require TS condition entry, the associated completion time is very short, and this condition is significant in that it threatens the fuel clad barrier in the event of an accident.
B. Incorrect – TS 3.6.1.4 is still met with Drywell pressure at 1.1 psig. Plausible because this is an elevated Drywell pressure, causes annunciator AR-112-D03 to alarm, and the associated TS does have a ≤1 hour requirement.
D. Incorrect – TS 3.1.5 allows 20 minutes to restore charging water pressure because Reactor pressure is >900 psig at 100% power. Plausible because a scram is required in less than 1 hour if the condition is not corrected, and if Reactor pressure were <900 psig, then an immediate Reactor scram would be required.

Technical Reference(s): TS 3.6.2.1, 3.2.1, 3.6.1.4, 3.1.5

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-8

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 2.2.36
	Importance Rating	3.1

Partial or Complete Loss of AC

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

Proposed Question: #55

A Unit 1 shutdown is in progress with the following:

- The Reactor Mode Switch is in STARTUP/STBY.
- All IRMs are on Range 9.
- Control rod insertion is in progress.
- Startup Transformer T-10 must be removed from service for emergent maintenance.
- Startup Bus 10 will be transferred to Startup Bus 20 when Startup Transformer T-10 is removed from service.

Which one of the following describes the applicable "AC Sources" Technical Specification for Unit 1 and if the number of required operable offsite circuits will still be satisfied once T-10 is removed from service?

	Applicable "AC Sources" Technical Specification for Unit 1	Once T-10 is removed from service, the required number of operable offsite circuits is...
A.	3.8.1, AC Sources – Operating	still satisfied.
B.	3.8.1, AC Sources – Operating	NO longer satisfied.
C.	3.8.2, AC Sources – Shutdown	still satisfied.
D.	3.8.2, AC Sources – Shutdown	NO longer satisfied.

Proposed Answer: B

Explanation: Per TS Table 1.1-1, the plant is currently in Mode 2. Therefore, TS 3.8.1 is currently applicable for Unit 1 (Modes 1, 2, and 3). TS 3.8.2 would be applicable in either Modes 4 or 5. TS 3.8.1 requires 2 operable offsite circuits (satisfied by having both Startup Transformer T-10 and T-20 in service). When Startup Transformer T-10 is removed from service, only 1 offsite circuit will be operable, so the requirement of LCO 3.8.1 will not be satisfied.

A. Incorrect – TS 3.8.1 requires 2 operable offsite circuits. Taking SUT T-10 out of service results in only 1 operable offsite circuit. Plausible because all the offsite power lines are still operable. Also plausible because TS 3.8.2 only requires 1 operable offsite circuit.
C. Incorrect – TS 3.8.1, not 3.8.2 is currently applicable. Plausible because a plant shutdown is in progress and the Reactor Mode Switch is out of RUN. TS 3.8.1 requires 2 operable offsite circuits. Taking SUT T-10 out of service results in only 1 operable offsite circuit. Plausible because all the offsite power lines are still operable. Also plausible because TS 3.8.2 only requires 1 operable offsite circuit.
D. Incorrect – TS 3.8.1, not 3.8.2 is currently applicable. Plausible because a plant shutdown is in progress and the Reactor Mode Switch is out of RUN.

Technical Reference(s): TS Table 1.1-1, TS 3.8.1, TS 3.8.2

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-003 RBO-8

Question Source: Bank – JAF 9/14 NRC #21

Question History: JAF 9/14 NRC #21

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 2.4.50
	Importance Rating	4.2

Loss of Shutdown Cooling

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Proposed Question: #56

Unit 1 is shutdown with the following:

- Annunciator AR-110-B01, RX LO LEVEL SIGNAL A CONFIRMED, is in alarm.
- Annunciator AR-110-B03, RX LO LEVEL SIGNAL B CONFIRMED, is NOT in alarm.
- Reactor water level is +10", down slow.
- RHR loop 1A is operating in the Shutdown Cooling (SDC) lineup.

Which one of the following describes the response of the given alarms and the required control of SDC?

A. Both alarms have responded properly. Isolate SDC now.

B. Both alarms have responded properly. Isolate SDC only if AR-110-B03 also alarms.

C. One of the alarms has NOT responded properly. Maintain SDC in service.

D. One of the alarms has NOT responded properly. Isolate SDC.

Proposed Answer: D

Explanation: AR-110-B01/B03 are required to alarm at +13" and signify that the low Reactor water level isolation of SDC is required. AR-110-B03 has failed to alarm as required and SDC has failed to isolate as required. ON-CONTISOL-101 entry and SDC isolation is required because Reactor water level is <+13".

Note: While the annunciator response does not contain the guidance for control of SDC, it does result in ON entry which does provide the proper direction. For purposes of matching the second half of the K/A, the ON guidance is considered an extension of the annunciator response in this case.

A. Incorrect – Annunciator AR-110-B03 has failed to alarm at +13". Plausible that this alarm would be received at a lower confirmation level (such as ADS confirmatory logic), but just receipt of AR-110-B01 would require isolation of SDC because SDC isolation logic is a special case in RHR. Also plausible because -129" is another Reactor water level associated with RHR logic.
B. Incorrect – Annunciator AR-110-B03 has failed to alarm at +13". Plausible that this alarm would be received at a lower confirmation level (such as ADS confirmatory logic). Also plausible because -129" is another Reactor water level associated with RHR logic.
C. Incorrect – SDC must be isolated because Reactor water level is <+13". Plausible that these alarms would be associated with the other Reactor water level associated with RHR operation (-129"), such that AR-110-B01 has alarmed when not required, and isolation would not be required.

Technical Reference(s): AR-110-B01/B03, ON-SDC-101, ON-CONTISOL-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295004 AA1.03
Importance Rating	3.4

Partial or Complete Loss of DC Power

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: A.C. electrical distribution

Proposed Question: #57

Unit 1 is operating at 50% power when 125VDC DC Bus 1D610 de-energizes due to a sustained electrical fault.

Which one of the following describes the status of control power to Aux Buses 11A?

Aux Bus 11A...

A. maintains its normal source of control power, but loses an alternate source.

B. loses its normal source of control power and does not have an alternate source.

C. loses its normal source of control power, but automatically transfers to an alternate control power supply.

D. loses its normal source of control power, but can be manually transferred to an alternate control power supply.

Proposed Answer: D

Explanation: Aux Bus 11A normally receives control power from 125 VDC Bus 1D610 (through 1D615). This control power source is lost. Backup control power supply is available, but must be manually transferred.

A. Incorrect – Aux Bus 11A normally receives control power from 125 VDC Bus 1D610 (through 1D615). Plausible because this would be true for another DC power loss.
B. Incorrect – Aux Bus 11A does have an alternate control power source. Plausible because not all control power is transferrable.
C. Incorrect – Backup control power supply is available, but must be manually transferred. Plausible because some power supplies automatically transfer to backup on loss of normal supply.

Technical Reference(s): ON-125VDC-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-002 RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 2.1.7
	Importance Rating	4.4

Control Room Abandonment

Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question: #58

A Control Room evacuation has been performed due to a fire with the following:

- ON-CREVAC-101, Control Room Evacuation, is being performed.
- All required actions were completed prior to exiting the Control Room.
- Reactor water level and pressure control have been established at the Remote Shutdown Panel.
- Reactor water level is +54", up slow.
- Reactor pressure is 700 psig, down slow.
- Reactor coolant temperature has lowered 37°F in the last 15 minutes.

Which one of the following describes the required control of Reactor water level and Reactor cooldown, in accordance with ON-CREVAC-101?

A. Continue to raise Reactor water level. Lower the cooldown rate.

B. Continue to raise Reactor water level. Maintain the cooldown rate.

C. Lower Reactor injection to stop the rise in level. Lower the cooldown rate.

D. Lower Reactor injection to stop the rise in level. Maintain the cooldown rate.

Proposed Answer: C

Explanation: ON-CREVAC-101 established control of Reactor water level and pressure from the Remote Shutdown Panel using RCIC and SRVs. ON-CREVAC-101 assumes EO-100-102 is utilized to procedurally control Reactor water level and pressure. Therefore, Reactor water level is required to be controlled +13" to +54" and Reactor cooldown is required to be controlled <100°F/hr. With level 54" up slow, Reactor injection must be throttled to lower level. With the cooldown rate at 37°F in the last 15 minutes (148°F/hr), the rate of depressurization must be slowed.

A. Incorrect – Reactor water level is required to be controlled +13" to +54". Plausible because later in ON-CREVAC-101, Reactor water level is intentionally raised to 90-100" to support placing Shutdown Cooling in service.
B. Incorrect – Reactor water level is required to be controlled +13" to +54". Plausible because later in ON-CREVAC-101, Reactor water level is intentionally raised to 90-100" to support placing Shutdown Cooling in service. Reactor cooldown is required to be controlled <100°F/hr. Plausible that a more rapid cooldown would be allowed during a Control Room Evacuation to reduce stored energy and allow injection with RHR from Remote Shutdown Panel.
D. Incorrect – Reactor cooldown is required to be controlled <100°F/hr. Plausible that a more rapid cooldown would be allowed during a Control Room Evacuation to reduce stored energy and allow injection with RHR from Remote Shutdown Panel.

Technical Reference(s): ON-CREVAC-101, EO-100-102

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	500000 EK1.01
	Importance Rating	3.3

High Containment Hydrogen Concentrations

Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATIONS: Containment integrity

Proposed Question: #59

Unit 1 has experience a loss of coolant accident and EO-100-103, Primary Containment Control, is being executed.

Which one of the following identifies the Hydrogen concentration threshold in the PC/G leg and the associated required action if this threshold is exceeded, in accordance with EO-100-103?

	Hydrogen Concentration Threshold	Required Action if Exceeded
A.	1%	Initiate Drywell spray
B.	1%	Vent the Primary Containment
C.	6%	Initiate Drywell spray
D.	6%	Vent the Primary Containment

Proposed Answer: B

Explanation: The Hydrogen concentration threshold used in the PC/G leg of EO-100-103 is 1%. If this is exceeded, the Primary Containment is vented.

A. Incorrect – Primary Containment venting is performed, not initiation of Drywell spray. Plausible because other legs of EO-100-103 initiate Drywell spray, the PC/G leg contains a conditions step about control of Drywell cooling relative to the status of Drywell spray, and initiation of Drywell spray may help prevent a combustion/explosion.
C. Incorrect – The threshold used in PC/G is 1%, not 6%. Plausible because 6% is the combustible limit and is used in EP-DS-001. Primary Containment venting is performed, not initiation of Drywell spray. Plausible because other legs of EO-100-103 initiate Drywell spray, the PC/G leg contains a conditions step about control of Drywell cooling relative to the status of Drywell spray, and initiation of Drywell spray may help prevent a combustion/explosion.
D. Incorrect – The threshold used in PC/G is 1%, not 6%. Plausible because 6% is the combustible limit and is used in EP-DS-001.

Technical Reference(s): EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295017 AK2.10
	Importance Rating	3.3

High Off-site Release Rate

Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Process radiation monitoring system

Proposed Question: #60

EO-100-105, Radioactivity Release Control, contains the following entry condition:

OFFSITE RAD RELEASE
RATE ABOVE ________________
ANTICIPATED
(TABLE 11)

Which one of the following identifies (1) a process radiation monitor used to assess this need for entry into EO-100-105, Radioactivity Release Control, and (2) the emergency classification level that fills in the blank above?

A. (1) Turbine Building VERMS
(2) Unusual Event

B. (1) Turbine Building VERMS
(2) Alert

C. (1) RHRSW discharge radiation monitor
(2) Unusual Event

D. (1) RHRSW discharge radiation monitor
(2) Alert

Proposed Answer: B

Explanation: The EO-100-105 entry condition is:

OFFSITE RAD RELEASE RATE ABOVE ALERT ANTICIPATED (TABLE 11)

Table 11 includes Turbine Building VERMS, but not RHRSW discharge radiation monitors.

A. Incorrect – Alert, not Unusual Event, correctly fills in the blank. Plausible because entry to EO-100-105 could be made at the Unusual Event level if a future Alert was anticipated. Also plausible because the Unusual Event level represents a significant degradation of plant release rate.
C. Incorrect – RHRSW discharge radiation monitors are not used to assess the need for entry into EO-100-105. Plausible because they do represent an offsite release and are used in the EALs for classification of emergency conditions. Alert, not Unusual Event, correctly fills in the blank. Plausible because entry to EO-100-105 could be made at the Unusual Event level if a future Alert was anticipated. Also plausible because the Unusual Event level represents a significant degradation of plant release rate.
D. Incorrect – RHRSW discharge radiation monitors are not used to assess the need for entry into EO-100-105. Plausible because they do represent an offsite release and are used in the EALs for classification of emergency conditions.

Technical Reference(s): EO-100-105

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295022 AK3.01
	Importance Rating	3.7

Loss of CRD Pumps

Knowledge of the reasons for the following responses as they apply to LOSS OF CRD PUMPS: Reactor SCRAM

Proposed Question: #61

Unit 1 is operating at 100% power with the following:

- Control Rod Drive (CRD) pump 1B is out of service for preventative maintenance.
- CRD pump 1A trips due to an overcurrent condition.
- Electrical Maintenance reports that the pump motor has failed.
- The following is a timeline of events:

Time (minutes)	Event
T_0	CRD pump 1A trips.
$T_0 + 10$	FIRST accumulator trouble alarm is received on Control Rod 22-23. Control Rod 22-23 is at position 10. HCU 22-23 accumulator pressure is 935 psig, down slow.
$T_0 + 15$	SECOND accumulator trouble alarm is received on Control Rod 42-15. Control Rod 42-15 is at position 48. HCU 42-15 accumulator pressure is 935 psig, down slow.

Which one of the following describes the action REQUIRED by ON-CRD-101, Control Rod Malfunction, and the reason for this action?

	Required Action	Reason
A.	Place the Reactor Mode Switch in SHUTDOWN immediately at time T_0 + 15 minutes.	Due to risk of drifting control rods.
B.	Place the Reactor Mode Switch in SHUTDOWN immediately at time T_0 + 15 minutes.	Due to risk of loss of scram capability.
C.	Restore charging water header pressure >940 psig or place the Reactor Mode Switch in SHUTDOWN by time T_0 + 35 minutes.	Due to risk of drifting control rods.
D.	Restore charging water header pressure >940 psig or place the Reactor Mode Switch in SHUTDOWN by time T_0 + 35 minutes.	Due to risk of loss of scram capability.

Proposed Answer: D

Explanation: The concern with low accumulator pressure is loss of scram capability. With Reactor pressure greater than 900 psig, a Reactor scram is required within 20 minutes of receipt of a second accumulator alarm unless charging water header pressure can be restored.

A. Incorrect – The concern is loss of scram capability, not drifting control rods. Plausible because drifting control rods is the concern with low scram air header pressure. A scram is not immediately required. Plausible because a scram would be required immediately if Reactor pressure was <900 psig.
B. Incorrect – A scram is not immediately required. Plausible because a scram would be required immediately if Reactor pressure was <900 psig.
C. Incorrect – The concern is loss of scram capability, not drifting control rods. Plausible because drifting control rods is the concern with low scram air header pressure.

Technical Reference(s): ON-CRD-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-055H RBO-7

Question Source: Bank – LOC26R NRC #34

Question History: LOC26R NRC #34

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295033 EA1.06
	Importance Rating	2.9

High Secondary Containment Area Radiation Levels

Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Portable radiation monitoring instruments

Proposed Question: #62

Unit 1 is operating at 75% power with the following:

- An un-isolable Reactor Water Cleanup leak into the Reactor Building is in progress.
- Area Radiation Monitors are unavailable due to power loss.
- A Radiation Protection technician obtains the following radiation data using a portable radiation monitor:
  - Reactor Building elevation 749' near RWCU pumps – 13 R/hr, up slow
  - Reactor Building elevation 749' near Reactor Building sample station – 12 R/hr, up slow
  - Reactor Building elevation 719' near the north HCUs – 8 R/hr, up slow

Given the following portion of EO-100-104, Secondary Containment Control:

TABLE 9

TABLE 9 REACTOR BUILDING RADIATION

RB AREA EL (FT)	ARM NUMBER LO RANGE	ARM NUMBER HIGH RANGE	ARM CHANNEL DESCRIPTION	MAX NORMAL RADIATION FIELD	MAX SAFE RADIATION FIELD (MR/HR)	MAX SAFE RADIATION EO104 (R/HR)	**RB RAD (R/HR)**	
818	35+	N/A	CASK STOR AREA	HI ALARM	10^4	10		
	14+	N/A	SPENT FUEL CRIT MON					
	15*	N/A	REFUEL FLOOR SOUTH					
	42+	N/A	REFUEL FLOOR WEST					
	47*	N/A	SPENT FUEL CRIT MON					
	N/A	49	REFUEL FLOOR AREA					
749	8+	52	RWCU RECIRC PP ACC	HI ALARM	10^4	10		
	10*	54	FUEL POOL PP AREA					
	11+	N/A	RX BLD SAMPLE ST					
719	5*	50	CRD NORTH	HI ALARM	10^4	10		
	6*	51	CRD SOUTH					
670	16+	53	REM SHDN ROOM ACC	HI ALARM	10^4	10		
645	3+	48	HPCI PP * TURB ROOM	HI ALARM	10^4	10		
645	2+	57	RCIC PP * TURB ROOM	HI ALARM	10^4	10		
645	25*	55	RHR A * C PP ROOM	HI ALARM	10^4	10		
645	1*	56	RHR B * D PP ROOM	HI ALARM	10^4	10		
645	4*	N/A	RB/RW SUMP ROOM	HI ALARM	10^4	10		

RANGE: + 0.01 - 10^2 MR/HR * 0.1 - 10^3 MR/HR

Which one of the following describes the required control of the Reactor, in accordance with EO-100-104 and EO-100-102, RPV Control?

A. The Reactor may continue to operate at power.

B. A Reactor scram is required, then the cooldown rate must be maintained <100°F/hr.

C. A Reactor scram is required, then Turbine Bypass Valves may be used to cooldown at a rate >100°F/hr. An Emergency RPV Depressurization is NOT required.

D. A Reactor scram is required, then an Emergency RPV Depressurization must be performed.

Proposed Answer: C

Explanation: Each Reactor Building elevation in EO-100-104 Table 9 (outlined by a box around one or more ARMs) is treated as on Reactor Building area. Therefore, one Reactor Building area is above the Max Safe rad level (elevation 749', >10 R/hr) and a second Reactor Building area is approaching the Max Safe rad level (elevation 719', <10 R/hr but close and rising). EO-100-104 requires a Reactor scram due to a primary system discharging into a Reactor Building area and one Reactor Building area above the Max Safe rad level. EO-100-104 does not yet require Emergency RPV Depressurization because a second area has not yet exceeded the Max Safe rad level. However, EO-100-102 allow a cooldown with TBVs >100°F/hr in anticipation of Emergency RPV Depressurization.

A. Incorrect – A Reactor scram is required because one area is above the Max Safe rad level. Plausible because this would be the correct answer if the leak was from a non-primary system or could be isolated.
B. Incorrect – The cooldown rate may exceed 100°F/hr. Plausible because this would be the correct answer if there was not a second area trending towards the Max Safe rad level.
D. Incorrect – An Emergency RPV Depressurization is not required. Plausible because two locations are reported above the Max Safe rad level, however these locations are each in the same Reactor Building area as defined by the EOPs.

Technical Reference(s): EO-100-104

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – LOC23 NRC #60

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

QUESTION 60

Unit 1 is operating in MODE 1 with reactor power at 75%. A primary leak into secondary containment is in progress.

Radiation levels in the reactor building are rising, with the following Area Radiation Monitor indications:

- ARM channel 50, "CRD NORTH" indicates 2×10^4 MR/HR
- ARM channel 52. "RWCU RECIRC PP ACC" indicates 3×10^4 MR/HR
- All other ARM channels are reading between 100 and 500 MR/HR

Refer to EO-100-004, "Secondary Containment Control", table 9 below and determine which one of the following describes the impact of these conditions on plant operation:

TABLE 9
REACTOR BUILDING RADIATION

RB AREA EL (FT)	ARM NUMBER LO RANGE	ARM NUMBER HIGH RANGE	ARM CHANNEL DESCRIPTION	MAX NORMAL RADIATION FIELD	MAX SAFE RADIATION FIELD (MR/HR)	MAX SAFE RADIATION EO104 (R/HR)	RB RAD (R/HR)
818	35*	N/A	CASK STOR AREA	HI ALARM	10^4	10	______ ______
	14*	N/A	SPENT FUEL CRIT MON				______ ______
	19*	N/A	REFUEL FLOOR NORTH				______ ______
	42*	N/A	REFUEL FLOOR WEST				______ ______
	47*	N/A	SPENT FUEL CRIT MON				______ ______
	N/A	48	REFUEL FLOOR AREA				
749	9*	52	RWCU RECIRC PP ACC				______ ______
	10*	54	FUEL POOL PP AREA	HI ALARM	10^4	10	______ ______
	11*	N/A	RX BLD SAMPLE ST				______ ______
719	5*	50	CRD NORTH				______ ______
	6*	51	CRD SOUTH	HI ALARM	10^4	10	______ ______

The crew is required to insert a reactor scram AND...

A.	stabilize RPV pressure below 1087 psig.
B.	perform Rapid Depressurization.
C.	anticipate Rapid Depressurization.
D.	force a cooldown of the RPV within limits.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295012 AA2.01
	Importance Rating	3.8

High Drywell Temperature

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Drywell temperature

Proposed Question: #63

Unit 1 is operating at 100% power with the following:

- Degraded Drywell Unit Cooler performance has been observed and a small steam leak has developed in the Drywell.
- Drywell average air temperature is 131°F and rising 2°F/min.
- Drywell pressure is 0.3 psig and rising 0.1 psig/min.

Note: Assume the rate of temperature and pressure rise remains constant.

Which one of the following describes the need to enter EO-100-103, Primary Containment Control?

EO-100-103 entry...

A. is currently required.

B. will first become required in approximately two (2) minutes.

C. will first become required in approximately ten (10) minutes.

D. will first become required in approximately fourteen (14) minutes

Proposed Answer: C

Explanation: EO-100-103 requires entry when either Drywell average air temperature exceeds 150°F or Drywell pressure exceeds 1.72 psig. Drywell pressure will reach and exceed 1.72 psig in approximately 14 minutes [(1.72-0.3)/0.1=14.2 minutes]. Drywell average air temperature will reach and exceed 150°F in approximately 10 minutes [(150-131)/2=9.5 minutes]. Therefore, EO-100-103 entry is first required in approximately 10 minutes.

A. Incorrect – Both Drywell pressure and average air temperature are below EO-100-103 entry conditions. Plausible because Drywell average air temperature is elevated and close to exceeding the TS limit.
B. Incorrect – EO-100-103 entry is not required until approximately 10 minutes. Plausible because in 2 minutes the TS 3.6.1.5 limit of 135°F will be exceeded.
D. Incorrect – EO-100-103 entry is first required in approximately 10 minutes based on Drywell average air temperature. Plausible because 14 minutes is when entry will first be required by Drywell pressure.

Technical Reference(s): EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295035 2.4.21
	Importance Rating	4.0

Secondary Containment High Differential Pressure

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: #64

Unit 1 has experienced a steam leak from HPCI into the Reactor Building with the following:

- Reactor Building Zone 1 HVAC has isolated.
- Standby Gas Treatment (SGTS) failed to start.
- Reactor Building Zone 1 differential pressure is +0.05" WG.

Which one of the following describes the status of the radioactive release from the Unit 1 Reactor Building?

A. Monitored, filtered

B. Monitored, unfiltered

C. Unmonitored, filtered

D. Unmonitored, unfiltered

Proposed Answer: D

Explanation: The purpose of the Secondary Containment and associated HVAC system is to provide a monitored, filtered release pathway for any airborne contamination. With Zone 1 HVAC isolated, Standby Gas Treatment failing to start, and positive pressure in the Secondary Containment, there is a release pathway through the Secondary Containment walls that is unmonitored and unfiltered (at ground level, not elevated through an exhaust stack and filtered).

A. Incorrect – With no ventilation running and positive pressure, the filtered, monitored release pathway is not functioning. Plausible because this would be correct if Standby Gas Treatment had automatically started as it should have when Zone 1 HVAC isolated.
B. Incorrect – The release is unmonitored because it is no longer being routed through an exhaust point that has radiation monitoring capability. Plausible if candidate believed release could still be classified as monitored due to either offsite monitoring equipment or area radiation monitors. Also plausible if candidate believed Secondary Containment was leak tight enough that release was still only through exhaust ducting, which is monitored.
C. Incorrect – The release is unfiltered, not filtered. Plausible if candidate believed release was classified as filtered because pressure would still force any release through the correct exhaust pathway (such as if Secondary Containment was leak tight design).

Technical Reference(s): TM-OP-034, TS 3.6.4.1 Bases, TS 3.6.4.3 Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-034 RBO-6

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295009 AA2.03
	Importance Rating	2.9

Low Reactor Water Level

Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL: Reactor water cleanup blowdown rate

Proposed Question: #65

A Unit 1 startup is in progress with the following:

- The Reactor is critical in the source range.
- Reactor coolant temperature is 126°F up slow.
- Reactor water level is 35", steady.
- RWCU is operating in the automatic letdown mode.
- Feedwater Startup Bypass Valve HV-10640 is currently closed.
- Condensate is in long path recirculation mode.

Then, the running CRD pump trips and Reactor water level begins to slowly lower.

Which one of the following identifies when the RWCU letdown flow rate first begins to lower?

A. Immediately

B. When Reactor water level reaches 33"

C. When Reactor water level reaches 30"

D. When Reactor water level reaches -38".

Proposed Answer: A

Explanation: When in automatic letdown mode, the RWCU letdown flow regulator receives a Reactor water level signal and automatically modulates the position of HV-144-F033 proportionally to the error between actual Reactor water level and setpoint. Therefore, as soon as Reactor water level began lowering below the 35" initial setpoint, letdown flow rate immediately begins lowering.

B. Incorrect – Letdown flow rate immediately begins lowering. Plausible because 33" is a level setpoint used in automatic RWCU letdown mode to restore automatic control after a level drop to 30".
C. Incorrect – Letdown flow rate immediately begins lowering. Plausible because 30" is a level setpoint used in automatic RWCU letdown mode to fully stop letdown flow.
D. Incorrect – Letdown flow rate immediately begins lowering. Plausible because -38" is the Reactor water level at which RWCU will isolate, which would serve as a backup means to secure letdown flow.

Technical Reference(s): OP-161-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-061 RBO-4

Question Source: Bank – LOC24 NRC #57

Question History: LOC24 NRC #57

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.15
	Importance Rating	2.7

Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

Proposed Question: #66

Given the following:

- You are the oncoming Plant Control Operator (PCO) for Unit 1.
- You and your shift have been off for the previous 7 days.
- Today is your first shift back on watch.

Which one of the following describes how far back you must review in the eSOMS logs and how a new Ops Supervisor Directive (standing order) will be communicated to you, in accordance with OP-AD-003, Shift Surveillance Scheduling, Log Sheets, Turnover Sheets and Rounds, and OI-AD-095, Operations Directives?

	Required Review Period in Logs	Communication Method for New Ops Supervisor Directive
A.	Only those from the past 24 hours	Pre-Shift Brief
B.	Only those from the past 24 hours	PCO Turnover Sheet
C.	All since last watch	Pre-Shift Brief
D.	All since last watch	PCO Turnover Sheet

Proposed Answer: A

Explanation: OP-AD-003 requires the oncoming PCO to review the eSOMS logs from the past 24 hours only. OI-AD-095 requires a new Ops Supervisor Directive to be communicated through the pre-shift brief.

B. Incorrect – A new Ops Supervisor Directive is required to be communicated at pre-shift brief, not placed on the PCO turnover sheet. Plausible because many other items required to be communicated to the oncoming PCO are placed on the PCO turnover sheet.
C. Incorrect – Only the logs from the last 24 hours are required to be reviewed. Plausible that all logs since the last watch would be required to be reviewed so that the PCO understood more history of plant conditions, especially since it has only been one week.
D. Incorrect – Only the logs from the last 24 hours are required to be reviewed. Plausible that all logs since the last watch would be required to be reviewed so that the PCO understood more history of plant conditions, especially since it has only been one week. A new Ops Supervisor Directive is required to be communicated at pre-shift brief, not placed on the PCO turnover sheet. Plausible because many other items required to be communicated to the oncoming PCO are placed on the PCO turnover sheet.

Technical Reference(s): OP-AD-003, OI-AD-095

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – NMP1 2009 NRC #66

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Given the following:

- The last watch you stood was day shift on May 1
- You have been on vacation and are preparing to assume the shift as the CRO on day shift on May 10

Which one of the following describes the Control Room Logs that must be reviewed before assuming the shift, in accordance with CNG-OP-1.01-2002?

A.	All Control Room Logs back to and including day shift on May 3. Further back is not required and anything less is not appropriate.
B.	All Control Room Logs back to and including day shift on May 5. Further back is not required and anything less is not appropriate.
C.	**All Control Room Logs back to and including day shift on May 8. Further back is not required and anything less is not appropriate.**
D.	Just the day and night shift Control Room logs for May 9. Further back is not required and anything less is not appropriate.

QUESTION	66

Given the following:

- The last watch you stood was day shift on May 1
- You have been on vacation and are preparing to assume the shift as the CRO on day shift on May 10

Which one of the following describes the Control Room Logs that must be reviewed before assuming the shift, in accordance with CNG-OP-1.01-2002?

A.	All Control Room Logs back to and including day shift on May 3. Further back is not required and anything less is not appropriate.
B.	All Control Room Logs back to and including day shift on May 5. Further back is not required and anything less is not appropriate.
C.	All Control Room Logs back to and including day shift on May 8. Further back is not required and anything less is not appropriate.
D.	Just the day and night shift Control Room logs for May 9. Further back is not required and anything less is not appropriate.

K&A #	G2.1.3
Importance Rating	3.7
QUESTION 66	
K&A Statement:	Knowledge of shift or short term relief turnover practices.
Justification:	
A.	Incorrect because the last 7 days is not the requirement, but plausible if the candidate does not know the requirement then the last 7 days seems reasonable.
B.	Incorrect because the last 5 days is not the requirement, but plausible if the candidate does not know the requirement then the last 5 days seems reasonable.
C.	Correct – Per CNG-OP-1.01-2002, the operator is required to "Review the Control Room logs going back to their last watch, or 48 hours, whichever is shorter."
D.	Incorrect because the last 24 hours is not the requirement for reading control room logs, but plausible because there are other requirements related to the last 24 hours.

Examination Outline Cross-Reference:

Level	RO
Tier #	3
Group #	
K/A #	2.1.29
Importance Rating	4.1

Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

Proposed Question: #67

Unit 1 is operating at 100% power with the following:

- SO-151-001A, Monthly Core Spray A Loop Discharge Line Filled & Valve Alignment Verification, is in progress.
- HV-152-F005A, CORE SPRAY LOOP A IB INJ SHUTOFF, must be determined to be CLOSED.

Given the following possible methods to determine HV-152-F005A is CLOSED:

(1) Observe position indicating lights on Control Room panel 601
(2) Observe local valve position indication in the Reactor Building
(3) Take manual handwheel control and attempt to re-position the valve in the CLOSED direction

Which one of the following identifies which of these methods is(are) used, in accordance with SO-151-001A?

A. (1) only

B. (2) only

C. (3) only

D. (1), (2), and (3)

Proposed Answer: A

Explanation: HV-152-F005A is a normally closed, motor-operated valve that has remote position indication and control in the Control Room. SO-151-001A utilizes the remote position indication in the Control Room (indicating lights) to verify correct alignment.

B. Incorrect – SO-151-001A utilizes the remote position indication in the Control Room (indicating lights) to verify correct alignment. Plausible because an alternate method could be to use local position indication.
C. Incorrect – SO-151-001A utilizes the remote position indication in the Control Room (indicating lights) to verify correct alignment. Plausible because an alternate method could be to use local manual handwheel control to check the valve in the CLOSED direction.
D. Incorrect – SO-151-001A utilizes only the remote position indication in the Control Room (indicating lights) to verify correct alignment. Plausible because an alternate method could be to use local position indication or local manual handwheel control to check the valve in the CLOSED direction.

Technical Reference(s): SO-151-001A

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.17
	Importance Rating	2.6

Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, coordination with the transmission system operator.

Proposed Question: #68

Unit 1 is operating at 100% power with the following:

- Leakage has been reported on Circulating Water 1A pump.
- The pump must be quickly removed from service to allow for isolation and maintenance.
- A rapid power reduction is directed per ON-RPR-101, Rapid Power Reduction.

Which one of the following identifies the need to notify GCC and/or TCC, in accordance with ON-RPR-101?

Notify...

A. GCC after initiating the power reduction.

B. GCC prior to initiating the power reduction.

C. TCC after initiating the power reduction.

D. TCC prior to initiating the power reduction.

Proposed Answer: A

Explanation: ON-RPR-101 requires immediate operator action to initiate a #2 Limiter on Reactor Recirculation, which initiates the power reduction. Subsequent action requires notifying GCC.

B. Incorrect – The notification is made after initiating the power reduction. Plausible because this power reduction is not needed for immediate Reactor safety that the notification would be made prior to reducing output to the grid.
C. Incorrect – GCC must be notified, not TCC. Plausible because these are both organizations dealing with the station's connection with the electrical grid and because other situations would require notification to TCC (eg. removing startup bus from service, ON-GENGRID-101).
D. Incorrect – GCC must be notified, not TCC. Plausible because these are both organizations dealing with the station's connection with the electrical grid and because other situations would require notification to TCC (eg. removing startup bus from service, ON-GENGRID-101). The notification is made after initiating the power reduction. Plausible because this power reduction is not needed for immediate Reactor safety that the notification would be made prior to reducing output to the grid.

Technical Reference(s): ON-RPR-101

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.20
	Importance Rating	2.6

Knowledge of the process for managing troubleshooting activities.

Proposed Question: #69

Unit 1 is operating at 100% power with the following:

- Electrical Maintenance is preparing to perform troubleshooting on a lighting panel.
- During Phase 1 of the troubleshooting, it is desired to have the panel's disconnect switch tagged open for personnel protection.
- During Phase 2 of the troubleshooting, it is desired to close the panel's disconnect switch.
- During Phase 3 of the troubleshooting, it is desired to have the panel's disconnect switch tagged open for personnel protection.

Which one of the following describes a tagging arrangement that will allow this troubleshooting to occur, in accordance with NDAP-QA-0322, Clearance and Tagging?

Tag the panel's disconnect switch with...

A. a Danger tag, only. A T-Lift is required during Phase 2.

B. a Danger tag, only. The tag may hang continuously during all phases of the troubleshooting.

C. a Test & Maintenance tag, only. The tag may hang continuously during all phases of the troubleshooting.

D. both a Danger tag and a Test & Maintenance tag. Both tags may hang continuously during all phases of the troubleshooting.

Proposed Answer: A

Explanation: To get the personnel protection required during Phases 1 and 3, a Danger tag is required. The disconnect cannot be closed while the Danger tag is still hanging during Phase 2, but a T-Lift allows temporarily removing the tag to allow the manipulation and then re-application of the tag for Phase 3.

B. Incorrect – The Danger tag cannot continue to hang during Phase 2 to allow closing the disconnect switch. Plausible because the Danger tag is the correct tag, and there is an administrative way to allow closing the disconnect, but it temporarily removes the tag.
C. Incorrect – A Test & Maintenance tag cannot be used for the personnel protection desired during Phases 1 and 2. Plausible because the Test & Maintenance tag would allow the manipulation required during Phase 2, and would allow opening the disconnect during Phases 1 and 3, however administratively this cannot be used to claim personnel protection.
D. Incorrect – A Danger tag and a Test & Maintenance tag are not allowed to be placed on the same component at the same time. Plausible because a Danger tag provides the required personnel protection and the Test & Maintenance tag separately provides the ability to manipulate a component.

Technical Reference(s): NDAP-QA-0322

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - NMP1 2013 NRC #71

Question History: NMP1 2013 NRC #71

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	3
Group #	
K/A #	2.3.7
Importance Rating	3.5

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Proposed Question: #70

Both Units are operating at 100% power with the following:

- Operator entry into a RWCU pump room is required to respond to an off-normal condition.
- The highest general area radiation level in that room is 120 mRem/hr.
- The highest loose surface contamination level is 2000 dpm/100 cm^2 Beta-Gamma.

Which one of the following describes the expected radiation and contamination designations on the Radiation Work Permit (RWP) for this activity, in accordance with NDAP-QA-0626?

	Radiation Designation	Contamination Designation
A.	Radiation Area	NOT a Contaminated Area
B.	Radiation Area	Contaminated Area
C.	High Radiation Area	NOT a Contaminated Area
D.	High Radiation Area	Contaminated Area

Proposed Answer: D

Explanation: This is a High Radiation Area because there are general area dose rates >100 mRem/hr. This is a Contaminated Area because there is loose surface contamination >1000 dpm/100 cm^2 Beta-Gamma.

A. Incorrect – This is a High Radiation Area, not just a Radiation Area. Plausible because general area radiation levels are only slightly above the threshold for a High Radiation Area and well below the level of a Locked High Radiation Area. This area is a Contaminated Area. Plausible because the contamination levels are only slightly above the threshold for a Contaminated Area dn well below the level of a Highly Contaminated Area.
B. Incorrect – This is a High Radiation Area, not just a Radiation Area. Plausible because general area radiation levels are only slightly above the threshold for a High Radiation Area and well below the level of a Locked High Radiation Area.
C. Incorrect – This area is a Contaminated Area. Plausible because the contamination levels are only slightly above the threshold for a Contaminated Area and well below the level of a Highly Contaminated Area.

Technical Reference(s): NDAP-QA-0626

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – JAF 9/12 NRC #72

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(12)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.11
	Importance Rating	3.8

Ability to control radiation releases.

Proposed Question: #71

Unit 1 is operating at 100% power with the following sequence:

Time (minutes)	Condition
0	• Annunciator AR-106-G03, OFF GAS HI RADIATION, alarms and is verified valid.
5	• Annunciator AR-111-C03, MN STM LINE RAD MONITOR HI RADIATION, alarms and is verified valid.
7	• Annunciator AR-106-F03, OFF GAS HI HI RADIATION, alarms and is verified valid.
12	• Annunciator AR-103-D01, MN STM LINE HI HI RADIATION, alarms and is verified valid. • Annunciator AR-104-D01, MN STM LINE HI HI RADIATION, alarms and is verified valid.

Which one of the following identifies the ***first*** time that a Reactor scram is ***required***, in accordance with ON-MSLRAD-101, Rising Offgas/MSL Rad Levels?

A. 0 minutes

B. 5 minutes

C. 7 minutes

D. 12 minutes

Proposed Answer: D

Explanation: All of the given alarms require entry into ON-MSLRAD-101. A Reactor scram is first required when Main Steam Line radiation monitors exceed the Hi-Hi alarm setpoint (12 minutes).

A. Incorrect – A Reactor scram is first required when Main Steam Line radiation monitors exceed the Hi-Hi alarm setpoint (12 minutes). Plausible because this alarm requires entry into ON-MSLRAD-101, extra monitoring, and actions to attempt to control release.
B. Incorrect – A Reactor scram is first required when Main Steam Line radiation monitors exceed the Hi-Hi alarm setpoint (12 minutes). Plausible because this alarm requires entry into ON-MSLRAD-101, extra monitoring, and actions to attempt to control release.
C. Incorrect – A Reactor scram is first required when Main Steam Line radiation monitors exceed the Hi-Hi alarm setpoint (12 minutes). Plausible because this alarm requires entry into ON-MSLRAD-101, extra monitoring, and actions to attempt to control release.

Technical Reference(s): ON-MSLRAD-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-079E RBO-7

Question Source: Modified Bank - LOC26 NRC #74

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Exam	RO	Tier	3	Group	N/A	Cognitive Level	Low	Level of Difficulty	3
K/A	2.3.11 Radiation Control					Importance	3.8		
Statement	Ability to control radiation releases.								

QUESTION 74

Unit 1 is operating at rated power when annunciator OFF-GAS HI RADIATION (AR-106-G03) is received.

The reading from Off Gas Pre Treatment Log Radiation Monitoring recorder (RR-D12-1R601) is determined to be valid and has just exceeded Lim1.

Which one of the following identifies the next action required due to exceeding Lim1?

A.	Scram the reactor and close the MSIVs and MSL drains
B.	Immediately reduce power to lower Offgas pretreatment activity to < 150,000 µCi/sec
C.	Contact Chemistry to obtain an Offgas pretreatment sample
D.	Verify the Offgas system is not bypassed immediately

Proposed Answer	C	
Applicant References	None	
Explanation	Offgas Hi alarm and Offgas readings exceeding Lim1 require entry into ON-179-002. The AR for the Offgas Hi alarm directs checking the readings on the Offgas pretreat recorder and evaluating entry into the ON. ON-179-002 describes Lim1 as set 50 percent above nominal steady-state background levels. With Lim1 set at a relatively low level this facilitates compliance with TS 3.7.5 for Offgas activity by ensuring pretreat samples are obtained to determine the actual Offgas activity level.	
	A	Incorrect. With Offgas pretreat readings just 50 percent higher than nominal background readings, MSL radiation levels will not have risen to the hi-hi alarm setpoint. Closure of the MSIVs is premature at this time.
	B	Incorrect. With Offgas pretreat readings just 50 percent higher than nominal background readings, actual Offgas activity levels remain at a very small fraction (<1 percent typically) of the TS 3.7.5 LCO limit. Action to reduce power to maintain Offgas activity less than half of the TS 3.7.5 limit will not be required with pretreat rad levels just exceeding Lim1.
	C	Correct. ON-179-001 Step 4.6 describes this action in response to Offgas pretreat readings above Lim1. Obtaining an Offgas pretreatment grab sample will allow determination of compliance with TS 3.7.5 limits.
	D	Incorrect. This is the TRM 3.7.7 Required Action and Completion Time for no operable Offgas pretreatment log radiation monitor. The question stem specifically identifies the reading as valid.
10CFR55	41.11	

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.22
	Importance Rating	3.6

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Proposed Question: #73

Unit 1 has experienced a loss of coolant accident with the following:

- Reactor pressure is 75 psig, down slow, with all ADS SRVs open.
- Reactor water level is -180", down slow, with Condensate injecting.
- Suppression Pool average water temperature is 180°F, up slow.
- Suppression Pool level is 23', steady.
- Suppression Chamber pressure is 23 psig, up slow.
- Drywell pressure is 25 psig, up slow.
- NO Core Spray pumps are operating.
- RHR pump A has just been made available.
- NO other RHR pumps are available.

Note: Heat Capacity Temperature Limit (HCTL) and Pressure Suppression Limit (PSL) are provided on the following page.

Which one of the following identifies the priority for use of RHR pump A for these conditions?

A. LPCI

B. Drywell spray

C. Suppression Pool cooling

D. Suppression Chamber spray

FIG 2 HCTL
HEAT CAPACITY TEMPERATURE LIMIT

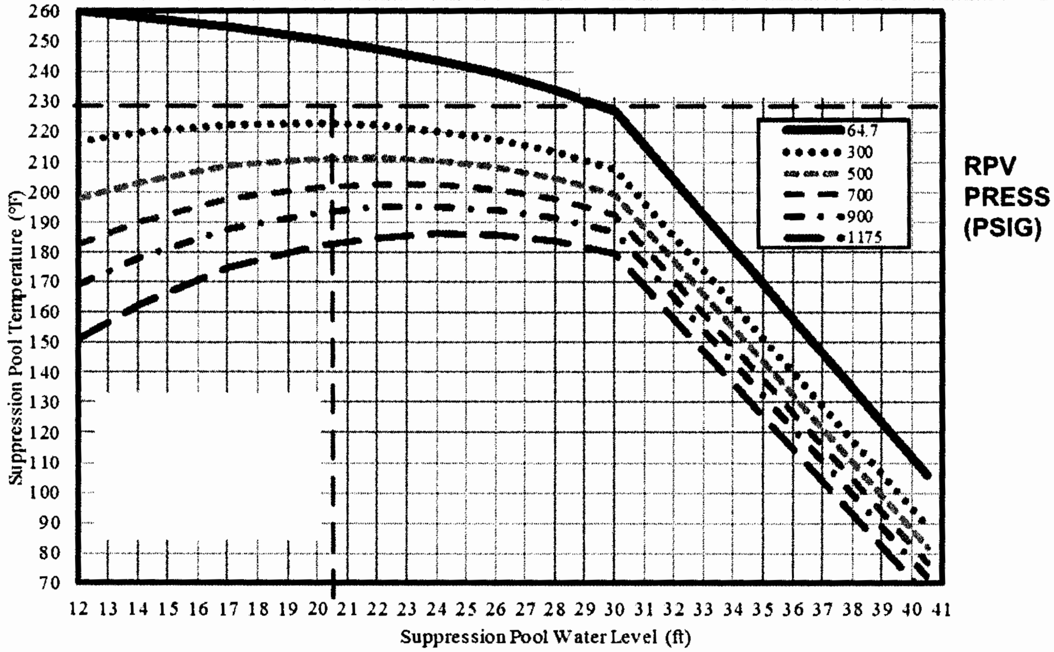
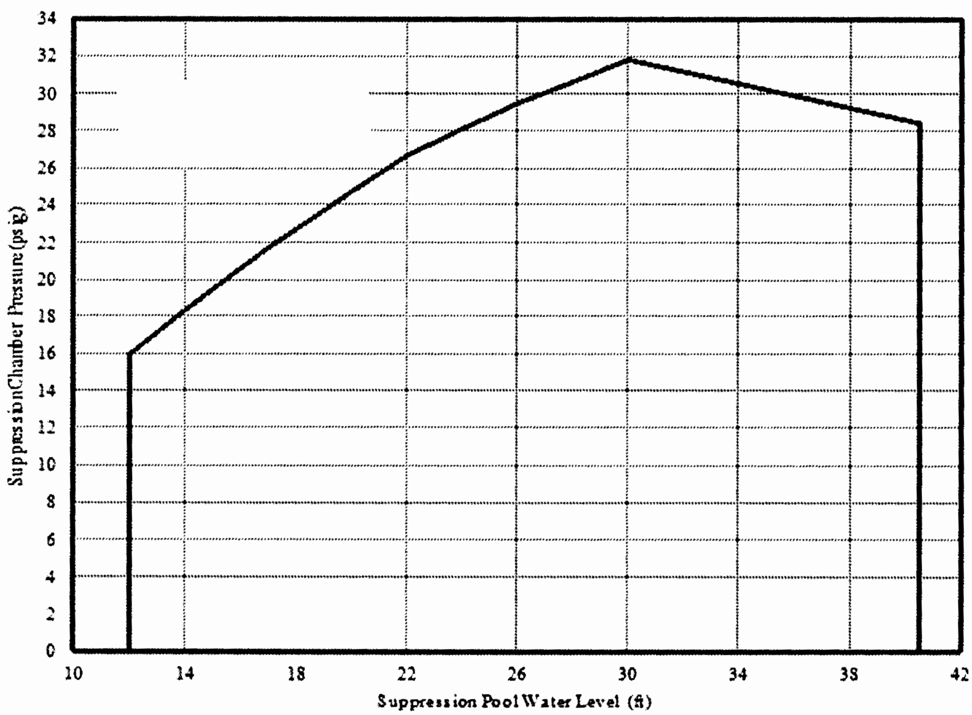


FIG. 4 PSL
PRESSURE SUPPRESSION LIMIT



Proposed Answer: A

Explanation: The given plant conditions present a loss of the RCS barrier and multiple challenges to the fuel clad and Primary Containment barriers. With Reactor water level <-179", down slow, and no Core Spray injection, adequate core cooling is not currently assured. Suppression Pool temperature is well above the 90°F threshold requiring maximized cooling and HCTL is a concern. Both Suppression Chamber and Drywell pressures are above thresholds requiring use of sprays. Therefore, there are steps in EO-100-102 and EO-100-103 that currently require use of the single RHR pump for LPCI, Suppression Pool cooling, Suppression Chamber spray, and Drywell spray. The bases for prioritizing these conflicting needs while in EO-100-102 and EO-100-103 is to prioritize adequate core cooling. Therefore, RHR pump A must first be used in LPCI mode to inject to the Reactor.

B. Incorrect – LPCI must be prioritized over Drywell spray. Plausible because EO-100-103 does required Drywell spray and PSP is being challenged. Also plausible because in some circumstances during severe accidents Primary Containment is prioritized over fuel clad.
C. Incorrect – LPCI must be prioritized over Suppression Pool cooling. Plausible because EO-100-103 does required Suppression Pool cooling and HCTL is being challenged. Also plausible because in some circumstances during severe accidents Primary Containment is prioritized over fuel clad.
D. Incorrect – LPCI must be prioritized over Suppression Chamber spray. Plausible because EO-100-103 does required Suppression Chamber spray and PSP is being challenged. Also plausible because in some circumstances during severe accidents Primary Containment is prioritized over fuel clad.

Technical Reference(s): EO-100-102, EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.2
	Importance Rating	4.5

Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Proposed Question: #74

Unit 1 is operating at 50% power with the following:

- A Reactor scram occurs.
- HPCI and RCIC automatically start.
- RCIC develops a steam leak.
- RCIC automatically isolates due to RCIC equipment area temperature.
- All actuations were based on valid signals.
- Core Spray and RHR pumps remain in standby.

Regarding the following EOP entry conditions:

(1) EO-100-102, RPV Control, low Reactor water level entry condition
(2) EO-100-104, Secondary Containment Control, high area temperature entry condition

Note: The EO-100-104 Reactor Building Temperatures table is provided on the next page.

Based on the system response to the given conditions, which one of the following identifies which of these EOP entry conditions **must** have been met, if any?

A. (1) only

B. (2) only

C. Both (1) and (2)

D. Neither (1) nor (2)

TABLE 8

TABLE 8 REACTOR BUILDING TEMPERATURE

RB AREA		MAX NORMAL ΔTEMP	TEMP	MAX SAFE TEMP	RECORDER POINTS	RB TEMP
EL (FT)		(°F)	(°F)	(°F)	Δ T, T	(°F)
818	GENERAL AREA	N/A	110	120		______ ______
779	GENERAL AREA	N/A	110	120		______ ______
749	GENERAL AREA	N/A	110	120		______ ______
	RWCU-PUMP ROOM	45	120	147	18,10	______ ______
	RWCU-HEAT EXCH ROOM	45	120	147	19,11	______ ______
	RWCU-PENETRATION ROOM	45	120	131	20,12	______ ______
719	GENERAL AREA	N/A	110	120		______ ______
	MAIN STEAM LINE TUN	60	157	177	7, 1	______ ______
683	GENERAL AREA	N/A	110	120		______ ______
	HPCI PIPE ROUTING AREA	45	120	167	17, 6	______ ______
	RCIC PIPE ROUTING AREA	45	120	167	9*, 3*	______ ______
670	GENERAL AREA	N/A	110	120		______ ______
645	HPCI-EQUIP AREA	45	120	167	15,4	______ ______
	HPCI-EMERG AREA COOLER	N/A	120	167	5	______ ______
645	RCIC-EMERG AREA COOLER	N/A	120	167	2*	______ ______
	RCIC-EQUIP AREA	45	120	167	7*,1*	______ ______
645	RHR EQUIP AREA 1	45	110	142	8,2	______ ______
645	RHR EQUIP AREA 2	45	110	142	9,3	______ ______
645	CS PUMP ROOM A	N/A	110	142		______ ______
645	CS PUMP ROOM B	N/A	110	142		______ ______
645	RB SUMP ROOM	N/A	110	125		______ ______

Proposed Answer: C

Explanation: RCIC automatically starts on Reactor water level less than -30" only. HPCI starts on Reactor water level less than -38" or Drywell pressure >1.72 psig. Since RHR and Core Spray have not started, Drywell pressure must not have exceeded 1.72 psig and Reactor water level must not have dropped to -129". The EO-00-102 low Reactor water level entry condition is +13". Since RCIC automatically started, the EO-100-102 low Reactor water level entry condition must have been met. RCIC automatically isolates on a high RCIC equipment area temperature of 167°F on the same temperature elements used in EO-100-104 Table 8. The EO-100-104 high area temperature entry condition includes these devices above 120°F. Since RCIC automatically isolated on RCIC equipment area temperature, the EO-100-104 high area temperature entry condition must have been met.

A. Incorrect – Since RCIC automatically isolated on RCIC equipment area temperature (167°F), the EO-100-104 high area temperature entry condition (120°F) must have been met also. Plausible that either different temperature elements would be used for isolation vs. EOP entry, or that the EOP entry would be at a higher temperature than the isolation, such that successful isolation would avoid EOP entry.
B. Incorrect – Since RCIC automatically started (-30"), the EO-100-102 low Reactor water level entry condition (+13") must have been met also. Plausible if candidate confuses RCIC auto start setpoint and EOP entry condition values, or believes RCIC may have started on an alternate signal.
D. Incorrect – Since RCIC automatically started (-30"), the EO-100-102 low Reactor water level entry condition (+13") must have been met also. Plausible if candidate confuses RCIC auto start setpoint and EOP entry condition values, or believes RCIC may have started on an alternate signal. Since RCIC automatically isolated on RCIC equipment area temperature (167°F), the EO-100-104 high area temperature entry condition (120°F) must have been met also. Plausible that either different temperature elements would be used for isolation vs. EOP entry, or that the EOP entry would be at a higher temperature than the isolation, such that successful isolation would avoid EOP entry.

Technical Reference(s): EO-100-102, EO-100-104, AR-108-F04/F05

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-050 RBO-7

Question Source: Bank – JAF 9/14 NRC #22

Question History: JAF 9/14 NRC #22

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.4
	Importance Rating	3.3

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: #75

A licensed Reactor Operator works the following schedule:

January 3	12-hour day shift as U-1 PCOM
January 4	12-hour day shift as U-1 PCOP
January 8	12-hour night shift as U-2 PCOP
January 9	12-hour night shift as U-2 PCOM

The Reactor Operator then is removed from watchstanding for an extended period of time, but maintains all training up to date and maintains medical clearance to stand watch.

Which one of the following identifies (1) when the Reactor Operator's license will first go inactive due to watchstanding hours and (2) the minimum number of watchstanding hours that will then be needed for license re-activation, in accordance with OP-AD-010, Control of Licensed Operator License Status, Restrictions, and Requirements?

	(1) The license will first go inactive...	(2) The minimum number of watchstanding hours for reactivation is then...
A.	April 1	40
B.	April 1	56
C.	July 1	40
D.	July 1	56

Proposed Answer: A

Explanation: The RO has stood watch for 4 12 hour shifts in the 1st quarter. A minimum of 5 12 hour shifts are required to maintain an active license. Therefore, the license first goes inactive on the first day of the 2nd quarter, or April 1. To re-activate, a minimum of 40 hours of watch standing is required.

B. Incorrect – The minimum hours for re-activation is 40, not 56. Plausible because 56 hours is the minimum needed for maintain active license status per quarter (7 8 hour shift).
C. Incorrect – The license first goes inactive on April 1, not July 1. Plausible because this would be correct if the RO had stood one more 12 hour watch in the first quarter.
D. Incorrect – The license first goes inactive on April 1, not July 1. Plausible because this would be correct if the RO had stood one more 12 hour watch in the first quarter. The minimum hours for re-activation is 40, not 56. Plausible because 56 hours is the minimum needed for maintain active license status per quarter (7 8 hour shift).

Technical Reference(s): OI-AD-044

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – LOC25 Cert #66

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.4	
	Importance Rating	3.3	

2.1.4 - Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: Common 66

A licensed Plant Control Operator worked the following schedule:

January 1 Off
January 2 Off
January 3 12-hour day shift as U-1 PCOM
January 4 12-hour day shift as U-1 PCOP
January 8 12-hour night shift as U-2 PCOP
January 9 12-hour night shift as U-2 PCOM

The PCO was then off-shift January 10 through March 31 due to illness.

All licensed operator training is up to date.

Medical clearance to stand watch with no restrictions has been obtained.

Which one of the following describes the status of the PCO's license and additional requirements, if any, to stand watch beginning April 1, 2013, in accordance with 10 CFR 55.53 – Conditions of Licenses?

A.	The license is Active because the PCO stood watch for at least 40 hours the previous quarter No additional requirements are needed to stand watch on 4/1/2013
B.	**The license is Inactive** **The license must be reactivated by performing shift functions, including a complete plant tour, under the direction of an active licensed PCO for at least 40 hours**
C.	The license is Inactive The license must be reactivated by performing shift functions, including a complete plant tour, under the direction of an active licensed PCO or SRO for at least 40 hours
D.	The license is Inactive