KEY
SUSQUEHANNA 2018 SRO NRC WRITTEN EXAM

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Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 2.2.22
	Importance Rating	4.0

Partial or Complete Loss of Forced Core Flow Circulation

Knowledge of limiting conditions for operations and safety limits.

Proposed Question: #1

Unit 1 is initially operating at 80% power.

- A malfunction causes Recirculation pump 1A speed to lower.
- An Operator locks the scoop tube for Recirculation pump 1A.
- Conditions stabilize as follows:
  - Total core flow is 70 Mlbm/hr, steady.
  - Recirculation loop 1A flow is 29 Mlbm/hr, steady.
  - Recirculation loop 1B flow is 41 Mlbm/hr, steady.
  - MCPR is 1.20.

Which one of the following identifies:

(1) whether a safety limit has been violated or NOT, and

(2) whether the loop flow mismatch has exceeded the requirements of Technical Specification (TS) 3.4.1, Recirculation Loops Operating, and OP-164-002, Reactor Recirculation System HMI Operations, or NOT?

	Has a Safety Limit been exceeded?	Has loop flow mismatch exceeded the requirements of TS 3.4.1 / OP-164-002?
A.	No	No
B.	No	Yes
C.	Yes	No
D.	Yes	Yes

Proposed Answer: B

Explanation: Safety Limit 2.1.1.2 requires MCPRS to be ≥1.09 during two Recirculation pump operation. With an actual MCPR of 1.20, this Safety Limit is not violated. No other Safety Limit violations are indicate either. TS 3.4.1 requires loop flow mismatch to be ≤ 10 Mlbm/hr with total core flow <75 Mlbm/hr. Loop flow mismatch is currently 12 Mlbm/hr, therefore the limit in TS 3.4.1 is exceeded.

A. Incorrect – Loop flow mismatch is currently 12 Mlbm/hr, therefore the limit in TS 3.4.1 is exceeded. Plausible if candidate misunderstands the flow mismatch requirement or miscalculates the flow mismatch.
C. Incorrect – MCPR is >1.09, therefore the associated Safety Limit is not exceeded. Plausible because MCPR is below the COLR limit required by TS 3.2.2. Loop flow mismatch is currently 12 Mlbm/hr, therefore the limit in TS 3.4.1 is exceeded. Plausible if candidate misunderstands the flow mismatch requirement or miscalculates the flow mismatch.
D. Incorrect – MCPR is >1.09, therefore the associated Safety Limit is not exceeded. Plausible because MCPR is below the COLR limit required by TS 3.2.2.

Technical Reference(s): TS 2.0, TS 3.4.1, OP-164-002

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-064 RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295003 AK2.04
Importance Rating	3.4

Partial or Complete Loss of AC Power

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following: A.C. electrical loads

Proposed Question: #2

Unit 1 is operating at 60% power with the following:

Time (hhmm)	Condition(s)
0800	• Condensate pump 1A is secured. • Condensate pumps 1B, 1C, and 1D are still operating.
0815	• A spurious Reactor scram occurs. • The Reactor Mode Switch is placed in SHUTDOWN. • Reactor water level reaches a low of -20", then recovers to the normal band. • An Operator secures Condensate pump 1D.
0817	• An Operator resets the Main Generator lockouts.
0819	• Aux Bus 11A de-energizes due to a sustained electrical fault.
0822	• A steam leak develops in the Drywell. • Drywell pressure is 4 psig, up slow.

Which one of the following describes the status of Condensate pumps 1B and 1C at time 0825?

	Condensate Pump 1B	Condensate Pump 1C
A.	Operating	Operating
B.	Operating	NOT operating
C.	NOT operating	Operating
D.	NOT operating	NOT operating

Proposed Answer: B

Explanation: Condensate pumps 1A and 1C are powered from Aux Bus 11A (1A101). Condensate pumps 1B and 1D are powered from Aux Bus 11B (1A102). Therefore, after Aux Bus 11A de-energizes at time 0820, only one Condensate pump remains running (1B). Drywell pressure exceeding 1.72 psig at time 0828 would trip these remaining pumps for load shedding to prevent overloading ESS transformers, except the Main Generator lockouts were previously reset at time 0818.

A. Incorrect – Condensate pump 1C is lost at time 0820 due to the loss of Aux Bus 11A. Plausible that Condensate pump 1C would be powered by a bus other than Aux Bus 11A (such as S/U Bus 10).
C. Incorrect – Condensate pump 1B remains operating. Plausible that either Aux Bus 11A would power the 1A and 1B pumps, or that Condensate pump 1B would trip on ESS load shedding at time 0822. Condensate pump 1C is lost at time 0820 due to the loss of Aux Bus 11A. Plausible that Condensate pump 1C would be powered by a bus other than Aux Bus 11A (such as S/U Bus 10).
D. Incorrect – Condensate pump 1B remains operating. Plausible that either Aux Bus 11A would power the 1A and 1B pumps, or that Condensate pump 1B would trip on ESS load shedding at time 0822.

Technical Reference(s): TM-OP-044, EO-000-102

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-003 RBO-4

Question Source: Modified Bank - LOC28 (2016) NRC #27

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AK1.05
	Importance Rating	3.3

Partial or Complete Loss of DC Power

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Loss of breaker protection

Proposed Question: #3

Unit 1 is operating at 100% power with RHR pump 1D running for Suppression Pool Cooling.

Then, 125 VDC Bus 1D644 de-energizes due to a sustained electrical fault.

Which one of the following describes the impact of this failure on RHR pump 1D?

RHR pump 1D...

A. trips, but can be re-started using the pump's Control Room switch without a manual realignment in the field.

B. remains running, CANNOT be stopped using the pump's Control Room switch, and loses automatic trip capabilities.

C. trips and can be re-started using the pump's Control Room switch, but only after a manual realignment in the field.

D. remains running, CANNOT be stopped using the pump's Control Room switch, but maintains automatic trip capabilities.

Proposed Answer: B

Explanation: 1D644 provides normal control power to the RHR pump 1D breaker. Alternate control power is available, but requires manual action to align. With the loss of control power, the RHR pump 1D breaker remains closed, loses the ability to be opened remotely (from the Control Room), and loses all automatic protective trips.

A. Incorrect – RHR pump 1D breaker remains closed. Plausible that the breaker would have a failsafe design that automatically opened the breaker due to the loss of all protective trips.
C. Incorrect – RHR pump 1D breaker remains closed. Plausible that the breaker would have a failsafe design that automatically opened the breaker due to the loss of all protective trips.
D. Incorrect – RHR pump 1D breaker loses all automatic protective trips. Plausible because some breakers (MCCs) have thermal cutouts that provide overcurrent protection even with a loss of control power. Also plausible that control power for automatic trips would have a backup for an ECCS breaker.

Technical Reference(s): ON-125VDC-101, RHR pump 1D control circuit print

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-002 RBO-6

Question Source: Bank - JAF 14-1 NRC #58

Question History: JAF 14-1 NRC #58

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:
TRH 2/6/18 – Revised answer choices based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 AK2.04
	Importance Rating	3.3

Main Turbine Generator Trip

Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Main generator protection

Proposed Question: #4

A Unit 1 startup is in progress with the following:

- Reactor power is 22%.
- The Main Generator is synchronized to the grid.

Then, the Main Turbine trips due to a thrust bearing wear detector signal.

Which one of the following describes the response of the Reactor and the Main Generator?

	The Reactor…	The Main Generator trips on a…
A.	scrams.	Turbine Stop Valve position signal.
B.	scrams.	anti-motoring / reverse power signal.
C.	does NOT scram.	Turbine Stop Valve position signal.
D.	does NOT scram.	anti-motoring / reverse power signal.

Proposed Answer: D

Explanation: The thrust bearing wear detector signal causes the Main Turbine to trip, rapidly closing Turbine Stop Valves. The Turbine Stop Valve closure scram is currently bypassed due to Reactor power <26%, therefore the Reactor does not scram. The Turbine Stop Valve position signal does not provide an input to the Main Generator trip / lockout circuitry. Instead, the Main Generator trips on reverse power as the Main Turbine coasts down.

A. Incorrect – The Reactor does not scram. Plausible because Turbine Stop Valve position does provide a scram signal, however this signal is currently bypassed due to Reactor power <26%. The Turbine Stop Valve position signal does not provide an input to the Main Generator trip / lockout circuitry. Instead, the Main Generator trips on reverse power as the Main Turbine coasts down. Plausible because the Turbine Stop Valve position signal does provide an input to the scram circuitry. Also plausible because a Main Generator lockout signal does directly trip the Main Turbine.
B. Incorrect – The Reactor does not scram. Plausible because Turbine Stop Valve position does provide a scram signal, however this signal is currently bypassed due to Reactor power <26%.
C. Incorrect – The Turbine Stop Valve position signal does not provide an input to the Main Generator trip / lockout circuitry. Instead, the Main Generator trips on reverse power as the Main Turbine coasts down. Plausible because the Turbine Stop Valve position signal does provide an input to the scram circuitry. Also plausible because a Main Generator lockout signal does directly trip the Main Turbine.

Technical Reference(s): AR-104-E02, AR-105-A01, AR-106-A08, AR-106-E08

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-098 RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 2.1.32
	Importance Rating	3.8

SCRAM

Ability to explain and apply system limits and precautions

Proposed Question: #5

Unit 1 has experienced a Reactor scram with the following:

- All control rods are in.
- Reactor water level is being controlled at +22" by Feedwater in the Level Setdown mode.
- The Reactor scram has NOT yet been reset.

Which one of the following describes the recommended control of the Reactor scram reset, in accordance with ON-SCRAM-101?

Reset the scram...

A. after raising Reactor water level to minimize the time level is below normal.

B. after raising Reactor water level to avoid receiving an additional scram signal.

C. prior to raising Reactor water level to avoid excessive RPV bottom head cooldown.

D. prior to raising Reactor water level to minimize introduction of debris into the Reactor.

Proposed Answer: B

Explanation: ON-SCRAM-101 note A.11 states, "Resetting the scram will result in a change in RPV makeup. Level control above Setpoint Setdown (i.e. +30 in.) is recommended prior to scram reset to avoid a second scram on low RPV level."

A. Incorrect – The reason is to avoid receiving an additional scram signal. Plausible because keeping Reactor water level low is generally undesirable, but just not the reason for the timing of the scram reset.
C. Incorrect – The scram is reset after raising Reactor water level. Plausible because leaving the scram NOT reset while Reactor water level is raised does result in more injection of cold water into the RPV bottom head, which does result in cooldown/stratification issues, and is a reason for wanting to reset the scram expeditiously.
D. Incorrect – The scram is reset after raising Reactor water level. Plausible because leaving the scram NOT reset while Reactor water level is raised does result in more time injecting water through CRDM seals, which has the potential to introduce debris into the Reactor.

Technical Reference(s): ON-SCRAM-101

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AA2.03
	Importance Rating	4.3

Control Room Abandonment

Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Reactor pressure

Proposed Question: #6

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

- A fire occurs requiring Control Room evacuation per ON-CREVAC-101, Control Room Evacuation.
- The Reactor is manually scrammed.
- A cooldown is initiated from the Remote Shutdown Panel at time 0000.
- Reactor pressure responds as follows:

Time (hhmm)	Reactor Pressure (psig)
0000	1025
0100	475

Note: ON-CREVAC-101 Attachment B, Pressure Vs. Temperature for Saturated Steam, is provided on the following page.

Which one of the following describes:

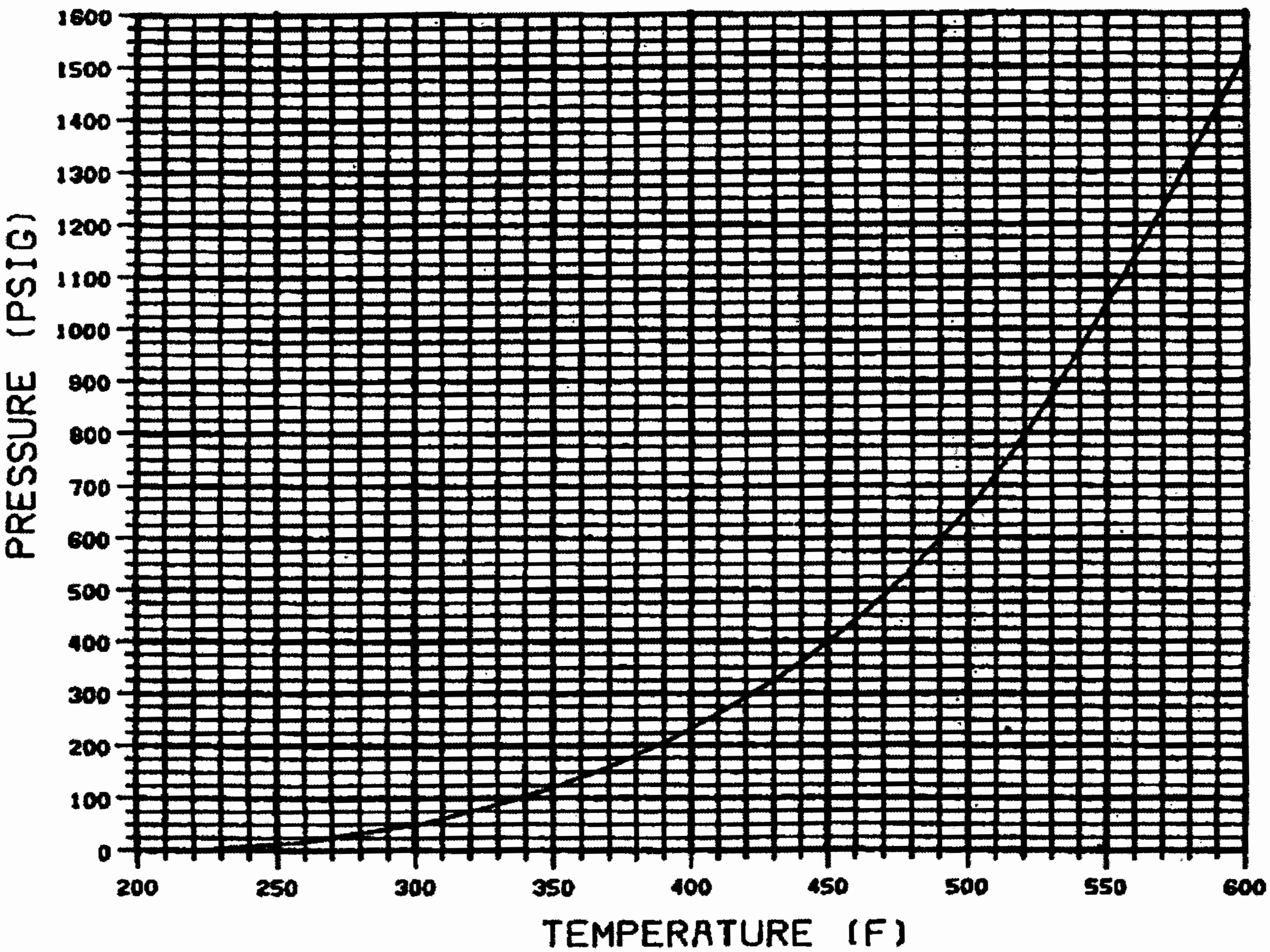
- the status of the cooldown rate in relation to the maximum cooldown rate allowed in GO-100-005, Plant Shutdown to Hot/Cold Shutdown,

  and

- the status of the cooldown rate in accordance with ON-CREVAC-101?

	The cooldown rate is...	In accordance with ON-CREVAC-101, the cooldown rate is...
A.	less than the GO-100-005 maximum rate.	acceptable
B.	less than the GO-100-005 maximum rate.	NOT acceptable
C.	greater than the GO-100-005 maximum rate.	acceptable
D.	greater than the GO-100-005 maximum rate.	NOT acceptable

Pressure Vs Temperature For Saturated Steam



Proposed Answer: A

Explanation: The given Reactor pressures indicate that temperature has lowered from approximately 550°F to approximately 465°F in one hour. This is a cooldown rate of approximately 85°F/hr and is less than 100°F/hr. 100°F/hr is the limit in GO-100-005, therefore this is below that limit. ON-CREVAC-101 assumes that EO-100-102 is entered concurrently, and therefore the cooldown rate limit is 100°F/hr. Therefore, the current cooldown rate is acceptable.

B. Incorrect – The cooldown rate is below the limit in GO-100-005. Plausible because the cooldown rate is close to 100°F/hr and is subject to plotting/calculation error.
C. Incorrect – The cooldown rate is acceptable. Plausible because it would be unacceptable if ON-CREVAC-101 required opening SRVs and leaving them open (rapid cooldown), as is required in some procedures.
D. Incorrect – The cooldown rate is below the limit in GO-100-005. Plausible because the cooldown rate is close to 100°F/hr and is subject to plotting/calculation error. The cooldown rate is acceptable. Plausible because it would be unacceptable if ON-CREVAC-101 required opening SRVs and leaving them open (rapid cooldown), as is required in some procedures.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-EO102, Objective 2

Question Source: Bank – LOC26R (1/2015) NRC #52

Question History: LOC26R (1/2015) NRC #52

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:
TRH 2/7/18 – Revised answer choices based on NRC comment.
TRH 2/22/18 – Revised explanations based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AK1.01
	Importance Rating	3.5

Partial or Complete Loss of CCW

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations

Proposed Question: #7

Unit 1 is operating at 50% power when the following occurs:

- RBCCW temperature control is malfunctioning.
- RBCCW supply temperature is 92°F, up slow.
- Initial attempts to manually control RBCCW temperature from the Control Room have been unsuccessful.
- Operators in the field are attempting further actions to restore RBCCW temperature.

Which one of the following identifies the first system load that should be secured, if necessary, to lower heat load on RBCCW, in accordance with ON-RBCCW-101, Loss of Reactor Building Closed Cooling Water?

A. Fuel Pool Cooling and Cleanup Heat Exchangers

B. A Reactor Recirculation Pump

C. Reactor Water Cleanup

D. Offgas Chillers

Proposed Answer: C

Explanation: ON-RBCCW-101 gives guidance for first removing RWCU, since it is a major heat load and operation can continue for a significant period of time without RWCU in-service.

A. Incorrect – ON-RBCCW-101 gives guidance for first removing RWCU. Plausible because Fuel Pool Cooling and Cleanup Heat Exchangers could be removed from service for a significant amount of time without adverse consequence and would reduce heat load on Service Water, which would preserve capacity for RBCCW.
B. Incorrect – ON-RBCCW-101 gives guidance for first removing RWCU. Plausible because Reactor power is at 50%, so a Recirculation pump could be removed from service and would significantly lower the heat load on RBCCW. Additionally, the procedure does provide guidance for removing a Recirculation pump from service if temperatures exceed limits.
D. Incorrect – ON-RBCCW-101 gives guidance for first removing RWCU. Plausible because Offgas Chillers could be removed from service for a period of time without significant adverse consequence and would reduce heat load on RBCCW.

Technical Reference(s): ON-RBCCW-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-014 RBO-7

Question Source: Bank – LOC26R (1/2015) NRC #41

Question History: LOC26R (1/2015) NRC #41

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AA1.04
	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: Service air isolations valves: Plant-Specific

Proposed Question: #8

Unit 1 is operating at 100% power when the following occurs:

- An air leak occurs on the header downstream of the Instrument Air receivers.
- Instrument Air header pressure lowers.
- PCV-12560, Service Air Crosstie to Instrument Air, automatically opens.
- An operator in the field isolates the air leak by closing a manual valve.
- Instrument Air header pressure returns to the normal band.

Which one of the following describes the resulting operation of PCV-12560?

PCV-12560...

A. automatically re-closes.

B. requires manual action from the Control Room to re-close.

C. requires manual action on Turbine Building elevation 676' to re-close.

D. requires manual action on Turbine Building elevation 729' to re-close.

Proposed Answer: A

Explanation: PCV-12560 automatically opens when pressure lowers to approximately 95 psig and then automatically re-closes when pressure returns to the normal band.

B. Incorrect – PCV-12560 automatically re-closes when pressure returns to the normal band. Plausible because other cross-ties require manual action. Also plausible because some air system components have controls in the Control Room.
C. Incorrect – PCV-12560 automatically re-closes when pressure returns to the normal band. Plausible because other cross-ties require manual action. Also plausible because most Instrument and Service Air components are located on Turbine Building elevation 676'.
D. Incorrect – PCV-12560 automatically re-closes when pressure returns to the normal band. Plausible because other cross-ties require manual action. Also plausible because the cross-tie between CIG and Instrument Air is manually performed on Turbine Building elevation 729'.

Technical Reference(s): OP-118-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-019 RBO-3

Question Source: Modified Bank – LOC25 (2013) Cert #21

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AA1.03
	Importance Rating	3.1

Loss of Shutdown Cooling

Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING: Component cooling water systems: Plant-Specific

Proposed Question: #9

A Unit 1 shutdown is in progress with the following:

- RHR loop A has been placed in Shutdown Cooling.
- RHR pump 1A is running.
- RHRSW pump 1A is running.
- Reactor pressure is 39 psig, steady.

Then, RHRSW pump 1A trips on overcurrent.
Reactor pressure is rising slowly.

Which one of the following describes the impact of this pump trip?

A. RHR loop A Shutdown Cooling lineup automatically isolates based on a signal from RHRSW pump 1A breaker position.

B. RHR loop A remains in the Shutdown Cooling lineup, but RHRSW flow to the heat exchanger CANNOT be restored.

C. RHR loop A remains in the Shutdown Cooling lineup and RHRSW flow to the heat exchanger can be restored using RHRSW pump 2A.

D. RHR loop A remains in the Shutdown Cooling lineup and RHRSW flow to the heat exchanger can be restored using RHRSW pump 1B.

Proposed Answer: C

Explanation: When RHRSW pump 1A trips, cooling water flow to RHR heat exchanger 1A is lost. AR-109-G02 and OP-116-001 provide guidance for aligning RHRSW pump 2A to supply cooling water to the Unit 1 RHR loop A heat exchanger.

A. Incorrect – RHR loop A remains running without cooling water to the heat exchanger. Plausible because RHR loop A will isolate if Reactor pressure reaches 98 psig due to loss of cooling, but does not automatically isolate based on trip of RHRSW pump 1A in anticipation of this temperature/pressure rise.
B. Incorrect – RHR loop A does remain in the SDC lineup, but RHRSW flow can be restored to the heat exchanger using RHRSW pump 2A. Plausible because the ability to cross-connect RHRSW pumps/loops is limited.
D. Incorrect – It is RHRSW pump 2A, not 1B, that can be cross-connected. Plausible that the cross-connect would be on the same unit, not on the opposite unit.

Technical Reference(s): OP-149-002, TM-OP-016, TM-OP-049, OP-116-001, AR-109-G02

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-016 RBO-03

Question Source: Bank – LOC26R (1/2015) NRC #11

Question History: LOC26R (1/2015) NRC #11

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:
TRH 2/6/18 – Changed "of this loss" to "of this pump trip" in question based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 AK2.02
	Importance Rating	2.9

Refueling Accidents

Knowledge of the interrelations between REFUELING ACCIDENTS and the following: Fuel pool cooling and cleanup system

Proposed Question: #10

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

- Fuel movement is in progress in the Unit 1 Spent Fuel Pool in preparation for a refueling outage.
- An irradiated fuel assembly has just been loaded on the Refuel Bridge main hoist and raised to the full up position.
- Fuel Pools are NOT cross-tied.

Then, a seismic event results in the following:

- The Refuel Bridge main hoist CANNOT be moved.
- The common discharge line from the Unit 1 Fuel Pool Cooling pumps completely ruptures.

Which one of the following describes the Unit 1 Spent Fuel Pool water level response to this event and the availability of Spent Fuel Pool makeup?

Spent Fuel Pool water level ___(1)___.

The normal method of Fuel Pool makeup ___(2)___ capable of adding water to the Spent Fuel Pool.

	(1)	(2)
A.	maintains the fuel assembly fully covered	is
B.	maintains the fuel assembly fully covered	is NOT
C.	lowers and partially uncovers the fuel assembly	is
D.	lowers and partially uncovers the fuel assembly	is NOT

Proposed Answer: B

Explanation: A complete rupture of the FPC pumps common discharge line results in loss of water circulation back to the SFP. This will result in SFP water level dropping about 2" due to the height of the weirs. However, siphoning of SFP water back through the rupture is prevented by vacuum breakers installed in the discharge piping. The design of the Refuel Bridge main hoist also ensures this SFP water level will maintain the elevated fuel assembly covered with water. The normal source of makeup water to FPC is from Condensate Transfer to the Skimmer Surge tanks. This makeup source is still available, but will not result in adding water to the Spent Fuel Pool because of the location of the leak.

Note: The question meets the K/A by presenting a refueling accident (breach of Fuel Pool piping during irradiated fuel movement that results in degraded Fuel Pool makeup capability) and requiring knowledge of how this refueling accident relates to maintaining a suspended fuel assembly covered and Fuel Pool makeup capability.

A. Incorrect – The normal source of makeup water to FPC is from Condensate Transfer to the Skimmer Surge tanks. This makeup source is still available, but will not result in adding water to the Spent Fuel Pool because of the location of the leak. Plausible that the normal makeup source would add water directly to the SFP (exactly because of this scenario).
C. Incorrect – SFP water level will lower about 2" to the top of the weirs, which will still maintain the fuel assembly fully covered with water. Plausible because the fuel assembly is raised. Also plausible because the leak location would siphon water from the SFP if not for vacuum breakers installed in the lines. Also plausible because if the FPC pumps took suction directly from the FPC, they could still pump it down. The normal source of makeup water to FPC is from Condensate Transfer to the Skimmer Surge tanks. This makeup source is still available, but will not result in adding water to the Spent Fuel Pool because of the location of the leak. Plausible that the normal makeup source would add water directly to the SFP (exactly because of this scenario).
D. Incorrect – SFP water level will lower about 2" to the top of the weirs, which will still maintain the fuel assembly fully covered with water. Plausible because the fuel assembly is raised. Also plausible because the leak location would siphon water from the SFP if not for vacuum breakers installed in the lines. Also plausible because if the FPC pumps took suction directly from the FPC, they could still pump it down.

Technical Reference(s): M-153 Sh 1, OP-135-001, USAR

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-035 RBO-3

Question Source: Bank – NMP1 2013 NRC #10

Question History: NMP1 2013 NRC #10

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(13)

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

High Drywell Pressure

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

Proposed Question: #11

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

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

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

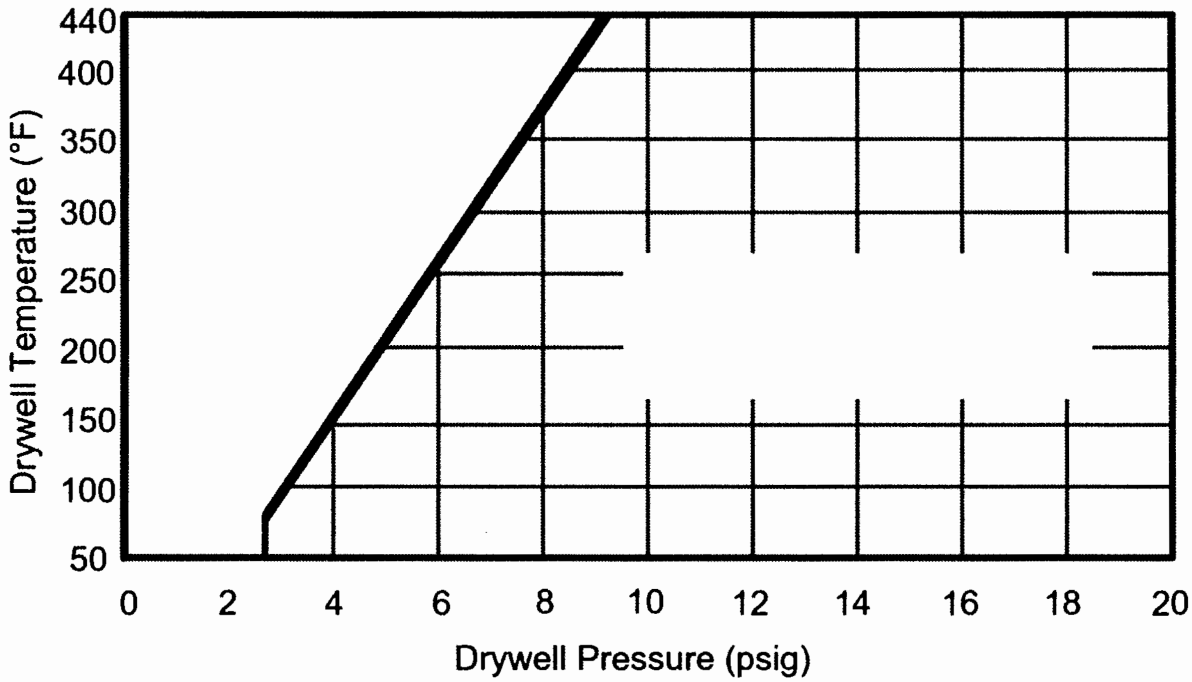
Which one of the following describes:

(1) whether Drywell spray is allowed or NOT, and

(2) the associated reason, in accordance with EO-100-103?

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

FIG. 6 DSIL
DRYWELL SPRAY INITIATION LIMIT



Proposed Answer: D

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

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-7

Question Source: Bank – LOC28R (2017) NRC #42

Question History: LOC28R (2017) NRC #42

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

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: #12

Unit 1 has experienced a scram with the following:

- MSIVs are closed.
- HPCI automatically started and is the only available injection source.
- HPCI injection has been throttled to 3000 gpm to the Reactor with the controller in AUTO.
- Reactor water level is +16", steady.
- Reactor pressure is 800 psig, up slow.

Which one of the following describes the response of HPCI flow rate if Reactor pressure rises to 1090 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 5000 gpm over a Reactor pressure range of 150 to 1210 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. Plausible because this would be the correct answer if the controller were in MAN.
B. Incorrect – Flow will remain approximately constant. Plausible that 1090 psig would be high enough to result in degraded HPCI flow because this is a Reactor pressure significantly above normal.
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 325 psig change in Reactor pressure. Plausible because rising steam supply pressure would partially offset the rising pump discharge pressure.

Technical Reference(s): HPCI DBD

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-052 RBO-4

Question Source: Bank - JAF 14-2 NRC #46

Question History: JAF 14-2 NRC #46

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EK3.02
	Importance Rating	3.9

Suppression Pool High Water Temperature

Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool cooling

Proposed Question: #13

Unit 1 has experienced a transient with the following:

- The Reactor is scrammed.
- MSIVs are closed.
- Reactor water level is being controlled between +13" and +54" with HPCI.
- Reactor pressure is being controlled 800-1050 psig with SRVs.
- Suppression Pool water temperature is 85°F, up slow.
- The Unit Supervisor has directed placing one loop of RHR in the Suppression Pool Cooling mode per OP-149-005, RHR Suppression Pool Cooling.

Which one of the following identifies the preferred loop of RHR to be placed in the Suppression Pool Cooling mode and the reason why, in accordance with OP-149-005?

The preferred loop to be placed in the Suppression Pool Cooling mode is RHR loop...

A. A due to the location of the RHR loop A suction.

B. B due to the location of the RHR loop B suction.

C. A due to greater heat rejection capacity of the RHR loop A heat exchanger.

D. B due to greater heat rejection capacity of the RHR loop B heat exchanger.

Proposed Answer: B

Explanation: OP-149-005 identifies RHR Loop B as the preferred loop when HPCI is in operation due to location of HPCI exhaust near the RHR Loop B suction.

A. Incorrect – RHR loop B is the preferred loop for Suppression Pool Cooling with HPCI in operation. Plausible because RHR loop A is the preferred loop for the RHR Fuel Pool Cooling mode.
C. Incorrect – RHR loop B is the preferred loop for Suppression Pool Cooling with HPCI in operation. Plausible because RHR loop A is the preferred loop for the RHR Fuel Pool Cooling mode. The reason is based on the location of the RHR loop B suction relative to the HPCI exhaust. Plausible because differences in heat exchanger capacity would also be a valid basis for preferring one loop over the other.
D. Incorrect – The reason is based on the location of the RHR loop B suction relative to the HPCI exhaust. Plausible because differences in heat exchanger capacity would also be a valid basis for preferring one loop over the other.

Technical Reference(s): OP-149-005

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-7

Question Source: Modified Bank - LOC26R (1/2015) NRC #39

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295028 EA2.06
	Importance Rating	3.4

High Drywell Temperature

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

Proposed Question: #14

Unit 1 is operating at 50% power when the following occurs:

- A small steam leak develops in the Drywell.
- The Reactor Mode Switch is placed in SHUTDOWN.
- Reactor water level is +35", stable, with Feedwater injecting.
- Reactor pressure is 920 psig, stable, on Turbine Bypass Valves.
- Drywell average temperature is 180°F, stable.
- Drywell pressure is 2.1 psig, stable.
- Suppression Pool water temperature is 80°F, stable.
- Suppression Chamber air temperature is 176°F, up slow.
- Suppression Chamber pressure 0.5 psig, up slow.

Which one of the following describes the Primary Containment response?

The Primary Containment response...

A. is per design for a small steam leak in the Drywell.

B. indicates a possible pair of stuck open SRV tail pipe vacuum relief valves.

C. indicates a possible pair of stuck open Suppression Chamber to Drywell vacuum breakers.

D. indicates a possible crack in the wall between the Suppression Chamber and the Reactor Building.

Proposed Answer: C

Explanation: A small steam leak in the Drywell causes Drywell pressure and temperature to rise. Eventually, Drywell pressure will rise high enough to relieve through the downcomers and the Suppression Pool water volume into the Suppression Chamber air space. This results in condensation of the relieved steam/gas mixture, such that Suppression Chamber pressure stays below Drywell pressure and Suppression Chamber air temperature is much lower than Drywell air temperature. The given data shows matched pressures and temperatures, indicating excessive bypass leakage between the Drywell and Suppression Chamber. This indicates a possible stuck open Suppression Chamber to Drywell vacuum breaker.

A. Incorrect – Suppression Chamber pressure and air temperature are too high. Plausible because all parameters are still within design limits. Also plausible if the Primary Containment response to a small steam leak is not fully understood.
B. Incorrect – A stuck open SRV tail pipe vacuum relief valve would only cause high Drywell pressure and temperature if SRVs had lifted. There is no evidence in the stem that SRVs have lifted. Plausible because this would be correct if Reactor pressure control were on SRVs.
D. Incorrect – A crack in the wall between the Suppression Chamber and the Reactor Building would result in lower Suppression Chamber pressure. Plausible because a crack in another location (the floor between the Suppression Chamber and Drywell) would be correct.

Technical Reference(s): AR-111-E04, Simulator data / USAR, Containment DBD

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-3

Question Source: Modified Bank – LOC26R (1/2015) NRC #78

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:
TRH 2/6/18 – Lowered SP pressure and temperature slightly below Drywell values, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 EA1.01
	Importance Rating	3.6

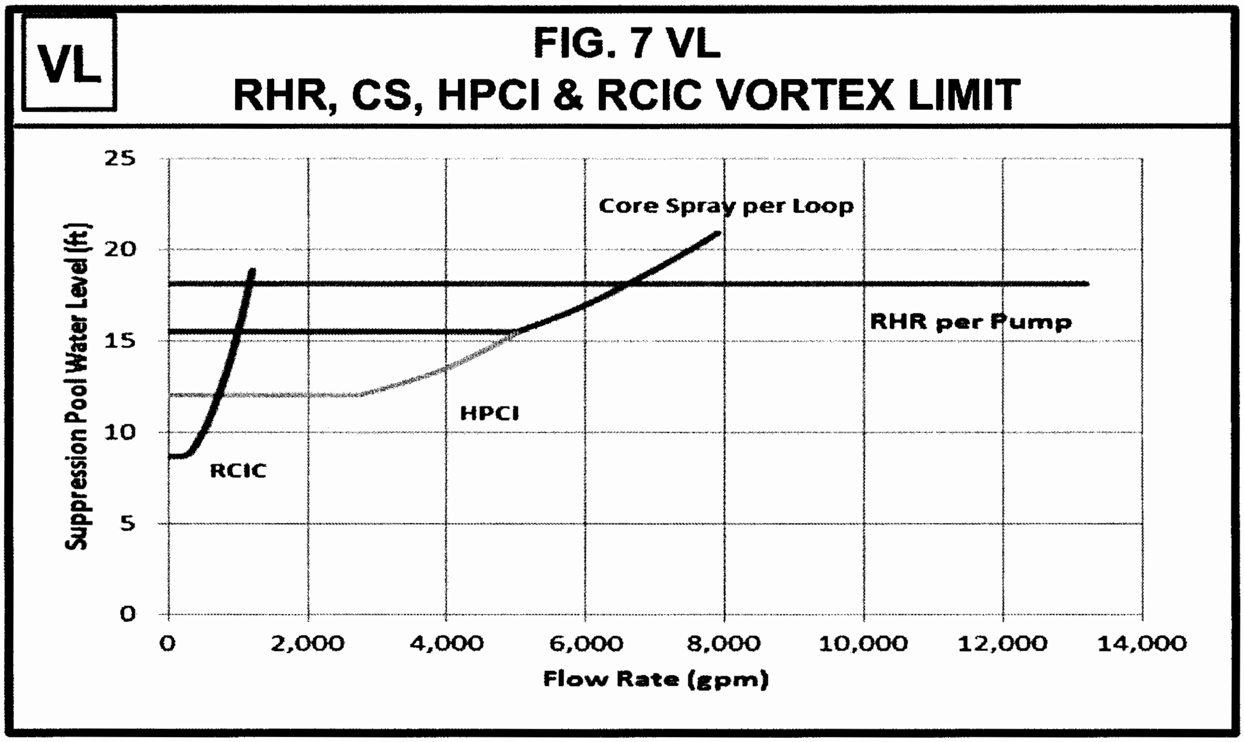
Low Suppression Pool Water Level

Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: ECCS systems (NPSH considerations): Plant-Specific

Proposed Question: #15

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

- Reactor water level is -150", up slow.
- Suppression Pool water level is 17', steady.
- RHR loop A flow rate is 8000 gpm, steady, with both pumps running.
- Core Spray loop A flow rate is 5000 gpm, steady, with both pumps running.



Which one of the following describes the status of Core Spray Loop A and RHR Loop A with respect to the Vortex Limit, in accordance with EO-000-103, Primary Containment Control?

	Vortex Limit for Core Spray Loop A	Vortex Limit for RHR Loop A
A.	SAT	SAT
B.	SAT	UNSAT
C.	UNSAT	SAT
D.	UNSAT	UNSAT

Proposed Answer: B

Explanation: Core Spray loop A flow is above the Vortex Limit curve for the given Suppression Pool water level, therefore it is SAT. RHR loop A flow is below the Vortex Limit curve for the given Suppression Pool water level, therefore it is UNSAT.

A. Incorrect – RHR loop A flow is below the Vortex Limit curve for the given Suppression Pool water level, therefore it is UNSAT. Plausible because this would be correct if Suppression Pool water level were slightly higher (~18-18.5'). Also plausible if wrong curve is used, operating point is mis-plotted, or if SAT area of graph is misunderstood.
C. Incorrect – Core Spray loop A flow is above the Vortex Limit curve for the given Suppression Pool water level, therefore it is SAT. Plausible because this would be correct if Suppression Pool water level were lower (~15-15.5') or if Core Spray flow were higher (~6000 gpm). Also plausible if wrong curve is used, operating point is mis-plotted, or if SAT area of graph is misunderstood. RHR loop A flow is below the Vortex Limit curve for the given Suppression Pool water level, therefore it is UNSAT. Plausible because this would be correct if Suppression Pool water level were slightly higher (~18-18.5'). Also plausible if wrong curve is used, operating point is mis-plotted, or if SAT area of graph is misunderstood.
D. Incorrect – Core Spray loop A flow is above the Vortex Limit curve for the given Suppression Pool water level, therefore it is SAT. Plausible because this would be correct if Suppression Pool water level were lower (~15-15.5') or if Core Spray flow were higher (~6000 gpm). Also plausible if wrong curve is used, operating point is mis-plotted, or if SAT area of graph is misunderstood.

Technical Reference(s): EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-051 RBO-7

Question Source: Bank – LOC25 (2013) Cert #42

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EA2.03
	Importance Rating	4.2

Reactor Low Water Level

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

Proposed Question: #16

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

- Reactor water level is -110", down slow.
- Reactor pressure is 400 psig, down slow.
- Drywell pressure is 7.5 psig, up slow.

Which one of the following describes the response of RHR?

RHR...

A. is currently injecting to the Reactor.

B. remains in a standby lineup until Reactor water level reaches -129".

C. pumps have automatically started, but will NOT inject until Reactor water level reaches -129".

D. pumps have automatically started, but will NOT inject until Reactor pressure lowers further.

Proposed Answer: D

Explanation: Reactor water level has not yet reached the LPCI initiation setpoint of -129". However, LPCI also initiates on the combination of high Drywell pressure (>1.72 psig) and low Reactor pressure (<420 psig). With both of these parameters met, RHR pumps have started. With Reactor pressure <420 psig, the LPCI injection valves have also opened. RHR is not yet injecting to the Reactor because Reactor pressure is above the shutoff head of the pumps. Once Reactor pressure lowers below approximately 300 psig, RHR will begin to inject, regardless of current Reactor water level.

A. Incorrect – RHR is not yet injecting to the Reactor. Plausible because RHR is currently running with the LPCI injection valves open.
B. Incorrect – LPCI has initiated on the combination of high Drywell pressure (>1.72 psig) and low Reactor pressure (<420 psig). Plausible because Reactor water level is still above the LPCI initiation setpoint of -129".
C. Incorrect – LPCI can and will inject even if Reactor water level stays above -129". Also LPCI will not inject at -129" unless Reactor pressure also lowers enough. Plausible because -129" is part of the LPCI initiation logic.

Technical Reference(s): OP-149-001, RHR DBD

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-4

Question Source: Bank – LOC26R (1/2015) NRC #17

Question History: LOC26R (1/2015) NRC #17

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK2.09
	Importance Rating	4.0

Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: Reactor water level

Proposed Question: #17

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

- Multiple control rods are stuck fully withdrawn.
- Reactor water level was intentionally lowered.
- Reactor power is now downscale on IRM range 1.
- Reactor water level is -65", up slow, with Feedwater injecting.
- Reactor pressure is 920 psig, steady, with Turbine Bypass Valves controlling.
- Standby Liquid Control (SLC) is injecting.
- Initial SLC tank level was 1900 gallons.
- Current SLC tank level is 900 gallons.

TABLE 19, HSBW Injected	
Initial Tank (Gal.)	Final Tank (Gal.)
2000	1068
1900	968
1800	868
1700	768
1600	668
1500	568
1400	468

Which one of the following describes the ability to:

(1) raise Reactor water level to a band of +13" to +54", and/or

(2) commence a Reactor cooldown, in accordance with EO-100-113, Level/Power Control?

	Raising Reactor water level to a band of +13" to +54" is currently...	Commencing a Reactor cooldown is currently...
A.	allowed.	allowed.
B.	allowed.	NOT allowed.
C.	NOT allowed.	allowed.
D.	NOT allowed.	NOT allowed.

Proposed Answer: B

Explanation: With Reactor water level intentionally lowered and multiple rods still stuck full out, EO-100-113 requires Hot Shutdown Boron Weight (HSBW) to be injected before allowing Reactor water level to be raised back to the normal band. HSBW has been injected based on Table 19, therefore Reactor water level restoration is allowed. With multiple rods still stuck full out and boron being injected, EO-100-113 requires Cold Shutdown Boron Weight (CSBW) (1650 gallons) to be injected before allowing a Reactor cooldown. CSBW has not yet been injected, therefore a Reactor cooldown is not currently allowed.

A. Incorrect – Reactor cooldown is not currently allowed. Plausible because if no boron had been injected, IRMs being downscale on range 1 would allow a Reactor cooldown. Also plausible because a Reactor cooldown will be allowed once more boron is injected.
C. Incorrect – Raising Reactor water level is currently allowed. Plausible because this would be the correct answer if SLC tank level was slightly higher. Reactor cooldown is not currently allowed. Plausible because if no boron had been injected, IRMs being downscale on range 1 would allow a Reactor cooldown. Also plausible because a Reactor cooldown will be allowed once more boron is injected.
D. Incorrect – Raising Reactor water level is currently allowed. Plausible because this would be the correct answer if SLC tank level was slightly higher.

Technical Reference(s): EO-100-113

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-053 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

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

High Offsite Radioactivity Release Rate

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

Proposed Question: #18

Which one of the following identifies a condition that results in an automatic realignment of CREOASS, and the associated reason for this CREOASS response per the FSAR?

	Condition	Associated Reason – Maintain Control Room...
A.	Reactor water level of -42"	isolated from the outside.
B.	Reactor water level of -42"	pressure higher than outside pressure.
C.	Control Structure outside air intake radiation level of 3 mR/hr	isolated from the outside.
D.	Control Structure outside air intake radiation level of 3 mR/hr	pressure higher than outside pressure.

Proposed Answer: B

Explanation: Reactor water level <-38" results in CREOASS automatically initiating in the Pressurization/Filtration mode, which also isolates normal Control Structure HVAC air intake. One of the reasons for this response is to ensure continued habitability for Control Room operators by keeping Control Room pressure higher than outside pressure.

A. Incorrect – Control Room pressure is maintained higher than outside pressure, but ventilation is not completely isolated from outside. Plausible because it is desired to limit introduction of contamination into the Control Room and normal Control Room ventilation is isolated.
C. Incorrect – Control Structure outside air intake radiation level of 3 mR/hr does not automatically initiate CREOASS in the Pressurization/Filtration mode. Plausible because a radiation level of 5 mr/hr or higher does cause this initiation. Control Room pressure is maintained higher than outside pressure, but ventilation is not completely isolated from outside. Plausible because it is desired to limit introduction of contamination into the Control Room and normal Control Room ventilation is isolated.
D. Incorrect – Control Structure outside air intake radiation level of 3 mR/hr does not automatically initiate CREOASS in the Pressurization/Filtration mode. Plausible because a radiation level of 5 mr/hr or higher does cause this initiation.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-030 RBO-7

Question Source: Modified Bank - LOC28R (2017) NRC #47

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:
TRH 2/6/18 – Re-worded part of question and 2nd half based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AK3.04
	Importance Rating	2.8

Plant Fire On Site

Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: Actions contained in the abnormal procedure for plant fire on site

Proposed Question: #19

Unit 1 is operating at 100% power when the following occurs:

- A fire occurs outside the Control Room.
- ON-013-001, Response to Fire, is being performed.
- The Reactor is scrammed in accordance with ON-SCRAM-101, Reactor Scram.
- The RCIC turbine initiates and CANNOT be overridden or isolated.
- Reactor water level is 56", up fast.

Which one of the following describes the required action and the reason for this action in accordance with ON-013-001?

A. Close the MSIVs to preserve Reactor coolant inventory.

B. Close the MSIVs to prevent damage to the Main Steam Lines.

C. Depressurize the Reactor to prevent damage to the RCIC turbine.

D. Depressurize the Reactor to prevent damage to the SRV tailpipes.

Proposed Answer: C

Explanation: ON-013-001 provides specific guidance to depressurize the Reactor before Reactor water level reaches +118" (Main Steam Lines) if RCIC injects and cannot be either overridden or isolated. The reason for this action is to prevent damage to the RCIC turbine.

A. Incorrect – The MSIVs are not closed. Plausible because this would assist in preserving Reactor coolant inventory from flowing to the Main Condenser or Suppression Pool.
B. Incorrect – The MSIVs are not closed. Plausible because this would prevent flooding the Main Steam Lines.
D. Incorrect – The reason is not to prevent damage to SRV tailpipes (a calculation has been done to determine damage is not expected). Plausible because rising level may result in water flow through the SRVs which would causes abnormal stresses in the tailpipes.

Technical Reference(s): ON-013-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-013A RBO-7

Question Source: Bank - LOC26R (1/2015) NRC #51

Question History: LOC26R (1/2015) NRC #51

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 2.4.8
	Importance Rating	3.8

Generator Voltage and Electric Grid Disturbances

Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Proposed Question: #20

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

- Thunderstorms in the area have caused a grid disturbance.
- ON-GENGRID-101, Main Generator or Grid Disturbance, has just been entered.

Then, a spurious Reactor scram occurs.
The Unit Supervisor enters EO-000-102, RPV Control.

Which one of the following describes the correct procedure implementation?

A. Continue performing ON-GENGRID-101. In the event of a conflict between ON-GENGRID-101 and EO-000-102, EO-000-102 is the overriding procedure.

B. Continue performing ON-GENGRID-101. In the event of a conflict between ON-GENGRID-101 and EO-000-102, ON-GENGRID-101 is the overriding procedure.

C. Exit ON-GENGRID-101. ON-GENGRID-101 is re-entered at the step in progress after exiting EO-000-102.

D. Exit ON-GENGRID-101. ON-GENGRID-101 entry conditions are re-evaluated after exiting EO-000-102.

Proposed Answer: A

Explanation: There is no requirement to exit ONs when EOs are entered. In fact, both procedures are executed concurrently. The EOs are higher-tiered documents than ONs, therefore in the event of a conflict, the EO must be followed.

B. Incorrect – The EOs are higher-tiered documents than ONs, therefore in the event of a conflict, the EO must be followed. Plausible because the ON is event-specific and the EO is more generic and symptom based.
C. Incorrect – There is no requirement to exit ONs when EOs are entered. Plausible because the EO is the higher-tiered document, however the ON is still performed.
D. Incorrect – There is no requirement to exit ONs when EOs are entered. Plausible because the EO is the higher-tiered document, however the ON is still performed.

Technical Reference(s): OP-AD-055

Proposed references to be provided to applicants during examination: None

Learning Objective: SY018

Question Source: Bank – LOC27 (2015) NRC #21

Question History: LOC27 (2015) NRC #21

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 2/6/18 – Changed classification to Bank based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295002 AA2.01
	Importance Rating	2.9

Loss of Main Condenser Vacuum

Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Condenser vacuum/absolute pressure

Proposed Question: #21

Unit 1 is operating at 30% power when the following occurs:

- Main Condenser pressure begins to rise due to air in-leakage.
- The Reactor mode switch is placed in SHUTDOWN.
- No other operator actions are performed.
- Reactor water level reaches a low value of +10" following the scram.
- Main Condenser pressure is 15.0" Hga, stable.

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

Feedwater pumps are...

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

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

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

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

Proposed Answer: C

Explanation: Multiple actuations occur based on rising Main Condenser pressure. At 7.5" Hga, the Main Turbine trips. At 11.8" Hga, the Feed pump turbines trip, therefore Feedwater pumps are unavailable for injection. Main Condenser pressure has not yet degraded to the point where MSIVs close (19.0" Hga) or TBVs trip (22.2" Hga), therefore TBVs remain available for pressure control.

A. Incorrect – Feedwater pumps are not available for injection because Main Condenser pressure is >11.8" Hga. Plausible because Reactor water level remained above the level that would automatically close MSIVs and Condensate remains available for injection through Feedwater valves.
B. Incorrect – Feedwater pumps are not available for injection because Main Condenser pressure is >11.8" Hga. Plausible because Reactor water level remained above the level that would automatically close MSIVs and Condensate remains available for injection through Feedwater valves. TBVs remain available for pressure control. Plausible because if Main Condenser pressure had degraded further (≥19.0" Hga), then they would be unavailable for pressure control due to MSIV closure.
D. Incorrect – TBVs remain available for pressure control. Plausible because if Main Condenser pressure had degraded further (≥19.0" Hga), then they would be unavailable for pressure control due to MSIV closure.

Technical Reference(s): ON-VACUUM-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-083 RBO-4

Question Source: Modified Bank – LOC28R (2017) NRC #32

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295007 AK2.04
	Importance Rating	3.2

High Reactor Pressure

Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: LPCS

Proposed Question: #22

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

- Reactor water level is -130", down slow.
- Reactor pressure is 800 psig, down slow.
- Drywell pressure is 10 psig, up slow.
- Core Spray pump 1A is running with a discharge pressure of 225 psig.
- Core Spray pump 1C is running with a discharge pressure of 290 psig.
- NO other ECCS pumps are running.
- NO high pressure injection sources are available.

Which one of the following describes the resulting status of the ADS valves two minutes later?

Two minutes later, the ADS valves will be ___(1)___ because the ADS logic senses ___(2)___.

	(1)	(2)
A.	open	adequate ECCS discharge pressure
B.	closed	inadequate ECCS discharge pressure
C.	open	an adequate number of closed ECCS pump breakers
D.	closed	an inadequate number of closed ECCS pump breakers

Proposed Answer: A

Explanation: With both Core Spray pumps 1A and 1C >160 psig, Reactor water level <-129", and Drywell pressure >1.72 psig, Div 1 ADS logic is satisfied. Div 2 ADS logic will not be satisfied. However, only one division is needed to open all ADS valves. Therefore, after 2 minutes, all ADS valves will open.

B. Incorrect – ADS valves open. Plausible because no RHR pumps are running, 2 Core Spray pumps are not running, and one of the running Core Spray pumps has abnormally low discharge pressure.
C. Incorrect – The ADS logic senses ECCS discharge pressures, not breaker positions. Plausible because this would be another method to determine the number of running pumps.
D. Incorrect – ADS valves open. Plausible because no RHR pumps are running, 2 Core Spray pumps are not running, and one of the running Core Spray pumps has abnormally low discharge pressure. The ADS logic senses ECCS discharge pressures, not breaker positions. Plausible because this would be another method to determine the number of running pumps.

Technical Reference(s): TM-OP-083E, AR-110-C01, AR-110-C03, E-324 Sheet 12

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-083E RBO-3

Question Source: Modified Bank – LOC27 (2015) NRC #11

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295012 AA1.02
	Importance Rating	3.8

High Drywell Temperature

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

Proposed Question: #23

Unit 1 is operating at 100% power when the following occurs:

- A steam leak develops in the Drywell.
- Drywell pressure is 2.0 psig, up slow.
- Drywell temperature is 155°F, up slow.

Which one of the following describes the status of Drywell cooling?

Drywell cooling fans are…

A. tripped with NO cooling water supplied to the coolers.

B. running with NO cooling water supplied to the coolers.

C. tripped with Reactor Building Chilled Water supplying the coolers.

D. running with Reactor Building Chilled Water supplying the coolers.

Proposed Answer: A

Explanation: The steam leak has caused Drywell pressure and temperature to rise. With Drywell pressure >1.72 psig, the Drywell cooling fans are tripped and RBCW to the cooling coils is isolated.

Note: The question meets the K/A by presenting a situation with elevated Drywell pressure and temperature and requiring knowledge of the response of Drywell cooling. Although it is Drywell pressure that causes the automatic response of Drywell cooling, the question still tests the applicant's ability to operate/monitor Drywell cooling as it relates to Drywell temperature, as the EO-100-103 Drywell temperature leg requires operating/monitoring Drywell cooling in response to the elevated Drywell temperature. Understanding the status of Drywell cooling is inherently important to proper control of the given elevated Drywell temperature.

B. Incorrect – Drywell cooling fans are tripped. Plausible because they are initially running and cooling is desirable to assist in mitigating the small steam leak.
C. Incorrect – RBCW to the coolers is isolated. Plausible because it is initially in service and cooling is desirable to assist in mitigating the small steam leak.
D. Incorrect – Drywell cooling fans are tripped. Plausible because they are initially running and cooling is desirable to assist in mitigating the small steam leak. RBCW to the coolers is isolated. Plausible because it is initially in service and cooling is desirable to assist in mitigating the small steam leak.

Technical Reference(s): OP-160-001, ON-CONTISOL-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-073 RBO-4

Question Source: Modified Bank – LOC23 (2/2011) Cert #62

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:
TRH 2/6/18 – Added K/A match statement.

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	2
K/A #	295013 AK1.04
Importance Rating	2.9

High Suppression Pool Temperature

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

Proposed Question: #24

A steam leak in the Unit 1 Main Steam Tunnel has resulted in the following:

- The Reactor has scrammed.
- The Unit Supervisor has directed Reactor pressure controlled 800-1050 psig using SRVs.

Which one of the following describes how SRVs should be operated, in accordance with the Emergency Operating Procedures?

A. Cycle through different SRVs sequentially, as needed, to verify they are each seated properly.

B. Cycle the same SRV repeatedly, as needed, to prevent thermal cycling of multiple SRV tailpipes.

C. Cycle through different SRVs sequentially, as needed, to distribute the heat load uniformly around the Suppression Pool.

D. Cycle the same SRV repeatedly, as needed, to maintain Suppression Pool heating as close as possible to the Suppression Pool Cooling injection point.

Proposed Answer: C

Explanation: SRV discharges are designed to be equally spaced at points around the Suppression Pool. EO-100-102 contains direction on the use of SRVs for Reactor pressure control based on this design to ensure more equal heat load distribution around the Suppression Pool by opening SRVs in a preferred sequence (alphabetical), rather than just opening the same SRV repeatedly.

Note: The question meets the K/A because it presents a situation that leads to rising Suppression Pool temperature (SRV opening for Reactor pressure control) and tests the operational implication (how to control SRVs and the reason why) of complete condensation (equally distributing heat load around Suppression Pool to avoid localized hot spots which could result in incomplete condensation).

A. Incorrect – The reason for operating SRVs sequentially is to distribute the heat load uniformly around the SP and equalize the number of actuations among the SRVs, NOT to ensure they are all seated. Plausible because cycling a leaking SRV is a method used to attempt to reseat the valve.
B. Incorrect – EO-100-102 contains a note that directs opening SRVs per a pre-defined sequence, vice opening the same SRV multiple times. Plausible because opening a new SRV will cause a larger thermal transient since its tailpipe is colder.
D. Incorrect – EO-100-102 contains a note that directs opening SRVs per a pre-defined sequence, vice opening the same SRV multiple times. Plausible because SP Cooling does discharge to one point in the SP, so this area will be cooled more directly.

Technical Reference(s): EO-100-102

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-EO102, Obj 2

Question Source: Bank - NMP1 2017 NRC #7

Question History: NMP1 2017 NRC #7

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:
TRH 2/6/18 – Added to K/A match statement based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295029 EK3.03
	Importance Rating	3.4

High Suppression Pool Water Level

Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Reactor SCRAM

Proposed Question: #25

Unit 1 is operating at 100% power when the following occurs:

- A malfunction has caused uncontrolled addition of water to the Suppression Pool.
- Suppression Pool water level is 23', up slow.
- Suppression Pool water temperature is 80°F, steady.
- Suppression Pool letdown is NOT available

Which one of the following identifies the threshold that must be used to determine the need for a Reactor scram on high Suppression Pool water level, in accordance with EO-100-103?

A. SRV Tailpipe Level Limit

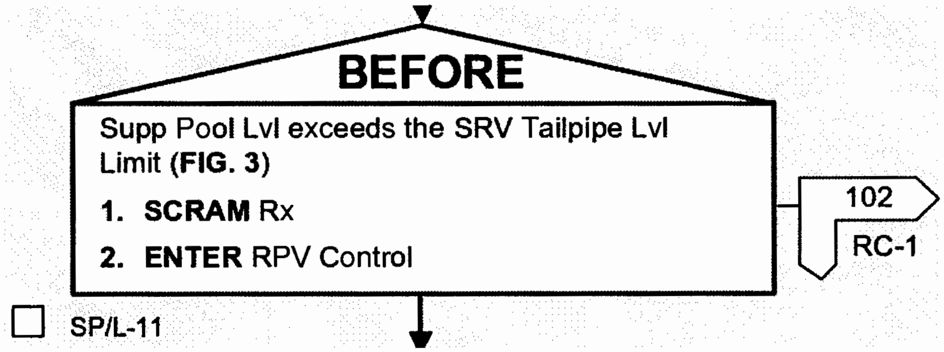
B. Heat Capacity Temperature Limit

C. Suppression Pool water level of 24.0'

D. Suppression Pool water level of 43.0'

Proposed Answer: A

Explanation: EO-100-103 contains the following step to control the Reactor scram based on the SRV Tailpipe Level Limit:



B. Incorrect – EO-100-103 controls the Reactor scram based on the SRV Tailpipe Level Limit. Plausible because EO-100-103 also uses the Heat Capacity Temperature Limit to determine the need for a scram, and this limit is impacted by Suppression Pool water level. However, with no heat addition to the Suppression Pool, HCTL will not be limiting, SRVTPLL will.

C. Incorrect – EO-100-103 controls the Reactor scram based on the SRV Tailpipe Level Limit. Plausible because EO-100-103 uses 24.0' as the entry condition and Technical Specifications use 24.0' as the LCO limit.

D. Incorrect – EO-100-103 controls the Reactor scram based on the SRV Tailpipe Level Limit. Plausible because EO-100-103 uses 43.0' to control securing Drywell sprays.

Technical Reference(s): EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295034 EK1.01
	Importance Rating	3.8

Secondary Containment Ventilation High Radiation

Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Personnel protection

Proposed Question: #26

Both Units are operating at 100% power when the following occurs:

- Fuel movement is in progress in the Unit 1 Spent Fuel Pool in preparation for an upcoming outage.
- A failure of the fuel grapple results in an irradiated fuel bundle being dropped.
- The following annunciators alarm in the Control Room:
  - AR-101-B05, RX BLDG AREA PANEL 1C605 HI RADIATION,
  - AR-112-D01, REFUEL FLOOR WALL EXH MON HI RADIATION.
- The Spent Fuel Pool Criticality area radiation monitor is in alarm.
- Refuel Floor Wall Exhaust radiation indicates 16 mr/hr, steady.
- The Unit Supervisor enters ON-FUEL-001, Fuel Handling System Malfunction.

Which one of the following describes the status of Reactor Building Zone III and the need for a Refuel Floor evacuation, in accordance with ON-FUEL-101?

	Reactor Building Zone III	Refuel Floor Evacuation
A.	Isolated	Required
B.	Isolated	NOT required
C.	NOT isolated	Required
D.	NOT isolated	NOT required

Proposed Answer: C

Explanation: Zone III has not isolated because Refuel Floor Wall Exhaust radiation is <21 mr/hr and the associated hi-hi alarm is not in alarm. ON-FUEL-001 requires a Refuel Floor evacuation based on the occurrence of a fuel handling accident and an unexpected rise in radiation levels while performing fuel moves.

A. Incorrect – Zone III has not isolated. Plausible because Refuel Floor Wall Exhaust radiation is above the hi alarm and the SFP criticality ARM is in alarm. This would be correct if Refuel Floor Wall Exhaust radiation were >21 mr/hr.
B. Incorrect – Zone III has not isolated. Plausible because Refuel Floor Wall Exhaust radiation is above the hi alarm and the SFP criticality ARM is in alarm. This would be correct if Refuel Floor Wall Exhaust radiation were >21 mr/hr. Refuel Floor evacuation is required. Plausible because conditions are not severe enough to result in a Zone III isolation, and ON-FUEL-001 uses Zone III isolation as an example of when to perform evacuation of the entire Reactor Building.
D. Incorrect – Refuel Floor evacuation is required. Plausible because conditions are not severe enough to result in a Zone III isolation, and ON-FUEL-001 uses Zone III isolation as an example of when to perform evacuation of the entire Reactor Building.

Technical Reference(s): AR-112-D01, AR-101-A05, ON-FUEL-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-070 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	500000 2.4.18
	Importance Rating	3.3

High Containment Hydrogen Concentration

Knowledge of the specific bases for EOPs.

Proposed Question: #27

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

Which one of the following describes the gas concentration parameter(s) used to determine the need for venting and the basis for the associated threshold value(s), in accordance with EO-100-103?

	Gas Concentration Parameter(s) Used to Determine the Need for Venting	Basis for Threshold Value(s)
A.	Hydrogen only	Minimum detectable
B.	Hydrogen only	Minimum to support deflagration
C.	Both Hydrogen and Oxygen	Minimum detectable
D.	Both Hydrogen and Oxygen	Minimum to support deflagration

Proposed Answer: A

Explanation: EO-100-103 uses only hydrogen concentration to determine the need for venting in the PC/G leg. The basis for the 1% H_2 concentration threshold is that this is considered the minimum detectable concentration.

B. Incorrect – The basis for the 1% H_2 concentration threshold is that this is considered the minimum detectable concentration. Plausible because the actual negative consequences of high H2 concentration are not possible until H_2 concentration reaches the higher minimum value to support deflagration. Also plausible because earlier revisions of the EOPs used the higher deflagration limit for this threshold.
C. Incorrect – EO-100-103 uses only hydrogen concentration to determine the need for venting in the PC/G leg. Plausible because oxygen must also be present in sufficient concentration for a deflagration to occur and the containment is normally inerted with nitrogen. Also plausible because earlier revisions of the EOPs used both hydrogen and oxygen concentrations to determine the need for venting.
D. Incorrect – EO-100-103 uses only hydrogen concentration to determine the need for venting in the PC/G leg. Plausible because oxygen must also be present in sufficient concentration for a deflagration to occur and the containment is normally inerted with nitrogen. Also plausible because earlier revisions of the EOPs used both hydrogen and oxygen concentrations to determine the need for venting. The basis for the 1% H_2 concentration threshold is that this is considered the minimum detectable concentration. Plausible because the actual negative consequences of high H2 concentration are not possible until H_2 concentration reaches the higher minimum value to support deflagration. Also plausible because earlier revisions of the EOPs used the higher deflagration limit for this threshold.

Technical Reference(s): EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	203000 K3.02
	Importance Rating	3.5

RHR/LPCI: Injection Mode

Knowledge of the effect that a loss or malfunction of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) will have on the following: Suppression pool level

Proposed Question: #28

Unit 1 experiences a small loss of coolant accident with the following:

- Drywell pressure is 4 psig, steady.
- Reactor water level is -50", steady.
- Reactor pressure is 700 psig, steady.
- Suppression Pool water level is 23', steady.

Then, a large leak develops from the discharge of RHR pump A into the RHR pump room.

- Attempts to isolate the leak from the Control Room have failed.
- The watertight door to RHR A pump room A is closed and CANNOT be opened due to a failure in the door opening mechanism.
- Suppression Pool water level is currently 20'.

Which one of the following describes the effect of these failures on Suppression Pool water level, in accordance with EO-100-103, Primary Containment Control?

Suppression Pool water level...

A. remains at the current level due to room size and elevation.

B. lowers, but will stabilize and NOT require an Emergency RPV Depressurization.

C. lowers, will require Emergency RPV Depressurization, but stays above SRV discharges.

D. lowers to below the level of the SRV discharges.

Proposed Answer: B

Explanation: A leak from RHR pump A into the RHR pump room with the water tight door will only lower Suppression Pool water level until the water levels in the pump room and Suppression Pool equalize. For RHR pump room A, this occurs at a Suppression Pool water level of 16'. This is above the level requiring an Emergency RPV Depressurization (12').

A. Incorrect – Suppression Pool level will lower to approximately 16'. Plausible because this would be correct for a Supp Pool leak into the RCIC room.
C. Incorrect – Suppression Pool level will stabilize at approximately 16'. This is above the level requiring Emergency RPV Depressurization. Plausible because this would be correct if level lowered to <12', which could happen if another leak developed in another area, the floor drain isolation valve was opened, or the watertight door failed.
D. Incorrect – Suppression Pool level will stabilize at approximately 16'. This is above the level of the SRV discharges (5'). Plausible because this would be correct if level lowered to <12', which could happen if another leak developed in another area, the floor drain isolation valve was opened, or the watertight door failed.

Technical Reference(s): EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:
TRH – Based on NRC comments: deleted bullet about door seal not leaking, added that current SP level is 19.5', and revised choices for length.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 2.1.27
	Importance Rating	3.9

Shutdown Cooling

Knowledge of system purpose and/or function.

Proposed Question: #29

A cooldown is being performed on Unit 1 with the following:

- RHR loop A is operating the Shutdown Cooling mode.
- HV-151-F048A, RHR HX A SHELL SIDE BYPS, is being throttled to control the cooldown rate.

The Unit Supervisor has directed ***lowering*** the cooldown rate.

Which one of the following describes the correct operation of HV-151-F048A to lower the cooldown rate and the effect on flow?

To lower the cooldown rate, HV-151-F048A must be throttled further ___(1)___ to adjust ___(2)___ flow through the heat exchanger.

	(1)	(2)
A.	open	RHR
B.	open	RHRSW
C.	closed	RHR
D.	closed	RHRSW

Proposed Answer: A

Explanation: To lower the cooldown rate, HV-151-F048A must be throttled further open. This allows more RHR flow to bypass the heat exchanger, which lowers the heat removal rate of the Shutdown Cooling loop.

Note: The question meets the K/A by giving manipulation of a component in the Shutdown Cooling lineup and requiring the candidate to understand overall system function (effect of component manipulation on line flow, relationship of bypass line to heat exchanger, relationship of two heat exchanger flow paths) to determine the end result on the controlled parameter (Reactor coolant temperature).

B. Incorrect – HV-151-F048A adjusts RHR flow, not RHRSW flow, through the heat exchanger. Plausible because some heat exchangers have a bypass line on the cooling water supply and this is viable design for controlling cooling rate.
C. Incorrect – HV-151-F048A must be throttled further open, not closed. Plausible because if the cooldown rate was being controlled by throttling HV-151-F003A or HV-151-F017A, the valve would need to be throttled further closed to lower the cooldown rate.
D. Incorrect – HV-151-F048A must be throttled further open, not closed. Plausible because if the cooldown rate was being controlled by throttling HV-151-F003A or HV-151-F017A, the valve would need to be throttled further closed to lower the cooldown rate. HV-151-F048A adjusts RHR flow, not RHRSW flow, through the heat exchanger. Plausible because some heat exchangers have a bypass line on the cooling water supply and this is viable design for controlling cooling rate.

Technical Reference(s): OP-149-002 Attachment E, M-151

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-03

Question Source: Modified Bank – JAF 14-1 NRC #22

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:
TRH 2/9/18 – Added K/A match statement based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 K3.05
	Importance Rating	2.6

Shutdown Cooling

Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Fuel pool cooling assist: Plant-Specific

Proposed Question: #30

Unit 1 is shutdown with the following:

- Reactor coolant temperature is 170°F, down slow.
- RHR loop B is operating in the Shutdown Cooling mode.
- SDC Hardening is NOT in effect.

RPS Bus A de-energizes due to a sustained electrical fault.

Which one of the following describes the status of Shutdown Cooling / RHR?

Shutdown Cooling...

A. remains in service on RHR loop B.

B. is lost. RHR loop A is available to be placed in Shutdown Cooling mode.

C. is lost. Neither loop of RHR is available to be placed in Shutdown Cooling mode. Only RHR loop B is available to be placed in Fuel Pool Cooling Assist mode.

D. is lost. Neither loop of RHR is available to be placed in Shutdown Cooling mode. Both loops of RHR are available to be placed in Fuel Pool Cooling Assist mode.

Proposed Answer: D

Explanation: Both loops of RHR are unavailable for Shutdown Cooling mode because Shutdown Cooling suction IV F009 closes due to loss of RPS Bus A. This failure does not affect the RHR Fuel Pool Cooling Assist mode, therefore it remains available with either RHR loop.

A. Incorrect – Although loss of RPS Bus A only causes one of the Shutdown Cooling suction IVs to close, this still causes loss of Shutdown Cooling on RHR loop B.
B. Incorrect – Although loss of RPS Bus A only causes one of the Shutdown Cooling suction IVs to close, this still causes loss of Shutdown Cooling on both RHR loops. The sustained electrical fault on RPS Bus A prevents transferring power to the alternate supply to restore SDC using RHR.
C. Incorrect – Loss of RPS A does not prevent placing RHR loop A in the Fuel Pool Cooling Assist mode.

Technical Reference(s): ON-RPS-101, ON-SDC-101, OP-149-003

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-6

Question Source: Bank - LOC26R (1/2015) NRC #57

Question History: LOC26R (1/2015) NRC #57

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 A1.05
	Importance Rating	4.1

HPCI

Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE COOLANT INJECTION SYSTEM controls including: Suppression pool temperature: BWR-2,3,4

Proposed Question: #31

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

- Surveillance SO-152-002, Quarterly HPCI Flow Verification, is in progress.
- Suppression Pool temperature is 112°F, up slow.

Which one of the following describes the required action, if any, in accordance with Technical Specifications?

The surveillance test...

A. may continue. Continued Reactor operator is acceptable at the current power level.

B. must be immediately suspended. The Reactor mode switch must be placed in SHUTDOWN immediately.

C. must be immediately suspended. Continued Reactor operation is acceptable at the current power level as long as Suppression Pool temperature does NOT exceed the scram limit.

D. must be immediately suspended. Suppression Pool temperature must be restored to within limits within one hour, or then the Reactor mode switch must be placed in SHUTDOWN.

Proposed Answer: B

Explanation: Technical Specification 3.6.2.1 allows Suppression Pool temperature to go as high as 105°F during SO-152-002 (up from the normal limit of 90°F). Once temperature exceeds 105°F, Condition C requires immediately suspending the surveillance test. Additionally, because temperature is above 110°F, Condition D requires immediately placing the Reactor mode switch in SHUTDOWN.

A. Incorrect – With Suppression Pool temperature >110°F, the test must be suspended and the Reactor mode switch in SHUTDOWN immediately. Plausible because this would be correct if temperature were ≤105°F. Also plausible because the normal limit is raised during testing that adds heat to the Suppression Pool.
C. Incorrect – With Suppression Pool temperature >110°F, the Reactor mode switch in SHUTDOWN immediately. Plausible because 120°F is an additional limit in TS 3.6.2.1 (but related to need for Reactor depressurization, not scram).
D. Incorrect – With Suppression Pool temperature >110°F, the Reactor mode switch in SHUTDOWN immediately. Plausible because the normal limit is raised during testing that adds heat to the Suppression Pool. Also plausible because TS 3.6.2.1 Condition A includes a 1 hour requirement.

Technical Reference(s): Technical Specification 3.6.2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-052 RBO-7

Question Source: Bank – JAF 16-1 NRC #74

Question History: JAF 16-1 NRC #74

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 2/6/18 – Revised wording of C and D slightly, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 K5.08
	Importance Rating	3.0

HPCI

Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM: Vacuum breaker operation: BWR-2,3,4

Proposed Question: #32

Unit 1 is operating at 100% power when it is determined that HPCI Turbine Exhaust Vacuum Breaker Check Valves 155F076 and 155F077 have failed open.

Which one of the following describes the operational concern due to these failures?

A. Containment isolation for that penetration is lost.

B. HPCI operation will pressurize the Suppression Chamber.

C. A loss of Suppression Chamber inventory following system operation.

D. Steam discharge into Secondary Containment would occur on system initiation.

Proposed Answer: B

Explanation: The two vacuum breaker valves (155F076, 155F077) are three-inch check valves that function to prevent steam from bypassing the Suppression Pool during turbine operation, and open to prevent the formation of a vacuum in the turbine exhaust header when the turbine is shut down. With both of them stuck open, if HPCI initiates, then turbine exhaust steam will flow through these valves directly into the Suppression Chamber airspace, causing direct pressurization.

A. Incorrect – Containment isolation for this penetration is still available via HV-155-F075 and HV-155-F079. Plausible that the check valves would be the containment isolation boundary on this relatively small line.
C. Incorrect – This line connects to the Suppression Chamber airspace, therefore Suppression Pool inventory will not be lost. Plausible because if this line discharged below the Suppression Pool water, then water would be lost through this line. Also plausible because a concern if these valves were failed closed is siphoning of water from the Suppression Pool into the HPCI exhaust line after turbine shutdown.
D. Incorrect – Steam discharge will be into the Suppression Chamber, not the Reactor Building. Plausible that this line would be connected to the Reactor Building airspace, since HPCI is located in the Reactor Building and this would provide the same design function.

Technical Reference(s): TM-OP-052, M-155

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-052 RBO-3

Question Source: Bank – Vision SYSID 1015

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 A3.06
	Importance Rating	3.6

LPCS

Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: Lights and alarms

Proposed Question: #33

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

- Flow verification testing is in progress on Core Spray loop A.
- HV-152-F004A, Core Spray Loop A Ob Inj Shutoff, is open.
- HV-152-F005A, Core Spray Loop A Ib Inj Shutoff, is closed.
- HV-152-F015A, Core Spray Loop A Test to Supp Pool, is full open.

Then, a loss of coolant accident results in the following:

- Reactor water level is -140", down slow.
- Reactor pressure is 475 psig, down slow.
- Drywell pressure is 15 psig, up slow.
- The following annunciators are in alarm:
  - o AR-109-A02, CORE SPRAY LOOP A ACTUATED
  - o AR-109-A03, ECCS LOOP A HI DRWL PRESS
  - o AR-109-A04, ECCS LOOP A RX LO LEVEL
- Core Spray loop A valve alignment is as shown on the next page.

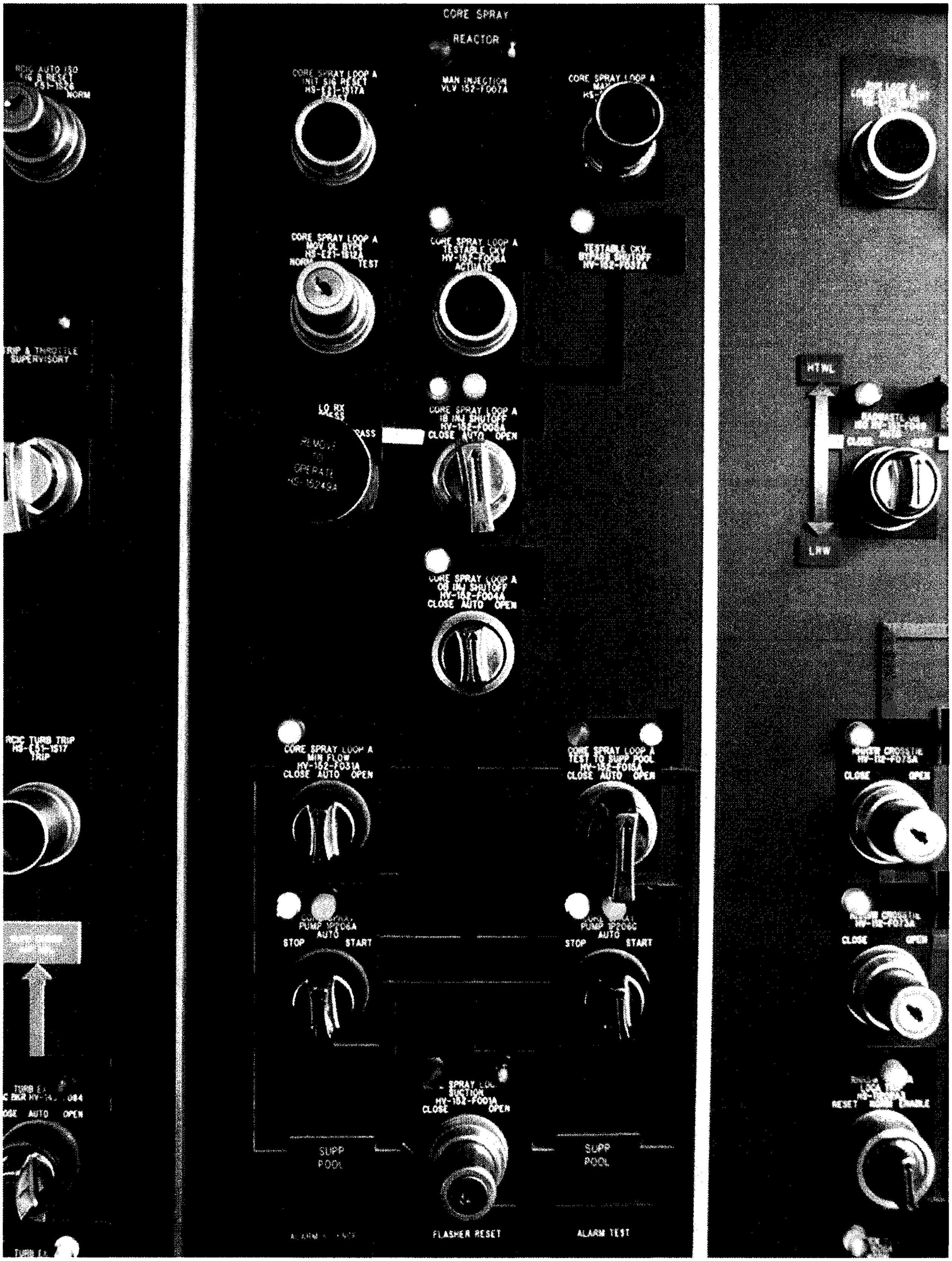
Which one of the following describes the status of Core Spray loop A valves F005A, F015A, and F031A?

A. F005A, F015A, and F031A have operated as designed.

B. F005A should be open, F015A should be closed, and F031A should be open.

C. F005A should be open and F015A should be closed. F031A is in the correct position.

D. F015A should be closed and F031A should be open. F005A is in the correct position.


CORE SPRAY
REACTOR
MAN INJECTION VLV 152-F007A
CORE SPRAY LOOP A
CORE SPRAY LOOP A MOV OL BYPS
NORM TEST
CORE SPRAY LOOP A TESTABLE CKV
ACTUATE
TESTABLE CKV BYPASS SHUTOFF
TRIP & THROTTLE SUPERVISORY
LO RX PRESS
REMOVE TO OPERATE
CORE SPRAY LOOP A IB INJ SHUTOFF
CLOSE AUTO OPEN
OB INJ SHUTOFF HV-152-F004A
CLOSE AUTO OPEN
RCIC TURB TRIP
TRIP
CORE SPRAY LOOP A MIN FLOW HV-152-F031A
CLOSE AUTO OPEN
CORE SPRAY LOOP A TEST TO SUPP POOL
CLOSE AUTO OPEN
PUMP 1P206A AUTO
STOP START
PUMP 1P206C AUTO
STOP START
SUCTION
CLOSE OPEN
SUPP POOL
SUPP POOL
FLASHER RESET
ALARM TEST
HTWL
CLOSE OPEN
CLOSE OPEN
CLOSE AUTO OPEN

Proposed Answer: D

Explanation: With flow verification testing initially in progress and HV-152F015A full open, Core Spray loop A injection valve F005A was initially closed, F004A was initially open, and the min flow valve (F031A) was initially closed. When Reactor water level went below -129", Core Spray received an automatic initiation signal. This signal is designed to cause the Core Spray test valve (F015A) to immediately close, but it has not. As this valve closes and Core Spray flow lowers, the min flow valve (F031A) should have automatically opened. F005A does not automatically open until Reactor pressure lowers further (~427 psig). F004A should be open based on normal lineup.

A. Incorrect – F015A should be closed and F031A should be open. Plausible because the given alignment is correct for the question's initial conditions, and Reactor pressure has not yet lowered enough to initiate CS injection.
B. Incorrect – F005A does not automatically open until Reactor pressure lowers further (~427 psig). Plausible because this would be correct if Reactor pressure was <427 psig but still above CS discharge pressure.
C. Incorrect – F005A does not automatically open until Reactor pressure lowers further (~427 psig). F031A should be open. Plausible because this would be correct if Reactor pressure was sufficiently below CS discharge pressure.

Technical Reference(s): OP-151-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-051 RBO-3

Question Source: Modified Bank – LOC26R (1/2015) NRC #8

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:
TRH 2/6/18 – Made editorial changes to last bullet and choice A based on NRC comments.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 2.1.28
	Importance Rating	4.1

SLC

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

Proposed Question: #34

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

- The SBLC MAN INITIATION keylock switch is placed in START A.

Which one of the following describes the resulting status of the white SBLC SQUIB READY A/B lights on Panel 1C601?

A. Both are illuminated.

B. Both are extinguished.

C. A is illuminated and B is extinguished.

D. A is extinguished and B is illuminated.

Proposed Answer: B

Explanation: When the SBLC MAN INITIATION keylock switch is placed in START A, SBLC pump 1A starts and both squib valves fire. When the valves fire, current rises to approximately 2 amps for a very short period of time and then falls to a steady-state value of 0 amps. The white lights are extinguished due to the resulting loss of continuity.

A. Incorrect – Both are extinguished. Plausible that the lights would illuminate when the valves fire.
C. Incorrect – A is extinguished also. Plausible that the lights would illuminate when the corresponding valve fires and that only A would fire since only pump 1A was started.
D. Incorrect – B is extinguished also. Plausible that B would not have fired since only pump 1A was started.

Technical Reference(s): OP-153-001, AR-107-A03, TM-OP-053

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-053 RBO-3

Question Source: Bank – NMP1 2015 Audit #3

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 K4.12
	Importance Rating	3.9

RPS

Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Bypassing of selected SCRAM signals (manually and automatically): Plant-Specific

Proposed Question: #35

A Unit 1 startup is in progress with the following:

- The Reactor Mode Switch has just been placed in RUN, in accordance with GO-100-002, Plant Startup, Heatup and Power Operation.
- NO further actions in GO-100-002 have been taken yet.
- All IRMs are indicating between 25 and 75 on Range 10.
- All APRMs are indicating between 7% and 8%.

Which one of the following describes the status of the IRM upscale scram?

The IRM upscale scram is...

A. bypassed based on a power signal from APRMs.

B. bypassed based on a signal from Reactor Mode Switch position.

C. NOT bypassed. It will be automatically bypassed based on a power signal from APRMs when Reactor power is higher.

D. NOT bypassed. If will be manually bypassed using switches on the IRM drawers in subsequent steps of GO-100-002.

Proposed Answer: B

Explanation: The IRM upscale scram is currently bypassed because the Reactor Mode Switch is in RUN.

A. Incorrect – The IRM upscale scram is currently bypassed because the Reactor Mode Switch is in RUN, not based on APRM power. Plausible because the APRM downscales are clear and APRMs are above the procedural limit for transitioning to Mode 1. Also plausible because TSV closure scram is bypassed based on a pressure signal that is proportional to Reactor power.
C. Incorrect – The IRM upscale scram is currently bypassed because the Reactor Mode Switch is in RUN. Plausible because power is still low enough on APRMs that IRM indications are in the normal range, such that further overlapping use is possible. Plausible because TSV closure scram is bypassed based on a pressure signal that is proportional to Reactor power and this does not occur until ~26% power.
D. Incorrect – The IRM upscale scram is currently bypassed because the Reactor Mode Switch is in RUN. Plausible because IRMs must be manually withdrawn and because the IRM drawers do have a mode switch for taking IRMs out of service.

Technical Reference(s): GO-100-002, AR-103-A04

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

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

RPS

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

Proposed Question: #36

Unit 1 has experienced a Reactor scram due to low Reactor water level with the following:

- The SCRAM DSCH VOL HI WTR LVL TRIP BYPS switch has been placed in BYPASS.
- All scram signals are clear.
- The PCO is performing OP-158-001, RPS System, section 2.6, Reset of Scram and ARI.
- The PCO takes the RPS SCRAM RESET switch to the GRP 1/4 position.

Which one of the following describes the response of annunciators AR-103-A01, RPS CHANNEL A1/A2 AUTO SCRAM, and AR-104-A01, RPS CHANNEL B1/B2 AUTO SCRAM?

A. Both annunciators reset.

B. AR-103-A01 resets, but AR-104-A01 remains in alarm.

C. AR-103-A01 remains in alarm, but AR-104-A01 resets.

D. NEITHER annunciator resets.

Proposed Answer: D

Explanation: When the RPS SCRAM RESET switch is taken to the GRP 1/4 position, it closes the scram inlet and outlet valves for two of the four rod groups in both RPS A and B. However, neither half scram logic is completely reset, therefore both of these annunciators remain in alarm until the switch is also taken to the GRP 2/3 position.

A. Incorrect – Both annunciators remain in alarm. Plausible that this one switch position clears enough of each RPS logic to reset both half scrams, since the switch position is not specific to either RPS channel.
B. Incorrect – Both annunciators remain in alarm. Plausible that this one switch position clears all of one side of RPS logic and not the other.
C. Incorrect – Both annunciators remain in alarm. Plausible that this one switch position clears all of one side of RPS logic and not the other.

Technical Reference(s): OP-158-001, TM-OP-058, RPS prints

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-058 RBO-4

Question Source: Bank – PB 2017 NRC #9

Question History: PB 2017 NRC #9

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 2/6/18 – Reduced wording in choice D based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K6.02
	Importance Rating	3.6

IRM

Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM: 24/48 volt D.C. power: Plant-Specific

Proposed Question: #37

A Unit 1 startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP
- Reactor power is midscale on IRM Range 6.

Then, 24 VDC Bus 1D682 de-energizes due to a sustained electrical fault.

Which one of the following describes the impact of this failure on the IRMs?

A. A, C, E & G IRM detectors CANNOT be moved.

B. B, D, F & H IRM detectors CANNOT be moved.

C. A, C, E & G IRM channels will de-energize resulting in a half scram and rod block.

D. B, D, F & H IRM channels will de-energize resulting in a half scram and rod block.

Proposed Answer: D

Explanation: 1D682 provides power to the IRM B, D, F, H drawers (division II). With loss of this power supply, these IRMs develop an INOP trip (half scram) and control rod block.

A. Incorrect – Power to Division I IRM detector drives is from 1Y218 and is not affected. Plausible because this would be correct for a loss of 1Y218.
B. Incorrect – Power to Division II IRM detector drives is from 1Y218 and is not affected. Plausible because this would be correct for a loss of 1Y218.
C. Incorrect – IRMs A, C, E, G are unaffected. Plausible because this would be correct for a loss of 1D672.

Technical Reference(s): AR-104-A06

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-078B RBO-3

Question Source: Bank – LOC24 (12/2011) Cert #11

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 A2.02
	Importance Rating	3.4

SRM

Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: SRM inop condition

Proposed Question: #38

A Unit 1 startup is in progress with SRM A bypassed due to a failed detector.

- The SRM B drawer mode switch is inadvertently bumped from OPERATE to STANDBY.
- The SRM B drawer mode switch is damaged such that it CANNOT be re-positioned back to OPERATE.

Which one of the following describes:

(1) whether a rod block is received, and

(2) the ability to bypass SRM B while SRM A remains bypassed, in accordance with the associated alarm response procedure?

	Rod Block Received?	While SRM A remains bypassed, SRM B...
A.	No	can also by bypassed.
B.	No	CANNOT also be bypassed.
C.	Yes	can also by bypassed.
D.	Yes	CANNOT also be bypassed.

Proposed Answer: D

Explanation: When the SRM mode switch is taken out of OPERATE, an INOP trip is received. This causes a rod block. All four SRMs are bypassed using a common joystick. Only one SRM can be bypassed at a time. Therefore, SRM B cannot be bypassed while SRM A remains bypassed.

A. Incorrect – An INOP rod block is received. Plausible that this would cause an alarm only, as the SRM will still indicate properly and two other SRMs from different divisions are still available. SRM B cannot be bypassed while SRM A remains bypassed. Plausible because these detectors are in different divisions (this is possible with IRMs).
B. Incorrect – An INOP rod block is received. Plausible that this would cause an alarm only, as the SRM will still indicate properly and two other SRMs from different divisions are still available.
C. Incorrect – SRM B cannot be bypassed while SRM A remains bypassed. Plausible because these detectors are in different divisions (this is possible with IRMs).

Technical Reference(s): AR-104-B06, TM-OP-078A

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 K5.04
	Importance Rating	2.9

APRM/LPRM

Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM: LPRM detector location and core symmetry

Proposed Question: #39

Which one of the following completes the below statement?

The LPRMs assigned to APRM 1 cover ___(1)___ quadrant(s) and ___(2)___ detector levels?

	(1) Number of Core Quadrants	(2) Number of Detector Levels
A.	1	1
B.	1	4
C.	4	1
D.	4	4

Proposed Answer: D

Explanation: APRM 1 receives inputs from LPRMs in each of the four core quadrants and at each of the four detector levels.

A. Incorrect – APRM 1 receives inputs from LPRMs in each of the four core quadrants, not just one. Plausible because each IRM and SRM channel only monitors one core quadrant. APRM 1 also receives inputs from LPRMs at each of the four detector levels, not just one. Plausible that LPRMs would be divisionalized to APRMs (A and C in one division, B and D in the other), such as other instruments are divisionalized to RPS.
B. Incorrect – APRM 1 receives inputs from LPRMs in each of the four core quadrants, not just one. Plausible because each IRM and SRM channel only monitors one core quadrant.
C. Incorrect – APRM 1 receives inputs from LPRMs at each of the four detector levels, not just one. Plausible that LPRMs would be divisionalized to APRMs (A and C in one division, B and D in the other), such as other instruments are divisionalized to RPS.

Technical Reference(s): OI-078-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-078D RBO-3

Question Source: Bank – LOC27 (2015) NRC #7

Question History: LOC27 (2015) NRC #7

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 2/6/18 – Revised all “2”s to “1”s in answer choices, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 K4.04
	Importance Rating	3.0

RCIC

Knowledge of REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) design feature(s) and/or interlocks which provide for the following: Prevents turbine damage: Plant-Specific

Proposed Question: #40

A seismic event results in the following:

- The Unit 1 Reactor is manually scrammed.
- Reactor water level is +25", stable, with RCIC injecting.
- Then, the following alarms are received:
  - AR-108-B01, RCIC TURB EXH HI PRESS
  - AR-108-C04, RCIC TURB HI PRESS BRG HI TEMP
  - AR-108-D05, RCIC TURB BRG OIL LO PRESS
- RCIC turbine exhaust pressure is 55 psig.
- RCIC turbine bearing temperatures are 200°F.
- RCIC turbine bearing lube oil pressure is 2 psig.

Which one of the following describes the resulting status of RCIC?

RCIC...

A. continues to operate.

B. trips on high exhaust pressure.

C. trips on high bearing temperature.

D. trips on low bearing lube oil pressure.

Proposed Answer: B

Explanation: RCIC trips because turbine exhaust pressure is >50 psig.

A. Incorrect – RCIC trips because turbine exhaust pressure is >50 psig. Plausible because the other given alarms do not result in a trip of RCIC. Also plausible that exhaust pressure would alarm at a lower pressure than the trip occurs.
C. Incorrect – RCIC trips turbine exhaust pressure, but not on high bearing temperature. Plausible because this is a valid alarm condition, would result in system damage, and requires RCIC shutdown per the AR.
D. Incorrect – RCIC trips turbine exhaust pressure, but not on low lube oil pressure. Plausible because this is a valid alarm condition, would result in system damage, and requires RCIC shutdown per the AR.

Technical Reference(s): AR-108-B01(C04)(D05)

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-050 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 A4.08
	Importance Rating	3.7

ADS

Ability to manually operate and/or monitor in the control room: Suppression pool level

Proposed Question: #41

Unit 1 is operating at 100% power when the following occurs:

- A seismic event results in a very large Suppression Pool break.
- The Reactor is scrammed.
- Reactor water level is +20", up slow.
- Suppression Pool water level is now 11.5', down slow.

Given the following possible pressure reduction methods:

(1) Opening SRVs
(2) Operating HPCI in pressure control mode

Which one of the following identifies which of these methods, if any, is currently allowable, in accordance with the Emergency Operating Procedures?

A. Only method (1)

B. Only method (2)

C. Both methods

D. NEITHER method

Proposed Answer: A

Explanation: EO-100-103 requires isolating HPCI because Suppression Pool water level is below 17' and adequate core cooling is assured. EO-100-112 allows opening ADS valves to reduce Reactor pressure because Suppression Pool water level is above 5'.

B. Incorrect – EO-100-103 requires isolating HPCI because Suppression Pool water level is below 17' and adequate core cooling is assured. Plausible because RCIC is still allowed to be used in pressure control mode. EO-100-112 allows opening ADS valves to reduce Reactor pressure because Suppression Pool water level is above 5'. Plausible because Suppression Pool water level is below the value requiring Emergency RPV Depressurization (12') and is approaching the level below which ADS valves are not allowed to be used (5').
C. Incorrect – EO-100-103 requires isolating HPCI because Suppression Pool water level is below 17' and adequate core cooling is assured. Plausible because RCIC is still allowed to be used in pressure control mode.
D. Incorrect – EO-100-112 allows opening ADS valves to reduce Reactor pressure because Suppression Pool water level is above 5'. Plausible because Suppression Pool water level is below the value requiring Emergency RPV Depressurization (12') and is approaching the level below which ADS valves are not allowed to be used (5').

Technical Reference(s): EO-100-103, EO-100-112

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-059 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 2/6/18 – Deleted “into multiple rooms” in first bullet and deleted last bullet, based on NRC comment.

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

Primary Containment Isolation/Nuclear Steam Supply Shutoff

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

Proposed Question: #42

Unit 1 is operating at 100% power when the following occurs:

- A Reactor scram occurs due to a loss of Feedwater.
- Reactor water level reaches a low of -42" and is being restored to the normal band.
- The picture on the following page shows the status of RWCU once Reactor water level is restored to normal and plant conditions stabilize.

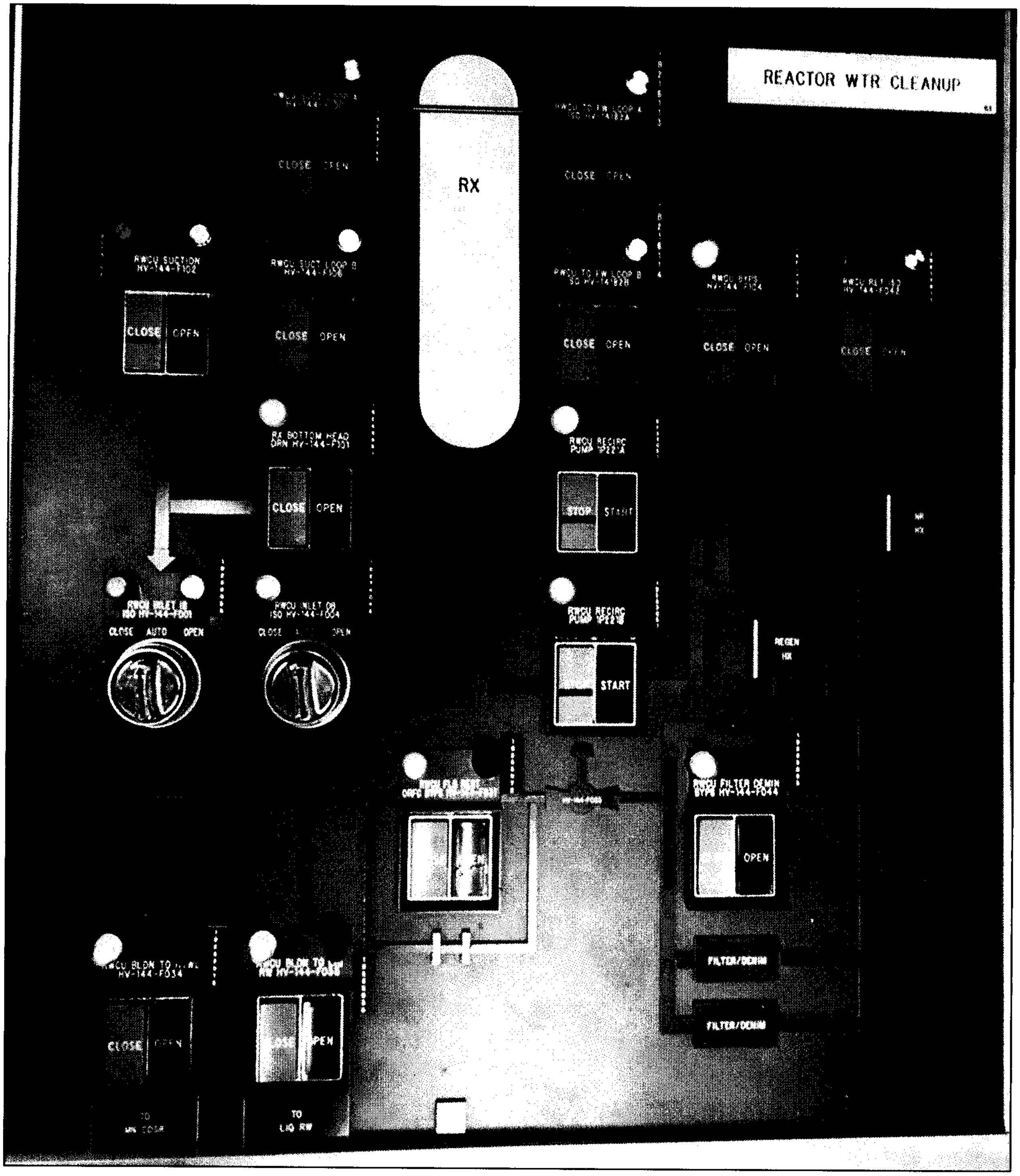
Which one of the following describes the status of HV-144-F001, RWCU INLET IB ISO, and HV-144-F004, RWCU INLET OB ISO?

A. Both valves have responded properly.

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

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

D. NEITHER valve has responded properly.


REACTOR WTR CLEANUP
RX
RWCU SUCTION HV-144-F102
CLOSE
OPEN
RX BOTTOM HEAD DRN HV-144-F101
RWCU RECIRC PUMP 1P221A
STOP
START
CLOSE AUTO OPEN
RWCU FILTER DEMIN BYPS HV-144-F044
FILTER/DEMIN
FILTER/DEMIN
REGEN HX
TO LIQ RW

Proposed Answer: C

Explanation: Reactor water level <-38" provides an isolation signal to both HV-144-F001 and HV-144-F004. The given indications show HV-144-F004 closed and HV-144-F001 open. Therefore, HV-144-F001 has failed to respond properly.

A. Incorrect – HV-144-F001 has failed to close as required. Plausible because this would be the correct answer for a RWCU isolation on high filter demineralizer inlet temperature.
B. Incorrect – HV-144-F001 has failed to close as required. Plausible because this would be the correct answer for a RWCU isolation on high filter demineralizer inlet temperature. HV-144-F004 has responded properly by closing. Plausible because this would be correct if Reactor water level had not gone <-38".
D. Incorrect – HV-144-F004 has responded properly by closing. Plausible because this would be correct if Reactor water level had not gone <-38". Also plausible if the logic and valve closures between the low Reactor water level and high filter demineralizer inlet temperature are mixed.

Technical Reference(s): ON-CONTISOL-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-061 RBO-4

Question Source: Modified Bank – LOC28R (2017) NRC #18

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 K2.01
	Importance Rating	2.8

SRVs

Knowledge of electrical power supplies to the following: SRV solenoids

Proposed Question: #43

Which one of the following describes the electrical power supply to the SRV solenoids that provide the pressure relief function?

A. One 125 VDC distribution panel supplies all 16 SRVs.

B. One 120 VAC distribution panel supplies all 16 SRVs.

C. Two 125 VDC distribution panels each supply half of the 16 SRVs.

D. Two 120 VAC distribution panels each supply half of the 16 SRVs.

Proposed Answer: A

Explanation: 125 VDC distribution panel 1D614 supplies electrical power to the pressure relief solenoids for all 16 SRVs.

B. Incorrect – The electrical power source is 125 VDC, not 120 VAC. Plausible because 120 VAC supplies other important solenoids (ex. RPS scram solenoids).
C. Incorrect – A single 125 VDC distribution panel provides electrical power for all pressure relief solenoids. Plausible because two 125 VDC distribution panels provides electrical power for the ADS solenoids.
D. Incorrect – A single 125 VDC distribution panel provides electrical power for all pressure relief solenoids. Plausible because two 125 VDC distribution panels provides electrical power for the ADS solenoids. The electrical power source is 125 VDC, not 120 VAC. Plausible because 120 VAC supplies other important solenoids (ex. RPS scram solenoids).

Technical Reference(s): E-180 Sh1

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-083E RBO-3

Question Source: Bank – LOC26R (1/2015) NRC #4

Question History: LOC26R (1/2015) NRC #4

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 K6.02
	Importance Rating	3.4

SRVs

Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES: Air (Nitrogen) supply: Plant-Specific

Proposed Question: #44

Unit 1 is operating at 100% power when the following occurs:

- A catastrophic piping rupture occurs just downstream of the CIG instrument dryers.
- Operators are unable to isolate the rupture.
- Downstream pressure lowers rapidly.

Which one of the following describes the impact of this failure on the ADS valves?

The ADS valves will...

A. lose all pneumatic operating pressure because their supply headers will depressurize through this leak location.

B. maintain pneumatic operating pressure due to automatic backup from Instrument Air.

C. maintain pneumatic operating pressure due to automatic backup from gas bottles.

D. maintain pneumatic operating pressure due to manual backup from Unit 2 CIG.

Proposed Answer: C

Explanation: The rupture location is upstream of the CIG 150 psig supply headers to the ADS valves. When header pressures lower to 142 psig, solenoid valves will automatically isolate the headers from this rupture location and aligned the headers to gas bottles.

A. Incorrect – When header pressures lower to 142 psig, solenoid valves will automatically isolate the headers from this rupture location and aligned the headers to gas bottles. Plausible because this leak location will cause header pressure to lower.
B. Incorrect – Automatic backup is from gas bottles, not Instrument Air. Plausible because Instrument Air is a manual backup supply to the 90# CIG header.
D. Incorrect – Automatic backup is from gas bottles, not Unit 2 CIG. Plausible because Unit 1 and 2 CRD can be cross-connected. Also plausible because a LOOP/LOCA on one Unit affects the other Unit's CIG system.

Technical Reference(s): ON-CIG-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-025 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 A1.03
	Importance Rating	3.8

Reactor Water Level Control

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: Reactor power

Proposed Question: #45

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

- The Reactor Mode Switch is in SHUTDOWN.
- ARI has been initiated.
- Reactor power is 8%, steady.
- Reactor water level is +25", up slow, with Feedwater injecting.
- Reactor pressure is 920 psig, steady, with Turbine Bypass Valves controlling.

Then, all injection to the RPV except RCIC, CRD and SBLC is stopped and prevented. Two minutes later (currently), conditions are as follows:

- Reactor power is 3%, down slow.
- Reactor water level is -45", down slow.
- Reactor pressure is 900 psig, down slow, with Turbine Bypass Valves available but closed.
- Suppression Pool temperature is 80°F, steady.
- Drywell pressure is 0.5 psig, steady.

Which one of the following describes the ability to re-commence injection with Feedwater, in accordance with EO-100-113, Level/Power Control?

Re-commencing injection with Feedwater is...

A. allowed based on the current value of Reactor power.

B. allowed based on the current value of Reactor water level.

C. NOT allowed based on the current value of Reactor power.

D. NOT allowed based on the current value of Reactor water level.

Proposed Answer: D

Explanation: EO-100-113 requires intentionally lowering Reactor water level in the initial conditions due to Reactor power >5%. Once all injection to the RPV except RCIC, CRD and SBLC is stopped and prevented, EO-100-113 requires lowering Reactor water level to at least -60" before re-assessing Reactor power to determine the need to further lower level or re-commence injection. Since Reactor water level is >-60", re-commencing Feedwater injection is not currently allowed. Reactor power is sufficiently low enough now that once level lowers to -60", Feedwater injection will be allowed.

A. Incorrect – Since Reactor water level is >-60", re-commencing Feedwater injection is not currently allowed. Plausible because Reactor power has lowered below the 5% threshold that originally required stopping and preventing injection.
B. Incorrect – Since Reactor water level is >-60", re-commencing Feedwater injection is not currently allowed. Plausible because Reactor water level has lowered over 60" and re-commencing injection is contingent upon Reactor water level.
C. Incorrect – Injection is not currently allowed because of Reactor water level, not power. Plausible because Reactor power has lowered below the 5% threshold that originally required stopping and preventing injection and Reactor power is sufficiently low enough now that once level lowers to -60", Feedwater injection will be allowed.

Technical Reference(s): EO-100-113

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-045 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 2/6/18 – Revised wording of answer choices based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 K1.01
	Importance Rating	3.4

Standby Gas Treatment

Knowledge of the physical connections and/or cause-effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: Reactor building ventilation system

Proposed Question: #46

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

- It is determined that HD-07543A, RB RECIRC SYS TO SGTS DMP, is mechanically bound closed.

Which one of the following describes the resulting availability of the Standby Gas Treatment (SGTS) trains to take suction on Reactor Building Zones I, II, and III?

A. NEITHER train is available to take suction on any Reactor Building Zone.

B. Only one train is available and it can take suction on all Reactor Building Zones.

C. Both trains are available, but can only take suction on two Reactor Building Zones.

D. Both trains are available to take suction on all Reactor Building Zones.

Proposed Answer: D

Explanation: The SGTS trains take suction from Reactor Building Ventilation through two parallel dampers, HD-07543A and HD-07543B. With HD-07543A failed closed, both trains can still take suction on all three RB Zones through HD-07543B.

A. Incorrect – With HD-07543A failed closed, both trains can still take suction on all three RB Zones through HD-07543B. Plausible because this would be correct if HD-07543A and HD-07543B were in series, such as HD-17508A and HD-17508B in the U1 DW/WW purge/burp suction path.
B. Incorrect – With HD-07543A failed closed, both trains can still take suction on all three RB Zones through HD-07543B. Plausible because this would be correct for a failure of HD-07553A.
C. Incorrect – With HD-07543A failed closed, both trains can still take suction on all three RB Zones through HD-07543B. Plausible because this would be correct if this damper only isolated one RB Zone (ex. Zone 1) from both trains.

Technical Reference(s): M-175 Sh2

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-070 RBO-3

Question Source: Modified Bank – LOC26 (2014) NRC #25

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 A2.10
	Importance Rating	2.9

AC Electrical Distribution

Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Exceeding current limitations

Proposed Question: #47

Unit 1 has experienced a loss of all offsite power (LOOP) with the following:

- DG A has started.
- DG A loading is currently 4600 KW.
- NO other DG is running.

Which one of the following describes the operation of DG A, in accordance with OP-024-001, Diesel Generators?

DG A loading is...

A. NOT exceeding a limit and may remain at this value indefinitely.

B. exceeding a limit, but can remain at this level for a maximum of 4 hours.

C. exceeding a limit, but can remain at this level for a maximum of 2 hours.

D. exceeding a limit and must be immediately lowered.

Proposed Answer: C

Explanation: OP-024-001 contains two DG load limits. There is a 4000 KW limit and a 4700 KW limit. Operation below 4000 KW is allowed indefinitely. Operation between 4000 and 4700 KW is allowed for short term operation. Short term operation is defined as a 2 hour limit. Operation above 4700 KW is not allowed for any period of time. Therefore, with load at 4600 KW, the 4000 KW limit is being exceeded, but operation can continue for 2 hours.

Note: The question meets the K/A because DG KW loading (KW) is directly related to ESS Bus current (amps) (Power = Current * Voltage, and voltage is held constant for a given bus, therefore Power ≈ Current). The candidate must assess loading and then both determine the effect (above or below any limits) and determine the length of time allowed (using procedure to control/mitigate). Additionally, there is no K/A under 262001 A2 that separately requires testing exceeding load/power limitations.

A. Incorrect – DG A loading is exceeding the 4000 KW continuous rating. Plausible because DG A loading is still below the 4700 KW short term rating.
B. Incorrect – The limit is 2 hours, not 4 hours. Plausible because 4 hours is the station blackout coping time.
D. Incorrect – Currently loading is acceptable for 2 hours. Plausible because this would be correct if loading was >4700 KW.

Technical Reference(s): OP-024-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-024 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 2/6/18 – Revised out “only” in stem, added bullet that no other DG is running, and made minor wording changes to answers based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 A3.01
	Importance Rating	2.8

Uninterruptable Power Supply (AC/DC)

Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: Transfer from preferred to alternate source

Proposed Question: #48

Unit 1 is operating at 100% power when Motor Control Center 1B246 de-energizes due to a sustained electrical fault.

Which one of the following describes the effect of this loss on Non-Class 1E Uninterruptible Power Supplies (UPS)?

Instrument AC UPS __(1)__ will be supplied by its __(2)__.

A. (1) 1D130
(2) battery until the battery voltage drops to less than 210 VDC, then it automatically transfers to its alternate supply, MCC 1B226.

B. (1) 1D130
(2) alternate supply, MCC 1B226, until MCC 1B246 is restored

C. (1) 1D240
(2) battery until the battery voltage drops to less than 210 VDC, then it automatically transfers to its alternate supply, MCC 1B226.

D. (1) 1D240
(2) alternate supply, MCC 1B226, until MCC 1B246 is restored

Proposed Answer: A

Explanation: MCC 1B246 supplies normal power to UPS 1D130. Upon loss of MCC 1B246, 1D130 first swaps to its battery. When battery voltage drop below 210 VDC, then 1D130 swaps to the alternate AC supply from 1B226.

B. Incorrect – Upon loss of MCC 1B246, the UPS first swaps to its battery. Plausible because 1B226 is the alternate AC supply and eventually does power the UPS.
C. Incorrect – MCC 1B246 supplies normal power to UPS 1D130, not UPS 1D240. Plausible because 1D240 is a related UPS.
D. Incorrect – MCC 1B246 supplies normal power to UPS 1D130, not UPS 1D240. Plausible because 1D240 is a related UPS. Upon loss of MCC 1B246, the UPS first swaps to its battery. Plausible because 1B226 is the alternate AC supply and eventually does power the UPS.

Technical Reference(s): ON-4KV

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-017 RBO-3

Question Source: Bank – LOC25 (2013) Audit #19

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 2.1.23
	Importance Rating	4.3

Uninterruptable Power Supply (AC/DC)

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: #49

Unit 1 is operating at 100% power when the following occurs:

- A failure of 1D130, Instrument AC UPS, results in Distribution Panel 1Y128 de-energizing.
- AR-102-C02, RECIRC MG WINDING A/B HI TEMP, alarms.
- Recirc MG Winding temperatures are 250°F, up slow, for both pumps.

Which one of the following describes the required operator action, in accordance with ON-YPNL-101, Loss of Instrument Bus?

A. Lower Recirc flow by manually adjusting scoop tube positioners.

B. Lower Recirc flow from the Control Room using ICS HMI.

C. Place alternate Recirc MG Set Ventilation in service.

D. Scram the Reactor and trip both Recirc pumps.

Proposed Answer: D

Explanation: Loss of 1Y128 results in loss of Recirc MG Set Ventilation and rising Recirc MG winding temperatures. Critical Condition A.1 requires scramming the Reactor and tripping both Recirc pumps if AR-102-C02 alarms.

A. Incorrect – The required action is to scram the Reactor and trip both Recirc pumps. Plausible because lowering Recirc flow would reduce the heat load in the Recirc MG set area. Also plausible because the loss of 1Y128 does result in a scoop tube lockup and a step in ON-YPNL-101 requires manually adjusting scoop tube positioners to change Recirc flow.
B. Incorrect – The required action is to scram the Reactor and trip both Recirc pumps. Plausible because lowering Recirc flow would reduce the heat load in the Recirc MG set area and flow is normally adjusted using ICS HMI.
C. Incorrect – The required action is to scram the Reactor and trip both Recirc pumps. Plausible because once 1Y128 is restored, a step in ON-YPNL-101 requires placing Recirc MG Set Ventilation in service per OP-133-002.

Technical Reference(s): ON-YPNL-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-017 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 K2.01
	Importance Rating	3.1

DC Electrical Distribution

Knowledge of electrical power supplies to the following: Major D.C. loads

Proposed Question: #50

Which one of the following identifies the electrical power supplies to the Unit 1 ATWS-ARI solenoids?

A. 1D614 and 1D624

B. 1D652 and 1D662

C. 1Y218 and 1Y219

D. 1Y201A and 1Y201B

Proposed Answer: A

Explanation: The Unit 1 ATSW-ARI solenoids are powered from 125 VDC panels 1D614 and 1D624.

B. Incorrect – The Unit 1 ATSW-ARI solenoids are powered from 125 VDC panels 1D614 and 1D624. Plausible because 1D652 and 1D662 are also Unit 1 DC panels.
C. Incorrect – The Unit 1 ATSW-ARI solenoids are powered from 125 VDC panels 1D614 and 1D624. Plausible because 1Y218 and 1Y219 are 120 VAC panels that supply important instrumentation.
D. Incorrect – The Unit 1 ATSW-ARI solenoids are powered from 125 VDC panels 1D614 and 1D624. Plausible because 1Y201A and 1Y201B are 120 VAC panels that supply power to the RPS scram pilot solenoids.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-058 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 K1.04
	Importance Rating	3.2

Emergency Generators (Diesel/Jet)

Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: Emergency generator cooling water system

Proposed Question: #51

A loss of all offsite power has occurred with the following:

- Diesel Generator (DG) A is running loaded to the 1A/2A ESS buses.
- DG B started and then tripped on generator differential.
- DG C failed to start and all attempts to start it manually have failed.
- DG D is running loaded to the 1D/2D ESS buses.
- Emergency Service Water (ESW) pump D failed to start.
- All other ESW pumps responded as designed per the given plant conditions.

Which one of the following describes the status of cooling water to DG D?

DG D...

A. is being supplied cooling water from ESW pump A.

B. is being supplied cooling water from ESW pump B.

C. is running without cooling water, but cooling water can be manually aligned.

D. is running without cooling water and cooling water CANNOT be manually aligned.

Proposed Answer: A

Explanation: With a loss of offsite power, each ESW pump is powered from it's respective DG. Since DGs B and C are not powering their buses, ESW pumps B and C are not running. With ESW pump D also failing, this leaves only ESW pump A running in ESW loop A. ESW loop A supplies cooling water to all four DGs through normally open MOVs HV-01112A(B)(C)(D). Therefore, even with only ESW pump A running, DG D has cooling water automatically supplied.

B. Incorrect – ESW pump B is not running because DG B tripped after start. Plausible because ESW pump B should be running and then would supply cooling water to DG D. Also plausible that ESW pump power supplies would be arranged in some other way than solely with same DG.
C. Incorrect – DG D has cooling water automatically supplied from ESW pump A. Plausible that ESW pump A would need to be manually aligned by opening HV-01112D since these components are in different divisions.
D. Incorrect – DG D has cooling water automatically supplied from ESW pump A. Plausible that ESW pump A would not be able to be manually aligned to the opposite division DG due to divisional separation criteria (ex. such as if only ESW B and ESW D could be cross-connected).

Technical Reference(s): TM-OP-054

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-054 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 K4.03
	Importance Rating	2.8

Instrument Air

Knowledge of INSTRUMENT AIR SYSTEM design feature(s) and or interlocks which provide for the following: Securing of IAS upon loss of cooling water

Proposed Question: #52

Unit 1 is operating at 100% power when the following occurs:

- The running TBCCW pump trips.
- The standby TBCCW pump fails to start.

Which one of the following describes the impact on the Instrument Air Compressors?

The Instrument Air Compressors...

A. trip on a low cooling water pressure signal.

B. trip only if discharge air temperature rises to 320°F.

C. are automatically supplied alternate cooling water from ESW.

D. are automatically supplied alternate cooling water from TBCW.

Proposed Answer: A

Explanation: IACs are supplied cooling water from TBCCW. IACs have a low cooling water pressure trip (<20 psig) that will trip the IACs upon the given total loss of all TBCCW pressure.

B. Incorrect – IACs trip on low cooling water pressure regardless of how high discharge air temperature gets. Plausible because if the low cooling water pressure trip were not part of the design, or if this was not a complete loss of TBCCW pressure, then the IACs would trip only if discharge air temperature rises to 320°F
C. Incorrect – There is no automatic cooling water backup from ESW. Plausible because ESW is a backup source of cooling water to the TBCCW heat exchangers on loss of Service Water and some plant loads have an automatic cooling water backup feature.
D. Incorrect – There is no automatic cooling water backup from TBCW. Plausible because some plant loads have an automatic cooling water backup feature. Also plausible because the similar RBCCW and RBCW systems have an automatic backup relationship for some loads.

Technical Reference(s): AR-124-D01, LA-1140-A05

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-018 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 K2.02
	Importance Rating	2.9

Component Cooling Water

Knowledge of electrical power supplies to the following: CCW valves

Proposed Question: #53

Unit 1 is operating at 100% power when 1B216 de-energizes due to a sustained electrical fault.

Which one of the following operations is affected by this electrical power loss?

A. Aligning RHRSW A cooling to the RHR Loop A heat exchanger

B. Aligning RHRSW B cooling to the RHR Loop B heat exchanger

C. Cross-tying RHRSW A to RHR Loop A for injection to the Reactor

D. Cross-tying RHRSW B to RHR Loop B for injection to the Reactor

Proposed Answer: C

Explanation: Loss of 1B216 results in no power to HV-112F073A, which is required to be opened to cross-tie RHRSW A to RHR Loop A for injection to the Reactor.

A. Incorrect – Aligning RHRSW A cooling to the RHR Loop A heat exchanger is not affected by this power loss. Plausible because this would be correct for a loss of 1B237.
B. Incorrect – Aligning RHRSW B cooling to the RHR Loop B heat exchanger is not affected by this power loss. Plausible because this would be correct for a loss of 1B247.
D. Incorrect – Cross-tying RHRSW B to RHR Loop B for injection to the Reactor is not affected by this power loss. Plausible because this would be correct for a loss of 1B247.

Technical Reference(s): ON-4KV-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-016 RBO-3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201001 A1.09
	Importance Rating	2.9

CRD Hydraulic

Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD DRIVE HYDRAULIC SYSTEM controls including: CRD drive water flow

Proposed Question: #54

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

- A control rod pattern adjustment is in progress.
- CRD drive water pressure is 250 psid.
- Control rod 30-31 is given a single notch insert signal.
- During insertion, drive water insert flow is approximately 4.8 gpm.
- The control rod double notches.

Which one of the following describes the required adjustment?

Adjust...

A. PV-146-F003, DRIVE WTR PRESS THTLG, further open.

B. PV-146-F003, DRIVE WTR PRESS THTLG, further closed.

C. HCU-30-31 INSERT DRIVE VALVE SPEED CONTROL VALVE further open.

D. HCU-30-31 INSERT DRIVE VALVE SPEED CONTROL VALVE further closed.

Proposed Answer: D

Explanation: Drive water pressure is appropriate, but drive water insert flow was higher than desired. To lower insert flow and slow down the rod to prevent double notching, the associated HCU's INSERT DRIVE VALVE SPEED CONTROL VALVE must be adjusted in the closed direction. This will lower the drive water insert flow without changing drive water pressure.

A. Incorrect – CRD drive water pressure is at the required value per OP-155-001 and adjusting PV-146-F003 would change drive water pressure. Plausible because if drive water pressure were high, it could cause double notching and the appropriate response would be to adjust PV-146-F003.
B. Incorrect – CRD drive water pressure is at the required value per OP-155-001 and adjusting PV-146-F003 would change drive water pressure. Plausible because if drive water pressure were high, it could cause double notching and the appropriate response would be to adjust PV-146-F003.
C. Incorrect – The associated HCU's INSERT DRIVE VALVE SPEED CONTROL VALVE must be adjusted further closed, not open. Plausible because if PV-146-F003 needed to be adjusted to prevent double notching, it would be opened further to lower drive water pressure.

Technical Reference(s): M-147 Sh2

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:
TRH 2/6/18 – Added control rod number to stem and C/D based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201002 2.4.49
	Importance Rating	4.6

Reactor Manual Control

Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Proposed Question: #55

Unit 1 is operating at 100% power with the following

- Control rod 10-21 was initially at position 12.
- Control rod 10-21 is now at position 16 and withdrawing out of the core.

Which one of the following describes the required ***immediate*** operator action(s), in accordance with ON-CRD-101, Control Rod Malfunction?

A. Scram the Reactor.

B. Reduce core flow to lower Reactor power 20%.

C. Select the control rod and insert it to position 00.

D. Select the control rod and give it a single notch insert signal to stop it at the current position.

Proposed Answer: C

Explanation: With a single control rod drifting out of the core, the immediate operator actions of ON-CRD-101 require inserting the control rod to position 00.

A. Incorrect – With a single control rod drifting out of the core, the immediate operator actions of ON-CRD-101 require inserting the control rod to position 00. Plausible because the immediate actions of ON-CRD-101 would require a scram if three or more control rods were drifting.
B. Incorrect – With a single control rod drifting out of the core, the immediate operator actions of ON-CRD-101 require inserting the control rod to position 00. Plausible because the subsequent actions of ON-CRD-101 require a Reactor power reduction of 20% using Recirc flow.
D. Incorrect – With a single control rod drifting out of the core, the immediate operator actions of ON-CRD-101 require inserting the control rod to position 00. Plausible because the immediate actions require selecting the control rod and giving it an insert signal, but to 00, not just to stop it at the current position. Also plausible because stopping the control rod at the current position minimizes the ensuing power transient and would prevent inserting it beyond it's initial position).

Technical Reference(s): ON-CRD-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-056 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 2/9/18 – Revised all references to “drifting” out of stem, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201003 K5.01
	Importance Rating	2.6

Control Rod and Drive Mechanism

Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD AND DRIVE MECHANISM: Hydraulics

Proposed Question: #56

A Unit 1 startup is in progress with Reactor pressure at 500 psig.

- The in-service CRD flow control valve fails closed.

Which one of the following describes the immediate effect of this failure on the ability to move control rods with RMCS and the ability to scram the Reactor?

Control rods ___(1)___ be moved with RMCS and control rods ___(2)___ insert on a scram.

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

Proposed Answer: C

Explanation: Closure of the in-service CRD FCV significantly lowers flow to the drive water and cooling water headers, but not the charging water header. The ability to drive rods with RMCS is lost (insufficient drive water pressure), but charging water header pressure is still available to ensure the scram function will work. HCU accumulators are charged in this question and are the credited method of rod insertion at the provided RPV pressure.

A. Incorrect – The ability to drive rods with RMCS is lost. Plausible because the FCV is designed to still pass some flow for CRDM cooling purposes, but this does not result in enough drive water pressure to move control rods with RMCS.
B. Incorrect – The ability to drive rods with RMCS is lost. Plausible because the FCV is designed to still pass some flow for CRDM cooling purposes, but this does not result in enough drive water pressure to move control rods with RMCS. Control rods will still insert on a scram. Plausible because Reactor pressure is low, such that accumulators are needed to ensure the scram function. This would be correct if the charging water header tapped off downstream of the FCV and accumulator alarms were received.
D. Incorrect – Control rods will still insert on a scram. Plausible because Reactor pressure is low, such that accumulators are needed to ensure the scram function. This would be correct if the charging water header tapped off downstream of the FCV and accumulator alarms were received.

Technical Reference(s): M-146

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-055H

Question Source: Bank – PB 2017 NRC #32

Question History: PB 2017 NRC #32

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202002 A4.03
	Importance Rating	3.1

Recirculation Flow Control System

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

Proposed Question: #57

Unit 1 is operating at 65% power when the following occurs:

- A loss of Control Room annunciators AR-101 through AR-103 has occurred.
- Then, during monitoring, the stable indications on the following page are observed for Recirc pump A.

Which one of the following describes the indicated condition of Recirc pump A?

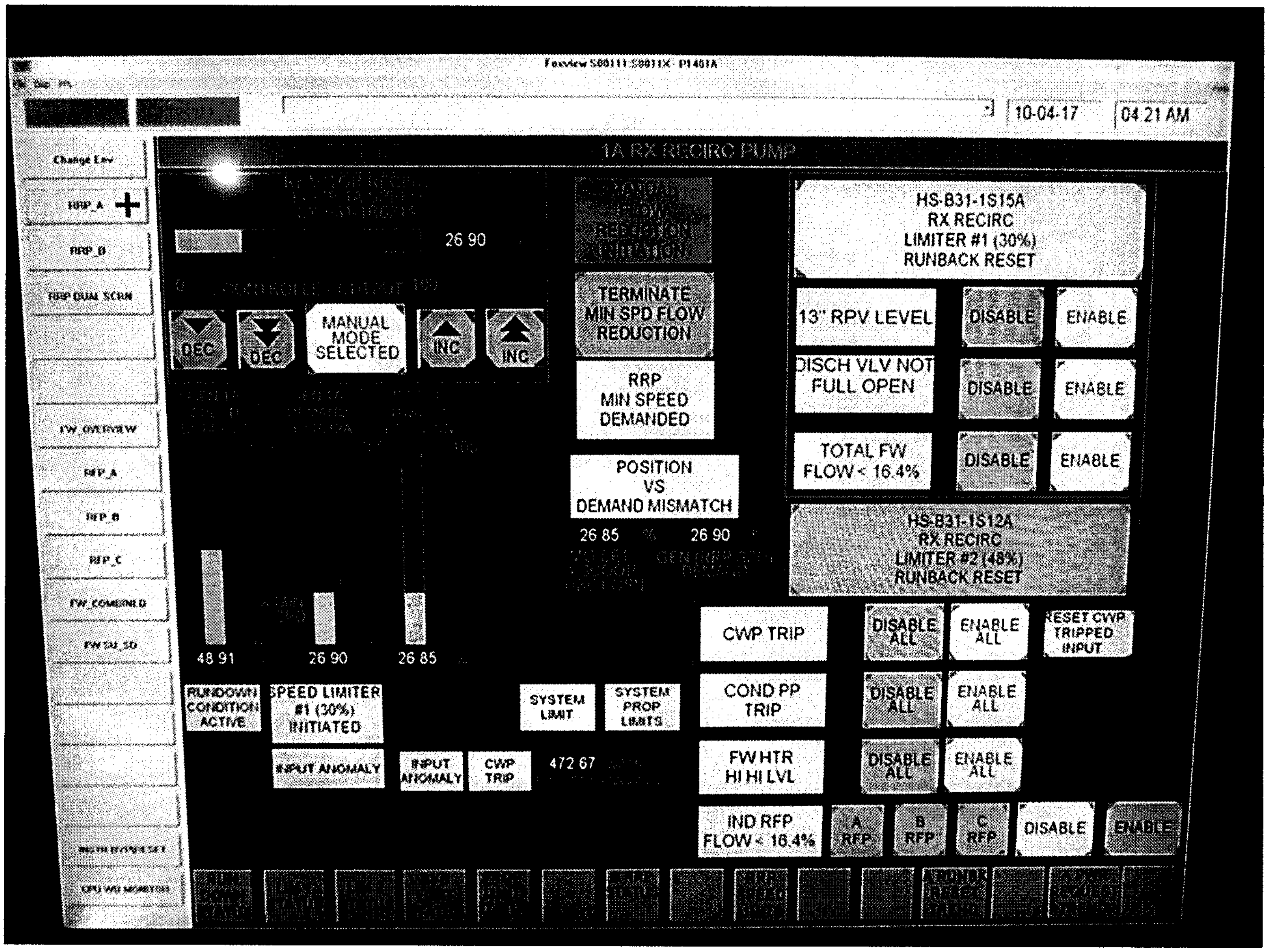
Recirc pump A...

A. has tripped.

B. has received a 30% runback signal.

C. has received a 48% runback signal.

D. is in a normal condition for 65% power operation.



10-04-17
04 21 AM
1A RX RECIRC PUMP
Change Env
RRP_A
RRP_B
RRP DUAL SCRN
FW_OVERVIEW
RFP_A
RFP_B
RFP_C
FW_COMBINED
26 90
DEC
DEC
MANUAL MODE SELECTED
INC
INC
TERMINATE MIN SPD FLOW REDUCTION
RRP MIN SPEED DEMANDED
POSITION VS DEMAND MISMATCH
26 85
26 90
HS-B31-1S15A RX RECIRC LIMITER #1 (30%) RUNBACK RESET
13" RPV LEVEL
DISABLE
ENABLE
DISCH VLV NOT FULL OPEN
DISABLE
ENABLE
TOTAL FW FLOW < 16.4%
DISABLE
ENABLE
HS-B31-1S12A RX RECIRC LIMITER #2 (48%) RUNBACK RESET
48 91
26 90
26 85
CWP TRIP
DISABLE ALL
ENABLE ALL
RESET CWP TRIPPED INPUT
RUNDOWN CONDITION ACTIVE
SPEED LIMITER #1 (30%) INITIATED
SYSTEM LIMIT
SYSTEM PROP LIMITS
COND PP TRIP
DISABLE ALL
ENABLE ALL
INPUT ANOMALY
INPUT ANOMALY
CWP TRIP
472 67
FW HTR HI HI LVL
DISABLE ALL
ENABLE ALL
IND RFP FLOW < 16.4%
A RFP
B RFP
C RFP
DISABLE
ENABLE

Proposed Answer: C

Explanation: The given indications show Recirc pump A has received a 48% runback signal, as evidenced by generator speed at approximately 48% and SPEED LIMTER #2 INITIATED light being illuminated in red.

A. Incorrect – Recirc pump A has runback to 48%, not tripped. Plausible because this would be correct if generator speed were indicating 0%.
B. Incorrect – Recirc pump A has runback to 48%, not 30%. Plausible because multiple indications read just below 30%.
D. Incorrect – Recirc pump A has runback to 48% and is not in a normal condition for 65% power. Plausible because this would be correct if pumps speeds were higher and SPEED LIMTER #2 INITIATED light was not illuminated.

Technical Reference(s): AR-102-C01, TM-OP-064

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-064 RBO-4

Question Source: Bank – JAF 16-1 NRC #50

Question History: JAF 16-1 NRC #50

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	215001 K6.04
	Importance Rating	3.1

Traversing In Core Probe

Knowledge of the effect that a loss or malfunction of the following will have on the TRAVERSING IN-CORE PROBE: Primary containment isolation system: Mark-I&II (Not-BWR1)

Proposed Question: #58

Unit 1 is operating at 100% power when the following occurs:

- Traversing In-Core Probe (TIP) scans are in progress with TIP channel A detector in the core.
- A coolant leak develops in the Drywell.
- Drywell pressure is 2 psig, up slow.
- The TIP channel A detector fails to retract on a signal from PCIS.

Which one of the following describes the resulting operation of the TIP channel A shear valve?

The TIP channel A shear valve…

A. automatically closes immediately.

B. automatically closes after a time delay.

C. must be manually closed from the TIP room.

D. must be manually closed from the Control Room.

Proposed Answer: D

Explanation: With the TIP A detector failing to retract on a PCIS signal (Drywell pressure >1.72 psig), the associated ball valve will not be able to close and isolate the penetration. While there is an automatic function for closing the ball valve, there is no automatic function for closing the shear valve. ON-CONT-ISOL directs isolating the TIP by manually firing the shear valve using a control on Control Room panel 1C607.

A. Incorrect – There is no automatic function for closing the shear valve. Plausible because there is an automatic function for closing the ball valve and closure of the shear valve is required because the detector has not retracted.
B. Incorrect – There is no automatic function for closing the shear valve. Plausible because there is an automatic function for closing the ball valve and closure of the shear valve is required because the detector has not retracted. Plausible that a time delay would be included to prevent unnecessary firing of the shear valve if the detector retraction was just slower than normal.
C. Incorrect – The shear valve control is on Control Room panel 1C607, not at the TIP room. Plausible because there is a provision at the TIP room for manual hand crank operation of the detector drive.

Technical Reference(s): OP-178-001, ON-CONTISOL-101

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank - JAF 12-2 NRC #57

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:
TRH 2/6/18 – Revised end of C and D based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	215002 K5.01
	Importance Rating	2.6

Rod Block Monitor

Knowledge of the operational implications of the following concepts as they apply to ROD BLOCK MONITOR SYSTEM: Trip reference selection: Plant-Specific

Proposed Question: #59

Which one of the following describes when the Rod Block Monitor is automatically bypassed and how the RBM upscale trip setpoints change as APRM power changes?

Note: All percentages given as Nominal Trip Setpoint, not Allowable Values.

	APRM Power Ranges			
	<24.9%	24.9%-61.0%	61.0%-81.0%	>81.0%
A.	109.2%	117.0%	123.0%	Bypassed
B.	123.0%	117.0%	109.2%	Bypassed
C.	Bypassed	109.2%	117.0%	123.0%
D.	Bypassed	123.0%	117.0%	109.2%

Proposed Answer: D

Explanation: The RBM automatically bypasses/unbypasses and adjusts the upscale trip setpoint based on a simulator thermal power value from APRMs. Below 24.9% on APRMs, the RBM is bypassed. From 24.9%-61.0%, the low power upscale setpoint of 123.0% is used. From 61.0-81.0%%, the intermediate power upscale setpoint of 117.0% is used. Above 81.0%, the high power upscale setpoint of 109.2% is used.

A. Incorrect – The RBM is bypassed at low powers, not high powers. Plausible because the RWM is bypassed at high power and not at low powers. The RBM upscale setpoint gets lower as APRM power rises, not higher. Plausible that RBM setpoint would get higher as APRM power rises, such as how APRM upscale setpoints get higher when Recirc flow gets higher.
B. Incorrect – The RBM is bypassed at low powers, not high powers. Plausible because the RWM is bypassed at high power and not at low powers.
C. Incorrect – The RBM upscale setpoint gets lower as APRM power rises, not higher. Plausible that RBM setpoint would get higher as APRM power rises, such as how APRM upscale setpoints get higher when Recirc flow gets higher.

Technical Reference(s): TRM 3.2.1 Table 7.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-078K RBO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	223001 A2.08
	Importance Rating	3.1

Primary Containment and Auxiliaries

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Compressor trips (loss of air): Plant-Specific

Proposed Question: #60

Unit 1 is operating at 25% power during a shutdown. Drywell venting and purging in progress.

- All Unit 1 Instrument Air and Service Air compressors trip.
- Unit 1 Instrument Air and Service Air pressures are lowering.
- The Unit Supervisor, with Shift Manager permission, has directed cross-tying Unit 1 and Unit 2 Instrument Air in accordance with OP-118-002, Instrument Air System Infrequent Operations.

Which one of the following describes (1) the response of the Drywell venting and purging lineups if air pressure drops to 0 psig and (2) on which elevation of the Turbine Building the cross-tying evolution is performed, in accordance with OP-118-002?

	(1)	(2)
A.	Still aligned	656’
B.	Still aligned	676’
C.	Isolated	656’
D.	Isolated	676’

Proposed Answer: C

Explanation: A complete loss of Instrument Air pressure will result in the Drywell venting and purging lineups isolating due to dampers that fail closed. Unit 1 and Unit 2 Instrument Air are cross-tied by opening 025091, which is located in the Turbine Building on elevation 656'.

A. Incorrect – The Drywell venting and purging lineups isolate due to dampers that fail closed. Plausible because these lineups could be made with MOVs, dampers that fail open on loss of IA, or could use the unaffected Containment Instrument Gas system.
B. Incorrect – The Drywell venting and purging lineups isolate due to dampers that fail closed. Plausible because these lineups could be made with MOVs, dampers that fail open on loss of IA, or could use the unaffected Containment Instrument Gas system. 025091 is located on elevation 656', not 676'. Plausible because the Instrument Air skids are located on elevation 676'.
D. Incorrect – 025091 is located on elevation 656', not 676'. Plausible because the Instrument Air skids are located on elevation 676'.

Technical Reference(s): ON-INSTAIR-101, OP-118-002

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-018 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	234000 A4.01
	Importance Rating	3.7

Fuel Handling Equipment

Ability to manually operate and/or monitor in the control room: Neutron monitoring system

Proposed Question: #61

Refueling is in progress on Unit 1 with the following:

- A new fuel bundle is being loaded in the general vicinity of SRM A, but NOT directly adjacent to SRM A.
- The fuel bundle has been lowered halfway into the core.
- The following SRM indications are observed:

SRM	Initial Count Rate (cps)	Current Count Rate (cps)	Period (sec)
A	50	300	20
B	60	65	400
C	45	55	200
D	70	75	400

- Annunciator AR-104-D06, SRM SHORT PERIOD, is in alarm.
- Refuel floor area radiation monitors indicate 2 mR/hr, steady.

Which one of the following describes the operational implications of these indications?

These SRM indications are…

A. normal. SRM A period will go to infinity after the fuel move.

B. normal. SRM A counts will lower to match other SRMs after the fuel move.

C. abnormal. Stop the fuel movement. Refuel floor evacuation is required.

D. abnormal. Stop the fuel movement. Refuel floor evacuation is NOT required.

Proposed Answer: D

Explanation: While some response on SRM A may be expected due to loading of a new fuel bundle in the same core quadrant, there are limitations as to how much response is expected. Annunciator AR-104-D06, SRM SHORT PERIOD, is received at a period of 50 seconds. Both high SRM count rates and short period are indications of an inadvertent criticality. ON-FUEL-001 only requires evacuation of the Refuel floor if unexpected changes in radiation level occur. Since no indication is given of rising radiation levels, evacuation is NOT required.

A. Incorrect – Some SRM A response may be expected and is allowed, however the amount of response exceeds limits. Plausible because the given alarm and such short periods can occur during control rod withdrawal during a startup, but are NOT normal during fuel loading.
B. Incorrect – Some SRM A response may be expected and is allowed, however the amount of response exceeds limits. Plausible because the given alarm and such short periods can occur during control rod withdrawal during a startup, but are NOT normal during fuel loading.
C. Incorrect – Evacuation would be required for the secondary containment if radiation alarms were to also occur.

Technical Reference(s): AR-104-D06, ON-FUEL-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-078A RBO-7

Question Source: Bank – NMP2 2014 Audit #39

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(13)

Comments:
TRH 2/6/18 – Added bullet regarding Refuel floor ARM and revised answer choices, based on NRC comments.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	241000 K3.08
	Importance Rating	3.7

Reactor/Turbine Pressure Regulating System

Knowledge of the effect that a loss or malfunction of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM will have on following: Control/governor valves

Proposed Question: #62

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

- EHC pressure regulator channel A is in control.

Then, the sensed pressure signal to EHC pressure regulator channel A fails upscale over two (2) minutes.

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

	TCVs...	The Reactor...
A.	open further.	scrams.
B.	open further.	does NOT scram.
C.	close further.	scrams.
D.	close further.	does NOT scram.

Proposed Answer: A

Explanation: EHC operates such that whichever pressure regulator demands the greatest opening of TCVs is in control. With EHC pressure regulator A initially in control, pressure setpoint bias is set such that EHC pressure regulator B will attempt to control Reactor pressure approximately 3 psig higher than A. When the sensed pressure signal to EHC pressure regulator A rises, it begins to open TCVs in an attempt to lower pressure. This causes actual Reactor pressure and PAM pressure to lower. EHC pressure regulator B does not take control because Reactor pressure lowers. EHC pressure regulator A will continue to lower Reactor pressure until 850 psig, when the MSIVs will close and the Reactor will scram.

B. Incorrect – The Reactor scrams. Plausible because on the opposite failure, EHC pressure regulator B will take control and prevent a Reactor scram.
C. Incorrect – TCVs open further. Plausible because on the opposite failure, TCVs would close further.
D. Incorrect – TCVs open further. Plausible because on the opposite failure, TCVs would close further. The Reactor scrams. Plausible because on the opposite failure, EHC pressure regulator B will take control and prevent a Reactor scram.

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

Proposed references to be provided to applicants during examination: None

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

Question Source: Modified Bank – LOC28R (2017) NRC #35

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 2/9/18 – Deleted “just”, changed “drifts” to “fails”, and deleted 2nd bullet, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	259001 A3.06
	Importance Rating	3.1

Reactor Feedwater System

Ability to monitor automatic operations of the REACTOR FEEDWATER SYSTEM including: Pump discharge pressure

Proposed Question: #63

Unit 1 is operating at 100% power when the following occurs:

- A manual Reactor scram is inserted.
- Reactor water level reaches a low of +5" before beginning to slowly rise.
- HPCI and RCIC remain in standby.
- NO operator actions have been taken with Feedwater or Condensate.

Which one of the following describes how the Feedwater pumps are operating ten (10) minutes later?

A. All three Feedwater pumps are injecting to the Reactor with speed being controlled based on a Reactor water level signal.

B. All three Feedwater pumps are injecting to the Reactor with speed being controlled based on a pump discharge pressure signal.

C. Only one Feedwater pump is injecting to the Reactor with speed being controlled based on a Reactor water level signal.

D. Only one Feedwater pump is injecting to the Reactor with speed being controlled based on a pump discharge pressure signal.

Proposed Answer: D

Explanation: With Reactor power initially at 100%, all three Feedwater pumps are operating with speeds controlled based on a Reactor water level signal. Reactor water level lowering below +13" causes the Setpoint Setdown logic to initiate. This logic shifts one Feedwater pump to the discharge pressure mode and shifts the other two Feedwater pumps to idle mode. The pumps in idle mode are not injecting to the Reactor. The pump in discharge pressure control mode has its speed controlled by a discharge pressure signal, while a Reactor water level signal is used to control injection by modulating Low Load Valve LV-10641.

Note: The question meets the K/A by testing knowledge of how Feedwater pump control automatically shifts modes following a scram. The automatic operation tested adjusts Feedwater pump speed to control pump discharge pressure.

A. Incorrect – Only one Feedwater pump is injecting to the Reactor with speed being controlled based on a pump discharge pressure signal. Plausible because this is the initial Feedwater lineup and would continue if Reactor water level did not lower below +13".
B. Incorrect – Only one Feedwater pump is injecting to the Reactor with speed being controlled based on a pump discharge pressure signal. Plausible because initially all three Feedwater pumps are injecting and one of the pumps is automatically swapped to the discharge pressure mode.
C. Incorrect – Only one Feedwater pump is injecting to the Reactor with speed being controlled based on a pump discharge pressure signal. Plausible because initially this Feedwater pump is being controlled by a Reactor water level signal and LV-10641 is still being controlled by a Reactor water level signal.

Technical Reference(s): OP-145-001, TM-OP-045

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-045 RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 2/6/18 – Added K/A match statement based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	272000 K1.08
	Importance Rating	3.6

Radiation Monitoring

Knowledge of the physical connections and/or cause-effect relationships between RADIATION MONITORING SYSTEM and the following: Reactor protection system

Proposed Question: #64

Unit 1 is operating at 100% power when RPS Bus A de-energizes due to a sustained electrical fault.

Which one of the following identifies a radiation monitor that is affected by this electrical loss and how the radiation monitor fails?

A. Service Water Effluent; fails tripped

B. Service Water Effluent; fails NOT tripped

C. Refuel Floor High Exhaust; fails tripped

D. Refuel Floor High Exhaust; fails NOT tripped

Proposed Answer: C

Explanation: RPS Bus A supplies power to Refuel Floor High Exhaust radiation monitors. On loss of power, these radiation monitors fail tripped.

A. Incorrect – Service Water Effluent radiation monitor is powered by 24 VDC Bus B. Plausible because many radiation monitors are powered from RPS.
B. Incorrect – Service Water Effluent radiation monitor is powered by 24 VDC Bus B. Plausible because many radiation monitors are powered from RPS. The Main Steam Line radiation monitors fail tripped. Plausible because some recorders fail as-is on loss of power.
D. Incorrect – The Refuel Floor High Exhaust radiation monitors fail tripped. Plausible because some recorders fail as-is on loss of power.

Technical Reference(s): TM-OP-079E

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-079E RBO-3

Question Source: Bank - PB 2017 NRC #31

Question History: PB 2017 NRC #31

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:
TRH 2/6/18 – Changed MSL rad monitors to Refuel Floor High Exhaust rad monitors, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	286000 K4.03
	Importance Rating	3.3

Fire Protection

Knowledge of FIRE PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Maintenance of fire header pressure

Proposed Question: #65

Both Units are operating at 100% power when the following occurs:

- A fire causes a loss of 125 VDC distribution panel 1D625.
- Fire Water header pressure lowers to 90 psig.

Which one of the following describes the response of the Motor Driven and Diesel Engine Driven Fire pumps?

A. Both the Motor Driven and Diesel Engine Driven Fire pumps start.

B. The Motor Driven Fire pump starts, only.

C. The Diesel Engine Driven Fire pump starts, only.

D. NEITHER the Motor Driven Fire pump NOR the Diesel Engine Driven Fire pump starts.

Proposed Answer: A

Explanation: 1D625 provide the alternate power supply to the fire detection and initiation circuitry. Since 1D615 is the normal power supply and is still available, detection and initiation proceeds normally. Both pumps start based on header pressure <100 psig.

B. Incorrect – The Diesel Engine Driven Fire pump starts. Plausible because this would be correct if header pressure only lowered due to jockey pump trip, because the Motor Driven Fire pump starts first and prevents pressure from lowering enough to start the Diesel Engine Driven Fire pump.
C. Incorrect – The Motor Driven Fire pump starts. Plausible if candidate believes loss of 1D625 prevents initiation or believes auto-start pressure is lower.
D. Incorrect – Both pumps start. Plausible if candidate believes loss of 1D625 prevents initiation or believes auto-start pressure is lower.

Technical Reference(s): OP-013-001, LA-0526

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-013 RBO-4

Question Source: Bank – LOC24 (12/2011) NRC #62

Question History: LOC24 (12/2011) NRC #62

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.29
	Importance Rating	4.1

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

Proposed Question: #66

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

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

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

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

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

A. (1) only

B. (2) only

C. (3) only

D. (1) and (3) only

Proposed Answer: A

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

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: AD044, Obj 14821

Question Source: Bank – LOC28R (2017) NRC #67

Question History: LOC28R (2017) NRC #67

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.3
	Importance Rating	3.7

Knowledge of shift or short-term relief turnover practices

Proposed Question: #67

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

- Shift turnover is in progress.
- You are one of the two oncoming PCOs.
- This is your third consecutive day shift as PCO.

Which one of the following describes

(1) how many of the oncoming PCOs are required to walk-down the Control Room panels with an offgoing PCO, and

(2) the need to review your TMX qualifications,

in accordance with OP-AD-003, Shift Surveillance Scheduling, Log Sheets, Turnover Sheets and Rounds?

	Required to Walk-down the Control Room Panels with an Offgoing PCO	TMX Qualification Review
A.	Only one oncoming PCO	NOT required
B.	Only one oncoming PCO	Required prior to taking the shift
C.	Both oncoming PCOs	NOT required
D.	Both oncoming PCOs	Required prior to taking the shift

Proposed Answer: D

Explanation: OP-AD-003 Attachment G requires both oncoming PCOs to complete a Control Room panel walk-down with an offgoing PCO. The procedure also requires reviewing TMX qualifications prior to taking each shift.

A. Incorrect – Both oncoming PCOs must walk-down the Control Room panels with an offgoing PCO. Plausible because only one PCO is required to be in the Control Room at a time and the two PCOs in general have different responsibilities. TMX qualifications must be reviewed prior to taking the shift. Plausible becausethis is the third consecutive day shift, such that qualifications were already recently reviewed and should not change day to day.
B. Incorrect – Both oncoming PCOs must walk-down the Control Room panels with an offgoing PCO. Plausible because only one PCO is required to be in the Control Room at a time and the two PCOs in general have different responsibilities.
C. Incorrect – TMX qualifications must be reviewed prior to taking the shift. Plausible because this is the third consecutive day shift, such that qualifications were already recently reviewed and should not change day to day.

Technical Reference(s): OP-AD-003

Proposed references to be provided to applicants during examination: None

Learning Objective: AD044, Obj 14798

Question Source: Modified Bank – LOC25 (2013) Cert #75

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.35
	Importance Rating	3.6

Ability to determine Technical Specification Mode of Operation.

Proposed Question: #68

A Unit 1 shutdown is in progress with the following:

- The Reactor Mode Switch has been transferred from RUN to STARTUP/STBY.
- Operators are inserting control rods using RMCS.
- Reactor power is:
  - o 5% on APRMs.
  - o Mid-scale on IRM Range 9.
- Reactor pressure is 975 psig.

Which one of the following is the Technical Specification Mode of Operation?

A. Mode 1

B. Mode 2

C. Mode 3

D. Mode 4

Proposed Answer: B

Explanation: Technical Specification Table 1.1-1 contains the criteria for Reactor Mode of Operation. With the Mode Switch in STARTUP/STBY, the Reactor is in Mode 2 (STARTUP).

A. Incorrect – With the Reactor Mode Switch not in RUN, the plant is not in Mode 1. Plausible because the plant was in Mode 1 just prior to the movement of the Reactor Mode Switch.
C. Incorrect – With control rods still not fully inserted, the plant is not yet in Mode 3. Plausible because if the Mode Switch was placed in SHUTDOWN instead of STARTUP/STBY, the plant would be in Mode 3. Also plausible because a plant shutdown is in progress.
D. Incorrect – With control rods still not fully inserted and the Reactor still at relatively high pressure, the plant is not yet in Mode 4. Plausible because a plant shutdown is in progress, the Mode Switch is not in RUN, and Reactor pressure is below the normal full power value.

Technical Reference(s): Technical Specification Table 1.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – LOC26R (1/2015) NRC #69

Question History: LOC26R (1/2015) NRC #69

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:
TRH 2/9/18 – Deleted noun names in choices, based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.13
	Importance Rating	4.1

Knowledge of tagging and clearance procedures.

Proposed Question: #69

Which one of the following identifies a tagging configuration that is prohibited simultaneously on the same component, in accordance with NDAP-QA-0322, Clearance and Tagging?

A. Two Caution Tags

B. A Danger Tag and a Caution Tag

C. A Danger Tag and a Test and Maintenance Tag

D. A Caution Tag and a Test and Maintenance Tag

Proposed Answer: C

Explanation: NDAP-QA-0322 does not allow a Danger Tag and a Test and Maintenance Tag to hang simultaneously on the same component.

A. Incorrect – Two Caution Tags are allowed to hang simultaneously on the same component. Plausible because some Tag types are only allowed to have a single Tag on one component.
B. Incorrect – A Danger Tag and a Caution Tag are allowed to hang simultaneously on the same component. Plausible because some different Tag types are not allowed to hang simultaneously on one component.
D. Incorrect – A Caution Tag and a Test and Maintenance Tag are allowed to hang simultaneously on the same component. Plausible because some different Tag types are not allowed to hang simultaneously on one component.

Technical Reference(s): NDAP-QA-0322

Proposed references to be provided to applicants during examination: None

Learning Objective: AD044, Obj 15063

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.41
	Importance Rating	3.5

Ability to obtain and interpret station electrical and mechanical drawings.

Proposed Question: #70

Unit 1 is operating at 100% power when SRV A opens due to a spurious signal from the Reactor overpressure relief logic.

Note: A portion of the associated electrical drawing is provided on the following page. Red marks on the drawing identify the cause of the spurious signal.

Given the following possible independent methods to close SRV A:

(1) Placing the SRV A control switch to OFF.
(2) Pull the SRV A fuses (B21C-F019 & B21C-F020) at Upper Relay Room panel 1C628.

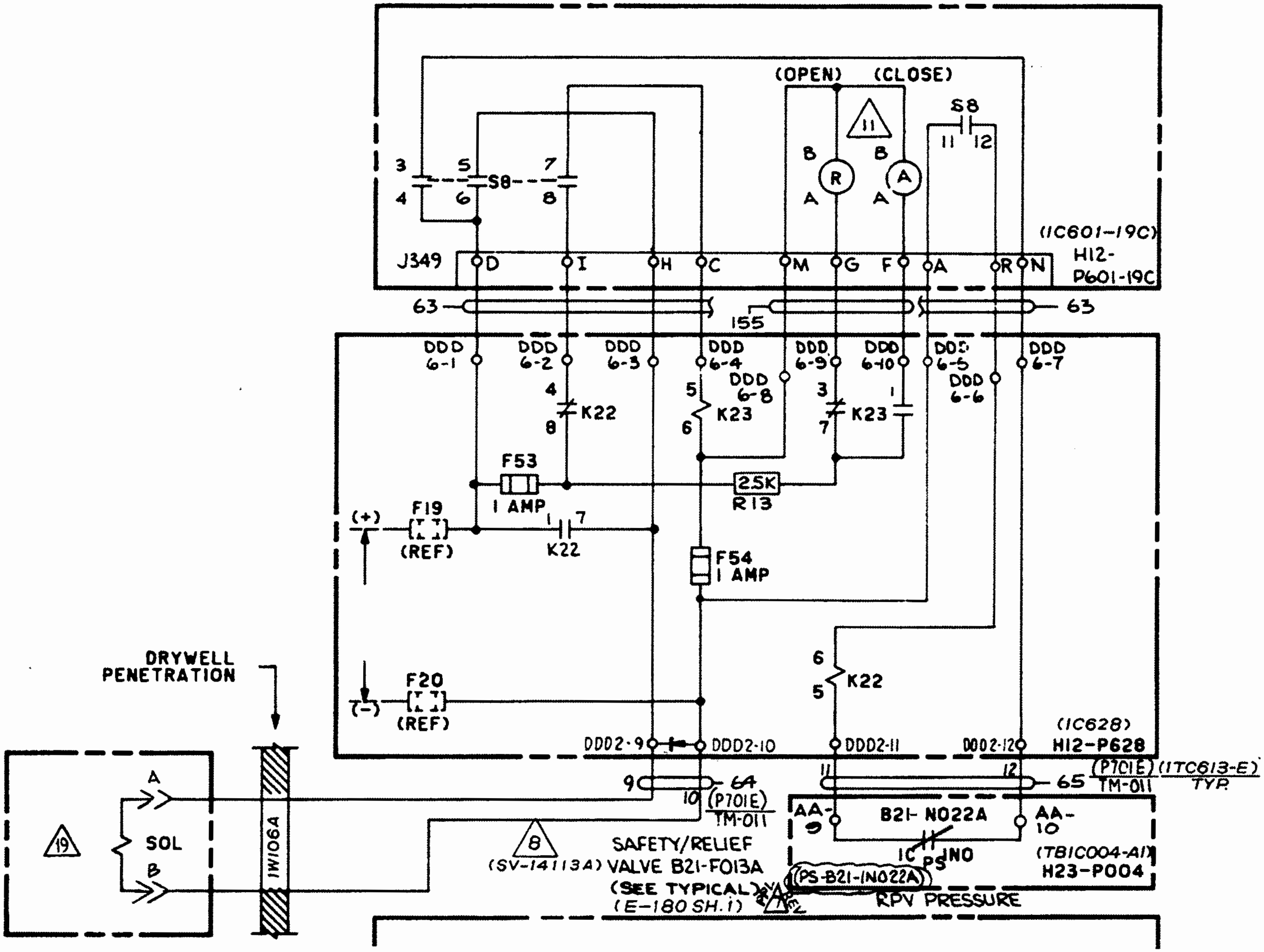
Which one of the following identifies which of these methods, if any, will close SRV A?

A. NEITHER method (1) NOR method (2) will close SRV A.

B. Method (1) will close SRV A, but method (2) will NOT.

C. Method (1) will NOT close SRV A, but method (2) will.

D. Both method (1) and method (2) will close SRV A.



(OPEN)
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DDD2-9
DDD2-10
DDD2-11
DDD2-12
(IC628)
H12-P628
64
(P701E)
TM-011
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TM-011
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TYP
SOL
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19
8
(SV-14113A)
SAFETY/RELIEF
VALVE B21-F013A
(SEE TYPICAL)
(E-180 SH.1)
AA-9
B21- N022A
AA-10
1C
PS
NO
(TBIC004-A1)
H23-P004
(PS-B21-1N022A)
RPV PRESSURE

Proposed Answer: D

Explanation: With the given fault in the overpressure relief logic, pulling SRV A fuses (B21C-F019 & B21C-F020) will isolate the spurious signal and result in the SRV closing. Taking the SRV control switch to OFF will also close the SRV because this switch is in-line with the fault.

A. Incorrect – Taking the SRV control switch to OFF will close the SRV because this switch is in-line with the fault. Plausible if the candidate believes this opens the contacts labeled "S8" in the given drawing. Pulling SRV A fuses (B21C-F019 & B21C-F020) will isolate the spurious signal and result in the SRV closing. Plausible because this would not close the SRV if the fault were in some other part of the logic.
B. Incorrect – Pulling SRV A fuses (B21C-F019 & B21C-F020) will isolate the spurious signal and result in the SRV closing. Plausible because this would not close the SRV if the fault were in some other part of the logic.
C. Incorrect – Taking the SRV control switch to OFF will close the SRV because this switch is in-line with the fault. Plausible because this would not close the SRV if the fault were in some other part of the logic.

Technical Reference(s): ON-SRV-101, SRV schematic

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-083 RBO-3

Question Source: Bank - JAF 16-1 NRC #69

Question History: JAF 16-1 NRC #69

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:
TRH 2/6/18 – Redacted power supply information from drawing based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	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: #71

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

- Annunciator AR-101-B05, RX BLDG AREA PANEL 1C605 HI RADIATION, alarms.
- The HPCI room area radiation monitor (ARM) is in alarm high.

Which one of the following identifies indications available locally in the Reactor Building?

	Warning Light	Meter Indication
A.	Available	Available
B.	Available	NOT available
C.	NOT available	Available
D.	NOT available	NOT available

Proposed Answer: A

Explanation: This ARM channel, like most other ARM channels, has a local alarm and auxiliary unit in the Reactor Building. This provides a warning light and also radiation level indication on an auxiliary unit meter.

B. Incorrect – Both of these indications are available locally in the Reactor Building. Plausible because meter indications for many parameters are limited in the plant.
C. Incorrect – Both of these indications are available locally in the Reactor Building. Plausible because local warning is desirable for safety purposes and other means of warning people in the plant of safety issues are audible only (evacuation alarms, for example).
D. Incorrect – Both of these indications are available locally in the Reactor Building. Plausible because local warning is desirable for safety purposes and other means of warning people in the plant of safety issues are audible only (evacuation alarms, for example). Plausible because meter indications for many parameters are limited in the plant.

Technical Reference(s): TM-OP-079B

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:
TRH 2/11/18 – Replaced question to keep more generic to ARMs in general, based on NRC comment.
TRH 2/22/18 – Deleted 2nd column based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.13
	Importance Rating	3.4

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: #72

A transient is in progress with the following:

- Operators are required to enter a radiologically posted area in order to manually close Primary Containment Isolation Valves.
- Radiation Physics (RP) has determined that the highest dose rate in the area is 750 mRem/hr.

Which one of the following describes the radiological posting requirement for the area and the need for the Operators to be continuously escorted by RP personnel, in accordance with NDAP-QA-0626, Radiologically Controlled Area Access and Radiation Work Permit (RWP) System?

This area is required to be posted as a __(1)__. The Operators __(2)__ required to be continuously escorted by RP personnel while in this area.

	(1)	(2)
A.	High Radiation Area	are
B.	High Radiation Area	are NOT
C.	Locked High Radiation Area	are
D.	Locked High Radiation Area	are NOT

Proposed Answer: B

Explanation: An area that has general dose rates above 100 mRem/hr but less than 1000 mRem/hr is classified as a High Radiation area. An RP escort would initially be required if dose rates have not already been determined. Since RP has already surveyed the area, the Operators do NOT need to have a continuous RP escort while in the area.

A. Incorrect – Continuous escort by RP is not required. Plausible because this is a High Radiation area and continuous escort would be required if dose rates had not already been determined.
C. Incorrect – This is not a Locked High Radiation area. Plausible because dose rates are well above the 100 mRem/hr threshold for a High Radiation area. Continuous escort by RP is not required. Plausible because this is a High Radiation area and continuous escort would be required if dose rates had not already been determined.
D. Incorrect – This is not a Locked High Radiation area. Plausible because dose rates are well above the 100 mRem/hr threshold for a High Radiation area.

Technical Reference(s): NDAP-QA-0626

Proposed references to be provided to applicants during examination: None

Learning Objective: AD044, Obj 15105

Question Source: Bank – LOC27 (2015) NRC #70

Question History: LOC27 (2015) NRC #70

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(12)

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.4
	Importance Rating	4.5

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

Proposed Question: #73

A transient on Unit 1 has resulted in the following:

- Reactor water level is +18", up slow.
- Reactor pressure is 1060 psig, up slow.
- Reactor power is downscale on APRMs.
- Drywell pressure is 1.4 psig, up slow.
- Drywell average temperature is 145°F, up slow.
- Suppression Pool water level is 23.7 feet, up slow.
- Suppression Pool temperature is 82°F, up slow.
- Annunciator AR-113-H01, CORE SPRAY LOOP B PUMP ROOM FLOODED, is in alarm.
- Annunciator AR-107-A05, DRYWELL FLOOR DRN SUMP A HI-HI LEVEL, is in alarm.
- An Operator in the field has confirmed this alarm is valid and that there is approximately 1" of water is on the floor.
- An Operator in the field has confirmed that the Core Spray loop B pump room door is closed.

Given the following procedures:

(1) ON-DWLEAK-101, Drywell Leakage
(2) ON-FLOOD-101, Flooding in Turbine or Reactor Building
(3) EO-100-102, RPV Control
(4) EO-100-103, Primary Containment Control
(5) EO-100-104, Secondary Containment Control

Which one of the following identifies which of these given procedures are required to be entered based on current conditions?

A. (1) and (4) only

B. (2) and (5) only

C. (1), (2), and (5) only

D. (1), (2), (3), (4), and (5)

Proposed Answer: C

Explanation: ON-DWLEAK-101 entry is required because Drywell pressure and temperature are rising and AR-112-D03 is in alarm based on Drywell pressure >1.0 psig. ON-FLOOD-101 entry is required because AR-113-H01 is in alarm and confirmed. EO-100-102 entry is not required because Reactor water level is >+13", Reactor pressure is <1087 psig, and Drywell pressure is <1.72 psig. EO-100-103 entry is not required because SP temperature is <90°F, SP water level is >22', Drywell temperature is <150°F, and Drywell pressure is <1.72 psig. EO-100-104 entry is required because AR-113-H01 is in alarm and confirmed.

A. Incorrect – ON-FLOOD-101 entry is required because AR-113-H01 is in alarm and confirmed. Plausible because leakage is contained to only the Core Spray room and there is only 1" of water on the floor. EO-100-103 entry is not required because SP temperature is <90°F, SP water level is >22', Drywell temperature is <150°F, and Drywell pressure is <1.72 psig. Plausible because Drywell temperature, Drywell pressure, and SP temperature are all elevated and rising. EO-100-104 entry is required because AR-113-H01 is in alarm and confirmed. Plausible because leakage is contained to only the Core Spray room and there is only 1" of water on the floor.
B. Incorrect – ON-DWLEAK-101 entry is required because Drywell pressure and temperature are rising and AR-112-D03 is in alarm based on Drywell pressure >1.0 psig. Plausible because Drywell pressure and temperature are below EO-100-103 entry conditions.
D. Incorrect – EO-100-102 entry is not required because Reactor water level is >+13", Reactor pressure is <1087 psig, and Drywell pressure is <1.72 psig. Plausible because Reactor water level is low, Reactor pressure is high, and Drywell pressure is high. EO-100-103 entry is not required because SP temperature is <90°F, SP water level is >22', Drywell temperature is <150°F, and Drywell pressure is <1.72 psig. Plausible because Drywell temperature, Drywell pressure, and SP temperature are all elevated and rising.

Technical Reference(s): ON-DWLEAK-101, ON-FLOOD-101, EO-100-102, EO-100-103, EO-100-104

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-EO-102, 103, 104 Obj 2

Question Source: Modified Bank – LOC26R (1/2015) NRC #72

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

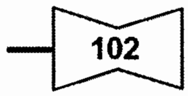
Comments:
TRH 2/6/18 – Revised wording of last bullet based on NRC comment.

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.19
	Importance Rating	3.4

Knowledge of EOP layout, symbols, and icons.

Proposed Question: #74

The crew has reached a step in EO-100-103, Primary Containment Control, attached to the following symbol:



Which one of the following describes this symbol?

A. Provides direction to immediately enter EO-100-102, RPV Control.

B. Provides direction to preferentially use Turbine Bypass Valves for Reactor pressure control.

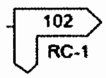
C. Serves as a reminder to check limits associated with injection sources being used in EO-100-102, RPV Control.

D. Serves as a reminder that Turbine Bypass Valves may be used to anticipate Emergency RPV Depressurization.

Proposed Answer: D

Explanation: The given symbol is used with steps that lead to Emergency RPV Depressurization and serves as a reminder that EO-100-102 contains a step that allows anticipating Emergency RPV Depressurization using Turbine Bypass Valves.

A. Incorrect – This symbol serves as a reminder that Turbine Bypass Valves may be used to anticipate Emergency RPV Depressurization. Plausible because EO-100-103 uses a similar symbol to indicate need for immediate entry into EO-100-102:



B. Incorrect – This symbol serves as a reminder that Turbine Bypass Valves may be used to anticipate Emergency RPV Depressurization. Plausible because the symbol is related to use of Turbine Bypass Valves and EO-100-103 does contain legs in which Turbine Bypass Valves are preferential for Reactor pressure control (such as high SP temperature).
C. Incorrect – This symbol serves as a reminder that Turbine Bypass Valves may be used to anticipate Emergency RPV Depressurization. Plausible because there are multiple warnings/cautions in EO-100-103 that reference limits on RHR, Core Spray, HPCI, and RCIC operation.

Technical Reference(s): NDAP-QA-0330

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-EO102, Obj 2

Question Source: Bank – LOC23 (2/2011) Cert #73

Question History: LOC23 (2/2011) Cert #73

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.50
	Importance Rating	4.2

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

Proposed Question: #75

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

Time (hhmm)	Condition
1400	• ON-SW-101, Loss of Service Water, is entered.
1415	• ON-SCRAM-101, Reactor Scram, is entered and the Reactor is manually scrammed.
1417	• EO-100-102, RPV Control, is entered.

Note: The Unit Supervisor has NOT made an announcement regarding "Transient Actions in Effect" during this transient.

Which one of the following describes the required method of alarm management during this transient, in accordance with OP-AD-300, Administration of Operations?

A. Normal alarm management remains in effect throughout the transient.

B. Transient condition alarm management automatically goes into effect at time 1400.

C. Transient condition alarm management automatically goes into effect at time 1415.

D. Transient condition alarm management automatically goes into effect at time 1417.

Proposed Answer: C

Explanation: OP-AD-300 provides two conditions for implementing transient condition alarm management. The Unit Supervisor can implement it by announcing "Transient Actions in Effect" using a crew update. Additionally, it automatically goes into effect when a Reactor scram occurs. Therefore, transient condition alarm management automatically goes into effect at time 1415.

Note: The question meets the K/A by testing general administrative requirements for alarm response, including knowing the requirements for how/when transient alarm response standards become active, which includes different expectations for verifying setpoints and operating any identified controls in the alarm response manuals.

A. Incorrect – Transient condition alarm management automatically goes into effect at time 1415. Plausible because the Unit Supervisor has NOT made an announcement regarding "Transient Actions in Effect" during this transient, and that is one method for entering transient condition alarm management.
B. Incorrect – Transient condition alarm management automatically goes into effect at time 1415, not 1400. Plausible because this is the first time an ON is entered.
D. Incorrect – Transient condition alarm management automatically goes into effect at time 1415, not 1417. Plausible because this is the first time an EOP is entered.

Technical Reference(s): OP-AD-300

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-AD-300, Obj 1

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	1
K/A #	295006 AA2.04
Importance Rating	4.1

SCRAM

Ability to determine and/or interpret the following as they apply to SCRAM: Reactor pressure

Proposed Question: #76

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

- I&C reports that a document review has revealed the high Reactor pressure scram setpoints are set according to the table below:

Instrument	Setpoint
PS-B21-1N023A	1078 psig
PS-B21-1N023B	1082 psig
PS-B21-1N023C	1098 psig
PS-B21-1N023D	1102 psig

Which one of the following describes the ***most restrictive requirement***, if any, based on these setpoints, in accordance with Technical Specification (TS) 3.3.1.1, Reactor Protection System (RPS) Instrumentation?

TS 3.3.1.1...

A. condition entry is NOT required.

B. requires placing a channel or the associated trip system in trip within a maximum of 12 hours.

C. requires placing a channel in one trip system in trip or trip system in trip within a maximum of 6 hours.

D. requires restoring trip capability within 1 hour, or then be in Mode 3 within 12 hours.

Proposed Answer: C

Explanation: TS 3.3.1.1 Table 3.3.1.1-1 Function 3 requires all four of these Reactor pressure inputs to be set ≤1093 psig. Therefore, both PS-B21-1N023C and PS-B21-1N023D are inoperable. PS-B21-1N023C inputs to RPS trip system A and PS-B21-1N023D inputs to RPS trip system B. TS 3.3.1.1 Conditions A and B are required to be entered, with the more restrictive Condition B requiring placing a channel in one trip system in trip or trip system in trip within a maximum of 6 hours.

Note: The provided reference must have the "required channels per trip system" column of Table 3.3.1.1-1 for APRMs redacted to prevent giving information required in Question 87.

A. Incorrect – TS 3.3.1.1 Conditions A and B are required to be entered. Plausible because this would be correct if all setpoints were ≤1093 psig. Also plausible because this would be the correct answer if PS-B21-1N023C were set slightly lower and TS 3.3.1.1 allowed one inoperable instrument without requiring Condition entry.
B. Incorrect – The most restrictive requirement is placing a channel in one trip system in trip or trip system in trip within a maximum of 6 hours. Plausible because this is also a requirement, just not the most restrictive. Also plausible because this would be correct if PS-B21-1N023C were set slightly lower.
D. Incorrect – The most restrictive requirement is placing a channel in one trip system in trip or trip system in trip within a maximum of 6 hours. Plausible because this would be correct if the two inoperable instruments input to the same trip system (A and C, or B and D).

Technical Reference(s): AR-103-B02, AR-104-B02, TS 3.3.1.1

Proposed references to be provided to applicants during examination: TS 3.3.1.1 (with allowable values except 1093 psig removed, and APRM required channels removed)

Learning Objective: TM-OP-058 RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295016 AA2.02
	Importance Rating	4.3

Control Room Abandonment

Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Reactor water level

Proposed Question: #77

Unit 1 is initially operating at 100%. A Control Room evacuation is being performed due to a fire in the Main Control Room with the following:

- ON-CREVAC-101, Control Room Evacuation, has been entered.
- All required actions were completed prior to exiting the Control Room.
- Reactor water level control has just been established at the Remote Shutdown Panel.
- Reactor water level is +25" and stable.

Which one of the following describes the status of Reactor water level relative to the required band and the procedure that specifies the required band?

Reactor water level is...

A. in the required band as specified by ON-CREVAC-101.

B. in the required band as specified by EO-100-102, RPV Control.

C. outside of the required band as specified by ON-CREVAC-101.

D. outside of the required band as specified by EO-100-102, RPV Control.

Proposed Answer: B

Explanation: ON-CREVAC-101 requires control of Reactor water level using RCIC from the Remote Shutdown Panel. ON-CREVAC-101 does not contain a specific Reactor water level band. ON-CREVAC-101 assumes EO-100-102 is utilized concurrently to control Reactor water level. Therefore, Reactor water level is required to be controlled +13" to +54" based on the band in EO-100-102

Note: The question meets SRO level guidelines because it requires assessment of abnormal plant conditions and determining what procedure(s) are appropriate to perform, including specific guidance contained in these procedure(s).

A. Incorrect – ON-CREVAC-101 does not contain a specific Reactor water level band. Plausible that ON-CREVAC-101 would contain a specific Reactor water level band due to the integrated nature of the procedure and the extremely abnormal conditions under which it would be implemented.
C. Incorrect – Reactor water level is in the required band of +13" to +54". Plausible that a higher band would be required based on either circulation requirements, inventory preservation, or in preparation for placing SDC in service (≥+45" is used in ON-SDC-101). ON-CREVAC-101 does not contain a specific Reactor water level band. Plausible that ON-CREVAC-101 would contain a specific Reactor water level band due to the integrated nature of the procedure and the extremely abnormal conditions under which it would be implemented.
D. Incorrect – Reactor water level is in the required band of +13" to +54". Plausible that a higher band would be required based on either circulation requirements, inventory preservation, or in preparation for placing SDC in service (≥+45" is used in ON-SDC-101).

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

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-EO102 Obj 2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295021 2.4.31
	Importance Rating	4.1

Loss of Shutdown Cooling

Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: #78

Unit 1 is operating in Mode 3 with the following:

- RHR pump 1A is operating in the Shutdown Cooling mode.
- RHR pumps 1B and 1D are unavailable due to maintenance.

Then, annunciator AR-109-H08, RHR LOOP A PUMP ROOM FLOODED, alarms with the following:

- Operators report there is an isolable leak in the RHR suction piping between HV-151-F006A, SHUTDOWN CLG SUCT, and HV-151-F008, SHUTDOWN CLG SUCT OB ISO.
- The floor drain isolation valve for RHR Loop A Pump Room has been opened.
- Operators report RHR Loop A Pump Room water level is 10", up slow.
- Operators have exited the RHR Loop A Pump Room and the watertight door has been closed.

Which one of the following describes the required control of RHR, in accordance with the applicable Off Normal Procedures (ONs) and Emergency Operating Procedures (EOPs)?

A. Isolate the leak now.

B. Maintain the leak un-isolated until water level exceeds the Max Safe value in RHR Loop A Pump Room, then isolate the leak.

C. Maintain the leak un-isolated unless water level exceeds the Max Safe value in RHR Loop A Pump Room AND at least one other Reactor Building area, then isolate the leak.

D. Maintain the leak un-isolated regardless of RHR Loop A Pump Room water level, but isolate the leak if Suppression Pool water level CANNOT be maintained above 22'.

Proposed Answer: A

Explanation: AR-109-H08 alarms when RHR Loop A Pump Room water level reaches 3.25" and requires entry into ON-FLOOD-101, Flooding in Turbine or Reactor Building, and EO-100-104, Secondary Containment Control. Both of these procedures contain conditional steps to control isolation of the leak. EO-100-104 requires isolation of the leak now because the Hi level alarm has been exceeded, available sump pumps have been aligned to the room by opening the room drain isolation valve, water level is still above the Hi level alarm and rising, and RHR loop A is not needed to support EOP/DSP actions or damage control. Similarly, ON-FLOOD-101 requires isolation of the leak now because RHR loop A is not needed to shutdown the Reactor, assure adequate core cooling, suppress a fire, or prevent primary containment failure. Loss of Shutdown Cooling complicates Reactor coolant temperature control, but ON-SDC-101, Loss of Shtudown Cooling, provides other methods of decay heat removal such that adequate core cooling is assured unless further failures occur.

Note: The question meets the K/A by presenting a failure that results in a partial loss of Shutdown Cooling (leak from running loop) and testing control of the loop that results in a complete loss of Shutdown Cooling (isolation will require closing HV-151-F008(9)) based on occurrence of an alarm. The alarm and related indication (water level) must be understood. Additionally, the applicable response procedures must be properly understood. In this case, ON-FLOOD-101 and EO-100-104 are extensions of the alarm response procedure based on how the facility's procedures are arranged.

Note: The question meets SRO level guidelines because it requires assessment of abnormal plant conditions (leakage that affects all available Shutdown Cooling) and determining when to select a section of a procedure (portions of ON-FLOOD-101 and EO-100-104 that controls isolation of leak). The question additionally tests ability of the SRO to prioritize competing issues (isolation fixes the leak issue, but exacerbates the loss of Shutdown Cooling).

B. Incorrect – EO-100-104 and ON-FLOOD-101 require isolating the leak now. Plausible because water level is still well below the Max Safe level of 84" and isolation of the leak will result in a loss of all Shutdown Cooling.
C. Incorrect – EO-100-104 and ON-FLOOD-101 require isolating the leak now. Plausible because water level is still well below the Max Safe level of 84" and isolation of the leak will result in a loss of all Shutdown Cooling. Also plausible because other conditional steps further in EO-100-104 require a 2nd Max Safe level to exceeded.
D. Incorrect – EO-100-104 and ON-FLOOD-101 require isolating the leak now. Plausible because water level is still well below the Max Safe level of 84" and isolation of the leak will result in a loss of all Shutdown Cooling. Also plausible because the watertight door is sealed, which should limit the extent of flooding, and a Suppression Pool water level of 22' indicates EO-100-103 entry is required.

Technical Reference(s): AR-109-H08, ON-FLOOD-101, EO-100-104, EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-049 RBO-7

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:
TRH 2/6/18 – Revised value in D from 17' to 22' to avoid any conflict with Question 28, based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295023 AA2.04
	Importance Rating	4.1

Refueling Accidents

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Occurrence of fuel handling accident

Proposed Question: #79

A refueling outage is in progress on Unit 1 with the following:

- Core shuffle phase II is in progress.
- A four cell section of the core is displayed on the following page.

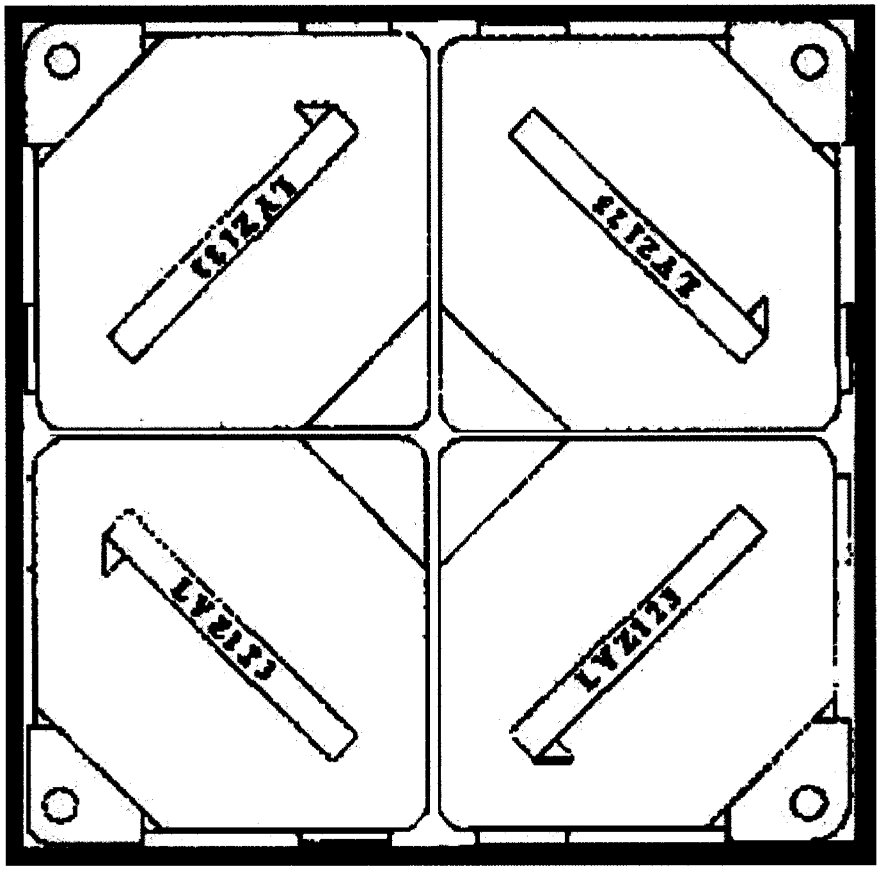
Which one of the following describes the status of this portion of the core and the required action(s), if any, in accordance with OP-0RF-008, Fuel and Blade Guide Handling Activities?

A. The fuel is loaded correctly. Core shuffle may continue with no additional required actions.

B. A discrepancy exists in the fuel loading. Core shuffle may continue without interruption as long as the discrepancy is fixed prior to startup.

C. A discrepancy exists in the fuel loading. The discrepancy must be immediately fixed and then fuel movements may continue with permission from the Refuel Floor Manager.

D. A discrepancy exists in the fuel loading. Fuel movements must be immediately stopped. Permission from the Assistant Operations Manager – Shift/WCC or Manager – Nuclear Operations is required before fuel movement may resume.


LYZ123
LYZ123
LYZ123
LYZ123

Proposed Answer: D

Explanation: All four fuel bundles are oriented incorrectly, as evidenced by the bail handle indicator pointing away from the center of the fuel cell. To correct this issue, all fuel bundles need to be re-positioned 180°. OP-0RF-081 requires immediately stopping fuel handling. This error meets the entry conditions of ON-FUEL-101 for a Refueling Platform Operation Anomaly. ON-FUEL-101 requires entry into AOP-081-001, which requires consulting Reactor Engineering for recommendations on how to proceed, and permission from the Assistant Operations Manager – Shift/WCC or Manager – Nuclear Operations prior to resuming fuel movement.

A. Incorrect – All four fuel bundles are oriented incorrectly, as evidenced by the bail handle indicator pointing away from the center of the fuel cell. Plausible because the error still maintains symmetry within the fuel cell.
B. Incorrect – Fuel movement must be stopped. Plausible because with all rods full inserted and the correct bundles just oriented incorrectly, the risk of inadvertent criticality is very low, such that fixing the discrepancy prior to startup is highly likely to be enough to prevent actual consequence.
C. Incorrect – Fuel movement must be stopped and immediate fixing of the discrepancy is not allowed by procedure (first Reactor Engineering must be consulted and then appropriate permission granted). Plausible because with all rods full inserted and the correct bundles just oriented incorrectly, the risk of inadvertent criticality is very low, and a method of fixing the discrepancy is readily apparent. Also plausible that it is desirable to fix the error quickly to minimize the time with the incorrect configuration.

Technical Reference(s): OP-0RF-008, ON-FUEL-101, AOP-081-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-081A RBO-7

Question Source: Modified Bank – NMP1 2013 NRC #80

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(6)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295025 2.1.7
	Importance Rating	4.7

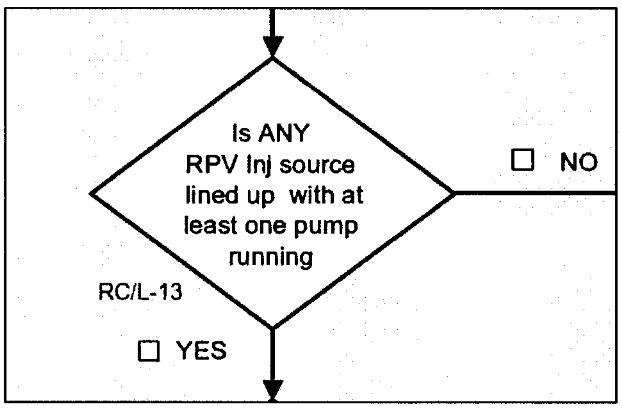
High Reactor Pressure

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

Proposed Question: #80

Unit 1 was operating at 100% power when a transient resulted in the following:

- Reactor water level is -178", down slow.
- Reactor pressure is 850 psig, down slow, with an SRV open.
- NO Reactor injection sources are available.
- EO-100-102, RPV Control, has been executed to the following step:



Which one of the following describes how Reactor pressure is now required to be controlled in accordance with the Emergency Operating Procedures?

A. Stabilize Reactor pressure around the current value.

B. Lower Reactor pressure. Do NOT exceed a cool-down rate of 100°F/hr.

C. Rapidly lower Reactor pressure. The cool-down rate is allowed to exceed 100°F/hr.

D. Close the SRV and allow Reactor pressure to rise until SRVs automatically actuate.

Proposed Answer: A

Explanation: With no injection sources available and Reactor water level -178" and lowering, EO-100-102 step RC/L-20 requires Steam Cooling. Steam Cooling requires stabilizing Reactor pressure, even though it is currently above the capacity of many low pressure injection systems. This is done to ensure the Minimum Zero Injection RPV Water Level calculation assumptions remain valid and Reactor inventory loss is minimized.

Note: Although Reactor pressure is not above the normal power operating range, this question meets the K/A because Reactor pressure is much higher than the injection limit of low pressure systems, which are required to inject if they become available based on Reactor water level.

B. Incorrect – Pressure must be stabilized near the current value. Plausible because lowering Reactor pressure would allow quicker injection with low pressure systems when they become available.
C. Incorrect – Pressure must be stabilized near the current value. Plausible because lowering Reactor pressure would allow quicker injection with low pressure systems when they become available. Also plausible because this would be correct if any injection source were available.
D. Incorrect – Pressure must be stabilized near the current value. Allowing SRVs to automatically actuate would result in Reactor pressure rising more than 100 psig. Plausible because Reactor water level is dangerously low and this would maximally conserve Reactor coolant inventory. Also plausible because the SRV will need to be intermittently closed to stabilize pressure and pressure may be allowed to rise up to 100 psig.

Technical Reference(s): EO-100-102

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-EO102 Obj 2

Question Source: Bank – NMP1 2015 NRC #82

Question History: NMP1 2015 NRC #82

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295030 2.4.47
	Importance Rating	4.2

Low Suppression Pool Water Level

Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: #81

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

- A seismic event caused a Reactor scram.
- Multiple control rods remain full out.
- Reactor water level is +25", steady, with Feedwater injecting.
- Reactor pressure is 700 psig, steady, with an SRV stuck open.
- Suppression Pool water temperature is 90°F, up fast at a rate of 2.0°F/min.
- Suppression Pool Cooling has been maximized.
- Suppression Pool water level is 19.0', down slow at a rate of 0.2'/min.
- Makeup to the Suppression Pool has been maximized.
- Damage control teams are attempting to stop the leakage from the Suppression Pool and close the SRV.

<u>**Note:**</u> Assume the rate of change in Suppression Pool water level and temperature remains constant. The above conditions are given at Time = 0 minutes. The Heat Capacity Temperature Limit is given on the following page.

Which one of the following identifies ***the latest time*** an Emergency RPV Depressurization will become ***required***, in accordance with EO-100-103, Primary Containment Control?

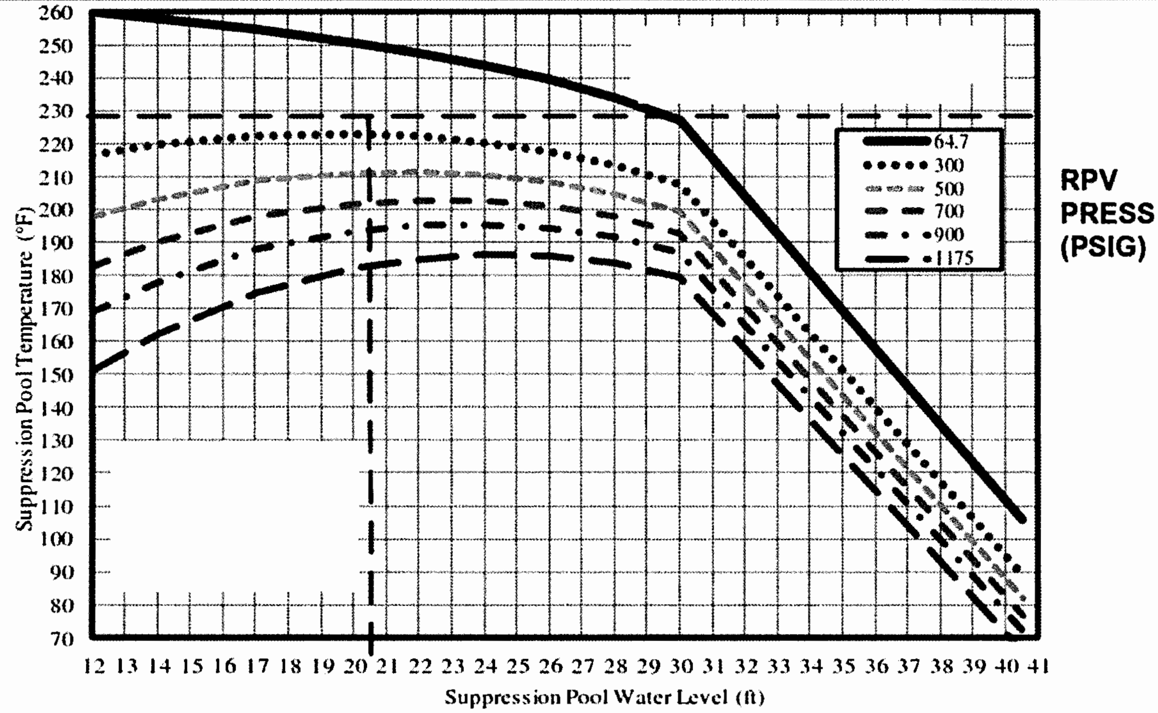
A. 10 minutes

B. 15 minutes

C. 35 minutes

D. 45 minutes

FIG 2 HCTL
HEAT CAPACITY TEMPERATURE LIMIT



Proposed Answer: C

Explanation: EO-100-103 requires Emergency RPV Depressurization either when Suppression Pool water temperature exceeds HCTL or when Suppression Pool water level reaches 12' (at the latest). With the given trends, the first requirement for Emergency RPV Depressurization will be on low Suppression Pool water level at time 35 minutes.

A. Incorrect – With the given trends, the first requirement for Emergency RPV Depressurization will be on low Suppression Pool water level at time 35 minutes. Plausible because at Time 10 minutes, Suppression Pool water level will reach 17', which requires isolation of HPCI.
B. Incorrect – With the given trends, the first requirement for Emergency RPV Depressurization will be on low Suppression Pool water level at time 35 minutes. Plausible because at Time 15 minutes, Suppression Pool water temperature will reach 120°F, which is a value in Technical Specifications that requires depressurization of the Reactor to <200 psig within 12 hours.
D. Incorrect – With the given trends, the first requirement for Emergency RPV Depressurization will be on low Suppression Pool water level at time 35 minutes. Plausible because at Time 45 minutes, Suppression Pool water temperature will reach 180°F, which is the lowest HCTL for a Reactor pressure of 700 psig.

Technical Reference(s): EO-100-103

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-EO103 Obj 3

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295037 2.4.34
	Importance Rating	4.1

Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Proposed Question: #82

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

- Control Room evacuation becomes necessary due to a fire.
- The Control Room is evacuated ***before*** the PCO takes any actions in ON-CREVAC-101, Control Room Evacuation.
- All required Local Actions of ON-CREVAC-101 are performed for response to Control Room evacuation prior to completing all PCO actions.
- Operators have arrived at the Remote Shutdown Panel (RSP), but NO actions have yet been taken at the RSP.

Which one of the following describes status of the plant based on the current place in ON-CREVAC-101?

The Reactor has...

A. NOT been scrammed, MSIVs are open, and Feedwater pumps are in service.

B. been scrammed, MSIVs are open, and Feedwater pumps are in service.

C. been scrammed, MSIVs are open, and Feedwater pumps are tripped.

D. been scrammed, MSIVs are closed, and Feedwater pumps are tripped.

Proposed Answer: D

Explanation: With no Control Room Initial Response actions taken, ON-CREVAC-101 requires the following Local Actions prior to arriving at the RSP:

- Open breakers 1Y201A02 and 1Y201B08 (both scrams the Reactor and causes MSIVs to close)
- Depress LOCAL TRIP pushbuttons for RFPTs 1A, 1B, and 1C (trips Feedwater pumps)

Note: The question meets SRO level guidelines because it requires assessment of plant conditions (Control Room evacuation with failure of scram from the Control Room) and selection/understanding of a section of a procedure to mitigate these conditions (alternate section of ON-CREVAC-101, including detailed knowledge of subsequent actions and effect on plant). The question cannot be answered solely using systems knowledge, knowing immediate operator actions, AOP/EOP entry conditions, or overall procedure purpose, sequence, or strategy.

A. Incorrect – The Reactor is scrammed. Plausible because the Reactor is initially at 100% when the Control Room is first evacuated. Also plausible if the candidate believes the action to scram the Reactor is taken at the RSP or after getting to the RSP.
B. Incorrect – MSIVs are closed. Plausible if the candidate does not understand that the action that scrams the Reactor also causes MSIVs to close. Feedwater pumps are tripped. Plausible because Feedwater pumps are still operating when the Control Room is first evacuated. Also plausible if the candidate believes the action to trip Feedwater pumps is taken at the RSP or after getting to the RSP.
C. Incorrect – MSIVs are closed. Plausible if the candidate does not understand that the action that scrams the Reactor also causes MSIVs to close.

Technical Reference(s): ON-CREVAC-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-EO102 Obj 2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:
TRH 2/7/18 – Revised question based on NRC comments.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295008 AA2.01
	Importance Rating	3.9

High Reactor Water Level

Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL: Reactor water level

Proposed Question: #83

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

- Annunciator AR-101-A17, RX WATER HI LEVEL, alarms.
- Investigation reveals that the Reactor Narrow Range level instrument PDT-C32-1N004B (Rosemount) has failed upscale.
- All other level instruments are functioning normally.
- The plant remains stable at 100% power.
- 2 minutes have elapsed since the alarm (AR-101-A17) was received.

Given the following Technical Specifications (TS):

(1) TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation
(2) TS 3.3.2.2, Feedwater – Main Turbine High Water Level Trip Instrumentation

Which one of the following describes the Condition entry requirements for these Technical Specifications based on the given failure?

Condition entry is required for...

A. NEITHER (1) NOR (2).

B. (1), but NOT for (2).

C. (2), but NOT for (1).

D. both (1) and (2).

Proposed Answer: C

Explanation: Narrow Range level instrument PDT-C32-1N004B inputs to multiple logics, including Feedwater level control, Main Turbine and Feedwater pump high level trips, RRP runbacks, and Level Setpoint Setdown. With the instrument failed upscale, it is initially providing a trip signal for Main Turbine and Feedwater pump high level trips. The plant response is correct (remains stable at 100%) because this logic is 2-out-of-3. However, after 90 seconds, ICS automatically bypasses the failed instrument, which makes it unable to perform it's protective function. This requires Condition entry in TS 3.3.2.2. This instrument does not input to the RPS low level scram logic, therefore Condition entry in TS 3.3.1.1 is also not required.

Note: The provided reference must have the "required channels per trip system" column of Table 3.3.1.1-1 for APRMs redacted to prevent giving information required in Question 87.

A. Incorrect – TS 3.3.2.2 Condition entry is required. Plausible because this would be correct if the instrument was not automatically bypassed by ICS after 90 seconds.
B. Incorrect – TS 3.3.1.1 Condition entry is not required. Plausible because this would be correct if Narrow Range level instrument 1N024B was failed upscale. TS 3.3.2.2 Condition entry is required. Plausible because this would be correct if the instrument was not automatically bypassed by ICS after 90 seconds.
D. Incorrect – TS 3.3.1.1 Condition entry is not required. Plausible because this would be correct if Narrow Range level instrument 1N024B was failed upscale.

Technical Reference(s): AR-101-A17, TM-OP-080, Technical Specifications 3.3.1.1 and 3.3.2.2

Proposed references to be provided to applicants during examination: Technical Specifications 3.3.1.1 and 3.3.2.2, with allowable values removed and APRM required channels removed; M-142 Sh1

Learning Objective: TM-OP-080 RBO-8

Question Source: Modified Bank – JAF 14-2 NRC #85

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295017 AA2.01
	Importance Rating	4.2

High Offsite Release Rate

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Off-site release rate: Plant-Specific

Proposed Question: #84

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

- A large spill of radioactive water has occurred in Radwaste.
- Turbine Building VERMS has indicated 2.1 E 7 μCi/min, steady, for the last 20 minutes.
- The running Radwaste Building Ventilation exhaust fan has tripped.

Which one of the following describes the need to enter EO-100-105, Radioactivity Release Control, and the required control of Radwaste Building Ventilation, in accordance with plant procedures?

	EO-100-105 Entry	Required Control of Radwaste Building Ventilation
A.	Required	Attempt to restart
B.	Required	Maintain shutdown
C.	NOT required	Attempt to restart
D.	NOT required	Maintain shutdown

Proposed Answer: A

Explanation: EO-100-105 entry is based on offsite radiation release rates relative to the Emergency Plan Alert levels. Turbine Building VERMs >1.9 E 7 μCi/min for >15 minutes exceeds the threshold for Alert EAL RA1.1, therefore EO-100-105 entry is required. EO-100-105 requires attempting to restart Radwaste ventilation.

Note: The EALs provided for Question 92 must have Table R-1 and EALS RG1.1, RS1.1, RA1.1, and RU1.1 redacted to avoid giving information related to this question.

B. Incorrect – Radwaste ventilation must be restarted if possible. Plausible that it would be maintained shutdown to limit the spread of contamination and to limit how much of the spilled water goes airborne.
C. Incorrect – EO-100-105 entry is required. Plausible because the given ventilation radiation level is just slightly above the threshold requiring entry and steady. Also plausible because this is not a primary system and is not in the main portion of the plant (TB/RB).
D. Incorrect – EO-100-105 entry is required. Plausible because the given ventilation radiation level is just slightly above the threshold requiring entry and steady. Also plausible because this is not a primary system and is not in the main portion of the plant (TB/RB). Radwaste ventilation must be restarted if possible. Plausible that it would be maintained shutdown to limit the spread of contamination and to limit how much of the spilled water goes airborne.

Technical Reference(s): EALs, EO-100-105, EP-RM-004

Proposed references to be provided to applicants during examination: None (see note above)

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:
TRH 2/17/18 – Resampled K/A and replaced question based on NRC comment.
TRH 2/22/18 – Added reference and edited provided reference, based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295020 2.1.20
	Importance Rating	4.6

Inadvertent Containment Isolation

Ability to interpret and execute procedure steps.

Proposed Question: #85

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

- The RWCU return flow instrument fails downscale.
- RWCU automatically isolates.
- RWCU flow on PPC OD3 display turns WHITE.
- RWCU is NOT restored within 15 minutes.

Which one of the following identifies the procedure that is required to be entered and the required action to be taken, in accordance with the Off-Normal Procedures?

Enter...

A. ON-PWR-101, Reactor Power. Suspend all reactivity changes.

B. ON-PWR-101, Reactor Power. Reduce core flow.

C. ON-CHEM-101, Chemistry Anomaly. Suspend all reactivity changes.

D. ON-CHEM-101, Chemistry Anomaly. Reduce core flow.

Proposed Answer: B

Explanation: The loss of RWCU flow results in an inaccurate heat balance calculation. ON-PWR-101 section B contains the appropriate guidance for unplanned loss of Reactor heat balance calculation. Since heat balance is not available for >15 minutes and Reactor power is >99.5%, the subsequent operator actions require lowering core flow by 0.5 Mlbm/hr.

A. Incorrect – It is required to perform a reactivity change by lowering core flow approximately 0.5 Mlbm/hr. Plausible because another step requires suspending all activities related to reactivity rise in the core. Also plausible because the heat balance is inaccurate, which makes monitoring reactivity changes more difficult.
C. Incorrect – ON-CHEM-101 entry is not required. Plausible because this procedure would be required to be entered for other RWCU abnormal indications and loss of RWCU does have an effect on coolant chemistry. It is required to perform a reactivity change by lowering core flow approximately 0.5 Mlbm/hr. Plausible because another step requires suspending all activities related to reactivity rise in the core. Also plausible because the heat balance is inaccurate, which makes monitoring reactivity changes more difficult.
D. Incorrect – ON-CHEM-101 entry is not required. Plausible because this procedure would be required to be entered for other RWCU abnormal indications and loss of RWCU does have an effect on coolant chemistry.

Technical Reference(s): ON-PWR-101, ON-CHEM-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-061 RBO-7

Question Source: Modified Bank - LOC26 (2014) NRC #83

Question History: LOC26 (2014) NRC #83

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:
TRH 2/9/18 – Revised 2nd half of B and D based on NRC comment.

Examination Outline Cross-Reference:

Level	SRO
Tier #	2
Group #	1
K/A #	209001 2.4.50
Importance Rating	4.0

LPCS

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

Proposed Question: #86

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

- Annunciator AR-109-E02, CORE SPRAY LOOP A HDR BREAK DETECT HI DIFF PRESS, alarms.
- The associated differential pressure detector (PDIS-E21-1N004A) indicates +4.0 psid.

Which one of the following identifies the validity of the alarm and the impact on compliance with the Technical Requirements Manual (TRM) or Technical Specifications (TS)?

	The alarm is...	Impact on Compliance with TRM / TS
A.	NOT valid.	NO TRM or TS Condition entry is required.
B.	NOT valid.	Condition entry is required in TRM 3.5.2, ECCS and RCIC System Monitoring Instrumentation.
C.	valid	The most limiting TS Condition entry requires restoration to operable status within a maximum of 7 days.
D.	valid	The most limiting TS Condition entry requires restoration to operable status within a maximum of 12 hours.

Proposed Answer: C

Explanation: The given indications show a valid Core Spray line break between the Reactor vessel and core shroud for Core Spray system A (pressure indication is above the 0.5 psig setpoint listed in the alarm response). This means that the design spray flow from Core Spray system A cannot be assured. This makes Core Spray system A inoperable. This requires entry into Technical Specification 3.5.1 Condition A, which requires restoring to operable status within 7 days.

A. Incorrect – The alarm is valid. Plausible because the indication is well within the instrument scale of -10 to +15 psid. Also plausible that no TS or TRM entry would be required because this is indication only.
B. Incorrect – The alarm is valid. Plausible because the indication is well within the instrument scale of -10 to +15 psid. Also plausible because if this instrument was inoperable, TRM 3.5.2 Condition A would apply.
D. Incorrect – The most restrictive Technical Specification Condition entry requires restoring to operable status within 7 days. Plausible because other required actions in TS 3.5.1 have a 12 hour completion time.

Technical Reference(s): AR-109-E02, Technical Specification 3.5.1

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-051 RBO-7

Question Source: Modified Bank – LOC26R (1/2015) NRC #86

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:
TRH 2/9/18 – Revised 2nd half of D based on NRC comment.
TRH 2/22/18 – Revised 12 days to 12 hours in choice D.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	215005 A2.03
	Importance Rating	3.8

APRM/LPRM

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

Proposed Question: #87

Unit 1 is operating at 100% power when the following occurs:

- All input power to APRM 3 is lost.
- NO other APRMs lose power.

Which one of the following describes:

(1) the plant response to this failure, and

(2) the need for a condition entry in Technical Specification (TS) 3.3.1.1, Reactor Protection System (RPS) Instrumentation?

	Plant Response	TS 3.3.1.1 Condition Entry
A.	Rod block, but NO APRM vote	Required
B.	Rod block, but NO APRM vote	NOT required
C.	Rod block and single APRM vote	Required
D.	Rod block and single APRM vote	NOT required

Proposed Answer: D

Explanation: Loss of all input power to the APRM causes an INOP trip. This causes a rod block and a single APRM scram vote. TS Table 3.3.1.1-1 only requires 3 of the 4 APRMs to be operable, so no Condition entry is required for any Function since the other 3 APRMs are unaffected.

Note: The question meets both parts of the K/A because it requires knowledge of how APRMs are affected by different conditions (bypass, loss of input power) and using this knowledge to determine how to use procedures to correct / control / mitigate based on the effects (application of TS).

Note: The reference for Questions 76 and 83 must have the "required channels per trip system" column of Table 3.3.1.1-1 for APRMs redacted based on this question.

A. Incorrect – A scram vote is received. Plausible because other APRM failures, such as downscale, cause a rod block but no scram vote. TS Table 3.3.1.1-1 only requires 3 of the 4 APRMs to be operable, so no Condition entry is required for any Function since the other 3 APRMs are unaffected. Plausible because some instrumentation TSs require Condition entry on a single failed instrument.
B. Incorrect – A scram vote is received. Plausible because other APRM failures, such as downscale, cause a rod block but no scram vote.
C. Incorrect – TS Table 3.3.1.1-1 only requires 3 of the 4 APRMs to be operable, so no Condition entry is required for any Function since the other 3 APRMs are unaffected. Plausible because some instrumentation TSs require Condition entry on a single failed instrument.

Technical Reference(s): Technical Specification 3.3.1.1, AR-103-A06

Proposed references to be provided to applicants during examination: None (see note above)

Learning Objective: TM-OP-078D RBO-8

Question Source: Modified Bank – LOC27 (2015) NRC #90

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:
TRH 2/9/18 – Agreed to redact the "required channels per trip system" column of Table 3.3.1.1-1 that is provided for Questions 76 and 83.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	223002 A2.06
	Importance Rating	3.2

Primary Containment Isolation/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: Containment instrumentation failures

Proposed Question: #88

Unit 1 is operating at 100% power when the following occurs:

- I&C reports that a contact failure has resulted in one Drywell pressure signal to the PCIS logic failing low.
- I&C reports that all Drywell pressure signals to RPS are still operable.

If NO action is taken to fix the transmitter or insert the associated trip, which one of the following identifies the ***maximum*** amount of time allowed to place the plant in Mode 3 (from when the determination of logic failure was made), in accordance with Technical Specifications?

A. 12 hours

B. 24 hours

C. 36 hours

D. 48 hours

Proposed Answer: B

Explanation: Technical Specification (TS) Table 3.3.6.1-1 requires all four Drywell pressure inputs to PCIS to be operable (Function 2.d). With one input failed low, TS 3.3.6.1 Condition A must be entered. This Condition allows 12 hours for to place the channel in trip for Function 2.d. Once this 12 hours elapses, Condition C must be entered, which requires immediately entering Condition H. Condition H requires being in Mode 3 in 12 hours. Therefore, a maximum of 24 hours (12+12) is allowed for entering Mode 3.

A. Incorrect – A maximum of 24 hours (12+12) is allowed for entering Mode 3. Plausible because this would be the answer if the candidate jumped from Table 3.3.6.1-1 directly to Condition H.
C. Incorrect – A maximum of 24 hours (12+12) is allowed for entering Mode 3. Plausible because this would be the answer if the wrong time from Condition A were used (24+12).
D. Incorrect – A maximum of 24 hours (12+12) is allowed for entering Mode 3. Plausible because this would be the answer if the wrong times from Condition A and H were used (12+36).

Technical Reference(s): TS 3.3.6.1

Proposed references to be provided to applicants during examination: TS 3.3.6.1 (with allowable values removed)

Learning Objective: TM-OP-059X RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(x)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	259002 A2.02
	Importance Rating	3.4

Reactor Water Level Control

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 reactor feedwater flow inputs

Proposed Question: #89

Unit 1 is operating at 100% power when the following occurs:

- One RFP 1A discharge flow transmitter is automatically removed from the ICS/DCS total feed flow calculation due to deviation >0.50 Mlbm/hr.
- LIC-C32-1R600, FW Level Ctl/Demand Signal Controller, is in AUTOMATIC.
- Feedwater Level Control (FWLC) is selected to FWLC-3E.
- Reactor water level is +35", steady.
- NO operator actions have been taken.

Then, a second RFP 1A discharge flow transmitter fails upscale.

Which one of the following describes the response of Reactor water level and the required control of FWLC, in accordance with ON-LVL-101, RPV Level Control System Malfunction?

	Reactor Water Level Response	Required Control of FWLC
A.	Lowers to approximately +32" and then returns to +35"	Place LIC-C32-1R600 in MANUAL and maintain it in this mode until at least one of the failed transmitters is fixed
B.	Lowers to approximately +32" and then returns to +35"	Place FWLC in FWLC-1E, place both failed transmitters in MAINTENANCE BYPASS, and then restore FWLC to FWLC-3E
C.	Remains stable at +35"	Place LIC-C32-1R600 in MANUAL and maintain it in this mode until at least one of the failed transmitters is fixed
D.	Remains stable at +35"	Place FWLC in FWLC-1E, place both failed transmitters in MAINTENANCE BYPASS, and then restore FWLC to FWLC-3E

Proposed Answer: B

Explanation: Reactor water level lowers to approximately +32" and then returns to +35" due to the steam/feed flow mismatch signal because the 2nd failed transmitter does not get automatically bypassed by FWLC. With two failed transmitters, ON-LVL-101 requires taking FWLC to FWLC-1E, placing both failed transmitters in MAINTENANCE BYPASS, and then restoring FWLC to FWLC-3E.

A. Incorrect – With two failed transmitters, ON-LVL-101 requires taking FWLC to FWLC-1E, placing both failed transmitters in MAINTENANCE BYPASS, and then restoring FWLC to FWLC-3E. Plausible because the procedure does require LIC-C32-1R600 in MANUAL as part of this operation, but then allows it to go back to AUTO after transmitters are bypassed.
C. Incorrect – Reactor water level lowers to approximately +32". Plausible because this is the response on the first failed transmitter. Also plausible that the second failed transmitter would also be automatically bypassed since a third transmitter exists. With two failed transmitters, ON-LVL-101 requires taking FWLC to FWLC-1E, placing both failed transmitters in MAINTENANCE BYPASS, and then restoring FWLC to FWLC-3E. Plausible because the procedure does require LIC-C32-1R600 in MANUAL as part of this operation, but then allows it to go back to AUTO after transmitters are bypassed.
D. Incorrect – Reactor water level lowers to approximately +32". Plausible because this is the response on the first failed transmitter. Also plausible that the second failed transmitter would also be automatically bypassed since a third transmitter exists.

Technical Reference(s): ON-LVL-101

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-045 RBO-7

Question Source: Bank - LOC24 (12/2011) Audit #86

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	262001 2.2.36
	Importance Rating	4.2

A.C. Electrical Distribution

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

Proposed Question: #90

Unit 1 is operating at 40% power with the following:

Date and Time (hhmm)	Condition(s)
3/1 at 0600	• A clearance is hung to allow inspection of Diesel Generator E jacket water heat exchanger tubing.
3/1 at 0800	• Startup Transformer T10 tap changer is placed in Manual.
3/1 at 1200	• Engineering reports that the Diesel Generator B air start system CANNOT perform its design function.

Which one of the following identifies the ***latest time*** by which the plant is required to enter Mode 3 if these conditions remain unchanged, in accordance with Technical Specifications?

A. 3/2 at 0100

B. 3/2 at 0200

C. 3/2 at 1200

D. 3/5 at 0000

Proposed Answer: C

Explanation: When Startup Transformer T10 tap changer is placed in Manual at time 0800 on 3/1, Technical Specification 3.8.1 Condition A must be entered for one offsite circuit inoperable. When Diesel Generator B is declared inoperable at time 1200 on 3/1, both Technical Specification 3.8.1 Conditions B and D must be entered. The most restrictive time requirement to be in Mode 3 is based on Condition D, which allows 12 hours to restore operability, followed by entry into Condition F, which allows 12 hour to be in Mode 3. This results in the latest time to enter Mode 3 being 3/2 at 1200 (3/1 at 1200 + 12 hours + 12 hours).

A. Incorrect – The latest time to enter Mode 3 is 3/2 at 1200. Plausible because this is the time requirement if Condition G is entered at time 3/1 at 1200.
B. Incorrect – The latest time to enter Mode 3 is 3/2 at 1200. Plausible because this is the time requirement based on two required DGs inoperable (3/1 at 1200 + 2 hours +12 hours).
D. Incorrect – The latest time to enter Mode 3 is 3/2 at 1200. Plausible because this is the time requirement based on only one Diesel Generator being inoperable. (3/1 at 1200 + 72 hours + 12 hours).

Technical Reference(s): Technical Specification 3.8.1

Proposed references to be provided to applicants during examination: Technical Specification 3.8.1 (without values, without above the line information, without SRs, with number of required DGs redacted in Required Action E.1)

Learning Objective: TM-OP-003 RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:
TRH 2/11/18 – Made multiple changes based on NRC comments: stated that T10 tap changer is taken to Manual, revised question wording, revised original choice C, and moved DG E info to new time block.
TRH 2/22/18 – Revised provided reference and 1st and 3rd conditions, based on NRC comment.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	202001 2.1.7
	Importance Rating	4.7

Recirculation System

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

Proposed Question: #91

Unit 1 is operating at 100% power when the following occurs:

- Reactor Recirc Pump (RRP) 1A trips.
- OPRMs are inoperable.
- APRM power is 62%, steady
- Total core flow is 50 Mlbm/hr, steady.
- The cause of the RRP 1A trip has been determined and corrected.

Note: The Power / Flow Map is provided on the following page.

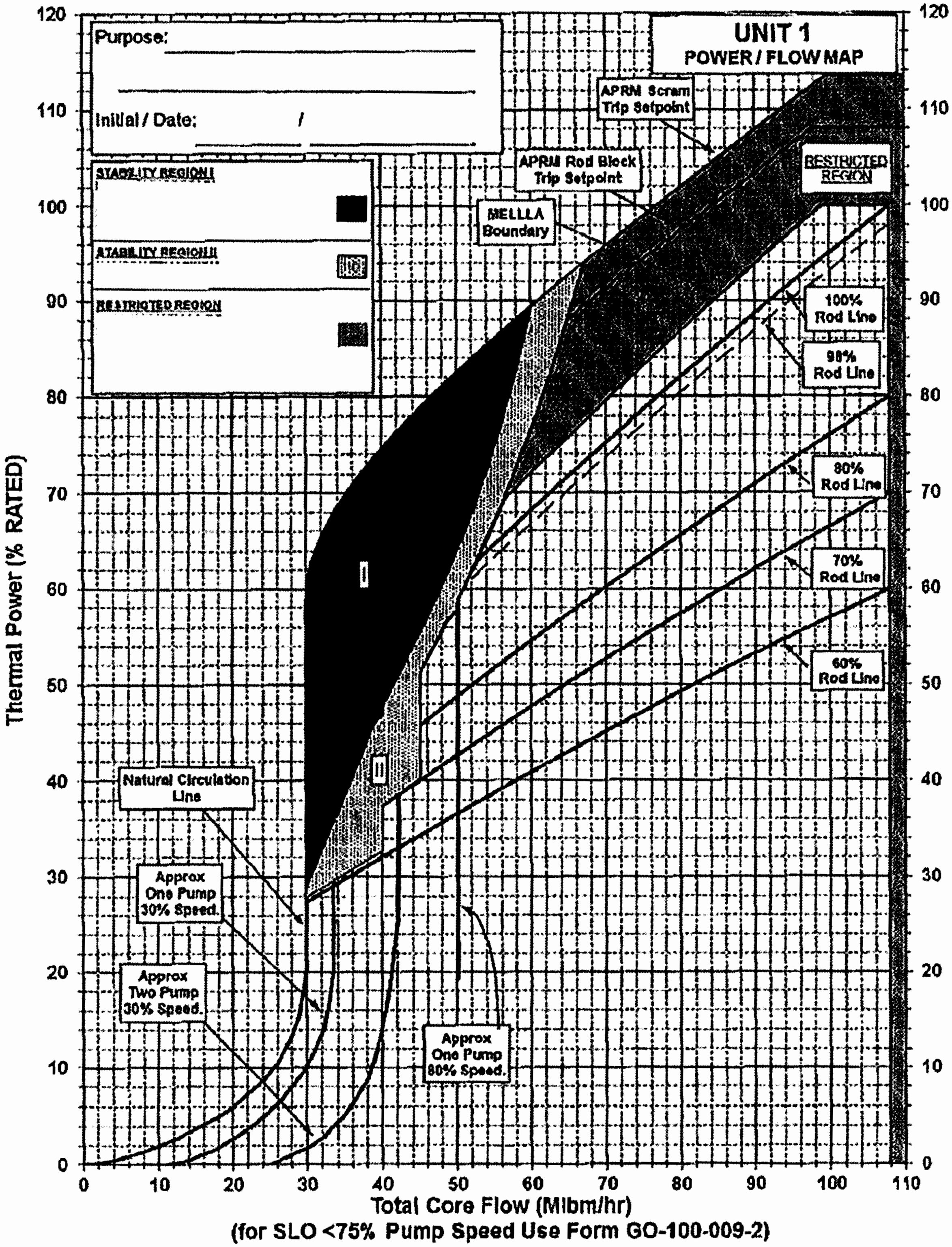
Which one of the following actions is required, in accordance with ON-RECIRC-101, Reactor Recirculation Malfunction?

A. Insert control rods in accordance with the CRC Book.

B. Place the Reactor Mode Switch in SHUTDOWN.

C. Raise total core flow by re-starting RRP 1A.

D. Raise total core flow using RRP 1B.



UNIT 1
POWER / FLOW MAP
Purpose:
Initial / Date: /
STABILITY REGION I
STABILITY REGION II
RESTRICTED REGION
APRM Scram Trip Setpoint
APRM Rod Block Trip Setpoint
MELLLA Boundary
RESTRICTED REGION
100% Rod Line
98% Rod Line
80% Rod Line
70% Rod Line
60% Rod Line
I
II
Natural Circulation Line
Approx One Pump 30% Speed.
Approx Two Pump 30% Speed.
Approx One Pump 80% Speed.
Thermal Power (% RATED)
Total Core Flow (Mlbm/hr)
(for SLO <75% Pump Speed Use Form GO-100-009-2)

Proposed Answer: A

Explanation: Plotting the given data on the power-flow map results in operation in Region 2. RRP 1B flow is already at the maximum allowed of 80%, therefore its flow cannot be raised to exit Region 2. RRP 1A re-start is not allowed to exit Region 2 because it is not rapid and because operation is above the 60% rod line. Therefore, the required action to exit Region 2 is to insert control rods using RMCS.

B. Incorrect – A Reactor scram is not required. Plausible because this would be correct if operation were in Region 1 of the power-flow map since OPRMs are inoperable.
C. Incorrect – Per OP-164-001, restart of the A pump is not allowed when above the 60% rod line, and this action will not permit RAPID exit of Region 2. Plausible because this would raise flow, which would exit Region 2 of the power-flow map.
D. Incorrect – RRP 1B flow cannot be raised because it is already at the maximum allowed speed of 80%. Plausible because this would exit Region 2 of the power-flow map if procedurally allowed.

Technical Reference(s): ON-RECIRC-101, TRM 9.0, OP-164-001

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-064 RBO-7

Question Source: Bank – LOC23 (2/2011) NRC #76

Question History: LOC23 (2/2011) NRC #76

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	271000 A2.04
	Importance Rating	4.1

Offgas

Ability to (a) predict the impacts of the following on the OFFGAS SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Offgas system high radiation

Proposed Question: #92

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

Time (hhmm)	Condition(s)
0900	• Annunciator AR-106-G03, OFF GAS HI RADIATION, alarms. • Chemistry reports that the last Reactor coolant activity sample was taken at 2200 the previous day and indicated 0.05 μCi/gm dose equivalent I-131. • Chemistry is directed to take another Reactor coolant activity sample.
1015	• Chemistry reports that Reactor coolant activity is 2.1 μCi/gm dose equivalent I-131.
1130	• Annunciator AR-106-F03, OFF GAS HI HI RADIATION, alarms.

Which one of the following describes the ***latest*** time by which notification to the NRC is required, in accordance with NDAP-QA-0720, Station Report Matrix and Reportabillity Evaluation Guidance?

A. 1130

B. 1245

C. 1415

D. 1530

Proposed Answer: B

Explanation: The EALs are affected by both Offgas pretreatment radiation readings and Reactor coolant activity levels. With the given conditions, an EAL is first met or exceeded at time 1130 when indications are received in the Control Room that Offgas pretreatment radiation readings are above the setpoint for Annunciator AR-106-F03, OFF GAS HI HI RADIATION (SU4.1). The EAL must be declared by 1145 at the latest (15 minutes from time indications are available in the Control Room). The NRC must be notified of this declaration by 1245 at the latest (1 hour from time of latest possible declaration).

Note: The question meets the K/A by presenting elevated Offgas radiation conditions and requiring the applicant to use these conditions to determine proper execution of the emergency plan and associated reportability requirements. The first half of the K/A could not be directly tested while maintaining the question level of difficulty high enough for an SRO, so the question scope is limited to the second half of the K/A, per NUREG 1021 ES-401 D.2.a.

Note: Table F-1 Row D in the provided reference to the applicants will be blocked out to prevent Question #98 from being a direct lookup. Additionally, Table R-1 and EALS RG1.1, RS1.1, RA1.1, and RU1.1 must be redacted to avoid giving information related to Question 84.

A. Incorrect – 1245 is the latest time notification to the NRC is required. Plausible because 1130 would be correct if an EAL were met or exceeded at time 1015 and this is when coolant activity is none to be elevated (above TS limit but below EAL threshold).
C. Incorrect – 1245 is the latest time notification to the NRC is required. Plausible because 1415 would be correct if no EAL were exceeded, but a scram were required at time 1015 based on elevated coolant activity (by TS or ON-MSLRAD-101).
D. Incorrect – 1245 is the latest time notification to the NRC is required. Plausible because 1530 would be correct if no EAL were exceeded, but a scram were required at time 1130 based on Offgas hi-hi radiation.

Technical Reference(s): Hot EAL Matrix, NDAP-QA-0720

Proposed references to be provided to applicants during examination: Hot and Cold EAL Matrices (Table F-1 Row D blocked out)

Learning Objective: TM-OP-072 RBO-07

Question Source: Modified Bank – NMP1 2009 NRC #91

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:
TRH 2/11/18 – Revised to add extra layer of required knowledge related to declaration/notification time requirements, based on NRC comment.

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	290001 A2.03
	Importance Rating	3.6

Secondary Containment

Ability to (a) predict the impacts of the following on the SECONDARY CONTAINMENT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High area radiation

Proposed Question: #93

Unit 1 is operating at 100% power when the following occurs:

- The watertight door between HPCI and RCIC was inadvertently left open.
- An un-isolable steam leak occurs from HPCI into the HPCI room.
- HPCI Equipment Area temperature is 160°F, steady.
- RCIC Equipment Area temperature is 125°F, steady.
- HPCI Room ARM indicates 1.2E+01 R/hr, steady.
- RCIC Room ARM indicates 2.0E+00 R/hr, steady.

__Note:__ EO-100-104, Secondary Containment Control, Tables 8 and 9 are provided on the following pages.

Which one of the following describes the required control of the Reactor, in accordance with the Emergency Operating Procedures?

A. Commence a controlled Reactor shutdown. A Reactor scram is NOT required.

B. Scram the Reactor and commence a cooldown at less than 100°F/hr.

C. Scram the Reactor and commence a cooldown at greater than 100°F/hr. Entry into EO-100-112, Emergency RPV Depressurization is NOT required.

D. Scram the Reactor and perform an Emergency RPV Depressurization per EO-100-112.

TABLE 8

TABLE 8 REACTOR BUILDING TEMPERATURE

RB AREA EL (FT)		MAX NORMAL ΔTEMP (°F)	MAX NORMAL TEMP (°F)	MAX SAFE TEMP (°F)	RECORDER POINTS ΔT, T	RB TEMP (°F)	
818	GENERAL AREA	N/A	110	120		____	____
779	GENERAL AREA	N/A	110	120		____	____
749	GENERAL AREA	N/A	110	120		____	____
	RWCU-PUMP ROOM	45	120	147	18,10	____	____
	RWCU-HEAT EXCH ROOM	45	120	147	19,11	____	____
	RWCU-PENETRATION ROOM	45	120	131	20,12	____	____
719	GENERAL AREA	N/A	110	120		____	____
	MAIN STEAM LINE TUN	60	157	177	7, 1	____	____
683	GENERAL AREA	N/A	110	120		____	____
	HPCI PIPE ROUTING AREA	60	120	167	17, 6	____	____
	RCIC PIPE ROUTING AREA	60	120	167	9*, 3*	____	____
670	GENERAL AREA	N/A	110	120		____	____
645	HPCI-EQUIP AREA	45	120	167	15,4	____	____
	HPCI-EMERG AREA COOLER	N/A	120	167	5	____	____
645	RCIC-EMERG AREA COOLER	N/A	120	167	2*	____	____
	RCIC-EQUIP AREA	45	120	167	7*,1*	____	____
645	RHR EQUIP AREA 1	45	110	142	8,2	____	____
645	RHR EQUIP AREA 2	45	110	142	9,3	____	____
645	CS PUMP ROOM A	N/A	110	142		____	____
645	CS PUMP ROOM B	N/A	110	142		____	____
645	RB SUMP ROOM	N/A	110	125		____	____

TABLE 9

TABLE 9 REACTOR BUILDING RADIATION

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

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

Proposed Answer: B

Explanation: Both of the given area temperatures are above Max Normal but below Max Safe. The HPCI Room ARM indication is above Max Safe, while the RCIC Room ARM indication is above Max Normal but below Max Safe. All indications are steady, so exceeding additional Max Safe values cannot be anticipated. EO-100-104 entry is required based on RB area temperatures and radiation levels above Max Normal values. Since the discharge is from a primary system, it cannot be isolated, and one RB area radiation level is above Max Safe, a Reactor scram is required and EO-100-102, RPV Control, must be entered. EO-100-112, Emergency RPV Depressurization, entry is not required because there is not a second Max Safe radiation level exceeded. EO-100-102 requires cooldown maintained less than 100°F/hr. Emergency RPV Depressurization cannot be anticipated because no additional parameter is approaching the Max Safe value.

Note: The first half of the K/A could not be directly tested while maintaining the question level of difficulty high enough for an SRO, so the question scope is limited mostly to the second half of the K/A, per NUREG 1021 ES-401 D.2.a.

A. Incorrect – A scram is required because the HPCI Room ARM indication is above Max Safe due to a primary system discharge that can't be isolated. Plausible because if the discharge were isolated or not from a primary system and another Max Safe rad level was exceeded, then this would be the correct answer.
C. Incorrect – Emergency RPV Depressurization cannot be anticipated because no additional parameter is approaching the Max Safe value. Plausible because if the RCIC ARM indication were approaching the Max Safe value, then this would be the correct answer.
D. Incorrect – EO-100-112, Emergency RPV Depressurization, entry is not required because there is not a second Max Safe radiation level exceeded. Plausible because this would be the correct answer if either two Max Safe temperatures were exceeded or a second Max Safe rad level was exceeded.

Technical Reference(s): EO-100-104, EO-100-102

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-EO102 Obj 2

Question Source: Modified Bank – LOC25 (2013) NRC #85

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:
TRH 2/7/18 – Changed wording/notation in last 2 bullets, based on NRC comments.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.39
	Importance Rating	4.3

Knowledge of conservative decision making practices.

Proposed Question: #94

Unit 1 is operating at 100% power when the following occurs:

- AR-102-G01, RECIRC PUMP A SEAL LEAKAGE HI FLOW, alarms.
- Recirc pump A seal leakage flow is 0.2 gpm, up slow.
- Recirc pump A second stage seal cavity pressure is 500 psig, steady.

Given the following procedures:

- ON-RECIRC-101, Reactor Recirculation Malfunction
- NDAP-00-0333, Operational Decision-Making Process

Which one of the following identifies the need for entry and/or applicability of these procedures for the given conditions?

	ON-RECIRC-101	NDAP-QA-0333
A.	Entry required	Applicable
B.	Entry required	NOT applicable
C.	Entry NOT required	Applicable
D.	Entry NOT required	NOT applicable

Proposed Answer: C

Explanation: ON-RECIRC-101 entry is not required because the given indications show degradation of only one Recirc pump seal (flow <1 gpm and second stage cavity pressure maintained). NDAP-00-0333 is applicable. Attachment A, SSES Emergent Issue Response Checklist, is used by the Shift Manager to determine the appropriate course of action. The Qualitative Risk Significance Table screens this issue as requiring being an Operation Decision Making Issue (ODMI) because the consequences are Catastrophic/Critical (significant loss of generating capacity if degradation continues, plant scram possible if DW leakage rises) and the probability is Probable (leak is already occurring and rate is rising).

A. Incorrect – ON-RECIRC-101 entry is not required because the given indications show degradation of only one Recirc pump seal (flow <1 gpm and second stage cavity pressure maintained). Plausible because ON-RECIRC-101 entry is required if evidence of dual seal failure is seen.
B. Incorrect – ON-RECIRC-101 entry is not required because the given indications show degradation of only one Recirc pump seal (flow <1 gpm and second stage cavity pressure maintained). Plausible because ON-RECIRC-101 entry is required if evidence of dual seal failure is seen. NDAP-QA-0333 is applicable. Plausible if candidate believed this process was only applicable if ON/EOP entry was warranted, or if candidate believed this process was only applicable where plant procedures did not address condition (AR-102-G01 provides guidance)
D. Incorrect – NDAP-QA-0333 is applicable. Plausible if candidate believed this process was only applicable if ON/EOP entry was warranted, or if candidate believed this process was only applicable where plant procedures did not address condition (AR-102-G01 provides guidance)

Technical Reference(s): AR-102-G01, ON-RECIRC-101, NDAP-00-0333

Proposed references to be provided to applicants during examination: None

Learning Objective: TM-OP-064 RBO-7

Question Source: Modified Bank - NMP1 2015 NRC #100

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.12
	Importance Rating	4.1

Knowledge of surveillance procedures.

Proposed Question: #95

Unit 1 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.
- A risk evaluation has been performed and the associated impact is being managed.

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. Delay greater than 24 hours is only allowed if a risk evaluation is performed and the impact is managed. Since the risk evaluation has been performed, 7 days are allowed in this case.

A. Incorrect – SR 3.0.3 allows a delay time to perform the missed surveillance before being required to declare the LCO not met. Plausible because the surveillance has been overdue for more than 25% of the normal time requirement, such as the grace period in SR 3.0.2.
B. Incorrect – SR 3.0.3 allows the longer of 24 hours or the specified frequency (7 days), but only if a risk evaluation is performed. Plausible because this would be the correct answer without the risk evaluation.
C. Incorrect – SR 3.0.3 allows the longer of 24 hours or the specified frequency (7 days). Plausible because 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:

Question Source: Bank – PB 2017 NRC SRO #20

Question History: PB 2017 NRC SRO #20

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.17
	Importance Rating	3.8

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

Proposed Question: #96

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

- Next week's work schedule has just been risk assessed.
- The risk assessment shows that performing the work will result in an Orange EOOS risk level.

Which one of the following describes the ability to perform this work, in accordance with NDAP-QA-1902, Integrated Risk Management?

This work...

A. CANNOT be performed until the risk level is reduced to at least Yellow or the plant is shutdown.

B. can be performed. A High Risk Activity Plan is required. A Risk Management Challenge Board is NOT required.

C. can be performed. A High Risk Activity Plan is NOT required. A Risk Management Challenge Board is required.

D. can be performed. Both a High Risk Activity Plan and a Risk Management Challenge Board is required.

Proposed Answer: D

Explanation: NDAP-QA-1902 allows work to be performed that results in an Orange EOOS risk level. This requires development of a High Risk Activity Plan and approval of that plan by the Risk Management Challenge Board.

A. Incorrect – NDAP-QA-1902 allows work to be performed that results in an Orange EOOS risk level. Plausible because this represents a higher than normal level of risk that should be avoided when possible and it does require additional approval.
B. Incorrect – A Risk Management Challenge Board is required. Plausible that this would only be required for the higher Red risk level. Also plausible because this is not required for Yellow risk.
C. Incorrect – A High Risk Activity Plan is required. Plausible that this would only be required for the higher Red risk level. Also plausible because this is not required for Yellow risk.

Technical Reference(s): NDAP-QA-1902

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.4
	Importance Rating	3.7

Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: #97

An emergency is in progress with the following:

- An Alert has been declared.
- One worker will be required to receive an emergency exposure to protect a valuable piece of plant equipment from an in-progress fire.
- Two workers are available.
- The workers have equal qualifications, lifetime occupational exposures, and current year occupational exposures.
- Worker A is a 55 year old male.
- Worker B is a 35 year old male.

Which one of the following is the maximum emergency dose that can be authorized for this task and the preferred worker to receive the exposure, in accordance with EP-PS-001 Attachment HH, Emergency Personnel Dose Assessment and Protective Action Guide?

A. 10 Rem, Worker A

B. 10 Rem, Worker B

C. 25 Rem, Worker A

D. 25 Rem, Worker B

Proposed Answer: A

Explanation: EP-PS-001 Attachment HH limits the maximum emergency dose that can be authorized for protecting equipment/facilities or to control a fire to 10 Rem. With other considerations equal, then a worker over 45 years old is the preferred individual to receive the dose (Worker A).

B. Incorrect – With other considerations equal, then a worker over 45 years old is the preferred individual to receive the dose (Worker A). Plausible that a younger worker would be desired due to better general physical health/ability.
C. Incorrect – EP-PS-001 Attachment HH limits the maximum emergency dose that can be authorized for protecting equipment/facilities or to control a fire to 10 Rem. Plausible because 25 Rem is allowed for life-saving or protection of large populations.
D. Incorrect – EP-PS-001 Attachment HH limits the maximum emergency dose that can be authorized for protecting equipment/facilities or to control a fire to 10 Rem. Plausible because 25 Rem is allowed for life-saving or protection of large populations. With other considerations equal, then a worker over 45 years old is the preferred individual to receive the dose (Worker A). Plausible that a younger worker would be desired due to better general physical health/ability.

Technical Reference(s): EP-PS-001 Attachment HH

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – NMP1 2015 Audit #97

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.14
	Importance Rating	3.8

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: #98

Which one of the following identifies a radiation monitor and associated threshold reading used to define Loss of the Fuel Clad Barrier, in accordance with the Emergency Plan?

A. Containment high range radiation; 7 R/hr

B. Containment high range radiation; 3,000 R/hr

C. Main Steam Line radiation; hi alarm setpoint

D. Main Steam Line radiation; hi-hi alarm setpoint

Proposed Answer: B

Explanation: Form EP-RM-004-F Table F-1, Fission Product Barrier Matrix, identifies Containment high range radiation >3000 R/hr as a Loss of the Fuel Clad Barrier.

Note: The provided reference for Question #92 is to be edited to avoid making this question a direct lookup.

A. Incorrect – 3000 R/hr is the threshold value, NOT 7 R/hr. 7 R/hr is the threshold value for Loss of the Reactor Coolant System Barrier.
C. Incorrect – MSL radiation monitoring is not used to define a Loss of the Fuel Clad Barrier. This indicates fuel clad degradation, but does not define loss of the barrier. Also plausible because MSL radiation monitoring is used to determine when to scram in ON-MSLRAD-101 given a fuel failure.
D. Incorrect – MSL radiation monitoring is not used to define a Loss of the Fuel Clad Barrier. This indicates fuel clad degradation, but does not define loss of the barrier. Also plausible because MSL radiation monitoring is used to determine when to scram in ON-MSLRAD-101 given a fuel failure.

Technical Reference(s): Form EP-RM-004-F

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – LOC28 (2016) NRC #93

Question History: LOC28 (2016) NRC #93

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

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: #99

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

- At time 0000 (hhmm), a hostile force attacks the Susquehanna Steam Electric Station Protected Area and Security immediately notifies the Control Room.

Which one of the following identifies the ***latest time*** by which the NRC must be notified of this event, in accordance with EP-PS-100, Emergency Director, Control Room - Emergency Plan Position Specific Instruction?

A. 0015

B. 0030

C. 0045

D. 0100

Proposed Answer: A

Explanation: This event requires emergency plan activation. Imminent (actual) security threats require an expedited notification to the NRC. The time requirement is within 15 minutes of the event.

B. Incorrect – The notification is required by 0015 at the latest. Plausible because 30 minutes is the latest time that notification would be required to the County and State.
C. Incorrect – The notification is required by 0015 at the latest. Plausible because 45 minutes would be the time requirement if the 15 minute clock for NRC notification was believed to start at the end of the 30 minute clock for County and State notification (and NRC notification normally occurs after County and State notification given an emergency event).
D. Incorrect – The notification is required by 0015 at the latest. Plausible because this is the correct answer for a non-security related emergency plan activation.

Technical Reference(s): EP-PS-100

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.40
	Importance Rating	4.5

Knowledge of SRO responsibilities in emergency plan implementation.

Proposed Question: #100

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

Time (hhmm)	Condition
0010	• Plant conditions justify the declaration of an Alert due to Drywell pressure. • Plant conditions also exceed the Unusual Event threshold due to coolant leakage rate into the Drywell.
0017	• Plant conditions change. • Plant conditions NO LONGER justify the declaration of an Alert due to Drywell pressure. • Plant conditions still exceed the Unusual Event threshold due to coolant leakage rate into the Drywell. • Emergency declaration has NOT yet been made.

Which one of the following describes the latest time at which the emergency declaration must be made and the level of emergency that must be declared, in accordance with EP-RM-004, EAL Classification Bases?

	Latest Time for Declaration	Level of Emergency Declaration
A.	0025	Unusual Event
B.	0032	Unusual Event
C.	0025	Alert
D.	0032	Alert

Proposed Answer: C

Explanation: When conditions exist for a declaration but then change prior to actual declaration and a different classification should be made based on current conditions, then the declaration is made at the higher classification level. The timeliness of the notification is based on the initial entry into an emergency condition. The change in conditions does not eliminate the requirement of the 15 minute declaration from the time indications were first available in the Control Room that an EAL was met or exceeded.

A. Incorrect – The declaration is made at the higher Alert level, even though the associated conditions are no longer present. Plausible because only the Unusual Event conditions are still present at the time of declaration.
B. Incorrect – The required time of declaration is based on when indications were first available in the Control Room that an EAL was met or exceeded, therefore 0025 is correct. Plausible because 0032 is 15 minutes after the Alert conditions cleared and only an Unusual Event applied. The declaration is made at the higher Alert level, even though the associated conditions are no longer present. Plausible because only the Unusual Event conditions are still present at the time of declaration.
D. Incorrect – The required time of declaration is based on when indications were first available in the Control Room that an EAL was met or exceeded, therefore 0025 is correct. Plausible because 0032 is 15 minutes after the Alert conditions cleared and only an Unusual Event applied.

Technical Reference(s): EP-RM-004

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - LOC24 (12/2011) NRC #100

Question History: LOC24 (12/2011) NRC #100

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)