Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AK3.06
	Importance Rating	2.9

Partial or Complete Loss of Forced Core Flow Circulation

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Core flow indication

Proposed Question: #1

The plant is in Single Loop operation after an unplanned Recirc Pump trip with the following:

- Indicated drive flow in the operating Recirculation loop is 21,000 gpm.
- Indicated total core flow on Control Room Panel P603 is 22 Mlbm/hr.

Which one of the following identifies the current relationship between indicated total core flow, as read on P603 flow chart recorder B22-R613, and actual total Core Flow and the reason for the difference?

Indicated total core flow is…

A. higher than actual due to idle loop flow being added to the total core flow summing network.

B. higher than actual due to idle jet pump flow being added to the total core flow summing network.

C. lower than actual due to idle loop flow being subtracted from the total core flow summing network.

D. lower than actual due to idle jet pump reverse flow being subtracted from the total core flow summing network.

Proposed Answer: C

Explanation: When one recirculation pump is not running, the forward/reverse flow logic network automatically subtracts the measured flow in the idle loop from the measured flow in the active loop and displays this core flow on the P603 chart recorder. However, with operating loop drive flow less than 22,000 gpm, jet pump flow in the shutdown loop will be positive instead of negative as assumed by the summing network. Therefore, the core flow on the P603 chart recorder is lower than actual.

A. Incorrect – The idle loop flow is subtracted, not added, by the summing network. Plausible because the idle loop flow should actually be added.
B. Incorrect – The idle loop flow is subtracted by the summing network. Plausible because jet pump flow is related to loop flow, and loop flow should actually be added.
D. Incorrect – The idle loop flow is subtracted, not jet pump reverse flow. Plausible because jet pump flow and loop flow are related.

Technical Reference(s): N2-OP-29 Section H.6.0

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-101001-RBO-11

Question Source: Bank – 2017 Cert #42

Question History: 2010 NRC #51

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(2)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 AA1.03
	Importance Rating	4.4

Partial or Complete Loss of AC Power

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Systems necessary to assure safe plant shutdown

Proposed Question: #2

A Station Blackout is in progress and RCIC is being utilized for Reactor injection.

Which one of the following describes:

(1) the Reactor water level indication to control by to avoid a RCIC high level trip

and

(2) action required to avoid a RCIC isolation on high room temperature

in accordance with N2-SOP-01, Station Blackout?

	(1) Reactor Water Level Indication to Control By	(2) Action to Avoid High Room Temperature Isolation
A.	Wide Range	Bypass isolation
B.	Wide Range	Cycle RCIC steam supply
C.	Narrow Range	Bypass isolation
D.	Narrow Range	Cycle RCIC steam supply

Proposed Answer: A

Explanation: N2-SOP-01 cautions to use wide range Reactor water level indication to avoid a RCIC trip on high level. N2-SOP-01 directs bypassing RCIC isolations to avoid an isolation on high room temperature.

B. Incorrect – N2-SOP-01 directs bypassing RCIC isolations to avoid an isolation on high room temperature. Plausible because cycling the steam supply closed when injection is not needed would limit temperature rise in the room.
C. Incorrect – N2-SOP-01 cautions to use wide range Reactor water level indication to avoid a RCIC trip on high level. Plausible because narrow range Reactor water level indication is commonly used for controlling Reactor water level.
D. Incorrect – N2-SOP-01 cautions to use wide range Reactor water level indication to avoid a RCIC trip on high level. Plausible because narrow range Reactor water level indication is commonly used for controlling Reactor water level. N2-SOP-01 directs bypassing RCIC isolations to avoid an isolation on high room temperature. Plausible because cycling the steam supply closed when injection is not needed would limit temperature rise in the room.

Technical Reference(s): N2-SOP-01

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP01C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AA2.01
	Importance Rating	3.2

Partial or Complete Loss of DC Power

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Cause of partial or complete loss of D.C. power

Proposed Question: #3

The plant is operating at 85% power when the over-voltage protection circuit actuates on the Division I safety related battery charger, 2BYS*CHGR2A1.

Which one of the following describes the effect on the battery charger and Division I safety related 125 VDC bus?

A. The DC output breaker from the charger trips and the bus deenergizes.

B. The AC input breaker to the charger trips and the battery will supply the bus.

C. Charger operates in voltage limiting mode and supplies the bus with the battery.

D. Charger DC output and AC input breakers trip and the standby charger automatically supplies the bus.

Proposed Answer: B

Explanation: Divisions I, II and III battery chargers and the normal 125 VDC battery chargers have an over voltage protection circuit which disconnects the AC input breaker to the chargers when the DC output voltage exceeds a manually preset value. Charger AC input breaker will open. The battery will supply the loads.

A. Incorrect – The AC input breaker to the chargers trips and the battery will supply the loads. Plausible because the trip is based on DC output voltage.
C. Incorrect – The AC input breaker to the chargers trips and the battery will supply the loads. Plausible because there is a current limiting mode based on low DC output voltage.
D. Incorrect – Only the AC input breaker to the chargers trips and the battery will supply the loads. Placing the standby charger in service requires manually closing the breakers. Plausible because the trip is based on DC output voltage.

Technical Reference(s): N2-OP-74A, ARP 852108

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-263000-RBO-05

Question Source: Bank - 2009 NRC #44

Question History: 2009 NRC #44

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 2.4.47
	Importance Rating	4.2

Main Turbine Generator Trip

Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Proposed Question: #4

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

- Annunciator 851140, TURBINE GENERATOR VIBRATION HIGH, alarms.
- The following vibration levels are observed on Turbine bearing #1:

Time (hhmm)	Vibration (mils)
0800	7.0
0801	7.5
0802	8.0

Note: Assume vibration levels continue to rise at the same rate.

Which one of the following identifies the earliest time at which a manual Turbine trip will become required, in accordance ARP 851140?

A. 0804

B. 0810

C. 0821

D. 0825

Proposed Answer: B

Explanation: ARP 851140 contains the following criteria for manually tripping the Turbine:

- Any vibration ≥10 mils for 15 minutes
- Any vibration ≥12 mils
- Any vibration >9 mils with >3 mils/minute rate of change

With the given information, a trip will first be required at Time 0810 when vibrations reach 12 mils.

A. Incorrect – A trip will first be required at Time 0810 when vibrations reach 12 mils. Plausible because this is the time when vibrations reach 9 mils (but rate of change is <3 mils/minute).
C. Incorrect – A trip will first be required at Time 0810 when vibrations reach 12 mils. Plausible because this is 15 minutes after vibrations will have reached 10 mils.
D. Incorrect – A trip will first be required at Time 0810 when vibrations reach 12 mils. Plausible because this is the time when vibrations reach 12 mils plus 15 minutes.

Technical Reference(s): ARP 851140

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-245000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295006 AK1.03
	Importance Rating	3.7

Scram

Knowledge of the operational implications of the following concepts as they apply to SCRAM: Reactivity control

Proposed Question: #5

The plant has just been manually scrammed from 100% power.

Which one of the following describes the time delay before the scram can be reset and the reason for this time delay?

	Time Delay	Reason
A.	10 seconds	Ensure all control rods fully insert
B.	10 seconds	Prevent rapid cycling of CRD and the RPV
C.	30 seconds	Ensure all control rods fully insert
D.	30 seconds	Prevent rapid cycling of CRD and the RPV

Proposed Answer: A

Explanation: RPS enforces a 10 second time delay for resetting a scram. The reason for this time delay is to ensure all control rods fully insert.

B. Incorrect – The reason for this time delay is to ensure all control rods fully insert. Plausible because resetting the scram does initiate a transient on both CRD and the RPV.
C. Incorrect – The time delay is 10 seconds. Plausible because 30 seconds is the time delay for resetting RRCS ARI.
D. Incorrect – The time delay is 10 seconds. Plausible because 30 seconds is the time delay for resetting RRCS ARI. The reason for this time delay is to ensure all control rods fully insert. Plausible because resetting the scram does initiate a transient on both CRD and the RPV.

Technical Reference(s): N2-OP-97, N2-212000-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-212000-RBO-5

Question Source: Modified Bank – 2016 NRC #41

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

SCRAM / 1

Knowledge of the operational implications of the following concepts as they apply to SCRAM : Reactivity control

Proposed Question: #41

The plant is operating at rated conditions with the following:

- A manual reactor scram is performed
- An RPS circuit failure has occurred which fails to energize time delay relays associated with Scram reset prevention

What are the operational implications of this failure if the SCRAM were to be manually reset before 10 seconds has elapsed?

A.	All control rods may fail to fully insert.
B.	The Scram Air header may not re-pressurize.
C.	Reactor water level may not recover above the scram setpoint.
D.	The Scram Discharge Volume vent and drain valves may not close.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AK2.02
	Importance Rating	4.0

Control Room Abandonment

Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Local control stations: Plant-Specific

Proposed Question: #6

The Control Room has been evacuated due to a fire with the following:

- N2-SOP-78, Control Room Evacuation, is being executed.
- The Unit Supervisor has directed the following:
  - Cycle SRVs to control Reactor pressure.
  - Start RHR pump RHS*P1A for Suppression Pool Cooling.

Which one of the following describes the location(s) where these actions are performed?

A. Cycling SRVs is performed at the Remote Shutdown Panels. Starting RHS*P1A is performed at Switchgear 2ENS*SWG101-3.

B. Cycling SRVs is performed at Reactor Building elevation 289'. Starting RHS*P1A is performed at the Remote Shutdown Panels.

C. Cycling SRVs is performed at Reactor Building elevation 289'. Starting RHS*P1A is performed at Switchgear 2ENS*SWG101-3.

D. Both actions are performed at the Remote Shutdown Panels.

Proposed Answer: D

Explanation: The Remote Shutdown Panels contain the controls to both cycle SRVs and to start RHS*P1A.

A. Incorrect – Both controls are at the Remote Shutdown Panels. Plausible because the breaker for RHS*P1A is located at Switchgear 2ENS*SWG101-3.
B. Incorrect – Both controls are at the Remote Shutdown Panels. Plausible because there are multiple SRV/ADS components on Reactor Building elevation 289'.
C. Incorrect – Both controls are at the Remote Shutdown Panels. Plausible because there are multiple SRV/ADS components on Reactor Building elevation 289'. Plausible because the breaker for RHS*P1A is located at Switchgear 2ENS*SWG101-3.

Technical Reference(s): N2-SOP-78, N2-OP-78

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP78C01 EO-2

Question Source: Modified Bank - JAF 14-1 NRC #42

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AK2.02
	Importance Rating	4.0

Control Room Abandonment

Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Local control stations: Plant-Specific

Proposed Question: #42

The plant was operating at 100% with the following:

- A fire developed in the Control Room and required evacuation.
- AOP-43, Plant Shutdown From Outside the Control Room, is being performed.
- The Shift Manager has directed starting RHRSW pump B and RHR pump D.

Which one of the following describes the panel location(s) where these actions are performed?

A. Both actions are performed at 25RSP (Reactor Building 300' North).

B. RHRSW pump B is started at 25RSP (Reactor Building 300' North) and RHR pump D is started at 25ASP-2 (East Crescent Stairway).

C. RHRSW pump B is started at 25ASP-2 (East Crescent Stairway) and RHR pump D is started at 25RSP (Reactor Building 300' North).

D. Both actions are performed at 25ASP-2 (East Crescent Stairway).

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AK3.02
	Importance Rating	3.3

Partial or Complete Loss of CCW

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Reactor power reduction

Proposed Question: #7

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

- Degraded performance has been observed in the Turbine Building Closed Loop Cooling system (CCS).
- N2-SOP-14, Loss or Degraded CCS System, has been entered.
- CCS heat exchanger discharge temperature is 102^{o}F and slowly rising.
- The Unit Supervisor has directed lowering Reactor power with Recirc flow based on guidance in N2-SOP-14.

Which one of the following identifies the reason for this action, in accordance with N2-SOP-14?

A. Lowers CCS heat load.

B. Prepares to scram the Reactor.

C. Prepares to trip one Recirc pump.

D. Allows securing one train of Feedwater pumps.

Proposed Answer: A

Explanation: N2-SOP-14 explains that the reason for lowering Reactor power is because it is "the only means to achieve a significant heat load reduction without securing major system loads".

B. Incorrect – The reason is to lower CCS heat load. Plausible because a Reactor scram could be required if conditions continue to degrade and lowering power is generally desired before scramming the Reactor when at 100% power.
C. Incorrect – The reason is to lower CCS heat load. Plausible because securing a Recirc pump may be desirable under related conditions, such as loss of CCP or SW.
D. Incorrect – The reason is to lower CCS heat load. Plausible because Feedwater pumps are CCS loads and could be secured after a power reduction.

Technical Reference(s): N2-SOP-14

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP14C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AA1.03
	Importance Rating	3.0

Partial or Complete Loss of Instrument Air

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air compressor power supplies

Proposed Question: #8

The plant is in an outage with the following:

- The Reactor is in Mode 4.
- 2NPS-SWG003 is de-energized due to a bus fault.
- 2NNS-SWG015 is being powered from Division II switchgear.
- All other electrical busses are in their normal lineup.
- Instrument Air Selector Switch is selected to CAB.

Then, a loss of Line 5 occurs.

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

A. C3C will continue to run in LEAD.
C3A will NOT be able to start in LAG.
C3B will NOT be able to start in BACKUP.

B. C3C will continue to run in LEAD.
C3A will NOT be able to start in LAG.
C3B remains available to start in BACKUP.

C. C3C will trip.
If IAS pressure lowers sufficiently, C3A will start in LAG.
C3B will NOT be able to start in BACKUP.

D. C3C will trip.
C3A will NOT be able to start in LAG.
C3B remains available to start in BACKUP.

Proposed Answer: D

Explanation: C3C is power from US10 Bus C which is powered from SWG001. C3A is powered from US5 which is powered from SWG014. C3B is powered from US6 which is powered from SWG015. With a loss of Line 5, SWG001, SWG014, and US10C all de-energize which causes C3A and C3C to trip. IAS pressure will lower until pressure reaches 85 psig in which time C3B will start in BACKUP.

A. Incorrect – C3C trips due to the loss of Line 5. It is C3A, not C3B, that also loses power. Plausible because the power supplies are similar.
B. Incorrect – C3C trips due to the loss of Line 5. C3A does lose power and C3B does remain available. Plausible because the power supplies are similar.
C. Incorrect – C3C will trip, however it is C3A, not C3B, that also loses power. Plausible because the power supplies are similar.

Technical Reference(s): N2-OP-19-Lineups

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-278001-RBO-11

Question Source: Bank – 2014 Cert #58

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AA2.05
	Importance Rating	3.4

Loss of Shutdown Cooling

Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor vessel metal temperature

Proposed Question: #9

A plant shutdown is in progress with the following:

- The Reactor Mode Switch is in SHUTDOWN.
- A loss of Shutdown Cooling occurs.
- The following Reactor coolant temperature data is observed:

Time (hhmm)	Reactor Coolant Temperature (^{o}F)
0830	176
0832	179
0834	182

Note: Assume Reactor coolant temperature continues to rise at the same rate.

Which one of the following describes whether the heatup rate is above or below 100^{o}F/hr and when a Reactor mode change will occur?

	Heatup Rate	Time of Mode Change
A.	Above 100^{o}F/hr	0846
B.	Above 100^{o}F/hr	0854
C.	Below 100^{o}F/hr	0846
D.	Below 100^{o}F/hr	0854

Proposed Answer: C

Explanation: The heatup rate is 3°F every 2 minutes. This equates to 90°F/hr, which is below 100°F/hr. The mode will change at 200°F. Given an initial temperature of 176°F at 0830, the mode change will occur at 0846 (176°F + (3°F/2min)*(16 min) = 200°F, 0830 + 16 min = 0846).

Note: In practice, Reactor vessel metal temperature is determined/interpreted indirectly through Reactor coolant temperature. Therefore, the question meets the K/A in an operationally oriented way by discussing Reactor coolant temperature as a proxy for Reactor vessel metal temperature.

A. Incorrect – The heatup rate is below 100°F/hr. Plausible because the heatup rate is close to 100°F/hr.
B. Incorrect – The heatup rate is below 100°F/hr. Plausible because the heatup rate is close to 100°F/hr. The mode change will occur at 0846. Plausible because 0854 is the time when the Reactor coolant will reach the boiling point (212°F).
D. Incorrect – The mode change will occur at 0846. Plausible because 0854 is the time when the Reactor coolant will reach the boiling point (212°F).

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-101001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 2.4.18
	Importance Rating	3.3

Refueling Accidents

Knowledge of the specific bases for EOPs.

Proposed Question: #10

Which one of the following describes the Minimum Safe Operating Spent Fuel Pool Level and the basis for this level, in accordance with NER-2M-039, NMP2 Emergency Operating Procedure and Severe Accident Procedures (EOP/SAP) Basis Document?

The Minimum Safe Operating Spent Fuel Pool Level is elevation…

A. 340’ and is based on maintaining the weirs covered.

B. 340’ and is based on providing adequate radiation shielding.

C. 352’ 8” and is based on maintaining the weirs covered.

D. 352’ 8” and is based on providing adequate radiation shielding.

Proposed Answer: B

Explanation: The Minimum Safe Operating Spent Fuel Pool Level is elevation 340'. This level is based on providing adequate radiation shielding for local operator actions and the Main Control Room.

A. Incorrect – This level is based on providing adequate radiation shielding for local operator actions and the Main Control Room. Plausible because keeping the weirs covered is part of the basis for the low level alarm of 352' 8", which is also used in N2-EOP-SC.
C. Incorrect – The level is 340'. Plausible because 352' 8" is the low level alarm and also used in N2-EOP-SC. This level is based on providing adequate radiation shielding for local operator actions and the Main Control Room. Plausible because keeping the weirs covered is part of the basis for the low level alarm of 352' 8", which is also used in N2-EOP-SC.
D. Incorrect – The level is 340'. Plausible because 352' 8" is the low level alarm and also used in N2-EOP-SC.

Technical Reference(s): NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(13)

Comments:

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

High Drywell Pressure

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

Proposed Question: #11

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

- Drywell spray has been initiated due to high Drywell pressure per N2-EOP-PC, Primary Containment Control.
- Drywell pressure is 6 psig and slowly lowering.
- Drywell temperature is 200^{o}F and lowering slowly.

Which one of the following identifies when Drywell spray is required to be secured, in accordance with N2-EOP-PC?

Secure Drywell spray before Drywell…

A. pressure drops to the threshold value of 1.68 psig.

B. pressure drops to the threshold value of 0 psig.

C. temperature drops to the threshold value of 150^{o}F.

D. temperature drops to the threshold value of 100^{o}F.

Proposed Answer: B

Explanation: N2-EOP-PC requires securing Drywell spray before Drywell pressure reaches the threshold value of 0 psig.

A. Incorrect – N2-EOP-PC requires securing Drywell spray before Drywell pressure reaches the threshold value of 0 psig. Plausible because Drywell pressure of 1.68 psig is the N2-EOP-PC entry condition.
C. Incorrect – N2-EOP-PC requires securing Drywell spray before Drywell pressure reaches the threshold value of 0 psig. Plausible because Drywell temperature is also elevated and 150^{o}F is the N2-EOP-PC entry condition.
D. Incorrect – N2-EOP-PC requires securing Drywell spray before Drywell pressure reaches the threshold value of 0 psig. Plausible because Drywell temperature is also elevated and 100^{o}F is a lower temperature constraint in N2-OP-60.

Technical Reference(s): N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Bank – PB 2017 NRC #63

Question History: PB 2017 NRC #63

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 EK2.01
	Importance Rating	4.1

High Reactor Pressure

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

Proposed Question: #12

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

- A Main Turbine pressure regulator malfunction causes Reactor pressure to rise.
- The following Reactor pressures are sensed by the transmitters inputting to RPS:
  - 2ISC*PIS1678A: 1038 psig
  - 2ISC*PIS1678B: 1075 psig
  - 2ISC*PIS1678C: 1046 psig
  - 2ISC*PIS1678D: 1083 psig

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

A. A half scram signal is received on RPS A, only.

B. A half scram signal is received on RPS B, only.

C. NO half scram signals are received.

D. A full scram signal is received.

Proposed Answer: B

Explanation: The high Reactor pressure scram setpoint is 1052 psig (must be ≤1072 per Technical Specifications). Both detectors 2ISC*PIS1678B and 2ISC*PIS1678D are above the setpoint. These detectors input to RPS B. Therefore, a half scram is received on RPS B. Neither of the other two detectors are above the setpoint, so a half scram is not received on RPS A.

A. Incorrect – A half scram is received on RPS B. Plausible because not all pressure transmitters have exceeded the scram setpoint and none have exceeded it by much. A half scram is not received on RPS A. Plausible if RPS detector assignments are confused or the setpoint is believed to be lower. Also plausible because the RPS A transmitters are above the high alarm setpoint (1035 psig).
C. Incorrect – A half scram is received on RPS B. Plausible because not all pressure transmitters have exceeded the scram setpoint and none have exceeded it by much.
D. Incorrect – A half scram is not received on RPS A. Plausible if RPS detector assignments are confused or the setpoint is believed to be lower. Also plausible because the RPS A transmitters are above the high alarm setpoint (1035 psig).

Technical Reference(s): ARPs 603103 and 603403, Technical Specification Table 3.3.1.1-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-212000-RBO-5

Question Source: Bank – 2019 NRC #12

Question History: 2019 NRC #12

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EK3.05
	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: Reactor SCRAM

Proposed Question: #13

Which one of the following describes:

(1) the limit used to determine when a Reactor scram is required based on high Suppression Pool water temperature

and

(2) the basis for this limit

in accordance with N2-EOP-PC, Primary Containment Control, and NER-2M-095, NMP2 Emergency Operating Procedure and Severe Accident Procedures (EOP/SAP) Basis Document?

A. (1) A constant limit of 110^{o}F
(2) Maintain adequate Net Positive Suction Head (NPSH) for ECCS pumps

B. (1) A constant limit of 110^{o}F
(2) Maintain the Primary Containment within design basis accident assumptions

C. (1) A variable limit based on the Heat Capacity Temperature Limit (HCTL)
(2) Maintain adequate Net Positive Suction Head (NPSH) for ECCS pumps

D. (1) A variable limit based on the Heat Capacity Temperature Limit (HCTL)
(2) Maintain the Primary Containment within design basis accident assumptions

Proposed Answer: B

Explanation: The decision to scram based on high Suppression Pool water temperature is based on a constant limit of 110°F. NER-2M-039 states that the basis for this is the Boron Injection Initiation Temperature. The basis for the Boron Injection Initiation Temperature at NMP2 is Technical Specification 3.6.2.1. The basis for this Technical Specification is related to ensuring the Containment is maintained within the assumptions of the design basis analysis.

A. Incorrect – The basis is related to ensuring the Containment is maintained within the assumptions of the design basis analysis. Plausible because high Suppression Pool water temperature also lowers available NPSH to the ECCS pumps.
C. Incorrect – The basis is related to ensuring the Containment is maintained within the assumptions of the design basis analysis. Plausible because high Suppression Pool water temperature also lowers available NPSH to the ECCS pumps. The decision to scram based on high Suppression Pool water temperature is based on a constant limit of 110°F. Plausible because the HCTL is another limit used in the SPT leg of N2-EOP-PC with respect of RPV Blowdown.
D. Incorrect – The decision to scram based on high Suppression Pool water temperature is based on a constant limit of 110°F. Plausible because the HCTL is another limit used in the SPT leg of N2-EOP-PC with respect of RPV Blowdown.

Technical Reference(s): N2-EOP-PC, NER-2M-039, Technical Specification 3.6.2.1 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295028 EA1.03
	Importance Rating	3.9

High Drywell Temperature

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

Proposed Question: #14

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

- Drywell Pressure is 3.0 psig
- Hottest Drywell Temperature is 275°F
- Drywell Cooling Fans are tripped

Which one of the following describes the effect on restoring Drywell Cooling System (DRS), in accordance with EOP support procedures?

A. Can be restored after defeating LOCA isolation interlocks

B. Can be restored without defeating LOCA isolation interlocks

C. CANNOT be restored, restoring CCP flow may result in water hammer

D. CANNOT be restored, restarting DRS fans may result in air duct damage

Proposed Answer: C

Explanation: Per N2-EOP-6.24, if DWT is above 250°F, the containment isolation MOVs cannot be reopened, because the water volume in the section of piping between the inboard and outboard isolation valves may have flashed to steam due to the elevated DWT.

A. Incorrect – Drywell cooling cannot be restored. Plausible because physically the system could be restored by defeating interlocks.
B. Incorrect – Drywell cooling cannot be restored. Plausible that partial restoration could be made without defeating interlocks by using the “fan only” mode.
D. Incorrect – The concern is water hammer from restoring CCP flow. Plausible because the other transient induced during restoration of Drywell Cooling is air flow/pressure change from starting fans.

Technical Reference(s): N2-EOP-6.24

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOP6HCC01, Objective # N2-EOP-6-CE3-A24

Question Source: Bank – 2016 NRC #60

Question History: 2016 NRC #60

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 EA2.02
	Importance Rating	3.9

Low Suppression Pool Water Level

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool temperature

Proposed Question: #15

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

- Suppression Pool water level is 196.5 feet and steady.
- "A" RHS is operating in LPCI mode.
- "B" RHS is operating in Suppression Pool Cooling mode.

Which one of the following describes the method for determining Suppression Pool temperature, in accordance with N2-EOP-PC, Primary Containment Control, and NER-2M-039, NMP2 Emergency Operating Procedure and Severe Accident Procedures (EOP/SAP) Basis Document?

Use…

A. RHS heat exchanger inlet temperature on "B" RHS.

B. RHS heat exchanger outlet temperature on "A" RHS.

C. Suppression Pool temperature indication on SPDS.

D. post-accident monitor Suppression Tool temperature indicators.

Proposed Answer: A

Explanation: With Suppression Pool water level <197', one of the given methods for determining Suppression Pool temperature is using RHS heat exchanger inlet temperature on a loop in Suppression Pool Cooling.

B. Incorrect – Heat exchanger outlet temperature on a loop of RHS in LPCI mode would not accurately reflect Suppression Pool temperature. Plausible because the correct answer is an RHS heat exchanger temperature.
C. Incorrect – This temperature indicator is uncovered due to low Suppression Pool water level. Plausible because normally this would be a valid indicator to use.
D. Incorrect – This temperature indicator is uncovered due to low Suppression Pool water level. Plausible because normally this would be a valid indicator to use.

Technical Reference(s): N2-EOP-PC, NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Bank - 2012 Cert #51

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 2.1.20
	Importance Rating	4.6

Reactor Low Water Level

Ability to interpret and execute procedure steps.

Proposed Question: #16

The plant has experienced a scram with the following:

- Both Recirculation pumps tripped due to low Reactor water level.
- All attempts to restart a Recirculation pump have failed.
- The scram has been reset.
- Control Rod Drive (CRD) flow is 63 gpm.
- Reactor Water Cleanup (WCS) is isolated.
- N2-SOP-29, Sudden Reduction in Core Flow, is being executed.

Which one of the following describes the required control of CRD flow and WCS to minimize thermal stratification, in accordance with N2-SOP-29?

A. Raise CRD flow and restore WCS to service.

B. Raise CRD flow and maintain WCS isolated.

C. Lower CRD flow and restore WCS to service.

D. Lower CRD flow and maintain WCS isolated.

Proposed Answer: C

Explanation: With both Recirculation pumps trips and unable to be restarted, N2-SOP-29 attachment 1 step 4.2 is applicable. WCS is restored to service as soon as possible to minimize thermal stratification effects. CRD flow is reduced to minimize injection of cold water to the lower portion of the Reactor vessel.

A. Incorrect – CRD flow is lowered. Plausible because raising core flow is desirable, but raising CRD flow only leads to greater thermal stratification, not higher core flow.
B. Incorrect – CRD flow is lowered. Plausible because raising core flow is desirable, but raising CRD flow only leads to greater thermal stratification, not higher core flow. WCS is restored to service. Plausible because WCS also isolated on low Reactor water level and WCS has a cooling effect, which could be thought to worsen thermal stratification in the Reactor vessel.
D. Incorrect – WCS is restored to service. Plausible because WCS also isolated on low Reactor water level and WCS has a cooling effect, which could be thought to worsen thermal stratification in the Reactor vessel.

Technical Reference(s): N2-SOP-29

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-SOP-29

Question Source: Bank – 2017 Cert #40

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK1.06
	Importance Rating	4.0

Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Cooldown effects on reactor power

Proposed Question: #17

The plant has experienced a failure to scram from 100% power with the following:

- Reactor pressure is 900 psig and stable.
- Standby Liquid Control has NOT yet been initiated.
- SRMs and IRMs are fully inserted.
- Power is on IRM range 4 and slowly lowering.
- The Unit Supervisor has given the order to cooldown to cold shutdown conditions.

While cooling down, the following is observed when Reactor pressure reaches 500 psig:

- Power on IRMs and SRMs begins to rise.
- Period indication is 100 seconds.

Which one of the following is required, in accordance with N2-EOP-C5, Failure to Scram?

Stop the cooldown and…

A. stabilize Reactor pressure near 500 psig.

B. raise pressure back to 900 psig.

C. initiate Standby Liquid Control. Once the Reactor becomes subcritical, continue the cooldown.

D. initiate Standby Liquid Control. Once SLS tank level drops to <2000 gallons, continue the cooldown.

Proposed Answer: A

Explanation: Per N2-EOP-C5, if the Reactor goes critical during cooldown, then pressure must be stabilized below 1052 psig. The EOP bases state that stabilizing pressure would be to pick a value close to the initial value but below the SCRAM setpoint. Since Reactor pressure at the time of criticality is 500 psig, then stabilizing at 500 psig would be appropriate.

B. Incorrect – Pressure must be stabilized, not raised significantly. Plausible because initial Reactor pressure was 900 psig.
C. Incorrect – There is no requirement to initiate Standby Liquid Control. Plausible because this would insert negative reactivity and is allowable.
D. Incorrect – There is no requirement to initiate Standby Liquid Control. Plausible because this would insert negative reactivity and is allowable.

Technical Reference(s): N2-EOP-C5, NER-2M-039, TMGs

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC5C01 EO-2

Question Source: Bank – 2012 NRC #39

Question History: 2012 NRC #39

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 EK2.05
	Importance Rating	3.7

High Offsite Radioactivity Release Rate

Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Site emergency plan

Proposed Question: #18

N2-EOP-RR, Radioactivity Release Control, contains a step that requires RPV Blowdown prior to exceeding a certain Emergency Action Level related to release rates.

Which one of the following identifies:

(1) the Emergency Action Level that is used in the step to control when RPV Blowdown is performed

and

(2) the reason for the RPV Blowdown

in accordance with N2-EOP-RR?

A. (1) Alert
(2) Maintain Control Room habitability

B. (1) Alert
(2) Limit dose to the general public

C. (1) General Emergency
(2) Maintain Control Room habitability

D. (1) General Emergency
(2) Limit dose to the general public

Proposed Answer: D

Explanation: N2-EOP-RR requires RPV Blowdown before offsite release rate reaches the General Emergency level. The basis is to limit dose to the general public.

A. Incorrect – N2-EOP-RR requires RPV Blowdown before offsite release rate reaches the General Emergency level. Plausible because N2-EOP-RR entry utilizes the Alert level. The basis is to limit dose to the general public. Plausible because Control Room habitability is also a concern during a release and forms the basis for Control Room HVAC and shielding design.
B. Incorrect – N2-EOP-RR requires RPV Blowdown before offsite release rate reaches the General Emergency level. Plausible because N2-EOP-RR entry utilizes the Alert level.
C. Incorrect – The basis is to limit dose to the general public. Plausible because Control Room habitability is also a concern during a release and forms the basis for Control Room HVAC and shielding design.

Technical Reference(s): N2-EOP-RR, NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRRC01 EO-2

Question Source: Bank - JAF 17-2 NRC #18

Question History: JAF 17-2 NRC #18

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

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

Following a fire in the Main Control Room, N2-SOP-78, Control Room Evacuation, directs that the Appendix R Disconnect switches on 2CES*PNL415/416 (Control Building 306' west and east cable chases) be placed in the ACTUATE position.

Which one of the following describes the reason for these actions with respect to operation of RCIC?

When placed in ACTUATE, these switches…

A. install separate auto isolation circuits to ensure that fire damage will NOT prevent required isolation.

B. install separate auto initiation circuits to ensure that fire damage will NOT prevent required system initiation.

C. enable Remote Shutdown Panel controls to allow operation of the system from outside of the Main Control Room.

D. eliminate auto isolation signals and inhibit auto start signals to prevent spurious isolation or initiation of the system.

Proposed Answer: D

Explanation: Placing the SW1-2CESA02 Appendix R Disconnect switch to ACTUATE eliminates RCIC auto isolation signals and inhibits the auto start on low level. These actions are desired to prevent spurious operation of RCIC due to fire induced circuit damage.

A. Incorrect – Separate auto isolation circuitry is not connected. Plausible because auto isolation circuitry is disconnected and connecting separate circuitry outside of the control room would maintain isolation function while still getting rid of fire induced damage concerns.
B. Incorrect – Separate auto initiation circuitry is not connected. Plausible because auto initiation circuitry is disconnected and connecting separate circuitry outside of the control room would maintain initiation function while still getting rid of fire induced damage concerns.
C. Incorrect – Remote Shutdown Panel controls are not enabled by this action. Plausible because this is analogous to switches on panels 2CES*PNL405.

Technical Reference(s): N2-SOP-78, N2-OP-78 Attachment 2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-296000-RBO-5

Question Source: Bank - 2017 NRC #47

Question History: 2017 NRC #47

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

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

Generator Voltage and Electric Grid Disturbances

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

Proposed Question: #20

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

- 2RHS*P1A, RHR pump A, is running for Suppression Pool Cooling.
- Then, a loss of offsite power (LOOP) occurs due to an electrical grid disturbance.
- All EDGs start and re-power their respective buses.

Which one of the following identifies the response of 2RHS*P1A?

2RHS*P1A…

A. automatically restarts immediately.

B. automatically restarts after a time delay.

C. does NOT automatically restart, but may be manually restarted at any time.

D. does NOT automatically restart, and may be manually restarted only after a time delay.

Proposed Answer: C

Explanation: 2RHS*P1A is powered from 2ENS*SWG101. On the loss of offsite power, this bus de-energizes and load shedding opens the breaker for 2RHS*P1A. Once the EDG starts and re-powers 2ENS*SWG101, 2RHS*P1A remains off until manually started. It is not blocked from being started for any amount of time.

A. Incorrect – 2RHS*P1A remains off until manually started. Plausible because the pump was initially running.
B. Incorrect – 2RHS*P1A remains off until manually started. Plausible because this would be correct if a LOCA signal were present.
D. Incorrect – It is not blocked from being started for any amount of time. Plausible because this is correct for some other loads (70 seconds).

Technical Reference(s): 05ENS021 sheets 1 and 2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	2
	K/A #	295002 AA2.04
	Importance Rating	2.8

Loss of Main Condenser Vacuum

Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM: Off-gas system flow

Proposed Question: #21

With the plant operating at 100% power, which one of the following is an indication of higher than normal Main Condenser air in-leakage?

	Offgas System Flow	Offgas System Pressure
A.	Lowering	Lowering
B.	Lowering	Rising
C.	Rising	Lowering
D.	Rising	Rising

Proposed Answer: D

Explanation: With the Main Condenser normally at a vacuum, rising air in-leakage causes vacuum to lower (or absolute pressure to rise) due to the presence of more non-condensable gases. This higher absolute pressure causes Offgas pressure to also rise. The rise in non-condensable gases re-sults in more flow from the Steam Jet Air Ejectors to the Offgas system, which causes Offgas flow to rise.

A. Incorrect – Offgas flow rises, not lowers. Plausible because other parameters (eg. Main Condenser vacuum) lower. Offgas pressure rises, not lowers. Plausible because other parameters (eg. Main Condenser vacuum) lower.
B. Incorrect – Offgas flow rises, not lowers. Plausible because other parameters (eg. Main Condenser vacuum) lower.
C. Incorrect – Offgas pressure rises, not lowers. Plausible because other parameters (eg. Main Condenser vacuum) lower.

Technical Reference(s): N2-SOP-9, ARP 851306, N2-ARP-OFG-IPNL-122

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-271000-RBO-8

Question Source: Bank - 2017 NRC #59

Question History: 2017 NRC #59

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	2
	K/A #	295008 AK1.02
	Importance Rating	2.8

High Reactor Water Level

Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL: Component erosion/damage

Proposed Question: #22

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

- A Feedwater Level Control malfunction has occurred.
- Reactor water level is 195" and rising quickly.
- Attempts to take manual control of Feedwater have failed.

Which one of the following describes an automatic protective action that is imminent and the reason for this protective action?

	Protective Action	Reason
A.	MSIV closure	Prevent flooding of the steam lines
B.	MSIV closure	Prevent moisture carryover to the Main Turbine
C.	Main Turbine trip	Prevent flooding of the steam lines
D.	Main Turbine trip	Prevent moisture carryover to the Main Turbine

Proposed Answer: D

Explanation: The Main Turbine trips on high Reactor water level at 202.3". This is to prevent moisture carryover from damaging Turbine components.

A. Incorrect – There is no automatic closure of MSIVs on high Reactor water level. Plausible because N2-SOP-101C does have guidance for manual closure of MSIVs on high Reactor water level.
B. Incorrect – There is no automatic closure of MSIVs on high Reactor water level. Plausible because N2-SOP-101C does have guidance for manual closure of MSIVs on high Reactor water level.
C. Incorrect – The reason is to prevent moisture carryover from damaging Turbine components. Plausible because this is a concern and the basis for manually closing MSIVs on high Reactor water level.

Technical Reference(s): N2-EOP-RPV, N2-245000-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245000-RBO-8

Question Source: Bank – NMP1 2013 Cert #21

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295010 AK2.01
	Importance Rating	3.2

High Drywell Pressure

Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: Suppression pool level

Proposed Question: #23

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

- RHR B is in the Suppression Pool Pumpdown lineup, in accordance with N2-OP-31, Residual Heat Removal System.
- 2RHS*MOV142 and 2RHS*MOV149, RHR DISCHARGE TO RADWASTE MOVs, are open.
- Suppression Pool water level is 200.7' and slowly lowering.

Then, a small steam leak in the Drywell causes Drywell pressure to reach 1.68 psig.

Which one of the following describes the resulting status of RHR B?

The RHR B Suppression Pool Pumpdown lineup…

A. automatically isolates. RHR B is capable of automatically realigning to the LPCI mode.

B. automatically isolates. RHR B is NOT capable of automatically realigning to the LPCI mode.

C. continues to lower Suppression Pool water level. RHR B is capable of automatically realigning to the LPCI mode.

D. continues to lower Suppression Pool water level. RHR B is NOT capable of automatically realigning to the LPCI mode.

Proposed Answer: A

Explanation: High Drywell pressure (1.68 psig) causes a Group 4 isolation signal, which closes 2RHS*MOV142 and 2RHS*MOV149 and isolates the Suppression Pool Pumpdown lineup. RHR B is capable of automatically realigning to the LPCI mode.

B. Incorrect – RHR B is capable of automatically realigning to the LPCI mode. Plausible that some other action in establishing the pumpdown lineup would prevent LPCI injection, such as in the procedure for RHR Suppression Pool Alternate Fill (N2-OP-31 H.6.0).
C. Incorrect – High Drywell pressure (1.68 psig) causes a Group 4 isolation signal, which closes 2RHS*MOV142 and 2RHS*MOV149 and isolates the Suppression Pool Pumpdown lineup. Plausible because without this isolation level would continue to lower.
D. Incorrect – High Drywell pressure (1.68 psig) causes a Group 4 isolation signal, which closes 2RHS*MOV142 and 2RHS*MOV149 and isolates the Suppression Pool Pumpdown lineup. Plausible because without this isolation level would continue to lower. RHR B is capable of automatically realigning to the LPCI mode. Plausible that some other action in establishing the pumpdown lineup would prevent LPCI injection, such as in the procedure for RHR Suppression Pool Alternate Fill (N2-OP-31 H.6.0).

Technical Reference(s): N2-OP-31, ARP 603101, N2-OP-83

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295032 EK3.01
	Importance Rating	3.5

High Secondary Containment Area Temperature

Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Emergency/normal depressurization

Proposed Question: #24

Which one of the following describes a reason for RPV Blowdown due to high Secondary Containment temperature, in accordance with N2-EOP-SC, Secondary Containment Control, and the associated bases?

RPV Blowdown is based on concerns for…

A. operability of the SRVs.

B. habitability of the Control Room.

C. operability of the Primary Containment.

D. loss of equipment located in the Reactor Building.

Proposed Answer: D

Explanation: NER-2M-039 contains the following bases for the RPV Blowdown on high Secondary Containment temperature: “A parameter above the Maximum Safe Value in two separate areas is indicative of a wide-spread problem posing a direct and immediate threat to secondary containment, **equipment in the secondary containment**, and safe operation of the plant. If a primary system is discharging into the secondary containment and area temperatures, radiation levels, or water levels are above their Maximum Safe Values in two or more areas, a blowdown must therefore be performed in accordance with EOP-C2.

A. Incorrect – This is not a stated reason for RPV Blowdown due to high Secondary Containment temperatures. Plausible because high temperatures in the Secondary Containment do impact the ability of personnel to enter the building.
B. Incorrect – This is not a stated reason for RPV Blowdown due to high Secondary Containment temperatures. Plausible because an un-isolable primary system discharging into the Reactor Building does lead to temperature and radiation concerns that could impact the Control Room.
C. Incorrect – This is not a stated reason for RPV Blowdown due to high Secondary Containment temperatures. Plausible because this is the reason for the RPV Blowdown on high Drywell temperature in EOP-PC.

Technical Reference(s): N2-EOP-SC, NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: Bank – JAF 18-1 NRC #24

Question History: JAF 18-1 NRC #24

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295014 AA1.06
	Importance Rating	3.3

Inadvertent Reactivity Addition

Ability to operate and/or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION: Reactor/turbine pressure regulating system

Proposed Question: #25

The plant is operating at 95% power when the following occurs:

- The center control rod drifts out from position 00 to 48.
- Attempts to insert the control rod have failed.
- NO other operator actions have yet been taken.
- Reactor power is 98% and stable.

Which one of the following describes how the EHC system controls Reactor pressure in response to this malfunction?

EHC throttles Turbine Control Valves further…

A. open and stabilizes Reactor pressure at the initial value.

B. closed and stabilizes Reactor pressure at the initial value.

C. open and stabilizes Reactor pressure slightly higher than the initial value.

D. closed and stabilizes Reactor pressure slightly lower than the initial value.

Proposed Answer: C

Explanation: The center control rod drifting from position 00 to 48 causes Reactor power to rise. This causes Reactor pressure to rise and Turbine inlet steam pressure to rise. EHC senses this higher pressure and throttle Turbine Control Valves further open to limit the pressure rise. The EHC system controls pressure proportional to steam flow, such that as steam flow rises, Reactor pressure will be controlled at a higher value.

A. Incorrect – The EHC system controls pressure proportional to steam flow, such that as steam flow rises, Reactor pressure will be controlled at a higher value. Plausible that the system would have an integral function to return Reactor pressure back to the initial value.
B. Incorrect – The given malfunction causes Turbine Control Valves to be throttled further open, not closed. Plausible because this would be correct for a transient that lowered Reactor power. The EHC system controls pressure proportional to steam flow, such that as steam flow rises, Reactor pressure will be controlled at a higher value. Plausible that the system would have an integral function to return Reactor pressure back to the initial value.
D. Incorrect – The given malfunction causes Turbine Control Valves to be throttled further open, not closed. Plausible because this would be correct for a transient that lowered Reactor power.

Technical Reference(s): N2-OP-23, N2-248000-RBO-2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-248000-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295020 AA2.03
	Importance Rating	3.7

Inadvertent Containment Isolation

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

Proposed Question: #26

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

- The Reactor Mode Switch is in RUN.
- An inadvertent isolation signal causes the following valves to close:
  - MSIV 2MSS*AOV7A
  - MSIV 2MSS*AOV7C
  - MSIV 2MSS*AOV6B

Which one of the following describes Reactor power as indicated on the APRMs one (1) minute later?

APRMs will indicate…

A. approximately 0%.

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

C. approximately 20%.

D. greater than 20%.

Proposed Answer: A

Explanation: The MSIV closure scram logic is arranged such that with the Reactor Mode Switch in RUN, a scram will occur if an MSIV is closed in at least 3 of the 4 Main Steam Lines. The given valves are distributed between 3 different Main Steam Lines, therefore a scram will occur, even though 20% power is within the capacity of a single Main Steam Line. APRMs drop to 0% following the scram.

B. Incorrect – Reactor power will indicate approximately 0% on APRMs. Plausible because Core Thermal Power will remain greater than 0% but below 20% for an extended period of time due to decay heat.
C. Incorrect – Reactor power will indicate approximately 0% on APRMs. Plausible because this would be correct if less than 3 Main Steam Lines were isolated.
D. Incorrect – Reactor power will indicate approximately 0% on APRMs. Plausible because Reactor pressure and power will initially rise when MSIVs first start to stroke closed and one Main Steam Line is capable of passing steam flow equivalent to greater than 20% Reactor power.

Technical Reference(s): 807E166TY sheet 9, PID-001E, PID-001F

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-212000-RBO-3

Question Source: Bank - JAF 18-1 NRC #22

Question History: JAF 18-1 NRC #22

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	2
	K/A #	295022 2.2.44
	Importance Rating	4.2

Loss of Control Rod Drive Pumps

Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question: #27

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

- The running CRD pump has tripped.
- The standby CRD pump has failed to start.
- Two accumulator trouble alarms have been received for withdrawn control rods.
- Both control rods have accumulator pressures of 920 psig.
- The CRD pumps cannot be started.

Which one of the following describes the requirement for a Reactor scram and the reason for this scram, in accordance with N2-SOP-30, Control Rod Drive Failures?

A Reactor scram is required ___(1)___. This scram is required due to degraded ___(2)___.

A. (1) immediately
   (2) control rod scram capability

B. (1) immediately
   (2) cooling to the CRD mechanisms

C. (1) 20 minutes after the loss of CRD Charging Header pressure
   (2) control rod scram capability

D. (1) 20 minutes after the loss of CRD Charging Header pressure
   (2) cooling to the CRD mechanisms

Proposed Answer: C

Explanation: With Reactor power at 100%, Reactor pressure is above 900 psig. With Reactor pressure above 900 psig, N2-SOP-30 requires a Reactor scram if two or more accumulators for withdrawn control rods are inoperable AND CRD charging water header pressure is <940 psig for 20 minutes. The reason for the scram is degraded control rod scram capability.

A. Incorrect – An immediate Reactor scram is not required. Plausible because this would be correct if Reactor pressure were <900 psig.
B. Incorrect – An immediate Reactor scram is not required. Plausible because this would be correct if Reactor pressure were <900 psig. The reason is degraded control rod scram capability. Plausible because cooling is lost to the CRD mechanisms with the loss of CRD pressure.
D. Incorrect – The reason is degraded control rod scram capability. Plausible because cooling is lost to the CRD mechanisms with the loss of CRD pressure.

Technical Reference(s): N2-SOP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP30C01 EO-2

Question Source: Bank - 2015 NRC #61

Question History: 2015 NRC #61

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	203000 K1.11
	Importance Rating	3.7

RHR/LPCI: Injection Mode

Knowledge of the physical connections and/or cause effect relationships between RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) and the following: Nuclear boiler instrumentation

Proposed Question: #28

The plant has experienced a loss of high pressure injection with the following:

- Reactor water level is 10".
- Reactor pressure is 450 psig.
- Drywell pressure is 0.6 psig.

Which one of the following describes the status of RHR?

RHR…

A. remains in a standby lineup.

B. is currently injecting to the Reactor.

C. pumps have automatically started, but will NOT inject until Reactor water level lowers further.

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

Proposed Answer: D

Explanation: Reactor water level <17.8” starts RHR pumps. Operating RHR pumps produce a discharge pressure of approximately 300 psig. With Reactor pressure at 450 psig, RHR will not inject to the Reactor until pressure lowers further.

A. Incorrect – RHR pumps have already started. Plausible that they would not start until Reactor water level lowers further with Drywell pressure below the initiation setpoint (1.68 psig).
B. Incorrect – RHR is operating in the LPCI mode, but Reactor pressure is still too high for injection to occur. Plausible because RHR heat exchanger pressure indication goes up to 600 psig.
C. Incorrect – Reactor water level is low enough, but Reactor pressure is not low enough. Plausible that they would not inject until Reactor water level lowers further with Drywell pressure below the initiation setpoint (1.68 psig). Also plausible because RHR heat exchanger pressure indication goes up to 600 psig.

Technical Reference(s): N2-OP-31, USAR

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 K2.01
	Importance Rating	3.1

Shutdown Cooling

Knowledge of electrical power supplies to the following: Pump motors

Proposed Question: #29

A plant shutdown is in progress when 2ENS*SWG101 de-energizes due to a sustained electrical fault.

Which one of the following describes the number of Shutdown Cooling loops that remain available and whether the preferred lineup of Alternate Shutdown Cooling would still be available?

	Number of Shutdown Cooling Loops Available	Is the Preferred Lineup of Alternate Shutdown Cooling Available?
A.	2	Yes
B.	2	No
C.	1	Yes
D.	1	No

Proposed Answer: C

Explanation: 2ENS*SWG101 is the power supply to RHR pump A. 2ENS*SWG103 is the power supply to RHR pumps B and C. RHR pumps B and C remain available, however only RHR pump B can be aligned for Shutdown Cooling. Therefore, only 1 loop of Shutdown Cooling is available. The preferred lineup of Alternate Shutdown Cooling can be accomplished using either RHR pump A or B. Therefore, the preferred lineup of Alternate Shutdown Cooling is still available using RHR pump B if needed.

A. Incorrect – Only 1 loop of Shutdown Cooling is available. Plausible because 2 RHR pumps still have power.
B. Incorrect – Only 1 loop of Shutdown Cooling is available. Plausible because 2 RHR pumps still have power. The preferred lineup of Alternate Shutdown Cooling is still available using RHR pump B if needed. Plausible because the preferred lineup of Alternate Shutdown Cooling can also be performed with RHR pump A, which is unavailable.
D. Incorrect – The preferred lineup of Alternate Shutdown Cooling is still available using RHR pump B if needed. Plausible because the preferred lineup of Alternate Shutdown Cooling can also be performed with RHR pump A, which is unavailable.

Technical Reference(s): N2-OP-31, N2-OP-31-LINEUPS, N2-SOP-31

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 K3.01
	Importance Rating	3.3

Shutdown Cooling

Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor pressure

Proposed Question: #30

The plant is in Mode 3 with the following:

- RHR B is operating in Shutdown Cooling (SDC).
- Both Recirc pumps are secured.
- Head spray is isolated.
- Reactor pressure is 20 psig and stable.
- 2RHS*MOV8B, Heat Exchanger 1B Inlet Bypass Valve Throttle, is throttled at an intermediate position:
- RHR Loop B flow indication on Panel 601 is 7000 gpm.

Then, the following occurs:

- 2RHS*MOV8B spuriously closes fully.
- 2RHS*MOV8B CANNOT be re-opened.

Which one of the following describes the resulting change in Reactor pressure and RHR Loop B flow indication on Panel 601?

	Reactor Pressure	RHR Loop B Flow Indication
A.	Rises	Rises
B.	Rises	Lowers
C.	Lowers	Rises
D.	Lowers	Lowers

Proposed Answer: D

Explanation: 2RHS*MOV8B closing causes flow through the RHR heat exchanger to rise. This causes more cooling. Since Reactor pressure was stable with the initial flows, raising the heat removal rate will cause pressure to lower. The RHR Loop B flow indication on Panel 601 includes both flow through the heat exchanger and through the bypass line. While heat exchanger flow rises, total flow indication lowers due to the given valve closure.

A. Incorrect – Reactor pressure lowers. Plausible because this malfunction does cause total system flow to lower, however flow through the heat exchanger rises. If flow through the heat exchanger lowered, Reactor pressure would rise. RHR Loop B flow indication lowers. Plausible because flow through the heat exchanger rises.
B. Incorrect – Reactor pressure lowers. Plausible because this malfunction does cause total system flow to lower, however flow through the heat exchanger rises. If flow through the heat exchanger lowered, Reactor pressure would rise.
C. Incorrect – RHR Loop B flow indication lowers. Plausible because flow through the heat exchanger rises.

Technical Reference(s): N2-OP-31, RHR PID

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-

Question Source: Modified Bank - 2017 NRC #15

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Proposed Question: #15

The plant is in Mode 3 with the following:

- RHR B is operating in Shutdown Cooling (SDC).
- Both Recirc pumps are secured.
- Head spray is isolated.
- Reactor coolant temperature is 250˚F and stable.
- The following valves are all throttled at an intermediate position:
  - 2RHS*MOV40B, SDC B Return Throttle
  - 2RHS*MOV8B, Heat Exchanger 1B Inlet Bypass Valve Throttle
  - 2SWP*MOV33B, Heat Exchanger 1B Service Water Outlet Valve
- RHR B total flow is 7000 gpm.
- Service Water to RHR B Heat Exchanger flow is 2000 gpm.
- Service Water temperature is 74°F.
- Five (5) Service Water pumps are in operation.

Then, the following occurs:

- 2RHS*MOV8B spuriously opens fully.
- RHR B total flow rises to 7450 gpm.
- 2RHS*MOV8B CANNOT be closed.

Which one of the following describes the resulting trend in Reactor coolant temperature and an allowable operator action to stabilize reactor coolant temperature, in accordance with N2-OP-31, Residual Heat Removal System?

	Reactor Coolant Temperature Trend		Allowable Operator Action
A.	Rises		Throttle 2RHS*MOV40B
B.	Rises		Throttle 2SWP*MOV33B
C.	Lowers		Throttle 2RHS*MOV40B
D.	Lowers		Throttle 2SWP*MOV33B

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 K4.02
	Importance Rating	3.0

Low Pressure Core Spray

Knowledge of LOW-PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: Prevents water hammer

Proposed Question: #31

The plant is operating at 100% power with all ECCS systems in the normal standby lineup.

Which one of the following describes the status of CSL*P2, LPCS/RHR A WTR LEG PMP, and where it discharges in the Low Pressure Core Spray system?

CSL*P2 is…

A. running. The pump discharges upstream of CSL*MOV104, PMP 1 INJECTION VLV.

B. running. The pump discharges downstream of CSL*MOV104, PMP 1 INJECTION VLV.

C. in standby. The pump discharges upstream of CSL*MOV104, PMP 1 INJECTION VLV.

D. in standby. The pump discharges downstream of CSL*MOV104, PMP 1 INJECTION VLV.

Proposed Answer: A

Explanation: CSL*P2 runs continuously when ECCS systems are in the normal standby lineup. The pump discharges upstream of CSL*MOV104, PMP 1 INJECTION VLV.

B. Incorrect – The pump discharges upstream of CSL*MOV104, PMP 1 INJECTION VLV. Plausible because the pump discharges downstream of the discharge check valve.
C. Incorrect – CSL*P2 runs continuously when ECCS systems are in the normal standby lineup. Plausible that it would only be run intermittently to fill the piping or that another source would be normally aligned to maintain pressure (ex. Condensate Transfer).
D. Incorrect – CSL*P2 runs continuously when ECCS systems are in the normal standby lineup. Plausible that it would only be run intermittently to fill the piping or that another source would be normally aligned to maintain pressure (ex. Condensate Transfer). The pump discharges upstream of CSL*MOV104, PMP 1 INJECTION VLV. Plausible because the pump discharges downstream of the discharge check valve.

Technical Reference(s): N2-OP-32, PID-032

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-209001-RBO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209002 K5.01
	Importance Rating	3.8

High Pressure Core Spray

Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE CORE SPRAY SYSTEM (HPCS): Adequate core cooling: BWR-5,6

Proposed Question: #32

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

- Indicated reactor water level is -63” and stable on Fuel Zone instruments.
- Reactor pressure is 50 psig and stable.
- HPCS is injecting 5500 gpm to the Reactor.
- NO other injection systems are available.

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

Adequate core cooling is…

A. NOT being provided.

B. being provided by Submergence.

C. being provided by Steam Cooling.

D. being provided by Core Spray Cooling.

Proposed Answer: A

Explanation: The given combination of indicated Fuel Zone level and Reactor pressure result in a corrected Fuel Zone water level of approximately -58". This is below the levels required for adequate core cooling to be provided by Submergence or Steam Cooling. This is above the level required for Core Spray Cooling. However, a malfunction of HPCS is causing its flow to be below that required for Core Spray Cooling (6350 gpm). With none of these three adequate core cooling mechanisms satisfied, adequate core cooling is not being provided.

B. Incorrect – Actual Reactor water level is approximately -58", which is below the level required for Submergence. Plausible because Submergence would be provided if Reactor water level were above -14". At higher Reactor pressures, the given indicated Fuel Zone level would result in an actual level much closer to -14".
C. Incorrect – Actual Reactor water level is approximately -58", which is below the level required for Steam Cooling. Plausible because Steam Cooling would be provided if Reactor water level were above -39" but below -14". At higher Reactor pressures, the given indicated Fuel Zone level would result in an actual level in this range.
D. Incorrect – A malfunction of HPCS is causing its flow to be below that required for Core Spray Cooling (6350 gpm). Plausible because actual Reactor water level is approximately -58", which is above the level required for Core Spray Cooling, and HPCS flow is close to the required value.

Technical Reference(s): N2-EOP-RPV, NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-209002-RBO-10

Question Source: Bank – 2017 Cert #5

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 K6.03
	Importance Rating	3.2

Standby Liquid Control

Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY LIQUID CONTROL SYSTEM: A.C. power

Proposed Question: #33

A failure to scram has resulted in the following:

- Reactor power is 25% and stable.
- 600 VAC 2EHS*MCC102 is de-energized due to a sustained electrical fault.

Which one of the following describes the Standby Liquid Control (SLS) pump that has power available and the number of explosive valves that have power available?

SLS pump ___(1)___ has power available.

___(2)___ explosive valve(s) has(have) power available.

	(1)	(2)
A.	A	Only one
B.	A	Two
C.	B	Only one
D.	B	Two

Proposed Answer: C

Explanation: 2EHS*MCC102 supplies power to SLS pump A and both squibs for 2SLS*VEX3A. With 2EHS*MCC102 de-energized, SLS pump B and only one explosive valve, 2SLS*VEX3B, have power available.

A. Incorrect – SLS pump B has power available. Plausible because the SLS pumps have similar power supplies.
B. Incorrect – SLS pump B has power available. Plausible because the SLS pumps have similar power supplies. Both squibs for 2SLS*VEX3A will not fire due to loss of 2EHS*MCC102. Plausible that the squibs would be powered by different sources to provide more redundancy.
D. Incorrect – Both squibs for 2SLS*VEX3A will not fire due to loss of 2EHS*MCC102. Plausible that the squibs would be powered by different sources to provide more redundancy.

Technical Reference(s): N2-OP-36A-Lineups, 807E161TY Sheet 4

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-211000-RBO-4

Question Source: Bank - 2017 Cert #4

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 A1.04
	Importance Rating	3.6

Standby Liquid Control

Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: Valve operations

Proposed Question: #34

The plant has experienced a failure to scram with the following:

- RRCS fails to actuate.
- The Standby Liquid Control (SLC) explosive valve current meters initially indicate as shown below:

- Then, the 2SLS*P1A, PMP 1A, keylock control switch is placed in the PUMP A RUN position.

Which one of the following describes the resulting steady-state indication of the current meters for SLC explosive valves 2SLS*VEX3A and 2SLS*VEX3B?

A. Both indicate upscale.

B. Both indicate downscale.

C. 3A indicates upscale and 3B indicates the initial value.

D. 3A indicates downscale and 3B indicates the initial value.

Proposed Answer: D

Explanation: When the SLC pump A key lock selector switch is taken to PUMP A RUN, SLC pump A starts and only explosive valve 2SLS*VEX3A fires. When the valve fires, current rises to approximately 2 amps (upscale indication) for a very short period of time and then falls to a steady-state value of 0 amps (downscale indication).

A. Incorrect – Explosive valve 3A will indicate downscale. Plausible because it will temporarily go upscale. Explosive valve 3B current indications will remain at the initial value. Plausible that both valves fire to provide extra redundancy.
B. Incorrect – Explosive valve 3B current indications will remain at the initial value. Plausible that both valves fire to provide extra redundancy.
C. Incorrect – Explosive valve 3A will indicate downscale. Plausible because it will temporarily go upscale.

Technical Reference(s): N2-OP-36A, 2101-211000C01

Proposed references to be provided to applicants during examination: Embedded picture

Learning Objective: N2-211000-RBO-5

Question Source: Bank – 2017 Cert #9

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 A2.15
	Importance Rating	3.7

Reactor Protection System

Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Load rejection

Proposed Question: #35

The plant is operating at 22% power with the following:

- Main Generator load is 250 MWe.
- Then, a Main Generator lockout occurs.

Which one of the following describes the need to execute N2-SOP-101C, Reactor Scram, and N2-SOP-21, Turbine Trip?

A. Both of these procedures need to be executed.

B. NEITHER of these procedures need to be executed.

C. N2-SOP-101C needs to be executed, but N2-SOP-21 does NOT.

D. N2-SOP-21 needs to be executed, but N2-SOP-101C does NOT.

Proposed Answer: D

Explanation: The Main Generator lockout causes a Main Turbine trip, therefore N2-SOP-21 must be executed. With Reactor power below 26%, the Reactor scrams on Turbine control and stop valve position are bypassed. Therefore, no automatic Reactor scram occurs. There is also no requirement for inserting a manual Reactor scram. Therefore, N2-SOP-101C does not need to be executed.

A. Incorrect – N2-SOP-101C does not need to be executed. Plausible because this would be correct if power was above 26%.
B. Incorrect – N2-SOP-21 must be executed. Plausible because power is so low and the Main Turbine is able to be online (not tripped) prior to the Main Generator being connected to the grid during a startup.
C. Incorrect – N2-SOP-101C does not need to be executed. Plausible because this would be correct if power was above 26%. N2-SOP-21 must be executed. Plausible because power is so low and the Main Turbine is able to be online (not tripped) prior to the Main Generator being connected to the grid during a startup. Also plausible that N2-SOP-21 would not need to be executed on a Main Turbine trip that was accompanied by a Reactor scram.

Technical Reference(s): ARP 852614, N2-SOP-21, ARP 603112, ARP 851109

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-212000-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 A3.01
	Importance Rating	3.3

Intermediate Range Monitor

Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including: Meters and recorders

Proposed Question: #36

The plant is operating in Mode 2 with IRM A indicating 12 on range 4.

Which one of the following describes if IRM A is currently causing a rod block and the required manipulation of the IRM A range switch to restore indication to the band required by N2-OP-101A, Plant Startup?

IRM A is…

A. causing a rod block. Place IRM A on range 3.

B. causing a rod block. Place IRM A on range 5.

C. NOT causing a rod block. Place IRM A on range 3.

D. NOT causing a rod block. Place IRM A on range 5.

Proposed Answer: C

Explanation: IRM A is indicating below the range of 25-75 required by N2-OP-101A, but above the downscale rod block setpoint of 5. Therefore, IRM A is NOT causing a rod block. To raise the IRM A indication, it must be placed on range 3.

A. Incorrect – IRM A is indicating above the downscale rod block setpoint of 5. Plausible because IRM A is indicating low and will cause a rod block if it goes below 5.
B. Incorrect – IRM A is indicating above the downscale rod block setpoint of 5. Plausible because IRM A is indicating low and will cause a rod block if it goes below 5. IRM A must be taken to range 3, not range 5. Plausible because if the IRM were indicating high, then it would need to be taken to range 5.
D. Incorrect – IRM A must be taken to range 3, not range 5. Plausible because if the IRM were indicating high, then it would need to be taken to range 5.

Technical Reference(s): ARP 603213, N2-OP-101A

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215002-RBO-5

Question Source: Bank - 2017 Cert #19

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 A4.04
	Importance Rating	3.2

Source Range Monitor

Ability to manually operate and/or monitor in the control room: SRM drive control switches

Proposed Question: #37

A plant startup is in progress with the following:

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

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

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

B. withdraws and immediately generates a rod block signal.

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

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

Proposed Answer: B

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

A. Incorrect – SRM A will withdraw. Plausible because the RETRACT PERMIT light will not be lit.
C. Incorrect – SRM A will withdraw. Plausible because the RETRACT PERMIT light will not be lit.
D. Incorrect – The SRM Detector Not Full In rod block will be received as soon as SRM A is not fully inserted. Plausible because counts <3 cps is another rod block signal.

Technical Reference(s): N2-OP-92, ARP 603216

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215002-RBO-5

Question Source: Bank – JAF 14-2 NRC #8

Question History: JAF 14-2 NRC #8

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 2.1.32
	Importance Rating	3.8

Source Range Monitor

Ability to explain and apply system limits and precautions.

Proposed Question: #38

A plant startup is in progress. All SRMs indicated 140 cps prior to commencing the startup.

Which one of the following identifies the SRM count rate to be used as the threshold for determining when single notch control rod movement is required, in accordance with N2-OP-101A, Plant Startup?

A. 280 cps

B. 560 cps

C. 840 cps

D. 1120 cps

Proposed Answer: D

Explanation: N2-OP-101A uses 3 SRM count rate doublings as the threshold for determining when single notch control rod movement is required. With an initial count rate of 140 cps, the first doubling occurs at 280 cps, the second doubling occurs at 560 cps, and the third doubling occurs at 1120 cps.

A. Incorrect – The threshold is 1120 cps. Plausible because this is when the first count rate doubling occurs.
B. Incorrect – The threshold is 1120 cps. Plausible because this is when the second count rate doubling occurs.
C. Incorrect – The threshold is 1120 cps. Plausible because this is 6x the initial count rate, which could be reached if the process for calculating 3 doublings was misunderstood.

Technical Reference(s): N2-OP-101A

Proposed references to be provided to applicants during examination: none

Learning Objective: N2-215002-RBO-9

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 K1.12
	Importance Rating	3.2

Average Power Range Monitor/Local Power Range Monitor

Knowledge of the physical connections and/or cause effect relationships between AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM and the following: Full core display

Proposed Question: #39

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

- Annunciator 603211, LPRM DOWNSCALE, alarms.
- LPRM 56-33A has failed downscale.

Which one of the following identifies automatic indications that occur, if any, on the full core display and the LPRM output meters surrounding the four rod display?

A. NO change occurs on the full core display. The LPRM output meters are automatically aligned to display the failed LPRM.

B. NO change occurs on the full core display. NO change occurs on the LPRM output meters unless an appropriate control rod is manually selected.

C. A white downscale light illuminates on the full core display. The LPRM output meters are automatically aligned to display the failed LPRM.

D. A white downscale light illuminates on the full core display. NO change occurs on the LPRM output meters unless an appropriate control rod is manually selected.

Proposed Answer: D

Explanation: The full core display includes a white downscale light for each LPRM that automatically illuminates if the LPRM fails downscale. The four rod display is surrounded by four sets of LPRM output meters that are aligned to LPRMs based on which control rod is currently selected. Failure of an LPRM does not automatically align these meters to display the affected LPRM.

A. Incorrect – The full core display does contain a white downscale light for each LPRM. Plausible if the candidate does not remember all of the indications available on the full core display. The LPRM output meters are not automatically aligned to the failed LPRM. Plausible because it is desired to confirm the failed LPRM output on one of these meters and no control rod is currently selected that would require use of these meters.
B. Incorrect – The full core display does contain a white downscale light for each LPRM. Plausible if the candidate does not remember all of the indications available on the full core display.
C. Incorrect – The LPRM output meters are not automatically aligned to the failed LPRM. Plausible because it is desired to confirm the failed LPRM output on one of these meters and no control rod is currently selected that would require use of these meters.

Technical Reference(s): ARP 603211

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215003-RBO-5

Question Source: Bank - 2017 Cert #18

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 K2.01
	Importance Rating	2.8

Reactor Core Isolation Cooling

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

Proposed Question: #40

The plant is operating at 100% power when 2EHS*MCC102 de-energizes due to a sustained electrical fault.

Which one of the following identifies a valve in the RCIC system that will NOT operate due to this power loss?

A. 2ICS*MOV120, TURB STM SUPPLY VLV

B. 2ICS*MOV121, TURB STM SUPPLY OUTBOARD ISOL VLV

C. 2ICS*MOV129, PMP 1 SUCT FROM CONDENSATE STOR TK

D. 2ICS*MOV150, TURBINE TRIP THROTTLE VLV

Proposed Answer: B

Explanation: 2ICS*MOV121 is powered by 2EHS*MCC102 and will not operate due to this power loss.

A. Incorrect – This valve is powered from 2DMS*MCCA1. Plausible because this is a motor operated valve.
C. Incorrect – This valve is powered from 2DMS*MCCA1. Plausible because this is a motor operated valve.
D. Incorrect – This valve is powered from 2DMS*MCCA1. Plausible because this is a motor operated valve.

Technical Reference(s): N2-OP-35-LINEUPS

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-217000-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

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

Reactor Core Isolation Cooling

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

Proposed Question: #41

The plant has scrammed with the following:

- Reactor pressure is 920 psig and stable.
- Reactor water level is 180" and stable.
- RCIC is injecting to the Reactor at 200 gpm.
- The RCIC flow controller is in Automatic.

Then, the RCIC pump discharge flow element fails low (senses zero flow).

Which one of the following describes the resulting change in the rate of decay heat removal by RCIC and the ability to control RCIC flow with the flow controller in Manual?

The rate of decay heat removal ___(1)___. The RCIC flow controller ___(2)___ function properly in Manual.

	(1)	(2)
A.	rises	will NOT
B.	rises	will
C.	lowers	will NOT
D.	lowers	will

Proposed Answer: B

Explanation: With the RCIC pump discharge flow element (2ICS*FT101) failed low, the flow input signal to the flow controller lowers. The flow controller senses that flow if below the demanded flow of 200 gpm and further opens the RCIC throttle valve 2ICS*MOV126. This removes more steam from the Reactor, which raises the rate of decay heat removal. The flow element failure has no adverse effect on the control of RCIC flow in Manual.

A. Incorrect – The flow controller will still function properly in Manual. Plausible because control room indicated flow will still be failed low. Also plausible because some failures would affected the controller in both Automatic and Manual.
C. Incorrect – The rate of decay heat removal rises, not lowers. Plausible indicated flow is low, which would correlate to lower steam flow to RCIC. Also plausible because this would be correct on the opposite failure. The flow controller will still function properly in Manual. Plausible because control room indicated flow will still be failed low. Also plausible because some failures would affected the controller in both Automatic and Manual.
D. Incorrect – The rate of decay heat removal rises, not lowers. Plausible indicated flow is low, which would correlate to lower steam flow to RCIC. Also plausible because this would be correct on the opposite failure.

Technical Reference(s): N2-OP-35, N2-217000-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-217000-RBO-11

Question Source: Modified Bank – JAF 17-2 NRC #39

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	217000 K3.01
	Importance Rating	3.7

RCIC

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

Proposed Question: #39

The plant has scrammed with the following:

- Reactor pressure is 920 psig and stable.
- Reactor water level is 190" and stable.
- RCIC is injecting to the Reactor at 300 gpm.
- The RCIC flow controller is in AUTO.

Then, the RCIC pump discharge flow element fails low (indicates zero flow).

Which one of the following describes the response of Reactor water level and the ability to control RCIC flow with the flow controller in MAN?

Reactor water level ___(1)___. The RCIC flow controller ___(2)___ function properly in MAN.

	(1)	(2)
A.	lowers	will
B.	lowers	will NOT
C.	rises	will
D.	rises	will NOT

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 K4.01
	Importance Rating	3.7

Automatic Depressurization System

Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following: Prevent inadvertent initiation of ADS logic

Proposed Question: #42

Which one of the following describes the ***minimum*** combination of Reactor water level instrument signals needed to cause ADS to open SRVs?

Note: Assume all ECCS pumps are running properly.

A. Two Level 1 signals, only

B. Two Level 3 signals, only

C. Two Level 1 signals and one Level 3 signal, only

D. Two Level 1 signals and two Level 3 signals

Proposed Answer: C

Explanation: Four level transmitters (2ISC*LT9A-D) provide Level 1 signals to the ADS logic. Two level transmitters (2ISC*LT12A/B) provide Level 3 signals to the ADS logic. For ADS to open SRVs, either Logic Channels A and E of Logic Channels B and F must be satisfied. For these Logic Channels to be satisfied, two of the Level 1 signals must be present and one of the Level 3 signals must be present.

A. Incorrect – One Level 3 signal is also required. Plausible because Level 3 (159.3") is much higher than the Reactor water level at which ADS actuates (Level 1 = 17.8").
B. Incorrect – Two Level 1 signals are required and only one level 3 signal. Plausible because this is still 2 different Reactor water level signals and provides some degree of protection against inadvertent initiation.
D. Incorrect – Only one Level 1 signal is required. Plausible because most logics require 2 of any given signal to actuate.

Technical Reference(s): ADS Logic Print

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-218000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 K4.06
	Importance Rating	3.4

Primary Containment Isolation/Nuclear Steam Supply Shutoff

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

Proposed Question: #43

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

- A Reactor scram occurs.
- PCIS Group 6 and 7 isolations occur on low Reactor water level.
- Reactor water level is restored and maintained in a band of 178-183".

Which one of the following describes the operation of the Reactor Water Cleanup (WCS) isolation valves?

WCS isolation valves…

A. automatically re-open.

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

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

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

Proposed Answer: C

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

A. Incorrect – The PCIS signals seal in and must be manually reset. Plausible because the initial system status had these valves open.
B. Incorrect – Even once the PCIS signals are manually reset, the WCS isolation valves remain closed until operator action is taken to re-open them. Plausible because the initial system status had these valves open.
D. Incorrect – The PCIS signals seal in and must be manually reset. Plausible because the valves would still be maintained closed until operator action was taken.

Technical Reference(s): N2-OP-37, N2-OP-83, N2-SOP-83

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223002-RBO-5

Question Source: Bank - JAF 14-2 NRC #7

Question History: JAF 14-2 NRC #7

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

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

Safety Relief Valves

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

Which one of the following describes the response of an SRV to a rising Reactor pressure if a pneumatic supply is NOT available from any source?

	SRV Safety Function	SRV Relief Function
A.	Will operate	Will operate
B.	Will operate	Will NOT operate
C.	Will NOT operate	Will operate
D.	Will NOT operate	Will NOT operate

Proposed Answer: B

Explanation: A pneumatic supply (accumulator / continuous) is required to lift an SRV in the Relief mode, but not in the Safety mode. The Safety mode is actuated when steam pressure overcomes spring force of the SRV.

A. Incorrect – A pneumatic supply (accumulator / continuous) is required to lift an SRV in the Relief mode. Plausible that an electrical signal alone could lift the valve.
C. Incorrect – The Safety mode will operate without a pneumatic supply. Plausible because other modes will not operate (Relief/ADS). A pneumatic supply (accumulator / continuous) is required to lift an SRV in the Relief mode. Plausible that an electrical signal alone could lift the valve.
D. Incorrect – The Safety mode will operate without a pneumatic supply. Plausible because other modes will not operate (Relief/ADS).

Technical Reference(s): N2-OP-34

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-218000-RBO-1

Question Source: Bank - 2010 NRC #1

Question History: 2010 NRC #1

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 A1.04
	Importance Rating	3.6

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 water level control controller indications

Proposed Question: #45

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

- The Feedwater Master M/A station setpoint is 185".
- A spurious Reactor scram occurs.
- All control rods insert.
- The Reactor Mode switch is placed in SHUTDOWN.
- Reactor water level reaches a low of 130" during the transient and stays at this level for approximately 10 seconds.

Which one of the following describes the response of the Feedwater Master M/A station ***setpoint***?

Feedwater Master M/A station ***setpoint***…

A. remains at 185" throughout the transient.

B. lowers to 153" and remains there until operator action is taken.

C. lowers to 153", but then rises and stabilizes at 175".

D. lowers to 153", but then rises and stabilizes at 185".

Proposed Answer: C

Explanation: Since Reactor water level goes below 159.3”, the Level Setpoint Setdown logic actuates. This lowers the Feedwater Master M/A station setpoint to 153”. After a 60 second time delay, the setpoint will slowly ramp to 175” and stabilize there.

A. Incorrect – Since Reactor water level goes below 159.3”, the Level Setpoint Setdown logic actuates. This lowers the Feedwater Master M/A station setpoint . Plausible because this would be correct if Reactor water level remained above 159.3” throughout the transient.
B. Incorrect – After a 60 second time delay, the setpoint will slowly ramp to 175” and stabilize there. Plausible because the setpoint initially goes to 153” and remains there for 60 seconds.
D. Incorrect – The setpoint stabilizes at 175”. Plausible because initial setpoint was 185”.

Technical Reference(s): N2-OP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-259002-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	261000 A2.01
	Importance Rating	2.9

Standby Gas Treatment

Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low system flow

Proposed Question: #46

The plant is operating at 100% power when the following occurs:

- Drywell pressure rises to 14 psig.
- GTS Trains A and B automatically start.
- GTS Train A is shut down by placing the Train A Initiation control switch in Auto After Stop.

Then, GTS Train B fan trips due to a blown control power fuse.

Which one of the following describes the impact of these events on Reactor Building differential pressure (D/P) and the actions required to restore RB differential pressure, in accordance with N2-OP-61B, Standby Gas Treatment System?

	Reactor Building D/P	Action Required
A.	Becomes less negative	Manually restart GTS Train A
B.	Becomes less negative	Confirm automatic restart of GTS Train A
C.	Becomes more negative	Manually restart GTS Train A
D.	Becomes more negative	Confirm automatic restart of GTS Train A

Proposed Answer: B

Explanation: When GTS Train B fan trips, GTS flow lowers. With less air being pulled out of the Reactor Building, pressure rises. With Reactor Building D/P negative, rising pressure equates to a less negative D/P. GTS Train A restarts because the high Drywell pressure signal is present and Reactor Building D/P will degrade to -0.25" WG.

A. Incorrect – Manual restart of GTS Train A is not required. Plausible because the train was manually stopped earlier.
C. Incorrect – Reactor Building D/P becomes less negative. Plausible if the direction of change of a negative pressure is confused. Manual restart of GTS Train A is not required. Plausible because the train was manually stopped earlier.
D. Incorrect – Reactor Building D/P becomes less negative. Plausible if the direction of change of a negative pressure is confused.

Technical Reference(s): N2-OP-61B

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-261000-RBO-5

Question Source: Bank - 2010 NRC #15

Question History: 2010 NRC #15

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 A3.01
	Importance Rating	3.1

AC Electrical Distribution

Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including: Breaker tripping

Proposed Question: #47

Given the following circuit breakers:

(1) 2NPS-SWG001-3, Supply to 2NPS-SWG001 from Normal Station Transformer
(2) 2NPS-SWG001-1, Supply to 2NPS-SWG001 from Reserve Station Transformer A
(3) 2NPS-SWG003-14, Supply to 2NPS-SWG003 from Normal Station Transformer
(4) 2NPS-SWG003-1, Supply to 2NPS-SWG003 from Reserve Station Transformer B
(5) 2NNS-SWG014-2, Supply to 2NNS-SWG014 from 2NPS-SWG001
(6) 2NNS-SWG015-3, Supply to 2NNS-SWG015 from 2NPS-SWG003

Which one of the following indicates which of these circuit breakers automatically OPEN following a Main Generator trip (Generator Lockout Trip) from rated conditions?

A. (1) and (3) only

B. (2) and (4) only

C. (1), (3), (5), and (6)

D. (2), (4), (5), and (6)

Proposed Answer: A

Explanation: On a main generator trip, a fast transfer occurs, transferring 2NPS-SWG001(003) from the Normal Station Transformer to the Reserve Station Transformers. Supply Breakers from the normal Station Transformer will automatically open.

B. Incorrect – 1-1, Supply to 2NPS-SWG001 from Reserve Station Transformer A, and 3-1, Supply to 2NPS-SWG003 from Reserve Station Transformer B, do not automatically open. Plausible because they do automatically close.
C. Incorrect – 13-6, Supply to 2NNS-SWG014 from 2NPS-SWG001, and 15-3, Supply to 2NNS-SWG015 from 2NPS-SWG003, do not automatically open. Plausible because these switchgears do momentarily de-energize.
D. Incorrect – 1-1, Supply to 2NPS-SWG001 from Reserve Station Transformer A, and 3-1, Supply to 2NPS-SWG003 from Reserve Station Transformer B, do not automatically open. Plausible because they do automatically close. 13-6, Supply to 2NNS-SWG014 from 2NPS-SWG001, and 15-3, Supply to 2NNS-SWG015 from 2NPS-SWG003, do not automatically open. Plausible because these switchgears do momentarily de-energize.

Technical Reference(s): N2-OP-68 Attachment 1-4, N2-ARP-852614, Drawing EE-MO1A

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-262001-RBO-5

Question Source: Bank - 2016 NRC #17

Question History: 2016 NRC #17

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 A4.04
	Importance Rating	3.6

AC Electrical Distribution

Ability to manually operate and/or monitor in the control room: Synchronizing and paralleling of different A.C. supplies

Proposed Question: #48

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

- The regular monthly start and load test is being performed for the Division I Diesel Generator (2EGS*EG1) per N2-OSP-EGS-M@001, Diesel Generator and Diesel Air Start Valve Operability Test – Division I, Section 8.2.

Which one of the following describes the governor mode used during this test and where the control switch to select this mode is located, in accordance with N2-OSP-EGS-M@001?

The 2EGS*EG1 governor is operated in…

A. parallel mode. The control switch used to select this mode is located at 2CEC*PNL852.

B. parallel mode. The control switch used to select this mode is located at 2CES*IPNL407.

C. isochronous mode. The control switch used to select this mode is located at 2CEC*PNL852.

D. isochronous mode. The control switch used to select this mode is located at 2CES*IPNL407

Proposed Answer: A

Explanation: During performance of N2-OSP-EGS-M@001, the EMERGENCY DSL GEN 1 PARALLEL switch is taken to ON. This places the governor in parallel mode prior to paralleling with the bus. This control is located at 2CEC*PNL852.

B. Incorrect – This control is located at 2CEC*PNL852. Plausible because other controls manipulated in N2-OSP-EGS-M@001 at located at 2CES*IPNL407.
C. Incorrect – The governor is operated in parallel mode. Plausible because the governor is normally in isochronous mode.
D. Incorrect – The governor is operated in parallel mode. Plausible because the governor is normally in isochronous mode. This control is located at 2CEC*PNL852. Plausible because other controls manipulated in N2-OSP-EGS-M@001 at located at 2CES*IPNL407.

Technical Reference(s): N2-OSP-EGS-M@001, N2-OP-100A

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 K6.02
	Importance Rating	2.8

Uninterruptable Power Supply (AC/DC)

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

Proposed Question: #49

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

- The breaker on 2VBB-PNL301 which supplies the normal AC power to 2VBB-UPS1B is open and tagged due to work on 2VBB-PNL301.
- 2VBB-UPS1B has been placed on battery power until the work is complete.

Then, DC Bus 2BYS-SWG001C de-energizes due to a sustained electrical fault.

Which one of the following describes the resulting status of Uninterruptible Power Supply (UPS) 2VBB-UPS1B loads?

A. De-energized, but can be manually aligned to another source.

B. De-energized and CANNOT be manually aligned to another source.

C. Energized because 2VBB-UPS1B inverter output is unaffected by this fault.

D. Energized because maintenance power automatically aligned through the static switch.

Proposed Answer: D

Explanation: 2BYS-SWG001C supplies the DC inverter input to 2VBB-UPS1B. With normal AC and DC power lost, the static switch and associated CB 3/4 breakers will automatically align maintenance power to UPS loads. Loads will remain energized.

A. Incorrect – Loads will remain energized. Plausible that the maintenance power supply would need to be manually aligned.
B. Incorrect – Loads will remain energized. Plausible that the maintenance power supply would also be de-energized by the given issues.
C. Incorrect – Inverter output is lost. Plausible because this would be correct for a different DC power loss or if another power source automatically supplied the inverter.

Technical Reference(s): N2-OP-71D, N2-OP-71D-LINEUPS

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-262002-RBO-3

Question Source: Bank - 2012 Cert #14

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 A2.01
	Importance Rating	2.8

DC Electrical Distribution

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

Proposed Question: #50

The plant is operating at rated conditions with the following:

- Annunciator 852108, DIV 1 EMER BUS BYS 002A 125 VDC SYSTEM TROUBLE alarms
- Computer point BYSIC05, 125VDC DIV 1 BAT 2A GROUND, caused the alarm
- The operator taking grounds on the bus places the selector switch to POS then to NEG with the following results:

    - POS 85 Volts
    - NEG 125 Volts

Which ground indication(s), if any, require(s) action in accordance with N2-OP-74A?

A. Positive bus only

B. Negative bus only

C. Both positive and negative busses

D. NEITHER positive NOR negative busses

Proposed Answer: C

Explanation: Annunciator 852108 alarms due to a battery ground at +/- 75 Volts. The associated ARP directs the operator to N2-OP-74A, section H.14. Per section H.14, a value of 75VDC or greater for either positive or negative ground readings indicates a ground exists and action is required.

A. Incorrect – The negative ground indication also requires action. Plausible because the ground indication is equivalent to the nominal bus voltage.
B. Incorrect – The positive ground indication also requires action. Plausible because the positive ground indication is much less than the negative ground indication
D. Incorrect – Both ground indications require action. Plausible if the candidate believed action was only required less than 75 V (farther from nominal bus voltage).

Technical Reference(s): ARP 852108, N2-OP-74A

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-263000-RBO-10

Question Source: Modified Bank – 2016 NRC #2

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	263000 K1.04
	Importance Rating	2.6

DC Electrical Distribution

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

Proposed Question: #2

The plant is operating at rated conditions with the following:

- Annunciator 852108, DIV 1 EMER BUS BYS 002A 125 VDC SYSTEM TROUBLE alarms
- Computer point BYSIC05, 125VDC DIV 1 BAT 2A GROUND, caused the alarm
- The operator taking grounds on the bus places the selector switch to POS then to NEG with the following results:
  - POS 85 Volts
  - NEG 25 Volts

Which ground indication(s), if any, require(s) action in accordance with N2-OP-74A?

A.	Positive Bus Only
B.	Negative Bus Only
C.	Both Positive and Negative Busses
D.	Neither Positive or Negative Busses

DC Electrical Distribution

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

Proposed Question: #2

The plant is operating at rated conditions with the following:

- Annunciator 852108, DIV 1 EMER BUS BYS 002A 125 VDC SYSTEM TROUBLE alarms
- Computer point BYSIC05, 125VDC DIV 1 BAT 2A GROUND, caused the alarm
- The operator taking grounds on the bus places the selector switch to POS then to NEG with the following results:
  - POS 85 Volts
  - NEG 25 Volts

Which ground indication(s), if any, require(s) action in accordance with N2-OP-74A?

A.	Positive Bus Only
B.	Negative Bus Only
C.	Both Positive and Negative Busses
D.	Neither Positive or Negative Busses

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 A3.02
	Importance Rating	3.1

Emergency Generators (Diesel/Jet)

Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Minimum time for load pick up

Proposed Question: #51

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

- The Division I EDG (EGS*EG1) is paralleled with 2ENS*SWG101 and loaded to 4400 KW.
- A Reactor scram occurs due to high Drywell pressure.

Then, two (2) minutes later, a loss of all offsite power occurs.

Which one of the following describes the response of the EGS*EG1 output breaker?

The EGS*EG1 output breaker…

A. remains closed throughout the transient.

B. trips open when the LOCA signal is received. It then recloses 10 seconds later.

C. trips open when the LOCA signal is received. It recloses 10 seconds after the loss of offsite power occurs.

D. trips open when the LOCA signal is received. It recloses immediately after the loss of offsite power occurs.

Proposed Answer: D

Explanation: The EDG output breaker receives a trip signal when the LOCA signal occurs. The EDG would then run unloaded. The EDG output breaker automatically recloses without a time delay when the loss of offsite power occurs.

A. Incorrect – The output breaker trips when the LOCA signal occurs. Plausible because the output breaker is initially closed and the LOCA signal also gives an EDG start signal.
B. Incorrect – The output breaker will not reclose given just the LOCA signal. Plausible because the output breaker is initially closed and the LOCA signal also gives an EDG start signal.
C. Incorrect – There is no 10 second time delay for reclosure of the output breaker in this situation. Plausible because normal EDG start and loading would result in the EDG closing in on the bus in approximately 10 seconds.

Technical Reference(s): N2-264001-RBO-5, ESK-5ENS021, ESK-5EGP001

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-264001-RBO-5

Question Source: Bank - 2012 Cert #15

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

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

Instrument Air

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

Proposed Question: #52

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

- Instrument Air compressor 2IAS-C3C is running and selected as the lead compressor.
- Instrument Air pressure is 120 psig.

Then, a large leak occurs in the Instrument Air system.

Which one of the following describes the Instrument Air pressures at which Instrument Air Compressors 2IAS-C3A and 2IAS-C3B will start?

	2IAS-C3A	2IAS-C3B
A.	100 psig	90 psig
B.	100 psig	85 psig
C.	90 psig	100 psig
D.	85 psig	100 psig

Proposed Answer: B

Explanation: With 2IAS-C3C running as the lead compressor, the compressor selector switch must be in the “CAB” position. This makes 2IAS-C3A the lag compressor, which will automatically start at 100 psig. This also makes 2IAS-C3B the backup compressor, which will automatically start at 85 psig.

A. Incorrect – 2IAS-C3B will start at 85 psig. Plausible because 90 psig is an alarm value in N2-SOP-19.
C. Incorrect – 2IAS-C3A will start at 100 psig. Plausible because if 2IAS-C3A was the backup compressor. Also plausible because 90 psig is an alarm value in N2-SOP-19. 2IAS-C3B will start at 85 psig. Plausible because this would be correct if 2IAS-C3B was the lag compressor.
D. Incorrect – 2IAS-C3A will start at 100 psig. Plausible because this would be correct if 2IAS-C3A was the backup compressor. 2IAS-C3B will start at 85 psig. Plausible because this would be correct if 2IAS-C3B was the lag compressor.

Technical Reference(s): N2-SOP-19, N2-OP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-278001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 2.4.9
	Importance Rating	3.8

Component Cooling Water

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

Proposed Question: #53

The plant is shutdown for a refueling outage and the Reactor is in Mode 5.

Which one of the following describes the minimum allowable running Service Water pump combination(s), in accordance with N2-OP-11, Service Water System?

A. 2-2

B. 1-2 or 2-1

C. 1-1

D. 0-1 or 1-0

Proposed Answer: D

Explanation: N2-OP-11 P&L 12.2 allows the following running Service Water pump configurations while in Mode 4 or 5:

- Pump configurations:
  - 0-1, 0-2, 0-3
  - 1-0, 1-1, 1-2, 1-3
  - 2-0, 2-1, 2-2, 2-3
  - 3-0, 3-1, 3-2, 3-3

Of these configurations, the minimum running Service Water pump combinations are 0-1 and 1-0.

Note: The question meets the K/A by testing a Component Cooling Water (Service Water) system requirement for shutdown conditions (Mode 5) that is directly tied to supporting LOCA response / mitigation strategy.

A. Incorrect – The minimum allowable running Service Water pump combinations are 0-1 and 1-0. Plausible because this is an allowable configuration and it is the minimum allowable in Modes 1, 2, or 3.
B. Incorrect – The minimum allowable running Service Water pump combinations are 0-1 and 1-0. Plausible because these are allowable configurations.
C. Incorrect – The minimum allowable running Service Water pump combinations are 0-1 and 1-0. Plausible because this is an allowable configuration.

Technical Reference(s): N2-OP-11 P&L 12.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-276000-RBO-9

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

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

CRD Hydraulic

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

Proposed Question: #54

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

A. 120 VAC, energize to cause a scram

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

C. 125 VDC, energize to cause a scram

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

Proposed Answer: C

Explanation: The backup scram valves are powered from 2BYS*PNL201A(B), which is a 125 VDC source. The valve solenoids are normally de-energized and energize to function.

A. Incorrect – The power supply is 125 VDC. Plausible because the normal scram solenoids are powered by 120 VAC.
B. Incorrect – The power supply is 125 VDC. Plausible because the normal scram solenoids are powered by 120 VAC. The valve solenoids are energized to function. Plausible because the normal scram solenoids are de-energized to function.
D. Incorrect – The valve solenoids are energized to function. Plausible because the normal scram solenoids are de-energized to function.

Technical Reference(s): N2-OP-97, N2-OP-97-LINEUPS

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-212000-RBO-4

Question Source: Bank - JAF 14-1 NRC #28

Question History: JAF 14-1 NRC #28

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	201003 K3.03
	Importance Rating	3.2

Control Rod and Drive Mechanism

Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on following: Shutdown margin

Proposed Question: #55

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

- A control rod pattern adjustment is in progress.
- Control rod 22-23 is given a signal to notch insert from position 24 to 22.
- When control rod 22-23 attempts to settle, the collet fingers fail to re-engage with the index tube.

Which one of the following describes the effect of the collet finger failures on shutdown margin if a Reactor scram occurs?

Given a Reactor scram prior to reset, shutdown margin…

A. is lowered, but will still meet the requirements of Technical Specifications.

B. is lowered and will NOT meet the requirements of Technical Specifications.

C. will be unaffected because control rod 22-23 fully inserts when the collet fingers fail and will remain fully inserted on a scram.

D. will be unaffected because, even though control rod 22-23 fully withdraws when the collet fingers fail, it will still fully insert on a scram.

Proposed Answer: D

Explanation: During the settle portion of the control rod notch insertion sequence, the control rod should move slightly downward until the collet fingers engage the index tube and hold the control rod in place. Since the collet fingers fail to engage the index tube, the control rod drifts down and out of the core due to the pull of gravity. This will add positive reactivity to the core, however a Reactor scram will apply hydraulic pressure to the CRDM and successfully insert the control rod fully into the core. This will result in the same post-scram shutdown margin as if the collet finger failure had never occurred.

A. Incorrect – Shutdown margin will be unaffected. Plausible because the control rod does drift fully out prior to the scram due to the collet finger malfunction.
B. Incorrect – Shutdown margin will be unaffected. Plausible because the control rod does drift fully out prior to the scram due to the collet finger malfunction.
C. Incorrect – The control rod drifts out of the core, not in, due to the collet finger malfunction. Plausible if the operation of the collet fingers is confused.

Technical Reference(s): N2-201001-RBO-2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-201001-RBO-2

Question Source: Bank - SSES LOC27 NRC #29

Question History: SSES LOC27 NRC #29

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202002 K4.07
	Importance Rating	2.9

Recirculation Flow Control

Knowledge of RECIRCULATION FLOW CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Minimum and maximum pump speed setpoints

Proposed Question: #56

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

- Both Reactor Recirc pumps are operating in fast speed.
- Then, a Feedwater level control malfunction occurs.
- Total Feedwater flow lowers to 6.3 Mlbm/hr.
- The Reactor is manually scrammed.
- Reactor water level reaches a low of 130” during the transient before being restored to the normal band.

Which one of the following describes the resulting status of the Reactor Recirc system?

Reactor Recirc pumps…

A. remain operating in fast speed with Flow Control Valve position unchanged.

B. remain operating in fast speed, but with Flow Control Valves runback.

C. are operating in slow speed.

D. are tripped.

Proposed Answer: C

Explanation: Reactor water level <159.3” causes the Recirc pumps to downshift from fast to slow speed. No other condition exists to trip the Recirc pumps.

A. Incorrect – The Recirc pumps are operating in slow speed. Plausible because the initial malfunction keeps total Feedwater flow above 3.44 Mlbm/hr, which is another setpoint that downshifts Recirc pumps from fast to slow speed.
B. Incorrect – The Recirc pumps are operating in slow speed. Plausible because the initial malfunction keeps total Feedwater flow above 3.44 Mlbm/hr, which is another setpoint that downshifts Recirc pumps from fast to slow speed. Also plausible because there are other conditions that result in a FCV runback.
D. Incorrect – No conditions are given that result in trip of the Reactor Recirc pumps. Plausible because there is a low Reactor water level trip at 108.8”.

Technical Reference(s): N2-OP-29

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-202002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	204000 K5.04
	Importance Rating	2.7

Reactor Water Cleanup

Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER CLEANUP SYSTEM: Heat exchanger operation

Proposed Question: #57

A plant startup is in progress with the following:

- Reactor Water Cleanup (WCS) is being used to reject water to the Main Condenser.
- 2WCS-FV135, REJECT FLOW CONTROL MANUAL CONTROL, is opened further.

Which one of the following identifies the change in WCS flow through the two sides of the Regenerative Heat Exchanger (RHX) when this valve is opened further?

	WCS Flow From Reactor Through RHX (Tube Side)	WCS Flow From Filter Demins Through RHX (Shell Side)
A.	Rises	Rises
B.	Rises	Lowers
C.	Lowers	Rises
D.	Lowers	Lowers

Proposed Answer: B

Explanation: When 2WCS-FV135 is opened, flow from the Reactor to WCS rises, including flow through the tube side of the RHX. WCS flow branches downstream of the filter demins. One branch goes to the reject line and the other branch is the return flow to the RHX shell side. Therefore, as reject flow goes up, RHX shell side flow goes down.

A. Incorrect – RHX return (shell side) flow goes down. Plausible because this would be correct if reject flow tapped off downstream of the RHX shell side.
C. Incorrect – RHX tube side flow goes up. Plausible because this would be correct if reject flow tapped off upstream of the RHX tube side. RHX return (shell side) flow goes down. Plausible because this would be correct if reject flow tapped off downstream of the RHX shell side.
D. Incorrect – RHX tube side flow goes up. Plausible because this would be correct if reject flow tapped off upstream of the RHX tube side.

Technical Reference(s): N2-OP-37, P&ID-37

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-204000-RBO-3

Question Source: Bank – 2010 NRC #31

Question History: 2010 NRC #31

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	214000 K6.02
	Importance Rating	2.7

Rod Position Information

Knowledge of the effect that a loss or malfunction of the following will have on the ROD POSITION INFORMATION SYSTEM: Position indication probe

Proposed Question: #58

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

- The Position Indication Probe (PIP) **connector** for control rod 30-31 has disconnected.

Given the following control rod 30-31 indications:

(1) Control rod position
(2) Control rod temperature
(3) Control rod HCU pressure

Which one of the following identifies which of these indications will be affected by this connector being disconnected?

A. (1) only

B. (1) and (2) only

C. (1) and (3) only

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

Proposed Answer: B

Explanation: The Position Indication Probe (PIP) connection passes signals related to both control rod position and temperature. A thermocouple is installed in the position indicator tube at the top to monitor drive temperature. Therefore, both position and temperature indications are affected.

A. Incorrect – Temperature indication is also affected. Plausible that the Position Indication Probe (PIP) connection would only be related to control rod position.
C. Incorrect – HCU pressure indication is not affected. Plausible that this signal would also be routed through this connector since temperature is. Temperature indication is also affected. Plausible that temperature indication would be completely separate from position indication.
D. Incorrect – HCU pressure indication is not affected. Plausible that this signal would also be routed through this connector since temperature is.

Technical Reference(s): N2-OP-96

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-201002-RBO-3

Question Source: Bank – 2016 NRC #32

Question History: 2016 NRC #32

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	215002 A1.01
	Importance Rating	2.7

Rod Block Monitor

Ability to predict and/or monitor changes in parameters associated with operating the ROD BLOCK MONITOR SYSTEM controls including: Trip reference: BWR-3,4,5

Proposed Question: #59

A plant startup is in progress.

Which one of the following describes how the Rod Block Monitor setpoint range (low, intermediate, high) is changed as the startup progresses?

The setpoint range is changed…

A. automatically based on an APRM signal.

B. automatically based on a Recirc flow signal.

C. automatically based on a Core Thermal Power signal.

D. manually using a pushbutton on Control Room Panel 603.

Proposed Answer: A

Explanation: An APRM provides Simulated Thermal Power to the RBM, which uses that power level to determine which RBM range setpoint is enabled.

B. Incorrect – An APRM signal is used. Plausible because Recirc flow is used to determine APRM trip setpoints.
C. Incorrect – An APRM signal is used. Plausible because this signal is described as Simulated Thermal Power. Also plausible because using Core Thermal Power calculation would provide the same functionality as using an APRM signal.
D. Incorrect – An APRM signal is used to automatically change the range. Plausible because this would be similar to how IRM ranges are manually selected as a startup progresses.

Technical Reference(s): N2-OP-92

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-215003-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	219000 A2.04
	Importance Rating	3.1

RHR/LPCI: Torus/Suppression Pool Cooling Mode

Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve openings

Proposed Question: #60

The plant is operating at 80% power with the following:

- Suppression Pool temperature is 92^{o}F and slowly rising due to a leaking SRV.
- RHR A has been placed in the Suppression Pool Cooling mode and normal flows have been established.
- RHR A and associated Service Water valves are aligned as follows:
  - 2RHS*MOV4A, PMP 1A MINIMUM FLOW VLV, is open
  - 2RHS*FV38A, RETURN TO SUPPR POOL COOLING, is throttled
  - 2SWP*MOV33A, HEAT EXCHANGER 1A SVCE WTR OUTLET VLV, is throttled

Which one of the following describes the status of these valves and, if necessary, the appropriate operator action per N2-OP-31, Residual Heat Removal System?

A. All valves are positioned as required.

B. 2RHS*MOV4A, PMP 1A MINIMUM FLOW VLV, should be closed. Manually close 2RHS*MOV4A.

C. 2RHS*FV38A, RETURN TO SUPPR POOL COOLING, should be fully open. Fully open 2RHS*FV38A.

D. 2SWP*MOV33A, HEAT EXCHANGER 1A SVCE WTR OUTLET VLV, should be fully open. Fully open 2SWP*MOV33A.

Proposed Answer: B

Explanation: 2RHS*MOV4A is open but should be closed since RHR flow has been established. Operator action is required to close 2RHS*MOV4A. 2RHS*FV38A and 2SWP*MOV33A are throttled appropriately. Throttling these valves is required to not exceed flow limitations in N2-OP-31.

A. Incorrect – 2RHS*MOV4A is open but should be closed since RHR flow has been established. Operator action is required to close 2RHS*MOV4A. Plausible because this valve is required to initially be open while establishing the lineup.
C. Incorrect – 2RHS*FV38A is throttled to establish correct flows and is not required to be fully open. Plausible because other valves in the lineup are required to be fully open.
D. Incorrect – 2SWP*MOV33A is throttled to establish correct flows and is not required to be fully open. Plausible because other valves in the lineup are required to be fully open.

Technical Reference(s): N2-OP-31

Proposed references to be provided to applicants during examination: Embedded picture

Learning Objective: N2-205000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	226001 A3.01
	Importance Rating	3.0

RHR/LPCI: Containment Spray Mode

Ability to monitor automatic operations of the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE including: Valve operation

Proposed Question: #61

The plant has experienced a steam leak in the Drywell with the following:

- Drywell spray is being initiated.
- 2RHS*MOV25B, Outlet to Drywell Spray, is open.
- 2RHS*MOV15B, Outlet to Drywell Spray, is stroking open.
- Before 2RHS*MOV15B is full open, Drywell pressure lowers to 1.2 psig.

Which one of the following describes the response of 2RHS*MOV15B?

A. Immediately reverses direction and closes.

B. Strokes full open and remains open.

C. Strokes full open and then closes.

D. Stops in an intermediate position.

Proposed Answer: D

Explanation: Relay K60A(B) appears in both Drywell spray valve circuits. If the high Drywell pressure signal clears (<1.68 psig), relay K60B will de-energize and the contacts in the open leg of the valve circuits open. This stops valve motion in an intermediate position.

A. Incorrect – The valve stops in an intermediate position. Plausible that a contact in the close circuitry would operate off of the K60B relay to close the valve.
B. Incorrect – The valve stops in an intermediate position. Plausible that the open signal would seal in until the valve was full open.
C. Incorrect – The valve stops in an intermediate position. Plausible that the open signal would seal in until the valve was full open and then a contact in the close circuitry would operate off of the K60B relay to close the valve.

Technical Reference(s): ASK 6RHS08, ASK 6RHS13, GE 807E170TY sheet 8

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-5

Question Source: Bank - Vision SYSID 34975

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	230000 A4.07
	Importance Rating	3.6

RHR/LPCI: Torus/Suppression Pool Spray Mode

Ability to manually operate and/or monitor in the control room: System flow

Proposed Question: #62

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

- Drywell pressure is 6 psig and stable.
- RHR A has been placed in Suppression Pool Spray mode.
- The Unit Supervisor has also directed utilizing RHR A for Suppression Pool Cooling.
- RHR A total flow is 4400 gpm.
- RHR A Suppression Chamber Spray header flow is 475 gpm.
- Suppression Pool temperature is 92^{o}F and rising slowly.

Which one of the following describes the status of the given flows, in accordance with N2-EOP-6.22, Containment Sprays?

A. Both of the given flows are acceptable.

B. Total flow is acceptable. Suppression Chamber Spray header flow is too low.

C. Total flow is too low. Suppression Chamber Spray header flow is acceptable.

D. Total flow is too low and Suppression Chamber Spray header flow is too low.

Proposed Answer: C

Explanation: N2-EOP-6.22 directs establishing a total flow of approximately 7450 gpm. The given flow of 4400 gpm is well below this desired value. N2-EOP-6.22 directs verifying Suppression Chamber Spray header flow is greater than or equal to 450 gpm. Therefore, the given flow of 475 gpm is acceptable.

A. Incorrect – N2-OP-31 directs establishing a total flow of approximately 7450 gpm. The given flow of 4400 gpm is well below this desired value. Plausible because a significant amount of flow has been established and Suppression Chamber Spray header flow is acceptable.
B. Incorrect – N2-OP-31 directs establishing a total flow of approximately 7450 gpm. The given flow of 4400 gpm is well below this desired value. Plausible because a significant amount of flow has been established. The given Suppression Chamber Spray header flow is acceptable. Plausible because this is well below the required Drywell Spray header flow.
D. Incorrect – The given Suppression Chamber Spray header flow is acceptable. Plausible because this is well below the required Drywell Spray header flow.

Technical Reference(s): N2-EOP-6.22

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-205000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	245000 2.1.27
	Importance Rating	3.9

Main Turbine Generator/Auxiliary

Knowledge of system purpose and/or function.

Proposed Question: #63

The plant is operating at 100% power when Stator Water Cooling system performance degrades.

Which one of the following would cause an automatic runback of the Main Generator?

Stator water…

A. inlet flow at 745 gpm

B. inlet pressure at 75 psig

C. return temperature at 87^{o}C

D. conductivity at 10 µmho/cm

Proposed Answer: C

Explanation: Stator water cooling return temperature >79^{o}C causes a Main Generator runback.

A. Incorrect – Stator water inlet flow at 745 gpm does not cause a Main Generator runback. Plausible because stator water inlet flow <634 gpm would cause a Main Generator runback.
B. Incorrect – Stator water inlet pressure at 75 psig does not cause a Main Generator runback. Plausible because stator water inlet pressure <64 psig would cause a Main Generator runback.
D. Incorrect – Stator water conductivity at 10 µmho/cm does not cause a Main Generator runback. Plausible because this is an alarm condition that would require removing the Main Generator from service.

Technical Reference(s): N2-253000-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-253000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	259001 K1.05
	Importance Rating	3.2

Feedwater

Knowledge of the physical connections and/or cause effect relationships between REACTOR FEEDWATER SYSTEM and the following: Condensate system

Proposed Question: #64

A plant startup is in progress with the following:

- Reactor power is approximately 1%.
- Reactor pressure is 245 psig and controlling with the bypass valve opening jack.
- Condensate and Feedwater pumps are operating ***as shown on the following page***.

Note: All of the illuminated light bulbs on the following page are labeled “Lit”. The extinguished light bulbs have NO corresponding label.

Then, 2NNS-SWG013 de-energizes due to a sustained electrical fault.

Which one of the following describes the impact of this malfunction on the Condensate and Feedwater systems?

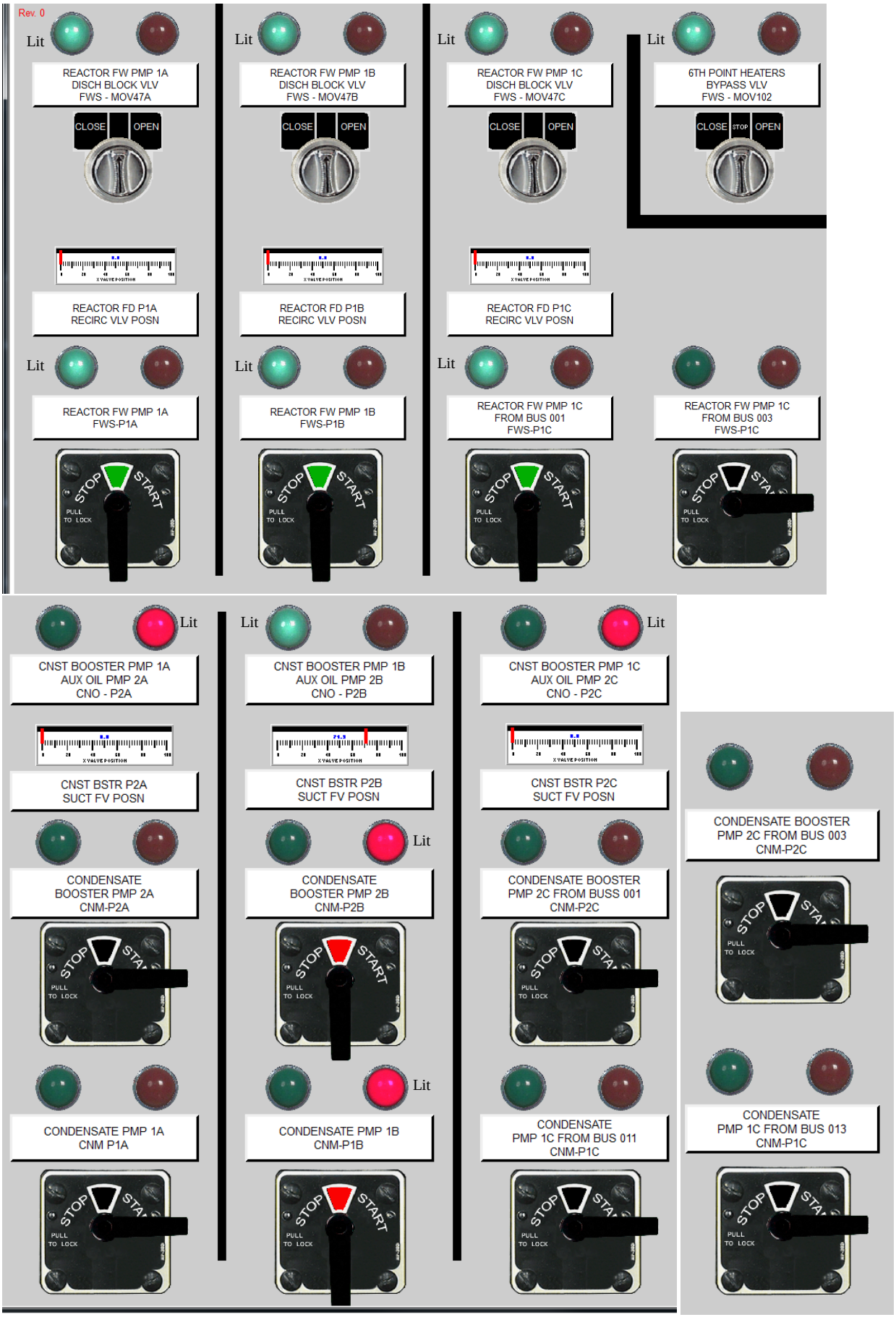
Feedwater flow to the Reactor is…

A. lost.

B. maintained with NO pump trips.

C. maintained due to auto-start of one (1) pump, only.

D. maintained due to auto-start of two (2) pumps.


Rev. 0
Lit
REACTOR FW PMP 1A
DISCH BLOCK VLV
FWS - MOV47A
CLOSE
OPEN
REACTOR FD P1A
RECIRC VLV POSN
REACTOR FW PMP 1A
FWS-P1A
STOP
START
PULL
TO LOCK
REACTOR FW PMP 1B
DISCH BLOCK VLV
FWS - MOV47B
REACTOR FD P1B
RECIRC VLV POSN
REACTOR FW PMP 1B
FWS-P1B
REACTOR FW PMP 1C
DISCH BLOCK VLV
FWS - MOV47C
REACTOR FD P1C
RECIRC VLV POSN
REACTOR FW PMP 1C
FROM BUS 001
FWS-P1C
6TH POINT HEATERS
BYPASS VLV
FWS - MOV102
STOP
REACTOR FW PMP 1C
FROM BUS 003
FWS-P1C
CNST BOOSTER PMP 1A
AUX OIL PMP 2A
CNO - P2A
CNST BSTR P2A
SUCT FV POSN
CONDENSATE
BOOSTER PMP 2A
CNM-P2A
CONDENSATE PMP 1A
CNM P1A
CNST BOOSTER PMP 1B
AUX OIL PMP 2B
CNO - P2B
CNST BSTR P2B
SUCT FV POSN
CONDENSATE
BOOSTER PMP 2B
CNM-P2B
CONDENSATE PMP 1B
CNM-P1B
CNST BOOSTER PMP 1C
AUX OIL PMP 2C
CNO - P2C
CNST BSTR P2C
SUCT FV POSN
CONDENSATE BOOSTER
PMP 2C FROM BUSS 001
CNM-P2C
CONDENSATE
PMP 1C FROM BUS 011
CNM-P1C
CONDENSATE BOOSTER
PMP 2C FROM BUS 003
CNM-P2C
CONDENSATE
PMP 1C FROM BUS 013
CNM-P1C

Proposed Answer: A

Explanation: Loss of 2NNS-SWG013 causes Condensate pump B to trip. The given alignment has Condensate pump A and C control switches in pull-to-lock, therefore they do not auto-start. This results in a loss of all Condensate flow, which causes a loss of Feedwater flow to the Reactor.

B. Plausible – Loss of 2NNS-SWG013 causes Condensate pump B to trip. Plausible because in a different lineup, this loss could result in no pump trips.
C. Plausible – The given alignment has Condensate pump A and C control switches in pull-to-lock, therefore they do not auto-start. Plausible because in a different lineup, one or both of these Condensate pumps could auto-start.
D. Plausible – The given alignment has Condensate pump A and C control switches in pull-to-lock, therefore they do not auto-start. Plausible because in a different lineup, one or both of these Condensate pumps could auto-start.

Technical Reference(s): N2-OP-3, N2-OP-3-LINEUPS

Proposed references to be provided to applicants during examination: Embedded picture

Learning Objective: N2-256000-RBO-4

Question Source: Bank – 2017 Cert #32

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	272000 K2.03
	Importance Rating	2.5

Radiation Monitoring

Knowledge of electrical power supplies to the following: Stack gas radiation monitoring system

Proposed Question: #65

Which one of the following identifies the power supply to the Stack Wide Range Gaseous Monitoring System (WRGM) sample cabinets?

A. 2VBB-UPS1H

B. 2VBA*UPS2D

C. 2BYS-SWG001A

D. 2BYS*SWG002A

Proposed Answer: A

Explanation: There are two main power supplies for the Stack Wide Range Gaseous Monitoring System (WRGM). Rad monitor control and main control room indication power comes from 2VBS-PNLA102 which is powered from 2VBB-UPS1A. The rad monitor sample cabinet power (2RMS-SKD170A & B) comes from 2VBB-UPS1H.

B. Incorrect – 2VBA*UPS2D does not supply the Stack Wide Range Gaseous Monitoring System (WRGM). Plausible because this bus supplies power to multiple other radiation monitoring systems.
C. Incorrect – 2BYS-SWG001A does not supply the Stack Wide Range Gaseous Monitoring System (WRGM). Plausible because this bus supplies power to many other instruments.
D. Incorrect – 2BYS*SWG002A does not supply the Stack Wide Range Gaseous Monitoring System (WRGM). Plausible because this bus supplies power to many other instruments.

Technical Reference(s): Loop Diagram 2RMS-l 70, EE-011DB

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-272000-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:

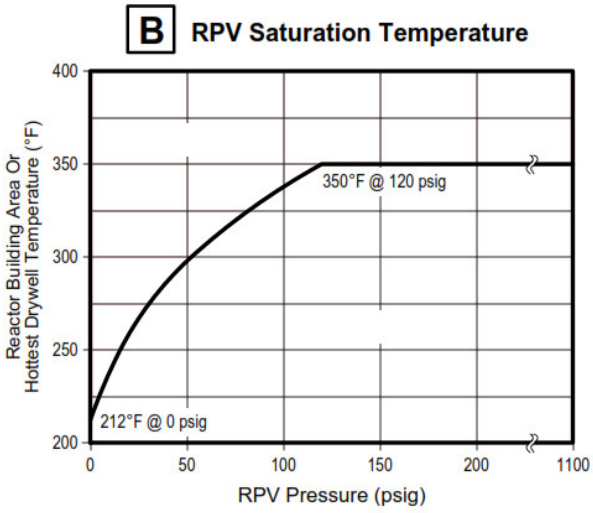
Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.45
	Importance Rating	4.3

Ability to identify and interpret diverse indications to validate the response of another indication.

Proposed Question: #66

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

- The hottest Reactor Building temperature is 110°F and slowly rising.
- The hottest Drywell temperature is 310°F and slowly rising.
- Reactor pressure is 50 psig and stable.
- There is no indication of instrument leg flashing for any Reactor water level instrument.
- The SRO has directed Reactor water level be controlled in the normal level control band given in N2-EOP-RPV, RPV Control.



Which one of the following identifies the Reactor water level instrument(s) (if any) that may be used to trend Reactor water level across the entire normal level band, in accordance with N2-EOP-RPV?

A. Wide Range, only

B. Wide Range and Narrow Range, only

C. Wide Range, Narrow Range, and Fuel Zone, only

D. No Reactor water level indications are valid across this entire level band

Proposed Answer: B

Explanation: The given Drywell / Reactor Building temperatures and Reactor pressure place operation in the BAD region of the RPV Saturation Temperature curve. Even though operation is in the BAD region of the RPV Saturation Temperature curve, instruments may still be used as long as indications of flashing are not present and as long as they are above the minimum indicated level. The normal Reactor water level band per N2-EOP-RPV is 159.3 inches to 202.3 inches (160-200 inches). Both Wide Range and Narrow Range instruments are capable of trending over the entire 160 to 200 inch band, however, Fuel Zone would be off scale high for this band. Upset and Shutdown Range cannot be used because 160 inches is below their minimum indicated levels per N2-EOP-RPV.

A. Incorrect – Narrow Range can also be used across the entire normal level control band. Plausible because Narrow Range would not indicate across an expanded band, such as in Alternate Level Control leg or in N2-EOP-C5.
C. Incorrect – Fuel Zone cannot be used across the entire normal level control band. Plausible because Fuel Zone instruments are available for use if level lowers.
D. Incorrect – Wide Range and Narrow Range may be used. Plausible because the given Drywell / Reactor Building temperatures and Reactor pressure place operation in the BAD region of the RPV Saturation Temperature curve.

Technical Reference(s): N2-EOP-RPV

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRPVC01 EO-2

Question Source: Bank - 2015 NRC #67

Question History: 2015 NRC #67

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.4
	Importance Rating	3.3

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: #67

The plant is operating at 100% power.

Which one of the following identifies:

(1) the ***minimum*** number of licensed Unit 2 Reactor Operators that must be on shift

and

(2) whether one of these minimum required Reactor Operators may leave the Unit 2 Control Room without relief

in accordance with OP-NM-103-102, Watch-Standing Practices at Nine Mile Point?

	(1) Minimum Number of Licensed Unit 2 Reactor Operators	(2) May One of the Minimum Required Reactor Operators Leave the Unit 2 Control Room Without Relief?
A.	2	Yes
B.	2	No
C.	3	Yes
D.	3	No

Proposed Answer: A

Explanation: The minimum number of licensed Unit 2 Reactor Operators that must be on shift with the plant at 100% power is 2. One of these ROs must remain within the Control Room ATC area. The other RO must normally remain within the Protected Area, but is allowed to leave the Unit 2 Control Room.

Note: The question meets the K/A by testing a shift staffing requirement (minimum number of ROs) and an individual operator responsibility related to this shift staffing requirement (whether one of these ROs may leave the Control Room without relief).

B. Incorrect – One of the required ROs may leave the Unit 2 Control Room. Plausible because these ROs are restricted on where they may go without relief, and the Unit 2 Control Room does require at least one of these ROs.
C. Incorrect – The minimum number of ROs is 2. Plausible because the minimum number is 3 under other situations.
D. Incorrect – The minimum number of ROs is 2. Plausible because the minimum number is 3 under other situations. One of the required ROs may leave the Unit 2 Control Room. Plausible because these ROs are restricted on where they may go without relief, and the Unit 2 Control Room does require at least one of these ROs.

Technical Reference(s): OP-NM-103-102

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - NMP1 2018 NRC #66

Question History: NMP1 2018 NRC #66

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.1
	Importance Rating	3.8

Knowledge of conduct of operations requirements.

Proposed Question: #68

Given the following:

- You are a licensed Reactor Operator.
- You have been on vacation for the previous 14 days.
- Today was your first shift back on watch.
- You worked the normal day shift from 0600 to 1800.
- Then you were required to stay a total of four (4) hours past end of shift due to an on-coming Operator calling in sick.
- You were originally scheduled to cover day shift tomorrow from 0600-1800.

Which one of the following identifies the earliest time you can return to work tomorrow without receiving a waiver in accordance with LS-AA-119, Fatigue Management and Work Hour Limits?

A. 0600

B. 0700

C. 0800

D. 1400

Proposed Answer: C

Explanation: LS-AA-119 Section 3.4 requires “a 10-hour break between the previous work period, or an 8-hour break between the previous work period when a break of less than 10 hours was necessary to accommodate a crew's scheduled transition between work schedules or shifts”. Since this situation is not related to a crew’s scheduled shift transition, the individual needs a minimum 10 hour break before returning to work. The individual worked 4 hours past 1800, so they left work at 2200. Ten hours later is 0800.

A. Incorrect – 0600 is only 8 hours after the individual left work and would not provide the required 10 hour break. Plausible because this would be acceptable if the break were required to accommodate a crew’s scheduled transition between shifts.
B. Incorrect – 0700 is only 9 hours after the individual left work and would not provide the required 10 hour break. Plausible because this would be correct if the requirement were 9 hours.
D. Incorrect – 0800 satisfies the requirement, therefore 1400 is not the earliest allowed time. Plausible because this would be correct if the requirement were 16 hours (plausible based on 16 hours just worked or 16/24 rule.).

Technical Reference(s): LS-AA-119

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - NMP1 2017 NRC #67

Question History: NMP1 2017 NRC #67

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.38
	Importance Rating	3.6

Knowledge of conditions and limitations in the facility license.

Proposed Question: #69

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

- A reduction in core flow transient results in core flow dropping to 65% of rated.
- Both Recirc pumps are still operating in high speed.

Which one of the following identifies the Tech Spec Thermal Limit LCO limit that will be affected?

A. Preconditioning Ratio (PCRAT)

B. Linear Heat Generation Rate (LHGR)

C. Minimum Critical Power Ratio (MCPR)

D. Average Planar Linear Heat Generation Rate (APLHGR)

Proposed Answer: C

Explanation: The MCPR LCO operating limit is HIGHER following a flow reduction. Per the COLR and TS 3.2.2 Bases “MCPR operating limit is the greater of either flow dependent MCPR limit MCPRf or power dependent MCPR limit MCPRp”.

A. Incorrect – PCRAT is not affected by a reduction in core flow. Plausible because it is another value calculated along with thermal limits in 3D Monicore.
B. Incorrect – LHGR limit is not affected by a reduction in core flow. Plausible because it is affected by other conditions, such as Turbine Bypass Valves being in operable.
D. Incorrect – APLHGR limit is not affected by a reduction in core flow. Plausible because it is affected by other conditions, such as being in single loop operation.

Technical Reference(s): TS Bases 3.2.2, COLR Sect 2

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank - 2009 NRC #37

Question History: 2009 NRC #37

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	3
	Group #	
	K/A #	2.2.42
	Importance Rating	3.9

Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Proposed Question: #70

The plant is operating at 100% power with NO evolutions in progress.

Which one of the following conditions would require entry in the ACTIONS section of a Technical Specification?

A. Drywell average temperature of 135^{o}F

B. Suppression Pool water level of 200’ 9”

C. Drywell pressure of 0.65 psig (15.35 psia)

D. Suppression Pool average temperature of 92^{o}F

Proposed Answer: D

Explanation: With no testing that adds heat to the Suppression Pool in progress and Suppression Pool average temperature >90^{o}F, Technical Specification 3.6.2.1 Condition A must be entered.

A. Incorrect – This would not require entry into Technical Specification ACTIONS. Plausible because it would require entry if Drywell average temperature was >150^{o}F.
B. Incorrect – This would not require entry into Technical Specification ACTIONS. Plausible because it would require entry if Suppression Pool water level was >201'.
C. Incorrect – This would not require entry into Technical Specification ACTIONS. Plausible because it would require entry if Drywell pressure was >15.45 psia.

Technical Reference(s): Technical Specifications 3.6.1.4, 3.6.1.5, 3.6.2.1, 3.6.2.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-223001-RBO-14

Question Source: Modified Bank - 2012 Cert #74

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Technical Specifications.

Proposed Question:	RO Question # 74

Which one of the following conditions would require entry into the ACTIONS section of a Technical Specification?

A.		Mode 2 and drywell average temperature is at 135°F
B.		Mode 1 and suppression pool average temperature is at 86°F
C.		Mode 1 and suppression pool water level is at 201 feet 5 inches
D.		Mode 3 and suppression pool water level is at 199 feet 8 inches

Proposed Answer:	C

Explanation (Optional):

A.		Incorrect. TS 3.6.1.5 requires entry into the action section if drywell temperature is >150F
B.		Incorrect. TS 3.6.2.1 requires entry into the action section if suppression pool temperature is >90F
C.		Correct. TS 3.6.2.2 requires suppression pool water level to be ≤201 feet. At 201 feet 5 inches, the LCO is not met and would require actions to be taken in the ACTIONS section.
D.		Incorrect. TS 3.6.2.2 requires entry into the action section if water level is <199 feet 6 inches.

Technical Reference(s):	TS 3.6.2.2	(Attach if not previously provided)

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.14
	Importance Rating	3.4

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Proposed Question: #71

Which one of the following describes the restriction on access to the Drywell during core shuffle, in accordance with N2-FHP-13.3, Core Shuffle?

Access is prohibited to…

A. the entire Drywell due to the potential for high radiation levels.

B. the entire Drywell due to the potential for high heat stress conditions.

C. only the elevations above 288’ due to the potential for high radiation levels.

D. only the elevations above 288’ due to the potential for high heat stress conditions.

Proposed Answer: C

Explanation: N2-FHP-13.3 prohibits personnel access to the Drywell only above elevation 288' during core shuffle due to the potential for high radiation levels in those areas.

A. Incorrect – Not all of the Drywell is restricted access during core shuffle. Plausible because access is restricted to certain portions of the Drywell.
B. Incorrect – Not all of the Drywell is restricted access during core shuffle. Plausible because access is restricted to certain portions of the Drywell. The concern is high radiation levels. Plausible because heat stress is also a concern during outage work in the Drywell.
D. Incorrect – The concern is high radiation levels. Plausible because heat stress is also a concern during outage work in the Drywell.

Technical Reference(s): N2-FHP-13.3 Sections 4.5.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-234000-RBO-9

Question Source: Bank - 2014 Cert #71

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(12)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.15
	Importance Rating	2.9

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

Proposed Question: #72

The plant is operating at 100% power.

Which one of the following radiation monitors would directly require entry into N2-EOP-SC, Secondary Containment Control, given a valid upscale reading?

A. 2RMS-RE103, RHS HX EQUIP ROOM, DIV II

B. 2RMS-RE116, CONDENSATE PUMP AREA

C. 2RMS-RE136, OFG-PNL122 AREA

D. 2RMS-RE152, RW BLDG FLOOR DRAIN SUMP AREA

Proposed Answer: A

Explanation: N2-EOP-SC entry is required given an area radiation monitor above an alarm setpoint in the Secondary Containment. Of the given radiation monitors, only 2RMS-RE103, RHS HX EQUIP ROOM, DIV II, is located in the Secondary Containment.

B. Incorrect – 2RMS-RE116, CONDENSATE PUMP AREA, does not require entry into N2-EOP-SC. Plausible because this is an area radiation monitor in the Turbine Building.
C. Incorrect – 2RMS-RE136, OFG-PNL122 AREA, does not require entry into N2-EOP-SC. Plausible because this is an area radiation monitor in the Turbine Building.
D. Incorrect – 2RMS-RE152, RW BLDG FLOOR DRAIN SUMP AREA, does not require entry into N2-EOP-SC. Plausible because this is an area radiation monitor in the Radwaste Building.

Technical Reference(s): N2-EOP-SC, ARP 851244, ARP 851245, ARP 851247

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: Bank - JAF 17-2 NRC #72

Question History: JAF 17-2 NRC #72

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.11
	Importance Rating	4.0

Knowledge of abnormal condition procedures.

Proposed Question: #73

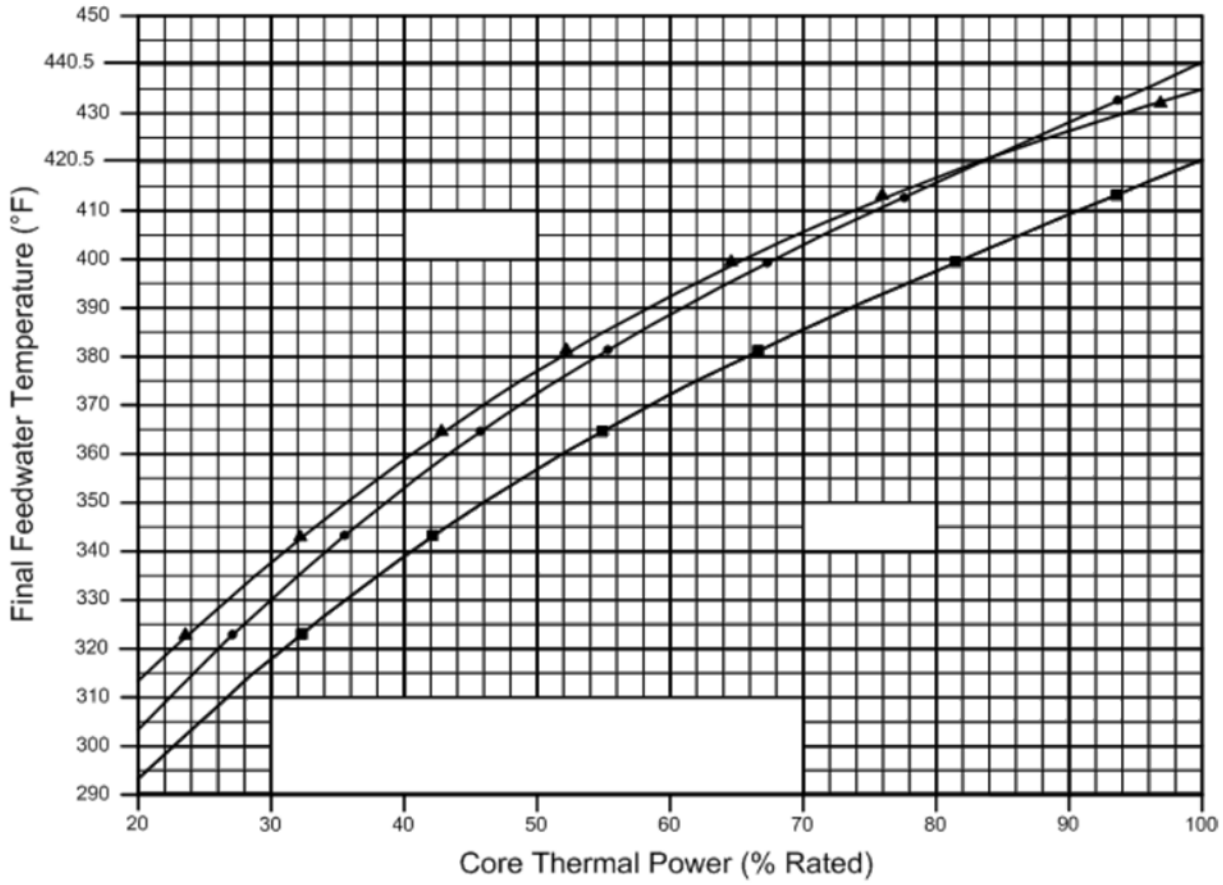
The plant is operating at 75% power with the following:

- 4th point Feedwater Heater A steam is removed per N2-OP-8, Feedwater Heaters and Extraction Steam System, due to a steam leak.
- Reactor power stabilizes at 78%.
- Feedwater temperature stabilizes at 390°F.
- N2-SOP-08, Unplanned Power Changes, has been entered due to the loss of Feedwater Heating.

Note: N2-SOP-08 Figure 1, Feedwater Temperature/Thermal Power Limit, is ***provided on the following page.***

Which one of the following describes the operating point on N2-SOP-08 Figure 1 and the required action for an unacceptable Feedwater temperature, in accordance with N2-SOP-08?

	The operating point on N2-SOP-08 Figure 1 is…	If the operating point is or becomes unacceptable, then it is required to perform a Reactor…
A.	acceptable.	power reduction per N2-SOP-101D, Rapid Power Reduction.
B.	acceptable.	scram per N2-SOP-101C, Reactor Scram.
C.	NOT acceptable.	power reduction per N2-SOP-101D, Rapid Power Reduction.
D.	NOT acceptable.	scram per N2-SOP-101C, Reactor Scram.


Final Feedwater Temperature (°F)
450
440.5
430
420.5
410
400
390
380
370
360
350
340
330
320
310
300
290
20
30
40
50
60
70
80
90
100
Core Thermal Power (% Rated)

Proposed Answer: C

Explanation: The given combination of Reactor power and Feedwater temperature is in the BAD region of the curve (unacceptable). If the operating point is unacceptable, then a Reactor power reduction is required per N2-SOP-101D, but NOT a Reactor scram.

A. Incorrect – Feedwater temperature is unacceptable. Plausible if the candidate mis-plots the operating point or does not understand which area of the curve is acceptable.
B. Incorrect – Feedwater temperature is unacceptable. Plausible if the candidate mis-plots the operating point or does not understand which area of the curve is acceptable. A Reactor power reduction would be required, NOT a Reactor scram. Plausible that a scram would be required due to operation below the analyzed Feedwater temperature for the safety analysis.
D. Incorrect – A Reactor power reduction would be required, NOT a Reactor scram. Plausible that a scram would be required due to operation below the analyzed Feedwater temperature for the safety analysis.

Technical Reference(s): N2-SOP-08

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: 2101-SOP08C01 EO-2

Question Source: Modified Bank - 2017 NRC #33

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.43
	Importance Rating	3.2

Knowledge of emergency communications systems and techniques.

Proposed Question: #74

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

- A plant transient has occurred and the Shift Manager has declared an Unusual Event.
- NO emergency response facilities have been activated.

Given the following actions:

(1) Transmit the NRC Event Notification Form
(2) Perform State and Local notifications
(3) Establish and maintain communications with the EOF and TSC

Which one of the following identifies which of these actions is the responsibility of the Shift Communicator, in accordance with EP-AA-112-100-F-05, Shift Communicator (CR) Checklist?

A. (1) and (2), only

B. (1) and (3), only

C. (2) and (3), only

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

Proposed Answer: A

Explanation: The Shift Communicator is responsible for both transmitting the NRC Event Notification Form and performing State and Local notifications. The Shift Communicator is not responsible for establishing and maintaining communications with the EOF and TSC.

B. Incorrect – The Shift Communicator is also responsible for performing State and Local notifications. Plausible that this would fall on another position, such as the Operations Communicator. The Shift Communicator is not responsible for establishing and maintaining communications with the EOF and TSC. Plausible because this is a responsibility of the Operations Communicator.
C. Incorrect – The Shift Communicator is also responsible for transmitting the NRC Event Notification Form. Plausible that this would fall on another position, such as the Operations Communicator or Shift Manager. The Shift Communicator is not responsible for establishing and maintaining communications with the EOF and TSC. Plausible because this is a responsibility of the Operations Communicator.
D. Incorrect – The Shift Communicator is not responsible for establishing and maintaining communications with the EOF and TSC. Plausible because this is a responsibility of the Operations Communicator.

Technical Reference(s): EP-AA-112-100, EP-AA-112-100-F-05

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP2 2019 NRC #75

Question History: NMP2 2019 NRC #75

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.22
	Importance Rating	3.6

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Proposed Question: #75

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

- Reactor pressure is 75 psig and lowering slowly with all ADS SRVs open.
- Reactor water level is -40" and lowering slowly with Condensate injecting.
- Suppression Pool average water temperature is 180^{o}F and rising slowly.
- Suppression Pool water level is 202' and stable.
- Suppression Chamber pressure is 18 psig and slowly rising.
- Drywell pressure is 20 psig and slowly rising.
- NO Core Spray pumps are available.
- RHR pump RHS*P1A has just been made available.
- NO other RHR pumps are available.

Which one of the following identifies the priority for use of RHS*P1A for these conditions?

A. LPCI

B. Drywell spray

C. Suppression Pool cooling

D. Suppression Chamber spray

Proposed Answer: A

Explanation: The given plant conditions present a loss of the RCS barrier and multiple challenges to the fuel clad and Primary Containment barriers. With Reactor water level <-39" and no Core Spray injecting, adequate core cooling is not currently assured. Suppression Pool temperature is well above the 90^{o}F threshold requiring use of all available Suppression Pool cooling and HCTL is a concern. Both Suppression Chamber and Drywell pressures are above the thresholds requiring use of sprays. Therefore, there are steps in N2-EOP-RPV and N2-EOP-PC that currently require use of the single RHR pump for LPCI, Suppression Pool cooling, Suppression Chamber spray, and Drywell spray. The bases for prioritizing these conflicting needs while in these procedures is to prioritize adequate core cooling. Therefore, RHR pump 1A must first be used in LPCI mode to inject to the Reactor.

B. Incorrect – LPCI must be prioritized over Drywell spray. Plausible because N2-EOP-PC does require Drywell spray and PSP is being challenged. Also plausible because in some circumstances during severe accidents Primary Containment is prioritized over fuel clad.
C. Incorrect – LPCI must be prioritized over Drywell spray. Plausible because N2-EOP-PC does require Suppression Pool cooling and HCTL is being challenged. Also plausible because in some circumstances during severe accidents Primary Containment is prioritized over fuel clad.
D. Incorrect – PCI must be prioritized over Suppression Chamber spray. Plausible because EO-100-103 does require Suppression Chamber spray and PSP is being challenged. Also plausible because in some circumstances during severe accidents Primary Containment is prioritized over fuel clad.

Technical Reference(s): N2-EOP-RPV, N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPRPVC01 EO-2

Question Source: Bank - SSES LOC28R NRC #73

Question History: SSES LOC28R NRC #73

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

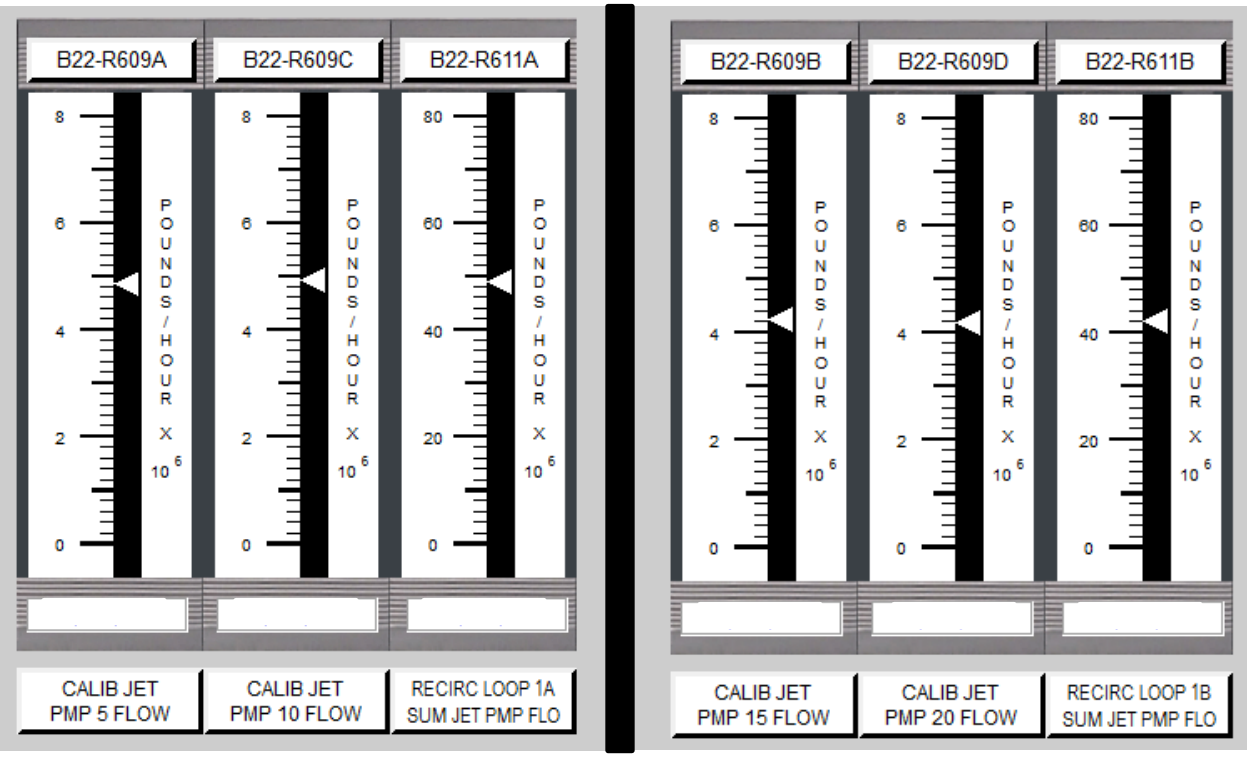
Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295001 2.2.40
	Importance Rating	4.7

Partial or Complete Loss of Forced Core Flow Circulation

Ability to apply Technical Specifications for a system.

Proposed Question: #76

The plant is stable at 52% power after an RCS flow control valve drift and automatic RCS flow control valve lockup. RCS loop flow indications on 2CEC*PNL602 are as follows:



Which one of the following describes the value of effective core flow and the need for Condition entry in Technical Specification (TS) 3.4.1, Recirculation Loops Operating?

Effective core flow is approximately…

A. 84 Mlbm/hr and TS 3.4.1 Condition entry is required.

B. 84 Mlbm/hr and TS 3.4.1 Condition entry is NOT required.

C. 98 Mlbm/hr and TS 3.4.1 Condition entry is required.

D. 98 Mlbm/hr and TS 3.4.1 Condition entry is NOT required.

Proposed Answer: A

Explanation: N2-OP-29 states, “The effective core flow shall be calculated by assuming both loops are at the smaller value of the two jet pump loop flows.” The given conditions show Recirc loop B has the lower jet pump loop flows. Therefore, the effective core flow is calculated by doubling the loop B summed jet pump flow (2 x 42 Mlbm/hr = 84 Mlbm/hr). This is greater than 70% of rated core flow (75.95 Mlbm/hr), therefore Technical Specification Surveillance Requirement 3.4.1.1.b applies and requires loop flow mismatch to be ≤ 5% or rated core flow (0.05 x 108.7 Mlbm/hr = 5.4 Mlbm/hr). The actual loop flow mismatch is greater than this limit (49 Mlbm/hr – 42 Mlbm/hr = 7 Mlbm/hr), therefore Technical Specification 3.4.1 Condition B entry is required.

B. Incorrect – TS 3.4.1 Condition B entry is required because loop flow mismatch is > 5.4 Mlbm/hr. Plausible because loop flow mismatch is less than the 10.8 Mlbm/hr limit that would apply if effective core flow were slightly lower (<70% of rated).
C. Incorrect – Effective loop flow is 84 Mlbm/hr, not 98 Mlbm/hr. Plausible because 98 Mlbm/hr is double the loop A flow.
D. Incorrect – Effective loop flow is 84 Mlbm/hr, not 98 Mlbm/hr. Plausible because 98 Mlbm/hr is double the loop A flow. TS 3.4.1 Condition B entry is required because loop flow mismatch is > 5.4 Mlbm/hr. Plausible because loop flow mismatch is less than the 10.8 Mlbm/hr limit that would apply if effective core flow were slightly lower (<70% of rated).

Technical Reference(s): Technical Specification 3.4.1 and associated bases

Proposed references to be provided to applicants during examination: Embedded picture

Learning Objective: N2-202001-RBO-14

Question Source: Bank - 2017 Cert #76

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295004 AA2.02
	Importance Rating	3.9

Partial or Complete Loss of DC Power

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Extent of partial or complete loss of D.C. power

Proposed Question: #77

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP/HOT STANDBY.
- The Reactor is critical above the point of adding heat.
- Reactor pressure is 700 psig and rising.

Then, the following occurs:

- Div I Battery Charger 2BYS*CHGR2A1 trips.
- Div I Battery Charger 2BYS*CHGR2A2 is unavailable.
- 2BYS*SWG002A voltage is 119 VDC and lowering slowly.

Which one of the following identifies the applicable Technical Specification for the current plant conditions and the need for Condition entry in this Technical Specification?

	Applicable Technical Specification	Condition Entry
A.	3.8.4, DC Sources – Operating	Required
B.	3.8.4, DC Sources – Operating	NOT required
C.	3.8.5, DC Sources – Shutdown	Required
D.	3.8.5, DC Sources – Shutdown	NOT required

Proposed Answer: A

Explanation: Technical Specification 3.8.4 is applicable in Modes 1, 2, and 3. Technical Specification 3.8.5 is applicable in Modes 4 and 5. The given plant conditions place the plant in Mode 2, therefore Technical Specification 3.8.4 is applicable. Technical Specification 3.8.4 bases require at least one battery charger to be operable for 2BYS*SWG002A. With 2BYS*CHGR2A1 tripped and 2BYS*CHGR2A2 unavailable, there is no operable battery charger for 2BYS*SWG002A. Therefore, Technical Specification 3.8.4 Condition entry is required.

B. Incorrect – Condition entry is required. Plausible because the bus remains energized and voltage is still close to nominal.
C. Incorrect – Technical Specification 3.8.4 is applicable. Plausible because the plant is not in Mode 1.
D. Incorrect – Technical Specification 3.8.4 is applicable. Plausible because the plant is not in Mode 1. Condition entry is required. Plausible because the associated bus remains energized and voltage is still close to nominal.

Technical Reference(s): Technical Specifications 3.8.4 and 3.8.5 and associated bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-263000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295006 2.1.25
	Importance Rating	4.2

Scram

Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: #78

A plant startup is in progress with the following:

- N2-OSP-RMC-@001, Control Rod Drive Scram Insertion Time Testing, is in progress.
- The following scram time data has been collected for four (4) adjacent control rods:

Notch Position	Control Rod 18-23 (sec)	Control Rod 22-23 (sec)	Control Rod 18-27 (sec)	Control rod 22-27 (sec)
45	0.510	0.271	0.601	0.489
39	0.832	0.448	0.978	0.854
25	1.899	1.017	2.098	2.716
05	3.375	1.923	3.625	7.143

Which one of the following describes the Technical Specification implications of this data, if any?

A. NO Technical Specification Condition entry is required.

B. Technical Specification 3.1.3 Condition C must be entered. NO other Technical Specification Condition entry is required.

C. Technical Specification 3.1.4 Condition A must be entered. NO other Technical Specification Condition entry is required.

D. Technical Specification 3.1.4 Condition A must be entered and Technical Specification 3.1.3 Condition C must be entered.

Proposed Answer: B

Explanation: The given scram times must be compared to the required values in Technical Specification Table 3.1.4-1. Based on this comparison, control rods 18-23 and 22-23 meet all required scram times at all notch positions. Control rod 18-27 does not meet any of the required scram times but does reach position 05 in less than 7 seconds. Therefore, control rod 18-27 is slow, but operable. Control rod 22-27 meets the first two required scram times, but not the last two. In addition, control rod 22-27 reaches position 05 in greater than 7 seconds. Therefore, control rod 22-27 is both slow and inoperable. With only two slow control rods, Technical Specification 3.1.4 LCO is met and Condition A is not required to be entered. However, with control rod 22-27 inoperable, Technical Specification 3.1.3 Condition C must be entered.

A. Incorrect – Technical Specification 3.1.3 Condition C must be entered. Plausible because this would be correct if only the control rod 22-27 position 05 time was slightly lower.
C. Incorrect – Technical Specification 3.1.4 Condition A entry is not required. Plausible because this would be correct if any of the scram times for control rods 18-23 or 22-23 exceeded limits. Technical Specification 3.1.3 Condition C must be entered. Plausible because this would be correct if only the control rod 22-27 position 05 time was slightly lower.
D. Incorrect – Technical Specification 3.1.4 Condition A entry is not required. Plausible because this would be correct if any of the scram times for control rods 18-23 or 22-23 exceeded limits.

Technical Reference(s): Technical Specifications 3.1.3 and 3.1.4

Proposed references to be provided to applicants during examination: Technical Specifications 3.1.3 and 3.1.4, including Table 3.1.4-1, but without bases

Learning Objective: N2-201001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295019 AA2.01
	Importance Rating	3.6

Partial or Complete Loss of Instrument Air

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air system pressure

Proposed Question: #79

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

- A significant Instrument Air leak has occurred.
- N2-SOP-19, Loss of Instrument Air, is being executed.
- 2IAS-PI194 (RB el. 261'), Inst Air Rcvr 2IAS-TK3 Pressure, indicates 68 psig and stable.
- 2IAS-PI101 (P851), Instrument Air Pressure, indicates 68 psig and stable.

Which one of the following describes the need for a Reactor scram and the operability of the outboard MSIVs based on the given air pressures, in accordance with N2-SOP-19?

	Reactor scram	Outboard MSIVs
A.	Required	Operable
B.	Required	Inoperable
C.	NOT required	Operable
D.	NOT required	Inoperable

Proposed Answer: B

Explanation: N2-SOP-19 includes the following condition steps:

IF	THEN
2IAS-PI101 (P851), Instrument air pressure, lowers to < 85 psig,	Verify 2IAS-AOV171 auto closes AND dispatch an Operator to monitor 2RDS-PI133 & 2IAS-PI194 (RB el. 261').
2IAS-PI194 (RB el. 261'), Inst air rcvr 2IAS-TK3 pressure, lowers to < 74 psig,	**SCRAM** the Reactor per N2-SOP-101C AND declare the OUTBOARD MSIV's inoperable.
2IAS-PI101 (P851), Instrument air pressure, lowers to < 70 psig,	**SCRAM** the Reactor per N2-SOP-101C.

Reactor Scram required ONLY if control rod(s) are withdrawn.

Due to air pressure on 2IAS-PI194 (RB el. 261') being below 74 psig, N2-SOP-19 requires both entering N2-SOP-101C to scram the Reactor and declaring the outboard MSIVs inoperable.

Note: The question meets SRO only requirements because the candidate must assess plant conditions (two given IA pressures) and then use this assessment to determine if an additional procedure is yet required to be implemented (N2-SOP-101C). This cannot be answered solely by knowing systems knowledge, immediate operator actions, SOP entry conditions, or overall mitigative strategy. Additional SRO knowledge is tested in the operability determination of the MSIVs.

A. Incorrect – The outboard MSIVs must be declared inoperable. Plausible because air pressures are still above the 65 psig threshold related to scram air header pressure.
C. Incorrect – A Reactor scram is required. The outboard MSIVs must be declared inoperable. Plausible because air pressures are still above the 65 psig threshold related to scram air header pressure.
D. Incorrect – A Reactor scram is required. Plausible because air pressures are still above the 65 psig threshold related to scram air header pressure.

Technical Reference(s): N2-SOP-19

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP19C01 EO-2

Question Source: Bank - 2017 NRC #76

Question History: 2017 NRC #76

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295024 2.4.41
	Importance Rating	4.6

High Drywell Pressure

Knowledge of the emergency action level thresholds and classifications.

Proposed Question: #80

Given the following categories from the NMP Unit 2 EAL Wallboard:

(1) Loss or Potential Loss of Fuel Clad
(2) Loss or Potential Loss of Reactor Coolant System
(3) Loss or Potential Loss of Containment

Which one of the following identifies which of these categories use Primary Containment pressure as a classifying parameter?

A. (2) only

B. (3) only

C. (2) and (3) only

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

Proposed Answer: C

Explanation: Primary Containment pressure is used as a classifying parameter for Loss of Reactor Coolant System (>1.68 psig) and Potential Loss of Containment (>45 psig). Primary Containment pressure is not used as a classifying parameter for Loss or Potential Loss of Fuel Clad.

A. Incorrect – Primary Containment pressure is also used as a classifying parameter for Potential Loss of Containment (>45 psig).
B. Incorrect – Primary Containment pressure is also used as a classifying parameter for Loss of Reactor Coolant System (>1.68 psig).
D. Incorrect – Primary Containment pressure is not used as a classifying parameter for Loss or Potential Loss of Fuel Clad. Plausible because Drywell radiation is used as a classifying for all three of the given categories.

Technical Reference(s): EP-AA-1013 Addendum 4 Appendix 1

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)

Comments:

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

High Drywell Temperature

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

Proposed Question: #81

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

- Reactor water level is 100 inches and slowly rising.
- Reactor pressure is 600 psig and slowly lowering.
- Drywell pressure is 14 psig and slowly rising
- Drywell temperature is 320^{o}F and slowly rising.
- Suppression Chamber pressure is 12 psig and slowly rising.
- Suppression Pool water level is 200.5 feet and stable.
- NO actions have yet been directed from N2-EOP-PC, Primary Containment Control.

Note: The Drywell Spray Initiation Limit and Pressure Suppression Pressure curves are ***provided on the next page***.

Which one of the following describes the required action(s), in accordance with N2-EOP-PC?

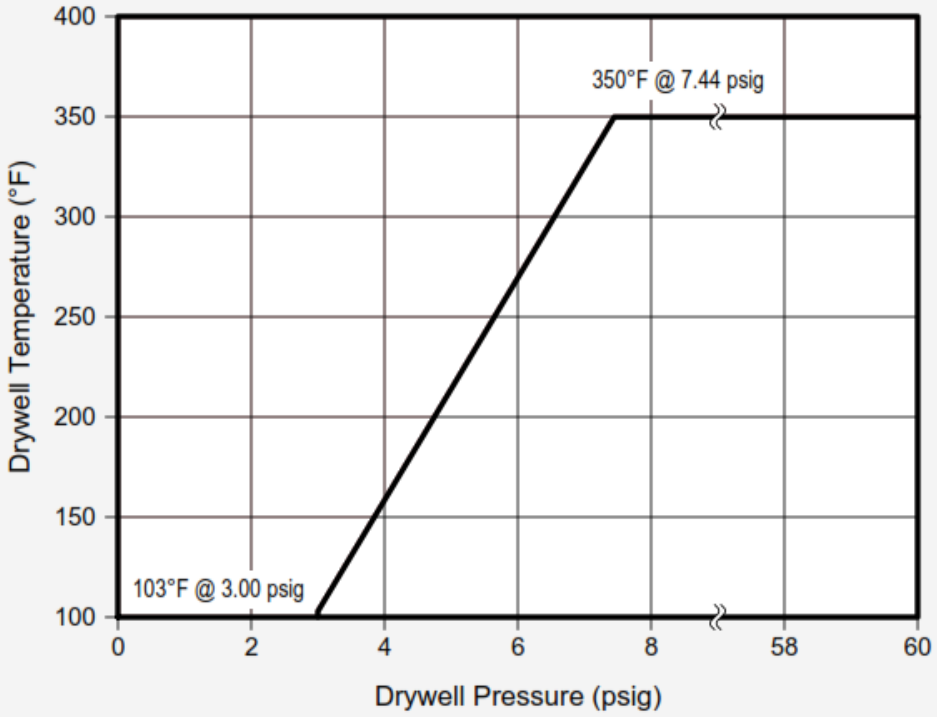
A. Spray the Suppression Chamber and then the Drywell. An RPV Blowdown is NOT yet required.

B. Spray the Suppression Chamber, but NOT the Drywell. An RPV Blowdown is NOT yet required.

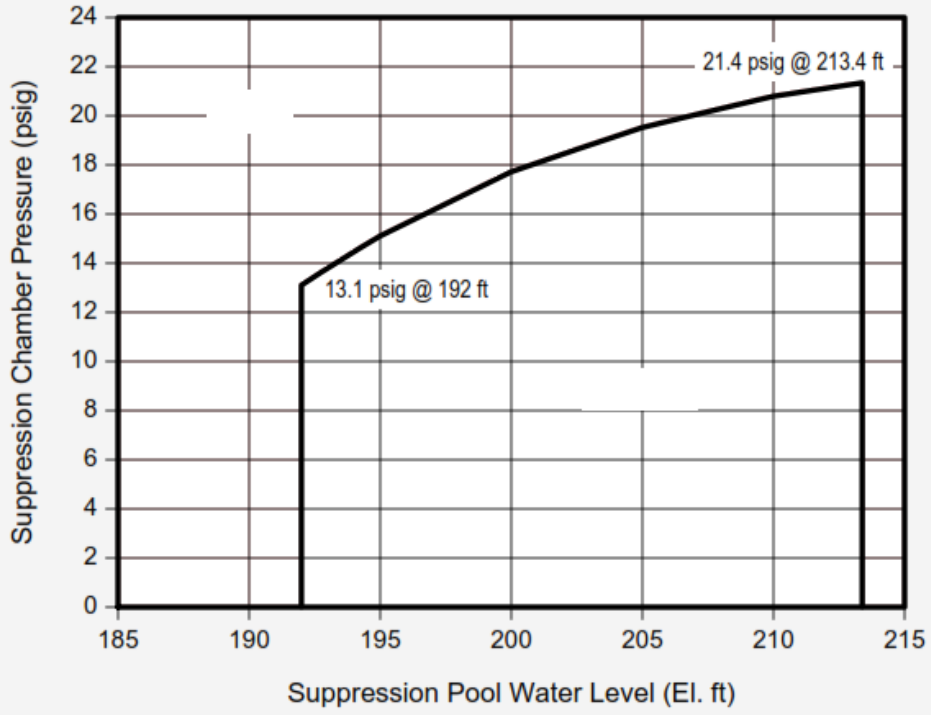
C. Spray the Suppression Chamber, but NOT the Drywell. Simultaneously perform an RPV Blowdown.

D. Perform an RPV Blowdown. Do NOT spray the Suppression Chamber or the Drywell.

K Drywell Spray Initiation Limit



L Pressure Suppression Pressure



Proposed Answer: A

Explanation: The given Suppression Chamber pressure is above the threshold requiring Suppression Chamber and Drywell sprays (10 psig). The combination of Drywell temperature and pressure are on the GOOD side of the Drywell Spray Initiation Limit. The combination of Suppression Chamber pressure and Suppression Pool water level are below the threshold requiring RPV Blowdown (PSP curve). Drywell temperature is highly elevated, but below the 340^{o}F threshold for RPV Blowdown. Therefore, the Suppression Chamber and Drywell need to be sprayed, but an RPV Blowdown is not yet required.

B. Incorrect – The Drywell should also be sprayed. Plausible because Drywell temperature is highly elevated and this would be correct if that resulted in being on the BAD side of the Drywell Spray Initiation Limit.
C. Incorrect – The Drywell should also be sprayed. Plausible because Drywell temperature is highly elevated and this would be correct if that resulted in being on the BAD side of the Drywell Spray Initiation Limit. An RPV Blowdown is not yet required. Plausible that this would be correct if the Drywell could not be sprayed. Also plausible because of how high Drywell temperature is.
D. Incorrect – The Suppression Chamber and Drywell should be sprayed. Plausible because Drywell temperature is highly elevated such that the Drywell Spray Initiation Limit would be challenged at lowering Drywell pressures. An RPV Blowdown is not yet required. Plausible that this would be correct if the Suppression Chamber and Drywell could not be sprayed. Also plausible because of how high Drywell temperature is.

Technical Reference(s): N2-EOP-PC

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Bank - JAF 14-2 NRC #78

Question History: JAF 14-2 NRC #78

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295038 2.4.30
	Importance Rating	4.1

High Offsite Radioactivity Release Rate

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: #82

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

Time (hh:mm)	Condition
12:00	Indications become available in the Control Room that require declaration of Site Area Emergency RS1: **RS1** 1 2 3 4 5 D *Bases: 2-28* **Release of gaseous radioactivity resulting in offsite dose greater than 100 mRem TEDE or 500 mRem CDE thyroid.** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 100 mRem TEDE** **OR** b. **> 500 mRem CDE Thyroid**
12:15	The Shift Manager declares Site Area Emergency RS1.
14:00	Indications become available in the Control Room that require declaration of General Emergency RG1: **RG1** 1 2 3 4 5 D *Bases: 2-26* **Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mRem TEDE or 5,000 mRem CDE thyroid.** 2. Dose assessment using actual meteorology indicates doses at or beyond the site boundary of **EITHER**: a. **> 1000 mRem TEDE** **OR** b. **> 5000 mRem CDE Thyroid**
14:15	The Shift Manager declares General Emergency RG1.

Which one of the following identifies the latest acceptable time for providing the initial Protective Action Recommendations (PARs) to the State and County, in accordance with EP-AA-111, Emergency Classification and Protective Action Recommendations, and EP-CE-114-100, Emergency Notifications?

A. 12:15

B. 12:30

C. 14:15

D. 14:30

Proposed Answer: D

Explanation: PARs are only made to the State and County upon declaration of a General Emergency. The PARs are made as part of the Part 1 Notification, which is required to be provided no later than 15 minutes after the declaration of the General Emergency.

A. Incorrect – PARs are NOT required to be provided until 14:30. Plausible that they would be required early due to a release in progress and a Site Area Emergency present.
B. Incorrect – PARs are NOT required to be provided until 14:30. Plausible that they would be required early due to a release in progress and a Site Area Emergency present.
C. Incorrect – PARs are NOT required to be provided until 14:30. Plausible that they would be required 15 minutes after indications are available in the Control Room of a General Emergency, as this is when the declaration is required and there is actual consequence to the health and safety of the general public.

Technical Reference(s): EP-AA-111, EP-CE-114-100

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – NMP1 2015 NRC #85

Question History: NMP1 2015 NRC #85

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295017 AA2.03
	Importance Rating	3.9

High Off-site Release Rate

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Radiation levels: Plant-Specific

Proposed Question: #83

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

- Off-gas is in single train operation, in preparation for a maintenance window.
- Annunciator 851326, OFFGAS RADIATION HIGH, is in alarm.
- Off-gas Radiation Monitors 2OFG-RU13A and 2OFG-RU13B are above their Alarm (RED) setpoints.
- Main Steam Line radiation monitors are NOT in alarm.

Which one of the following describes the required execution of the Special Operating Procedures, if any?

A. NO Special Operating Procedure entry is required unless conditions further degrade.

B. Enter N2-SOP-17, Fuel Failure or High Activity in Rx Coolant or Offgas. NO other Special Operating Procedure entry is required unless conditions further degrade.

C. Enter N2-SOP-17, Fuel Failure or High Activity in Rx Coolant or Offgas, and N2-SOP-101D, Rapid Power Reduction. Entry into N2-SOP-101C, Reactor Scram, is NOT required.

D. Enter N2-SOP-17, Fuel Failure or High Activity in Rx Coolant or Offgas, and N2-SOP-101C, Reactor Scram.

Proposed Answer: D

Explanation: With Annunciator 851326 in alarm, entry into N2-SOP-17 is required. With both 2OFG-RU13A and 2OFG-RU13B in alarm, Offgas isolates and N2-SOP-17 requires entry into N2-SOP-101C.

A. Incorrect – With Annunciator 851326 in alarm, entry into N2-SOP-17 is required. Plausible because Main Steam Line radiation monitors are another entry condition, but are not in alarm.
B. Incorrect – With both 2OFG-RU13A and 2OFG-RU13B in alarm, Offgas isolates and N2-SOP-17 requires entry into N2-SOP-101C. Plausible because Main Steam Line radiation monitors are not in alarm.
C. Incorrect – With both 2OFG-RU13A and 2OFG-RU13B in alarm, Offgas isolates and N2-SOP-17 requires entry into N2-SOP-101C. Plausible because Main Steam Line radiation monitors are not in alarm.

Technical Reference(s): N2-SOP-17

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-SOP17C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:

Level	SRO
Tier #	1
Group #	2
K/A #	295033 2.4.6
Importance Rating	4.7

High Secondary Containment Area Radiation Levels

Knowledge of EOP mitigation strategies.

Proposed Question: #84

The plant is operating at 100% power when the following sequence occurs:

Time (hh:mm)	Condition
00:00	An un-isolable Reactor Water Cleanup (WCS) leak develops.
00:07	Radiation monitor RE2A, RCS Pump Instrument Panel A, on Reactor Building Elevation 215 is reading 9.2 E+03 mR/hr and rising and indicating RED on DRMS display.
00:11	WCS pump room A temperature is 212^{o}F and rising.
00:16	Radiation monitor RE149, RWCU Valve Area, on Reactor Building Elevation 328 is reading 8.0 E+03 mR/hr and rising and indicating RED on DRMS display.
00:22	WCS pump room B temperature is 212^{o}F and rising.

Which one of the following identifies the ***earliest*** time at which an RPV Blowdown is required, in accordance with N2-EOP-SC, Secondary Containment Control?

A. 00:07

B. 00:11

C. 00:16

D. 00:22

Proposed Answer: C

Explanation: Per OP-NM-101-111-1001, each area radiation monitor is treated as a separate area. Since RE2A and RE149 are reading above Max Safe Values, two General Area radiation levels are above Max Safe at time 00:16. Because this is caused by a primary system that cannot be isolated, a Blowdown is required by N2-EOP-SC Step SC-12.

A. Incorrect – An RPV Blowdown is first required at time 00:16. Plausible because this is when a General Area exceeds the first Max Safe value, which does require a React scram.
B. Incorrect – An RPV Blowdown is first required at time 00:16. Plausible because this is when two Max Safe values are exceeded, however they are in different parameters.
D. Incorrect – An RPV Blowdown is first required at time 00:16. Plausible because this is when two Max Safe values are exceeded in both of the affected parameters.

Technical Reference(s): N2-EOP-SC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295036 EA2.02
	Importance Rating	3.1

Secondary Containment High Sump/Area Water Level

Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Water level in the affected area

Proposed Question: #85

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

- A fire header has ruptured in the Reactor Building.
- Annunciator 851453, RX BLDG FL DRAIN SYS TROUBLE, alarms.
- The Radwaste Operator reports the following annunciators are in alarm:
  - 513113, RX BLDG FL DR SUMP TK2A LEVEL HIGH/HIGH
  - 513117, RX BLDG FL DR SUMP TK2E LEVEL HIGH/HIGH
- Annunciator 851443, REACTOR BLDG GENERAL AREA TANK 2A FLOODING, alarms.
- Annunciator 851444, REACTOR BLDG GENERAL AREA TANK 2E FLOODING, alarms.
- Attempts to isolate the leak and remove water from these areas have been unsuccessful.

Which one of the following describes the required control of the Reactor, in accordance with the Emergency Operating Procedures?

A. The Reactor may continue to operate at the current power level.

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

C. A Reactor scram is required, but an RPV Blowdown is NOT required.

D. A Reactor scram and an RPV Blowdown are required.

Proposed Answer: B

Explanation: The high area water levels require entry into N2-EOP-SC, Secondary Containment Control. The two flooding annunciators indicate that two General Areas have exceeded Max Safe water levels. Since the flooding is caused by a fire header break, this is a non-primary system discharge. N2-EOP-SC therefore requires a Reactor shutdown, but not a Reactor scram or RPV Blowdown.

A. Incorrect – A Reactor shutdown is required. Plausible because this would be correct if only one area was above Max Safe water level. Also plausible because this is a non-primary system discharge.
C. Incorrect – A Reactor scram is not required. Plausible because this would be correct for a primary system discharge with only one area above Max Safe water level.
D. Incorrect – A Reactor scram and an RPV Blowdown are not required. Plausible because this would be correct for a primary system discharge.

Technical Reference(s): N2-EOP-SC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPSCC01 EO-2

Question Source: Bank - JAF 16-1 NRC #85

Question History: JAF 16-1 NRC #85

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

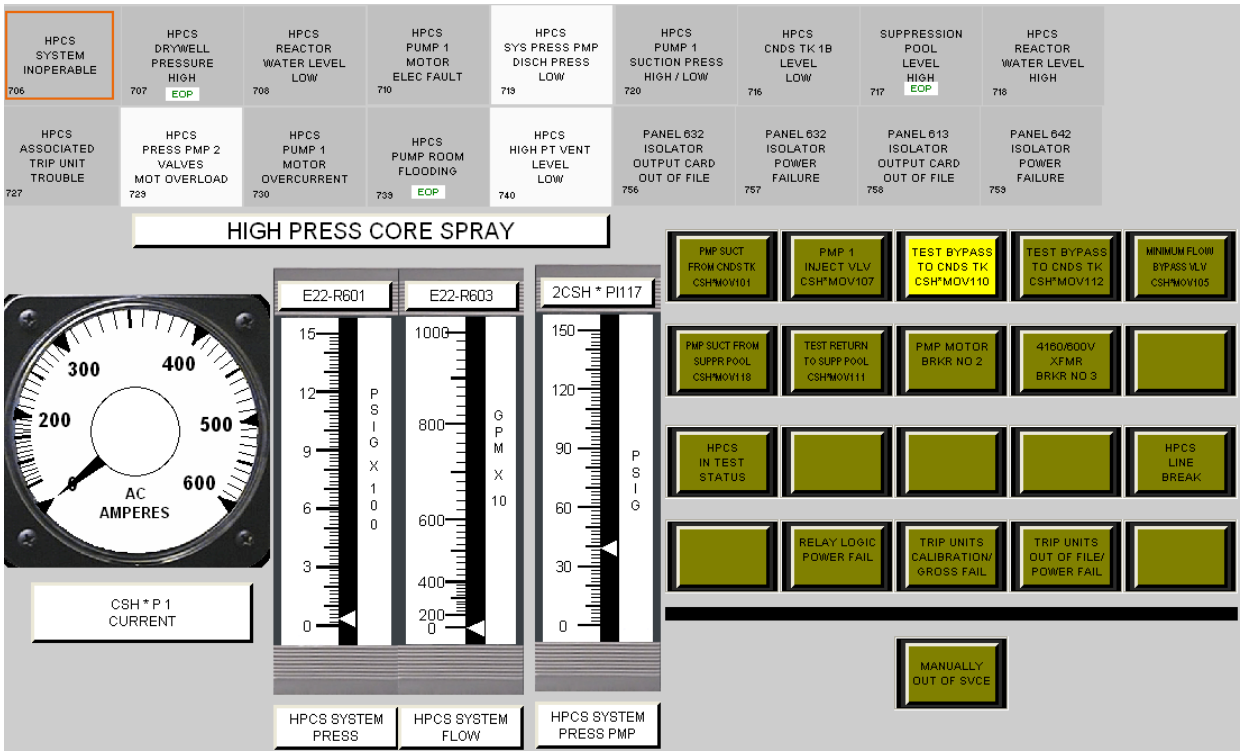
Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	209002 2.2.44
	Importance Rating	4.4

High Pressure Core Spray

Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions.

Proposed Question: #86

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



Which one of the following describes the status of HPCS and the required action(s)?

HPCS is…

A. INOPERABLE because 2CSH*MOV110 has lost control power. Dispatch an operator to locally verify shut 2CSH*MOV110.

B. INOPERABLE. Place HPCS in Pull to Lock per ARP 601740 and perform actions within N2-OP-33 for CSH Operation with WTR LEG PMP2 Out of Service.

C. INOPERABLE. Leave 2CSH*P1 in NORMAL AFTER STOP and perform actions within N2-OP-33 for CSH Operation with WTR LEG PMP2 Out of Service.

D. OPERABLE because CST B Tank level is above the low level alarm setpoint. Perform actions within N2-OP-33 for CSH Operation with WTR LEG PMP2 Out of Service.

Proposed Answer: B

Explanation: The given indications show that the HPCS system pressure pump (CSH*P2) is tripped. Normal system pressure would be approximately 75 psig with the pump running. N2-OP-33 requires declaring HPCS inoperable. Additionally, ARP 601740 requires placing the HPCS pump in pull to lock.

A. Incorrect – During power operations, the 2CHS*MOV110 breaker is normally open for Appendix R purposes. Plausible based on the valve motor overload alarm and lit CSH*MOV110 light.
C. Incorrect – The HPCS pump must be placed in pull to lock. Plausible that other procedural actions would mitigate the issue such that the HPCS pump could remain in standby.
D. Incorrect – HPCS is not operable. Plausible because annunciator 601716 is not in alarm.

Technical Reference(s): ARP 601740, N2-OP-33

Proposed references to be provided to applicants during examination: Embedded picture

Learning Objective: N2-209002-RBO-14

Question Source: Bank – 2012 NRC #88

Question History: 2012 NRC #88

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	212000 A2.08
	Importance Rating	4.2

Reactor Protection System

Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor level

Proposed Question: #87

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

- Current time is 0800.
- A review of calibration data for the Reactor water level inputs to the RPS scram function shows the following setpoints:

Reactor Water Level Instrument	Scram Setpoint
2ISC*LIS1680A	155.5”
2ISC*LIS1680B	161.8”
2ISC*LIS1680C	160.5”
2ISC*LIS1680D	160.1”

Which one of the following identifies the latest time to enter Mode 3, if any, if the setpoints remain at these values, in accordance Technical Specification 3.3.1.1, Reactor Protection System (RPS) Instrumentation?

A. 2100 today

B. 0200 tomorrow

C. 0800 tomorrow

D. There is NO need to enter Mode 3.

Proposed Answer: D

Explanation: Technical Specification Table 3.3.1.1-1 requires the Reactor water level scram setpoints to be set greater than or equal to 157.8". The given information shows that 2ISC*LIS1680B, 2ISC*LIS1680C, and 2ISC*LIS1680D are set correctly, but 2ISC*LIS1680A setpoint is too low. This requires entering Technical Specification 3.3.1.1 Condition A. This Condition requires inserting a half scram within 12 hours. Once this is carried out, there is no need to enter Mode 3.

A. Incorrect – There is no need to enter Mode 3. Plausible because this would be the result if Condition C were entered, followed by Conditions D and H (1 hour + 12 hours).
B. Incorrect – There is no need to enter Mode 3. Plausible because this would be the result if Condition B were entered, followed by Conditions D and H (6 hours + 12 hours).
C. Incorrect – There is no need to enter Mode 3. Plausible because this would be correct if a half scram was not inserted per Condition A and then Conditions D and H were entered (12 hours + 12 hours).

Technical Reference(s): Technical Specification 3.3.1.1

Proposed references to be provided to applicants during examination: Technical Specification 3.3.1.1 without allowable values or bases

Learning Objective: N2-212000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	2
	Group #	1
	K/A #	259002 2.2.37
	Importance Rating	4.6

Reactor Water Level Control

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

Proposed Question: #88

A loss of coolant accident has occurred with the following:

- Containment Sprays are in service
- The hottest Drywell temperature is 325^{o}F and slowly lowering.
- Reactor pressure is 75 psig and slowly lowering.
- Reactor water level indicates:
  - 150 inches and stable on Narrow Range indicators.
  - 50 inches and slowly lowering on Wide Range indicators.
- Fuel Zone indicators are inoperable.
- N2-EOP-RPV, RPV Control, and N2-EOP-PC, Primary Containment Control, are being executed.

Note: EOP Detail A is ***provided on the next page***.

Which one of the following describes the required action, in accordance with the Emergency Operating Procedures?

A. Remain in N2-EOP-RPV and control Reactor water level using the Narrow Range indicators.

B. Remain in N2-EOP-RPV and control Reactor water level using the Wide Range indicators.

C. Exit N2-EOP-RPV and enter N2-EOP-C2, RPV Blowdown.

D. Exit N2-EOP-RPV and enter N2-EOP-C4, RPV Flooding.

A RPV Water Level Instrument Restrictions

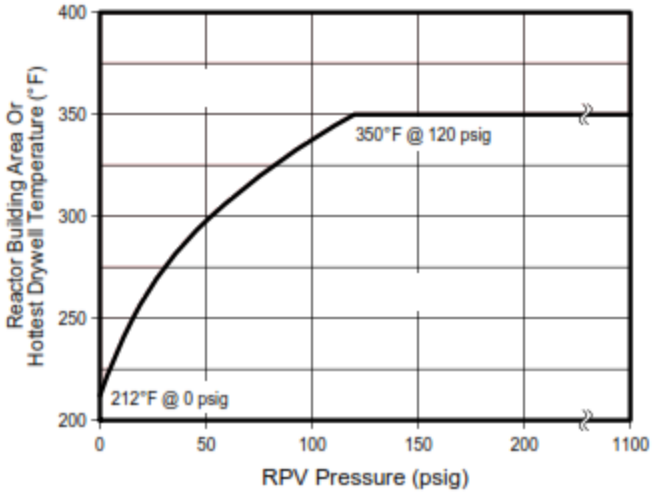
⚠ **If the temperature near any RPV water level instrument run is above the RPV Saturation Temperature (Fig B) (EOP-6.28), the instrument may be unreliable due to boiling in the run.**

An RPV water level instrument may be used only when:

- There is no evidence of instrument leg flashing,

  AND
- The instrument reads above the Minimum Indicated Level (Table C).

B RPV Saturation Temperature



C Minimum Indicated Levels

Instrument	Hottest Reactor Building or Drywell Area Temperature	
	≤ 350°F	> 350°F
Shutdown Range	200 in.	250 in.
Upset Range	190 in.	260 in.
Narrow Range	150 in.	155 in.
Wide Range	25 in.	25 in.
Fuel Zone	-150 in.	-155 in.

Proposed Answer: B

Explanation: The elevated Drywell temperature and low Reactor water level requires use of Detail A to determine usability of Reactor water level instrumentation. Narrow Range indicators are indicating at their Minimum Indicated Levels, and therefore cannot be used to determine and control Reactor water level. Wide Range indicators are indicating above their Minimum Indicated Levels. The combination of Drywell temperature and Reactor pressure is on the BAD side of the RPV Saturation Temperature curve. However, indicators can still be used as long as there is no evidence of instrument leg flashing. Therefore, Wide Range indicators can still be used to determine and control Reactor water level. It is not appropriate to exit N2-EOP-RPV and enter N2-EOP-C4.

A. Incorrect – Narrow Range indicators are indicating at their Minimum Indicated Levels, and therefore cannot be used to determine and control Reactor water level. Plausible because the nominal range for Narrow Range indicators goes down to 145".
C. Incorrect – Wide Range indicators can still be used to determine and control Reactor water level. It is not appropriate to exit N2-EOP-RPV. Plausible because the given combination of Drywell temperature and Reactor pressure is on the BAD side of the RPV Saturation Temperature curve. Also plausible that N2-EOP-C2 would be used before N2-EOP-C4 to open ADS valves.
D. Incorrect – Wide Range indicators can still be used to determine and control Reactor water level. It is not appropriate to exit N2-EOP-RPV and enter N2-EOP-C4. Plausible because the given combination of Drywell temperature and Reactor pressure is on the BAD side of the RPV Saturation Temperature curve.

Technical Reference(s): N2-EOP-RPV

Proposed references to be provided to applicants during examination: Embedded graphic

Learning Objective: 2101-EOPC4C01 EO-2

Question Source: Bank - NMP1 2017 NRC #76

Question History: NMP1 2017 NRC #76

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	262002 A2.01
	Importance Rating	2.8

Uninterruptable Power Supply (AC/DC)

Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Under voltage

Proposed Question: #89

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

- 2VBA*UPS2B, Division II UPS, is in operation.
- The DC supply voltage to 2VBA*UPS2B is lost because DC supply breaker trips open and CANNOT be re-closed.

Which one of the following describes the required action(s), in accordance with Technical Specifications?

A. 2VBA*UPS2D must be placed in operation within 8 hours or the reactor must be in Mode 3 within 20 hours.

B. 2VBA*UPS2D must be placed in operation within 24 hours or the reactor must be in Mode 3 within 36 hours.

C. The reactor must be in Mode 3 within 20 hours. 2VBA*UPS2D cannot be placed in operation to satisfy the requirements of Technical Specifications.

D. The reactor must be in Mode 3 within 36 hours. 2VBA*UPS2D cannot be placed in operation to satisfy the requirements of Technical Specifications.

Proposed Answer: B

Explanation: Without a DC power supply, UPS2B is inoperable. Per TS Bases for 3.8.7, either UPS2B or 2D need to be operable and in operation in order to meet LCO 3.8.7. Per TS 3.8.7 you have 24 hours to restore an operable UPS to service. If you don't restore an UPS to service, you must be in mode 3 within 12 hours. So 24 + 12 hours is 36 hours.

A. Incorrect – 24 hours is allowed, not 8. Plausible because the related TS 3.8.8 Condition A completion time is 8 hours.
C. Incorrect – 2VBA*UPS2D can be placed in service to satisfy TS requirements. Plausible that the loss of 2VBA*UPS2B would still require the same TS actions regardless of 2VBA*UPS2D status.
D. Incorrect – 2VBA*UPS2D can be placed in service to satisfy TS requirements. Plausible that the loss of 2VBA*UPS2B would still require the same TS actions regardless of 2VBA*UPS2D status.

Technical Reference(s): T.S. 3.8.7 and bases

Proposed references to be provided to applicants during examination: T.S. 3.8.7 and 3.8.8 without bases

Learning Objective: N2-262002-RBO-14

Question Source: Bank – 2016 NRC #89

Question History: 2016 NRC #89

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	2
	Group #	1
	K/A #	400000 2.1.27
	Importance Rating	4.0

Component Cooling Water

Knowledge of system purpose and/or function.

Proposed Question: #90

The plant is operating at 100% power. Given the following Service Water system valves:

(1) 2SWP*MOV50A(B), DISCH HEADER CROSS-TIE ISOL VLV
(2) 2SWP*MOV19A(B), OUTLET TO REACTOR BLDG
(3) 2SWP*MOV3A(B), OUTLET TO TURBINE BLDG

Which one of the following identifies which of these valves must be operable for the Service Water system to be considered operable, in accordance with Technical Specification 3.7.1, Service Water (SW) System and Ultimate Heat Sink (UHS), bases?

A. (1) only

B. (2) only

C. (1) and (2) only

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

Proposed Answer: D

Explanation: Technical Specification 3.7.1 Bases require the SW supply header cross connect valves and supply isolation valves (including non-essential header) to be operable for the system to be considered operable. This includes all of the given valves.

A. Incorrect – 2SWP*MOV19A(B) and 2SWP*MOV3A(B) also need to be operable. Plausible because 2SWP*MOV50A(B) are required.
B. Incorrect – 2SWP*MOV50A(B) and 2SWP*MOV3A(B) also need to be operable. Plausible because 2SWP*MOV19A(B) are required.
C. Incorrect – 2SWP*MOV3A(B) also need to be operable. Plausible because 2SWP*MOV50A(B) and 2SWP*MOV19A(B) are required.

Technical Reference(s): Technical Specification 3.7.1 Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-276000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	216000 2.2.38
	Importance Rating	4.5

Nuclear Boiler Instrumentation

Knowledge of conditions and limitations in the facility license.

Proposed Question: #91

The plant is operating at 100% power when 2ISC*PT6A (Wide range Reactor pressure transmitter that feeds Panel 601 recorder) fails downscale. ***See the next page*** for a picture of the associated recorder and the relevant section of PID.

Which one of the following describes the Technical Specification implication of this failure, if any?

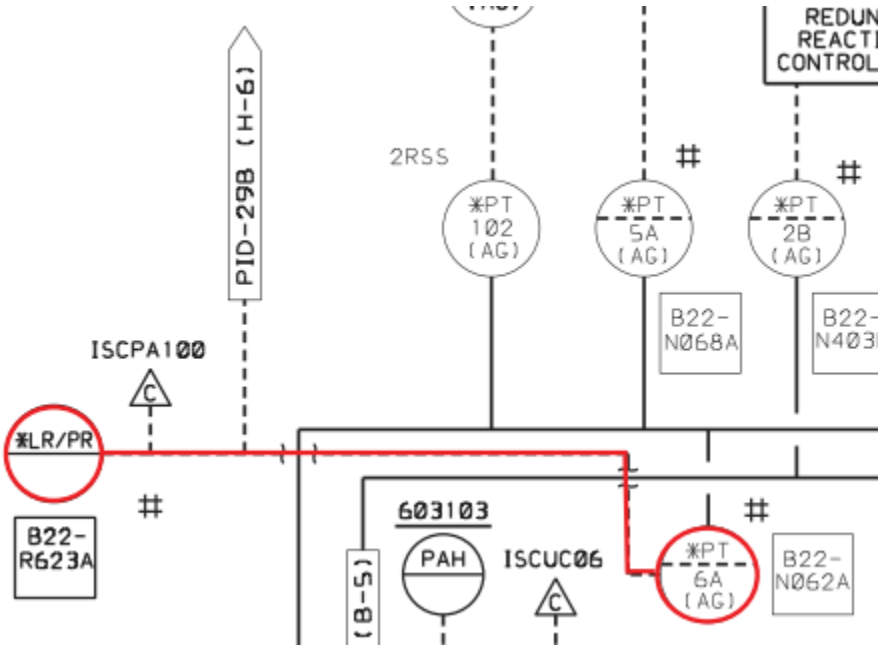
A. NO Technical Specification Condition entry is required.

B. Technical Specification Condition entry is required. The most limiting Completion Time is 30 days.

C. Technical Specification Condition entry is required. The most limiting Completion Time is 7 days.

D. Technical Specification Condition entry is required. The most limiting Completion Time is 12 hours.


YOKOGAWA
Press + lvl ALL DISP
Jan.19.2021 09:31:43 EVENT
Rx vessel press
0
PSIG
H
Rx wtr lvl wide
169.59
Inches
L
DISP/ ENTER
2ISC*LR1623A/2ISC*PR1623A
(B22-R623A)
REACTOR
2ISC*PRI623A
PRESS
2ISC*LIS1691A
WATER LEVEL
RESET


REDUN
REACTI
CONTROL
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PID-298 (H-6)
*PT
102
(AG)
*PT
5A
(AG)
*PT
2B
(AG)
B22-
N068A
B22-
N403
ISCPA100
C
*LR/PR
B22-
R623A
603103
PAH
(B-5)
ISCUC06
C
*PT
6A
(AG)
B22-
N062A

Proposed Answer: B

Explanation: 2ISC*PT6A provides indication in the Control Room, but does not provide any protective functions. 2ISC*PT6A is one of two required channels of post-accident instrumentation per Technical Specification Table 3.3.3.1-1. With one channel inoperable, Technical Specification 3.3.3.1 Condition A entry is required. The associated Completion Time is 30 days.

A. Incorrect – Technical Specification Condition entry is required. Plausible because this would be correct if there were more than 2 available channels to provide for Function 1 in Technical Specification Table 3.3.3.1-1.
C. Incorrect – With one channel inoperable, Technical Specification 3.3.3.1 Condition A entry is required. The associated Completion Time is 30 days. Plausible because Technical Specification 3.3.3.1 Condition C has a 7 day Completion Time.
D. Incorrect – With one channel inoperable, Technical Specification 3.3.3.1 Condition A entry is required. The associated Completion Time is 30 days. Plausible because Technical Specification 3.3.3.1 Condition E has a 12 hour Completion Time, and this would take effect if the 30 day Completion Time from Condition A was not satisfied. Also plausible because this would be the Completion Time if Technical Specification 3.3.1.1 was entered due to inoperable Reactor pressure instrumentation to RPS.

Technical Reference(s): Technical Specification 3.3.3.1, PID-28A, N2-216000-RBO-5

Proposed references to be provided to applicants during examination: Technical Specification 3.3.1.1 w/o AVs, Technical Specification 3.3.3.1

Learning Objective: N2-216000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	234000 A2.01
	Importance Rating	3.7

Fuel Handling Equipment

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

Proposed Question: #92

A refueling outage is in progress with the following:

- The Reactor Mode Switch is in ***REFUEL***.
- A fuel bundle is latched on the Refuel Bridge main hoist.
- The Refuel Bridge is located over the center of the Reactor.
- The Refuel Bridge main hoist is in the normal-up position.

Then, the following occurs:

- Control rod 02-43 is inadvertently given a withdraw signal.
- Control rod block interlocks fail (Rod Block Interlocks #1 and #2) and control rod 02-43 withdraws to position 06.

Which one of the following describes:

- the impact on the ability to lower the fuel bundle

  and

- the allowance to override interlocks, in accordance with Technical Specifications?

A. All lowering of the fuel bundle is blocked. This interlock is allowed to be overridden since only one control rod is withdrawn.

B. All lowering of the fuel bundle is blocked. This interlock is NOT allowed to be overridden while control rod 02-43 is withdrawn.

C. The fuel bundle can be lowered to the top of the core, but then further lowering will be blocked. This interlock is allowed to be overridden since only one control rod remains withdrawn.

D. The fuel bundle can be lowered to the top of the core, but then further lowering will be blocked. This interlock is NOT allowed to be overridden while control rod 02-43 remains withdrawn.

Proposed Answer: B

Explanation: An interlock prevents lowering the main hoist while over the core with a fuel bundle with a control rod not fully inserted. It will prevent any lowering of the fuel bundle from the current position. Technical Specification 3.9.1 does allow further fuel movement into the core with the failed interlock, but only if all control rods are fully inserted and a control rod block is inserted. Therefore, there is no allowance to override the interlock while a control rod remains partially withdrawn.

A. Incorrect – There is no allowance to override the interlock while a control rod remains partially withdrawn. Plausible because Technical Specification 3.9.1 does allow further fuel movement into the core with the failed interlock, but only if all control rods are fully inserted and a control rod block is inserted.
C. Incorrect – All lowering of the fuel bundle is blocked. Plausible because reactivity would not be added to the core until the fuel was lowered well below the current position. There is no allowance to override the interlock while a control rod remains partially withdrawn. Plausible because Technical Specification 3.9.1 does allow further fuel movement into the core with the failed interlock, but only if all control rods are fully inserted and a control rod block is inserted.
D. Incorrect – All lowering of the fuel bundle is blocked. Plausible because reactivity would not be added to the core until the fuel was lowered well below the current position.

Technical Reference(s): N2-OP-39, Technical Specification 3.9.1 and associated bases

Proposed references to be provided to applicants during examination: Technical Specification 3.9.1

Learning Objective: N2-234000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(7)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	288000 2.1.23
	Importance Rating	4.4

Plant Ventilation

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

Proposed Question: #93

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

- An irradiated fuel bundle has been dropped in the Spent Fuel Pool.
- 2HVR*RE14A(B), Reactor Building Above Refueling Floor Exhaust Rad Monitors, indicate 7.2 x 10^{-5} µCi/cc and stable.
- 2HVR*RE32A(B), Reactor Building Below Refueling Floor Exhaust Rad Monitors, indicate 5.1 x 10^{-7} µCi/cc and stable.
- Reactor Building Ventilation (HVR) is in service.
- The Standby Gas Treatment (GTS) system is in standby.

Which one of the following describes (1) the status of HVR and GTS and (2) how these systems are required to be controlled?

A. (1) The systems have responded properly.
(2) No change in system alignment is required.

B. (1) The systems have responded properly.
(2) Start GTS and secure HVR.

C. (1) The systems have NOT responded properly.
(2) Start GTS and maintain HVR in operation.

D. (1) The systems have NOT responded properly.
(2) Secure HVR and maintain GTS secured to isolate the Reactor Building.

Proposed Answer: A

Explanation: The given radiation monitors will auto-start GTS and isolate HVR if they exceed 1.7 x 10-3 µCi/cc. Since all radiation monitors are below this value, HVR should still be running and GTS should still be in standby. Therefore, the systems have responded correctly. N2-SOP-39, Refuel Floor Events, must be entered due to the dropped fuel bundle. Additionally, the above refuel floor monitors show elevated readings. However, there is no requirement to start GTS and shutdown HVR based on either of these conditions.

B. Incorrect – No procedure requires GTS start or HVR isolation in this situation. Plausible because a fuel bundle has been dropped and Above Refuel Floor Radiation is elevated. Also plausible because other plants do require operator action to start GTS and isolate HVR on a dropped fuel bundle regardless of rad levels.
C. Incorrect – The systems have responded properly. Plausible because Above Refuel Floor Radiation is elevated. No procedure requires GTS start in this situation. Plausible because a fuel bundle has been dropped and Above Refuel Floor Radiation is elevated. Also plausible because other plants do require operator action to start GTS on a dropped fuel bundle regardless of rad levels.
D. Incorrect – The systems have responded properly. Plausible because Above Refuel Floor Radiation is elevated. No procedure requires HVR isolation in this situation. Plausible because a fuel bundle has been dropped and Above Refuel Floor Radiation is elevated. Also plausible because other plants do require operator action to isolate HVR on a dropped fuel bundle regardless of rad levels.

Technical Reference(s): N2-OP-61B Section B, N2-SOP-39

Proposed references to be provided to applicants during examination: None

Learning Objective: N2-261000-RBO-5, 2101-SOP39C01 EO-2

Question Source: Bank – 2014 Cert #81

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(7)

Comments:

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.7
	Importance Rating	4.7

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

Proposed Question: #94

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

- No Reactor injection sources are currently available.
- N2-EOP-C3, Steam Cooling, is being executed.
- Reactor water level is -35 inches (actual) and lowering slowly.

Then five minutes later…

- A Control Rod Drive pump is restored and is now injecting to the Reactor.
- Reactor water level is -42 inches (actual) and continuing to lower.

Which one of the following describes the status of adequate core cooling and the required action, in accordance with the Emergency Operating Procedures?

	Status of Adequate Core Cooling	Required Action
A.	Assured	Return to N2-EOP-RPV, RPV Control.
B.	Assured	Remain in N2-EOP-C3, Steam Cooling.
C.	NOT assured	Enter N2-EOP-C2, RPV Blowdown.
D.	NOT assured	Enter the Severe Accident Procedures.

Proposed Answer: C

Explanation: Initially, adequate core cooling (ACC) was assured because there was no RPV injection and Reactor water level was above -58 inches. However, once CRD began injecting, adequate core cooling is not assured between -39 and -58 inches. With an RPV injection source injecting and Reactor water not able to be restored/maintained above -39 inches, N2-EOP-C3 requires entering N2-EOP-C2, RPV Blowdown.

A. Incorrect – ACC is not currently assured. Plausible because ACC was assured until the CRD pump was injecting with Reactor water level < -39 inches.
B. Incorrect – ACC is not currently assured. Plausible because ACC was assured until the CRD pump was injecting with Reactor water level < -39 inches.
D. Incorrect – N2-EOP-C3 requires entering N2-EOP-C2, RPV Blowdown. Plausible because if core cooling cannot be restored after the Blowdown, the Severe Accident Procedures would then be entered from N2-EOP-RPV.

Technical Reference(s): N2-EOP-C3, NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC3C01 EO-2

Question Source: Bank - NMP2 2014 NRC #99

Question History: NMP2 2014 NRC #99

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.42
	Importance Rating	3.4

Knowledge of new and spent fuel movement procedures.

Proposed Question: #95

The plant is in Mode 5 with the following:

- Core shuffle is in progress.
- A four cell section of the core is ***displayed on the following page***.

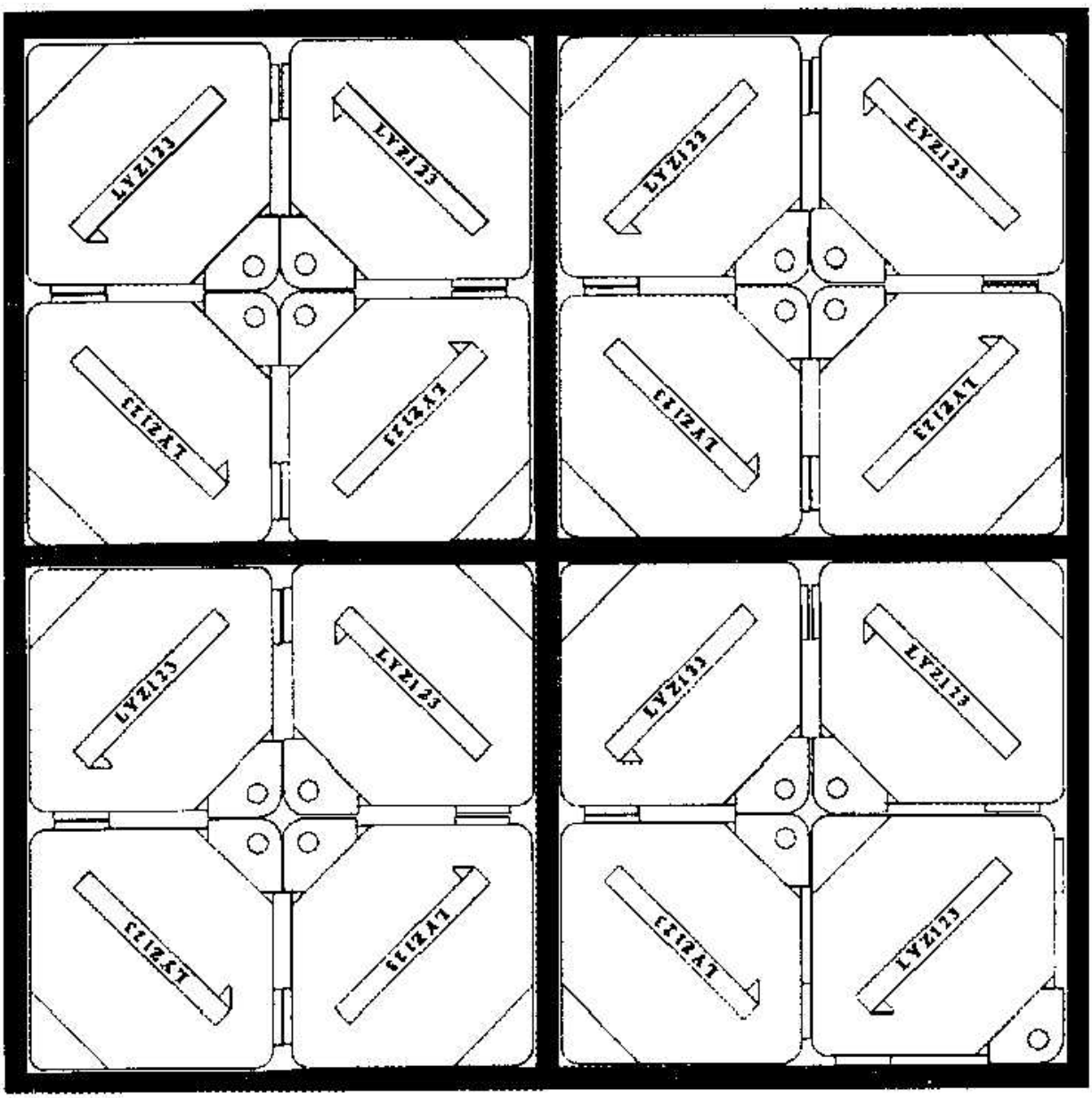
Which one of the following describes the status of this portion of the core and the required action(s), if any, in accordance with N2-FHP-13.3, Core Shuffle?

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 SRO.

D. A discrepancy exists in the fuel loading. Fuel movements must be immediately stopped. Completion of a formal evaluation is required before fuel movement may resume.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Proposed Answer: D

Explanation: The fuel assembly in the lower right corner is oriented incorrectly, as evidenced by the bail handle indicator pointing away from the center of the fuel cell. N2-FHP-13.3 section 4.1.4 defines this as a Fuel Movement Discrepancy (FMD). N2-FHP-13.3 section 4.3.2 requires fuel movement to be stopped and requires completion of a formal evaluation, via Attachments 3 and 4, prior to resuming fuel movements.

A. Incorrect – There is a discrepancy in the fuel loading. Plausible because 15 of the 16 assemblies are loaded correctly and the overall shape is symmetrical.
B. Incorrect – Core shuffle cannot continue with the given error. Plausible because no significant adverse consequence would be expected from this fuel movement discrepancy until Reactor operation recommenced.
C. Incorrect – The discrepancy cannot be immediately fixed without a formal evaluation. Plausible because the issue would be fairly simple to fix.

Technical Reference(s): N2-FHP-13.3

Proposed references to be provided to applicants during examination: None

Learning Objective: S-101002-RBO-9

Question Source: Bank - 2015 NRC #98

Question History: 2015 NRC #98

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(6)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.43
	Importance Rating	3.3

Knowledge of the process used to track inoperable alarms.

Proposed Question: #96

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

- Annunciator 601537, ADS Valves/Safety Valves Leaking, has been received multiple times.
- The cause of the alarm has been determined to be a malfunctioning SRV tailpipe temperature detector.
- It is desired to disable the faulty annunciator input.

Which one of the following describes:

(1) the proper tag to be used

and

(2) where the disabled alarm shall be noted

in accordance with S-ARP-001, Disabling Nuisance Annunciators?

	(1) Tag to Be Used on Annunciator	(2) Where Disabled Alarm Shall Be Noted
A.	Caution Tag	Control Room log
B.	Caution Tag	Operator work around log
C.	Temporary Configuration Change Tag	Control Room log
D.	Temporary Configuration Change Tag	Operator work around log

Proposed Answer: C

Explanation: S-ARP-001 provides guidance to eliminate the source of a nuisance alarm. The process is carried out by OP-AA-108-101 and utilizes a Temporary Configuration Change Tag for the annunciator. Additionally, S-ARP-001 requires noting the disabled annunciator in the Control Room log.

A. Incorrect – A Temporary Configuration Change Tag is used for the annunciator. Plausible because Caution Tags are commonly used on the Control Room boards for off-normal controls or indications.
B. Incorrect – A Temporary Configuration Change Tag is used for the annunciator. Plausible because Caution Tags are commonly used on the Control Room boards for off-normal controls or indications. S-ARP-001 requires noting the disabled annunciator in the Control Room log. Plausible because this is another tracking mechanism that is used for degraded plant conditions that challenge operators.
D. Incorrect – S-ARP-001 requires noting the disabled annunciator in the Control Room log. Plausible because this is another tracking mechanism that is used for degraded plant conditions that challenge operators.

Technical Reference(s): S-ARP-001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank – JAF 17-2 NRC #42

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 2.2.43
	Importance Rating	3.0

SRVs

Knowledge of the process used to track inoperable alarms

Proposed Question: #42

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

- Annunciator 09-4-1-16, SRV Leaking, has been received multiple times.
- The cause of the alarm has been determined to be a malfunctioning SRV tailpipe temperature detector.
- The Unit Supervisor has directed disabling the faulty annunciator input using the T-mod process.

Which one of the following describes the proper tag to be used on this annunciator, in accordance with OP-AA-103-102, Watch-Standing Practices, and OP-AA-108-101, Control of Equipment and System Status?

A. Caution Tag

B. Equipment Status Tag

C. Special Condition Tag

D. Temporary Configuration Change Tag

Examination Outline Cross-Reference:		
	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.36
	Importance Rating	4.2

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

Proposed Question: #97

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

Time	Condition
0800	• Standby Diesel Generator 2EGS*EG3 is removed from service for maintenance. • 2EGS*EG3 is scheduled to be returned to service in 48 hours.
1000	• Annunciator 601701, DIVISION I STBY LIQ CONT SYSTEM INOPERABLE, alarms. • Investigation reveals a blown fuse in the control circuit for 2EHS*MCC102 Bkr.17D, 2SLS*MOV1A, SLC STORAGE TK OUTLET VALVE, Power Supply.

Which one of the following describes the operability of the Division II Standby Liquid Control subsystem, in accordance with Technical Specifications?

The Division II Standby Liquid Control subsystem…

A. must be declared inoperable at time 0800.

B. must be declared inoperable at time 1000.

C. may be considered operable up to time 1400, but then must be declared inoperable.

D. may be considered operable for the duration of the maintenance on 2EGS*EG3.

Proposed Answer: C

Explanation: The Division II Standby Liquid Control subsystem is supported by 2EGS*EG3. Technical Specification LCO 3.0.6 does not require declaring the supported system inoperable when the support system is made inoperable (Time 0800). At time 1000, the redundant Standby Liquid Control subsystem becomes inoperable. Technical Specification 3.8.1 Required Action B.2 gives 4 hours from this time before the Division II Standby Liquid Control subsystem must be declared inoperable. Therefore, the Division II Standby Liquid Control subsystem becomes inoperable at Time 1400.

A. Incorrect – The Division II Standby Liquid Control subsystem becomes inoperable at Time 1400. Plausible because this is the time when the support system becomes inoperable.
B. Incorrect – The Division II Standby Liquid Control subsystem becomes inoperable at Time 1400. Plausible because this is the time when the redundant system becomes inoperable.
D. Incorrect – The Division II Standby Liquid Control subsystem becomes inoperable at Time 1400. Plausible because LCO 3.0.6 normally does not require declaring a supported system inoperable based on an inoperable support system.

Technical Reference(s): Technical Specifications 3.0.6, 3.1.7, and 3.8.1

Proposed references to be provided to applicants during examination: Technical Specifications 3.1.7 and 3.8.1 without bases

Learning Objective: N2-211000-RBO-14

Question Source: Modified Bank - 2010 Cert SRO #14

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

	Importance Rating		4.6

SLC: Equipment Control: Ability to recognize system parameters that are entry level conditions for Technical Specifications.

Question:	SRO #14

The plant is at rated conditions on day one of an LCO due to scheduled maintenance on Standby Diesel 2EGS*EG3. The diesel is scheduled to be returned to service in 48 hours.

With this condition, alarm DIVISION I STBY LIQ CONT SYSTEM INOPERABLE, 601701, annunciates. The alarm is determined to be due to a blown fuse in the control circuit for 2EHS*MCC102 Bkr.17D, 2SLS*MOV1A, SLC STORAGE TK OUTLET VALVE, Power Supply.

The SRO is required to declare Division I Standby Liquid Control System inoperable AND

A.		If the diesel LCO is exited as scheduled, the plant can continue to operate for 7 days while the subsystem is being repaired.
B.		Commence a reactor plant shutdown and be in mode 3 within the next 12 hours.
C.		If not restored within the next 8 hours commence a reactor plant shutdown and be in mode 3 within 12 hours.
D.		Declare Division II Standby Liquid Control subsystem inoperable if not restored within the next 4 hours.

Proposed Answer:	D

Explanation (Optional):

A.		Incorrect: Division II SLC must also be declared inoperable due to its diesel being out of service. Plausible in that this would be true if the other subsystem was not affected by the diesel being out of service.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.6
	Importance Rating	3.8

Ability to approve release permits.

Proposed Question: #98

A loss of coolant accident has resulted in the following:

- An RPV Blowdown has been performed due to exceeding the Pressure Suppression Pressure.
- Reactor pressure is 75 psig and stable.
- Reactor water level is being controlled 160" to 200" using Condensate.
- RHR and LPCS are available for injection.
- Drywell pressure is 46 psig and slowly rising.
- Suppression Chamber pressure is 44 psig and slowly rising.
- Suppression Pool water level is 210 feet and slowly rising.
- Chemistry reports that venting of the Primary Containment will exceed release rate limits.

Note: The Primary Containment Pressure Limit (PCPL) curve is ***provided on the next page***.

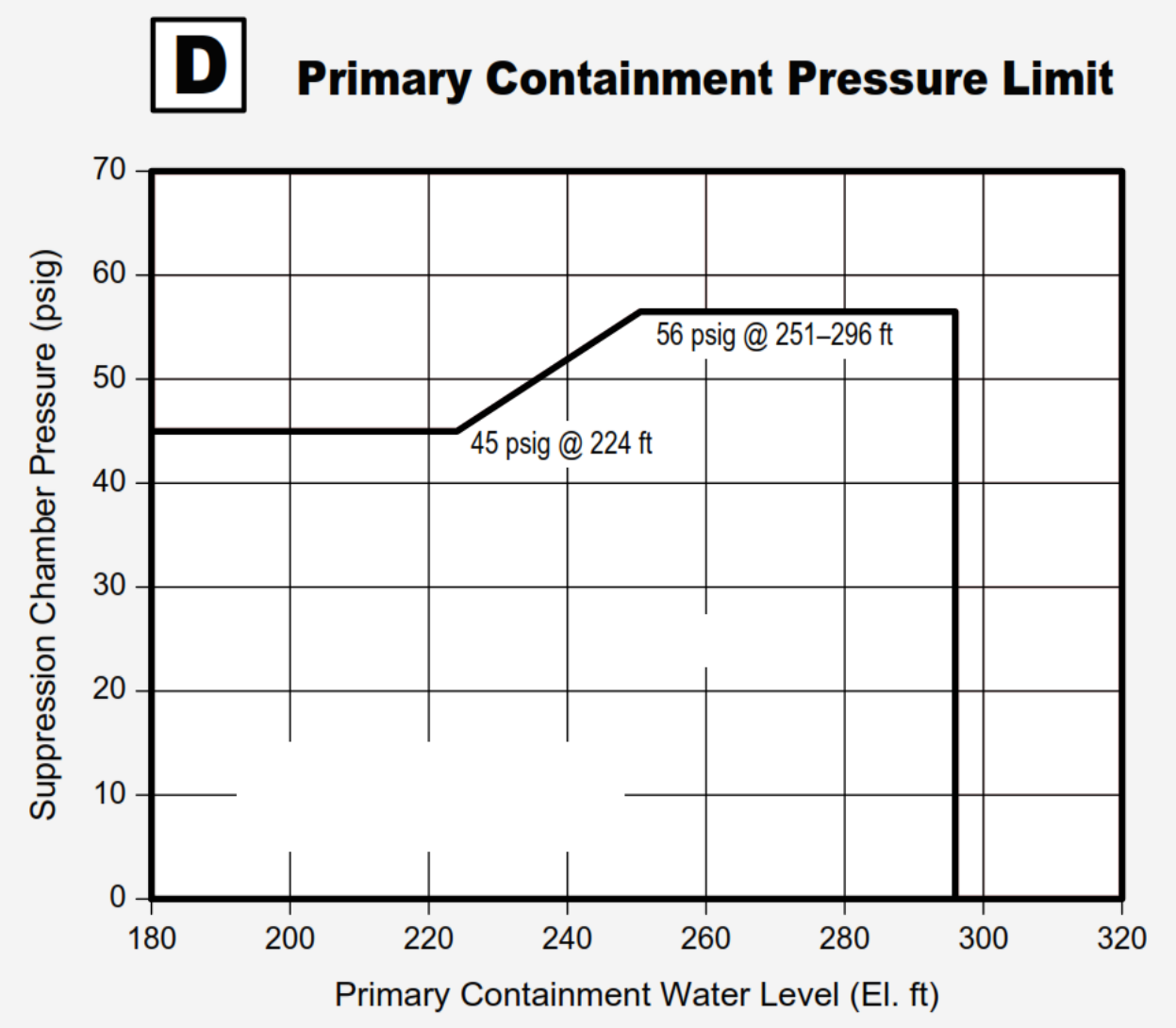
Which one of the following describes the required control of Condensate injection and Primary Containment venting, in accordance with the Emergency Operating Procedures?

A. Continue injection with Condensate. Direct Primary Containment venting.

B. Continue injection with Condensate. Do NOT direct Primary Containment venting.

C. Transition injection from Condensate to another system. Direct Primary Containment venting.

D. Transition injection from Condensate to another system. Do NOT direct Primary Containment venting.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Primary Containment Pressure Limit
70
60
50
40
30
20
10
0
Suppression Chamber Pressure (psig)
56 psig @ 251–296 ft
45 psig @ 224 ft
180
200
220
240
260
280
300
320
Primary Containment Water Level (El. ft)

Proposed Answer: C

Explanation: The given values of Suppression Chamber pressure and Suppression Pool water level show that the Primary Containment Pressure Limit (PCPL) is about to be exceeded. N2-EOP-RPV requires stopping injection from sources outside the Primary Containment not needed for core cooling if you cannot stay below PCPL. Reactor water level is well above top of active fuel, other systems are available for injection, and Condensate injects water from outside the Primary Containment. Therefore, injection must be transitioned from Condensate to another system. N2-EOP-PC requires venting the Primary Containment even if release rate limits will be exceeded.

Note: This question meets the K/A by testing the candidate's knowledge of when to approve a release through Primary Containment venting in excess of normal release rate limits.

A. Incorrect – Condensate injection must be terminated because it injects from outside the Primary Containment and PCPL is being challenged. Plausible because PCPL has not yet been exceeded and the PCPL curve actually rises as SP level rises.
B. Incorrect – Condensate injection must be terminated because it injects from outside the Primary Containment and PCPL is being challenged. Plausible because PCPL has not yet been exceeded and the PCPL curve actually rises as SP level rises. Primary Containment venting is required even though release rates will be exceeded because PCPL is being challenged. Plausible because PCPL has not yet been exceeded and release rate limits will be violated.
D. Incorrect – Primary Containment venting is required even though release rates will be exceeded because PCPL is being challenged. Plausible because PCPL has not yet been exceeded and release rate limits will be violated.

Technical Reference(s): N2-EOP-RPV, N2-EOP-PC

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPPCC01 EO-2

Question Source: Bank – 2017 Cert #96

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.21
	Importance Rating	4.6

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

Proposed Question: #99

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

- A Feedwater Level Control malfunction causes Reactor water level to lower.
- An Operator manually scrams the Reactor in accordance with N2-SOP-101C, Reactor Scram.
- Ten (10) control rods stick at position 02.
- One (1) control rods sticks at position 24.
- All other control rods fully insert.
- Reactor water level reaches a low of 110” and then begins to rise.
- APRMs are downscale.
- IRMs are being inserted.

Which one of the following describes the required strategies for Reactor water level and pressure control, in accordance with the Emergency Operating Procedures?

	Reactor Water Level Control	Reactor Pressure Control
A.	Reactor water level may be restored and maintained 159.3” to 202.3”	Reactor cooldown is currently allowed
B.	Reactor water level may be restored and maintained 159.3” to 202.3”	Reactor cooldown is NOT currently allowed
C.	Reactor water level must be lowered to at least 100”	Reactor cooldown is currently allowed
D.	Reactor water level must be lowered to at least 100”	Reactor cooldown is NOT currently allowed

Proposed Answer: B

Explanation: Upon entry to N2-EOP-RPV, diagnostic questions related to the success of the scram determine which EOP mitigation strategy will be followed. If the Reactor will stay shutdown under all conditions without boron, N2-EOP-RPV strategies are used and N2-EOP-C5 is NOT entered. In this case, the Reactor is NOT guaranteed to stay shutdown under all conditions without boron based on multiple rods stuck out. The Reactor water level control strategy in N2-EOP-C5 is then contingent upon Reactor power. With APRMs downscale (<4%), Reactor water level control in a band of -39-202.3" is allowed. Therefore, Reactor water level may be raised to a band of 159.3-202.3", and does not have to be lowered to below 100". Reactor pressure control is contingent upon determination if the Reactor is currently shutdown. This determination can be made by all IRMs being fully inserted and on range 6 or below. With IRMs not yet fully inserted, this determination cannot be made, and Reactor cooldown is NOT yet allowed.

A. Incorrect – Reactor cooldown is NOT yet allowed. Plausible because once IRMs are fully inserted, power may be below Range 6, which would then allow cooldown.
C. Incorrect – Reactor water level does not need to be lowered. Plausible because this would be correct if APRMs were above 4%. Reactor cooldown is NOT yet allowed. Plausible because once IRMs are fully inserted, power may be below Range 6, which would then allow cooldown.
D. Incorrect – Reactor water level does not need to be lowered. Plausible because this would be correct if APRMs were above 4%.

Technical Reference(s): N2-EOP-RPV, N2-EOP-C5, NER-2M-039

Proposed references to be provided to applicants during examination: None

Learning Objective: 2101-EOPC5C01 EO-2

Question Source: Bank - NMP1 2015 Cert #80

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

ES-401	Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.37
	Importance Rating	4.1

Knowledge of the lines of authority during implementation of the emergency plan.

Proposed Question: #100

An accident has occurred with the following:

- A General Emergency has been declared.
- Protective Action Recommendations (PARs) were initially made from Main Control Room.
- All Emergency Response Facilities are now fully staffed and have assumed all duties.
- Wind direction has changed enough to require new PARs.

Which one of the following identifies the facility that is required to make the new PAR notification to the State and County?

A. Main Control Room (MCR)

B. Technical Support Center (TSC)

C. Operations Support Center (OSC)

D. Emergency Operations Facility (EOF)

Proposed Answer: D

Explanation: The Main Control Room initially has responsibility for determining PARs and making the associated notifications to the State and County. Once all Emergency Response Facilities are activated, this authority transfers to personnel in the Emergency Operations Facility (EOF).

A. Incorrect – The authority for determining PARs and making the associated notification is from the EOF. Plausible because this authority rests in the Main Control Room initially.
B. Incorrect – The authority for determining PARs and making the associated notification is from the EOF. Plausible because the TSC has many other roles and responsibilities once activated.
C. Incorrect – The authority for determining PARs and making the associated notification is from the EOF. Plausible because the OSC has many other roles and responsibilities once activated.

Technical Reference(s): EP-AA-111, EP-AA-114

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(4)

Comments: