Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	203000 K1.01
	Importance Rating	2.8

RHR/LPCI: Injection Mode

Knowledge of the physical connections and/or cause-effect relationships between RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) and the following: Condensate storage and transfer system: Plant-Specific

Proposed Question: #1

The plant is operating at 100% power with the following:

- RHR is in a normal standby lineup.
- One Condensate Transfer pump is running and one is in standby.
- Then, the running Condensate Transfer pump trips.

Which one of the following describes the effect of this pump trip on RHR?

The normal keep-full source is...

A. maintained due to automatic start of the standby Condensate Transfer pump.

B. lost until the standby Condensate Transfer pump is manually started, but the backup keep-full source remains available.

C. maintained in service uninterrupted, but the backup keep-full source is lost until the standby Condensate Transfer pump is manually started.

D. maintained in service uninterrupted, and the backup keep-full source is maintained due to automatic start of the standby Condensate Transfer pump.

Proposed Answer: D

Explanation: The normal RHR keep-full source is the RHR keep-full pumps, with Condensate transfer available as a backup keep-full source. When the running Condensate Transfer pump trips, the standby Condensate Transfer pump auto-starts on low discharge pressure.

A. Incorrect – The normal keep-full source is NOT from Condensate Transfer.
B. Incorrect – The normal keep-full source is from the RHR keep-full pumps, NOT from Condensate Transfer.
C. Incorrect – The standby Condensate Transfer pump auto-starts on low discharge pressure.

Technical Reference(s): SDLP-10 Section C.12, OP-5 Section B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.10.d

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 #	400000 K1.04
	Importance Rating	2.9

Component Cooling Water

Knowledge of the physical connections and / or cause-effect relationships between CCWS and the following: Reactor coolant system, in order to determine source(s)

Proposed Question: #2

The plant is operating at 100% power with the following:

- RBCLC leakage is suspected.
- Given the following RBCLC loads:

  (1) Drywell Equipment Sump Cooler
  (2) RWCU Pump Coolers
  (3) Recirc Pump and Motor Coolers

Which one of the following lists which of these loads may be isolated from the RBCLC system using valves controlled from the Control Room?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: B

Explanation: Both the Drywell Sump Cooler and the Recirc Pump and Motor Coolers have RBCLC isolation valves that can be operated from Control Room panel 09-75. The RWCU Pump Coolers do NOT have RBCLC isolation valves that can be operated from the Control Room.

Note: This question meets the K/A by testing knowledge of the physical connections between the RBCLC system and the RWCU/Recirc system (which, in the event of a cooler leak, could cause RCS leakage into the RBCLC system). The question specifically tests knowledge of the valves provided to allow attempted isolation of RBCLC system components. These valves could be used as part of a leak identification and isolation process given RCS leakage into the RBCLC system.

A. Incorrect – The Recirc Pump and Motor Coolers have RBCLC isolation valves that can be operated from Control Room panel 09-75. The RWCU Pump Coolers do NOT have RBCLC isolation valves that can be operated from the Control Room.
C. Incorrect – The RWCU Pump Coolers do NOT have RBCLC isolation valves that can be operated from the Control Room.
D. Incorrect – The RWCU Pump Coolers do NOT have RBCLC isolation valves that can be operated from the Control Room.

Technical Reference(s): OP-40 Section B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15 1.11.b

Question Source: Modified Bank – NMP2 2013 Audit #23

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 #	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: #3

The plant is operating at 100% power with the following:

- MCC-152 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 A CANNOT automatically inject to the Reactor.

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

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

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

Proposed Answer: A

Explanation: MCC-152 supplies power to 14MOV-12A, the Core Spray A inboard injection valve. This valve is normally closed and must open for Core Spray A to automatically inject to the Reactor. Without power from MCC-152, Core Spray A CANNOT automatically inject to the Reactor.

B. Incorrect – Core Spray B injection valves are powered by MCC-162, not MCC-152.
C. Incorrect – Core Spray A minimum flow valve, 14MOV-5A, is normally open, but is powered from MCC-153, not MCC-152.
D. Incorrect – Core Spray B minimum flow valve, 14MOV-5B, is normally open, but is powered from MCC-163, not MCC-152.

Technical Reference(s): OP-14 Attachments 2A and 2B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14 1.04.c

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 #	212000 K2.01
	Importance Rating	3.2

RPS

Knowledge of electrical power supplies to the following: RPS motor-generator sets

Proposed Question: #4

The plant is operating at 100% power with the following:

- MCC-251 de-energizes due to a sustained electrical fault.

Which one of the following describes the effect of this fault on the Reactor Protection System (RPS)?

A. The backup power supply is lost to RPS A.

B. The backup power supply is lost to RPS B.

C. A half scram occurs on RPS A.

D. A half scram occurs on RPS B.

Proposed Answer: C

Explanation: MCC-251 supplies power to RPS MG Set A. When MCC-251 de-energizes, RPS MG Set A will stop supplying power to RPS A and a half scram will occur on RPS A.

A. Incorrect – The RPS A backup power supply is a transformer powered from MCC-252, not MCC-251.
B. Incorrect – The RPS B backup power supply is a transformer powered from MCC-262, not MCC-251.
D. Incorrect – RPS MG Set B is powered by MCC-261, not MCC-251.

Technical Reference(s): OP-18 Attachment 1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05 1.03.a

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 8/29/14 – Swapped order of answer choices based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 K3.11
	Importance Rating	2.8

UPS (AC/DC)

Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following: MSIVs: Plant-Specific

Proposed Question: #5

The plant is operating at 100% power with the following:

- A complete loss of the UPS occurs.
- UPS distribution bus 71ACUPS de-energizes.

Which one of the following describes the effect of this loss on the Main Steam Isolation Valves (MSIVs)?

A. The MSIVs automatically close.

B. MSIV automatic isolation capability is lost.

C. All MSIV position indication in the Main Control Room is lost.

D. MSIV position indication on the PCIS Graphic Display is lost, only.

Proposed Answer: D

Explanation: The only effect on loss of UPS power to the MSIVs is loss of indication on the PCIS graphic display panel on 09-3.

A. Incorrect – MSIV isolation logic is powered from the RPS buses, not the UPS bus, therefore the MSIVs remain open.
B. Incorrect – MSIV isolation logic is powered from the RPS buses, not the UPS bus, therefore the MSIVs isolation capability is unaffected.
C. Incorrect – Only the MSIV position indication on the PCIS graphic display is lost. Position indication near the control switches is still available.

Technical Reference(s): AOP-21

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16C 1.10.b

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 #	205000 K3.02
	Importance Rating	3.2

Shutdown Cooling

Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor water level: Plant-Specific

Proposed Question: #6

The plant is shutdown with the following:

- RHR B is operating in Shutdown Cooling mode.
- Both Recirculation loops are in service.
- Reactor water level is 195 inches and stable.

Then, a spurious signal causes isolation of Shutdown Cooling.

Which one of the following describes the status of Reactor water level, in accordance with AOP-30, Loss of Shutdown Cooling?

Reactor water level...

A. is within the preferred control band of 177 to 270 inches.

B. is within the preferred control band of 177 to 222.5 inches.

C. should be raised to the preferred control band of 200 to 270 inches.

D. should be raised to the preferred control band of 234.5 to 270 inches.

Proposed Answer: C

Explanation: With a loss of Shutdown Cooling, AOP-30 directs control of Reactor water level between 200 and 270 inches since Recirculation pumps are running. The initial Reactor water level of 195 inches is below this preferred band.

A. Incorrect – 177 inches is the low end of the Reactor water level control band in EOP-2. 177 inches is also an alternate Reactor water level limit in AOP-30 if the preferred band cannot be maintained.
B. Incorrect – 177 to 222.5 inches is the Reactor water level control band in EOP-2.
D. Incorrect – 234.5 to 270 inches is the preferred Reactor water level control band in AOP-30 only if no Recirculation pumps are running. The wider preferred band of 200 to 270 inches is given with Recirculation pumps running.

Technical Reference(s): AOP-30 Attachment 2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.15.a

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 #	223002 K4.06
	Importance Rating	3.4

PCIS/Nuclear Steam Supply Shutoff

Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: Once initiated, system reset requires deliberate operator action

Proposed Question: #7

The plant is operating at 100% power with the following:

- A Reactor scram occurs.
- A PCIS Group 2 isolation occurs on low Reactor water level.
- Reactor water level is restored and maintained in a band of 180-220 inches.

Which one of the following describes the operation of the Reactor Water Cleanup (RWCU) isolation valves when Reactor water level is restored above the PCIS group 2 isolation reset point?

RWCU isolation valves...

A. automatically re-open.

B. remain closed until the PCIS signal is manually reset. The valves then automatically re-open.

C. remain closed until the PCIS signal is manually reset and the RWCU isolation valve control switches are taken to OPEN.

D. remain closed until the RWCU isolation valve control switches are taken to OPEN. Manual reset of the PCIS signal is NOT required.

Proposed Answer: C

Explanation: The PCIS Group 2 isolation signal seals in. Once Reactor water level is restored above the reset point, the isolation signal can be manually reset, but does not automatically reset. Once the isolation signal is manually reset, the RWCU isolation valves can be re-opened by taking their control switches to OPEN, but they do not automatically re-open.

A. Incorrect – The PCIS signal seals in and must be manually reset.
B. Incorrect – Even once the PCIS signal is manually reset, the RWCU isolation valves remain closed until operator action is taken to re-open them.
D. Incorrect – The PCIS signal seals in and must be manually reset.

Technical Reference(s): AOP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16C 1.05.a.4

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 #	215004 K4.04
	Importance Rating	2.8

Source Range Monitor

Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Changing detector position

Proposed Question: #8

A plant startup is in progress with the following:

- All SRMs are fully inserted and indicating approximately 50 cps.
- All IRMs are on range 1.
- SRM A is selected and given a withdraw signal.

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

SRM A...

A. remains fully inserted and generates a rod block signal.

B. withdraws and immediately generates a rod block signal.

C. remains fully inserted and does NOT generate a rod block signal.

D. withdraws and generates a rod block signal only if indication drops below 3 cps.

Proposed Answer: B

Explanation: With SRM A counts below 100 cps, the RETRACT PERMIT light will NOT be lit, however the SRM will still withdraw if given a signal. With SRM A counts less than 100 cps and IRMs below Range 3, the SRM Detector Not Full In rod block will be received as soon as SRM A is not fully inserted.

A. Incorrect – While the RETRACT PERMIT light will NOT be lit, this is indication only and does NOT actually stop SRM A from withdrawing.
C. Incorrect – While the RETRACT PERMIT light will NOT be lit, this is indication only and does NOT actually stop SRM A from withdrawing.
D. Incorrect – SRM A would also generate a rod block if counts lowered below 3 cps, however the SRM Detectors Not Full In rod block will occur first.

Technical Reference(s): OP-16, SDLP-07B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.05.a.1.d

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 #	259002 K5.03
	Importance Rating	3.1

Reactor Water Level Control

Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM: Water level measurement

Proposed Question: #9

The plant is operating at 100% power.

Which one of the following describes the negative consequence of the Reactor Water Level Reference Leg Backfill System being out of service for an extended period of time per OP-27A, Reactor Water Level Reference Leg Backfill System?

Reactor water level inputs to the Feedwater Level Control system may indicate…

A. lower than actual during plant depressurization.

B. lower than actual during steady-state operation.

C. higher than actual during plant depressurization.

D. higher than actual during steady-state operation.

Proposed Answer: C

Explanation: The Reactor Water Level Reference Leg Backfill System is required to flush non-condensable gases out of the level instrumentation reference legs. If these non-condensable gases are NOT removed, then level indication may rise during Reactor depressurization as the gases come out of solution and affect the height of water in the reference leg. This causes the level indication to be higher than actual. These erroneous Reactor water level indications in turn affect how the Feedwater level control system controls actual Reactor water level.

A. Incorrect – Reactor water level will indicate higher than actual, not lower.
B. Incorrect – Reactor water level will indicate higher than actual, not lower. The effect is seen during depressurization, not during steady-state operation at 100% power.
D. Incorrect – The effect is seen during depressurization, not during steady-state operation at 100% power.

Technical Reference(s): OP-27A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02B 1.09.a

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 8/29/14 – Added "per OP-27A" to question based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 K5.01
	Importance Rating	3.8

ADS

Knowledge of the operational implications of the following concepts as they apply to AUTOMATIC DEPRESSURIZATION SYSTEM: ADS logic operation

Proposed Question: #10

The plant is operating at 100% power with the following:

- RHR pumps B and D are tagged out of service.
- A coolant leak develops in the Drywell.
- Core Spray pump A fails to start.
- The 10600 bus de-energizes.
- RHR pump A is running with a discharge pressure of 195 psig.
- Reactor water level is 59 inches and lowering slowly (time 0 seconds).

90 seconds later, RHR pump A shows signs of cavitation:

- RHR pump A discharge pressure is oscillating between 85-115 psig.
- Reactor water level is 20 inches and continues to slowly lower.
- No EOP actions have been taken yet.

Which one of the following describes the response of the Automatic Depressurization System (ADS) timer and the ADS valves?

	ADS Timer	ADS Valves
A.	Resets to zero	Open at time 120 seconds
B.	Resets to zero	Do NOT open at time 120 seconds
C.	Continues to time out	Open at time 120 seconds
D.	Continues to time out	Do NOT open at time 120 seconds

Proposed Answer: D

Explanation: The ADS timer (~120 seconds) begins timing when Reactor water level drops below 59.5 inches. The timer continues timing unless Reactor water level rises back above 59.5 inches. Since Reactor water level is still below 59.5 inches, the timer continues timing. However, ADS valves do NOT open when the timer times out unless either an RHR or Core Spray pump operating with discharge pressure above 125 or 100 psig, respectively. RHR pumps B and D are unavailable due to tagout. RHR pump C and Core Spray pump B are unavailable due to loss of 10600 bus. Core Spray pump A failed to start. Since only RHR pump A is available, and it is not providing a discharge pressure of 125 psig, the ADS valves do NOT open when the ADS timer times out.

A. Incorrect – Reactor water level remains below 59.5 inches, therefore the ADS timer does NOT reset to zero. ADS valves do NOT open due to lack of adequate ECCS pressure.
B. Incorrect – Reactor water level remains below 59.5 inches, therefore the ADS timer does NOT reset to zero.
C. Incorrect – ADS valves do NOT open due to lack of adequate ECCS pressure.

Technical Reference(s): OP-68, SDLP-02J

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J 1.10.a

Question Source: Bank – March 2012 NRC #38

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 #	261000 K6.05
	Importance Rating	3.1

SGTS

Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM: Reactor protection system: Plant-Specific

Proposed Question: #11

The plant is operating at 100% power with the following:

- Standby Gas Treatment (SGT) train A is running for performance of ST-7BA, Monthly SGT Train A Run.
- Then, a sustained electrical fault causes RPS bus A to de-energize.

Which one of the following describes the effect of this malfunction on SGT trains A and B?

	SGT Train A	SGT Train B
A.	Remains in operation	Remains in standby
B.	Remains in operation	Automatically initiates
C.	Automatically shuts down	Remains in standby
D.	Automatically shuts down	Automatically initiates

Proposed Answer: A

Explanation: Loss of RPS bus A causes automatic isolation of Reactor Building ventilation train A and automatic initiation of SGT train A. Reactor Building ventilation train B and SGT train B are unaffected. Therefore, SGT train A remains running (now with an initiation signal) and SGT train B remains in standby.

B. Incorrect – SGT train B is unaffected by the loss of RPS bus A.
C. Incorrect – While Reactor Building ventilation train A automatically shuts down, SGT train A remains in service.
D. Incorrect – While Reactor Building ventilation train A automatically shuts down, SGT train A remains in service. SGT train B is unaffected by the loss of RPS bus A.

Technical Reference(s): AOP-59, AOP-60

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05 1.09.a

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 #	217000 K6.03
	Importance Rating	3.5

RCIC

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Suppression pool water supply

Proposed Question: #12

The plant has experienced a loss of coolant accident with the following:

- RCIC is injecting 400 gpm to the Reactor.
- CST level is 57 inches and slowly lowering.
- Torus water level is 10.8 feet and slowly lowering.

Which one of the following describes the effect of these conditions on the operation of RCIC?

RCIC is currently operating with suction from the...

A. CSTs. If CST level continues to lower, the RCIC suction path will automatically swap to the Torus.

B. Torus. If Torus level continues to lower, the RCIC suction path will automatically swap to the CSTs.

C. CSTs. If CST level continues to lower, the RCIC suction path will stay aligned to the CSTs and RCIC pump vortexing may become a concern.

D. Torus. If Torus level continues to lower, the RCIC suction path will stay aligned to the Torus and RCIC pump vortexing may become a concern.

Proposed Answer: D

Explanation: RCIC suction is normally aligned to the CSTs. However, RCIC suction has already swapped to the Torus due to CST level < 59.5 inches. Torus water level is below the EOP entry and Tech Spec low level and approaching the 10.75' action level in EOP-4, but there is no automatic swap back to the CSTs once RCIC suction has swapped to the Torus. RCIC pump vortexing becomes a concern if Torus water level continues to lower below 5.7 feet.

A. Incorrect – RCIC suction is currently from the Torus because CST level is < 59.5 inches.
B. Incorrect – While there is an auto swap from CSTs to Torus, there is no auto swap from the Torus to the CSTs.
C. Incorrect – RCIC suction is currently from the Torus because CST level is < 59.5 inches.

Technical Reference(s): OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13 1.05.b.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 #	264000 A1.01
	Importance Rating	3.0

EDGs

Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including: Lube oil temperature

Proposed Question: #13

The plant is shutdown following a small LOCA with the following:

- 115 KV Lines 3 and 4 de-energize.
- Annunciator 09-8-2-9, EDG A ENG TROUBLE OR SHUTDOWN, alarms.
- Annunciator 93ECP-A-03, High Lube Oil Temp, alarms.

Which one of the following describes if EDG A will automatically trip on high lube oil temperature signal?

EDG A will...

A. NOT automatically trip.

B. automatically trip, regardless of Drywell pressure and Reactor water level.

C. automatically trip, but only if Drywell pressure is below 2.7 psig AND Reactor water level is above 59.5".

D. automatically trip, but only if Drywell pressure is below 2.7 psig OR Reactor water level is above 59.5".

Proposed Answer: A

Explanation: High lube oil temperature is an alarm condition for EDG A, but NOT an automatic trip (such as low lube oil pressure or high jacket water temperature). Therefore, EDG A will NOT automatically trip on a high lube oil temperature signal.

B. Incorrect – EDG A will NOT automatically trip on a high lube oil temperature signal. Engine overspeed and incomplete start cause EDG A to automatically trip regardless of Drywell pressure and Reactor water level.
C. Incorrect – EDG A will NOT automatically trip on a high lube oil temperature signal. Low lube oil pressure and high jacket water temperature trips require both Drywell pressure and Reactor water level to be SAT.
D. Incorrect – EDG A will NOT automatically trip on a high lube oil temperature signal. Low lube oil pressure and high jacket water temperature trips require both Drywell pressure and Reactor water level to be SAT.

Technical Reference(s): ARP 09-8-2-09, ARP 93ECP-A-03

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-93 1.05.c.1

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:
TRH 8/29/14 – Deleted extra "will" in choice A based on NRC comment.

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 A1.06
	Importance Rating	3.3

IRM

Ability to predict and/or monitor changes in parameters associated with operating the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM controls including: Lights and alarms

Proposed Question: #14

A plant startup is in progress with the following:

- The Mode Switch is in START & HOT STBY.
- All Intermediate Range Monitor (IRM) detectors are indicating mid-scale on range 5.
- Then, the IRM A mode switch is moved from OPERATE to STANDBY on Panel 09-12.

Which one of the following describes the resulting status of Panel 09-5 IRM A indicating lights?

	IRM A UPSC TR OR INOP light	IRM A BYP light
A.	Lit	Lit
B.	Lit	NOT lit
C.	NOT lit	Lit
D.	NOT lit	NOT lit

Proposed Answer: B

Explanation: When IRM A mode switch is taken out of OPERATE, an INOP signal is generated. This illuminates the IRM A UPSC TR OR INOP light on Panel 09-5, but not the IRM A BYP light.

A. Incorrect – The IRM A BYP light is not lit.
C. Incorrect – The IRM A UPSC TR OR INOP light is lit and the IRM A BYP light is not lit.
D. Incorrect – The IRM A UPSC TR OR INOP light is lit.

Technical Reference(s): ARP-09-5-2-52, 1.66-180

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.05.a.6.a, 1.11.a.7.a, 1.11.a.7.d

Question Source: Bank – 9/12 NRC #38

Question History: 9/12 NRC #38

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 #	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: #15

The plant is operating at 25% power with the following:

- Feedwater pump A and the Main Turbine are in service.
- 4160V Buses have been transferred to Normal Station Service and Bus 10700 has been energized.

Then, the following annunciators alarm:

- 09-8-1-19, 125VDC BATT CHGR A AC SUPP TROUBLE
- 09-8-1-20, 125VDC BATT A VOLT LO

Station Battery A volts indicate 0.

Which one of the following actions will occur in accordance with AOP-45, Loss of DC Power System A?

A. RWR MG Set A runs back to the 30% limiter.

B. EDGs A and C start and energize Bus 10500.

C. ADS Logic A transfers to DC Power System B.

D. UPS 71UPP transfers from AC to DC input power.

Proposed Answer: C

Explanation: The given indications show a complete loss of DC Power System A. This will result in many automatic actions, including ADS Logic A transferring to DC Power System B.

A. Incorrect – The Reactor will scram, which would normally result in a runback of RWR MG Set A. However, the loss of DC Power System A causes RWR MG Set scoop tube to lock up, which will prevent a runback.
B. Incorrect – Bus 10500 does de-energize due to the loss of DC Power System A, however EDGs A and C also lose control power, and therefore fail to start and re-energize the bus.
D. Incorrect – The UPS does lose control power, but DC input power is also lost.

Technical Reference(s): AOP-45

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B 1.14.a

Question Source: Bank - 2010 NRC #45

Question History: 2010 NRC #45

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 #	215005 A2.07
	Importance Rating	3.2

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: Recirculation flow channels flow mismatch

Proposed Question: #16

The plant is operating at 90% power with the following:

- Annunciator 09-5-2-25, Flow Ref Off Norm, alarms.
- Total core flow indicates 85% of rated.
- The Recirculation flow units indicate as follows:

Flow Unit	Flow %
A	85
B	97
C	86
D	84

Which one of the following describes the plant impact of these conditions and the method used to bypass the Flow Unit B input to the flow comparator in accordance with OP-16, Neutron Monitoring?

These conditions result in a __(1)__. The Flow Unit B input to the flow comparator is bypassed per OP-16 by repositioning a __(2)__.

	(1)	(2)
A.	rod block, but NO half scram	joystick on Panel 09-5
B.	rod block, but NO half scram	switch on Panel 09-14
C.	rod block and a half scram	joystick on Panel 09-5
D.	rod block and a half scram	switch on Panel 09-14

Proposed Answer: A

Explanation: The given conditions show Flow Unit B is indicating higher than the rest of the Flow Units. The outputs of Flow Units B and D are compared and if they differ by more than 10%, they result in a rod block. The output of Flow Units B and D are also auctioneered to supply the flow reference signal to the APRM B, D, and F flow biased scram circuitry. The lower flow of Flow Unit B and D is used, so in this case, Flow Unit D continues to supply accurate indication to the APRMs and no half scram would result. OP-16 section E.21 provides the direction for bypassing a Flow Unit. A joystick on Panel 09-5 is used to bypass the flow unit from the comparator function. A switch is used at panel 09-14 if it is also desired to bypass other functions of the Flow Unit.

B. Incorrect – A joystick on Panel 09-5 is used to bypass the flow comparator function of Flow Unit B, NOT a switch on Panel 09-14.
C. Incorrect – The flow comparator >10% difference function only causes a rod block. A half scram would occur if a flow unit failed low enough to cause an APRM flow-biased scram.
D. Incorrect – The flow comparator >10% difference function only causes a rod block. A half scram would occur if a flow unit failed low enough to cause an APRM flow-biased scram. A joystick on Panel 09-5 is used to bypass the flow comparator function of Flow Unit B, NOT a switch on Panel 09-14.

Technical Reference(s): ARP 09-5-2-25, OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07C 1.10.d

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 A3.02
	Importance Rating	3.8

HPCI

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

Proposed Question: #17

The plant is operating at 50% power with the following:

- ST-4N, HPCI Quick-Start, Inservice, and Transient Monitoring Test, is in progress.
- HPCI is running with its discharge aligned to the CSTs.
- HPCI flow and flow controller conditions are as shown in the pictures on the following page.

Then, the following occur:

- The Reactor scrams.
- Reactor water level lowers to a low of 120 inches.

Which one of the following describes the resulting operation of HPCI?

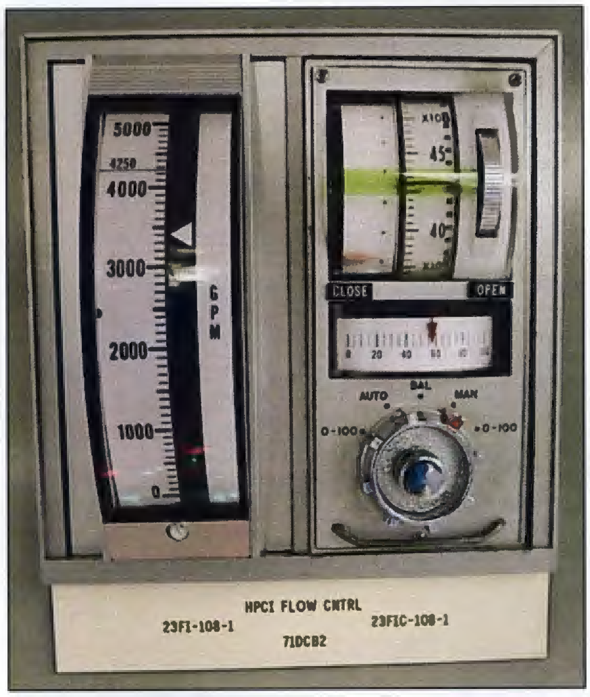
HPCI discharge flow...

A. continues to be aligned to the CSTs and is controlled by the manual demand signal.

B. automatically re-aligns to the Reactor and is controlled by the manual demand signal.

C. continues to be aligned to the CSTs and automatically controls at approximately 4250 gpm.

D. automatically re-aligns to the Reactor and automatically controls at approximately 4250 gpm.


5000
4250
4000
3000
2000
1000
0
GPM
CLOSE
OPEN
AUTO
BAL
MAN
0-100
HPCI FLOW CNTRL
23FI-108-1
23FIC-108-1
71DCB2


TO CST
HPCI & RCIC TEST VLV TO CST 23MOV-24
FROM RCIC
TEST VLV TO CST 23MOV-21
DC
AUTO


PMP DISCH VLV 23MOV-20
DC
HPCI PMP
INJ VLV 23MOV-19
DC
CLOSE
AUTO
OPEN
TO B FDWTR

Proposed Answer: B

Explanation: The initial conditions show HPCI running with approximately 3400 gpm to the CSTs in manual flow control mode. In this alignment, 23MOV-21 and 23MOV-24 are open to allow HPCI discharge flow to go to the CSTs and 23MOV-19 is closed to block HPCI discharge flow from going to the Reactor. When Reactor water level lowers below 126.5 inches, HPCI receives an automatic initiation signal. This causes 23MOV-21 and 23MOV-24 to close and 23MOV-19 to open. This shifts all HPCI flow from the CSTs to the Reactor. However, the HPCI flow controller does NOT automatically shift into automatic on the low Reactor water level signal, therefore HPCI flow control remains in manual.

A. Incorrect – The HPCI test line isolates and the discharge to the Reactor automatically opens.
C. Incorrect – The HPCI test line isolates and the discharge to the Reactor automatically opens. HPCI continues to operate in manual flow control mode unless operator action is taken.
D. Incorrect – HPCI continues to operate in manual flow control mode unless operator action is taken.

Technical Reference(s): ST-4N, OP-15, SDLP-23

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23 1.05.c.4 and 1.05.c.6

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 #	239002 A3.04
	Importance Rating	3.6

SRVs

Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including: Acoustical monitor noise: Plant-Specific

Proposed Question: #18

The plant is operating at 100% power with the following indications for SRV A:

Which one of the following describes the status of the SRV A indications?

SRV A...

A. has normal indications for a valve that cycled open then closed.

B. indicates open by acoustic monitor but closed by solenoid position.

C. indicates open by solenoid position but closed by acoustic monitor.

D. indicates open by both acoustic monitor and solenoid position.

Proposed Answer: B

Explanation: The SRV A white light being illuminated indicates the valve is open as sensed by the acoustic monitor. However, the green light being illuminated and the red light being extinguished indicate that the SRV A is not open based on solenoid position.

A. Incorrect – The SRV A white light is normally extinguished when the valve is closed. The white light does not seal-in after valve opening.
C. Incorrect – The SRV A green light would be extinguished and red light would be illuminated if solenoid position indicated open.
D. Incorrect – The SRV A green light would be extinguished and red light would be illuminated if solenoid position indicated open.

Technical Reference(s): AOP-36

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J 1.05.a.5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 8/29/14 – Revised choice A to raise plausibility based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 A4.07
	Importance Rating	3.6

SLC

Ability to manually operate and/or monitor in the control room: Lights and alarms

Proposed Question: #19

A failure to scram has occurred with the following:

- The CRS has directed initiation of Standby Liquid Control (SLC) pump A.
- The SLC keylock switch is rotated to the START SYS A position.

The following indications are now present:

- Annunciator 09-3-3-30, SLC SQUIB VLV CONTINUITY LOSS, is NOT in alarm.
- Both SQUIB VLVS READY white lights are illuminated.

Which one of the following describes the status of these indications?

A. Both Annunciator 09-3-3-30 and the white lights responded properly.

B. Annunciator 09-3-3-30 responded properly, but the white lights did NOT respond as expected.

C. The white lights responded properly, but Annunciator 09-3-3-30 did NOT respond as expected.

D. NEITHER Annunciator 09-3-3-30 NOR the white lights responded properly.

Proposed Answer: A

Explanation: Annunciator 09-3-3-30, SLC SQUIB VLV CONTINUITY LOSS, is normally NOT in alarm, and is NOT expected to alarm when the SLC keylock switch is taken to either the START SYS A or START SYS B position. The SQUIB VLVS READY white lights are normally illuminated and are both expected to remain illuminated when the SLC keylock switch is taken to either the START SYS A or START SYS B position. This annunciator will alarm and these lights will extinguish only once the keylock switch is taken back to STOP.

B. Incorrect – Both squib valves fire when the SLC keylock switch is taken to either START SYS A or START SYS B. This results in a loss of continuity for the valve monitoring circuit, but the white lights are designed to only extinguish once the keylock switch is taken back to STOP.
C. Incorrect – Both squib valves fire when the SLC keylock switch is taken to either START SYS A or START SYS B. This results in a loss of continuity for the valve monitoring circuit, but the annunciator is designed to only alarm once the keylock switch is taken back to STOP.
D. Incorrect – Both squib valves fire when the SLC keylock switch is taken to either START SYS A or START SYS B. This results in a loss of continuity for the valve monitoring circuit, but the white lights are designed to extinguish and the annunciator is designed to alarm only once the keylock switch is taken back to STOP.

Technical Reference(s): ARP 09-3-3-30, OP-17, SDLP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-17 1.05.b.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 #	300000 A4.01
	Importance Rating	2.6

Instrument Air

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

Proposed Question: #20

The plant is operating at 100% power with the following:

- Air Compressor A is operating as the lead compressor.
- Air Compressor B is aligned as the 1st standby compressor.
- Then, Air Compressor A trips.
- 39PI-102, Instrument Air Header Pressure, indicates 109 psig and slowly lowering.

Which one of the following describes the response of Air Compressor B, per OP-39, Breathing, Instrument, and Service Air System?

Air Compressor B first receives an automatic start signal when ___(1)___ and unloads when Instrument Air pressure rises to ___(2)___.

	(1)	(2)
A.	Air Compressor A trips	110 psig
B.	Air Compressor A trips	120 psig
C.	Instrument Air pressure lowers to 107 psig	110 psig
D.	Instrument Air pressure lowers to 107 psig	120 psig

Proposed Answer: D

Explanation: The 1st standby Air Compressor is set to automatically start at 107 psig. The 2nd standby Air Compressor is set to automatically start at 104 psig. These setpoints are based on ensuring the Air Compressors are able to start and load by 100 psig. Once started, the standby Air Compressors unload at the same setpoint as the lead Air Compressor, 120 psig. 110 psig is the normal loading pressure for the lead Air Compressor

A. Incorrect – Air Compressor B will first receive a start signal at 107 psig and unload at 120 psig.
B. Incorrect – Air Compressor B will first receive a start signal at 107 psig
C. Incorrect – Air Compressor B will unload at 120 psig.

Technical Reference(s): OP-39

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39 1.05.c.3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:
TRH 8/29/14 – Deleted "expected" from and added "per OP-39" to question based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 2.4.21
	Importance Rating	4.0

AC Electrical Distribution

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: #21

A plant shutdown is in progress with the following:

- The Reactor Mode Switch is in START & HOT STBY.
- All IRMs are on range 9.
- Control rod insertion is in progress.

Then, 115 KV Line 3 de-energizes.

Which one of the following describes the applicable "AC Sources" Technical Specification and the status of its LCO?

	Applicable Technical Specification	Status of LCO
A.	3.8.1, AC Sources – Operating	Met
B.	3.8.1, AC Sources – Operating	NOT Met
C.	3.8.2, AC Sources – Shutdown	Met
D.	3.8.2, AC Sources – Shutdown	NOT Met

Proposed Answer: B

Explanation: Technical Specification 3.8.1 applies in Modes 1, 2, and 3, while Technical Specification 3.8.2 applies in Modes 4 and 5. A shutdown is in progress, however the plant is still in Mode 2 based on the Reactor Mode Switch being in START & HOT STBY. Therefore, Technical Specification 3.8.1, AC Sources – Operating, applies. The LCO requires two 115 KV lines to be operable. With Line 3 de-energized, only one 115 KV line (Line 4) is operable, therefore the LCO is NOT met.

A. Incorrect – The LCO is NOT met because it requires two 115 KV lines. TS 3.8.2 only requires one 115 KV line.
C. Incorrect – TS 3.8.1 applies, not TS 3.8.2. The applicable LCO is NOT met because it requires two 115 KV lines. TS 3.8.2 only requires one 115 KV line.
D. Incorrect – TS 3.8.1 applies, not TS 3.8.2.

Technical Reference(s): Technical Specifications 3.8.1 and 3.8.2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71D 1.16

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 9/8/14 – Discussed with NRC Chief Examiner and confirmed this is a fair question for an RO candidate.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 2.4.2
	Importance Rating	4.5

RCIC

Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Proposed Question: #22

The plant is operating at 50% power with the following:

- A Reactor scram occurs.
- RCIC automatically starts.
- RCIC develops a steam leak.
- RCIC automatically isolates due to RCIC room temperature.
- All RCIC actuations were based on valid signals.

Given the following EOP entry conditions:

(1) EOP-2, RPV Control, low Reactor water level entry condition
(2) EOP-5, Secondary Containment Control, high area temperature entry condition

Note: The EOP-5 Reactor Building Area Temperatures table is provided on the next page.

Based on the response of RCIC 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)

REACTOR BUILDING AREA TEMPERATURES

AREA	INSTRUMENT	MAXIMUM NORMAL	MAXIMUM SAFE
Reactor Building 369 ft elevation 66RTD-106 66RTD-108	66TI-106, Panel 09-75 66TI-108, Panel 09-75	104°F	112°F
Outside 'A' LPCI Battery Enclosure 66RTD-115	EPIC Only	104°F	113°F
Below Refuel Floor Exhaust 66RTD-105	66TI-105, Panel 09-75	104°F	113°F
Outside 'B' LPCI Battery Enclosure 66RTD-116	EPIC Only	104°F	113°F
SLC Pump Area 66RTD-114	EPIC Only ❶	104°F	133°F
Fuel Pool Cooling Pump Room 66RTD-113	EPIC Only	104°F	133°F
Reactor Building 300 ft elevation northeast 66RTD-112	EPIC Only ❶	104°F	158°F
RWCU Heat Exchanger Room 12TE-117E 12TE-117F	Panel 09-21 Panel 09-21	115°F	203°F
'B' RWCU Pump Room 12TE-117C 12TE-117D	Panel 09-21 Panel 09-21	135°F	225°F
'A' RWCU Pump Room 12TE-117A 12TE-117B	Panel 09-21 Panel 09-21	125°F	225°F
Reactor Building 300 ft elevation southwest 66RTD-111	EPIC Only ❶	104°F	173°F
'B' RHR Heat Exchanger Room 23RTD-02A 23RTD-02B	23-203A, Panel 09-95 23-203B, Panel 09-96	130°F	242°F
Reactor Building 272 ft elevation southeast 23RTD-02C 23RTD-02D	23-204A, Panel 09-95 23-204B, Panel 09-96	104°F	153°F
HPCI Drywell Entrance 13RTD-102C 13RTD-102D	13-202C, Panel 09-95 13-202D, Panel 09-96	120°F	251°F
RCIC Drywell Entrance 13RTD-102A 13RTD-107B	13-202A, Panel 09-95 13-207B, Panel 09-96	120°F	218°F
Reactor Building 272 ft elevation soutwest 23RTD-01C 23RTD-01D	23-202A, Panel 09-95 23-202B, Panel 09-96	104°F	196°F
'A' RHR Heat Exchanger Room 23RTD-01A 23RTD-01B	23-201A, Panel 09-95 23-201B, Panel 09-96	130°F	242°F
Torus Room - South HPCI Steamline 13RTD-107C 13RTD-107D	13-207C, Panel 09-95 13-207D, Panel 09-96	120°F	280°F
Torus Room - Southwest RCIC Steamline 13RTD-107A 13RTD-102B	13-207A, Panel 09-95 13-202B, Panel 09-96	120°F	280°F
East Crescent 66RTD-109B	66TI-109B, Panel 09-75	104°F	137°F
HPCI Room 23RTD-94A 23RTD-94B 23RTD-117A 23RTD-117B	23-294A, Panel 09-95 23-294B, Panel 09-96 23-217A, Panel 09-95 23-217B, Panel 09-96	104°F	137°F
RCIC Room 13RTD-89A 13RTD-89B	13-289A, Panel 09-95 13-289B, Panel 09-96	104°F	137°F
West Crescent 13RTD-76A 13RTD-76B	13-276A, Panel 09-95 13-276B, Panel 09-96	104°F	137°F

Proposed Answer: C

Explanation: RCIC automatically starts on Reactor water level less than 126.5 inches, only. The EOP-2 low Reactor water level entry condition is 177 inches. Since RCIC automatically started, the EOP-2 low Reactor water level entry condition must have been met. RCIC automatically isolates on a high RCIC room temperature of 133°F on devices 13RTD-89A and 13RTD-89B. The EOP-5 high area temperature entry condition includes these devices above 104°F. Since RCIC automatically isolated on RCIC room temperature, the EOP-5 high area temperature entry condition must have been met.

A. Incorrect – Since RCIC automatically isolated on RCIC room temperature (133°F), the EOP-5 high area temperature entry condition (104°F) must have been met also.
B. Incorrect – Since RCIC automatically started (126.5"), the EOP-2 low Reactor water level entry condition (177") must have been met also.
D. Incorrect – Since RCIC automatically started (126.5"), the EOP-2 low Reactor water level entry condition (177") must have been met. Since RCIC automatically isolated on RCIC room temperature (133°F), the EOP-5 high area temperature entry condition (104°F) must have been met.

Technical Reference(s): OP-19, ARP 09-3-3-2, ARP 09-3-3-12, EOP-2, EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13 1.07.a and 1.07.b

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 #	263000 K1.01
	Importance Rating	3.3

DC Electrical Distribution

Knowledge of the physical connections and/or cause-effect relationships between D.C. ELECTRICAL DISTRIBUTION and the following: A.C. electrical distribution

Proposed Question: #23

The plant is operating at 100% power with the following:

- 71DC-A3 Relay Room Distribution Cabinet Breaker #20 (DC Power to 10100 Breakers) trips.

Which one of the following describes the effect of this breaker trip on the 10100 bus?

The 10100 bus...

A. de-energizes due to trip of breaker 10102 (NSS to Bus 10100).

B. de-energizes due to trip of breaker 10112 (RSS to Bus 10100).

C. remains energized through breaker 10102 (NSS to Bus 10100).

D. remains energized through breaker 10112 (RSS to Bus 10100).

Proposed Answer: C

Explanation: At 100% power, the 10100 bus is powered from normal station service (NSS) through breaker 10102. A loss of DC control power to breaker 10102 will cause a loss of automatic and remote trip capability, but the breaker will remain closed. The breaker can only be opened manually / locally. Therefore, the 10100 bus remains energized through the 10102 breaker.

A. Incorrect – The 10100 bus remains energized.
B. Incorrect – The 10100 bus remains energized.
D. Incorrect – The 10100 bus remains energized through breaker 10112, not 10102, based on plant conditions.

Technical Reference(s): OP-46A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71E 1.10.c

Question Source: Bank – March 2012 NRC #46

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 K4.08
	Importance Rating	4.2

SLC

Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: System initiation upon operation of SBLC control switch

Proposed Question: #24

A failure to scram and loss of all offsite power (LOOP) has occurred with the following:

- Emergency Diesel Generators B and D have failed to start.
- The Standby Liquid Control (SLC) keylock switch is taken to START SYS A, and then taken to START SYS B.

Which one of the following describes the SLC system response?

A. Neither SLC pump is running.

B. Only SLC pump A is running.

C. Only SLC pump B is running.

D. Both SLC pumps are running.

Proposed Answer: A

Explanation: When the SLC keylock switch is taken to START SLC A, SLC pump A starts and SLC pump B remains off. When the SLC keylock switch is taken to START SLC B, SLC pump A stops. Additionally, with a loss of all offsite power and failure of EDGs B and D to start, there is no power to SLC pump B (10600 bus, 11600 bus, and MCC 162 are de-energized). Therefore SLC pump B does NOT run.

B. Incorrect – SLC pump A started when the keylock switch was taken to START SLC A, however when the keylock switch is taken to START SLC B, SLC pump A stops.
C. Incorrect – Due to loss of all offsite power and failure of EDGs B and D to run, SLC pump B has no power to run.
D. Incorrect – SLC pump A started when the keylock switch was taken to START SLC A, however when the keylock switch is taken to START SLC B, SLC pump A stops. Due to loss of all offsite power and failure of EDGs B and D to run, SLC pump B has no power to run.

Technical Reference(s): OP-17

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-11 1.05.b.2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:
TRH 8/29/14 – Revised wording of question based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 A4.09
	Importance Rating	3.8

HPCI

Ability to manually operate and/or monitor in the control room: Suppression pool level: BWR-2,3,4

Proposed Question: #25

The plant has experienced a seismic event with the following:

- The Reactor has scrammed.
- HPCI is injecting to the Reactor.
- Torus water level is 11.5 feet and lowering.
- Attempts to align Torus makeup and isolate the leakage have failed.

Which one of the following describes the threshold Torus water level that requires HPCI to be shutdown and the method to shutdown HPCI, in accordance with EOP-4, Primary Containment Control?

EOP-4 requires initiating a manual HPCI ___(1)___ if Torus water level cannot be maintained above the threshold of ___(2)___.

	(1)	(2)
A.	turbine trip	9.58 feet
B.	steam line isolation	9.58 feet
C.	turbine trip	10.75 feet
D.	steam line isolation	10.75 feet

Proposed Answer: C

Explanation: EOP-4 contains the following steps:

IF	THEN
Torus water level cannot be maintained above 10.75 ft	➢ Irrespective of whether the core will be adequately cooled, ensure a manual HPCI turbine trip has been initiated. ➢ Enter EOP-2, "RPV Control," and execute it concurrently with this procedure.
Torus water level cannot be maintained above 9.58 ft	**EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.**

The threshold Torus water level for HPCI shutdown is 10.75 feet and the method for HPCI shutdown is initiating a manual HPCI turbine trip.

A. Incorrect – The threshold Torus water level for HPCI shutdown is 10.75 feet, NOT 9.58 feet. 9.58 feet is the threshold Torus water level for emergency RPV depressurization.
B. Incorrect – The threshold Torus water level for HPCI shutdown is 10.75 feet, NOT 9.58 feet. 9.58 feet is the threshold Torus water level for emergency RPV depressurization. The method for HPCI shutdown is initiating a manual HPCI turbine trip, NOT a manual steam line isolation.
D. Incorrect – The method for HPCI shutdown is initiating a manual HPCI turbine trip, NOT a manual steam line isolation.

Technical Reference(s): EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11E 4.05

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 #	2
	Group #	1
	K/A #	259002 K6.01
	Importance Rating	3.2

Reactor Water Level Control

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM: Plant air systems

Proposed Question: #26

A plant startup is in progress with the following:

- Reactor water level is 200 inches and stable.
- Reactor Feedwater pump B is in service with its flow controller in MANUAL.
- 34FCV-137, Feedwater System Low Flow Control Valve, is partially open with its controller in MANUAL.
- 34FCV-135B, RFP B Min Flow Valve, is fully open.

Then, Instrument Air is lost to just 34FCV-137 and 34FCV-135B.

Which one of the following describes the effect of these malfunctions on Reactor water level?

Reactor water level...

A. rises due to 34FCV-137 failing open and 34FCV-135B failing in its initial position.

B. rises due to 34FCV-135B failing closed and 34FCV-137 failed in its initial position.

C. lowers due to 34FCV-137 failing closed and 34FCV-135B failing in its initial position.

D. remains approximately constant with both of these valves failed in their initial positions.

Proposed Answer: D

Explanation: Upon loss of instrument air, 34FCV-137 fails as-is and 34FCV-135B fails open (which is its initial position in this case). With neither valve moving and Reactor water level initially stable, Reactor water level will remain approximately constant.

A. Incorrect – 34FCV-137 fails as-is on loss of instrument air, and therefore does NOT re-position.
B. Incorrect – 34FCV-135B fails open, NOT closed.
C. Incorrect – 34FCV-137 fails as-is on loss of instrument air, and therefore does NOT re-position.

Technical Reference(s): ARP 09-5-1-24, AOP-12, FM-34A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-33 1.10.a

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 #	233000 K1.14
	Importance Rating	2.5

Fuel Pool Cooling/Cleanup

Knowledge of the physical connections and/or cause-effect relationships between FUEL POOL COOLING AND CLEAN-UP and the following: Reactor building ventilation

Proposed Question: #27

The plant is operating at 100% power with the following:

- An un-isolable leak has developed in the Spent Fuel Pool liner.
- Spent Fuel Pool water level is lowering.
- The Refuel Floor Exhaust radiation monitors indicate as follows:
  - 17RM-456A (Refuel Floor Exhaust Radiation Monitor) – 8×10^3 cpm
  - 17RM-456B (Refuel Floor Exhaust Radiation Monitor) – 2×10^4 cpm
  - 17RM-452A (Below Refuel Floor Exhaust Radiation Monitor) – 6×10^3 cpm
  - 17RM-452B (Below Refuel Floor Exhaust Radiation Monitor) – 3×10^3 cpm

Which one of the following describes the response of the Reactor Building Ventilation system fans to these conditions?

A. All running fans trip.

B. All running fans remain in service.

C. The Above Refuel Floor Exhaust fan trips. The Below Refuel Floor Exhaust fan and Supply Fans remain in service.

D. The Above Refuel Floor Exhaust fan and Below Refuel Floor Exhaust fan trip. The Supply Fans remain in service.

Proposed Answer: C

Explanation: 17RM-456B indicates above the high-high setpoint of 1×10^4 cpm. This signal alone causes a full Reactor Building Ventilation isolation. A Reactor Building Ventilation isolation results in trip of the running Above Refuel Floor Exhaust fan, however the Below Refuel Floor Exhaust fan and the Supply fans remain in service to provide a recirculation flow path through the Reactor Building.

A. Incorrect – Although a full Reactor Building Ventilation isolation occurs, the Below Refuel Floor Exhaust fan and the Supply fans remain in service.
B. Incorrect – A full Reactor Building Ventilation isolation occurs on a single exhaust radiation monitor exceeding 1×10^4 cpm. This trips the running Above Refuel Floor Exhaust fan.
D. Incorrect – Although a full Reactor Building Ventilation isolation occurs, the Below Refuel Floor Exhaust fan remains in service.

Technical Reference(s): ARP 09-3-2-40, OP-51A

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-66A 1.05.c.1 and 1.05.c.4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:
TRH 8/29/14 – Deleted extra words at end of choice B based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	259001 K1.20
	Importance Rating	3.1

Reactor Feedwater

Knowledge of the physical connections and/or cause-effect relationships between REACTOR FEEDWATER SYSTEM and the following: Main steam system: TDRFPs-Only

Proposed Question: #28

Which one of the following describes:

(1) the location of the high pressure (HP) steam supply tap off for the Reactor Feed Pump Turbines (RFPTs) relative to the MSIVs, and

(2) if the Total Steam Flow indication on panel 09-5 includes the HP steam supply flow to the RFTPs?

	Location of Steam Supply Tap Off For RFPTs	Total Steam Flow Indication on Panel 09-5
A.	Upstream of MSIVs	Includes RFPT HP steam flow
B.	Upstream of MSIVs	Does NOT include RFPT HP steam flow
C.	Downstream of MSIVs	Includes RFPT HP steam flow
D.	Downstream of MSIVs	Does NOT include RFPT HP steam flow

Proposed Answer: C

Explanation: The RFPT HP steam supply tap off is located downstream of the MSIVs, but upstream of the Main Turbine Stop and Control valves. The Total Main Steam flow indication on Panel 09-5 is based on flow through the Main Steam Line flow restrictors, which are located upstream of both the MSIVs and the RFPT HP steam supply tap off. Therefore, the Total Main Steam flow indication does include steam flow to the RFPT HP steam supply line.

A. Incorrect – The RFPT HP steam supply tap off is located downstream of the MSIVs, NOT upstream.
B. Incorrect – The RFPT HP steam supply tap off is located downstream of the MSIVs, NOT upstream. The Total Main Steam flow indication does include steam flow to the RFPT HP steam supply line.
D. Incorrect – The Total Main Steam flow indication does include steam flow to the RFPT HP steam supply line.

Technical Reference(s): FM-29A, FM-29B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-33 1.09.c

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 #	2
	Group #	2
	K/A #	239001 K3.02
	Importance Rating	3.1

Main and Reheat Steam

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

Proposed Question: #29

A plant startup is in progress with the following:

- Reactor power is mid-scale on IRM range 9.
- Reactor pressure is 800 psig and slowly rising.
- All Main Steam Isolation Valves (MSIVs) are open.

Then, Main Steam Line D ruptures in the steam tunnel.

Which one of the following describes the response of the Reactor Protection System (RPS) and the status of the Main Condenser as a heat sink for the Reactor?

	Does RPS Scram The Reactor On An MSIV Closure Signal?	**Does The Main Condenser Remain In Service As The Reactor's Heat Sink?**
A.	No	No
B.	No	Yes
C.	Yes	No
D.	Yes	Yes

Proposed Answer: A

Explanation: Rupture of a single Main Steam Line downstream of the MSIVs will cause isolation of all MSIVs, NOT just the MSIVs in the effected Main Steam Line. Since all MSIVs close, the Main Condenser is no longer in service as the Reactor's heat sink. The Torus becomes the Reactor's heat sink through the SRVs, and possibly HPCI/RCIC. With Reactor power mid-scale on IRM Range 9, the Reactor Mode Switch must still be in STARTUP/HOT STANDBY. Since the Reactor Mode Switch is NOT in RUN, the RPS MSIV Closure scram is bypassed. Therefore, even with MSIVs closed, RPS does NOT cause a Reactor scram based on an MSIV Closure signal.

B. Incorrect – Even though the malfunction is on a single Main Steam Line, all MSIVs close due to the resulting high temperature and/or flow signal(s). This removes the Main Condenser from service as the Reactor's heat sink.
C. Incorrect – Since the Reactor Mode Switch is NOT in RUN, RPS does NOT cause a Reactor scram on an MSIV Closure signal.
D. Incorrect – Since the Reactor Mode Switch is NOT in RUN, RPS does NOT cause a Reactor scram on an MSIV Closure signal. Even though the malfunction is on a single Main Steam Line, all MSIVs close due to the resulting high temperature and/or flow signal(s). This removes the Main Condenser from service as the Reactor's heat sink.

Technical Reference(s): OP-1, ARP-09-5-1-12

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-29 1.05.b.1, 1.05.b.3, and 1.05.c.1

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 #	202002 K4.03
	Importance Rating	3.0

Recirculation Flow Control

Knowledge of RECIRCULATION FLOW CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Signal failure detection: Plant-Specific

Proposed Question: #30

The plant is operating at 100% power with the following:

- The signal to the Recirculation MG Set A scoop tube positioner fails to 0 mA.

Which one of the following describes the resulting operation of the Recirculation MG Set A scoop tube?

The scoop tube automatically ...

A. locks up due to loop flow mismatch.

B. locks up due to signal failure detection.

C. re-positions to the minimum Recirculation flow position.

D. re-positions to the maximum Recirculation flow position.

Proposed Answer: B

Explanation: A loss of signal (< 1 mA) to the scoop tube positioner causes a scoop tube lock up.

A. Incorrect – The scoop tube locks up based on detection of the signal failure, NOT based on loop flow mismatch, which would result if the scoop tube was allowed to re-position.
C. Incorrect – A loss of signal (< 1 mA) to the scoop tube positioner causes a scoop tube lock up. Therefore, the scoop tube is prevented from re-positioning to the minimum flow position, as it would otherwise.
D. Incorrect – A loss of signal (< 1 mA) to the scoop tube positioner causes a scoop tube lock up. Therefore, the scoop tube is prevented from re-positioning, as it would otherwise.

Technical Reference(s): OP-27, ARP 09-4-3-11

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02I 1.05.c.3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:
TRH 8/29/14 – Moved "automatically" from beginning of choices A and B to common part of answer choices based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	241000 K5.03
	Importance Rating	3.5

Reactor/Turbine Pressure Regulating System

Knowledge of the operational Implications of the following concepts as they apply to REACTOR/TURBINE PRESSURE REGULATING SYSTEM: Reactor power vs. reactor pressure

Proposed Question: #31

The plant is operating at 45% power during a startup with the following:

- The second Feedwater pump has just been started
- Reactor pressure is approximately 990 psig and stable.

Then, Reactor power is raised to 75% using Recirculation flow and control rod withdrawals.

Which one of the following describes the expected response of Reactor pressure and the necessity of adjusting the EHC pressure setpoint during this power ascension, in accordance with OP-65, Startup and Shutdown Procedure?

Reactor pressure...

A. rises. The EHC pressure setpoint is required to be adjusted periodically.

B. rises. The EHC pressure setpoint is NOT required to be adjusted periodically.

C. remains near 990 psig due to periodic adjustment of the EHC pressure setpoint.

D. remains at 990 psig. The EHC pressure setpoint is NOT required to be adjusted periodically.

Proposed Answer: B

Explanation: The Turbine pressure regulator is designed to maintain a relatively constant Turbine inlet pressure as Reactor power changes. To accomplish this goal while steam flow also rises, Reactor pressure rises as Reactor power rises. The EHC pressure setpoint is set to approximately 970 psig even before the Main Turbine is rolled and remains there until a final adjustment at 100% power. No periodic changes in EHC pressure setpoint are required during power ascension based on the operating characteristics of the regulator.

A. Incorrect – No changes in EHC pressure setpoint are required.
C. Incorrect – Reactor pressure rises as the EHC system controls to maintain relatively constant Turbine inlet pressure. No changes in EHC pressure setpoint are required.
D. Incorrect – Reactor pressure rises as the EHC system controls to maintain relatively constant Turbine inlet pressure.

Technical Reference(s): SDLP-94C, OP-65

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94C 1.05.a.4

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 #	2
	Group #	2
	K/A #	286000 K6.01
	Importance Rating	3.1

Fire Protection

Knowledge of the effect that a loss or malfunction of the following will have on the FIRE PROTECTION SYSTEM: A.C. electrical distribution: Plant-Specific

Proposed Question: #32

A plant startup is in progress with the following:

- The Reactor has just reached the point of adding heat (POAH).
- 115KV Lines 3 and 4 de-energize.

Which one of the following describes the effect(s) on the Fire Protection system?

The Makeup Fire Pump (P-3)...

A. maintains header pressure. The Motor Driven Fire Pump (P-2) loses power.

B. loses power. The Motor Driven Fire Pump (P-2) auto-starts on low header pressure.

C. and the Motor Driven Fire Pump (P-2) lose power. The Diesel Driven Fire Pump (P-1) auto-starts on low header pressure.

D. and the Motor Driven Fire Pump (P-2) lose power. The Diesel Driven Fire Pump (P-1) loses control power and must be manually started.

Proposed Answer: C

Explanation: At the current point in the plant startup, 115KV Lines 3 and 4 are providing power to Buses 10100, 10200, 10300, and 10400. When Lines 3 and 4 de-energize, all of these buses de-energize. The loss of Bus 10400 causes the loss of L34 and L44. The loss of L34 causes the loss of MCC-343. MCC-343 is the power supply to the Makeup Fire Pump (P-3), therefore this pump loses power. The Motor Driven Fire Pump (P-2) would normally auto-start on a low header pressure of 109 psig, however its power supply, L44, is de-energized. The Diesel Driven Fire Pump (P-1) auto-starts on a low header pressure of 101 psig. This pump uses 24 VDC to start, and is therefore NOT prohibited from starting with the loss of Lines 3 and 4.

A. Incorrect – The Makeup Fire Pump (P-3) loses power from MCC-343.
B. Incorrect – The Motor Driven Fire Pump (P-2) loses power from L44.
D. Incorrect – Diesel Driven Fire Pump control power is from 24 VDC and remains available after the loss of Lines 3 and 4.

Technical Reference(s): OP-33

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-76 1.10.a

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 #	272000 A1.01
	Importance Rating	3.2

Radiation Monitoring

Ability to predict and/or monitor changes in parameters associated with operating the RADIATION MONITORING SYSTEM controls including: Lights, alarms, and indications associated with normal operations

Proposed Question: #33

The plant is operating at 100% power with the following:

- Area Radiation Monitor (ARM) #25, RX BLDG EL 272 EAST HCU AREA, goes into alarm high.
- Annunciator 09-3-1-40, RX BLDG ARM RAD HI, is received and acknowledged.

Which one of the following describes how Annunciator 09-3-1-40 will respond to the following two **independent** conditions?

If a second ARM input exceeds its high alarm setpoint, then Annunciator 09-3-1-40 __(1)__.

If ARM #25 returns to normal, then Annunciator 09-3-1-40 __(2)__ be reset before depressing ARM RESET pushbutton(s).

	(1)	(2)
A.	re-flashes	can
B.	re-flashes	can NOT
C.	does NOT re-flash	can
D.	does NOT re-flash	can NOT

Proposed Answer: D

Explanation: Annunciator 09-3-1-40 will actuate upon receipt of a high alarm condition from any of 16 Area Radiation Monitors (ARMs). However, once the annunciator is in, it will not re-flash for any subsequent high alarm conditions from any of the other ARMs. Additionally, once the high radiation condition clears, the high alarm trip is sealed in until the RESET pushbutton is depressed on the corresponding ARM trip unit on control room panel 09-11. Until this RESET pushbutton is depressed, both the ARM trip unit amber HIGH light and Annunciator 09-3-1-40 will be sealed in.

A. Incorrect – The annunciator does not have re-flash capability. The annunciator cannot be reset until the ARM RESET pushbutton is depressed.
B. Incorrect – The annunciator does not have re-flash capability.
C. Incorrect – The annunciator cannot be reset until the ARM RESET pushbutton is depressed.

Technical Reference(s): ARP 09-3-1-40, 1.78-98, UFSAR Fig 7.13-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-17 1.14.d.5

Question Source: Bank – September 2012 NRC #73

Question History: September 2012 NRC #73

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	2
K/A #	234000 A2.01
Importance Rating	3.3

Fuel Handling Equipment

Ability to (a) predict the impacts of the following on the FUEL HANDLING EQUIPMENT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Interlock failure

Proposed Question: #34

Refueling is in progress and bundle 37-42 is scheduled to be moved from the core to the Spent Fuel Pool. The following sequence occurs:

- Upon receiving "Grapple Down" and "Slack Cable" indications, the grapple switch is actuated.
- The "Grapple Closed" light is illuminated.
- Upon lifting bundle 37-42, the hoist load cell registers 100 pounds.

Which one of the following describes:

(1) the effect, if any, on refueling interlocks, and

(2) the action the Refueling Bridge operator must take, in accordance with OSP-66.001, Management of Refueling Activities?

A. (1) All refueling interlocks are operable.
(2) Continue fuel movement.

B. (1) All refueling interlocks are operable.
(2) Stop fuel movement.

C. (1) One or more refueling interlocks are inoperable.
(2) Continue fuel movement with an independent verifier.

D. (1) One or more refueling interlocks are inoperable.
(2) Stop fuel movement.

Proposed Answer: D

Explanation: Improper operation of the hoist load cell affects the control rod block associated with fuel movement and it affects control of Refueling Bridge movement from the SFP to the core. The load cell must register >540 pounds with a fuel bundle loaded to properly enforce these interlocks. Since the load cell only registers 100 pounds with a fuel bundle loaded, these interlocks are inoperable. OSP-66.001 requires "If any abnormal condition, fuel bundle damage, or evidence of interference or binding develops, then refueling operations shall be immediately stopped and the condition resolved before continuing".

A. Incorrect – This failure would still allow further lifting of the fuel bundle and transport to the SFP. However, this failure still affects other refueling interlocks (control rod block and bridge motion).
B. Incorrect – This failure would still allow further lifting of the fuel bundle and transport to the SFP. However, this failure still affects other refueling interlocks (control rod block).
C. Incorrect – OSP-66.001 requires stopping refueling operations and resolution of the condition before continuing. No provision is made allowing independent verification to replace inoperable refueling interlocks.

Technical Reference(s): OP-66A, OSP-66.001

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-08B 1.05.b.4.a

Question Source: Modified Bank – 2008 NRC #93

Question History: 2008 NRC #93

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 #	288000 A3.01
	Importance Rating	3.8

Plant Ventilation Systems

Ability to monitor automatic operations of the PLANT VENTILATION SYSTEMS including: Isolation/initiation signals

Proposed Question: #35

The plant is operating at 100% power with the following:

- 67FN-6A, Turbine Building Ventilation Exhaust Fan A, is in service.
- 67FN-6B, Turbine Building Ventilation Exhaust Fan B, is in standby.

Then, 67FN-6A trips on motor overcurrent.

Which one of the following describes the response of 67AOD-138A, 67FN-6A discharge damper, and the response of 67FN-6B?

	67AOD-138A	67FN-6B
A.	Closes	Automatically starts
B.	Closes	Remains in standby
C.	Remains open	Automatically starts
D.	Remains open	Remains in standby

Proposed Answer: A

Explanation: The Turbine Building exhaust fan discharge dampers, 67AOD-138A(B), are interlocked with their respective fan to automatically close when the fan is NOT running. When the running Turbine Building exhaust fan trips, the standby exhaust fan automatically starts.

B. Incorrect – The standby exhaust fan automatically starts.
C. Incorrect – The discharge damper automatically closes.
D. Incorrect – The discharge damper automatically closes. The standby exhaust fan automatically starts.

Technical Reference(s): OP-52, ARP HV-1-1-03, SDLP-67

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-67 1.05.b.1

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 #	2
	Group #	2
	K/A #	271000 A4.09
	Importance Rating	3.3

Offgas

Ability to manually operate and/or monitor in the control room: Offgas system controls/components

Proposed Question: #36

The plant is operating at 100% power when annunciator 09-3-2-10, OFF-GAS TIMER INITIATED, alarms.

Which one of the following describes the cause of this alarm and the automatic action that occurs based on receipt of this alarm?

This alarm is caused by...

A. high radiation in Offgas. If the timer times out, 38AOV-113A(B), CNDSR ISOL VLVs, close.

B. high radiation in Offgas. If the timer times out, 01-107AOV-100A, OFF GAS DISCH TO STACK, closes.

C. high temperature or pressure in Offgas. If the timer times out, 38AOV-113A(B), CNDSR ISOL VLVs, close.

D. high temperature or pressure in Offgas. If the timer times out, 01-107AOV-100A, OFF GAS DISCH TO STACK, closes.

Proposed Answer: B

Explanation: Annunciator 09-3-2-10 is caused by high radiation as sensed by the Offgas radiation monitors. If the associated timer times out, 01-107AOV-100A, OFF GAS DISCH TO STACK, closes.

A. Incorrect – 01-107AOV-100A, OFF GAS DISCH TO STACK, closes. 38AOV-113A(B), CNDSR ISOL VLVs, would close on high temperature or pressure.
C. Incorrect – High temperature or pressure cause annunciators 09-6-1-07 and 09-6-1-15, respectively, NOT 09-3-2-10. 01-107AOV-100A, OFF GAS DISCH TO STACK, closes. 38AOV-113A(B), CNDSR ISOL VLVs, would close on high temperature or pressure.
D. Incorrect – High temperature or pressure cause annunciators 09-6-1-07 and 09-6-1-15, respectively, NOT 09-3-2-10.

Technical Reference(s): ARPs 09-3-2-10, 09-6-1-07, 09-6-1-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-01A 1.05.c.1

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 #	2
	Group #	2
	K/A #	290003 2.4.47
	Importance Rating	4.2

Control Room HVAC

Emergency Procedures / Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: #37

The plant is operating at 100% power with the following:

- Control Room Ventilation performance is degraded.
- Temperature in the at-the-controls area of the Control Room is rising as follows:

Time (hh:mm)	Temperature (°F)
00:00	68
00:15	74

Note: Assume the temperature trend remains constant.

Which one of the following describes the approximate earliest time at which Control Room temperature will cause all available cooling to auto-initiate, per OP-55B, Control Room Ventilation and Cooling?

A. 00:45

B. 01:00

C. 01:15

D. 01:30

Proposed Answer: C

Explanation: OP-55B states that all available Control Room cooling will auto-initiate when Control Room temperature reaches 98°F. Given the current trend, Control Room temperature will reach 98°F at 01:15.

A. Incorrect – At 00:45, temperature will be approximately 86°F, NOT 98°F. This is close to the 85°F lake temperature used in an associated Control Room temperature calculation.
B. Incorrect – At 01:00, temperature will be approximately 92°F, NOT 98°F. This is close to the 90°F temperature limit in Technical Specification 3.7.4 for entry into a shutdown Required Action.
D. Incorrect – At 01:30, temperature will be approximately 104°F, NOT 98°F. This is the temperature referenced in OP-55B Precaution C.2.3 to ensure proper operation of equipment in the Control Room.

Technical Reference(s): OP-55B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-70 1.13.b

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 9/2/14 – Revised question to focus on more operationally oriented Control Room temperature, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	215001 A2.01
	Importance Rating	2.7

Traversing In-core Probe

Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor water level: Mark-I&II(Not-BWR1)

Proposed Question: #38

The plant is operating at 100% power with the following:

- Traversing In-core Probe (TIP) calibration of LPRMs is in progress.
- The TIP detector is stationary at the top of the core.
- Then, a Reactor scram occurs.
- Reactor water level lowers to 130 inches.
- The TIP detector retracts to the shield chamber.
- The associated TIP ball and shear valves remain open.
- Reactor water level is now 160 inches and slowly rising.

Which one of the following describes the response of the TIP system and the required operator action?

The TIP system...

A. responded properly. Restore the TIP system to the standby lineup per RAP-7.3.14, Traversing In-Core Probe System.

B. did NOT respond properly. Close the ball valve only per AOP-15, Isolation Verification and Recovery.

C. did NOT respond properly. Close the shear valve only per AOP-15, Isolation Verification and Recovery.

D. did NOT respond properly. Close both the ball valve and the shear valve per AOP-15, Isolation Verification and Recovery.

Proposed Answer: B

Explanation: With Reactor water level <177 inches, the TIP should retract and then the ball valve should close. AOP-15 directs verifying the ball valve closed. The shear valve should remain open, and would only be fired if the ball valve subsequently fails to close.

A. Incorrect – The TIP system did NOT respond properly because the ball valve should be closed.
C. Incorrect – The ball valve, NOT the shear valve, should be closed.
D. Incorrect – The shear valve should NOT be closed.

Technical Reference(s): AOP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07F 1.05.c.1

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 #	1
	Group #	1
	K/A #	295001 AK1.01
	Importance Rating	3.5

Partial or Complete Loss of Forced Core Flow Circulation

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Natural circulation

Proposed Question: #39

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP/HOT STANDBY.
- Reactor pressure is 50 psig and slowly rising.

Then, both Recirculation pumps trip.

Which one of the following describes the required control of the Reactor in accordance with AOP-8, Unexpected Change in Core Flow?

A. A manual Reactor scram is required.

B. A manual Reactor scram is NOT required, but a plant shutdown is required.

C. Reactor operation can continue as long as NO indications of thermal-hydraulic instability occur.

D. Reactor operation can continue as long as core conditions satisfy the natural circulation line of the Power-Flow map.

Proposed Answer: B

Explanation: If both Recirculation pumps trip and the Reactor Mode Switch is in STARTUP/HOT STANDBY, AOP-8 requires commencing a normal plant shutdown per OP-65 and achieving Mode 3 within 12 hours. The trip of both Recirculation pumps places the plant in natural circulation. AOP-8 does not allowed continued operation in natural circulation.

A. Incorrect – AOP-8 only requires a manual Reactor scram if the Reactor Mode Switch is in RUN, THI is detected, or Power-Flow map requires a scram.
C. Incorrect – The Reactor must be shutdown within 12 hours. Evidence of THI is a reason requiring a manual Reactor scram.
D. Incorrect – The Reactor must be shutdown within 12 hours. Improper position on Power-Flow map is a reason requiring a manual Reactor scram.

Technical Reference(s): AOP-8

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 8/29/14 – Added to question explanation to strengthen K/A match based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 AK1.03
	Importance Rating	3.7

Refueling Accidents

Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS: Inadvertent criticality

Proposed Question: #40

The plant is shutdown with the following:

- Core shuffle is in progress.
- RHR is operating in the Shutdown Cooling mode.

Which one of the following describes the lower limit on Reactor coolant temperature, in accordance with OP-13D, RHR – Shutdown Cooling, and an adverse consequence of exceeding this limit?

	Lower Limit on Reactor Coolant Temperature	Adverse Consequence of Exceeding This Limit
A.	60°F	Exceed Shutdown Margin (SDM) calculation assumption and risk inadvertent criticality
B.	60°F	Exceed Pressure-Temperature Limits Report (PTLR) limit and risk excessive vessel stresses
C.	68°F	Exceed Shutdown Margin (SDM) calculation assumption and risk inadvertent criticality
D.	68°F	Exceed Pressure-Temperature Limits Report (PTLR) limit and risk excessive vessel stresses

Proposed Answer: C

Explanation: OP-13D requires Reactor coolant temperature to be maintained greater than 68°F. Going below 68°F would exceed the minimum temperature assumption used in the Shutdown Margin (SDM) calculation.

A. Incorrect – 60°F is the lower limitation for RPV head installation from the PTLR, but OP-13D limits Reactor coolant temperature to 68°F.
C. Incorrect – 60°F is the lower limitation for RPV head installation from the PTLR, but OP-13D limits Reactor coolant temperature to 68°F. Going below 68°F would exceed the minimum temperature assumption used in the Shutdown Margin (SDM) calculation.
D. Incorrect – Going below 68°F would exceed the minimum temperature assumption used in the Shutdown Margin (SDM) calculation.

Technical Reference(s): OP-13D, Technical Specification 1.1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.13.d

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 #	600000 AK1.02
	Importance Rating	2.9

Plant Fire On-site

Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: Fire Fighting

Proposed Question: #41

The plant is operating at 100% power with the following:

- A serious fire develops in the Reactor Building West Crescent area.
- The Fire Protection Panel is in alarm and indicates actuation of fire suppression in the area.
- Numerous unexplained EPIC alarm and annunciators have been received in the Control Room.

Which one of the following describes the required action(s) in accordance with AOP-28, Operation During Plant Fires?

A. Commence a normal Reactor Shutdown.

B. Perform a rapid power reduction to 55% core flow.

C. Scram the Reactor and trip both Recirculation pumps.

D. Maintain Reactor power stable and dispatch the Fire Brigade.

Proposed Answer: C

Explanation: The given conditions require entering AOP-28 Attachment 2 to respond to a fire in the Reactor Building West Crescent. The immediate actions of this attachment require a Reactor scram and tripping of Recirculation pumps.

A. Incorrect – A manual Reactor scram is required, NOT a normal shutdown.
B. Incorrect – A manual Reactor scram is required. A rapid power reduction to 55% core flow before the scram is allowable, but NOT required.
D. Incorrect – Although normally it would be desirable to minimize a plant transient, in this case a manual Reactor scram is required.

Technical Reference(s): AOP-28

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03

Question Source: Bank – March 2012 NRC #73

Question History: March 2012 NRC #73

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 #	295006 AK2.07
	Importance Rating	4.0

SCRAM

Knowledge of the interrelations between SCRAM and the following: Reactor pressure control

Proposed Question: #42

The plant is operating at 100% power with the following:

- A Reactor scram occurs.
- The Main Turbine trips.
- The 4160V buses fail to transfer to reserve power.
- APRMs are downscale.
- RPS buses remain energized throughout the transient.
- Three minutes later, the immediate actions of AOP-1, Reactor Scram, have been completed by the At-The-Controls (ATC) operator.

Which one of the following describes the resulting Reactor pressure control strategy?

Reactor pressure will be controlled...

A. by SRVs because the MSIVs automatically closed.

B. by SRVs because the MSIVs were manually closed.

C. at approximately 970 psig by Turbine Bypass Valves.

D. at approximately 1040 psig by Turbine Bypass Valves.

Proposed Answer: B

Explanation: Due to the failure of the 4160V buses to transfer to reserve power, all Circulating Water pumps are de-energized. Therefore, AOP-1 immediate actions require manually closing the MSIVs. With the MSIVs closed, Turbine Bypass Valves are unavailable to control Reactor pressure and SRVs will either automatically cycle or be manually cycled to control Reactor pressure.

A. Incorrect – Since the timing of the EDG loading kept the RPS buses energized throughout the transient, the MSIVs would not automatically close.
C. Incorrect – AOP-1 immediate actions led to MSIVs being closed, therefore Turbine Bypass Valves are unavailable to control Reactor pressure. 970 psig is the normal post-scram pressure at which Turbine Bypass Valves would control.
D. Incorrect – AOP-1 immediate actions led to MSIVs being closed, therefore Turbine Bypass Valves are unavailable to control Reactor pressure. 1040 psig is the normal pressure at which the Turbine Bypass Valves would control at 100% power.

Technical Reference(s): AOP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03

Question Source: Bank – March 2012 NRC #5

Question History: March 2012 NRC #5

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

Which one of the following describes the significance of the 9.58' Torus water level used in EOP-4, Primary Containment Control?

This is the level of the...

A. downcomer openings.

B. the HPCI exhaust pipe opening.

C. the RCIC exhaust pipe opening.

D. the SRV tailpipe sparger openings.

Proposed Answer: A

Explanation: 9.58' is the Torus water level at which the downcomer openings would first come uncovered. EOP-4 uses this Torus water level as the threshold value for performing an emergency RPV depressurization.

B. Incorrect – The HPCI exhaust line opening is at 10.75'. This level is used in EOP-4 as the threshold for tripping HPCI and entering EOP-2.
C. Incorrect – The RCIC exhaust line opening is at 10.75'. This level is used in EOP-4 as the threshold for tripping HPCI and entering EOP-2.
D. Incorrect – 9.58' is based on downcomer vent openings, not SRV sparger openings. SRV sparger openings are lower than 9.58'.

Technical Reference(s): MIT-301.11E

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11E 4.05

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 #	295004 AK2.01
	Importance Rating	3.1

Partial or Complete Loss of DC Power

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: Battery charger

Proposed Question: #44

A plant shutdown is in progress with the following:

- A loss of coolant accident develops.
- Drywell pressure is 3.1 psig and slowly rising.
- Reactor pressure is 460 psig and slowly lowering.
- Breaker 71MCC-252-OD1, 71BC-1A 125V DC Battery Charger A, trips.

Which one of the following describes the effect of these conditions on 71BAT-3B, LPCI MOV Battery B, and 71SB-1A, Station Battery A?

	71BAT-3B, LPCI MOV Battery B	71SB-1A, Station Battery A
A.	Remains on float charge	Remains on float charge
B.	Remains on float charge	Begins to discharge
C.	Begins to discharge	Remains on float charge
D.	Begins to discharge	Begins to discharge

Proposed Answer: D

Explanation: With breaker 71MCC-252-OD1 tripped, AC power to the battery charger for 71SB-1A, Station Battery A, is lost. There is no alternate AC power source for this battery charger to shift to, therefore 71SB-1A begins to discharge. Drywell pressure >2.7 psig causes the AC input to the battery charger for 71BAT-3B, LPCI MOV Battery B, to open. With this separation from the battery charger, 71BAT-3B will also begin to discharge.

A. Incorrect – Both batteries are normally on a float charge, but lose or are separated from their chargers under these conditions and begin to discharge.
B. Incorrect – 71BAT-3B is normally on a float charge, but is separated from its charger under these conditions and begin to discharge.
C. Incorrect – 71SB-1A is normally on a float charge, but loses its charger under these conditions and begin to discharge.

Technical Reference(s): OP-43A, OP-43C

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B 1.05.c.1 and 1.05.c.3

Question Source: Modified Bank – March 2012 NRC #3

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 9/2/14 – Modified last bullet in question stem to directly test loss of battery charger vice loss of an upstream AC board, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AK3.03
	Importance Rating	3.5

Control Room Abandonment

Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT: Disabling control room controls

Proposed Question: #45

AOP-43, Plant Shutdown from Outside the Control Room, requires the At-The-Controls (ATC) operator to place SRV switches at panel 25ASP-5 (300' Admin Building Hallway) in LOCAL.

Given the following possible reasons:

(1) Enable panel 02ADS-71 (300' Reactor Building) SRV control switches.
(2) Disable Control Room SRV control switches.
(3) Disable automatic opening of SRVs due to ADS signal.

Which one of the following identifies which of these reasons for placing the SRV switches in LOCAL are correct?

A. (1) and (2) only

B. (1) and (3) only

C. (2) and (3) only

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

Proposed Answer: C

Explanation: Placing SRV switches at panel 25ASP-5 (300' Admin Building Hallway) in LOCAL is designed to prevent a hot short in the Control Room from opening an SRV. This is accomplished by disabling the Control Room SRV control switch, as well as the automatic ADS function of the SRVs. The panel 02ADS-71 SRV control switches are normally enabled and are unaffected by taking the SRV switches at panel 25ASP-5 to LOCAL.

A. Incorrect – The automatic ADS function of the SRVs is disabled. The panel 02ADS-71 SRV control switches are normally enabled and are unaffected by taking the SRV switches at panel 25ASP-5 to LOCAL.
B. Incorrect – The Control Room SRV control switches are disabled. The panel 02ADS-71 SRV control switches are normally enabled and are unaffected by taking the SRV switches at panel 25ASP-5 to LOCAL.
D. Incorrect – The panel 02ADS-71 SRV control switches are normally enabled and are unaffected by taking the SRV switches at panel 25ASP-5 to LOCAL.

Technical Reference(s): AOP-43, SDLP-02J

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J 1.05.a.11

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 #	1
	Group #	1
	K/A #	295025 EK3.03
	Importance Rating	3.8

High Reactor Pressure

Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: HPCI operation: Plant-Specific

Proposed Question: #46

A scram has occurred with the following:

- MSIVs are closed.
- HPCI automatically started and is the only available injection source.
- HPCI injection has been throttled to 200 gpm with the controller in AUTO.
- Reactor water level is 180 inches and stable.
- Reactor pressure is 900 psig and slowly rising.

Which one of the following describes the response of HPCI flow rate if Reactor pressure rises to 1100 psig?

HPCI flow rate will...

A. lower because HPCI turbine speed is controlled at a constant value.

B. lower because the design discharge pressure range of the HPCI pump is exceeded.

C. remain approximately constant because the control system will throttle the governor based on a flow feedback signal.

D. remain approximately constant because rising Reactor steam supply pressure balances rising pump discharge pressure without the need for governor adjustment.

Proposed Answer: C

Explanation: The HPCI pump is designed to supply 4250 gpm over a Reactor pressure range from 150 to 1195 psig. With rising Reactor pressure, HPCI flow will tend to lower. However, with the controller in AUTO, the HPCI governor valve will be automatically adjusted to maintain constant flow based on a flow feedback signal.

A. Incorrect – Flow will remain approximately constant. When in AUTO, the controller maintains constant flow, not speed.
B. Incorrect – Flow will remain approximately constant. HPCI is rated for 4250 gpm up to 1195 psig.
D. Incorrect – With rising Reactor pressure, HPCI flow will tend to lower based on the natural interplay of steam supply pressure vs. pump discharge pressure. Governor response is required to maintain flow approximately constant over this 200 psig change in Reactor pressure.

Technical Reference(s): OP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23 1.05.a.22

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 #	295031 EK3.05
	Importance Rating	4.2

Reactor Low Water Level

Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL: Emergency depressurization

Proposed Question: #47

A loss of coolant accident has occurred with the following:

- Reactor water level is 20 inches and lowering slowly.
- Reactor pressure is 700 psig and lowering slowly.
- RCIC and CRD are injecting.

Which one of the following describes when an emergency RPV depressurization based on low Reactor water level is required and the associated reason for waiting until this Reactor water level, in accordance with EOP-2, RPV Control?

	When Emergency RPV Depressurization Is Required	Reason For Waiting Until This Reactor Water Level
A.	Before Reactor water level reaches the top of active fuel	Maximize time available to line up additional injection systems
B.	Before Reactor water level reaches the top of active fuel	Maximize time available for decay heat to lower
C.	After Reactor water level reaches the top of active fuel, but before it reaches the minimum steam cooling water level	Maximize time available to line up additional injection systems
D.	After Reactor water level reaches the top of active fuel, but before it reaches the minimum steam cooling water level	Maximize time available for decay heat to lower

Proposed Answer: C

Explanation: EOP-2 requires waiting until Reactor water level is below 0" (top of active fuel) before performing an emergency RPV depressurization. EOP-2 also requires performing the emergency RPV depressurization before Reactor water level drops to -19" (minimum steam cooling RPV water level). The emergency RPV depressurization is delayed until this Reactor water level to allow the maximum amount of time to line up additional injection systems, while adequate core cooling is still assured, in the hope of avoiding the need for the emergency RPV depressurization altogether.

A. Incorrect – Emergency RPV depressurization is delayed until Reactor water level drops below the top of active fuel to maximize the time for restoring injection systems.
B. Incorrect – Emergency RPV depressurization is delayed until Reactor water level drops below the top of active fuel to maximize the time for restoring injection systems.
D. Incorrect – The reason is to provide time for lining up additional injection systems, not to allow decay heat to lower.

Technical Reference(s): EOP-2, MIT-301.11c

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11c 1.07

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 AA1.02
	Importance Rating	4.2

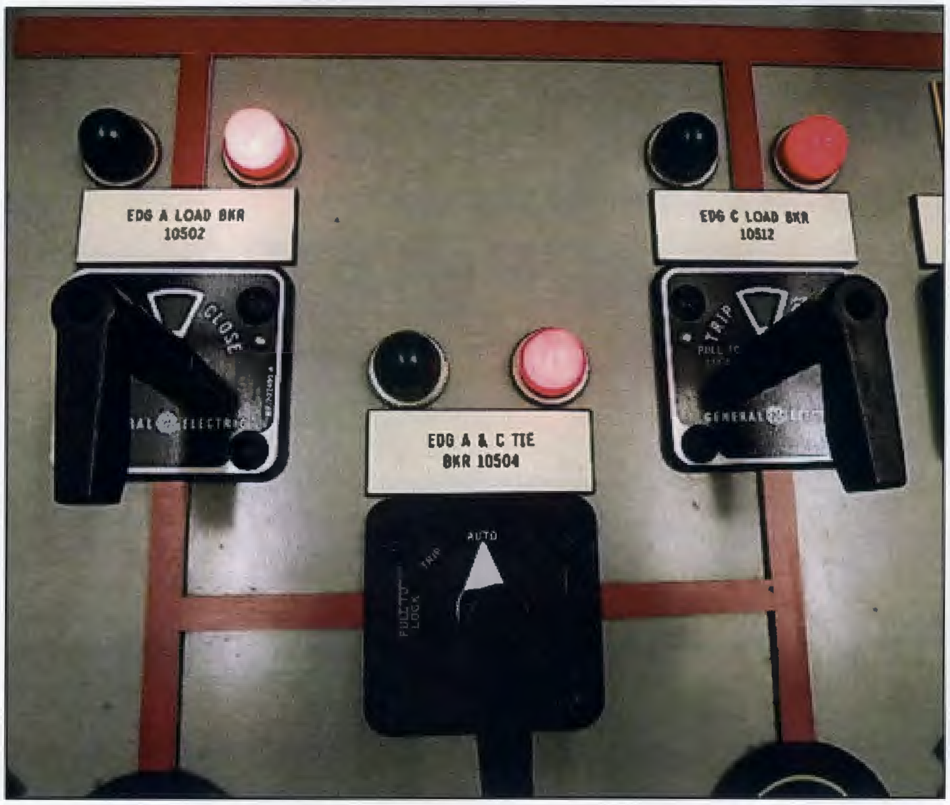
Partial or Complete Loss of AC Power

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

Proposed Question: #48

The plant is operating at 100% power with the following:

- Annunciator 09-8-3-26, BUS 10300 NORM SUPP BKR 10302 TRIP, alarms.
- Bus 10300 Reserve Supply Breaker 10312 is open and remains open.
- There is no fault on the 10300 bus.
- The following picture shows the status of Emergency Diesel Generator (EDG) A and C output and tie breakers **one minute** later:



Which one of the following describes the status of these breakers?

A. All three breakers are in the expected position.

B. The output breakers are in the expected position, but the tie breaker is NOT in the expected position.

C. The tie breaker is in the expected position, but the output breakers are NOT in the expected position.

D. NONE of the breakers are in the expected position.

Proposed Answer: B

Explanation: When breaker 10302 trips with no fault on the 10300 bus, breaker 10312 should close to re-energize the 10300 bus. Since 10312 is open and remains open, the 10300 bus de-energizes. This also causes the 10500 bus to de-energize. Undervoltage on the 10500 bus should cause EDGs A and C to start, their output breakers to close, and their tie breaker to open. The picture shows the output breakers closed as expected, but the tie breaker failed to open as expected.

A. Incorrect – The tie breaker should be open because the EDGs started due to undervoltage.
C. Incorrect – The output breakers should be closed because stem conditions resulted in undervoltage on the 10500 bus. The tie breaker should be open because the EDGs started due to undervoltage.
D. Incorrect – The output breakers should be closed because stem conditions resulted in undervoltage on the 10500 bus.

Technical Reference(s): OP-22

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-93 1.05.b.3

Question Source: Modified Bank – 2008 NRC #2

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 #	295019 AA1.02
	Importance Rating	3.3

Partial or Complete Loss of Instrument Air

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air system valves: Plant-Specific

Proposed Question: #49

The plant is operating at 50% power during a startup with the following:

- An operator reports a major Instrument Air (IA) leak in the plant.
- IA header pressure is 106 psig and lowering.

Which one of the following describes the response of 39FCV-110, Service Air Header Auto Isolation Valve?

39FCV-110 will close when header pressure reaches __(1)__. If header pressure is later recovered to the normal band, 39FCV-110 __(2)__.

	(1)	(2)
A.	95 psig	automatically re-opens
B.	95 psig	requires manual action to re-open
C.	85 psig	automatically re-opens
D.	85 psig	requires manual action to re-open

Proposed Answer: B

Explanation: Lowering IA header pressure results in the following automatic actions:

- 107 psig - first standby IA compressor starts
- 104 psig - second standby IA compressor starts
- 95 psig - 39FCV-110 closes
- 85 psig - 39AOV-111 closes

Therefore, 39FCV-110 closes when header pressure reaches 95 psig. Once air header pressure is back in the normal band, 39FCV-110 remains closed until a local reset switch is depressed.

A. Incorrect – Once air header pressure is back in the normal band, 39FCV-110 remains closed until a local reset switch is depressed.
C. Incorrect – 39FCV-110 closes when header pressure reaches 95 psig. 39AOV-111 (Breathing Ai) closes at 85 psig. Once air header pressure is back in the normal band, 39FCV-110 remains closed until a local reset switch is depressed.
D. Incorrect – 39FCV-110 closes when header pressure reaches 95 psig. 39AOV-111 (Breathing Ai) closes at 85 psig.

Technical Reference(s): AOP-12, OP-39

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39 1.05.c.1

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 #	1
	Group #	1
	K/A #	295038 EA1.07
	Importance Rating	3.6

High Off-site Release Rate

Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: Control room ventilation: Plant-Specific

Proposed Question: #50

The plant is operating at 100% power with the following:

- A Loss of Coolant Accident (LOCA) at Nine Mile Point Unit 2 is causing high radiation levels on-site at JAF.
- The Control Room Ventilation Supply radiation monitor is reading 100 cpm and rising.
- Annunciator 09-75-1-20, CNTRL RM SUPP RAD MON INOP OR HI, is clear.

Which one of the following describes the radiation level at which Annunciator 09-75-1-20 will alarm and the required response upon receipt of the alarm?

Annunciator 09-75-1-20 will alarm when the Control Room Ventilation Supply radiation monitor reaches __(1)__ cpm. Then, the Control Room Ventilation System __(2)__.

	(1)	(2)
A.	500	automatically goes into the ISOLATE mode due to the high radiation signal
B.	500	must be manually placed in the ISOLATE mode using the ISOL & PURGE CNTRL switch on Panel 09-75
C.	1000	automatically goes into the ISOLATE mode due to the high radiation signal
D.	1000	must be manually placed in the ISOLATE mode using the ISOL & PURGE CNTRL switch on Panel 09-75

Proposed Answer: D

Explanation: Annunciator 09-75-1-20 alarms on high Control Room Ventilation Supply radiation level at 1000 cpm. This alarm requires placing Control Room Ventilation in the ISOLATE mode as soon as practicable and in all cases within 30 minutes. However, this alarm does NOT automatically place Control Room Ventilation in the ISOLATE mode. OP-55B gives the operator direction on how to place the system in the ISOLATE mode. This is accomplished by placing the ISOL & PURGE CONTRL switch in ISOL. From this one switch manipulation, the system enters the ISOLATE mode.

A. Incorrect – The high radiation alarm setpoint is 1000 cpm. The system does NOT automatically enter the ISOLATE mode upon alarm.
B. Incorrect – The high radiation alarm setpoint is 1000 cpm.
C. Incorrect – The system does NOT automatically enter the ISOLATE mode upon alarm.

Technical Reference(s): ARP 09-75-1-20, OP-55B

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-70 1.07.a and 1.14

Question Source: Bank – September 2012 NRC #64

Question History: September 2012 NRC #64

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EA2.02
	Importance Rating	3.9

High Drywell Pressure

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

Proposed Question: #51

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

- Torus Spray has just been placed service.
- Drywell pressure is 8 psig and rising slowly.
- Torus pressure is 6 psig and rising slowly.
- Drywell temperature is 340°F and rising slowly.
- Torus water level is 14.1 feet and stable.

Note: The Drywell Spray Initiation Limit is provided on the next page.

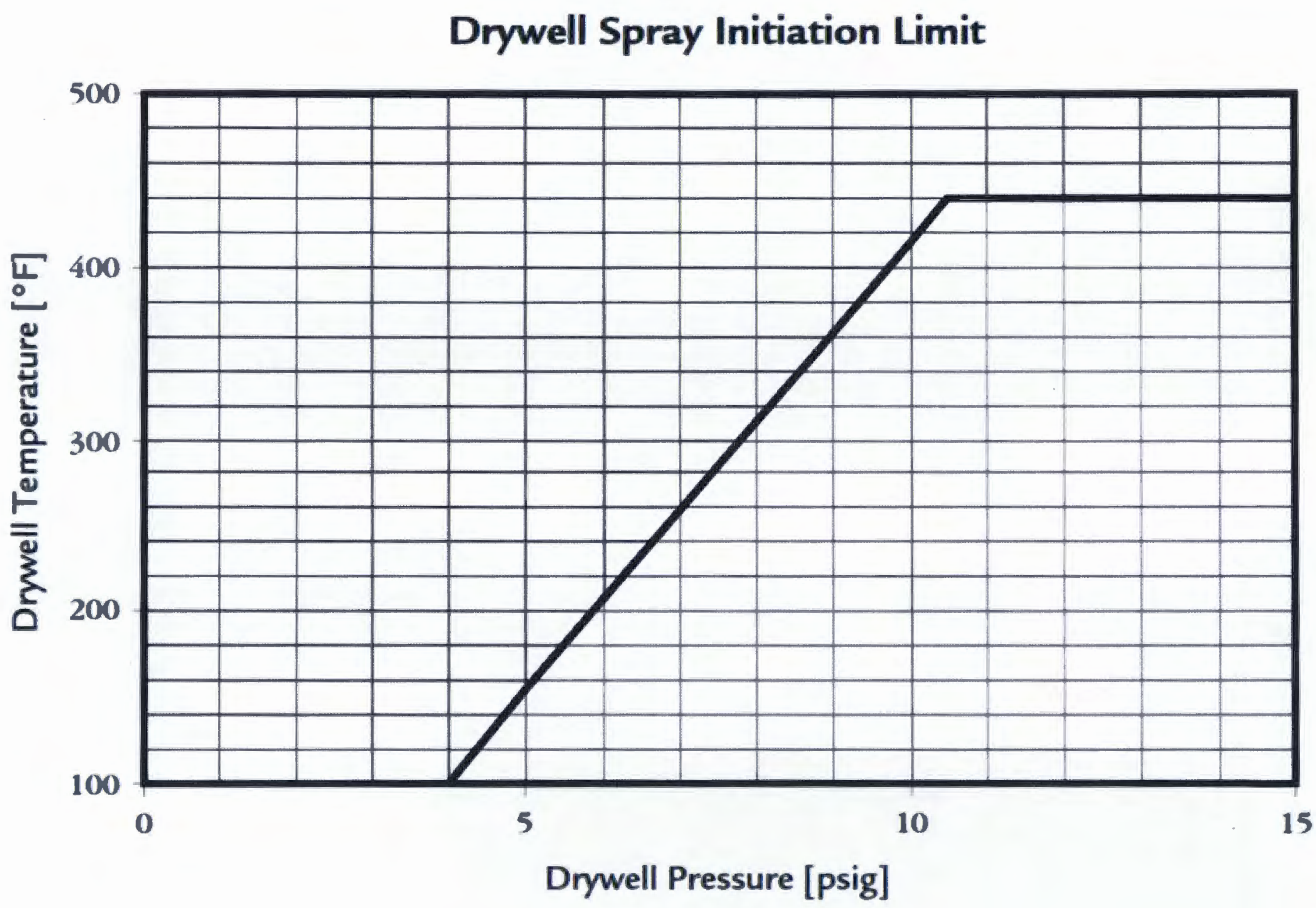
In accordance with EOP-4, Primary Containment Control, the crew shall...

A. initiate Drywell Spray because Drywell pressure is above 2.7 psig.

B. initiate Drywell Spray because Drywell temperature is above 309°F.

C. NOT initiate Drywell Spray because Torus pressure is below 15 psig.

D. NOT initiate Drywell Spray because of the potential for an evaporative pressure drop.


Drywell Spray Initiation Limit
500
400
300
200
100
Drywell Temperature [°F]
0
5
10
15
Drywell Pressure [psig]

Proposed Answer: D

Explanation: With Drywell temperature above 135°F, Drywell pressure above 2.7 psig, and Torus water level above 14.0 feet, EOP-4 is required to be entered. EOP-4 requires Drywell Spray before Drywell temperature reaches 309°F. Since Drywell temperature is 340°F, it is above the threshold requiring initiation of Drywell spray. The given combination of Drywell pressure and temperature exceed the limits of the Drywell Spray Initiation Limit curve, therefore Drywell Sprays CANNOT be placed in service ("NOT OK to spray"). The reason for NOT initiating Drywell Spray when exceeding the Drywell Spray Initiation Limit Curve is to avoid a potential rapid pressure drop due to evaporative cooling.

A. Incorrect – The given combination of Drywell pressure and temperature exceed the limits of the Drywell Spray Initiation Limit curve, therefore Drywell Sprays CANNOT be placed in service ("NOT OK to spray"). Drywell pressure >2.7 psig is a reason for initiating Torus Spray.
B. Incorrect – The given combination of Drywell pressure and temperature exceed the limits of the Drywell Spray Initiation Limit curve, therefore Drywell Sprays CANNOT be placed in service ("NOT OK to spray"). Otherwise, Drywell Spray would be required due to Drywell temperature >309°F.
C. Incorrect – Drywell Spray must not be initiated, but the reason is not that Torus pressure is <15 psig. While this parameter does not itself require Drywell Spray, it also does not preclude Drywell Spray. Even with Torus pressure <15 psig, Drywell temperature >309°F normally makes Drywell Spray required.

Technical Reference(s): EOP-4, MIT-301.11E

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11E 4.05

Question Source: Bank – NMP1 2013 NRC #11

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 9/2/14 – Revised question to ask whether to spray or not, plus the associated reason, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AA2.04
	Importance Rating	2.9

Partial or Complete Loss of CCW

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: System flow

Proposed Question: #52

The plant is operating at 100% power with the following:

- Reactor Building Closed Loop Cooling (RBCLC) pumps A and B are operating.
- RBCLC pump C is in standby.

Then, Bus 10400 de-energizes due to a sustained electrical fault.

Which one of the following describes the effect of this loss on RBCLC system flow?

RBCLC system flow...

A. lowers to and remains at zero.

B. remains the same throughout the transient.

C. lowers and stabilizes at a lower value, but remains above zero.

D. lowers, but then returns to approximately the pre-transient value.

Proposed Answer: D

Explanation: Bus 10400 supplies power to Bus L-14, which supplies power to RBCLC pump B. When RBCLC pump B stops operating, system flow begins to lower. However, RBCLC pump C automatically starts due to the loss of RBCLC pump B and restores RBCLC system flow to approximately the pre-transient value.

A. Incorrect – RBCLC pump A is powered from Bus L-13 and maintains system flow above zero throughout the transient. RBCLC pump C (also powered from Bus L-13) auto-starts to return system flow to approximately the pre-transient value.
B. Incorrect – RBCLC pump B stops operating due to loss of power from Bus L-14, which causes RBCLC system flow to momentarily lower.
C. Incorrect – RBCLC pump C auto-starts to return system flow to approximately the pre-transient value.

Technical Reference(s): OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15 1.05.c.1

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 #	1
	Group #	1
	K/A #	295021 AA2.07
	Importance Rating	2.9

Loss of Shutdown Cooling

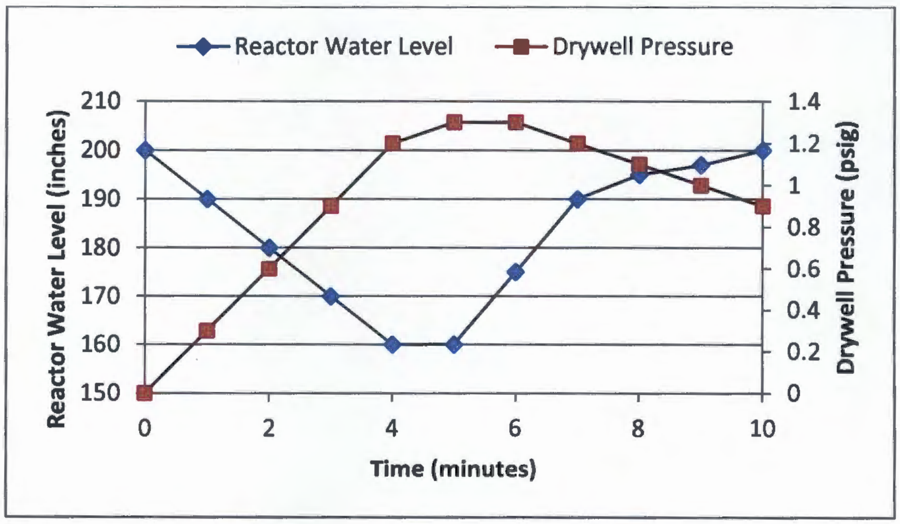
Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor recirculation flow

Proposed Question: #53

The plant is shutdown with the following:

- Shutdown Cooling is in service.
- Both Recirculation loops are in service.

Then, an event results in the following Reactor water level and Drywell pressure transients:



Which one of the following describes the resulting status of forced core circulation at Time = 10 minutes?

Forced core circulation...

A. is lost.

B. is maintained by Recirculation flow, only.

C. is maintained by Shutdown Cooling flow, only.

D. is maintained by both Recirculation and Shutdown Cooling flow.

Proposed Answer: B

Explanation: The given transient results in Reactor water level lowering to a minimum of 160 inches and Drywell pressure rising to a maximum of 1.3 psig. Reactor water level less than 177 inches causes a loss of Shutdown Cooling flow. Reactor water level less than 196.5 inches also would cause a Recirculation runback to 44%, if Recirculation speed is initially above that value. However, Recirculation pumps do NOT trip since Reactor water level remains above 105.4 inches. Therefore, forced core circulation is maintained by Recirculation, but not Shutdown Cooling.

A. Incorrect – Since Reactor water level does not lower to 105.4 inches, Recirculation pumps remain operating.
C. Incorrect – Since Reactor water level lowers below 177 inches, Shutdown Cooling flow is lost due to closure of the LPCI inboard injection valve. Since Reactor water level does not lower to 105.4 inches, Recirculation pumps remain operating.
D. Incorrect – Since Reactor water level lowers below 177 inches, Shutdown Cooling flow is lost due to closure of the LPCI inboard injection valve.

Technical Reference(s): AOP-15, OP-27

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H 1.05.c.2, SDLP-10 1.10.h

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 #	700000 2.4.30
	Importance Rating	2.7

Generator Voltage and Electric Grid Disturbances

Emergency Procedures / Plan; Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

Proposed Question: #54

The plant is operating at 100% power with the following:

- A problem is discovered with Breaker 10022, LHH-FITZ 115 KV LINE 3 BKR 10022.
- The Shift Manager has directed removing Line 3 from service per OP-44, 115 KV System, Section G.3, Removing Lighthouse Hill-Fitzpatrick Line 3 from Service.

Given the following:

(1) Power Control
(2) Nine Mile Point Unit 1 Control Room
(3) Nine Mile Point Unit 2 Control Room

Which one of the following identifies which of these locations are required to be notified of this evolution, in accordance with OP-44?

A. (1) only

B. (1) and (2) only

C. (1) and (3) only

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

Proposed Answer: B

Explanation: OP-44 requires notification of both Power Control and Nine Mile Point Unit 1 Control Room for any operations that are known to affect availability or continuity of 115 KV power. This includes removing Line 3 from service.

A. Incorrect – The Nine Mile Point Unit 1 Control Room is also required to be notified.
C. Incorrect – The Nine Mile Point Unit 1 Control Room is also required to be notified. The Nine Mile Point Unit 2 Control Room is not required to be notified.
D. Incorrect – The Nine Mile Point Unit 2 Control Room is not required to be notified.

Technical Reference(s): OP-44

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71D 1.13.a

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 #	295005 2.1.20
	Importance Rating	4.6

Main Turbine Generator Trip

Conduct of Operations: Ability to interpret and execute procedure steps.

Proposed Question: #55

A plant startup is in progress with the following:

- Reactor power is 24%.
- The Main Generator is on-line.

Then, Main Turbine bearing header pressure drops to 10 psig.

Which one of the following describes (1) the automatic Main Turbine response and (2) the required operator action?

A. (1) The Main Turbine trips.
(2) Execute AOP-1, Reactor Scram.

B. (1) The Main Turbine trips.
(2) Execute AOP-2, Main Turbine Trip Without Scram.

C. (1) The Main Turbine remains online.
(2) Scram the Reactor and execute AOP-1, Reactor Scram.

D. (1) The Main Turbine remains online.
(2) Trip the Main Turbine and execute AOP-2, Main Turbine Trip Without Scram.

Proposed Answer: B

Explanation: Main Turbine bearing oil pressure less than 17 psig causes annunciator 09-5-2-07 to alarm and an automatic Main Turbine trip. The corresponding ARP includes the following steps:

```
1.    IF full scram,
      THEN execute AOP-1, Reactor Scram.
NOTE:     Loss of feedwater heating due to turbine
          trip will cause reactor power to rise.
2.    IF half scram or no scram,
      THEN execute AOP-2, Main Turbine Trip without
      Scram.
```

With Reactor power less than 29%, the Main Turbine trip does NOT cause a Reactor scram. Therefore AOP-2, NOT AOP-1, must be executed.

A. Incorrect – AOP-1, Reactor Scram, would be executed if initial Reactor power was above 29%. However, with Reactor power below 29%, the Main Turbine trip does NOT cause a Reactor scram, so AOP-2 is executed.
C. Incorrect – Main Turbine bearing oil pressure less than 17 psig causes an automatic Main Turbine trip.
D. Incorrect – Main Turbine bearing oil pressure less than 17 psig causes an automatic Main Turbine trip.

Technical Reference(s): ARP 09-5-2-07

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94A 1.05.c.1

Question Source: Modified Bank – 2010 NRC #76

Question History: 2010 NRC #76

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 2.4.18
	Importance Rating	3.3

Suppression Pool High Water Temperature

Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.

Proposed Question: #56

Which one of the following describes (1) when Emergency Depressurization is required based on high Torus temperature and (2) the basis for this requirement, in accordance with EOP-4, Primary Containment Control?

A. (1) Before Torus temperature reaches the Boron Injection Initiation Temperature (BIIT)
(2) Maintain adequate Net Positive Suction Head (NPSH) for ECCS pumps

B. (1) Before Torus temperature reaches the Boron Injection Initiation Temperature (BIIT)
(2) Avoid failure of the Containment or equipment inside the Containment

C. (1) When Torus temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit (HCTL)
(2) Maintain adequate Net Positive Suction Head (NPSH) for ECCS pumps

D. (1) When Torus temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit (HCTL)
(2) Avoid failure of the Containment or equipment inside the Containment

Proposed Answer: D

Explanation: The EOP-4 Torus temperature legs requires Emergency RPV Depressurization if Torus temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit (HCTL). The basis is to not raise Torus water temperature or pressure above limits before the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent, such that failure of the containment and equipment inside the containment is avoided.

A. Incorrect – Entering EOP-2, not Emergency RPV Depressurization, is required before Torus temperature reaches BIIT. ECCS pump NPSH is lowered as Torus temperature rises, however it is not the specific basis for the Emergency RPV Depressurization requirement.
B. Incorrect – Entering EOP-2, not Emergency RPV Depressurization, is required before Torus temperature reaches BIIT.
C. Incorrect – ECCS pump NPSH is lowered as Torus temperature rises, however it is not the specific basis for the Emergency RPV Depressurization requirement.

Technical Reference(s): EOP-4, MIT-301.11e

Proposed references to be provided to applicants during examination: x

Learning Objective: MIT-301.11e 4.05

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 #	295028 EK2.03
	Importance Rating	3.6

High Drywell Temperature

Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Reactor water level indication

Proposed Question: #57

The plant was operating at 100% power when a significant event resulted in the following:

- Reactor pressure is 225 psig and slowly lowering.
- Drywell temperature is 405°F and slowly rising.
- Narrow range Reactor water level instruments indicate 175 inches and stable.
- Wide range Reactor water level instruments indicate 50 inches and stable.
- Fuel zone Reactor water level indicator and recorder are indicating erratically.
- Refuel zone Reactor water level instrument indicates 225 inches and stable.
- EPIC is not operating properly.
- Portions of EOP-11, EOP & SAOG Graphs, are provided on the following page.

Which one of the following identifies which Reactor water level instrument(s), if any, is(are) indicating a valid reading?

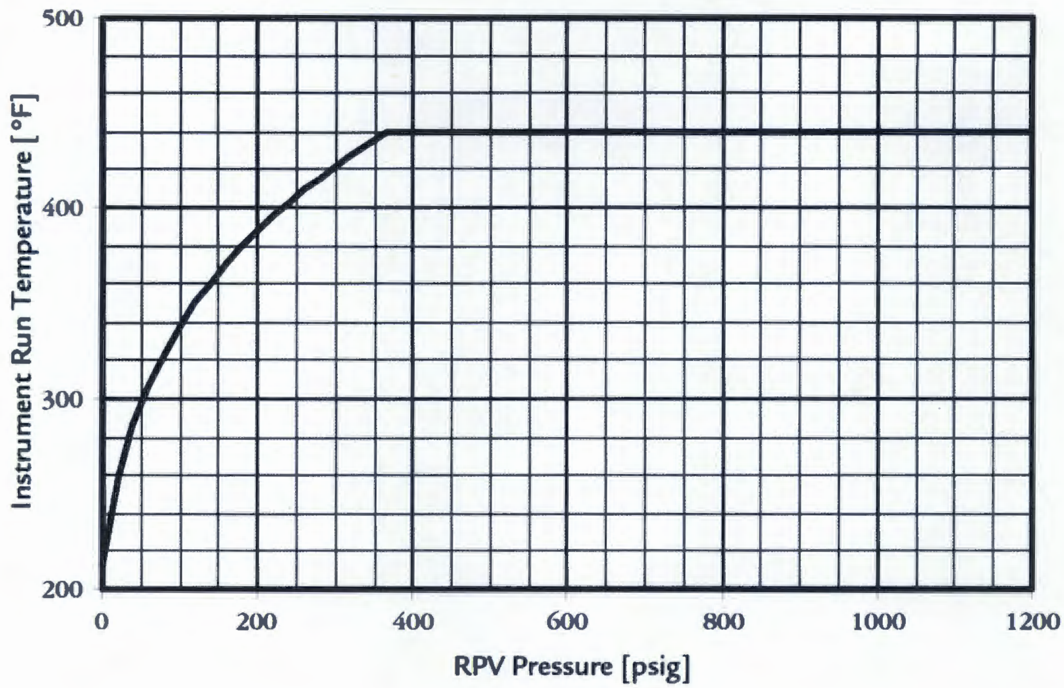
A. None

B. Wide Range

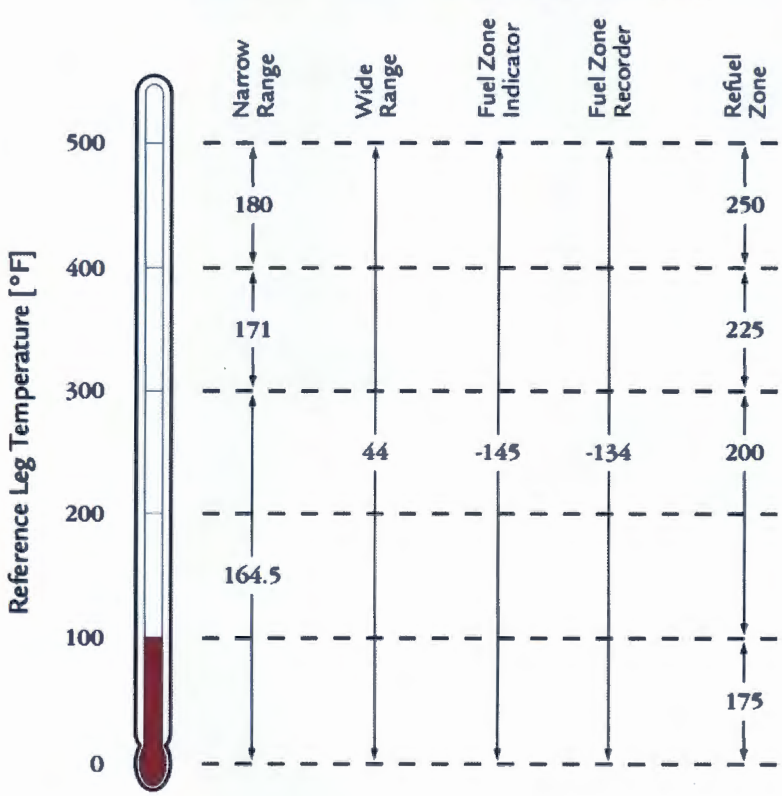
C. Refuel Zone

D. Narrow Range

RPV Saturation Temperature



Minimum Usable Indicating Levels [in.]



Proposed Answer: B

Explanation: The given combination of Reactor pressure and Drywell temperature are on the "bad" side of the RPV Saturation Temperature curve. This means Reactor water level instruments may be unreliable due to boiling, but are not invalid for use until the effects of boiling are observed. Since Wide range, Narrow range, and Refuel zone instruments are indicating stable level, no evidence of boiling is present. With Drywell temperature above 400oF, both Narrow range and Refuel zone indicators are below their Minimum Usable Indicating Levels of 180 inches and 250 inches, respectively. This makes both of these indicators invalid for use. Wide range indicators are above their Minimum Usable Indicating Level of 44 inches and remain valid for use.

A. Incorrect – Even though Reactor pressure and Drywell temperature are on the "bad" side of the RPV Saturation Temperature curve, they are not invalid for use based on this unless indications of boiling are observed.
C. Incorrect – Even though 225 inches is within the normal range for Refuel zone indicators, it is below the current Minimum Usable Indicating Level of 250 inches based on Drywell temperature.
D. Incorrect – Even though 175 inches is within the normal range for Narrow range indicators, it is below the current Minimum Usable Indicating Level of 180 inches based on Drywell temperature.

Technical Reference(s): EOP-11 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11b 1.01

Question Source: Modified Bank – 3/12 NRC #68

Question History: 3/12 NRC #68

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(2)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 2.4.4
	Importance Rating	4.5

SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown

Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: #58

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

- All APRM indication is lost.
- All control rod position indication is lost.

Then, a Reactor scram and Main Turbine trip occurs.

- Reactor water level is maintained within the green band throughout the transient.
- Reactor pressure is 980 psig and stable with two Turbine Bypass Valves open.

Which one of the following describes the EOP entry requirements for these conditions?

A. No EOP entry is required.

B. Enter EOP-2, RPV Control, and remain in this EOP.

C. Enter EOP-2, RPV Control, and then exit to EOP-3, Failure to Scram.

D. Enter EOP-3, Failure to Scram. Entry into EOP-2, RPV Control, is NOT required.

Proposed Answer: C

Explanation: EOP-2 must be entered due to Reactor power above 2.5% or unknown when a scram is required. Reactor power can be determined to be above 2.5%, even with the loss of APRMs, based on Reactor pressure being stable with two Turbine Bypass Valves open. Once in EOP-2, a series of diagnostic questions requires exiting EOP-2 and entering EOP-3 because control rod position indication cannot prove that the Reactor will remain shutdown under all conditions without boron.

A. Incorrect – EOP-2 must be entered based on Reactor power. No other EOP-2 entry conditions are met (Reactor water level, Reactor pressure, or Drywell pressure).
B. Incorrect – With no control rod position indication, the series of diagnostic questions at the beginning of EOP-2 requires exiting EOP-2 and entering EOP-3.
D. Incorrect – Although EOP-3 will eventually be entered, it is only entered after EOP-2 is entered.

Technical Reference(s): EOP-2, EOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11c 1.02, MIT-301.11d 1.02

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 #	295029 EK1.01
	Importance Rating	3.4

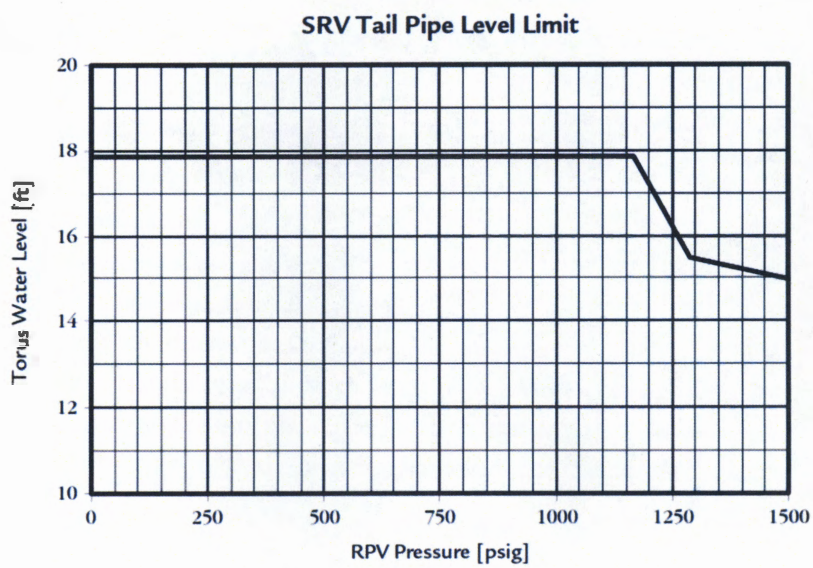
High Suppression Pool Water Level

Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Containment integrity

Proposed Question: #59

A loss of coolant accident has resulted in the following:

- Reactor water level is 150 inches and slowly rising.
- Reactor pressure is 250 psig and slowly lowering.
- Condensate is injecting to the Reactor.
- Core Spray and RHR are available for injection.
- Torus water level is 17.5 feet and slowly rising.
- EOP-2, RPV Control, and EOP-4, Primary Containment Control, are being executed.
- The CRS has determined that Reactor injection sources must be controlled to maintain Torus water level below the SRV Tail Pipe Level Limit.



Which one of the following describes (1) the required control of Condensate injection, in accordance with EOP-4, and (2) the basis for the SRV Tail Pipe Level Limit at this Reactor pressure?

A. (1) Continued injection with Condensate is acceptable.
(2) Prevent damage to the SRV tail pipes.

B. (1) Continued injection with Condensate is acceptable.
(2) Prevent damage to the Primary Containment.

C. (1) Condensate injection must be terminated.
(2) Prevent damage to the SRV tail pipes.

D. (1) Condensate injection must be terminated.
(2) Prevent damage to the Primary Containment.

Proposed Answer: D

Explanation: With Torus water level approaching 17.85 feet, the SRV Tail Pipe Level Limit is about to be exceeded. In this situation, EOP-4 requires terminating injection into the RPV from sources external to the Primary Containment as long as the core will still be adequately cooled. Since Reactor water level is well above the top of active fuel and low pressure ECCS systems are available for injection, Condensate injection can be terminated without jeopardizing adequate core cooling. At Reactor pressures below ~1170 psig, the basis for the SRV Tail Pipe Level Limit is to prevent damage to the Primary Containment by not exceeding the Maximum Pressure Suppression Primary Containment Water Level of 17.85 feet. At Reactor pressures above ~1170 psig, the basis for the SRV Tail Pipe Level Limit is to prevent damage to SRV tailpipes.

A. Incorrect – Condensate injects water from outside of the Primary Containment, and must be terminated to prevent further rise in Torus water level. Damage to the SRV tailpipes is the basis for the limit only at higher Reactor pressures.
B. Incorrect – Condensate injects water from outside of the Primary Containment, and must be terminated to prevent further rise in Torus water level.
C. Incorrect – Damage to the SRV tailpipes is the basis for the limit only at higher Reactor pressures.

Technical Reference(s): EOP-4, EOP-11, MIT-301.11b

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e 4.05

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 #	295002 AK2.05
	Importance Rating	2.7

Loss of Main Condenser Vacuum

Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following: Feedwater system

Proposed Question: #60

A loss of Main Condenser vacuum has resulted in the following:

- The Reactor has been manually scrammed.
- Main Condenser vacuum is 22 inches Hg and slowly lowering.
- Reactor pressure is 800 psig and slowly lowering.
- Reactor water level is 190 inches and slowly rising.

Which one of the following describes the availability of the Condensate and Feedwater system for injection to the Reactor?

Feedwater pumps are currently...

A. available to inject to the Reactor. Feedwater pumps will become unavailable when Main Condenser vacuum lowers to approximately 20 inches Hg.

B. available to inject to the Reactor. Feedwater pumps will become unavailable when Main Condenser vacuum lowers to approximately 8 inches Hg.

C. NOT available to inject to the Reactor. Injection through the Feedwater system will become available when Reactor pressure reaches approximately 700 psig.

D. NOT available to inject to the Reactor. Injection through the Feedwater system will become available when Reactor pressure reaches approximately 255 psig.

Proposed Answer: A

Explanation: At 22 inches Hg, the Feedwater pumps have not yet tripped on low vacuum and the MSIVs have not yet closed on low vacuum. Additionally, Reactor water level is below the Feedwater pump high level trip. With no indications to the contrary, Feedwater pumps are still available for injection to the Reactor. The Feedwater pumps will trip when Main Condenser vacuum reaches 20 inches Hg.

B. Incorrect – Feedwater pumps will become unavailable for injection when Main Condenser vacuum reaches 20 inches. 8 inches Hg corresponds to the MSIV closure, which would make the Feedwater pumps unavailable if the earlier Feedwater pump trip were not in place.
C. Incorrect – At 22 inches Hg, the Feedwater pumps have not yet tripped on low vacuum and the MSIVs have not yet closed on low vacuum. Additionally, Reactor water level is below the Feedwater pump high level trip. With no indications to the contrary, Feedwater pumps are still available for injection to the Reactor.
D. Incorrect – At 22 inches Hg, the Feedwater pumps have not yet tripped on low vacuum and the MSIVs have not yet closed on low vacuum. Additionally, Reactor water level is below the Feedwater pump high level trip. With no indications to the contrary, Feedwater pumps are still available for injection to the Reactor. Condensate Booster pumps become available for injection to the Reactor through the Feedwater system at approximately 700 psig. 255 psig is the Reactor pressure at which Condensate pumps become available for injection through the Feedwater system.

Technical Reference(s): AOP-31, AOP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-33 1.05.c.3.a

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 #	1
	Group #	2
	K/A #	295009 AK3.01
	Importance Rating	3.2

Low Reactor Water Level

Knowledge of the reasons for the following responses as they apply to LOW REACTOR WATER LEVEL: Recirculation pump run back: Plant-Specific

Proposed Question: #61

The plant is operating at 100% power with the following:

- Feedwater pump A trips due to low bearing oil pressure.
- Reactor water level lowers to 190 inches and then recovers to the normal control band.

Which one of the following describes the automatic Recirculation system response and the reason for this response?

Recirculation pump speeds run back to…

A. 30% to maintain adequate Recirculation pump NPSH.

B. 44% to maintain adequate Recirculation pump NPSH.

C. 30% to lower Reactor power within single Feedwater pump capacity.

D. 44% to lower Reactor power within single Feedwater pump capacity.

Proposed Answer: D

Explanation: With less than two Feedwater pumps operating and Reactor water level less than 196.5 inches, Recirculation pump speed is automatically run back to 44%. This runback is designed to rapidly lower Reactor power such that a single Feedwater pump can inject enough water to prevent a Reactor scram on low Reactor water level.

A. Incorrect – The Recirculation pumps run back to 44%. The 30% run back is based on total Feedwater flow less than 20% of rated. Adequate Recirculation pump NPSH is part of the basis for the 30% runback, not the 44% runback.
B. Incorrect – Adequate Recirculation pump NPSH is part of the basis for the 30% runback, not the 44% runback.
C. Incorrect – The Recirculation pumps run back to 44%. The 30% run back is based on total Feedwater flow less than 20% of rated.

Technical Reference(s): OP-27, SDLP-02I

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02I

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 #	2
K/A #	295010 AA1.01
Importance Rating	3.4

High Drywell Pressure

Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: Drywell ventilation/cooling

Proposed Question: #62

A small loss of coolant accident has resulted in the following:

- Drywell pressure is 1.9 psig and slowly rising.
- Drywell temperature is 130°F and slowly rising.
- The CRS has directed you to ensure Drywell cooling is maximized.
- The following Drywell cooling lineup is observed:

DW CLG 68FN-2A	Running	DW CLG 68FN-4A	Running
DW CLG 68FN-2B	Running	DW CLG 68FN-4B	Running
DW CLG 68FN-2C	Running	DW CLG 68FN-4C	Secured
DW CLG 68FN-2D	Secured	DW CLG 68FN-4D	Secured

Which one of the following describes the status of the Drywell cooling lineup, in accordance with OP-53, Drywell Ventilation and Cooling?

A. Drywell cooling is already maximized on both Drywell cooling assemblies.

B. One additional fan should be started on both Drywell cooling assembly A and B.

C. One additional fan should be started on Drywell cooling assembly B. Cooling is already maximized on Drywell cooling assembly A.

D. Two additional fans should be started on Drywell cooling assembly B. One additional fan should be started on Drywell cooling assembly A.

Proposed Answer: C

Explanation: Drywell cooling is maximized by having three fans running on each assembly. The given lineup has three fans running on Drywell cooling assembly A and two fans running on Drywell cooling assembly B. Therefore, one additional fan should be started on Drywell cooling assembly B and Drywell cooling assembly A is maximized.

A. Incorrect – One additional fan should be started on Drywell cooling assembly B.
B. Incorrect – Drywell cooling assembly A already has the maximum of three fans running. Starting a 4^{th} fan would violate the precaution in OP-53.
D. Incorrect – Starting four fans in each assembly would violate the precaution in OP-53.

Technical Reference(s): OP-53

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16B 1.05.a.5.c

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 #	295020 AA2.03
	Importance Rating	3.7

Inadvertent Containment Isolation

Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: Reactor power

Proposed Question: #63

The plant is operating at 20% power with the following:

- The Reactor Mode Switch is in RUN.
- An inadvertent isolation signal causes the following valves to close:
    - MSIV 29AOV-80A
    - MSIV 29AOV-80C
    - MSIV 29AOV-86B

Which one of the following describes Reactor power as indicated on the APRMs one (1) minute after this transient begins?

APRMs will indicate...

A. approximately 0%.

B. greater than 0% but less than 20%.

C. approximately 20%.

D. greater than 20%.

Proposed Answer: A

Explanation: On an isolation signal, MSIVs are required to close within 5 seconds. The MSIV closure scram logic is arranged so that with the Mode Switch in RUN, a scram will occur if an MSIV is closed in at least 3 of the 4 Main Steam Lines. The given valves are distributed between Main Steam Lines A, B, and C, therefore a Reactor scram will occur, even though 20% Reactor steam flow is well within the capacity of a single Main Steam Line. APRMs drop to 0% following the Reactor scram.

B. Incorrect – Reactor power lowers to 0% on APRMs due to a scram. While Core Thermal Power will indicate above 0% for an extended period of time, APRMs will indicate 0% within one minute.
C. Incorrect – Reactor power lowers to 0% on APRMs due to a scram. If only two Main Steam Lines isolated, the Reactor would NOT scram and power would stabilize back at approximately 20% within one minute. While a single Main Steam Line can handle 20% Reactor steam flow, the RPS logic will still enforce a scram in this situation.
D. Incorrect – Reactor power lowers to 0% on APRMs due to a scram. Reactor power will initially rise as MSIVs stroke closed and Reactor pressure rises, however the scram occurs well within one minute. While a single Main Steam Line can handle 20% Reactor steam flow, the RPS logic will still enforce a scram in this situation.

Technical Reference(s): OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-29 1.05.b.1

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:
TRH 9/2/14 – Revised question at a lower power level to raise plausibility of distractors and enhanced explanation, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	500000 2.1.31
	Importance Rating	4.6

High Containment Hydrogen Concentration

Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Proposed Question: #64

A loss of coolant accident has resulted in the following:

- Reactor water level is -60 inches and slowly rising.
- Drywell pressure is 12 psig and slowly rising.
- All required EOP and EP actions have been completed.

Which one of the following describes where position indication for the Hydrogen and Oxygen Sampling System isolation valves can be found and the current position of these isolation valves?

Position indication for the Hydrogen and Oxygen Sampling System isolation valves can be found...

A. in the Relay Room, only. These valves are currently open.

B. in the Relay Room, only. These valves are currently closed.

C. in both the Relay Room and the Control Room. These valves are currently open.

D. in both the Relay Room and the Control Room. These valves are currently closed.

Proposed Answer: C

Explanation: The Hydrogen and Oxygen Sampling System isolation valves are controlled from the Relay Room. Position indication is located both at the control station in the Relay Room and on the PCIS display in the Control Room. The Hydrogen and Oxygen Sampling System isolated due to both Reactor water level less than 177 inches and Drywell pressure greater than 2.7 psig. The PC/G leg of EOP-4 requires placing the systems back in service to monitor gas concentrations, including bypassing interlocks. EP-2 contains the specific steps for restoring the systems. Since all required EOP and EP actions have been completed, the isolation valves are currently open.

A. Incorrect – The controls for the isolation valves are found in only the Relay Room, but there are position indications in both the Relay Room and Control Room.
B. Incorrect – The controls for the isolation valves are found in only the Relay Room, but there are position indications in both the Relay Room and Control Room. While the isolation valves automatically closed on high Drywell pressure and low Reactor water level, all required EOP actions have been completed. This includes re-opening the isolation valves per EOP-4 Section PC/G and EP-2.
D. Incorrect – While the isolation valves automatically closed on high Drywell pressure and low Reactor water level, all required EOP actions have been completed. This includes re-opening the isolation valves per EOP-4 Section PC/G and EP-2.

Technical Reference(s): AOP-15, EOP-4, EP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e 4.05

Question Source: Modified Bank – 2010 NRC #27

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:
TRH 9/2/14 – Revised question to raise plausibility of distractors, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295008 AK2.09
	Importance Rating	3.1

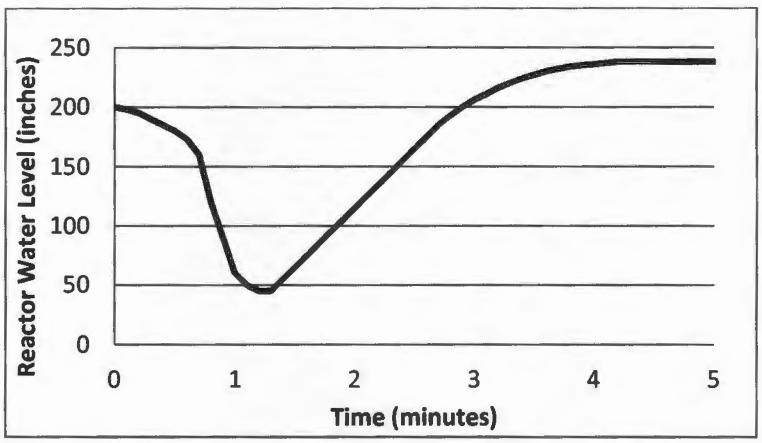
High Reactor Water Level

Knowledge of the interrelations between HIGH REACTOR WATER LEVEL and the following: Reactor water cleanup system (ability to drain): Plant-Specific

Proposed Question: #65

The plant is operating at 100% power with the following:

- A Reactor scram occurs due to low Reactor water level following a Feedwater pump trip.
- 71ACUPS Distribution Bus de-energizes due to a sustained electrical fault.
- The following graph shows the Reactor water level transient:



Which one of the following describes the ability to re-open MSIVs for Reactor pressure control and/or establish Reactor Water Cleanup (RWCU) blowdown flow to lower Reactor water level at ~~time~~ 5 minutes?

A. MSIVs can be reopened for Reactor pressure control and RWCU blowdown flow can be established to lower Reactor water level.

B. MSIVs can be reopened for Reactor pressure control, but RWCU blowdown flow CANNOT be established to lower Reactor water level.

C. RWCU blowdown flow can be established to lower Reactor water level, but MSIVs CANNOT be reopened for Reactor pressure control.

D. MSIVs CANNOT be reopened for Reactor pressure control and RWCU blowdown flow CANNOT be established to lower Reactor water level.

Proposed Answer: ~~B~~ D is correct, per revised key.

Explanation: Due to the Reactor water level transient, MSIVs have closed (59.5 inches) and RWCU has isolated (177 inches). Loss of 71ACUPS also causes an isolation of RWCU due to loss of power to the demin inlet temperature switch. With Reactor water level now high, both of the PCIS isolation signals can be reset, but the RWCU isolation will still be enforced due to the loss of 71ACUPS. Therefore RWCU blowdown flow cannot be established, whereas MSIVs can be reopened to establish TBV flow.

A. Incorrect – RWCU blowdown flow cannot be established due to the sustained loss of 71ACUPS.
C. Incorrect – RWCU blowdown flow cannot be established due to the sustained loss of 71ACUPS. The MSIVs can be reopened after resetting the PCIS Group 1 isolation signal.
~~D~~. ~~Incorrect~~ Correct – The MSIVs ~~can~~ cannot be reopened after resetting the PCIS Group 1 isolation signal.

Technical Reference(s): AOP-15, AOP-21

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-12 1.10.d

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 #	3
	Group #	
	K/A #	2.1.32
	Importance Rating	3.8

Ability to explain and apply all system limits and precautions.

Proposed Question: #66

The plant is operating at 60% power with the following:

- Recirculation pump A has tripped.
- AOP-8, Unexpected Change in Core Flow, has been entered.
- AOP-32, Unplanned Power Change, has been entered.
- The CRS has directed monitoring for indications of thermal-hydraulic instability (THI) per the OP-16, Neutron Monitoring, Posted Attachment.

Which one of the following describes:

(1) the threshold limit for APRM peak to peak oscillations that is used to define THI,

(2) the required action if this limit is exceeded,

in accordance with OP-16 and AOP-32?

	Threshold Limit For APRM Peak To Peak Oscillations	Required Action If Limit Is Exceeded
A.	10%	Manually scram the Reactor
B.	10%	Insert CRAM rods using RMCS
C.	25%	Manually scram the Reactor
D.	25%	Insert CRAM rods using RMCS

Proposed Answer: A

Explanation: The OP-16 Posted Attachment sets 10% APRM peak to peak oscillations as the threshold limit for defining THI. AOP-32 requires inserting a manual Reactor scram if this limit is exceeded.

B. Incorrect – If this limit is exceeded, AOP-32 requires inserting a manual Reactor scram, not inserting CRAM rods. Other steps in AOP-32 require checking the position on the Power-Flow map and inserting CRAM rods if needed.
C. Incorrect – The threshold for APRM peak to peak oscillations is 10%. 25% is the value used in EOP-3 for requiring boron injection.
D. Incorrect – The threshold for APRM peak to peak oscillations is 10%. 25% is the value used in EOP-3 for requiring boron injection. If this limit is exceeded, AOP-32 requires inserting a manual Reactor scram, not inserting CRAM rods. Other steps in AOP-32 require checking the position on the Power-Flow map and inserting CRAM rods if needed.

Technical Reference(s): OP-16, AOP-32

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H 1.09.f

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(2)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.26
	Importance Rating	3.4

Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).

Proposed Question: #67

A tagout is being developed for a water system with the following conditions:

- Highest system water temperature: 250°F
- Highest system water pressure: 250 psig

In accordance with EN-OP-102, Protective and Caution Tagging, the system is...

A. NOT a "High Energy System".

B. a "High Energy System" due to pressure only.

C. a "High Energy System" due to temperature only.

D. a "High Energy System" due to both pressure and temperature.

Proposed Answer: C

Explanation: EN-OP-102 defines a "High Energy System" as having a temperature greater than 200°F or pressure greater than 500 psig. Therefore, this system is a "High Energy System" based on a temperature of >200°F, but not based on pressure.

A. Incorrect – This is a "High Energy System" based on a temperature of >200°F.
C. Incorrect – The system is a "High Energy System", but based on temperature, not pressure. The highest system water pressure is below the threshold of 500 psig.
D. Incorrect – The system is a "High Energy System", but based on temperature, not pressure. The highest system water pressure is below the threshold of 500 psig.

Technical Reference(s): EN-OP-102

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP 44.09.d

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 8/29/14 - Revised question for clarity and to avoid a subset issue based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.1
	Importance Rating	4.5

Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Proposed Question: #68

The plant is shutdown with the following:

- Pre-startup checks are being performed per OP-65, Startup and Shutdown Procedure.
- The following SRM indications are observed:

SRM	Count Rate	Position	Bypassed?
A	2 cps	Fully inserted	Yes
B	120 cps	Fully inserted	No
C	10 cps	Fully inserted	No
D	50 cps	Fully inserted	No

Which one of the following describes the status of the SRMs, in accordance with OP-65?

These SRM indications are...

A. satisfactory to commence Reactor startup.

B. NOT satisfactory to commence Reactor startup. The number of satisfactory SRMs is one less than required.

C. NOT satisfactory to commence Reactor startup. The number of satisfactory SRMs is two less than required.

D. NOT satisfactory to commence Reactor startup. The number of satisfactory SRMs is three less than required.

Proposed Answer: A

Explanation: OP-65 and Technical Specification 3.3.1.2 require at least 3 SRMs indicating greater than 3 cps. SRMs B, C, and D meet this requirement, therefore SRMs support commencing Reactor startup.

B. Incorrect – SRMs are SAT for commencing Reactor startup. SRM A does not have the required number of counts and is bypassed, but only 3 SRMs are required.
C. Incorrect – SRMs are SAT for commencing Reactor startup. 2 less than required would be the answer if the minimum count rate was 100 cps (which is the setpoint for the SRM downscale rod block with detectors not fully inserted).
D. Incorrect – SRMs are SAT for commencing Reactor startup. 3 less than required would be the answer if the minimum count rate was 100 cps and all 4 SRMs were required.

Technical Reference(s): OP-65, Technical Specification 3.3.1.2

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B 1.16

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 #	3
	Group #	
	K/A #	2.2.43
	Importance Rating	3.0

Knowledge of the process used to track inoperable alarms.

Proposed Question: #69

The plant is operating at 100% power with the following:

- I&C is performing troubleshooting on an SDIV level instrument.
- Annunciator 09-5-1-44, SDIV A OR B NOT DRAINED, will be received intermittently for the next four (4) hours.
- The troubleshooting activity will extend into the next shift.

Which one of the following describes requirements for this annunciator per EN-OP-115-08, Annunciator Response?

The annunciator...

A. must be flagged and logged in either the annunciator log or turnover sheet.

B. must be flagged, but does **NOT** need to be logged in either the annunciator log or turnover sheet.

C. must be logged in either the annunciator log or turnover sheet, but does **NOT** need to be flagged.

D. does **NOT** need to be flagged nor logged in either the annunciator log or the turnover sheet.

Proposed Answer: A

Explanation: EN-OP-115-08 Section 5.2[12] describes requirements for flagging of annunciators. For short duration alarms (less than 15 minutes), the CRS/SM may waive flagging requirements. For activities that cause an expected alarm and the annunciator is associated with equipment required in current mode, the following is required:

- Install an annunciator flag on the expected alarm.
- If projected activity duration will exceed remaining time of current shift, then update annunciator log or SRO/RO turnover sheet as appropriate with alarm status.

B. Incorrect – Since the activity is projected to go into the next shift, then the annunciator log or turnover sheet must be updated.
C. Incorrect – Since the activity is projected to last longer than 15 minutes, it must be flagged.
D. Incorrect – Since the activity is projected to last longer than 15 minutes, it must be flagged. Since the activity is projected to go into the next shift, then the annunciator log or turnover sheet must be updated.

Technical Reference(s): EN-OP-115-08

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-Admin 1.02

Question Source: Modified Bank # - September 2012 NRC #69

Question History: September 2012 NRC #69

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 #	2.3.15
	Importance Rating	2.9

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: #70

The plant is operating at 100% power.

Which one of the following radiation monitors would directly require entry into EOP-5, Secondary Containment Control, given a valid upscale reading?

A. 18RIA-051-21, RBC HX AREA

B. 17RM-150A, OFF GAS RAD MON A

C. 18RIA-050-07, RX FDWTR PMP AREA

D. 17RM-050A, STACK GAS RAD MON A

Proposed Answer: A

Explanation: EOP-5 entry is required given any Reactor Building Area Radiation Level above the Maximum Normal value, as given in the associated EOP-5 table. 18RIA-051-21, RBC HX AREA, reading above 5 mr/hr requires entry into EOP-5.

B. Incorrect – The Offgas rad monitors relate to AOP-3 and EAL entry conditions, but are not part of EOP-5 entry conditions.
C. Incorrect – This is an area radiation monitor, but since it is in the Turbine Building, it is not one of the area radiation monitors that require EOP-5 entry.
D. Incorrect – The Stack rad monitors relate to EOP-6 and EAL entry conditions, but are not part of EOP-5 entry conditions.

Technical Reference(s): EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11f 1.04

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.12
	Importance Rating	3.2

Knowledge of Radiological Safety Principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: #71

The plant is operating at 100% power with the following:

- A Main Steam Line break occurs.
- The CRS enters AOP-40, Main Steam Line Break.
- The Reactor is scrammed.
- The MSIVs are closed.

Which one of the following describes a requirement that must be completed within 30 minutes, in accordance with AOP-40?

A. Activate TSC Filtered Ventilation.

B. Activate Standby Gas Treatment.

C. Isolate Turbine Building Ventilation.

D. Isolate Control and Relay Room Ventilation.

Proposed Answer: D

Explanation: AOP-40 requires isolating Control and Relay Room Ventilation within 30 minutes to ensure continued habitability of these areas.

> **Note:** The question meets the K/A by testing knowledge of requirements pertaining to licensed operator duties for aligning ventilation/filtration in response to an adverse radiological situation.

A. Incorrect – AOP-40 requires activating TSC Filtered Ventilation within 60 minutes, not 30 minutes.
B. Incorrect – AOP-40 does not require placing SGT in service. AOP-39, Loss of Coolant, requires activating SGT as part of placing MSLCS in service, but AOP-40 does not mirror these requirements.
C. Incorrect – AOP-40 requires evacuating the Turbine Building, but not isolating Turbine Building Ventilation. This is a possible misconception based on a desire to contain contamination within the Turbine Building, such as with isolating the Reactor Building given a steam leak in that building.

Technical Reference(s): AOP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03.a

Question Source: Modified Bank – 2010 NRC #61

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

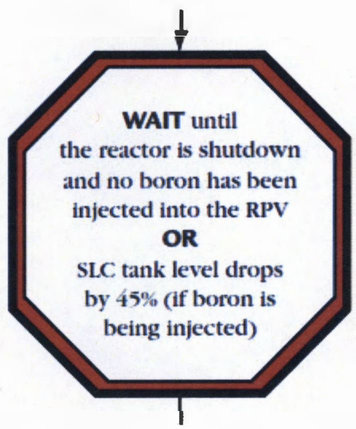
Comments:
TRH 8/29/14 – Added to question explanation to strengthen K/A match based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.21
	Importance Rating	4.0

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: #72

EOP-3, Failure to Scram, contains the following step:



Which one of the following is the threshold for declaring the Reactor "shutdown" for the purpose of this step, in accordance with EOP-3?

A. All APRMs indicate 2.5% or below.

B. All APRMs indicate 10% or below.

C. All IRMs indicate on Range 6 or below.

D. All IRMs indicate on Range 9 or below.

Proposed Answer: C

Explanation: IRM Range 6 or below is the threshold used to make the "shutdown" determination in EOP-3.

A. Incorrect – APRMs < 2.5% power is used as a threshold in determining the level control strategy, but does not provide enough information to make the "shutdown" determination in EOP-3.
B. Incorrect – APRMs <10% power is used as a threshold for taking the Mode Switch out of RUN during an OP-65 shutdown, but does not provide enough information to make the "shutdown" determination in EOP-3.
D. Incorrect – IRMs on Range 9 or below would prohibit taking the Mode Switch to RUN during an OP-65 startup, but does not provide enough information to make the "shutdown" determination in EOP-3.

Technical Reference(s): EP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11d 1.07

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 #	2.4.27
	Importance Rating	3.4

Knowledge of "fire in the plant" procedures.

Proposed Question: #73

The plant is operating at 100% power with the following:

- A fire has developed in the Relay Room.
- The fire requires CO_2 flooding of the Relay Room.
- Annunciator FPP-C, CO_2 System High Temperature, is in alarm.
- The CRS has entered AOP-63, Relay Room CO_2 Discharge.

Which one of the following describes the required operator response, in accordance with AOP-63?

A. Manually initiate Relay Room CO_2. Evacuate all personnel from the Control Room and execute AOP-43, Plant Shutdown From Outside the Control Room.

B. Manually initiate Relay Room CO_2. Evacuate unnecessary personnel from the Control Room, but do NOT execute AOP-43, Plant Shutdown From Outside the Control Room.

C. Ensure automatic initiation of Relay Room CO_2. Evacuate all personnel from the Control Room and execute AOP-43, Plant Shutdown From Outside the Control Room.

D. Ensure automatic initiation of Relay Room CO_2. Evacuate unnecessary personnel from the Control Room, but do NOT execute AOP-43, Plant Shutdown From Outside the Control Room.

Proposed Answer: B

Explanation: With Annunciator FPP-C, CO_2 System High Temperature, in alarm, any CO_2 system other than for the Relay Room would automatically initiate after a 60 second time delay. However, the Relay Room CO_2 system is unique in that it does not automatically initiate. AOP-63 directs manual initiation of Relay Room CO_2 once it is determined that flooding is required. AOP-63 directs evacuating all unnecessary personnel from the Control Room, but allows operators to remain in the Control Room with the use of precautionary breathing apparatus. This prevents the need to perform AOP-43.

A. Incorrect – Operators remain in the Control Room, therefore AOP-43 is not required.
C. Incorrect – The Relay Room CO_2 system does not automatically initiate like other CO_2 systems. Operators remain in the Control Room, therefore AOP-43 is not required.
D. Incorrect – The Relay Room CO_2 system does not automatically initiate like other CO_2 systems.

Technical Reference(s): AOP-63

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03.a

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

See Separate Attachment for Question #74
Due to Sensitive Security Information

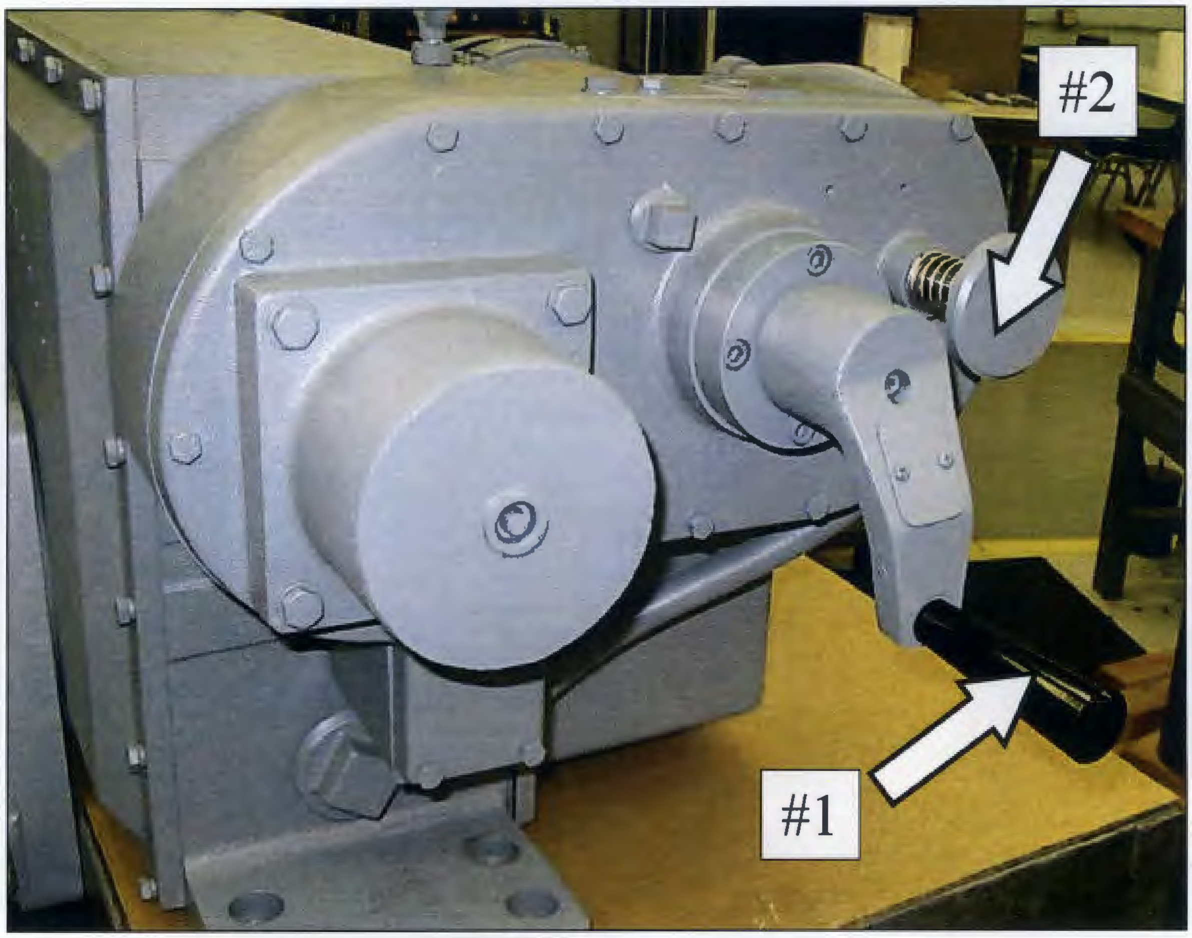
Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.30
	Importance Rating	4.4

Ability to locate and operate components, including local controls.

Proposed Question: #75

The plant is operating at 75% power with the following:

- The RWR MG set B scoop tube has been manually locked due to controller oscillations.
- RWR MG set B speed must be lowered due to high generator temperatures.
- At panel 02-184ACT-1BAMP, the power OFF/ON switch has been taken to the OFF position.
- Refer to the pictures below of RWR MG set B scoop tube positioner, with two controls labeled #1 and #2:



Which one of the following describes the control(s) required to be operated to adjust RWR MG set A speed locally, in accordance with OP-27, Recirculation System?

A. Operate control #1, only.

B. Operate control #2, only.

C. Operate control #1, then operate control #2.

D. Operate control #2, then operate control #1.

Proposed Answer: D

Explanation: To locally change RWR MG set B speed, first the plunger (labeled control #2) must be depressed, then the hand crank (labeled control #1) must be rotated.

A. Incorrect – Both controls must be operated.
B. Incorrect – Both controls must be operated.
C. Incorrect – Control #2 must be operated first, then control #1.

Technical Reference(s): OP-27

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02I 1.13.c

Question Source: Bank – 2013 NMP1 NRC #68

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295030 EA2.03
	Importance Rating	3.9

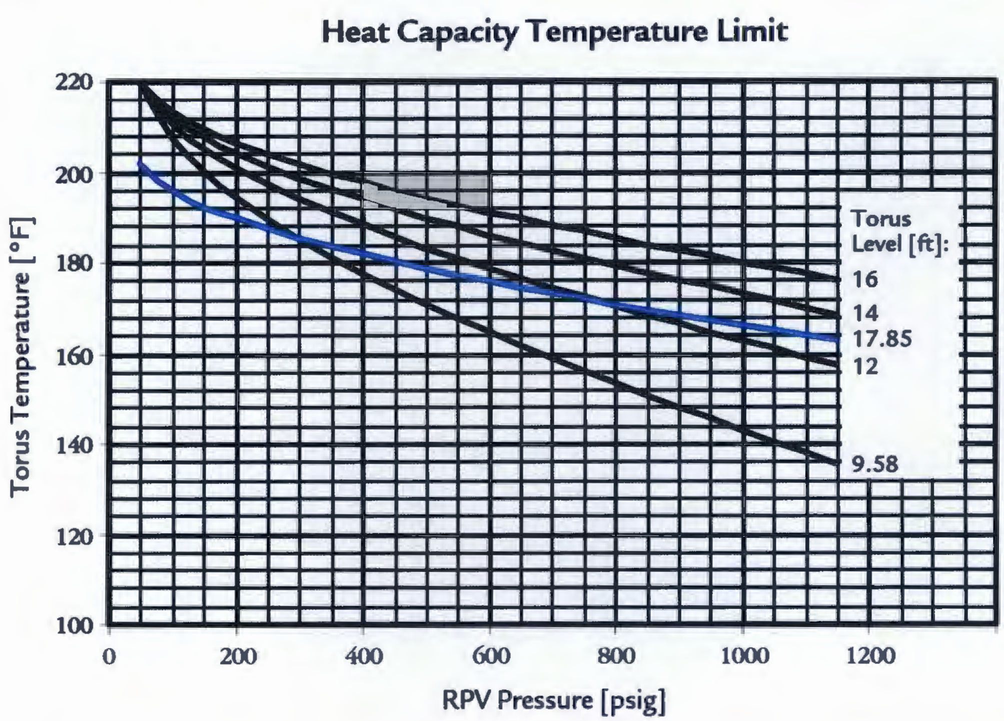
Low Suppression Pool Water Level

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Reactor pressure

Proposed Question: #76

The plant has experienced a seismic event with the following:

- A failure to scram occurred.
- Reactor power is approximately 30% and stable.
- Reactor water level is 70 inches and slowly lowering.
- Reactor pressure is 850 psig and slowly rising.
- Reactor pressure is being controlled in a band of 800-1000 psig using Turbine Bypass Valves and SRVs.
- Torus water temperature is 160°F and slowly rising.
- Torus water level is 11.5 feet and stable.
- EOP-3, Failure to Scram, and EOP-4, Primary Containment Control, are being executed.



Which one of the following describes the impact of these conditions on the current Reactor pressure control strategy, in accordance with the EOPs?

A. Enter EOP-3a, Failure to Scram – ED, based on the EOP-4 Torus Water Level leg (T/L).

B. Enter EOP-3a, Failure to Scram – ED, based on the EOP-4 Torus Temperature leg (T/T).

C. Continue to execute the EOP-3 RPV Pressure leg (RPV/P) step to stabilize Reactor pressure below 1080 psig.

D. Execute the EOP-3 RPV Pressure leg (RPV/P) override for lowering Reactor pressure, exceeding a cooldown rate of 100°F/hr if necessary. Entering EOP-3a, Failure to Scram – ED, is NOT required.

Proposed Answer: B

Explanation: The given conditions give low Torus water level and high Torus water temperature. Since Torus water level is below 12 feet, the 9.58 feet curve of the HCTL graph applies. For the given Reactor pressure and Torus water temperature, this HCTL curved is violated. Since Reactor power is well above the capacity of the Turbine Bypass Valves, Reactor pressure cannot be lowered to restore below HCTL below the limit (any pressure reduction with SRVs will be accompanied by even higher Torus water temperatures. Since Torus water temperature and Reactor pressure cannot be restored and maintained below HCTL, the EOP-4 temperature leg requires entering EOP-3a.

A. Incorrect – While Torus water level is low, it is stable and not threatening the 9.58 foot threshold requiring ED.
C. Incorrect – With Torus water level below 12 feet, Torus water temperature/Reactor pressure well above the 9.58 feet HCTL curve, and Reactor power well above the capacity of the TBVs, ED is required.
D. Incorrect – With Torus water level below 12 feet, Torus water temperature/Reactor pressure well above the 9.58 feet HCTL curve, and Reactor power well above the capacity of the TBVs, ED is required. Since ED is required, EOP-3 is exited, therefore execution of the EOP-3 override is NOT correct.

Technical Reference(s): EOP-3, EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e 4.05

Question Source: Modified Bank – 2009 NMP1 NRC #15

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295019 AA2.02
	Importance Rating	3.7

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: #77

The plant is operating at 100% power with the following:

- An air leak has developed in the Reactor Building.
- Annunciator 09-5-1-54, SCRAM AIR HDR PRESS HI OR LO, alarms.
- 03PI-210, SCRAM AIR HDR PRESS, on Panel 09-5 indicates 64 psig and slowly lowering.

Which one of the following describes the procedure to be entered and the required action?

	Procedure To Be Entered	Required Action
A.	AOP-12, Loss of Instrument Air	A manual Reactor scram is required now.
B.	AOP-69, Control Rod Drive Pump Trouble	A manual Reactor scram is required now.
C.	AOP-12, Loss of Instrument Air	A manual Reactor scram is NOT currently required, but will be if any control rod drift occurs.
D.	AOP-69, Control Rod Drive Pump Trouble	A manual Reactor scram is NOT currently required, but will be if any control rod drift occurs.

Proposed Answer: C

Explanation: Annunciator 09-5-1-54 indicates a low air pressure to the CRD scram air header. AOP-12, Loss of Instrument Air, provides guidance for this situation, but AOP-69, Control Rod Drive Pump Trouble, does NOT. AOP-12 contains scram criteria for this loss of air. There is no scram requirement solely on low scram air header pressure, therefore a scram is not currently REQUIRED. However, if any CRD drift annunciator is received, then a manual Reactor scram is required.

A. Incorrect – A manual scram is not currently required.
B. Incorrect – AOP-12, not AOP-69, provides the guidance for this situation. A manual scram is not currently required.
D. Incorrect – AOP-12, not AOP-69, provides the guidance for this situation.

Technical Reference(s): ARP 09-5-1-54, AOP-12, AOP-69

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03.a

Question Source: Bank – 2013 NMP1 Audit #80

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295028 EA2.05
	Importance Rating	3.8

High Drywell Temperature

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: Torus/suppression chamber pressure: Plant-Specific

Proposed Question: #78

A loss of coolant accident has resulted in the following:

- Reactor water level is 100 inches and slowly rising.
- Reactor pressure is 700 psig and slowly lowering.
- Drywell pressure is 30 psig and slowly rising.
- Drywell temperature is 220°F and slowly rising.
- Torus pressure is 27 psig and slowly rising.
- Torus water level is 13.5 feet and stable.
- No actions have yet been directed from EOP-4, Primary Containment Control.

__Note:__ The Drywell Spray Initiation Limit and Pressure Suppression Pressure curves are provided on the following page.

Which one of the following describes the required direction(s) to be given by the CRS, in accordance with EOP-4?

Direct...

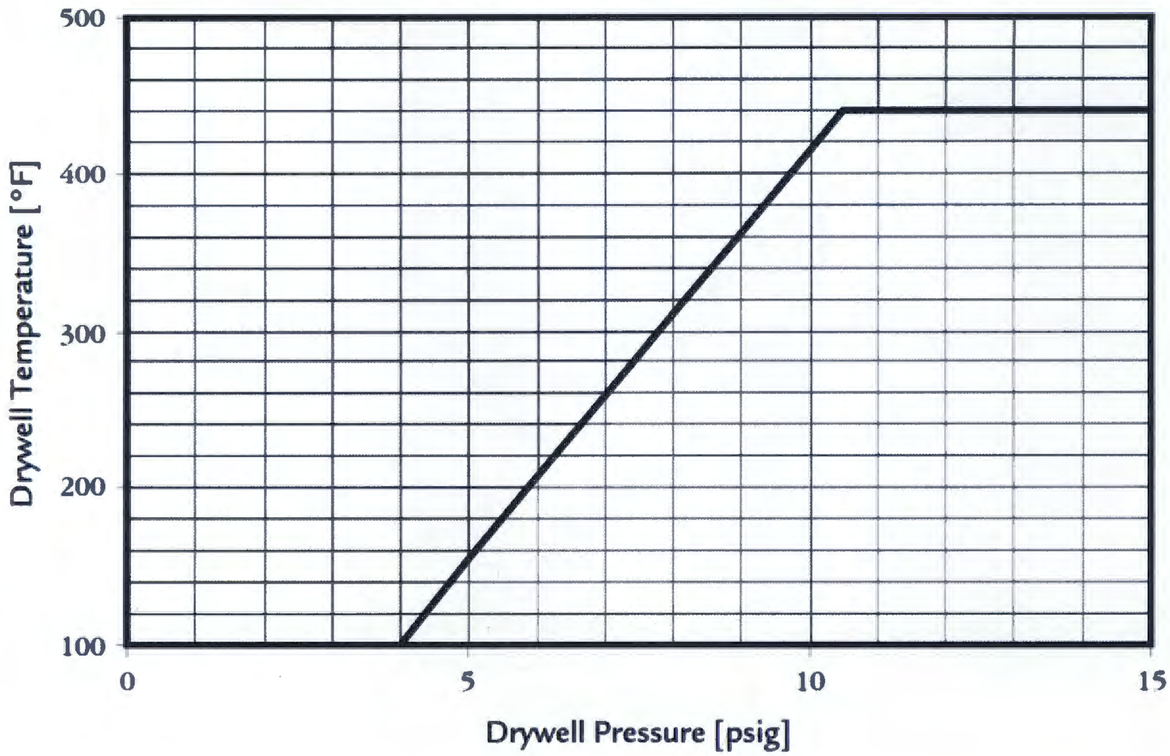
A. initiation of Torus spray and Drywell spray. Also direct an Emergency RPV Depressurization.

B. initiation of Torus spray, but NOT Drywell spray. Also direct an Emergency RPV Depressurization.

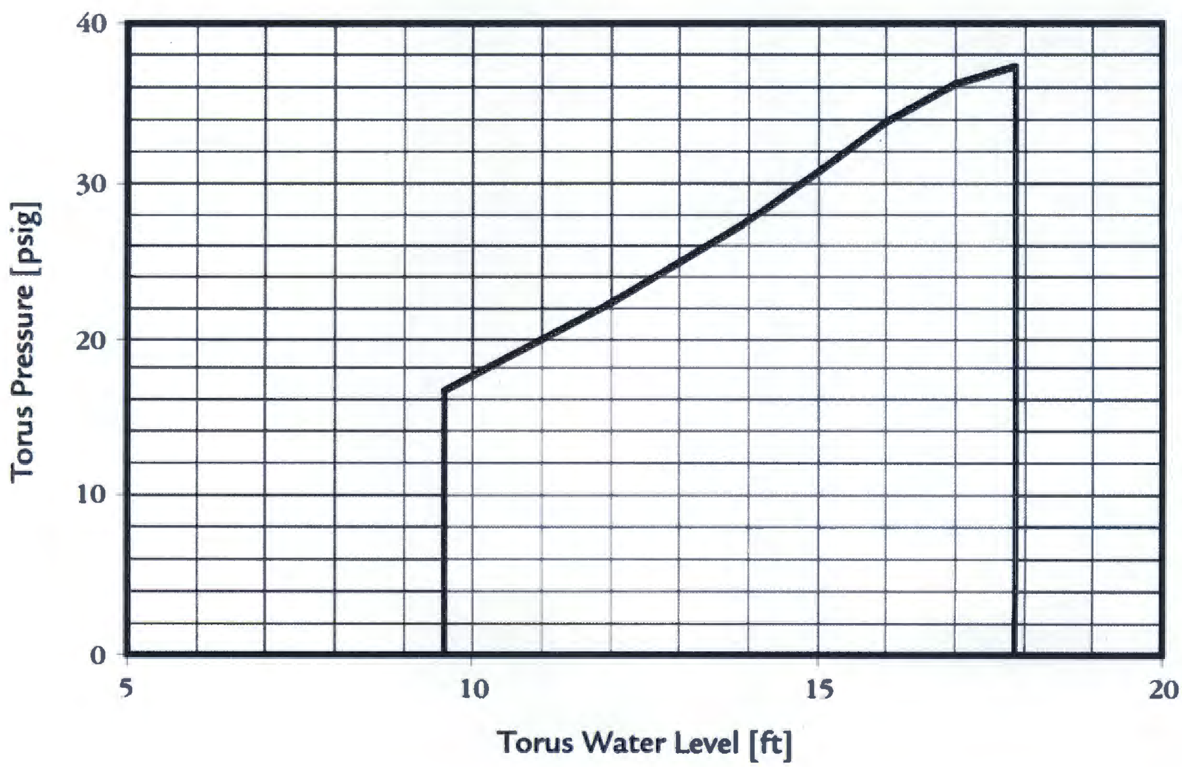
C. initiation of Torus spray and Drywell spray. Once sprays are in service, re-assess the need for an Emergency RPV Depressurization.

D. initiation of Torus spray, but NOT Drywell spray. Once Torus spray is in service, re-assess the need for an Emergency RPV Depressurization.

Drywell Spray Initiation Limit



Pressure Suppression Pressure



Proposed Answer: C

Explanation: The given Torus pressure is above the thresholds for Torus spray (2.7 psig), Drywell spray (15 psig), and Emergency RPV Depressurization (PSP curve). However, since no actions have yet been performed in EOP-4, sprays must be placed in service before evaluating the PSP curve for determination of the need for an Emergency RPV Depressurization. Both Torus and Drywell spray are allowed with the given conditions (Torus water level < 26 feet, Drywell pressure/temperature within DWSIL curve).

A. Incorrect – The need for an Emergency RPV Depressurization cannot yet be determined. This determination is made after sprays have been placed in service.
B. Incorrect – Drywell sprays should be directed since Torus pressure is above 15 psig and Drywell pressure/temperature are within the DWSIL curve. The need for an Emergency RPV Depressurization cannot yet be determined. This determination is made after sprays have been placed in service.
D. Incorrect – Drywell sprays should be directed since Torus pressure is above 15 psig and Drywell pressure/temperature are within the DWSIL curve.

Technical Reference(s): EOP-4, EOP-11, ODSO-49

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11e 4.05

Question Source: Modified Bank - 2008 NRC #81

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295038 2.4.4
	Importance Rating	4.7

High Off-site Release Rate

Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: #79

The plant is operating at 100% power with the following:

- Annunciator 09-3-2-27, Off Gas Rad Mon Hi, is in alarm.
- Annunciator 09-3-2-28, Stack Rad Mon Hi, is in alarm.
- Offgas Radiation Monitors, 17RM-150A and B, are indicating 700 mR/hr and slowly rising.
- Stack Gas Radiation Monitors, 17RM-050A and B, are indicating 2 x 10^5 cps and slowly rising.
- Stack High Range Radiation Monitors, 17RM-53A and B, are indicating 10 mR/hr and slowly rising.

Which one of the following describes the entry requirements for AOP-3, High Activity in Reactor Coolant or Offgas, and EOP-6, Radioactivity Release Control?

	AOP-3 Entry Required?	EOP-6 Entry Required?
A.	No	No
B.	No	Yes
C.	Yes	No
D.	Yes	Yes

Proposed Answer: C

Explanation: AOP-3 entry is required due to Annunciator 09-3-2-27 being in alarm (and Offgas radiation levels supporting the alarm as valid), even without the corresponding Hi-Hi alarm in (09-3-2-38). EOP-6 entry is not required because the given Stack radiation levels are not above the Alert level for Emergency Plan offsite release rate.

Note: The question satisfies SRO level question guidelines by requiring assessment of Emergency Action Levels (EALs) to determine need for EOP-6 entry.

A. Incorrect – AOP-3 entry is required.
B. Incorrect – AOP-3 entry is required. EOP-6 entry is not required.
D. Incorrect – EOP-6 entry is not required.

Technical Reference(s): ARP 09-3-2-27, ARP 09-3-2-28, AOP-3, EOP-6, Hot EAL Chart

Proposed references to be provided to applicants during examination: Hot EAL Chart, with Table F-1 Row D blocked out

Learning Objective: LP-AOP 1.01 and 1.12, MIT-301.11g 6.05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	700000 2.2.37
	Importance Rating	4.6

Generator Voltage and Electric Grid Disturbances

Ability to determine operability and/or availability of safety related equipment.

Proposed Question: #80

The plant is operating at 100% power with the following:

- 115 KV System voltage is 114.5 KV and slowly lowering.
- All load tap changers are operating in Auto.
- AOP-72, 115 KV Grid Loss, Instability, or Degradation, is being executed.
- Power Control reports that the Low Voltage Post Contingency Alarm voltage is 105 KV.

Which one of the following describes the parameter used by AOP-72 to determine operability of the 115 KV Lines and current operability of the 115 KV Lines based on this parameter, in accordance with AOP-72?

	Parameter Used By AOP-72 To Determine Operability of 115 KV Lines	Current Operability of 115 KV Lines Based On This Parameter
A.	Actual system voltage	Operable
B.	Actual system voltage	Inoperable
C.	Post contingency voltage	Operable
D.	Post contingency voltage	Inoperable

Proposed Answer: D

Explanation: AOP-72 and Technical Specifications base 115 KV Line operability on the status of the post contingency voltage. With load tap changers in Auto and post contingency voltage less than 109.3 KV, the 115 KV Lines are inoperable.

A. Incorrect – AOP-72 bases operability on post contingency voltage.
B. Incorrect – AOP-72 bases operability on post contingency voltage.
C. Incorrect – Since load tap changers are in Auto, the 115 KV Lines would remain operable with post contingency voltage less than 111.1 KV but greater than 109.3 KV. However, with post contingency voltage less than 109.3 KV, the 115 KV lines are inoperable.

Technical Reference(s): AOP-72, Technical Specification 3.8.1 Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71D 1.16

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:
TRH 8/29/14 – Revised 55.43 designation from (5) to (2) based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295016 2.4.11
	Importance Rating	4.2

Control Room Abandonment

Knowledge of abnormal condition procedures.

Proposed Question: #81

The plant is operating at 100% power with the following:

- A fire in the Control Room requires execution of AOP-43, Plant Shutdown From Outside The Control Room.
- Operators have completed all required immediate actions of AOP-43.

Which one of the following describes the required direction to be given to control Reactor pressure and water level, in accordance with AOP-43?

Direct...

A. stabilizing Reactor pressure below 1080 psig and raising Reactor water level above 222.5 inches.

B. commencing a rapid Reactor depressurization and raising Reactor water level above 222.5 inches.

C. stabilizing Reactor pressure below 1080 psig and stabilizing Reactor water level between 177 and 222.5 inches.

D. commencing a rapid Reactor depressurization and stabilizing Reactor water level between 177 and 222.5 inches.

Proposed Answer: B

Explanation: AOP-43 requires a series of actions that result in rapidly lowering Reactor pressure by opening seven SRVs and injecting with RHR to raise Reactor water level. No direct is given to subsequently throttle RHR, resulting in Reactor water level rising above the normal upper limit of 222.5 inches.

A. Incorrect – Reactor pressure is required to be rapidly lowered by opening SRVs, not stabilized.
C. Incorrect – Reactor pressure is required to be rapidly lowered by opening SRVs, not stabilized. Reactor water level is required to be raised using LPCI injection, not stabilized in the normal control band.
D. Incorrect – Reactor water level is required to be raised using LPCI injection, not stabilized in the normal control band.

Technical Reference(s): AOP-43

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP 1.03.a

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295023 2.4.41
	Importance Rating	4.6

Refueling Accidents

Knowledge of the emergency action level thresholds and classifications.

Proposed Question: #82

The plant is shutdown for a refueling outage with the following:

- Core shuffle is in progress.
- A seismic event is felt by Control Room operators.
- The EPIC seismic activity alarm is received.
- An irradiated fuel bundle is dropped and damaged in the Spent Fuel Pool.
- Refuel Floor Exhaust radiation monitors indicate 500 mR/hr and slowly rising.
- The Spent Fuel Pool area radiation monitor indicates upscale.

Which one of the following describes the highest Emergency Action Level (EAL) met or exceeded and the time limit for declaring this EAL, in accordance with IAP-2, Classification of Emergency Conditions?

	Highest EAL Met or Exceeded	Maximum Time Limit for Declaring EAL
A.	Alert	15 minutes
B.	Alert	30 minutes
C.	Unusual Event	15 minutes
D.	Unusual Event	30 minutes

Proposed Answer: A

Explanation: A seismic event that is both felt by Control Room operators and results in the EPIC alarm meets the criteria for Unusual Event HU1.1. Insufficient information is available to determine the seismic event has also met the criteria for Alert HA1.1. The Refuel Floor Exhaust radiation monitor readings are below the Unusual Event Level. The damaged irradiated fuel causing the Spent Fuel Pool area radiation monitor to exceed 1000 mR/hr (upscale) meets the criteria for Alert AA2.1. Therefore, the highest EAL met is an Alert. IAP-2 requires the EAL to be declared within 15 minutes. The Part 1 notification is required to be initiated within a total of 30 minutes.

B. Incorrect – IAP-2 requires the EAL to be declared within 15 minutes. The Part 1 notification is required to be initiated within a total of 30 minutes.
C. Incorrect – Unusual Event HU1.1 is met, but so is the higher Alert EAL AA2.1.
D. Incorrect – Unusual Event HU1.1 is met, but so is the higher Alert EAL AA2.1. IAP-2 requires the EAL to be declared within 15 minutes. The Part 1 notification is required to be initiated within a total of 30 minutes.

Technical Reference(s): IAP-2, EAL Chart

Proposed references to be provided to applicants during examination: Hot and Cold EAL Charts with Table F-1 Row D blocked out

Learning Objective: LP-AOP 1.12

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295015 AA2.02
	Importance Rating	4.2

Incomplete SCRAM

Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM: Control rod position

Proposed Question: #83

The plant is operating at 50% power with the following:

- Feedwater pump A trips.
- An Operator places the Reactor Mode Switch in SHUTDOWN.
- Ten control rods stick at position 02.
- All other control rods fully insert.
- Reactor water level reaches a low of 130 inches and then begins rising.

Which one of the following describes the required direction to be given for Reactor water level and pressure control, in accordance with the Emergency Operating Procedures?

	Reactor Water Level Control	Reactor Pressure Control
A.	Restore and maintain Reactor water level 177-222.5 inches.	Reactor cooldown is allowed.
B.	Restore and maintain Reactor water level 177-222.5 inches.	Reactor cooldown is NOT allowed.
C.	Terminate and prevent injection and lower Reactor water level to at least 110 inches.	Reactor cooldown is allowed.
D.	Terminate and prevent injection and lower Reactor water level to at least 110 inches.	Reactor cooldown is NOT allowed.

Proposed Answer: A

Explanation: With Reactor water level below 177 inches, EOP-2 must be entered. At the beginning of EOP-2, there is a series of diagnostic steps that determine if the Reactor will remain shutdown under all conditions without boron. These steps determine if the operator remains in EOP-2 or transitions to EOP-3. One criteria that allows the operator to determine that the Reactor will remain shutdown under all conditions without boron is if all control rods inserted to or beyond position 02 (defined as the Maximum Subcritical Banked Withdrawal Position at JAF). With all control rods inserted to position 02 or beyond, the operator remains in EOP-2. EOP-2 requires Reactor water level restored and maintained 177 to 222.5 inches and allows Reactor cooldown, even with some control rods still not fully inserted.

B. Incorrect – EOP-2 allows Reactor cooldown, even with some control rods still not fully inserted.
C. Incorrect – EOP-2 requires Reactor water level restored and maintained 177 to 222.5 inches.
D. Incorrect – EOP-2 requires Reactor water level restored and maintained 177 to 222.5 inches. EOP-2 allows Reactor cooldown, even with some control rods still not fully inserted.

Technical Reference(s): EOP-2, EOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11b 1.03 and 1.07

Question Source: Bank - NMP1 2013 NRC #5

Question History: NMP1 2013 NRC #5

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295036 2.4.47
	Importance Rating	4.2

Secondary Containment High Sump/Area Water Level

Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: #84

The plant is operating at 100% power with the following:

- A fire header break has occurred in the Reactor Building.
- EOP-5, Secondary Containment Control, is being executed.
- Attempts to isolate the break have been unsuccessful.
- West Crescent area water level is 8 inches and rising at a rate of 1.0 inch/minute.
- East Crescent area water level is 3 inches and rising at a rate of 0.5 inches/minute.

Note: Assume the area water level rates of rise remain constant.

Which one of the following describes a required direction to be given, in accordance with EOP-5?

In approximately ___(1)___, it will be necessary to direct ___(2)___.

	(1)	(2)
A.	10 minutes	a Reactor shutdown, but NOT an emergency RPV depressurization
B.	10 minutes	an emergency RPV depressurization
C.	30 minutes	a Reactor shutdown, but NOT an emergency RPV depressurization
D.	30 minutes	an emergency RPV depressurization

Proposed Answer: C

Explanation: 18 inches is the Maximum Safe Area Water Level in both the East and West Crescents. With the given conditions, the West and East Crescent area water levels will reach 18 inches in approximately 10 minutes and 30 minutes, respectively. If a primary system were discharging, a Reactor scram is required before either water level reaches 18 inches (~10 minutes) and an emergency RPV depressurization is required if both water levels reach 18 inches (~30 minutes). Since the leak is from a fire header break, this is NOT a primary system discharge. Therefore, EOP-5 requires shutting down the Reactor if both water levels reach 18 inches (~30 minutes).

A. Incorrect – In approximately 10 minutes, only one area water level will be at or above 18 inches. The Reactor shutdown is not required until both area water levels reach 18 inches.
B. Incorrect – In approximately 10 minutes, only one area water level will be at or above 18 inches. The Reactor shutdown is not required until both area water levels reach 18 inches. An emergency RPV depressurization will not be required because the leakage is from a non-primary system.
D. Incorrect – An emergency RPV depressurization will not be required because the leakage is from a non-primary system.

Technical Reference(s): EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11f 1.03 and 1.07

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295009 2.4.31
	Importance Rating	4.1

Low Reactor Water Level

Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: #85

The plant is operating at 100% power with the following:

- Feedwater Level Control is selected to column A.
- Annunciator 09-5-1-28, RX WTR LVL ALARM HI OR LO, alarms.
- Narrow range Reactor water level indication from 06LT-52B indicates downscale on Panel 09-5.
- All other Reactor water level indications are stable in the normal band.

Given the following Technical Specifications (TS):

(1) TS 3.3.2.2, Feedwater and Main Turbine High Water Level Trip Instrumentation
(2) TS 3.3.5.1, Emergency Core Cooling System Instrumentation

Which one of the following describes the condition entry requirements for these Technical Specification based on this failure?

Condition entry is...

A. NOT required for (1) or (2).

B. required for (1), but NOT for (2).

C. required for (2), but NOT for (1).

D. required for both (1) and (2).

Proposed Answer: B

Explanation: The given level transmitter failure prevents one channel of Feedwater and Main Turbine high level trip capability. TS 3.3.2.2 requires all three channels to be operable and condition entry is required with the given failure. The HPCI high level trip is from a different set of narrow range instruments, therefore all required channels are operable and TS 3.3.5.1 entry is NOT required.

A. Incorrect – TS 3.3.2.2 Condition A must be entered.
C. Incorrect – TS 3.3.2.2 Condition A must be entered. No condition entry is required for TS 3.3.5.1.
D. Incorrect – No condition entry is required for TS 3.3.5.1.

Technical Reference(s): SDLP-02B, Technical Specifications 3.3.2.2 and 3.3.5.1

Proposed references to be provided to applicants during examination: Technical Specifications 3.3.2.2 and 3.3.5.1, with allowable value column blocked out

Learning Objective: SDLP-02B 1.18.c and 1.18.f

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	259002 A2.01
	Importance Rating	3.4

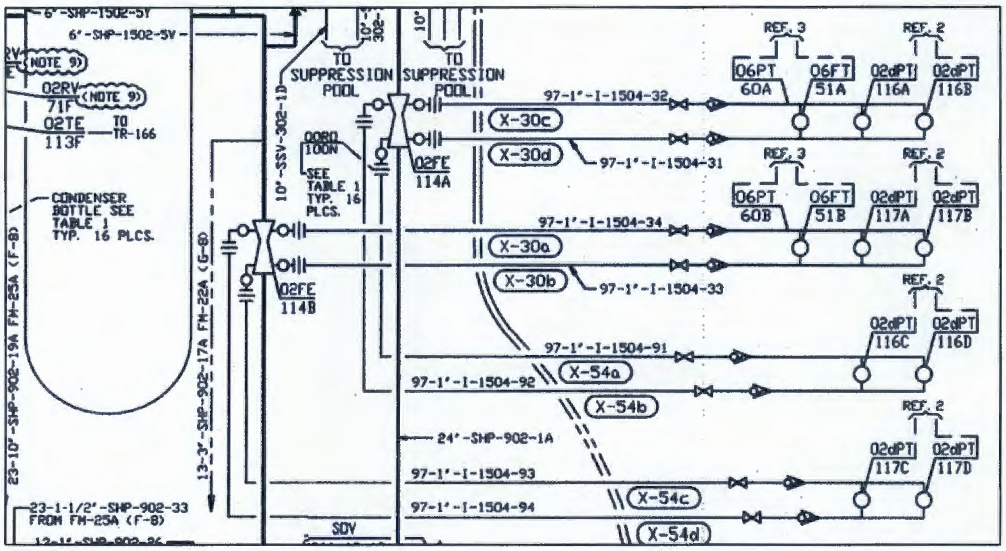
Reactor Water Level Control System

Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of any number of main steam flow inputs

Proposed Question: #86

The plant is operating at 100% power with the following:

- Feedwater level control is operating in three-element control.
- 06FT-51A, the Main Steam Line A flow transmitter that inputs to Feedwater Level Control, fails downscale (shown in a portion of FM-29A below).
- AOP-41, Feedwater Malfunction, is entered.
- Feedwater level control is placed in MAN and Reactor water level is stabilized in the normal band.



Which one of the following describes:

(1) the initial direction of the Reactor water level change due to the steam flow transmitter failure, and

(2) the Technical Specification impact of the steam flow transmitter failure?

(1) Reactor water level...
(2) Technical Specification condition entry is...

A. (1) rises.
(2) required.

B. (1) rises.
(2) NOT required.

C. (1) lowers.
(2) required.

D. (1) lowers.
(2) NOT required.

Proposed Answer: D

Explanation: With 06FT-51A failing downscale, the Feedwater level control system detects less steam leaving the Reactor vessel and anticipates Reactor water level rising. Therefore, the Feedwater level control system lowers Feedwater flow, causing actual Reactor water level to lower. Technical Specification Table 3.3.6.1-1 Function 1.c requires 2 flow instruments per trip system per Main Steam Line while in Modes 1, 2, and 3. Therefore, all four flow instruments for each Main Steam Line that input to PCIS are required. However, the steam flow transmitter to Feedwater level control is separate from the four transmitters to PCIS (02dPT-116A,B,C,D), therefore Technical Specification 3.3.6.1 is still satisfied and condition entry is NOT required. There is no other Technical Specification impacted by loss of this steam flow transmitter.

A. Incorrect – Reactor water level lowers, not rises. Technical Specification 3.3.6.1 condition entry would be required if any of the other Main Steam flow transmitters failed downscale, but this transmitter does NOT required Technical Specification condition entry.
B. Incorrect – Reactor water level lowers, not rises.
C. Incorrect – Technical Specification 3.3.6.1 condition entry would be required if any of the other Main Steam flow transmitters failed downscale, but this transmitter does NOT required Technical Specification condition entry.

Technical Reference(s): OP-2A, FM-29A, Technical Specification 3.3.6.1

Proposed references to be provided to applicants during examination: Technical Specification 3.3.6.1 and Table 3.3.6.1 with the allowable value column blocked out

Learning Objective: SDLP-06 1.10.e

Question Source: Modified Bank – 2010 NRC #41

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	223002 A2.11
	Importance Rating	3.9

PCIS/Nuclear Steam Supply Shutoff

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Standby liquid initiation

Proposed Question: #87

The plant is operating at 100% power with the following:

- I&C and Engineering have determined that a wiring problem will prevent all input from the Standby Liquid Control (SLC) pump control switch to the Reactor Water Cleanup Isolation circuitry from re-positioning upon SLC pump start.
- I&C and Engineering have determined that all other SLC pump control switch wiring is connected properly.

Which one of the following describes the impact of this wiring problem on Technical Specifications (TS)?

A. TS 3.1.7 Conditions A and B must be entered.

B. TS 3.3.6.1 Conditions A and B must be entered.

C. TS 3.1.7 Condition A must be entered. TS Condition B does NOT need to be entered.

D. TS 3.3.6.1 Condition A must be entered. TS Condition B does NOT need to be entered.

Proposed Answer: B

Explanation: With the input from the Standby Liquid Control (SLC) pump control switch to the Reactor Water Cleanup Isolation circuitry not re-positioning upon SLC pump start, TS table 3.3.6.1-1 Function 5.d is NOT met and this isolation function is NOT maintained, requiring TS 3.3.6.1 Conditions A and B to be entered.

A. Incorrect – TS 3.1.7 entry is not required unless the required completion time of TS 3.3.6.1 Condition A or B is not met, in which case TS 3.3.6.1 Condition I may require declaring a SLC subsystem inoperable.
C. Incorrect – TS 3.1.7 entry is not required unless the required completion time of TS 3.3.6.1 Condition A or B is not met, in which case TS 3.3.6.1 Condition I may require declaring a SLC subsystem inoperable.
D. Incorrect – Since all input from the SLC switch to the RWCU isolation logic is not working, TS table 3.3.6.1-1 Function 5.d is not capable of occurring. This results in the need to enter TS 3.3.6.1 Condition B.

Technical Reference(s): Technical Specifications 3.1.7 and 3.3.6.1, ESK-6RE, 1.70-110

Proposed references to be provided to applicants during examination: Technical Specifications 3.1.7 and 3.3.6.1 (allowable value column blocked out)

Learning Objective: SDLP-11 1.18

Question Source: Bank – March 2014 NRC #86

Question History: March 2014 NRC #86

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	262001 2.2.38
	Importance Rating	4.5

AC Electrical Distribution

Knowledge of conditions and limitations in the facility license.

Proposed Question: #88

The plant is operating at 100% power with the following:

- Core Spray pump A has been out of service for maintenance for 24 hours.
- Then, Lines 3 and 4 de-energize.
- All Emergency Diesel Generators start and load their respective buses.

Which one of the following describes the latest time Technical Specifications require entering Mode 3 based on these conditions?

A. In 13 hours

B. In 25 hours

C. In 6 days and 12 hours

D. In 7 days and 12 hours

Proposed Answer: B

Explanation: When Lines 3 and 4 de-energize, both offsite circuits required by LCO 3.8.1 are inoperable. TS 3.8.1 Condition C requires declaring required features inoperable when the redundant required feature is inoperable in 12 hours. Since Core Spray B now has no offsite power supply and its redundant feature is inoperable (Core Spray A), Core Spray B must be declared inoperable at 12 hours. With Core Spray A and B inoperable, TS 3.5.1 Condition H must be entered, which requires entering LCO 3.0.3 immediately. LCO 3.0.3 requires being in Mode 3 13 hours later, for a total of 25 hours.

A. Incorrect – Mode 3 is not required until 25 hours. 13 hours is based on entering LCO 3.0.3 immediately.
C. Incorrect – Mode 3 is required in 25 hours. 6 days plus 12 hours is based on TS 3.5.1 Conditions A and B, which would also be applicable, but not the most restrictive.
D. Incorrect – Mode 3 is required in 25 hours. 7 days plus 12 hours is based on TS 3.8.1 Conditions C and F, which would also be applicable, but not the most restrictive.

Technical Reference(s): Technical Specifications 3.0.3, 3.5.1, and 3.8.1

Proposed references to be provided to applicants during examination: Technical Specifications 3.0.3, 3.5.1, and 3.8.1, with TS 3.8.1 LCO and applicability blocked out.

Learning Objective: SDLP-71D 1.18

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	215003 2.1.25
	Importance Rating	4.2

IRM

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

Proposed Question: #89

A plant startup is in progress with the following:

- All control rods are inserted.
- Reactor coolant temperature is 110°F and stable.
- The Reactor Mode Switch has just been taken to Startup/Hot Standby.
- IRM A fails downscale.

Which one of the following describes the impact of this failure on Technical Specification 3.3.1.1, RPS Instrumentation?

Technical Specification 3.3.1.1 Condition entry is...

A. NOT required because enough IRMs remain operable.

B. NOT required because of the current plant operating mode.

C. required. If the Required Action and associated Completion Time is NOT met, then Condition G will apply.

D. required. If the Required Action and associated Completion Time is NOT met, then Condition H will apply.

Proposed Answer: A

Explanation: With the Reactor Mode Switch in Startup/Hot Standby, the plant is in Mode 2, even with no control rods yet withdrawn. Therefore, the IRM upscale scram is required to be operable. However, only three IRMs are required per trip channel. IRMs C, E, and G maintain the required number of operable IRMs in RPS A. Therefore no TS 3.3.1.1 Condition entry is required.

B. Incorrect – Condition entry is NOT required because sufficient IRMs remain operable. The current plant operating mode (2) would require condition entry if another IRM were inoperable.
C. Incorrect – Condition entry is NOT required because sufficient IRMs remain operable. Condition G would be the correct secondary Condition if another IRM were inoperable.
D. Incorrect – Condition entry is NOT required because sufficient IRMs remain operable. Condition H would be the correct secondary Condition if another IRM were inoperable and the plant was still in Mode 5.

Technical Reference(s): Technical Specification Table 1.1-1, Technical Specification 3.3.1.1 and Table 3.3.1.1-1

Proposed references to be provided to applicants during examination: Technical Specification 3.3.1.1 (with SR 3.3.1.1.2 blocked out) and Table 3.3.1.1-1 (IRM Section only)

Learning Objective: SDLP-07B 1.18

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	215005 2.1.7
	Importance Rating	4.7

APRM / LPRM

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

Proposed Question: #90

The plant is operating near rated power with the following:

- ST-5D, APRM Calibration, has been performed to meet the requirements of Technical Specification 3.3.1.1.
- The procedure is complete up to the Management SRO Review.
- Core Thermal Power is 2510.6 MWth.
- Final APRM readings are as follows:

APRM	Indication
A	97.5%
B	99.0%
C	100.5%
D	99.5%
E	101.5%
F	99.5%

Which one of the following describes the operability of the APRMs based on these indications, in accordance with Technical Specifications?

A. All APRMs are operable.

B. One APRM is inoperable. All other APRMs are operable.

C. Two APRMs are inoperable. All other APRMs are operable.

D. Three APRMs are inoperable. All other APRMs are operable.

Proposed Answer: B

Explanation: Technical Specification Surveillance Requirement 3.3.1.1.2 and ST-5D require APRMs to indicate ±2% of the Reactor power based on the core thermal power heat balance. Rated core thermal power is 2536 MWth, therefore the plant is currently operating at 99% power (2510.6/2536 x 100%). This makes the allowable band for APRM readings 97-101%. APRMs B, D, and F are in the middle of this band. APRMs A and C are within the band, but close to the limits. APRM E is indicating higher than this allowable band. Therefore, one APRM is inoperable and all the rest are operable.

A. Incorrect – APRM E is indicating 0.5% higher than allowed for operability.
C. Incorrect – Only APRM E is out of tolerance. APRMs A and C are near the limit, but still in tolerance.
D. Incorrect – Only APRM E is out of tolerance. APRMs A and C are near the limit, but still in tolerance.

Technical Reference(s): Technical Specification Surveillance Requirement 3.3.1.1.2, ST-5D

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07C 1.16

Question Source: Modified Bank – NMP1 2008 NRC #20

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	202002 A2.09
	Importance Rating	3.3

Recirculation Flow Control

Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Recirculation flow mismatch: Plant-Specific

Proposed Question: #91

The plant is operating at 70% power with the following:

- Recirculation pump A spuriously runs back to the 44% limiter.
- Attempts to clear the runback have failed.
- Attempts to lower Recirculation loop B flow have failed.
- The speed difference between the Recirculation MG sets is approximately 30%.

Which one of the following describes the potential negative impact of these conditions and the required action to be directed, in accordance with AOP-8, Unexpected Change in Core Flow?

Operating with significant speed difference causes potentially damaging…

A. jet pump vibration. Direct shutdown of Recirculation pump A.

B. jet pump vibration. Direct shutdown of Recirculation pump B.

C. core flow oscillations. Direct shutdown of Recirculation pump A.

D. core flow oscillations. Direct shutdown of Recirculation pump B.

Deleted, per revised key.

Proposed Answer: B

Explanation: A caution in AOP-8 states, "Operating with significant speed difference between recirculation pumps causes jet pump vibration which can damage jet pump components and supports." AOP-8 section E.7 gives guidance on response to a runback. Step E.7.1 requires shutting down the higher speed Recirculation pump if speed difference cannot be maintained less than 10%. Since neither Recirculation pump A flow can be raised with the locked in runback nor Recirculation pump B flow can be lowered, and speed difference is greater than 10%, the higher speed Recirculation pump B must be shutdown.

A. Incorrect – AOP-8 requires directing shutdown of the higher speed Recirculation pump, which in this case is B.
C. Incorrect – The concern listed in AOP-8 for such a speed difference is jet pump vibration, not core flow oscillation. AOP-8 does direct monitoring for core flow oscillations. AOP-8 requires directing shutdown of the higher speed Recirculation pump, which in this case is B.
D. Incorrect – The concern listed in AOP-8 for such a speed difference is jet pump vibration, not core flow oscillation. AOP-8 does direct monitoring for core flow oscillations.

Technical Reference(s): AOP-8

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H 1.14.a

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	245000 2.4.50
	Importance Rating	4.0

Main Turbine Generator and Auxiliary Systems

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

Proposed Question: #92

A plant startup is in progress with the following:

- The Reactor Mode Switch is taken from STARTUP/HOT STANDBY to RUN.
- Reactor power is approximately 10% and stable.
- The crew is continuing with Reactor power ascension.
- Annunciator 09-5-1-52, TCV FAST CLOSURE & TSV TRIP BYPASSED, is in alarm.

Which one of the following describes the status of this annunciator and the impact on Technical Specification 3.3.1.1, RPS Instrumentation?

Annunciator 09-5-1-52...

A. has failed to clear. Technical Specification 3.3.1.1 condition entry is required.

B. has failed to clear. Technical Specification 3.3.1.1 condition entry is NOT required.

C. will clear when Reactor power is approximately 29%. Failure of the associated trip bypass to clear would require Technical Specification 3.3.1.1 condition entry.

D. will clear when Reactor power is approximately 29%. Failure of the associated trip bypass to clear would NOT require Technical Specification 3.3.1.1 condition entry.

Proposed Answer: C

Explanation: Placing the Reactor Mode Switch in RUN should clear annunciator 09-5-1-12, MSIV CLOSURE TRIP IN BYPASS, but NOT annunciator 09-5-1-52. Annunciator 09-5-1-52 should remain in alarm and clear when Reactor power reaches approximately 29%. Failure of the associated trip bypass to clear with Reactor power above 29% would require entry into Technical Specification 3.3.1.1 Conditions A, B, and C.

A. Incorrect – Annunciator 09-5-1-52 should be in alarm and clear when Reactor power reaches approximately 29%.
B. Incorrect – Annunciator 09-5-1-52 should be in alarm and clear when Reactor power reaches approximately 29%. Failure of the TCV/TSV closure scram bypass to clear would require condition entry for Technical Specification 3.3.1.1.
D. Incorrect – Failure of the TCV/TSV closure scram bypass to clear would require condition entry for Technical Specification 3.3.1.1.

Technical Reference(s): ARP 09-5-1-52, ARP 09-5-1-12, Technical Specifications 3.3.1.1

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94A 1.18

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	2
K/A #	288000 A2.01
Importance Rating	3.4

Plant Ventilation Systems

Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High drywell pressure: Plant-Specific

Proposed Question: #93

The plant has experienced a loss of coolant accident with the following:

- Reactor water level is 180 inches and slowly rising.
- Drywell pressure is 15 psig and slowly lowering.
- Refuel Floor and Reactor Building Exhaust radiation monitors are indicating 85 cpm and stable.
- The offsite radioactivity release rate is at the Site Area Emergency level and slowly rising due to Drywell venting.
- Multiple Reactor Building area radiation monitors are in alarm due to the Drywell venting operation.

Which one of the following describes the status of Reactor Building Ventilation and the associated action required to be directed, in accordance with the Abnormal and Emergency Operating Procedures?

	Status of Reactor Building Ventilation	Action to Be Directed
A.	In service	Isolate Reactor Building Ventilation.
B.	Isolated	Bypass interlocks and restore Reactor Building Ventilation to service.
C.	In service	Bypass interlocks as necessary to maintain Reactor Building Ventilation in service.
D.	Isolated	Verify the Reactor Building Ventilation isolation and maintain the system out of service.

Proposed Answer: B

Explanation: Reactor Building Ventilation is isolated due to Drywell pressure above 2.7 psig. EOP-2, EOP-4, EOP-5, EOP-6, AOP-15, and AOP-39 are all required to be executed in this situation. AOP-15 requires verifying the Reactor Building Ventilation isolation. Since Reactor Building ventilation exhaust radiation is well below 10^4 cpm, EOP-5 requires restarting Reactor Building ventilation, defeating interlocks if necessary. Defeating the high Drywell pressure interlock will be necessary in this situation.

A. Incorrect – Reactor Building Ventilation is already isolated due to high Drywell pressure.
C. Incorrect – Reactor Building Ventilation is already isolated due to high Drywell pressure.
D. Incorrect – Under these conditions, EOP-5 entry is required and a step in EOP-5 directs restoring the system, bypassing interlocks if necessary.

Technical Reference(s): AOP-15, EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-66A 1.05.c.2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.34
	Importance Rating	3.5

Knowledge of primary and secondary plant chemistry limits.

Proposed Question: #94

Technical Specification 3.4.6, RCS Specific Activity, limits the allowable specific activity in the Reactor coolant.

Which one of the following describes the limitation and the basis behind this limitation, in accordance with Technical Specifications?

Limits specific activity of dose equivalent __(1)__ in the Reactor coolant to prevent exceeding 10 CFR limits during a design basis __(2)__.

	(1)	(2)
A.	Cesium 137	Recirc loop rupture
B.	Cesium 137	Main Steam Line break
C.	Iodine 131	Recirc loop rupture
D.	Iodine 131	Main Steam Line break

Proposed Answer: D

Explanation: TS 3.4.6 limits the dose equivalent I-131 specific activity to ensure that 10 CFR dose limits are not exceeding during a Main Steam Line break.

A. Incorrect – Dose equivalent I-131, not Cs-137, is limited by TS 3.4.6. Cs-137 is a significant radioisotope referenced in ODCM. The basis behind the limitation is a Main Steam Line break, not a Recirc loop rupture.
B. Incorrect – Dose equivalent I-131, not Cs-137, is limited by TS 3.4.6. Cs-137 is a significant radioisotope referenced in ODCM.
C. Incorrect – The basis behind the limitation is a Main Steam Line break, not a Recirc loop rupture.

Technical Reference(s): Technical Specification 3.4.6 and Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-ITS02 1.05

Question Source: Modified Bank – March 2012 NRC #94

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.18
	Importance Rating	3.9

Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

Proposed Question: #95

The plant is shutdown for a refueling outage.

Which one of the following describes the minimum requirements for review of the outage risk assessment against actual plant conditions, in accordance with AP-10.09, Outage Risk Assessment?

At least once per...

A. day by a minimum of one (1) individual

B. day by a minimum of two (2) individuals

C. shift by a minimum of one (1) individual

D. shift by a minimum of two (2) individuals

Proposed Answer: D

Explanation: AP-10.09 requires the Safety Shutdown Manager to verify the current plant conditions are in compliance with the outage risk assessment at least once each shift. AP-10.09 also requires a Shift Manager to independently review this verification at least once each shift. Therefore, at least two (2) individuals must perform the review.

A. Incorrect – AP-10.09 requires the review at least once per shift by two (2) individuals.
B. Incorrect – AP-10.09 requires the review at least once per shift.
C. Incorrect – AP-10.09 requires the review by two (2) individuals.

Technical Reference(s): AP-10.09

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP 41.03b

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(6)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.14
	Importance Rating	3.8

Knowledge of radiation or containment hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: #96

The plant is operating at 100% power with the following:

- Fuel movements are in progress to optimize the Spent Fuel Pool configuration for an upcoming outage.

Which one of the following describes a restriction on placement of irradiated fuel bundles in the Spent Fuel Pool storage racks and the reason for this restriction, in accordance with OSP-66.001, Management of Refueling Activities?

No irradiated fuel shall be placed in the...

A. peripheral row of the Spent Fuel Pool due to criticality control concerns.

B. peripheral row of the Spent Fuel Pool due to radiation streaming concerns.

C. first two rows of the storage rack adjacent to the Spent Fuel Pool gates due to criticality control concerns.

D. first two rows of the storage rack adjacent to the Spent Fuel Pool gates due to radiation streaming concerns.

Proposed Answer: D

Explanation: OSP-66.001 step 6.11.4 states, "No irradiated fuel shall be placed in the first two rows of the fuel rack adjacent to the fuel pool gates at any time..." The associated WARNING states that the concern is radiation streaming through the air gap between the SFP gates.

A. Incorrect – Only irradiated fuel with less than 1 year of decay time is not allowed in the peripheral row of the SFP. The concern is radiation streaming through the air gap between the Spent Fuel Pool gates, not criticality control. Criticality control is a general concern in the SFP addressed by design of the storage racks.
B. Incorrect – Only irradiated fuel with less than 1 year of decay time is not allowed in the peripheral row of the SFP.
C. Incorrect – The concern is radiation streaming through the air gap between the Spent Fuel Pool gates, not criticality control. Criticality control is a general concern in the SFP addressed by design of the storage racks.

Technical Reference(s): OSP-66.001

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-08A 1.13.b

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(7)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.35
	Importance Rating	4.0

Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Proposed Question: #97

The plant has scrammed from 100% power with the following:

- The Shift Manager has declared a Site Area Emergency due to a primary system discharging into the Reactor Building.
- Two NPOs must be dispatched to the Reactor Building to attempt manual closure of a valve to stop the discharge.
- All emergency response facilities are fully operational.

Which one of the following describes the required method of dispatching the NPOs, in accordance with EAP-13, Damage Control?

The Shift Manager...

A. directly dispatches the NPOs from the Control Room.

B. coordinates with the Technical Support Center (TSC) to dispatch the NPOs.

C. coordinates with the Operations Support Center (OSC) to dispatch the NPOs.

D. coordinates with the Emergency Operations Facility (EOF) to dispatch the NPOs.

Proposed Answer: C

Explanation: Once all emergency response facilities are activated, dispatching of NPOs is controlled/coordinated through the OSC.

A. Incorrect – Once all emergency response facilities are activated, dispatching of NPOs is controlled/coordinated through the OSC. Prior to full activation, the NPOs would be dispatched directly by the Control Room staff.
B. Incorrect – Once all emergency response facilities are activated, dispatching of NPOs is controlled/coordinated through the OSC. The TSC may be used to come up with strategies that NPOs implement, but is not directly involved with dispatching NPOs.
D. Incorrect – Once all emergency response facilities are activated, dispatching of NPOs is controlled/coordinated through the OSC. The EOF maintains overall oversight of the emergency response, but is not directly involved with dispatching NPOs.

Technical Reference(s): EAP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-12.5.4.2 EO1.48

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.30
	Importance Rating	4.1

Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

Proposed Question: #98

The plant is operating at 100% power with the following:

- A plant transient with multiple system malfunctions occurs.
- Plant conditions result in the Shift Manager declaring a Site Area Emergency (SAE).

Which one of the following describes the time requirement for notifying the County, State, and NRC, in accordance with EAP-1.1, Offsite Notifications?

The County and State must be notified within __(1)__ of the SAE declaration.

The NRC must be notified immediately after notification of the County and State, not to exceed __(2)__ from the SAE declaration.

	(1)	(2)
A.	15 minutes	1 hour
B.	15 minutes	30 minutes
C.	30 minutes	1 hour
D.	30 minutes	30 minutes

Proposed Answer: A

Explanation: The County and State must be notified through transmittal of the Part 1 notification within 15 minutes of the emergency declaration. The NRC must be notified through transmittal of the NRC Event Notification Worksheet immediately after notification of the County and State and not later than 1 hour after the emergency declaration.

B. Incorrect – The NRC notification is required within a maximum of 1 hour, not 30 minutes, such as if the two notifications each were given a 15 minute maximum.
C. Incorrect – The County and State notifications are required within 15 minutes of declaration, not 30 minutes. 30 minutes is the maximum time from when conditions requiring declaration are present in the Control Room (15 minutes max for declaration + 15 minutes max for notification).
D. Incorrect – The County and State notifications are required within 15 minutes of declaration, not 30 minutes. 30 minutes is the maximum time from when conditions requiring declaration are present in the Control Room (15 minutes max for declaration + 15 minutes max for notification). The NRC notification is required within a maximum of 1 hour, not 30 minutes, such as if the two notifications each were given a 15 minute maximum.

Technical Reference(s): EAP-1.1

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-12.5.4.1 EO 4.01

Question Source: Bank – NMP2 2012 NRC #93

Question History: NMP2 2012 NRC #93

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.40
	Importance Rating	4.7

Ability to apply technical specifications for a system.

Proposed Question: #99

The plant is operating at 100% power with the following:

- It is discovered that a surveillance has not been completed on time for a Technical Specification required system.
- The surveillance frequency is 7 days.
- The surveillance was last performed 11 days ago.

Which one of the following describes the status of the system, in accordance with Technical Specifications?

A. The associated LCO must be declared NOT met at this time.

B. Complete the surveillance within a maximum of 24 hours from the time of discovery or then the associated LCO must be declared NOT met.

C. Complete the surveillance within a maximum of 2 days from the time of discovery or then the associated LCO must be declared NOT met.

D. Complete the surveillance within a maximum of 7 days from the time of discovery or then the associated LCO must be declared NOT met.

Proposed Answer: D

Explanation: Surveillance Requirement (SR) 3.0.3 applies given discovery of a missed surveillance after the required frequency has elapsed. The requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency (in this case, 7 days), whichever is greater. Therefore up to 7 days are allowed to perform the missed surveillance before being required to declare the LCO not met.

A. Incorrect – SR 3.0.3 allows a delay time to perform the missed surveillance before being required to declare the LCO not met.
B. Incorrect – SR 3.0.3 allows the longer of 24 hours or the specified frequency (7 days).
C. Incorrect – SR 3.0.3 allows the longer of 24 hours or the specified frequency (7 days). 2 days is based on the grace period (25% of 7 days) allowed by SR 3.0.2.

Technical Reference(s): Technical Specification Surveillance Requirement 3.0.3

Proposed references to be provided to applicants during examination: None

Learning Objective: JLP-OPS-ITS02 1.01

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.5
	Importance Rating	2.9

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: #100

Which one of the following identifies a radiation monitor and associated threshold reading used to define a Potential Loss of the Primary Containment Barrier, in accordance with IAP-2, Classification of Emergency Conditions, Table F-1, Fission Product Barrier Matrix?

A. Drywell radiation; 450 R/hr

B. Drywell radiation; 250,000 R/hr

C. Reactor Building Exhaust radiation; 1,000 cpm

D. Reactor Building Exhaust radiation; 100,000 cpm

Proposed Answer: B

Explanation: IAP-2 Table F-1, Fission Product Barrier Matrix, identifies Drywell radiation > 250,000 R/hr as the threshold value for a Potential Loss of the Primary Containment Barrier.

A. Incorrect – The threshold value is 250,000 R/hr, not 450 R/hr. 450 R/hr is the hi-hi alarm setpoint for the Drywell rad monitors.
C. Incorrect – While Reactor Building Exhaust ventilation is both used in the EAL chart and would likely rise in the event of a Primary Containment failure during a LOCA, it is not provided as an indication of a Primary Containment Barrier failure on IAP-2 Table F-1.
D. Incorrect – While Reactor Building Exhaust ventilation is both used in the EAL chart and would likely rise in the event of a Primary Containment failure during a LOCA, it is not provided as an indication of a Primary Containment Barrier failure on IAP-2 Table F-1.

Technical Reference(s): Hot EAL Chart

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-12.5.4.2 EO 1.03

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments: