Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 A2.08
	Importance Rating	3.3

Shutdown Cooling

Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of heat exchanger cooling

Question: #1

A plant shutdown is in progress with the following:

- Shutdown Cooling (SDC) loop 12 is in service.
- Reactor water level is 90”.
- Reactor coolant temperature is 340^{o}F.
- Reactor pressure is 115 psig.

Then, a spurious valve closure results in a loss of all Reactor Building Closed Loop Cooling (RBCLC) flow to the SDC heat exchangers.

Which one of the following describes how SDC will respond and the subsequent required control of Reactor Water Cleanup (RWCU) flow, in accordance with N1-SOP-6.1, Loss of Rx Cavity/Decay Heat Removal?

	SDC Response	Control of RWCU Flow
A.	SDC isolation valves (38-01, 38-02, 38-13) close and trip the SDC pump	Maximize
B.	SDC isolation valves (38-01, 38-02, 38-13) close and trip the SDC pump	Minimize
C.	SDC pump trips and SDC isolation valves (38-01, 38-02, 38-13) remain open	Maximize
D.	SDC pump trips and SDC isolation valves (38-01, 38-02, 38-13) remain open	Minimize

Proposed Answer: C

Explanation: With loss of RBCLC flow to the SDC heat exchangers, Reactor coolant temperature and pressure rise. The SDC pumps trip when Reactor coolant temperature reaches 350°F. The isolation valves remain open. N1-SOP-6.1 requires maximizing RWCU flow to provide additional decay heat removal.

A. Incorrect – The isolation valves remain open. Plausible because these valves cannot be opened until Reactor pressure is below 120 psig, but there is no isolation signal in the opposite direction.
B. Incorrect – The isolation valves remain open. Plausible because these valves cannot be opened until Reactor pressure is below 120 psig, but there is no isolation signal in the opposite direction. N1-SOP-6.1 requires maximizing RWCU flow to provide additional decay heat removal. Plausible that RWCU flow would be lowered to prevent either thermal stratification or high NRHX outlet temperatures (with Reactor coolant temperature rising).
D. Incorrect – N1-SOP-6.1 requires maximizing RWCU flow to provide additional decay heat removal. Plausible that RWCU flow would be lowered to prevent either thermal stratification or high NRHX outlet temperatures (with Reactor coolant temperature rising).

Technical Reference(s): N1-OP-4, N1-SOP-6.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-5

Question Source: Modified Bank – 2008 NRC #9

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) : Heat removal mechanisms	
Proposed Question:	Common 9

The plant is in Hot Shutdown with Shutdown Cooling (SDC) Loop 12 in operation. The following conditions exist:

- RPV water level is 90 inches
- RPV coolant temperature is 347°F
- RPV pressure is 117 psig

Which one of the following describes how SDC will respond to a sustained loss of Reactor Building Closed Loop Cooling (RBCLC)?

A. SDC System isolation valves (38-01, 38-02, 38-13) close on high pressure and trip the SDC Pump.

B. SDC System isolation valves (38-01, 38-02, 38-13) close on high temperature and trip the SDC Pump.

C. SDC Pump trips on high temperature and SDC System isolation valves (38-01, 38-02, 38-13) remain open.

D. SDC Pump trips on high pressure and SDC System isolation valves (38-01, 38-02, 38-13) remain open.

Proposed Answer:	C.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 A3.05
	Importance Rating	4.3

High Pressure Coolant Injection

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

Question: #2

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

- Feedwater flow control valve 13 fails closed.
- The Reactor is manually scrammed.
- HPCI initiates.
- Reactor water level is 68" and rising.

Which one of the following describes the status of injection from Feedwater pumps 11 and 12 to the Reactor?

A. NEITHER Feedwater pump is injecting

B. Only Feedwater pump 11 is injecting

C. Only Feedwater pump 12 is injecting

D. Both Feedwater pumps are injecting

Proposed Answer: C

Explanation: While operating in the HPCI mode, Feedwater pump 11 will attempt to control Reactor water level at 65" and Feedwater pump 12 will attempt to control Reactor water level at 72". With Reactor water level at 68", Feedwater pump 11 will not be injecting and Feedwater pump 12 will be injecting.

A. Incorrect – Feedwater pump 12 is injecting. Plausible because this would be correct if Reactor water level was above 72".
B. Incorrect – Feedwater pump 11 is not injecting and Feedwater pump 12 is injecting. Plausible because this would be the choice if pump operating characteristics were confused and reversed.
D. Incorrect – Feedwater pump 11 is not injecting. Plausible because this would be correct if Reactor water level was below 65".

Technical Reference(s): N1-OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-5

Question Source: Bank - 2009 Cert #17

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 #	207000 A4.04
	Importance Rating	3.8

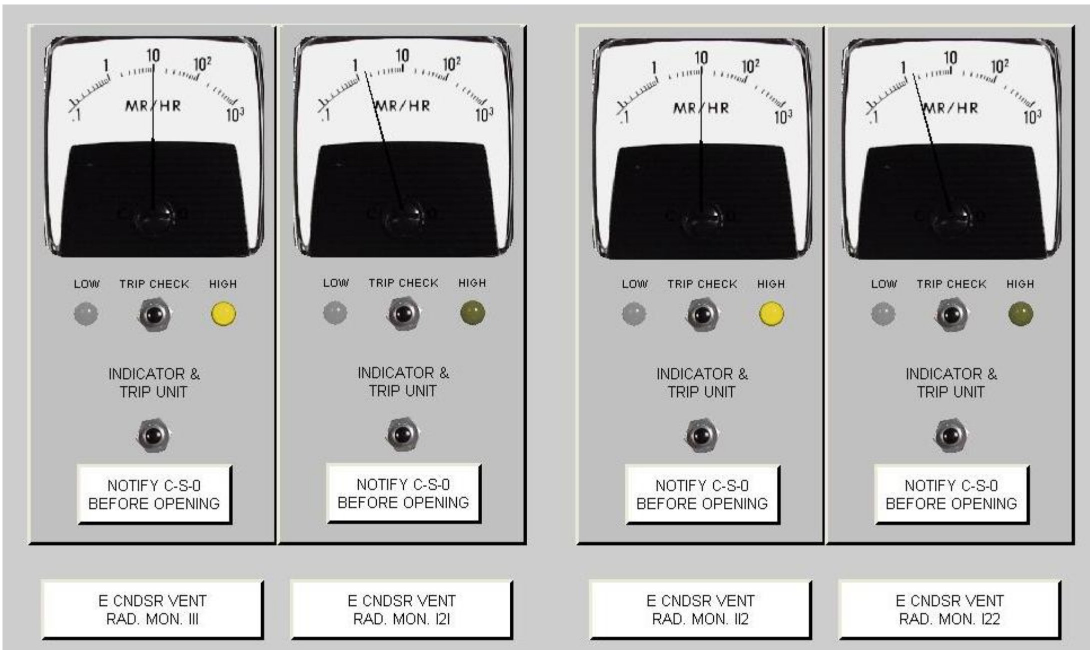
Isolation (Emergency) Condenser

Ability to manually operate and/or monitor in the control room: Vent line radiation levels: BWR-2,3

Question: #3

The plant has scrammed from 100% power with the following:

- Emergency Condenser (EC) 11 has been placed in service.
- EC 12 remains in standby.
- Annunciator K1-1-2, EMER COND VENT 11 RAD MONITOR, is in alarm.
- The following indications are present for the EC vent radiation monitors:



Which one of the following describes the status of these indications, in accordance with ARP K1-1-2?

These indications are…

A. normal for an in-service EC. The alarm can currently be reset.

B. normal for an in-service EC. The alarm CANNOT currently be reset.

C. NOT normal for an in-service EC. The EC 11 vent radiation monitors should be reading lower.

D. NOT normal for an in-service EC. The EC 11 vent radiation monitors should be reading higher.

Proposed Answer: C

Explanation: The given indications show two EC vent radiation monitors in alarm high (111 and 112 – both associated with EC 11) and two EC vent radiation monitors slightly elevated above normal readings but below the high alarm setpoint (121 and 122 – both associated with EC 12). ARP K1-1-2 provides specific guidance on how to interpret EC vent radiation monitors alarms. With both EC 11 vent radiation monitors (111 and 112) in alarm high, an EC tube leak is suspected. These are NOT normal indications for an in-service EC.

A. Incorrect – These are NOT normal indications for an in-service EC. Plausible that an in-service EC would have slightly elevated rad levels given Reactor coolant is flowing through the tubes.
B. Incorrect – These are NOT normal indications for an in-service EC. Plausible that an in-service EC would have slightly elevated rad levels given Reactor coolant is flowing through the tubes. The alarm cannot currently be reset because rad levels are still above the high alarm setpoint of 5 mr/hr.
D. Incorrect – EC 11 radiation monitors are indicating higher than normal. Plausible because they are reading lower than normal as some rise in indication is expected for an in-service EC.

Technical Reference(s): ARP K1-1-2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-10

Question Source: Modified Bank – 2015 NRC #87

Question History:

Question Cognitive Level: Comprehension or Analysis

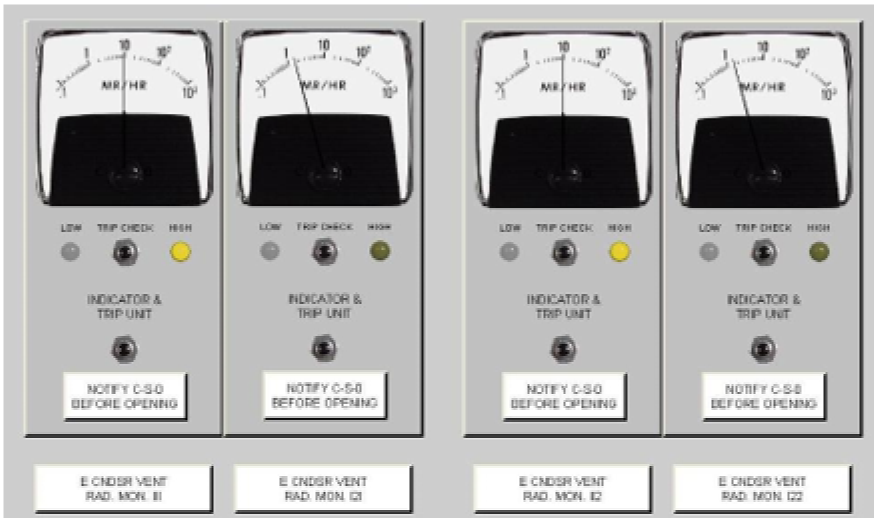
10 CFR Part 55 Content: 55.41(8)

Comments:

Proposed Question: #87

The plant has scrammed from 100% power with the following:

- The MSIVs are closed and CANNOT be re-opened.
- Emergency Condenser (EC) 12 is out of service for maintenance.
- Emergency Condenser 11 has been placed in service.
- Annunciator K1-1-2, EMER COND VENT 11 RAD MONITOR, is in alarm.
- The following indications are present for the EC vent radiation monitors:



Which one of the following describes the impact of these indications on operation of EC 11, in accordance with ARP K1-1-2?

A. These are valid indications of an EC 11 tube leak. Direct isolating EC 11.

B. One or more indications have failed. Direct RP to monitor dose rates at EC piping.

C. These indications are expected for an in service EC. Direct continued use of EC 11 unless vent radiation levels exceed a threshold of 30 mr/hr.

D. These indications are higher than expected for an in service EC, but do NOT validate an EC 11 tube leak. Direct Chemistry to obtain an EC 11 shell side sample.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 2.4.50
	Importance Rating	4.2

Low Pressure Core Spray

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

Question: #4

The plant is operating at 100% power when Annunciator K3-4-1, CORE SPRAY SPARGER DIFF PRESS, alarms.

Which one of the following identifies the location(s) for verifying the associated differential pressure indications and a normal value for this parameter?

The associated differential pressure indications are verified in the…

A. North Instrument Room and normally indicate a positive value.

B. North Instrument Room and normally indicate a negative value.

C. Reactor Building SW and SE Corner Rooms and normally indicate a positive value.

D. Reactor Building SW and SE Corner Rooms and normally indicate a negative value.

Proposed Answer: B

Explanation: The Core Spray sparger D/P indications are located in the North Instrument Rooms, per N1-ARP-K3. These indicators normally show a negative value when at power because they're calibrated to indicate no d/p when the reactor is in cold shutdown conditions. This results in a negative pressure indication with the plant at power because the pressure inside the core spray sparger line is greater than the pressure above the core plate, with respect to cold shutdown conditions.

A. Incorrect – These indicators normally show a negative value. Plausible because most plant indications normally indicate a positive value.
C. Incorrect – The Core Spray sparger D/P indications are located in the North Instrument Rooms. Plausible because the Core Spray pumps are in the Reactor Building SW and SE Corner Rooms. These indicators normally show a negative value. Plausible because most plant indications normally indicate a positive value.
D. Incorrect – The Core Spray sparger D/P indications are located in the North Instrument Rooms. Plausible because the Core Spray pumps are in the Reactor Building SW and SE Corner Rooms.

Technical Reference(s): ARP K3-4-1, N1-209001-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-209001-RBO-5

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 #	211000 K1.06
	Importance Rating	3.7

Standby Liquid Control

Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Reactor vessel

Question: #5

Which one of the following describes where the Liquid Poison system injects into the Reactor vessel?

A. Through a Feedwater line

B. Through the Core Spray nozzles

C. Into a dedicated sparger located below the core plate

D. Into a dedicated sparger located above the core plate

Proposed Answer: C

Explanation: The Liquid Poison system injects through a dedicated sparger that is located below the core support plate.

A. Incorrect – The Liquid Poison system injects through a dedicated sparger that is located below the core plate. Plausible because the boron solution could be injected through Feedwater into the annulus region and then forced through the core by Recirc pumps. Such mixing is necessary to ensure proper boron concentration in the core and is the basis behind raising Reactor water level in N1-EOP-3.
B. Incorrect – The Liquid Poison system injects through a dedicated sparger that is located below the core plate. Plausible because this would get boron into the Reactor and there is an interrelationship between Liquid Poison and Core Spray through the pipe-within-a-pipe arrangement.
D. Incorrect – The Liquid Poison system injects through a dedicated sparger that is located below the core plate. Plausible because part of the pipe-within-a-pipe arrangement communicates with a location above the core plate.

Technical Reference(s): N1-OP-12

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-211000-RBO-2

Question Source: Bank – JAF 16-1 NRC #7

Question History: JAF 16-1 NRC #7

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 K2.01
	Importance Rating	3.2

Reactor Protection

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

Question: #6

Which one of the following describes the 600VAC power source that normally supplies power to the motor generator set associated with Reactor Protection System (RPS) channel 11?

A. Powerboard 16

B. Powerboard 17

C. Powerboard 131

D. Powerboard 141

Proposed Answer: C

Explanation: The motor generator set associated with Reactor Protection System (RPS) channel 11 is MG Set 131. This MG is powered by PB 131.

A. Incorrect – The motor generator set associated with Reactor Protection System (RPS) channel 11 is powered by PB 131. Plausible because PB 16 supplies power to RPS 11's UPS 162.
B. Incorrect – The motor generator set associated with Reactor Protection System (RPS) channel 11 is powered by PB 131. Plausible because PB 17 supplies power to RPS 12's UPS 172.
D. Incorrect – The motor generator set associated with Reactor Protection System (RPS) channel 11 is powered by PB 131. Plausible because PB 141 supplies power to RPS 12's motor generator set.

Technical Reference(s): N1-OP-40, C-19409-C, C-19859-C Sh4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-4

Question Source: Modified Bank – 2015 NRC #4

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

RPS

Knowledge of electrical power supplies to the following: Analog trip system logic cabinets

Proposed Question: #4

Which one of the following describes the 600VAC power source that normally supplies power for the Reactor Protection System (RPS) Analog Trip System (ATS) cabinet D?

A. Powerboard 16

B. Powerboard 17

C. Powerboard 131

D. Powerboard 141

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 K3.02
	Importance Rating	3.6

Intermediate Range Monitor

Knowledge of the effect that a loss or malfunction of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM will have on following: Reactor manual control

Question: #7

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- All Intermediate Range Monitors (IRM) detectors are fully inserted and indicating mid-scale on range 2.

Then, a shorted contact causes IRM 11 detector to withdraw.

Which one of the following describes the resulting status of control rod blocks?

A. NO rod block is received.

B. A rod block is received as soon as IRM 11 leaves the fully inserted position.

C. A rod block is first received when IRM 11 exits the core region.

D. A rod block is first received when IRM 11 indication lowers to below 7.5/125 of scale.

Proposed Answer: B

Explanation: Per N1-OP-38B, Section B, a position switch on the detector retraction mechanism blocks control rod withdrawal with the Reactor Mode Switch in STARTUP unless the detectors are inserted to the startup position. Per F.1, all detectors must be fully inserted to perform a startup. Therefore, IRM 11 causes a rod block as soon as the detector begins withdrawing.

A. Incorrect – With the Reactor Mode Switch in STARTUP, IRM 11 causes a rod block as soon as the detector begins withdrawing. Plausible that only a single IRM would not cause a rod block.
C. Incorrect – The rod block is received earlier, as soon as the IRM leaves the fully inserted position. Plausible that as long as it remains in the core region it would satisfy interlocks, such as SRMs are allowed to be partially withdrawn.
D. Incorrect – The rod block is received earlier, as soon as the IRM leaves the fully inserted position. Plausible that no block would be received until it lowers below the downscale setpoint.

Technical Reference(s): N1-OP-38B, ARP F2-3-6

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: Bank – JAF 12-2 NRC #37

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 #	215004 K4.01
	Importance Rating	3.7

Source Range Monitor

Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Rod withdrawal blocks

Question: #8

Which one of the following describes conditions in which Source Range Monitor (SRM) rod blocks are bypassed?

The SRM downscale rod block is bypassed when ___(1)___. The SRM upscale and downscale rod blocks are bypassed when ___(2)___.

	(1)	(2)
A.	the SRM is fully inserted	the SRM is fully withdrawn
B.	the SRM is fully inserted	all associated IRMs are on Range 8 or above
C.	all associated IRMs are on Range 2	the SRM is fully withdrawn
D.	all associated IRMs are on Range 2	all associated IRMs are on Range 8 or above

Proposed Answer: B

Explanation: Per N1-OP-38A, one SRM trip function is a Downscale intermediate trip. This is a count-rate level interlock to bypass a position switch on the detector retraction mechanism which blocks control rod withdrawal in the START-UP mode unless the detectors are inserted to the startup position. The bypass permits control rod withdrawal with the detector withdrawn as long as a count rate above 100 counts per second is maintained. Therefore, SRM downscale rod blocks are bypassed when the SRM is fully inserted to allow startup with less than 100 cps. N1-OP-43A, E.2.18, directs fully withdrawing SRMs when IRMs are on range 8 or above. This is because all SRM rod blocks are bypassed once the associated IRM are on Range 8 or above.

A. Incorrect – All associated IRMs are on Range 8 or above bypasses all SRM rod blocks. Plausible that SRM detector position would input to the bypass circuitry, such that once the SRM is fully out of the core, it could no longer cause spurious signals.
C. Incorrect – The SRM being fully inserted bypasses the downscale rod blocks. Plausible because IRM detector position does input to other bypasses. All associated IRMs are on Range 8 or above bypasses all SRM rod blocks. Plausible that SRM detector position would input to the bypass circuitry, such that once the SRM is fully out of the core, it could no longer cause spurious signals.
D. Incorrect – The SRM being fully inserted bypasses the downscale rod blocks. Plausible because IRM detector position does input to other bypasses.

Technical Reference(s): N1-OP-38A, N1-OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: Bank – 2013 NRC #36

Question History: 2013 NRC #36

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 #	215005 K5.01
	Importance Rating	2.8

Average Power Range Monitor/Local Power Range Monitor

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

Question: #9

Which one of the following describes the method used to adjust APRM readings to offset the effects of LPRM detector aging and the required range for APRM readings relative to core thermal power, in accordance with Reactor Engineering Procedures?

Note: Assume FPAPDR is less than or equal to 1.

__(1)__ is adjusted to achieve a reading that is __(2)__ of core thermal power.

	(1)	(2)
A.	Flux amplifier gain	+2%, -0%
B.	Flux amplifier gain	+0%, -2%
C.	Power supply voltage	+2%, -0%
D.	Power supply voltage	+0%, -2%

Proposed Answer: A

Explanation: APRMs are adjusted to offset the effects of LPRM aging by changing the flux amplifier gain. N1-REP-12 requires the APRM reading to be adjusted to +2.0%, -0.0% of core thermal power.

B. Incorrect – N1-REP-12 requires the APRM reading to be adjusted to +2.0%, -0.0% of core thermal power. Plausible because this is very similar to the requirement with just the +/- switched and understanding which range is more/less conservative requires understanding of the system.
C. Incorrect – APRMs are adjusted to offset the effects of LPRM aging by changing the flux amplifier gain. Plausible because changing power supply voltage would also affect instrument indication.
D. Incorrect – APRMs are adjusted to offset the effects of LPRM aging by changing the flux amplifier gain. Plausible because changing power supply voltage would also affect instrument indication. N1-REP-12 requires the APRM reading to be adjusted to +2.0%, -0.0% of core thermal power. Plausible because this is very similar to the requirement with just the +/- switched and understanding which range is more/less conservative requires understanding of the system..

Technical Reference(s): N1-REP-12

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-2

Question Source: Bank – 2017 NRC #8

Question History: 2017 NRC #8

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	218000 K6.06
	Importance Rating	3.4

Automatic Depressurization

Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM: D.C. power: Plant-Specific

Question: #10

Which one of the following describes the effect a loss of 125 VDC will have on the Electromatic Relief Valves (ERVs)?

A. The F-panel control, Reactor over-pressure protection, and ADS function becomes inoperable.

B. The ADS function and Reactor over-pressure protection becomes inoperable, but the F-panel control remains operable.

C. The ADS function and F-panel control becomes inoperable, but the Reactor over-pressure protection remains operable.

D. The F-panel control and Reactor over-pressure protection becomes inoperable, but the ADS function remains operable.

Proposed Answer: A

Explanation: The loss of 125 VDC power makes the F-panel control, Reactor over-pressure protection, and ADS function become inoperable because, per N1-OP-47A, Attachments 7 and 8, power to the ERV solenoid is lost (Circuits B03 and C03). All three of these functions require operation of the same DC powered solenoid.

B. Incorrect – All three functions are lost. Plausible that different power supplies would support the different functions.
C. Incorrect – All three functions are lost. Plausible that different power supplies would support the different functions.
D. Incorrect – All three functions are lost. Plausible that different power supplies would support the different functions.

Technical Reference(s): N1-OP-47A Attachment 7, 8, C-19859-C Sh 24

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-4

Question Source: Bank – 2008 NRC #4

Question History: 2008 NRC #4

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 #	223002 A1.04
	Importance Rating	2.6

Primary Containment Isolation/Nuclear Steam Supply Shutoff

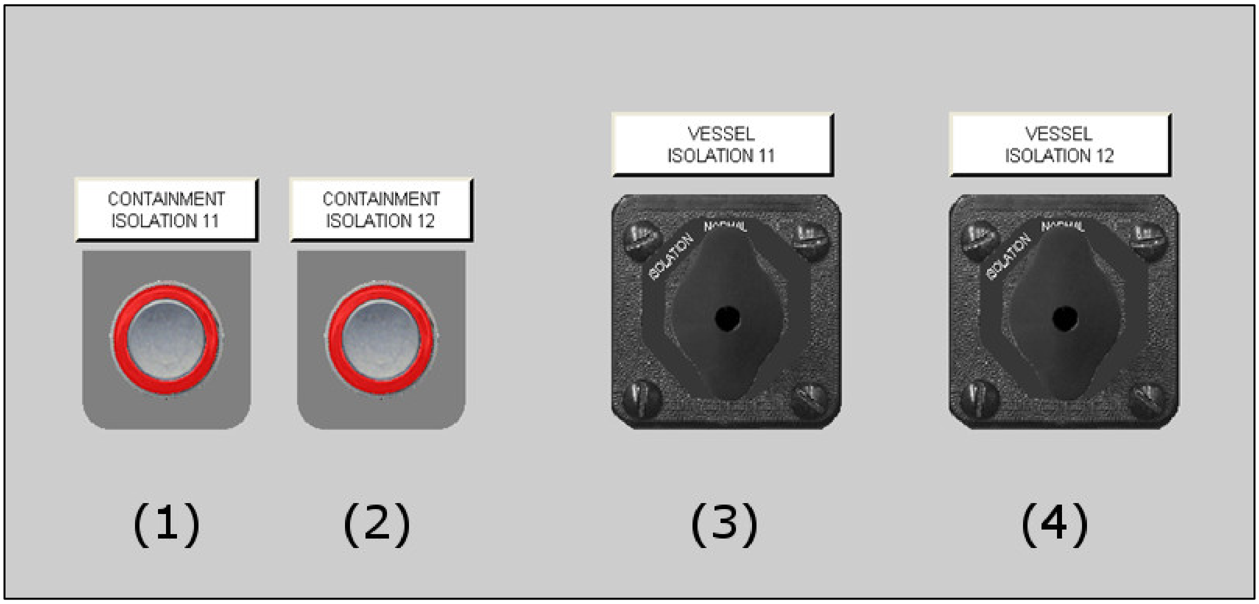
Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: Individual system relay status

Question: #11

The plant has experienced an accident with the following:

- Two Post LOCA Vent Valves (201.1-09 and 201.1-11) have failed to close as required.
- The US has directed attempting to close these valves using the isolation controls on E-console.

Given the following picture with controls labeled (1), (2), (3), and (4):



Which one of the following describes the manipulations required to initiate the manual isolation the US has requested?

A. Depress pushbuttons (1) and (2) to energize the associated isolation relays.

B. Depress pushbuttons (1) and (2) to de-energize the associated isolation relays.

C. Rotate control switches (3) and (4) to energize the associated isolation relays.

D. Rotate control switches (3) and (4) to de-energize the associated isolation relays.

Proposed Answer: B

Explanation: The Post LOCA Vent isolation valves close as part of the Containment Isolation, not the Vessel Isolation. A Containment Isolation is initiated by depressing the pushbuttons labeled (1) and (2). From drawing C-19859-S sh 13, this opens contacts 1S5a and 1S5B, de-energizing relays in the isolation circuitry to cause an isolation.

A. Incorrect – Depressing these pushbuttons de-energizes the associated relays. Plausible because some RPS control functions, such as ATWS circuitry act by energizing their respective relays.
C. Incorrect – The correct controls are (1) and (2). Plausible because (3) and (4) would be correct for other isolation valves. The associated relays de-energize to function. Plausible because some RPS control functions, such as ATWS circuitry act by energizing their respective relays.
D. Incorrect – The correct controls are (1) and (2). Plausible because (3) and (4) would be correct for other isolation valves.

Technical Reference(s): N1-SOP-40.2, C-19859-C sheet 13

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: Modified - 2017 NRC #14

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

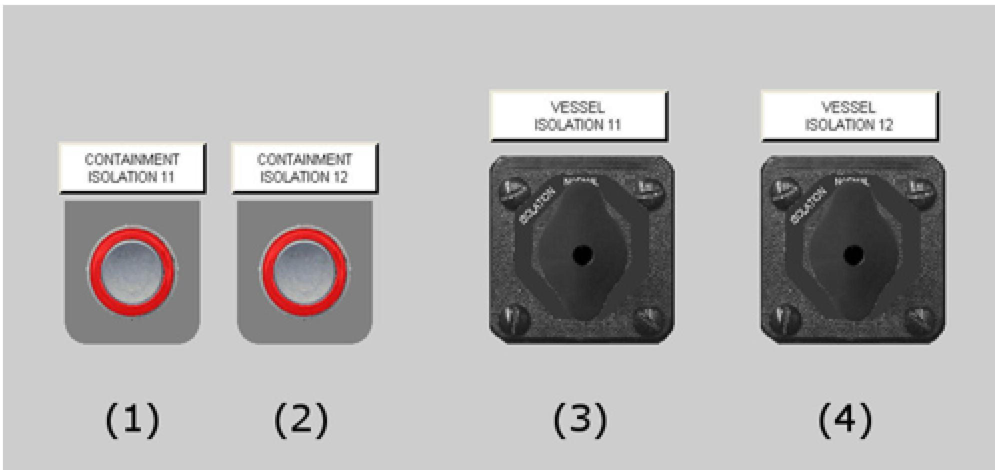
Comments:

Proposed Question: #14

The plant has experienced a Reactor Water Cleanup (RWCU) leak into the Reactor Building with the following:

- The Reactor has been scrammed.
- RWCU isolation valves have failed to automatically close.
- RWCU isolation valves have failed to close using the K-panel controls.
- The US has directed attempting to close the RWCU isolation valves using the isolation controls on E-console.

Given the following picture with controls labeled (1), (2), (3), and (4):



Which one of the following describes the manipulations required to initiate the manual isolation the US has requested?

A. Depress pushbuttons (1) and (2) to energize the associated isolation relays.

B. Depress pushbuttons (1) and (2) to de-energize the associated isolation relays.

C. Rotate control switches (3) and (4) to energize the associated isolation relays.

D. Rotate control switches (3) and (4) to de-energize the associated isolation relays.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 A2.10
	Importance Rating	3.9

Primary Containment Isolation/Nuclear Steam Supply Shutoff

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of coolant accidents

Question: #12

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

- A loss of coolant accident occurs.
- HPCI fails to initiate.
- Reactor water level is 0" and slowly lowering.
- Annunciator F1-4-2, RPS CH 11 MAIN STEAM ISOLATION AUTO OPERATE, alarms.
- Annunciator F4-4-7, RPS CH 12 MAIN STEAM ISOLATION AUTO OPERATE, does NOT alarm.

Which one of the following describes the effect of these indications on the Main Steam Isolation Valves (MSIVs) and the required action(s)?

	MSIVs	Required Action(s)
A.	Remain open	Manually close the MSIVs
B.	Remain open	Install MSIV isolation jumpers
C.	Automatically close	Verify the MSIV isolation
D.	Automatically close	Install MSIV isolation jumpers and re-open MSIVs

Proposed Answer: A

Explanation: With Reactor water level less than 5 inches, both annunciators F1-4-2 and F4-4-7 should be in alarm due to relays 11K73, 11K74, 12K73, and 12K74 de-energizing. With F4-4-7 NOT in alarm, 12K73 and 12K74 have failed to de-energize. With both of these relays still energized, none of the MSIVs will automatically close as required with Reactor water level less than 5 inches. N1-EOP-2 requires verifying isolations per N1-SOP-40.2. N1-SOP-40.2 directs manually closing the MSIVs.

B. Incorrect – The MSIVs do remain open. That part is correct. Plausible because MSIV isolation jumpers are installed if it is desired to have the MSIVs remain open with low reactor water level, such as in N1-EOP-3. Installation of the MSIV jumpers would bypass lo lo MSIV isolation. However an automatic isolation is required based on Reactor water level, and procedures require manually closing the MSIVs.
C. Incorrect – Plausible because only two of the four MSIV isolation relays need to de-energize to cause the MSIVs to automatically isolate. However, one of the relays must de-energize in each of the two RPS channels. With annunciator F4-4-7 NOT in alarm, neither RPS channel 12 relay de-energized. Therefore MSIVs did not close.
D. Incorrect – Plausible because only two of the four MSIV isolation relays need to de-energize to cause the MSIVs to automatically isolate. However, one of the relays must de-energize in each of the two RPS channels. With annunciator F4-4-7 NOT in alarm, neither RPS channel 12 relay de-energized. Therefore MSIVs did not close. Also plausible because MSIV isolation jumpers are installed if it is desired to have the MSIVs remain open with low reactor water level, such as in N1-EOP-3. Installation of the MSIV jumpers would bypass lo lo MSIV isolation. However an automatic isolation is required based on Reactor water level, and procedures require manually closing the MSIVs.

Technical Reference(s): C-19859-C sheets 10 & 11, ARPs F1-4-2 & F4-4-7, N1-EOP-2, N1-SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: Bank – LOR bank question

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 #	239002 A2.05
	Importance Rating	3.2

Safety Relief Valves

Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor pressure

Question: #13

The plant has experienced a steam leak in the Reactor Building during a startup with the following:

- The Reactor has been scrammed.
- N1-EOP-5, Secondary Containment Control, requires an RPV Blowdown.
- N1-EOP-8, RPV Blowdown, is entered.
- Emergency Condensers are initiated
- Reactor pressure is 40 psig and slowly lowering.
- Torus pressure is 1 psig and stable.

Which one of the following describes:

(1) the required execution of N1-EOP-8

and

(2) the response of ERVs if their control switches are taken to OPEN?

	(1) Required Execution of N1-EOP-8	(2) Response of ERVs If Their Control Switches Are Taken to OPEN
A.	Place ERV control switches in OPEN and leave them in OPEN	ERVs open
B.	Place ERV control switches in OPEN but return them to AUTO when they close	ERVs open
C.	Place ERV control switches in OPEN	ERVs remain closed
D.	Maintain ERV control switches in AUTO	ERVs remain closed

Proposed Answer: C

Explanation: For an ERV to actually open, it needs at least 50 psig differential pressure (between Reactor pressure and Torus pressure). With differential pressure at only 39 psig, the ERVs will not open if their control switches are taken to OPEN. However, N1-EOP-8 still requires placing their control switches in OPEN in this situation to preclude re-pressurization of the Reactor.

A. Incorrect – ERVs will not open due to insufficient differential pressure between the Reactor and Torus. Plausible because this would be correct for most valves and for ERVs if Reactor pressure were higher.
B. Incorrect – ERVs will not open due to insufficient differential pressure between the Reactor and Torus. Plausible because this would be correct for most valves and for ERVs if Reactor pressure were higher.
D. Incorrect – ERV control switches must be taken to OPEN. Plausible because this action will not actually open the ERVs and other systems could alternately be used to lower Reactor pressure or prevent re-pressurization.

Technical Reference(s): N1-218000-RBO-2, N1-EOP-8, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

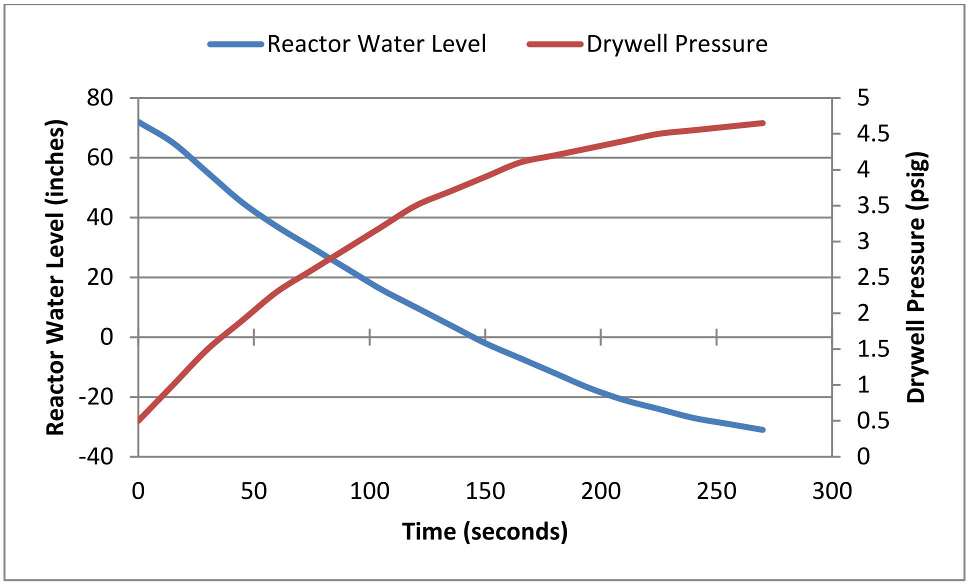
Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	239002 A3.01
	Importance Rating	3.8

Safety Relief Valves

Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including: SRV operation after ADS actuation

Question: #14

The plant is operating at 100% power when a loss of coolant accident and loss of high pressure injection results in the following Reactor water level and Drywell pressure response:



Which one of the following describes the status of the Automatic Depressurization System (ADS) and Electromatic Relief Valves (ERVs) at time = 300 seconds?

ADS…

A. is in standby and all ERVs are closed.

B. timers are counting down and all ERVs are closed.

C. timers have timed out, but all ERVs remain closed.

D. timers have timed out and three ERVs are open.

Proposed Answer: D

Explanation: ADS timers begin timing when both Drywell pressure is ≥3.5 psig and Reactor water level is ≤ -10 inches. This occurs at approximately Time = 180 seconds. The ADS timers count down for 111 seconds. Therefore, at approximately Time ≥ 291 seconds, the ADS timers will be timed out and three ERVs will open.

A. Incorrect – ADS will have timed out and ERVs will be open. Plausible because this would be correct if Drywell pressure stayed below 3.5 psig or if Reactor water level returned above -10”.
B. Incorrect – ADS will have timed out and ERVs will be open. Plausible because this would be correct at a slightly earlier time or if ADS logic actuated slightly later.
C. Incorrect – ERVs will be open. Plausible because this would be correct if Reactor pressure was below 50 psig, as in the bank question.

Technical Reference(s): N1-OP-2, N1-OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-5

Question Source: Modified Bank - 2015 Cert #23

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

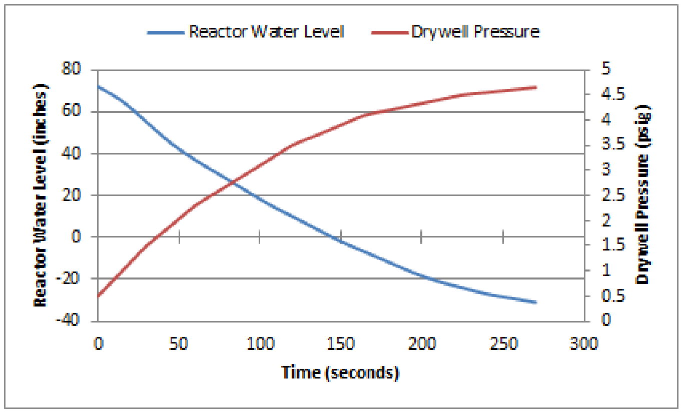
Proposed Question: #23

A plant startup is in progress with the following:

- The Reactor is at the point of adding heat.
- Reactor pressure is 25 psig and slowly rising.

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

Area



Which one of the following describes the status of the Automatic Depressurization System (ADS) and Electromatic Relief Valves (ERVs) at Time = 300 seconds?

ADS...

A. is in standby and all ERVs are closed.

B. timers are counting down and all ERVs are closed.

C. timers have timed out, but all ERVs remain closed.

D. timers have timed out and three ERVs are open.

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	259002 A3.07
	Importance Rating	3.5

Reactor Water Level Control

Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: FWRV lockup

Question: #15

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

- Feedwater pump 13 is operating with its flow control valve in automatic.
- Feedwater Level Control (FWLC) is selected to channel 11.

Which one of the following identifies:

(1) a condition that causes automatic lockup of Feedwater flow control valve 13

and

(2) how this lockup responds when the condition is corrected?

	(1) Feedwater flow control valve 13 automatically locks up on loss of…	(2) When this condition is corrected, the lockup…
A.	RPS Bus 11	automatically resets
B.	RPS Bus 11	must be manually reset
C.	Instrument Air	automatically resets
D.	Instrument Air	must be manually reset

Proposed Answer: D

Explanation: Feedwater flow control valve 13 automatically locks up on a loss of Instrument Air. When Instrument Air pressure is restored, the lock remains sealed in until manually reset.

A. Incorrect – Feedwater flow control valve 13 does not automatically lock up on loss of RPS Bus 11. Plausible because with FWLC selected to channel 11, RPS Bus 11 is supplying power. However, on loss of RPS Bus 11, the power supply to FWLC automatically swaps to RPS Bus 12. Also plausible because loss of electrical signal to Feedwater flow control valve 13 would also cause a lock up. The lock remains sealed in until manually reset. Plausible because some automatic actions do automatically reset upon restoration of the initiating condition.
B. Incorrect – Feedwater flow control valve 13 does not automatically lock up on loss of RPS Bus 11. Plausible because with FWLC selected to channel 11, RPS Bus 11 is supplying power. However, on loss of RPS Bus 11, the power supply to FWLC automatically swaps to RPS Bus 12. Also plausible because loss of electrical signal to Feedwater flow control valve 13 would also cause a lock up.
C. Incorrect – The lock remains sealed in until manually reset. Plausible because some automatic actions do automatically reset upon restoration of the initiating condition.

Technical Reference(s): N1-OP-16, ARP H3-4-6

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259002-RBO-8

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 #	1
	K/A #	259002 A4.06
	Importance Rating	3.1

Reactor Water Level Control

Ability to manually operate and/or monitor in the control room: DP/Single/three element control selector switch: Plant-Specific

Question: #16

The plant is operating at 10% power.

Which one of the following identifies the parameters used by the Feedwater Level Control System (FWLC) to control Reactor water level at this power level, in accordance with N1-OP-16, Feedwater System Booster Pump to Reactor?

A. Reactor water level, only

B. Reactor water level and feed flow, only

C. Reactor water level and steam flow, only

D. Reactor water level, feed flow, and steam flow

Proposed Answer: A

Explanation: Below 25% power, N1-OP-16 requires FWLC to be in single element control. In this mode, only Reactor water level is used by FWLC to control Reactor water level.

B. Incorrect – Only Reactor water level is used at this power level. Plausible because feed flow is used at higher power levels.
C. Incorrect – Only Reactor water level is used at this power level. Plausible because steam flow is used at higher power levels.
D. Incorrect – Only Reactor water level is used at this power level. Plausible because feed and steam flow are used at higher power levels. This would be correct >25% power.

Technical Reference(s): N1-OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259002-RBO-5

Question Source: Bank - 2010 NRC #7

Question History: 2010 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 #	261000 A4.09
	Importance Rating	2.7

Standby Gas Treatment

Ability to manually operate and/or monitor in the control room: Ventilation valves/dampers

Question: #17

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

- Operators are venting the Drywell with RBEVS in accordance with N1-OP-9.
- Drywell pressure is 1.8 psig and lowering.

Then, a seismic event results in the following:

- Drywell pressure is 3.6 psig and rising slowly.
- Reactor Building Ventilation Exhaust radiation is 10 mr/hr and rising slowly.

Which one of the following describes the response of valves 201-18, EM VENTILATION FROM DW & TORUS BV, and 202-36, EM VENTILATION FROM REACTOR BLDG BV?

	201-18	202-36
A.	Remains open	Remains closed
B.	Remains open	Automatically opens
C.	Automatically closes	Remains closed
D.	Automatically closes	Automatically opens

Proposed Answer: B

Explanation: Per N1-OP-9, Section H, valve 201-18 remains open despite the Containment Isolation signal (Drywell pressure >3.5 psig) and must be manually closed in this situation. Per N1-OP-10, Section B.3.0, valve 202-36 will automatically open due to the RBEVS initiation signal (>5 mr/hr).

A. Incorrect – 202-36 receives an automatic open signal on high radiation (>5 mr/hr). Plausible because 202-36 is initially closed for the venting lineup and 201-18 does not automatically re-position.
C. Incorrect – 201-18 remains open. Plausible because 201-18 is required to be closed given the Containment isolation and RBEVS initiation, however there is no automatic feature to close this valve. 202-36 receives an automatic open signal on high radiation (>5 mr/hr). Plausible because 202-36 is initially closed for the venting lineup and 201-18 does not automatically re-position.
D. Incorrect – 201-18 remains open. Plausible because 201-18 is required to be closed given the Containment isolation and RBEVS initiation, however there is no automatic feature to close this valve.

Technical Reference(s): N1-OP-9, N1-OP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-5

Question Source: Bank – 2017 Cert #35

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	212000 2.4.6
	Importance Rating	3.7

Reactor Protection System

Knowledge of EOP mitigation strategies.

Question: #18

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- All APRMs are downscale.
- All IRMs are indicating mid-scale on Range 7.

Then, a Feedwater malfunction results in the following:

- Reactor water level lowers to 40".
- The Reactor Mode Switch is placed in SHUTDOWN.
- The RPS pushbuttons are depressed.
- ARI is manually initiated.
- All RPS white lights remain illuminated.
- NO control rods scram.

Which one of the following describes the required mitigating strategy?

Enter…

A. N1-SOP-1, Reactor Scram. NO EOP entry is required.

B. N1-EOP-2, RPV Control. NO entry into N1-EOP-3, Failure to Scram, is required.

C. N1-EOP-2, RPV Control, and remain in this procedure. Additionally, enter N1-EOP-3, Failure to Scram.

D. N1-EOP-2, RPV Control. Then exit N1-EOP-2 and enter N1-EOP-3, Failure to Scram.

Proposed Answer: D

Explanation: The given conditions show a failure of RPS to insert control rods (on an automatic scram, a manual scram, and manual ARI initiation). The Reactor is initially critical above the POAH (IRMs on range 7) but below the APRM downscales. With Reactor water level <53”, entry into N1-EOP-2 is required. All control rods are not inserted to at least position 04. With the Reactor critical and above the POAH, the Reactor will not stay shutdown without boron. Therefore, N1-EOP-2 requires exiting N1-EOP-2 and entering N1-EOP-3.

A. Incorrect – N1-EOP-2 and N1-EOP-3 must be entered. Plausible because this would be correct if Reactor water level had stayed above 53”.
B. Incorrect – N1-EOP-3 must be entered. Plausible because APRMs are downscale (N1-EOP-2 entry on Reactor power is not even required).
C. Incorrect – N1-EOP-2 must be exited. Plausible based on low initial power.

Technical Reference(s): N1-EOP-2, N1-EOP-3

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP3C01 EO-2

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 #	262001 2.1.7
	Importance Rating	4.4

AC Electrical Distribution

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

Question: #19

A loss of coolant accident has occurred with the following:

- A manual Reactor scram was inserted.
- Following the scram, Lines 1 and 4 de-energized.
- N1-SOP-33A.1, Loss of 115 KV, has been entered.
- EDG 102 load is 2250 KW.
- EDG 103 load is 2450 KW.

Which one of the following describes the status of EDG load, in accordance with N1-SOP-33A.1?

A. Both EDGs are below the load limit requiring operator action.

B. EDG 102 is below the load limit requiring operator action. EDG 103 is above the load limit requiring operator action, but below the emergency load limit.

C. Both EDGs are above the load limit requiring operator action, but below the emergency load limit.

D. Both EDGs are above the load limit requiring operator action. EDG 102 is below the emergency load limit, but EDG 103 is above the emergency load limit.

Proposed Answer: B

Explanation: N1-SOP-33A.1 requires operator action if EDG load exceeds 2300 KW and lists the EDG emergency load limit as 2845 KW. EDG 102 is below both of those limits. EDG 103 is greater than 2300 KW, but less than 2845 KW.

A. Incorrect – EDG 103 is above the limit of 2300 KW. Plausible because is still below the continuous rating of 2586 KW.
C. Incorrect – EDG 102 is below the limit of 2300 KW. Plausible because EDG 102 load is relatively high.
D. Incorrect – EDG 102 is below the limit of 2300 KW. Plausible because EDG 102 load is relatively high. EDG 103 is below the emergency load limit of 2845 KW. Plausible because EDG 103 is above the continuous load limit of 2586 KW.

Technical Reference(s): N1-SOP-33A.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-264001-RBO-9

Question Source: Modified Bank – 2017 NRC #19

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 # 262001 A4.05
Importance Rating 3.3

AC Electrical Distribution

Ability to manually operate and/or monitor in the control room: Voltage, current, power, and frequency on A.C. buses

Proposed Question: #19

A loss of coolant accident has occurred with the following:

- A manual Reactor scram was inserted.
- Following the scram, Lines 1 and 4 de-energized.
- N1-SOP-33A.1, Loss of 115 KV, has been entered.
- EDG 102 load is 2450 KW.
- EDG 103 load is 2750 KW.

Which one of the following describes the status of EDG load, in accordance with N1-SOP-33A.1?

A. Both EDGs are below the load limit requiring operator action.

B. EDG 102 is below the load limit requiring operator action. EDG 103 is above the load limit requiring operator action, but below the emergency load limit.

C. Both EDGs are above the load limit requiring operator action, but below the emergency load limit.

D. Both EDGs are above the load limit requiring operator action. EDG 102 is below the emergency load limit, but EDG 103 is above the emergency load limit.

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	1
K/A #	262001 K1.02
Importance Rating	3.3

AC Electrical Distribution

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

Question: #20

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

- A complete loss of Battery Board 12 occurs.
- Then, an Operator transfers the DC supply switch from “Battery Board 12” to “Battery Board 11” at Powerboard (PB) 103.

Which one of following describes the resulting status of control power for Breaker R-1013, RESERVE SUPPLY TO PB 103, and CORE SPRAY PUMP 112 MOTOR BREAKER?

	Breaker R-1013, RESERVE SUPPLY TO PB 103	CORE SPRAY PUMP 112 MOTOR BREAKER
A.	Has control power	Has control power
B.	Has control power	Does NOT have control power
C.	Does NOT have control power	Has control power
D.	Does NOT have control power	Does NOT have control power

Proposed Answer: C

Explanation: DC control power is divisionalized such that Battery Board 11 normally supplies Powerboard 102 and Battery Board 12 normally supplies Powerboard 103. Some provisions are installed to cross-tie divisions. With a loss of Battery Board 12, the control power for Powerboard 103 can be partially transferred to Battery Board 11. All of the load breakers, such as for Core Spray pump 112, are transferrable, but the supply breakers, such as R-1013, are non-transferrable.

A. Incorrect – R-1013 does not have control power. Plausible because all of the load breakers do have control power after the manual swap.
B. Incorrect – R-1013 does not have control power. Plausible because all of the load breakers do have control power after the manual swap. The Core Spray breaker does have control power. Plausible because the supply breakers do not have control power.
D. Incorrect – The Core Spray breaker does have control power. Plausible because the supply breakers do not have control power.

Technical Reference(s): N1-SOP-47A.1, N1-OP-47A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001-RBO-8

Question Source: Bank – 2015 NRC #40

Question History: 2015 NRC #40

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 #	262002 K1.17
	Importance Rating	3.1

Uninterruptable Power Supply (AC/DC)

Knowledge of the physical connections and/or cause-effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: Scram solenoid valves: Plant-Specific

Question: #21

The plant is operating at 100% power when Motor-Generator (MG) Set 131 trips.

Which one of the following describes the response of scram solenoids (SOV-139(A)s & SOV-139(B)s), scram pilot solenoid valves (SOV-139s), and control rods?

	Scram Solenoids (SOV-139(A)s & SOV-139(B)s)	Scram Pilot Valves (SOV-139s)	Control Rods
A.	NONE change state	NONE re-position	NONE re-position
B.	Half change state	NONE re-position	NONE re-position
C.	Half change state	Half re-position	NONE re-position
D.	Half change state	Half re-position	Half re-position

Proposed Answer: B

Explanation: With the trip of MG Set 131, Reactor Trip Bus 131 de-energizes. This causes half of the normally energized scram solenoids to de-energize (one at each HCU). With the other scram solenoid still energized from Reactor Trip Bus 141, none of the scram pilot solenoids valves actually re-position, therefore no control rods re-position.

Note: The question meets the K/A because MG Set 131 is the equivalent of a UPS in this application at NMP1. Additionally, note how multiple sub-K/As for this system discuss motor-generator operation (eg. 262002 K5.02, K5.03, A1.02, etc.).

A. Incorrect – Half of the normally energized scram solenoids de-energize (change state). Plausible that only logic would be affected but the actual solenoids would not change state without the other half of the logic changing.
C. Incorrect – None of the scram pilot solenoids valves actually re-position. Plausible because this used to be the case before a change in system design.
D. Incorrect – None of the scram pilot solenoids valves actually re-position and no control rods re-position. Plausible because is used to be the case that half of the scram pilot solenoid valves would re-position. Plausible that this could result in control rod motion.

Technical Reference(s): C-19859-C sheet 4 and 7, C-18016-C

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-11

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 #	263000 K3.02
	Importance Rating	3.5

DC Electrical Distribution

Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Components using D.C. control power (i.e. breakers)

Question: #22

The plant is shutdown with the following:

- Shutdown Cooling (SDC) loops 11 and 12 are in service.
- Then, Battery Board 12 de-energizes due to a sustained electrical fault.

Which one of the following describes the effect on SDC?

A. The system continues to operate. The only affect is that 38-02, SDC Inlet IV (Outside), loses power.

B. A SDC Isolation signal occurs. SDC pump 12 does NOT trip and 38-02, SDC Inlet IV (Outside), does NOT close.

C. A SDC Isolation signal occurs. SDC pump 12 trips, however 38-02, SDC Inlet IV (Outside), does NOT close.

D. The system continues to operate. 38-02, SDC Inlet IV (Outside), loses power, and the ability to operate SDC Pump 12 from the Control Room is lost.

Proposed Answer: D

Explanation: A loss of 125 VDC does NOT cause a condition that will cause an SDC Isolation. RPS supplies the isolation logic, so with just a loss of DC, no isolation signal occurs. The DC distribution system supplies 125 VDC to operate the DC motor-operated isolation valve (38-02) and control power for the pumps. On a loss of the respective power supply, 38-02 will fail as-is. On a loss of DC control power, the ability to operate the pump from the control room will be lost.

A. Incorrect – SDC pump 12 also loses control power and cannot be operated from the Control Room. Plausible that SDC pump 12 would retain control power from another source or automatically swap to another source.
B. Incorrect – No isolation signal occurs. Plausible because this would be correct for a loss of RPS bus 12.
C. Incorrect – No isolation signal occurs. Plausible because this would be correct for a loss of RPS bus 12.

Technical Reference(s): N1-SOP-47A.1, C-19845-C sheet 1, C-19439-C sheet 10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-4

Question Source: Bank – 2010 NRC #11

Question History: 2010 NRC #11

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 #	264000 K4.01
	Importance Rating	3.5

Emergency Generators (Diesel/Jet) EDG

Knowledge of EMERGENCY GENERATORS (DIESEL/JET) design feature(s) and/or interlocks which provide for the following: Emergency generator trips (normal)

Question: #23

The plant is operating at 100% power with Emergency Diesel Generator (EDG) 102 running during performance of N1-ST-M4A.

Given the following signals:

(1) Low cooling water pressure
(2) High engine temperature

Which one of the following describes which of these signal(s) will automatically trip EDG 102, if any?

A. Neither

B. (1) only

C. (2) only

D. (1) and (2)

Proposed Answer: B

Explanation: Per N1-OP-45, section B, low cooling water pressure automatically trips the EDG. High engine temperature causes an alarm, but does not automatically trip the EDG.

A. Incorrect – Low cooling water pressure automatically trips the EDG. Plausible this would be alarm only as it does not necessarily mean that the engine is experiencing high temperature.
C. Incorrect – Low cooling water pressure automatically trips the EDG. Plausible this would be alarm only as it does not necessarily mean that the engine is experiencing high temperature. High engine temperature does not automatically trip the EDG. Plausible because it does cause an alarm and does indicate a direct problem with the EDG.
D. Incorrect – High engine temperature does not automatically trip the EDG. Plausible because it does cause an alarm and does indicate a direct problem with the EDG.

Technical Reference(s): N1-OP-45, ARP A4-3-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-264001-RBO-5

Question Source: Modified Bank – 2015 Cert #24

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 # 264000 K4.02
Importance Rating 4.0

EDGs

Knowledge of EMERGENCY GENERATORS (DIESEL/JET) design feature(s) and/or interlocks which provide for the following: Emergency generator trips (emergency/LOCA)

Proposed Question: #24

Given the following Emergency Diesel Generator (EDG) trip signals:

(1) Low cooling water pressure
(2) Engine overspeed

Which one of the following describes which trip signal(s) is(are) bypassed upon automatic EDG initiation, if any?

A. Neither

B. (1) only

C. (2) only

D. (1) and (2)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 K5.01
	Importance Rating	2.5

Instrument Air

Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Air compressors

Question: #24

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

- Instrument Air Compressor (IAC) 13 is in service.
- IAC 11 is green flagged as the standby compressor.
- IAC 12 is red flagged as the backup compressor.

Then, IAC 13 trips on overcurrent. Instrument Air pressure is 101 psig and slowly lowering.

Which one of the following describes the automatic operation of the Instrument Air system?

A. IAC 11 loads immediately.

B. IAC 12 loads immediately.

C. IAC 11 loads when Instrument Air pressure reaches 98 psig.

D. IAC 12 loads when Instrument Air pressure reaches 98 psig.

Proposed Answer: D

Explanation: As the backup compressor, IAC 12 has its control switch red flagged and its Loading Selector Switches in the 1 position. This sets up IAC 12 to auto-start if IA pressure reaches 98 psig. As the standby compressor, IAC 11 has its control switch green flagged and its Loading Selector Switches in the 2 position. This sets up IAC 11 to auto-start if IA pressure reaches 93 psig.

A. Incorrect – IAC 11 will load if needed, but not until IA pressure lowers to 93 psig. Plausible because loading of at least one IAC will be required and could be controlled directly off of IAC 13 breaker position.
B. Incorrect – IAC 12 will load, but not until IA pressure lowers to 98 psig. Plausible because loading will be required and could be controlled directly off of IAC 13 breaker position.
C. Incorrect – IAC 11 will load if needed, but not until IA pressure lowers to 93 psig. Plausible if candidate confuses Loading Selector Switch positions for the standby and backup IACs.

Technical Reference(s): N1-OP-20

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278001-RBO-5

Question Source: Bank – 2017 Cert #11

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 K6.06
	Importance Rating	2.9

Component Cooling Water

Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: Heat exchangers and condensers

Question: #25

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

- All five Reactor Recirculation pumps (RRPs) are operating.
- A leak develops in the RRP 11 pump cooler.

Which one of the following identifies the response of RBCLC radiation monitor indication and CLC Makeup Tank level?

	RBCLC Radiation Monitor Indication	CLC Makeup Tank Level
A.	Rises	Rises
B.	Rises	Remains the same
C.	Remains the same	Rises
D.	Remains the same	Remains the same

Proposed Answer: B

Explanation: The RRP 11 pump cooler has Reactor coolant (~1020 psig, 545ºF) on one side of the heat exchanger and RBCLC water (~100 psig, 90ºF) on the other side. With a leak in the heat exchanger, the higher pressure and contaminated Reactor coolant enters the RBCLC system. This causes the RBCLC radiation monitor indication to rise. It also causes excess RBCLC water to flow through the overflow line near the CLC makeup tank. However, a check valve prevents this water from entering the CLC makeup tank and causing level to rise.

A. Incorrect – CLC Makeup Tank level remains the same. Plausible because water is flowing from the Reactor coolant system into RBCLC.
C. Incorrect – RBCLC rad monitor indication rises. Plausible because some RBCLC load heat exchanger leaks would cause water to flow from RBCLC into the other process stream, which would result in no change in the RBCLC rad monitor indication. CLC Makeup Tank level remains the same. Plausible because water is flowing from the Reactor coolant system into RBCLC.
D. Incorrect – RBCLC rad monitor indication rises. Plausible because some RBCLC load heat exchanger leaks would cause water to flow from RBCLC into the other process stream, which would result in no change in the RBCLC rad monitor indication.

Technical Reference(s): C-18022-C sheets 2 and 3, N1-OP-1 Attachment 11

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-208000-RBO-11

Question Source: Bank – 2015 Cert #2

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 #	207000 A1.01
	Importance Rating	3.7

Isolation (Emergency) Condenser

Ability to predict and/or monitor changes in parameters associated with operating the ISOLATION (EMERGENCY) CONDENSER controls including: Isolation condenser level: BWR-2,3

Question: #26

The plant has scrammed with the following:

- Emergency Condenser (EC) 11 shell level is 6.5 feet and stable.
- Then, EC 11 is placed in service.

Which one of the following describes the response of EC 11 shell level?

EC 11 shell level…

A. rises and overflow first starts at 7 feet.

B. rises and overflow first starts at 9.5 feet.

C. lowers and automatic makeup first starts at 6 feet.

D. lowers and automatic makeup first starts at 4.5 feet.

Proposed Answer: C

Explanation: When an Emergency Condenser is placed in service, shell side water level begins to lower due to boil off of water from the EC to the atmosphere. Automatic makeup first begins to restore water level at 6 feet.

A. Incorrect – Shell side water level initially begins to lower. Plausible because Reactor water level rises when an EC is initiated. Also plausible that condensation of Reactor steam would raise level if construction of the EC was not well understood. 7 feet is plausible because it is higher than the normal level.
B. Incorrect – Shell side water level initially begins to lower. Plausible because Reactor water level rises when an EC is initiated. Also plausible that condensation of Reactor steam would raise level if construction of the EC was not well understood. 9.5 feet is plausible because it is higher than the normal level and a water level associated with the related EC makeup water tank.
D. Incorrect – Automatic makeup first begins to restore water level at 6 feet. Plausible because 4.5 feet is below the normal value of EC shell water level but well within the indicating range of the level instrument. Also plausible because 4.5 feet is a level value that corresponds to the Reactor low level scram on wide range level instrumentation.

Technical Reference(s): ARP K1-2-3, N1-OP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201001 A3.08
	Importance Rating	3.0

CRD Hydraulic

Ability to monitor automatic operations of the CONTROL ROD DRIVE HYDRAULIC SYSTEM including: Drive water flow

Question: #27

A startup is in progress with the following:

- Control rod 22-07 is given a notch withdraw signal.
- Initially, the control rod inserts and CRD drive water flow indicates 4 gpm.
- Next, the control rod withdraws and CRD drive water flow indicates 2 gpm.
- Then, the control rod settles and CRD drive water flow indicates 0 gpm.

Which one of the following describes these CRD drive water flow indications?

A. All of these flow indications are as expected for a notch withdraw signal.

B. While the control rod was inserting, the flow indication was too high.

C. While the control rod was withdrawing, the flow indication was too low.

D. While the control rod was settling, the flow indication was too low.

Proposed Answer: A

Explanation: Normal drive water flow is 4 gpm during inserting and 2 gpm during withdrawing. Drive water flow should be 0 gpm during settling. Therefore, all indications were correct.

B. Incorrect – Normal drive water flow is 4 gpm during inserting. Plausible because this is significantly higher than the flow during withdrawing.
C. Incorrect – Normal drive water flow is 2 gpm during withdrawing. Plausible because this is significantly lower than the flow during inserting.
D. Incorrect – Normal drive water flow is 0 gpm during settling. Plausible that some flow would still be indicated as the rod is still moving.

Technical Reference(s): N1-OP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-3

Question Source: Bank - SSES LOC27 NRC #33

Question History: SSES LOC27 NRC #33

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 #	201003 A4.02
	Importance Rating	3.5

Control Rod and Drive Mechanism

Ability to manually operate and/or monitor in the control room: CRD mechanism position: Plant-Specific

Question: #28

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

- Rod Select Power is turned OFF.
- Control rod 02-19 is at position 48.
- Then, control rod 02-19 reed switch S47 (for position 47) fails in the closed position.
- NO actual control rod movement occurs.

Which one of the following describes the impact of this switch failure?

F3-2-6, CONTROL ROD DRIFT, will annunciate…

A. at this time.

B. only when Rod Select Power is turned ON.

C. only when control rod 02-19 is selected for movement.

D. only when control rod 02-19 is selected for movement and the rod movement timer has started.

Proposed Answer: A

Explanation: The rod drift alarm will be received due to an odd reed switch being closed with the rod not selected for movement.

B. Incorrect – The rod drift alarm will come in as soon as a non-selected control rod has an odd position reed switch close. Plausible because the S48 reed switch remains closed. Also plausible that Rod Select Power would be involved in this alarm circuitry.
C. Incorrect – The rod drift alarm will come in as soon as a non-selected control rod has an odd position reed switch close. Plausible because the S48 reed switch remains closed. Also plausible because the rod select circuitry does interact with the rod drift circuitry.
D. Incorrect – The rod drift alarm will come in as soon as a non-selected control rod has an odd position reed switch close. Plausible because the S48 reed switch remains closed. Also plausible because the rod select circuitry and timer status does interact with the rod drift circuitry.

Technical Reference(s): SDBD 303, N1-201002-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201002-RBO-11

Question Source: Bank – 2010 Cert #29

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	204000 2.2.42
	Importance Rating	3.9

Reactor Water Cleanup

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

Question: #29

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

- Reactor Water Cleanup (RWCU) isolates and CANNOT be restored.
- The continuous conductivity meter is operable and aligned to Recirc loop 11.

Which one of the following describes the need for entry into Technical Specification 3.2.3, Coolant Chemistry?

Entry into TS 3.2.3 is…

A. immediately required due to the loss of RWCU.

B. NOT immediately required but will be required if RWCU remains out of service for 8 hours, regardless of actual Reactor coolant chemistry.

C. NOT required based on the loss of RWCU alone. Entry will be required if fluoride ion concentration exceeds limits.

D. NOT required based on the loss of RWCU alone. Entry will be required if chloride ion concentration exceeds limits.

Proposed Answer: D

Explanation: The loss of RWCU alone does not require entry into TS 3.2.3. However, TS 3.2.3 entry will be required if chloride ion concentration exceeds limits.

A. Incorrect – The loss of RWCU alone does not require entry into TS 3.2.3. Plausible that TS 3.2.3 would require RWCU to be in service at 100% power, because coolant chemistry will eventually exceed limits without RWCU in service.
B. Incorrect – The loss of RWCU alone does not require entry into TS 3.2.3. Plausible that TS 3.2.3 would require RWCU to be in service at 100% power, because coolant chemistry will eventually exceed limits without RWCU in service. An 8 hour allowance is plausible since chemistry will not immediately exceed limits and TS 3.2.3 surveillance requirements use an 8 hour time.
C. Incorrect – Flouride ion concentration exceeding limits does not require entry into TS 3.2.3. Plausible because fluoride is an element of concern with regards to coolant chemistry and is tested for by Chemistry.

Technical Reference(s): Technical Specification 3.2.3

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-204000-RBO-14

Question Source: New

Question History:

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 #	215001 K1.05
	Importance Rating	3.3

Traversing In Core Probe

Knowledge of the physical connections and/or cause-effect relationships between TRAVERSING IN-CORE PROBE and the following: Primary containment isolation system: (Not-BWR1)

Question: #30

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

- Traversing In-Core Probe (TIP) scans are in progress with TIP 1 detector in the core.
- Then, a coolant leak develops in the Drywell.
- Drywell pressure is 3.6 psig and rising.
- TIP 1 detector fails to retract automatically and manually.

Which one of the following describes the resulting operation of the TIP 1 shear valve?

The TIP 1 shear valve…

A. automatically closes immediately.

B. automatically closes after a time delay.

C. must be manually closed using a switch at the TIP Room.

D. must be manually closed using a switch on Control Room J panel.

Proposed Answer: D

Explanation: Upon receipt of a high Drywell pressure signal (3.5 psig), the TIP should automatically retract and then the ball valve should automatically close. Since the TIP failed to retract, the ball valve cannot close for the containment isolation. The shear valve is provided as a backup, but does not automatically isolate. The shear valve is closed using a control on Control Room J Panel.

A. Incorrect – The TIP shear valve does not have an automatic closure feature. Plausible because the related TIP ball valve does have an automatic closure feature.
B. Incorrect – The TIP shear valve does not have an automatic closure feature. Plausible because the related TIP ball valve does have an automatic closure feature.
C. Incorrect – The manual shear valve closure is accomplished using a switch on Control Room J Panel. Plausible that this action would be accomplished locally at the TIP room since it is an infrequently used control.

Technical Reference(s): N1-OP-39, N1-SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215001-RBO-5

Question Source: Bank – 2017 Cert #38

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	216000 K2.01
	Importance Rating	2.8

Nuclear Boiler Instrumentation

Knowledge of electrical power supplies to the following: Analog trip system: Plant-Specific

Question: #31

Which one of the following describes the power source that normally supplies power for the Analog Trip System (ATS) cabinet C?

A. RPS Bus 11

B. RPS Bus 12

C. Battery Board 11

D. Battery Board 12

Proposed Answer: A

Explanation: RPS Bus 11 is the normal power supply to ATS cabinet C.

B. Incorrect – RPS Bus 11 is the normal power supply to ATS cabinet C. Plausible because this would be correct for ATS cabinets B and D.
C. Incorrect – RPS Bus 11 is the normal power supply to ATS cabinet C. Plausible because Battery Board 11 is the backup power supply to RPS Bus 11, and therefore also a backup to ATS cabinet C. Also plausible because Battery Board 11 does directly supply other important loads.
D. Incorrect – RPS Bus 11 is the normal power supply to ATS cabinet C. Plausible because Battery Board 12 is the backup power supply to RPS Bus 12, and therefore also a backup to ATS cabinets B and D. Also plausible because Battery Board 12 does directly supply other important loads.

Technical Reference(s): N1-212000-RBO-4, C-19409-C Sh 1B, N1-OP-40, N1-SOP-40.1, C-19957-C Sh 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-4

Question Source: Modified Bank - 2015 NRC #4

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 # 212000 K2.02
Importance Rating 2.7

RPS

Knowledge of electrical power supplies to the following: Analog trip system logic cabinets

Proposed Question: #4

Which one of the following describes the 600VAC power source that normally supplies power for the Reactor Protection System (RPS) Analog Trip System (ATS) cabinet D?

A. Powerboard 16

B. Powerboard 17

C. Powerboard 131

D. Powerboard 141

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	2
	Group #	2
	K/A #	223001 K3.03
	Importance Rating	3.4

Primary Containment and Auxiliaries

Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES will have on following: Containment/drywell pressure: Plant-Specific

Question: #32

Which one of the following describes the effect that a failed open Drywell vacuum breaker would have on Primary Containment pressure response to a design-basis loss of coolant accident?

A. Both peak Torus and Drywell pressures would be higher than in design-basis analysis.

B. Both peak Torus and Drywell pressures would be lower than in design-basis analysis.

C. Peak Torus pressure would be higher and peak Drywell pressure would be lower than in design-basis analysis.

D. Peak Torus pressure would be lower and peak Drywell pressure would be higher than in design-basis analysis.

Proposed Answer: A

Explanation: With a failed open Drywell vacuum breaker, the Torus and Drywell air spaces are directly connected and the pressure-suppression function of the Primary Containment is bypassed. In the event of a LOCA, this means steam will not be forced under the water in the Torus and condensed. Therefore, for anything but a very small LOCA, pressure will be higher in both the Torus and the Drywell as compared to the design-basis analysis.

B. Incorrect – Both pressures will be higher. Plausible because the effect would be different for a very small LOCA and if the candidate does not fully understand the pressure suppression function of the Primary Containment.
C. Incorrect – Both pressures will be higher. Plausible because the effect would be different for a very small LOCA and if the candidate does not fully understand the pressure suppression function of the Primary Containment.
D. Incorrect – Both pressures will be higher. Plausible because the effect would be different for a very small LOCA and if the candidate does not fully understand the pressure suppression function of the Primary Containment.

Technical Reference(s): C-18006-C sheet 2, UFSAR Ch 15

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223001-RBO-11

Question Source: Bank – NMP2 2019 NRC #57

Question History: NMP2 2019 NRC #57

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	233000 K4.06
	Importance Rating	2.9

Fuel Pool Cooling/Cleanup

Knowledge of FUEL POOL COOLING AND CLEAN-UP design feature(s) and/or interlocks which provide for the following: Maintenance of adequate pool level

Question: #33

Which one of the following describes the normal makeup water flow path to the Spent Fuel Pool?

A. Surge tank level is sensed which opens a level control valve to admit water directly to the surge tank.

B. Spent Fuel Pool level is sensed which opens a level control valve to admit water directly to the surge tank.

C. Surge tank level is sensed which opens a level control valve to admit water directly to the Spent Fuel Pool.

D. Spent Fuel Pool level is sensed which opens a level control valve to admit water directly to the Spent Fuel Pool.

Proposed Answer: C

Explanation: Surge tank level is sensed by LT 54-27 and opens the normal makeup valve LCV 57-25, which discharges directly to the Spent Fuel Pool.

A. Incorrect – Makeup discharges directly to the Spent Fuel Pool. Plausible because the backup makeup source discharges directly to the surge tank.
B. Incorrect – Surge tank level is sensed for normal makeup control. Plausible because Spent Fuel Pool level is the direct parameter of concern. Makeup discharges directly to the Spent Fuel Pool. Plausible because the backup makeup source discharges directly to the surge tank.
D. Incorrect – Surge tank level is sensed for normal makeup control. Plausible because Spent Fuel Pool level is the direct parameter of concern.

Technical Reference(s): N1-OP-6

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-233000-RBO-3

Question Source: Bank – 2009 NRC #59

Question History: 2009 NRC #59

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(13)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	239001 K5.09
	Importance Rating	3.4

Main and Reheat Steam

Knowledge of the operational implications of the following concepts as they apply to MAIN AND REHEAT STEAM SYSTEM: Decay heat removal

Question: #34

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

- An unplanned degradation of Main Condenser vacuum has occurred.
- A manual Reactor scram is inserted.
- The Reactor Mode Switch is in SHUTDOWN.
- Emergency Condensers and ERVs are unavailable.
- Main Condenser vacuum is 9” Hg and stable.

Given the following possible decay heat removal options:

(1) Turbine Bypass Valves
(2) Offgas Mixing Jet

Which one of the following identifies the decay heat removal options currently available for use, if any?

A. (1) only

B. (2) only

C. (1) and (2)

D. Neither is available

Proposed Answer: B

Explanation: With Main Condenser vacuum at 9" Hg and stable, Turbine Bypass Valves are tripped closed, but MSIVs remain open. Use of the Offgas Mixing Jet for decay heat removal requires MSIVs to be open to supply steam. Therefore, Turbine Bypass Valves are unavailable for decay heat removal but the Offgas Mixing Jet is available.

A. Incorrect – Turbine Bypass Valves are not available. Plausible because they would be available if Main Condenser vacuum was slightly higher, plus MSIVs are open. The Offgas Mixing Jet is available. Plausible that use of the Offgas Mixing Jet would be precluded in some way due to the degraded Main Condenser vacuum.
C. Incorrect – Turbine Bypass Valves are not available. Plausible because they would be available if Main Condenser vacuum was slightly higher, plus MSIVs are open.
D. Incorrect – The Offgas Mixing Jet is available. Plausible that use of the Offgas Mixing Jet would be precluded in some way due to the degraded Main Condenser vacuum.

Technical Reference(s): N1-SOP-25.1, N1-EOP-1 Attachment 21

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-10

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 #	2
	K/A #	241000 K6.08
	Importance Rating	3.6

Reactor/Turbine Pressure Regulating

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

Question: #35

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

- A manual Reactor scram is inserted.
- The Reactor Mode Switch is in SHUTDOWN.
- Multiple control rods fail to scram.
- All APRM indications are lost.
- Reactor water level is 70” and stable with Feedwater injecting.
- Reactor pressure is 925 psig and stable.
- Four (4) Turbine Bypass Valves are full open and all other Turbine Bypass Valves are full closed.

Which one of the following ranges contains the approximate value of Reactor power?

A. 1-10%

B. 11-20%

C. 21-30%

D. 31-40%

Proposed Answer: B

Explanation: Total Turbine Bypass Valve capacity is approximately 40% of rated steam flow for all nine Turbine Bypass Valves. This represents a capacity of approximately 4.4% of rated steam flow for one Turbine Bypass Valve. With four valves full open, Reactor power is approximately 17.7%, or in the range of 11-20%.

A. Incorrect – Reactor power is in the range of 11-20%. Plausible because this would be correct for 2 TBVs open.
C. Incorrect – Reactor power is in the range of 11-20%. Plausible because this would be correct for 5-6 TBVs open.
D. Incorrect – Reactor power is in the range of 11-20%. Plausible because this would be correct for 7+ TBVs open.

Technical Reference(s): N1-OP-31, N1-239001-RBO-2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-248000-RBO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	256000 A1.03
	Importance Rating	2.8

Reactor Condensate System

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CONDENSATE SYSTEM controls including: System pressure

Question: #36

A plant startup is in progress with the following:

- Condensate pump 11 is running.
- Condensate pumps 12 and 13 are in standby with their control switches green-flagged.
- Condensate pump discharge pressure is 170 psig.

Then, Powerboard 11 de-energizes due to a sustained electrical fault.

Which one of the following describes the resulting status of Condensate system pressure?

Condensate system pressure…

A. is restored by automatic start of Condensate pump 12, only.

B. is restored by automatic start of Condensate pumps 12 and 13.

C. lowers to 0 psig until an additional Condensate pump is manually started. Only Condensate pump 12 is available for manual start.

D. lowers to 0 psig until an additional Condensate pump is manually started. Both Condensate pumps 12 and 13 are available for manual start.

Proposed Answer: B

Explanation: When Powerboard 11 de-energizes, power is lost to Condensate pump 11. Condensate system pressure lowers. At 120 psig, Condensate pumps 12 and 13 both automatically start to restore system pressure. These pumps are powered from Powerboards 101 and 12, respectively.

A. Incorrect – Condensate pump 13 also starts. Plausible that only Condensate pump 12 starts because only one (HPCI preferred) Feedwater Booster Pump would start under these conditions. However, both condensate pumps receive a start signal.
C. Incorrect – Condensate pumps automatically start on low system discharge pressure. Plausible because not all pumps have this automatic feature. Also plausible that more than one condensate pump is affected by the power board loss and only one is available for restart. As is the case with other system pumps, such as RBCLC.
D. Incorrect – Condensate pumps automatically start on low system discharge pressure. Plausible because not all pumps have this automatic feature.

Technical Reference(s): N1-OP-15A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-256000-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 #	2
	K/A #	288000 A2.05
	Importance Rating	2.6

Plant Ventilation

Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Extreme outside weather conditions: Plant-Specific

Question: #37

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

- A tornado warning is in effect.
- The crew has entered N1-SOP-64, High Winds.

Which one of the following describes a required action, in accordance with N1-SOP-64?

Verify all Turbine Building…

A. roof vents are open.

B. roof vents are closed.

C. ventilation fans are in fast speed.

D. ventilation fans are in slow speed.

Proposed Answer: B

Explanation: With a severe weather warning in effect, N1-SOP-64 requires verifying closed all Turbine Building roof vents.

<u>Note:</u> The first half of the K/A could not be directly tested due to lack of procedure basis information and the simplistic nature of the K/A. The question therefore focuses on testing the second half of the K/A.

A. Incorrect – N1-SOP-64 requires verifying closed all Turbine Building roof vents. Plausible that they would be opened if there was a concern about high D/P across Turbine Building walls from high winds, which could damage the walls.
C. Incorrect – N1-SOP-64 does not require the Turbine Building ventilation fans to be in a specific speed. Plausible that the fans would need to be in a specific speed to adjust building pressure in response to the high winds.
D. Incorrect – N1-SOP-64 does not require the Turbine Building ventilation fans to be in a specific speed. Plausible that the fans would need to be in a specific speed to adjust building pressure in response to the high winds.

Technical Reference(s): N1-SOP-64

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP64C01 EO-2

Question Source: Bank - 2008 NRC #34

Question History: 2008 NRC #34

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 #	290003 A3.01
	Importance Rating	3.3

Control Room Ventilation

Ability to monitor automatic operations of the CONTROL ROOM HVAC including: Initiation/reconfiguration

Question: #38

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

- Control Room Ventilation is operating in a normal alignment.
- Control Room Ventilation fan 11 motor is running and fan 12 motor is secured.

Then, a loss of coolant accident occurs.

- Reactor water level is 20" and slowly lowering.
- Drywell pressure is 5.0 psig and slowly rising.
- Control Room Ventilation radiation monitor 11 indicates 7.98 E 01 cpm and stable.
- Control Room Ventilation radiation monitor 12 indicates 8.56 E 01 cpm and stable.

Which one of the following describes the resulting status of Control Room Ventilation normal supply dampers and the Control Room Emergency Ventilation fans?

	Control Room Ventilation Normal Supply Dampers	Control Room Emergency Ventilation Fans
A.	Closed	One running, only
B.	Closed	Both running
C.	Open	One running, only
D.	Open	Both running

Proposed Answer: A

Explanation: Control Room Ventilation automatically transfers to the emergency mode when Drywell pressure exceeds 3.5 psig. Control Room Ventilation normal supply dampers close and only one Control Room Emergency Ventilation fan starts.

B. Incorrect – Only one Control Room Emergency Ventilation fan starts. Plausible both would start, such as how both RBEVS fans start on automatic initiation.
C. Incorrect – Control Room Ventilation normal supply dampers close. Plausible because they are initially open. Plausible that they would remain open to supply fresh air.
D. Incorrect – Control Room Ventilation normal supply dampers close. Plausible because they are initially open. Plausible that they would remain open to supply fresh air. Only one Control Room Emergency Ventilation fan starts. Plausible both would start, such as how both RBEVS fans start on automatic initiation.

Technical Reference(s): N1-OP-49

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288003-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AK1.04
	Importance Rating	2.5

Partial or Complete Loss of Forced Core Flow Circulation

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Limiting cycle oscillation: Plant-Specific

Question: #39

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

- Four Recirculation pumps are operating.
- The fifth Recirculation pump is available for start.
- A malfunction in the master Recirculation controller causes Recirculation flow and Reactor power to lower.
- N1-SOP-1.5, Unplanned Rx Power Change, is entered.
- A Reactor Operator places all individual Recirculation M/A stations in MANUAL and the flow/power reduction ceases.
- Annunciators F2-4-5, LPRM CH 11, and F3-4-2, LPRM CH 12, are repeatedly alarming and clearing due to LPRM upscale and downscale alarms.
- APRM indications are rising and falling with increasing magnitude and frequency.
- Reactor operation is in the uncolored region of the Four Loop Power to Flow Map.

Which one of the following actions is required?

A. Manually scram the Reactor.

B. Raise recirculation flow or insert rods with RMCS.

C. Perform a normal plant shutdown per N1-OP-43C.

D. Raise Recirculation flow by starting the idle Recirculation pump.

Proposed Answer: A

Explanation: Periodic LPRM upscale and downscale alarms indicate thermal hydraulic instability. Due to the thermal hydraulic instability, an override in N1-SOP-1.5 requires a manual Reactor scram.

B. Incorrect – A manual Reactor scram is required. Plausible because other conditions in N1-SOP-1.5 (operation in the Restricted Zone) would require exiting the region by either raising Recirculation flow or inserting control rods with RMCS.
C. Incorrect – A manual Reactor scram is required. Plausible because plant shutdown is required due to thermal hydraulic instability. However, a scram, not a normal shutdown, is required.
D. Incorrect – A manual Reactor scram is required. Plausible because this would raise flow and help mitigate the power oscillations, which are brought on by low flow conditions.

Technical Reference(s): N1-SOP-1.5, ARPs F2-4-5 and F3-4-2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP1.5C01 EO-2

Question Source: Bank – 2015 Cert #56

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 AK2.04
	Importance Rating	3.4

Partial or Complete Loss of AC Power

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

Question: #40

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

- CRD pump 11 is operating.
- CRD pump 12 is in standby.
- Liquid Poison (LP) pump 11 is injecting.
- Then, a loss of 115 KV offsite power occurs.
- Both Emergency Diesel Generators start and power their respective powerboards.

Which one of the following describes the resulting status of LP pump 11 and CRD pumps?

	LP Pump 11	CRD Pumps
A.	Running	Only CRD pump 11 is running
B.	Running	Both CRD pumps are running
C.	Tripped	Only CRD pump 11 is running
D.	Tripped	Both CRD pumps are running

Proposed Answer: B

Explanation: The loss of 115 KV power results in loss of power to PBs 102 and 103. 86-16 and 86-17 actuate on the loss of power to PBs 102 and 103 to initiate load shedding to reduce the loading on the diesels when they start and energize their busses. Liquid Poison pump 11 and CRD pump 11 are not affected by the trip of 86-16. CRD Pump 12 receives a start signal from 86-17. When power is restored to the powerboards, LP pump 11, CRD pump 11, and CRD pump 12 will start.

A. Incorrect – Both CRD pumps are running. Plausible because only one was initially running.
C. Incorrect – LP pump 11 is running. Plausible because it temporarily de-energizes due to the loss of 115 KV power and some loads are shed by 86-16/17 operation. Both CRD pumps are running. Plausible because only one was initially running.
D. Incorrect – LP pump 11 is running. Plausible because it temporarily de-energizes due to the loss of 115 KV power and some loads are shed by 86-16/17 operation.

Technical Reference(s): N1-OP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001-RBO-11

Question Source: Bank - 2010 Cert #47

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AK3.01
	Importance Rating	2.6

Partial or Total Loss of DC Power

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Load shedding: Plant-Specific

Question: #41

Which one of the following describes the reason for performing Battery load reductions during a Station Blackout?

A. Support a manual dead bus transfer to MG Set 167.

B. Maintain power to Reactor instrumentation, EC controls, and to start an EDG.

C. Avoid a loss of critical Battery Board loads due to breaker trips on over-current.

D. Maintain power to the Process Computer and Annunciators for the entire coping period.

Proposed Answer: B

Explanation: During a Station Blackout, Static Battery Chargers are lost and Batteries begin to discharge. Per N1-OP-47A, B.1, load reductions are performed to preserve Battery capacity, specifically for Reactor instrumentation, Emergency Condenser control, and EDG starting.

A. Incorrect – Plausible because MG Set 167 does operate from DC power during a Station Blackout. However, it is not transferred to any other buses. N1-SOP-33A.2 requires load shedding MG Set 167 within 2 hours of event start.
C. Incorrect – Breakers on the SR Battery Boards have been bypassed. Fuses have been installed in their place in the back of the Battery Boards. Both the Negative and Positive legs to each load are fused. Plausible if believe lowering voltage will draw more current and cause trips.
D. Incorrect – PPC and annunciators are de-energized when MG 167 is tripped 2 hours into the 4 hour coping period. Plausible because these loads are maintained for a longer period of time than most other loads.

Technical Reference(s): N1-OP-47A, N1-SOP-33A.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-10

Question Source: Bank - 2017 NRC #45

Question History: 2017 NRC #45

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295005 AA1.04
	Importance Rating	2.7

Main Turbine Generator Trip

Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: Main generator controls

Question: #42

The plant is operating at 80% power when a Main Turbine trip results in a Main Generator trip.

Which one of the following is the required operator response to this event in accordance with N1-SOP-31.1, Turbine Trip?

A. Manually close breakers R-915 and R-925 when MOD-18 is open.

B. Confirm automatic closure of R-915 and R-925 when MOD-18 is open.

C. Manually reset Generator Trip Relays 86G1 & 86G2 and manually restart Stator Water Cooling.

D. Confirm automatic reset of Generator Trips Relays 86G1 & 86G2 and automatic restart of Stator Water Cooling.

Proposed Answer: C

Explanation: N1-SOP-31.1 requires manually resetting 86G1 and 86G2. There is no automatic reset of these relays. Once these relays are reset, Stator Water Cooling is required to be manually restarted.

A. Incorrect – Plausible because R-915 and R-925 eventually need to be re-closed, but have sealed in trip signals until the generator fault is correct and 86G1 and 86G2 are reset.
B. Incorrect – Plausible because R-915 and R-925 eventually need to be re-closed, but have sealed in trip signals until the generator fault is correct and 86G1 and 86G2 are reset.
D. Incorrect – Plausible because 86G1 and 86G2 do need to be reset and Stator Water Cooling must be restarted, however these must be accomplished manually.

Technical Reference(s): N1-SOP-31.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245001-RBO-10

Question Source: Bank – 2015 Cert #58

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 #	295006 AA2.06
	Importance Rating	3.5

Scram

Ability to determine and/or interpret the following as they apply to SCRAM: Cause of reactor SCRAM

Question: #43

A plant startup is in progress with the following:

- The Reactor Mode Switch is in STARTUP.
- All IRMs indicate approximately 30 out of 125 on Range 8
- APRMs indicate 3%
- Reactor pressure is 450 psig

Then, the MSIVs close due to a spurious isolation signal, resulting in the following conditions:

- IRMs spike to 120 out of 125 on Range 8
- APRMs spike to 9%
- Reactor pressure spikes to 550 psig

Which one of the following describes the RPS signal that initiates a Reactor scram, if any?

A. IRM Hi-Hi

B. APRM Hi-Hi

C. MSIV Closure

D. NO Reactor scram occurs

Proposed Answer: A

Explanation: IRMs cause a Reactor scram on a Hi-Hi signal due to exceeding 117.5 out of 125 on Range 8. APRMs do not spike high enough to cause a Reactor scram given the startup conditions. MSIV Closure does not cause a direct scram due to Reactor pressure being below 600 psig.

B. Incorrect – IRM Hi-Hi initiates the Reactor scram. Plausible because APRMs spike high enough to clear the downscales.
C. Incorrect – IRM Hi-Hi initiates the Reactor scram. Plausible because MSIV closure would cause an anticipatory scram if Reactor pressure was >600 psig (589 psig).
D. Incorrect – IRM Hi-Hi initiates the Reactor scram. Plausible because this would be correct if IRMs stayed below 117.5 out of 125 or if the reactor mode switch were in RUN, bypassing IRM protective functions.

Technical Reference(s): ARPs F3-1-1, F3-3-1, F3-3-6,

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 2.1.32
	Importance Rating	3.8

Control Room Abandonment

Ability to explain and apply system limits and precautions.

Question: #44

A Control Room evacuation has been performed with the following:

- Control is being established at Remote Shutdown Panel 12.
- Reactor pressure is 1050 psig and slowly rising.
- Reactor water level is 90” and slowly rising.
- Emergency Condenser 12 is in standby.

Which one of the following describes the required control of Emergency Condenser 12, in accordance with N1-SOP-21.2, Control Room Evacuation?

Emergency Condenser 12…

A. must NOT be manually initiated until Reactor water level is lowered. The steam isolation valves may remain open.

B. must NOT be manually initiated until Reactor water level is lowered. At least one steam isolation valve must be manually closed.

C. may be manually initiated. If Reactor water level rises more than 5”, then the Emergency Condenser must be secured.

D. may be manually initiated. If Reactor water level rises more than 5”, then the Emergency Condenser may be left in service.

Proposed Answer: D

Explanation: N1-SOP-21.2 cautions against initiating an Emergency Condenser with Reactor water level above 95”. Since Reactor water level is less than 95”, the Emergency Condenser may be manually initiated. Once initiated, there is no requirement to remove the Emergency Condenser from service if Reactor water level reaches 95”.

Note: The question tests the “apply” part of the generic K/A but not the “explain” part. This is in accordance with NUREG 1021 ES-401 Section D.2.a and tests the subject at the higher cognitive level.

A. Incorrect – Since Reactor water level is less than 95”, the Emergency Condenser may be manually initiated. Plausible because N1-OP-13 section H.2.0 contains a band of 85-90” in relation to initiating an EC, but only if level is initially >95”.
B. Incorrect – Since Reactor water level is less than 95”, the Emergency Condenser may be manually initiated. Plausible because N1-OP-13 section H.2.0 contains a band of 85-90” in relation to initiating an EC, but only if level is initially >95”.
C. Incorrect – Once initiated, there is no requirement to remove the Emergency Condenser from service if Reactor water level reaches 95”. Plausible because as Reactor water level rises, the efficiency of the EC is reduced.

Technical Reference(s): N1-SOP-21.2, N1-OP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-10

Question Source: Bank – 2015 Cert #53

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AA2.05
	Importance Rating	2.9

Partial or Complete Loss of CCW

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

Question: #45

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

- Turbine Building Closed Loop Cooling (TBCLC) pump 11 is in service.
- TBCLC pump 12 is in standby.
- TBCLC discharge header pressure is 110 psig and stable.
- Then, TBCLC pump 11 begins to fail.
- TBCLC discharge header pressure lowers to 75 psig.

Which one of the following describes the resulting status of TBCLC pump 12?

TBCLC pump 12…

A. auto-starts due to a low discharge header pressure signal.

B. remains in standby and will only auto-start if TBCLC pump 11 breaker trips.

C. remains in standby, but will auto-start if discharge header pressure lowers further.

D. remains in standby unless Operators manually start it.

Proposed Answer: A

Explanation: TBCLC pumps have an auto-start feature on a low discharge pressure of 80 psig. Since header pressure is below this value, TBCLC pump 12 auto-starts.

B. Incorrect – TBCLC pump 12 auto-starts. Plausible because not all pumps have auto-starts on discharge pressure.
C. Incorrect – TBCLC pump 12 auto-starts. Plausible because this would be correct if pressure were degraded but still above 80 psig.
D. Incorrect – TBCLC pump 12 auto-starts. Plausible because this would be correct for the RBCLC system.

Technical Reference(s): N1-OP-24

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-274000-RBO-5

Question Source: Bank – 2015 NRC #7

Question History: 2015 NRC #7

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 #	295019 2.4.47
Importance Rating	4.2

Partial or Complete Loss of Instrument Air

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

Question: #46

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

- An Instrument Air leak has developed.
- All available Instrument Air compressors are running.
- The following Instrument Air system pressures are observed:

Time (hh:mm)	Instrument Air System Pressure (PI-94-15A) (psig)
08:00	98
08:02	94
08:04	90

Note: Assume the Instrument Air system pressure trend remains constant.

Which one of the following identifies when the threshold for inserting a manual Reactor scram will first be reached, in accordance with N1-SOP-20.1, Instrument Air Failure?

A. 08:05

B. 08:14

C. 08:17

D. 08:19

Proposed Answer: B

Explanation: Instrument Air pressure is lowering 4 psig every 2 minutes, or 2 psig/minute. The threshold requiring a Reactor scram is 70 psig. This is 20 psig below the value at 08:04. It will take 10 minutes to lower 20 psig. Therefore, 70 psig will be reached at 08:14.

A. Incorrect – The threshold of 70 psig will be reached at 08:14. Plausible because this is the first time pressure will be less than 89 psig, which is a value in N1-SOP-20.1 for verifying valve closures.
C. Incorrect – The threshold of 70 psig will be reached at 08:14. Plausible because this is the first time pressure will be less than 65 psig, which is a related alarm value in ARP F3-3-2.
D. Incorrect – The threshold of 70 psig will be reached at 08:14. Plausible because this is the first time pressure will reach 60 psig, which is a value in N1-SOP-20.1 requiring a scram for scram air header pressure.

Technical Reference(s): N1-SOP-20.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP20.1C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AK1.04
	Importance Rating	3.6

Loss of Shutdown Cooling

Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING: Natural circulation

Question: #47

A refueling outage is in progress with the following:

- The Reactor head is still installed.
- Reactor water level is at flange level and stable.
- All Reactor Recirculation pumps have tripped.
- All Reactor Recirculation pump discharge valves have been closed.
- All Reactor Recirculation pump suction and discharge bypass valves are open.
- The only running Shutdown Cooling pump has tripped.

Which one of the following identifies the adequacy of current conditions for maintaining valid Reactor water level indication and preventing thermal stratification, in accordance with N1-OP-4, Shutdown Cooling System?

Current conditions are…

A. adequate to both maintain valid Reactor water level indication and prevent thermal stratification.

B. adequate to maintain valid Reactor water level indication, but NOT to prevent thermal stratification.

C. adequate to prevent thermal stratification, but NOT to maintain valid Reactor water level indication.

D. NEITHER adequate to maintain valid Reactor water level indication NOR prevent thermal stratification.

Proposed Answer: B

Explanation: To maintain valid Reactor water level indication, N1-OP-4 P&L 7 requires one of the following conditions:

- the suction and discharge valves of at least two Recirc loops to be maintained open (not currently satisfied)
- Reactor water level to be above the Main Steam Line nozzles (satisfied since Reactor water level is at flange level and stable)
- The Steam Dryer and Separator removed (not currently satisfied since the Reactor head is still installed)

To prevent thermal stratification, N1-OP-4 P&L 8 requires one of the following conditions:

- Forced circulation with at least one Reactor Recirc pump running (not currently satisfied, although natural circulation exists)
- Reactor water level above Main Steam Line nozzles with Shutdown Cooling in service and all Recirc suction or discharge and discharge bypass valves closed (not currently satisfied)
- Subsection H.4.0 to be performed (not possible with Reactor head installed)

A. Incorrect – Conditions are not adequate to ensure thermal stratification is prevented. Plausible because some amount of natural circulation does exist, but not enough to satisfy the requirements of N1-OP-4.
C. Incorrect – Conditions are adequate to ensure valid Reactor water level instrumentation. Plausible because two of the possible ways to ensure this are NOT met. Conditions are not adequate to ensure thermal stratification is prevented. Plausible because some amount of natural circulation does exist, but not enough to satisfy the requirements of N1-OP-4.
D. Incorrect – Conditions are adequate to ensure valid Reactor water level instrumentation. Plausible because two of the possible ways to ensure this are NOT met.

Technical Reference(s): N1-OP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-9

Question Source: Bank - 2017 Cert #41

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295023 AK2.01
	Importance Rating	3.3

Refueling Accidents

Knowledge of the interrelations between REFUELING ACCIDENTS and the following: Fuel handling equipment

Question: #48

A refueling outage is in progress with the following:

- The Reactor Mode Switch is in REFUEL.
- All control rods are inserted.
- The Refuel Bridge is located over the core.
- The Refuel Bridge grapple is loaded with a fuel bundle.
- The Refuel Bridge grapple is in the full up position.

Which one of the following describes the ability to withdraw a control rod?

A. All control rod withdrawal is blocked to prevent inadvertent large reactivity additions to the core.

B. All control rod withdrawal is blocked to prevent over-exposure of personnel on the Refueling Bridge.

C. One control rod can be withdrawn because accident analysis for a dropped fuel bundle assures any resulting reactivity excursion remains within limits.

D. One control rod can be withdrawn, but only if the grapple remains in the full up position, to stay within the limits of the accident analysis for a dropped fuel bundle.

Proposed Answer: A

Explanation: All control rod withdrawal is blocked due to the Refuel Bridge being over the core with the grapple loaded. The basis for this interlock is to prevent inadvertent large reactivity additions to the core.

B. Incorrect – The basis for this interlock is to prevent inadvertent large reactivity additions to the core. Plausible because withdrawing a control rod could cause some additional rise in count rate and additional exposure to personnel on the Refuel Bridge, however this is not the basis for the interlock.
C. Incorrect – All control rod withdrawal is blocked due to the Refuel Bridge being over the core with the grapple loaded. Plausible because with the grapple in the full up position, two simultaneous positive reactivity additions to the core are prevented. However, the interlock is not set to include grapple position to allow control rod withdrawal in this situation.
D. Incorrect – All control rod withdrawal is blocked due to the Refuel Bridge being over the core with the grapple loaded. Plausible because with the grapple in the full up position, two simultaneous positive reactivity additions to the core are prevented. However, the interlock is not set to include grapple position to allow control rod withdrawal in this situation.

Technical Reference(s): Technical Specification 3.5.2 and associated Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-234001-RBO-5

Question Source: Bank - 2015 Cert #47

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EK3.06
	Importance Rating	4.0

High Drywell Pressure

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Reactor SCRAM

Question: #49

Which one of the following describes:

(1) the basis for the Reactor scram on high Drywell pressure, in accordance with the Updated Final Safety Analysis Report (UFSAR),

and

(2) the reason for including high Drywell pressure as an entry condition for N1-EOP-2, RPV Control, in accordance with NER-1M-095, NMP1 Emergency Operating Procedures and Severe Accident Procedures Basis Document?

The basis for the Reactor scram on high Drywell pressure is the analysis for a ___(1)___.

The reason for including high Drywell pressure as an entry condition for N1-EOP-2 is because it may indicate a challenge to ___(2)___.

A. (1) complete loss of Drywell cooling.
(2) adequate core cooling.

B. (1) complete loss of Drywell cooling.
(2) Reactor vessel level indication.

C. (1) Reactor coolant system break in the Drywell.
(2) adequate core cooling.

D. (1) Reactor coolant system break in the Drywell.
(2) Reactor vessel level indication.

Proposed Answer: C

Explanation: UFSAR chapter 8 states that the high Drywell pressure scram is based on the analysis for a Reactor coolant system break in the Drywell. NER-1M-095 states that the reason for including high Drywell pressure as an entry condition for N1-EOP-2 is because it may indicate a challenge to adequate core cooling.

A. Incorrect – UFSAR chapter 8 states that the high Drywell pressure scram is based on the analysis for a Reactor coolant system break in the Drywell. Plausible because another transient that could lead to high Drywell pressure is a complete loss of Drywell cooling.
B. Incorrect – UFSAR chapter 8 states that the high Drywell pressure scram is based on the analysis for a Reactor coolant system break in the Drywell. Plausible because another transient that could lead to high Drywell pressure is a complete loss of Drywell cooling. NER-1M-095 states that the reason for including high Drywell pressure as an entry condition for N1-EOP-2 is because it may indicate a challenge to adequate core cooling. Plausible because high Drywell pressure/temperature do affect Reactor water level indications and this is addressed in N1-EOP-2.
D. Incorrect – NER-1M-095 states that the reason for including high Drywell pressure as an entry condition for N1-EOP-2 is because it may indicate a challenge to adequate core cooling. Plausible because high Drywell pressure/temperature do affect Reactor water level indications and this is addressed in N1-EOP-2.

Technical Reference(s): UFSAR Chapter 8, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank - JAF 16-1 NRC #46

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295024 EK3.06
	Importance Rating	4.0

High Drywell Pressure

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Reactor SCRAM

Proposed Question: #46

Which one of the following describes the reason for the Reactor scram on Drywell high pressure, in accordance with the Final Safety Analysis Report (FSAR)?

A. Decrease the probability of exceeding Primary Containment design limits following a complete loss of Drywell cooling.

B. Prevent the loss of equipment inside the Drywell needed for accident mitigation following a complete loss of Drywell cooling.

C. Prevent the loss of equipment inside the Drywell needed for accident mitigation following a break in the Reactor Coolant Pressure Boundary.

D. Decrease the probability of fuel damage following a break in the Reactor Coolant Pressure Boundary.

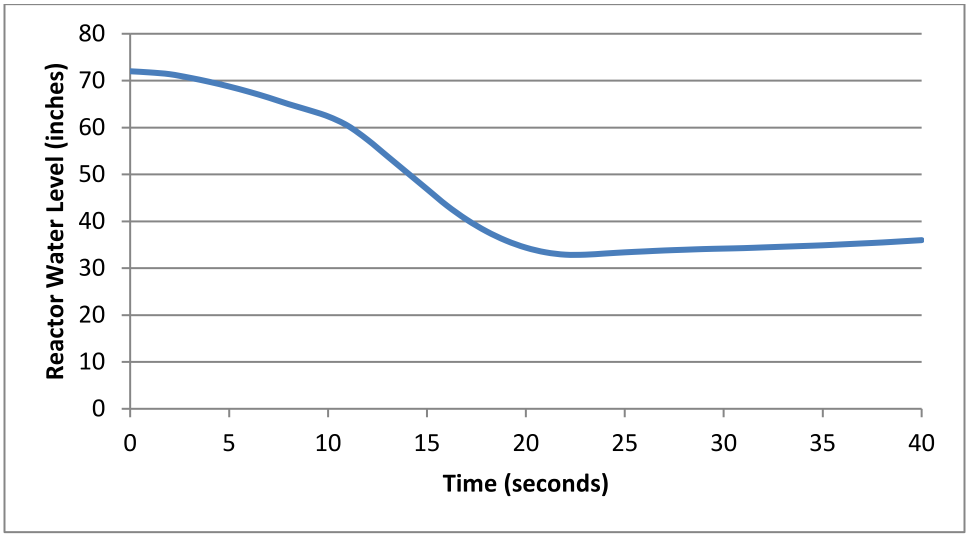
Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 EA1.06
	Importance Rating	4.5

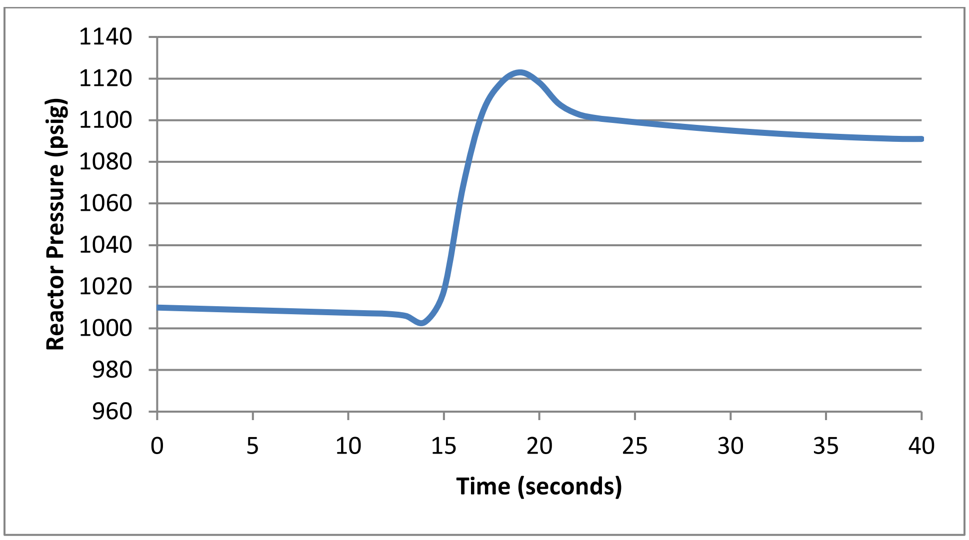
High Reactor Pressure

Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: Isolation condenser: Plant-Specific

Question: #50

The plant is operating at 30% power when a transient causes the following Reactor water level and pressure responses:





Which one of the following describes the status of Emergency Condenser logic during this event?

Emergency Condensers…

A. do NOT receive an initiation signal during the event.

B. receive an initiation signal based on high Reactor pressure, only.

C. receive an initiation signal based on low Reactor water level, only.

D. receive initiation signals based on both high Reactor pressure and low Reactor water level.

Proposed Answer: B

Explanation: Emergency Condensers initiate on either high Reactor pressure (>1080 psig for 12 seconds) or low Reactor water level (<+5" for 12 seconds). The given graphs show that Reactor pressure is >1080 psig for >12 seconds, therefore the Emergency Condensers receive an initiation signal based on high Reactor pressure. However, Reactor water level does not lower below +5", therefore an initiation signal based on low Reactor water level is NOT received.

A. Incorrect – The given graphs show that Reactor pressure is >1080 psig for >12 seconds, therefore the Emergency Condensers receive an initiation signal based on high Reactor pressure. Plausible because Reactor pressure spikes over the highest ERV set pressure (1100 psig) for less than 12 seconds.
C. Incorrect – The given graphs show that Reactor pressure is >1080 psig for >12 seconds, therefore the Emergency Condensers receive an initiation signal based on high Reactor pressure. Plausible because Reactor pressure spikes over the highest ERV set pressure (1100 psig) for less than 12 seconds. Reactor water level does not lower below +5", therefore an initiation signal based on low Reactor water level is NOT received. Plausible because Reactor water level does go below the scram and HPCI initiation setpoint of 53".
D. Incorrect – Reactor water level does not lower below +5", therefore an initiation signal based on low Reactor water level is NOT received. Plausible because Reactor water level does go below the scram and HPCI initiation setpoint of 53".

Technical Reference(s): N1-OP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-5

Question Source: Bank - 2017 Cert #46

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EA2.03
	Importance Rating	3.9

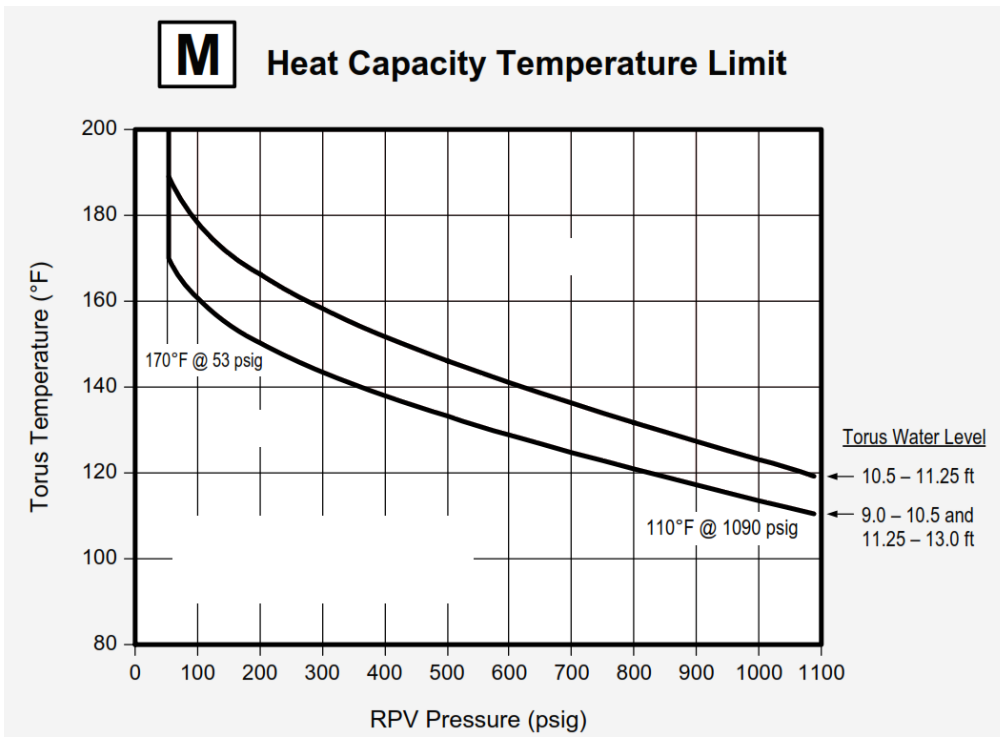
Suppression Pool High Water Temperature

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

Question: #51

The plant has experienced an accident with the following conditions:

Parameter	Time 1	Time 2
Reactor Pressure	675 psig	825 psig
Torus Water Temperature	133ºF	135ºF
Torus Water Level	11.0’	11.1’



Which one of the following describes the operating point on the Heat Capacity Temperature Limit (HCTL) at Times 1 and 2, in accordance with N1-EOP-4, Primary Containment Control?

	Time 1	Time 2
A.	GOOD	GOOD
B.	GOOD	BAD
C.	BAD	GOOD
D.	BAD	BAD

Proposed Answer: B

Explanation: At Time 1, with Torus water temperature at 133°F and Reactor pressure at 675 psig, operation is below the applicable Torus level curve (10.5-11.25'). This is the GOOD side of the curve. At Time 2, with Torus water temperature at 135°F and Reactor pressure at 825 psig, operation is above the applicable Torus level curve (10.5-11.25'). This is the BAD side of the curve.

A. Incorrect – Conditions are BAD at Time 2. Plausible if candidate mis-plots the point or misunderstands which side of the curve is GOOD/BAD.
C. Incorrect – Conditions are GOOD at Time 1. Plausible if candidate mis-plots the point, uses the wrong curve, or misunderstands which side of the curve is GOOD/BAD. Conditions are BAD at Time 2. Plausible if candidate mis-plots the point or misunderstands which side of the curve is GOOD/BAD.
D. Incorrect – Conditions are GOOD at Time 1. Plausible if candidate mis-plots the point, uses the wrong curve, or misunderstands which side of the curve is GOOD/BAD.

Technical Reference(s): N1-EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank – PB 2017 NRC #57

Question History: PB 2017 NRC #57

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295028 2.4.3
	Importance Rating	3.7

High Drywell Temperature

Ability to identify post-accident instrumentation.

Question: #52

The plant is shutdown with the following:

- A loss of coolant accident has occurred.
- All Drywell temperature indication in the Control Room has been lost.

Which one of the following describes an alternate location for determining Drywell temperature, in accordance with N1-SOP-29.1, EOP Key Parameter – Alternate Instrumentation?

A. East Instrument Room

B. West Instrument Room

C. North Instrument Room

D. Remote Shutdown Panels

Proposed Answer: D

Explanation: There is alternate Drywell temperature indication at the Remote Shutdown Panels, but not in any of the Instrument Rooms.

A. Incorrect – There is no alternate Drywell temperature indication in the East Instrument Room. Plausible because this room does contain many alternate indicators listed in N1-SOP-29.1.
B. Incorrect – There is no alternate Drywell temperature indication in the West Instrument Room. Plausible because this room does contain many alternate indicators listed in N1-SOP-29.1.
C. Incorrect – There is no alternate Drywell temperature indication in the North Instrument Room. Plausible because this room does contain many alternate indicators listed in N1-SOP-29.1.

Technical Reference(s): N1-SOP-29.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223000-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 #	1
	Group #	1
	K/A #	295030 EK1.02
	Importance Rating	3.5

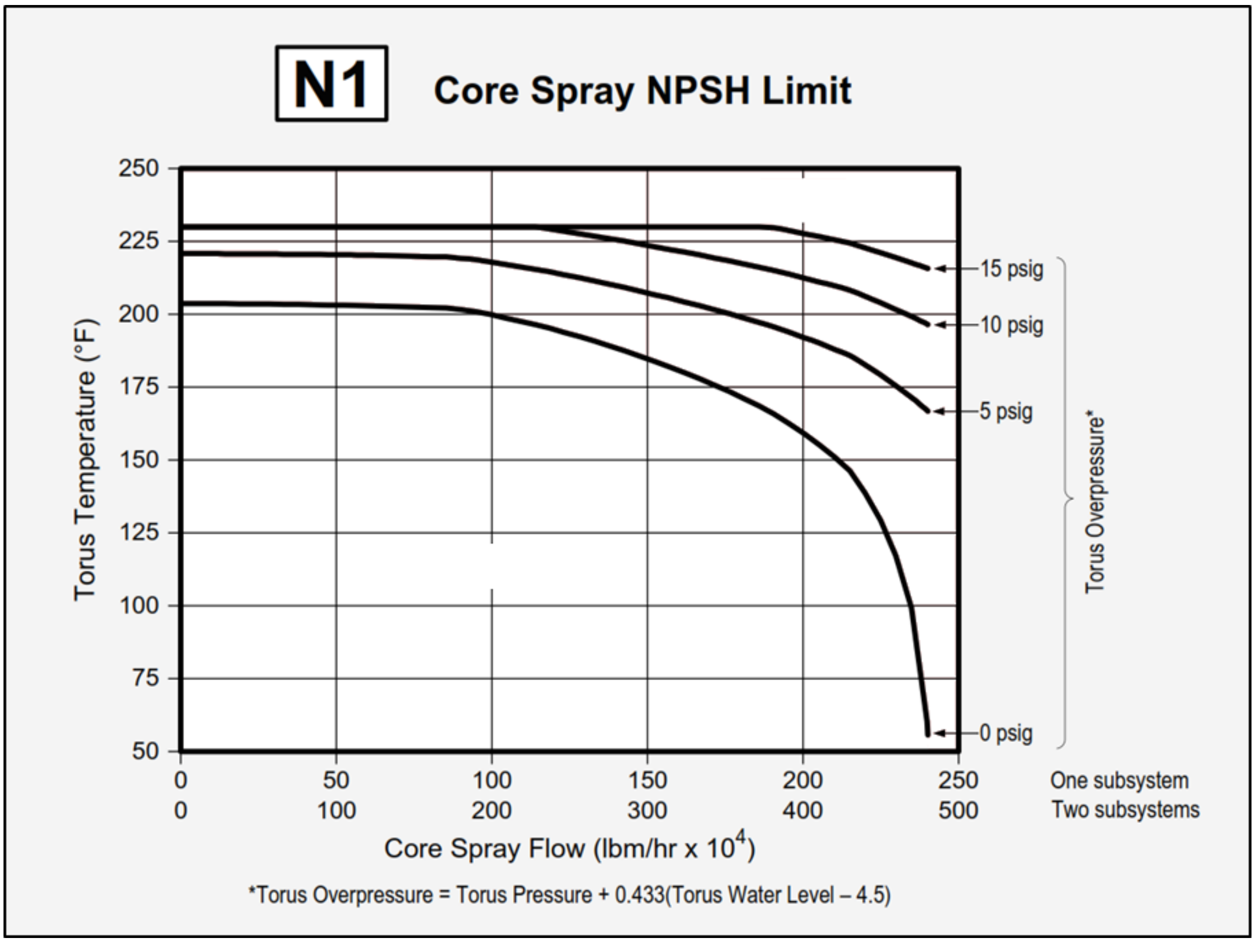
Low Suppression Pool Water Level

Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: Pump NPSH

Question: #53

A loss of coolant accident has resulted in the following:

- The Reactor has been depressurized using Emergency Condensers and ERVs.
- Core Spray is injecting and maintaining Reactor water level.
- Containment Sprays have been utilized to lower Containment pressure.
- Torus water temperature is 175°F and stable.
- Torus water level is 8.5 feet and stable.
- Torus pressure is 4 psig and slowly rising.
- Drywell pressure is 5 psig and slowly rising.



Which one of the following states the maximum Core Spray flow (lbm/hr x 10^4) that may be used for Reactor injection while maintaining Core Spray within the NPSH limit?

A. 170

B. 230

C. 340

D. 460

Proposed Answer: D

Explanation: The given combination of Torus pressure and Torus water level result in a Torus overpressure of 5.7 psig (4 psig + 0.433(8.5-4.5)). Since this is greater than 5 psig but less than 10 psig, the 5 psig curve must be used on the Core Spray NPSH Limit. With a Torus temperature of 175ºF and all Core Spray subsystems available, this allows a maximum Core Spray flow of 460 lbm/hr x10^4.

A. Incorrect – The given conditions allow a maximum Core Spray flow of 460 lbm/hr x10^4. 170 lbm/hr x10^4 would be the limit if only one Core Spray subsystem was available.
B. Incorrect – The given conditions allow a maximum Core Spray flow of 460 lbm/hr x10^4. 230 lbm/hr x10^4 would be the limit if only one Core Spray subsystem was available and the 5 psig overpressure curve were applicable.
C. Incorrect – The given conditions allow a maximum Core Spray flow of 460 lbm/hr x10^4. 340 lbm/hr x10^4 would be the limit if the 0 psig overpressure curve were applicable.

Technical Reference(s): N1-EOP-2, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: Modified Bank - 2017 NRC #50

Question History:

Question Cognitive Level: Comprehension or Analysis

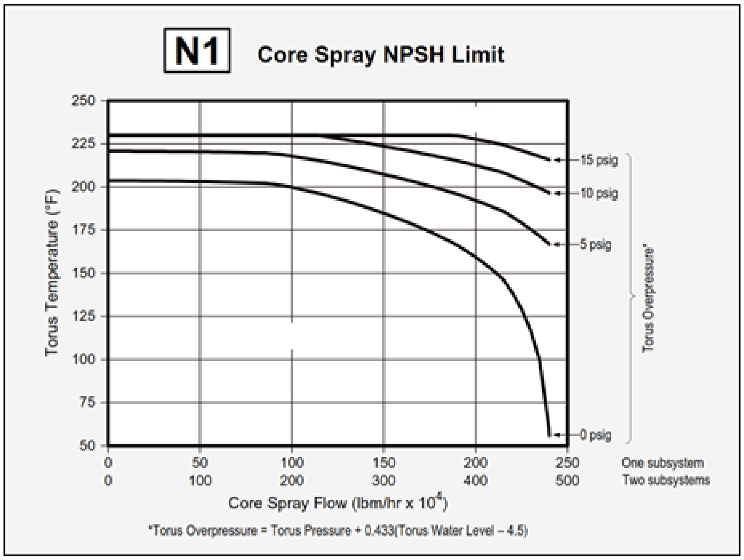
10 CFR Part 55 Content: 55.41(10)

Comments:

Proposed Question: #50

A loss of coolant accident has resulted in the following:

- The Reactor has been depressurized using Emergency Condensers and ERVs.
- Core Spray is injecting and maintaining Reactor water level.
- Containment Sprays have been utilized to lower Containment pressure.
- Torus water temperature is 175°F and stable.
- Torus water level is 8.5 feet and stable.
- Torus pressure is 2 psig and slowly rising.
- Drywell pressure is 3 psig and slowly rising.



Which one of the following states the maximum Core Spray flow (lbm/hr x 10^4) that may be used for Reactor injection while maintaining Core Spray within the NPSH limit?

A. 170

B. 230

C. 340

D. 460

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK2.02
	Importance Rating	3.8

Reactor Low Water Level

Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Reactor pressure

Question: #54

N1-EOP-2 contains the following step:

“If RPV Blowdown (EOP-8) is anticipated, then rapidly depressurize the RPV using EC and the main turbine bypass valves.”

Which one of the following identifies a parameter on which anticipating RPV Blowdown per this step is NOT appropriate, in accordance with OP-NM-101-111-1001, Transient Mitigation Guidelines?

A. Rising Torus pressure

B. Lowering Torus water level

C. Lowering Reactor water level

D. Rising Reactor Building temperature

Proposed Answer: C

Explanation: OP-NM-101-111-1001 provides guidance on how to implement the anticipatory Blowdown in N1-EOP-2. OP-NM-101-111-1001 specifically prohibits use of anticipatory Blowdown in the case of Reactor water level lowering. Low Reactor water level is called out as the lone case where anticipatory Blowdown is not appropriate since inventory conservation is the key to mitigating low Reactor water level.

A. Incorrect – OP-NM-101-111-1001 does not caution against anticipating Blowdown on rising Torus pressure. Plausible because this is a parameter in the EOPs that can lead to an RPV Blowdown.
B. Incorrect – OP-NM-101-111-1001 does not caution against anticipating Blowdown on lowering Torus water level. Plausible because this is a parameter in the EOPs that can lead to an RPV Blowdown.
D. Incorrect – OP-NM-101-111-1001 does not caution against anticipating Blowdown on rising Reactor Building temperature. Plausible because this is a parameter in the EOPs that can lead to an RPV Blowdown.

Technical Reference(s): OP-NM-101-111-1001

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: Modified Bank – 2013 Cert #85

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference: Level SRO

Tier #	1
Group #	2
K/A #	295009 2.1.20
Importance Rating	4.6

Low Reactor Water Level

Ability to interpret and execute procedure steps.

Proposed Question: #85

N1-EOP-2, RPV Control, contains the following step:

IF	THEN
RPV Blowdown (EOP-8) is <u>anticipated</u>	Rapidly depressurize the RPV using EC and the main turbine bypass valves. ⚠ **Hi/Lo – Lo/Lo Rosemounts may be unreliable following rapid depressurization below 500 psig.** ☛ OK to exceed 100°F/hr cooldown.

Given the following parameters approaching limits that will require RPV Blowdown:

(1) Low Torus water level
(2) Low Reactor water level

Which one of the following identifies the ability to anticipate RPV Blowdown and rapidly depressurize the RPV per this step, in accordance with GAI-OPS-20, Transient Mitigation Guidelines?

	<u>Low Torus Water Level</u>	<u>Low Reactor Water Level</u>
A.	Rapid depressurization NOT allowed	Rapid depressurization NOT allowed
B.	Rapid depressurization NOT allowed	Rapid depressurization allowed
C.	Rapid depressurization allowed	Rapid depressurization NOT allowed
D.	Rapid depressurization allowed	Rapid depressurization allowed

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

Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

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

Question: #55

A failure to scram has occurred with the following:

- Reactor power is 15% and stable.
- The Reactor Mode Switch is in SHUTDOWN.
- The RPS scram pushbuttons have been depressed.
- ARI has been initiated.
- No control rods have inserted.
- All of the white Scram Solenoid Group lights are extinguished.
- All of the blue SCRAM lights on the full core display are extinguished.
- All of the amber accumulator lights on the full core display are extinguished.
- Reactor pressure is 425 psig and stable.
- No CRD pumps are available.

Which one of the following methods in N1-EOP-3.1, Alternate Control Rod Insertion, is available to insert control rods?

A. Attachment 1, Scram Control Rods Electrically

B. Attachment 2, Scram Control Rods by Venting the Scram Air Header

C. Attachment 4, Scram Control Rods by Repeated Manual Scram Signals

D. Attachment 6, Scram Control Rods by Increasing Cooling Water Differential Pressure

Proposed Answer: B

Explanation: The given indications show that the RPS scram groups de-energized (all white lights extinguished), but the scram valves did not open (all blue lights extinguished) and the accumulators did not discharge (all amber lights extinguished). This is indicative of a failure of the scram air header to depressurize. Venting the scram air header per EOP-3.1 attachment 2 is available to insert control rods.

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

A. Incorrect – The given indications show that the RPS scram groups are already de-energized (all white lights extinguished) without causing control rod insertion. Pulling RPS fuses per EOP-3.1 attachment 1 will not result in any change since RPS scram groups are already de-energized. Plausible because this is the correct answer in other failure to scram scenarios.
C. Incorrect – The given indications show that the RPS scram groups are already de-energized (all white lights extinguished) without causing control rod insertion. Repeating manual scrams per EOP-3.1 attachment 4 will not do anything to correct the failure that prevented the scram air header from depressurizing on the first scram. Plausible because this is the correct answer in other failure to scram scenarios.
D. Incorrect – Raising cooling water D/P per EOP-3.1 attachment 6 would work, however with no CRD pumps available, this method cannot be accomplished. Plausible because this is the correct answer in other failure to scram scenarios.

Technical Reference(s): N1-EOP-3.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP3C01 EO-2

Question Source: Bank – 2015 NRC #44

Question History: 2015 NRC #44

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295038 EA1.06
	Importance Rating	3.5

High Offsite Radioactivity Release Rate

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

Question: #56

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

- A plant transient is in progress.
- Annunciator H1-1-8, STACK GAS MONITORS HIGH RADIATION, is in alarm.
- The running Turbine Building Ventilation fans have tripped.
- The Unit Supervisor has entered N1-EOP-6, Radioactivity Release Control.

Which one of the following operator actions is required in accordance with N1-EOP-6 and why?

A. Restart the Turbine Building Ventilation system to direct any radioactivity release through an elevated, monitored path.

B. Restart the Turbine Building Ventilation system to minimize transferring contamination from the Turbine Building to the Reactor Building.

C. Verify the Turbine Building Ventilation system isolated to minimize the overall radiological release from the Turbine Building.

D. Verify the Turbine Building Ventilation system isolated to minimize transferring contamination from the Turbine Building to the Reactor Building.

Proposed Answer: A

Explanation: The basis document for N1-EOP-6 states the Turbine Building Ventilation is restarted to prevent an unmonitored ground release. N1-EOP-6 is entered when the ALERT condition based on off-site release rates is exceeded. The Turbine Building Ventilation system maintains a negative pressure in the Turbine Building to ensure releases from or through systems that pass through secondary containment are captured for release through the plant stack.

B. Incorrect – Turbine Building Ventilation is restarted to prevent an unmonitored ground release, not to control transfer of contamination from the Turbine Building to the Reactor Building. Plausible because if the Turbine Building is allowed to be at a more positive pressure (ventilation isolated), then leakage from the Turbine Building to the Reactor Building would increase, which would introduce more contamination into the Reactor Building.
C. Incorrect – Turbine Building Ventilation must be restarted. Plausible because isolating Turbine Building Ventilation does slow the overall release to the environment.
D. Incorrect – Turbine Building Ventilation must be restarted. Plausible because isolating Turbine Building Ventilation does slow the overall release to the environment, which may also slow the amount of contamination that is introduced to the Reactor Building through the supply ventilation intake.

Technical Reference(s): N1-EOP-6, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288002-RBO-12

Question Source: Bank - 2017 NRC #70

Question History: 2017 NRC #70

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	600000 AA2.03
	Importance Rating	2.8

Plant Fire On Site

Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Fire alarm

Question: #57

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

- Main Fire Panel Annunciator 2-1-1-3, TURB. BLDG 261 LOCAL PNL NO.3 FIRE, alarms.
- Fire Zone D-2263, TURB BLDG 289 RB CHARCOAL FILTERS, is in alarm.

Which one of the following describes the associated fire-fighting system and how it is initiated?

A. Manually initiated CO_2

B. Automatically initiated CO_2

C. Manually initiated water deluge

D. Automatically initiated water deluge

Proposed Answer: C

Explanation: This alarm is based on detectors in the RBEVS charcoal filters. These detectors initiate the given alarms, but do not automatically initiate fire suppression. The associated fire suppression system is a water deluge system that must be initiated manually by opening valves.

A. Incorrect – The associated fire suppression system is a water deluge system. Plausible because adding water to a charcoal filter is generally undesirable, such that use of CO_2 would be a likely alternative.
B. Incorrect – The associated fire suppression system is a water deluge system. Plausible because adding water to a charcoal filter is generally undesirable, such that use of CO_2 would be a likely alternative. The associated fire suppression system is manually initiated. Plausible that the same detectors that initiate the given alarms would also automatically initiate suppression, as in many other locations.
D. Incorrect – The associated fire suppression system is manually initiated. Plausible that the same detectors that initiate the given alarms would also automatically initiate suppression, as in many other locations.

Technical Reference(s): ARP 2-1-1-3, N1-OP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-8

Question Source: Bank - PB 2017 NRC #58

Question History: PB 2017 NRC #58

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 2.2.12
	Importance Rating	3.7

Generator Voltage and Electric Grid Disturbances

Knowledge of surveillance procedures.

Question: #58

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

- N1-ST-M4B, Emergency Diesel Generator 103 and PB 103 Operability Test, is in progress.
- The next step in N1-ST-M4B is to manually start Emergency Diesel Generator (EDG) 103.

Then, 115KV power Lines 1 and 4 de-energize.

Which one of the following describes the response of EDG 103?

EDG 103…

A. remains in standby.

B. starts but then trips.

C. starts and runs, but does NOT automatically load onto the bus.

D. starts, runs, and automatically loads onto the bus.

Proposed Answer: D

Explanation: Given the current place in N1-ST-M4B, multiple actions have been taken for EDG 103. The Remote Auto/Local Start Selector switch remains in REMOTE AUTO. Speed Droop has been set to 65 (normally at 0). These conditions still allow EDG 103 to start and load the bus in the event of a loss of Lines 1 and 4.

A. Incorrect – EDG 103 starts and loads onto the bus. Plausible because this would be correct if the Remote Auto/Local Start Selector switch was taken to local for this test.
B. Incorrect – EDG 103 starts and loads onto the bus. Plausible because the droop setting has been set for operation in parallel with offsite power, versus the normal setting for operation as the sole power source on the bus.
C. Incorrect – EDG 103 starts and loads onto the bus. Plausible that actions taken in the ST to this point would still allow automatic starting of the EDG but require manual loading onto the bus.

Technical Reference(s): N1-ST-M4B, N1-OP-45

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-264001-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 #	1
	Group #	2
	K/A #	295012 AK3.01
	Importance Rating	3.5

High Drywell Temperature

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE: Increased drywell cooling

Question: #59

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

- RBCLC cooling capacity has degraded.
- Drywell average temperature is 120°F and slowly rising.
- Only four (4) Drywell cooling fans are currently operating.
- N1-OP-8, Primary Containment Area Cooling System, Section H, Maintaining Drywell Average Temperature During Steady State Power Operations, is being performed.

Which one of the following describes:

(1) the upper Drywell average temperature limit to be maintained by providing additional Drywell Cooling

and

(2) the approximate Drywell average temperature change for each Drywell Cooling fan started in accordance with N1-OP-8, Primary Containment Area Cooling System?

	(1) Upper Drywell Average Temperature Limit	(2) Approximate Drywell Average Temperature Change For Each Drywell Cooling Fan Started
A.	125°F	5°F
B.	125°F	15°F
C.	150°F	5°F
D.	150°F	15°F

Proposed Answer: C

Explanation: N1-OP-8 section H sets the upper limit on Drywell average temperature as 150°F. A 5°F change in Drywell average temperature is expected for each Drywell Cooling fan that is started.

Note: The question meets the K/A by giving a rising Drywell temperature and asking about the reason for starting additional Drywell Cooling (upper temperature limit in associated procedure), as well as testing a piece of knowledge that supports understanding the reason for starting one versus two additional fans (the approximately temperature change per fan).

A. Incorrect – N1-OP-8 section H sets the upper limit on Drywell average temperature as 150°F. Plausible because 125°F is a related temperature alarm value in ARP L1-4-4.
B. Incorrect – N1-OP-8 section H sets the upper limit on Drywell average temperature as 150°F. Plausible because 125°F is a related temperature alarm value in ARP L1-4-4. A 5°F change in Drywell average temperature is expected for each Drywell Cooling fan that is started. Plausible because 15°F is close in magnitude to 5°F and could be confused with the 115°F lower Drywell average temperature limit.
D. Incorrect – A 5°F change in Drywell average temperature is expected for each Drywell Cooling fan that is started. Plausible because 15°F is close in magnitude to 5°F and could be confused with the 115°F lower Drywell average temperature limit.

Technical Reference(s): N1-OP-8

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223000-RBO-10

Question Source: Modified Bank – 2010 NRC #42

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Facility:	Nine Mile Point Unit 1
Vendor:	GE
Exam Date:	2010
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295028	EK2.04	
	Importance Rating	3.6		

Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell ventilation

Proposed Question: RO Question # 42

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

- Six (6) Drywell cooling fans are running
- Drywell average temperature is 134ºF and stable

Then, Drywell cooling fan 11 trips and cannot be restarted.

Which one of the following describes the expected magnitude of the Drywell temperature change and, based on this expected change, whether entry into EOP-4, Primary Containment Control, will be required?

	Expected Drywell Temperature Change	Entry into EOP-4
A.	5ºF	Will be required
B.	5ºF	Will NOT be required
C.	20ºF	Will be required
D.	20ºF	Will NOT be required

Proposed Answer: B

Explanation (Optional):

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295015 AA1.02
	Importance Rating	4.0

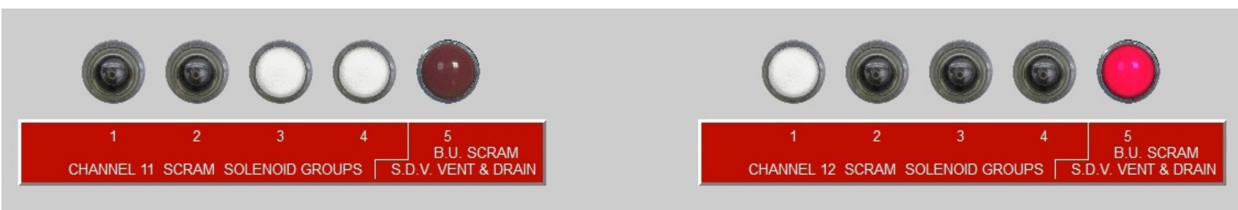
Incomplete Scram

Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: RPS

Question: #60

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

- A manual Reactor scram is required.
- An Operator has placed the Reactor Mode Switch in SHUTDOWN and depressed both manual scram pushbuttons.
- The following RPS indications are available in the Main Control Room:



Note: Assume all control rods respond properly for the status of RPS indications.

Which one of the following describes the status of control rods?

A. All control rods have inserted.

B. NO control rods have inserted.

C. Only one group of control rods have inserted.

D. Only two groups of control rods have inserted.

Proposed Answer: C

Explanation: The given indications show that RPS channel 11 has failed to de-energize the Group 3 and 4 scram solenoids, and RPS channel 12 has failed to de-energize the Group 1 scram solenoids and the channel 12 backup scram valve solenoids. This results in only the Group 2 control rods inserting because (1) all other Groups have one side of RPS still energizing their scram solenoids and (2) one backup scram solenoid remains energized.

A. Incorrect – Only Group 2 control rods have inserted. Plausible because many lights are extinguished, including one of the backup scram valve solenoids. However, both backup scram vale solenoids must be de-energized to insert all control rods.
B. Incorrect – Group 2 control rods have inserted. Plausible because many lights are still lit when they should be extinguished, however the combination is sufficient to at least insert Group 2 control rods.
D. Incorrect – Only Group 2 control rods have inserted. Plausible because many lights are extinguished, however the given combination gives only one Group of control rods with both sides of RPS de-energizing their scram solenoids.

Technical Reference(s): N1-OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-5

Question Source: Bank – 2017 Cert #21

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 #	295022 AA2.01
	Importance Rating	3.5

Loss of Control Rod Drive Pumps

Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS: Accumulator pressure

Question: #61

A plant startup is in progress with the following:

Time (hh:mm)	Conditions
09:27	• Preparations are in progress for starting the first Feedwater pump.
09:30	• CRD pump 11 trips. • Annunciators F3-1-2, CONTROL ROD DRIVE PUMP 11 TRIP-VIB, and F3-1-5, CRD CHARGING WTR PRESSURE HI/LO, alarm.
09:32	• CRD pump 12 does not start when its control switch is placed to START.
09:34	• Annunciator F3-2-5, CRD ACCUMULATOR LEVEL HIGH PRESS LOW, alarms. • Control rod 10-19 accumulator light is lit. • Control rod 10-19 accumulator pressure is reported as 900 psig and lowering.
09:37	• Control rod 34-35 accumulator light is lit • Control rod 34-35 accumulator pressure is reported as 920 psig and lowering.

Which one of the following lists the time at which a manual scram is first required, in accordance with N1-SOP-5.1, Loss of Control Rod Drive?

A. 09:32

B. 09:34

C. 09:52

D. 09:54

Proposed Answer: B

Explanation: Reactor pressure is below 350 psig based on the point in the startup when the first Feedwater pump is started. With Reactor pressure less than 900 psig, no CRD pump running, and an accumulator low pressure alarm, an immediate Reactor scram is first required at time 09:34.

A. Incorrect – An immediate Reactor scram is first required at time 09:34. Plausible because at this time Reactor pressure is below 900 psig and no CRD pump can be started.
C. Incorrect – An immediate Reactor scram is first required at time 09:34. Plausible because this is 20 minutes after it is determined that no CRD pump can be started. 20 minutes is a time used in N1-SOP-5.1 to determine scram requirements if Reactor pressure is higher.
D. Incorrect – An immediate Reactor scram is first required at time 09:34. Plausible because this would be correct if Reactor pressure was above 900 psig.

Technical Reference(s): N1-SOP-5.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP5.1C01 EO-2

Question Source: Bank – 2009 NRC #25

Question History: 2009 NRC #25

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295032 2.1.20
	Importance Rating	4.6

High Secondary Containment Area Temperature

Ability to interpret and execute procedure steps.

Question: #62

The plant has experienced a scram with the following:

- A break has developed in the Scram Discharge Volume.
- The Reactor scram CANNOT be reset.
- Reactor Building elevation 237' west general area temperature is 120°F and slowly rising.
- Reactor Building elevation 237' east general area temperature is 95°F and slowly rising.
- Reactor Building north-west corner room area water level is 2 feet and slowly rising.
- Reactor Building south-west corner room area water level is 3 inches and slowly rising.

Which one of the following describes the need for an RPV Blowdown, in accordance with N1-EOP-5, Secondary Containment Control?

An RPV Blowdown is…

A. required due to high area water levels.

B. required due to high general area temperatures.

C. NOT required because the break is NOT in a primary system.

D. NOT required because area temperatures and water levels are below the Max Safe Values.

Proposed Answer: D

Explanation: N1-EOP-5 would be entered in this situation due to high water levels and high area temperatures. N1-EOP-5 requires an RPV Blowdown if all of the following conditions are met:

- Primary system discharging into the Reactor Building
- The discharge CANNOT be isolated
- Two or more General areas are above Maximum Safe Value of the same parameter (temperature - 135°F, water level - 5 feet)

The given situation meets the first two criteria, but not the last. When a scram is present, the Scram Discharge Volume is directly connected to the RPV such that lowering RPV pressure would reduce the leak rate from the break. This meets the definition of a primary system. Since the scram cannot be reset, the break cannot be isolated. However, neither the given temperatures or water levels are above the Maximum Safe Values. Therefore, no RPV Blowdown is required.

A. Incorrect – An RPV Blowdown is not currently required. Plausible because an unisolable primary system is discharging into the Reactor Building and causing elevated area water levels.
B. Incorrect – An RPV Blowdown is not currently required. Plausible because an unisolable primary system is discharging into the Reactor Building and causing elevated area temperatures.
C. Incorrect – The Scram Discharge Volume is a primary system when a scram is present. Plausible because when the scram is reset, the Scram Discharge Volume is not in contact with the primary system (normal condition).

Technical Reference(s): N1-EOP-5, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO-2

Question Source: Bank - 2013 Cert #26

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295035 EK1.01
	Importance Rating	3.9

Secondary Containment High Differential Pressure

Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Secondary containment integrity

Question: #63

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

- A malfunction of the Reactor Building differential pressure controller occurs.
- Reactor Building differential pressure rises to +0.1” H_2O and remains there for two (2) minutes.
- Then, Operators take manual control of Reactor Building differential pressure and restore it to a normal value.

Which one of the following describes:

(1) whether the Reactor Building blowout panels ruptured during this transient,

and

(2) whether entry into N1-EOP-5, Secondary Containment Control, is required based on this transient?

	(1) Reactor Building Blowout Panels	(2) N1-EOP-5 Entry
A.	Did NOT rupture	Required
B.	Did NOT rupture	NOT required
C.	Ruptured	Required
D.	Ruptured	NOT required

Proposed Answer: A

Explanation: The Reactor Building blowout panels are designed to rupture at 65 psf. This equates to well over +0.1” H_2O. Therefore the blowout panels did not rupture. N1-EOP-5 entry is required based on Reactor Building D/P above 0”.

B. Incorrect – N1-EOP-5 entry is required. Plausible because D/P was only above 0” for a short period of time and no other adverse conditions exist in the Reactor Building (leakage, high rad, etc.).
C. Incorrect – The Reactor Building blowout panels did not rupture. Plausible because they would if D/P rose further.
D. Incorrect – The Reactor Building blowout panels did not rupture. Plausible because they would if D/P rose further. N1-EOP-5 entry is required. Plausible because D/P was only above 0” for a short period of time and no other adverse conditions exist in the Reactor Building (leakage, high rad, etc.).

Technical Reference(s): N1-EOP-5, EP-CE-114-100-F-5, N1-290001-RBO-2, EP-AA-114-F-07

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295036 EK2.01
	Importance Rating	3.1

Secondary Containment High Sump/Area Water Level

Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL and the following: Secondary containment equipment and floor drain system

Question: #64

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

- Annunciator H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH, alarms.
- Computer point B128, RBFDT 11 (NW) LVL, is in alarm high.
- The alarm has been confirmed by an operator in the field

Which one of the following describes the requirement to enter N1-EOP-5, Secondary Containment Control, based on these conditions and the required operation of the associated sump pump?

N1-EOP-5 entry is ___(1)___.

The associated sump pump is required to be ___(2)___.

A. (1) required based on these conditions alone
(2) running

B. (1) required based on these conditions alone
(2) isolated

C. (1) NOT required based on these conditions unless they are also accompanied by area water level above floor level
(2) running

D. (1) NOT required based on these conditions unless they are also accompanied by area water level above floor level
(2) isolated

Proposed Answer: A

Explanation: Any one Reactor Building floor drain water level above the H2-2-1 alarm setpoint requires entry into N1-EOP-5. There is a separate N1-EOP-5 entry condition based on area water level above 0" (floor level). With sump water level high, the associated sump pump should be running.

B. Incorrect – The sump pump is required to be running and not isolated. Plausible because Drywell floor drain sump pumps automatically isolate based on a LOCA signal to contain leakage.
C. Incorrect – N1-EOP-5 entry is required. Plausible because there is a separate N1-EOP-5 entry condition based on area water level above 0" (floor level) that is not currently indicated.
D. Incorrect – N1-EOP-5 entry is required. Plausible because there is a separate N1-EOP-5 entry condition based on area water level above 0" (floor level) that is not currently indicated. The sump pump is required to be running and not isolated. Plausible because Drywell floor drain sump pumps automatically isolate based on a LOCA signal to contain leakage.

Technical Reference(s): ARP H2-2-1, N1-EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01

Question Source: Bank – 2017 Cert #61

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	500000 EK3.06
	Importance Rating	3.1

High Containment Hydrogen Concentration

Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Operation of wet well vent

Question: #65

A loss of coolant accident has resulted in the following:

- Hydrogen concentrations in the Containment require venting.
- Torus water level is 21' and stable.

Which one of the following describes the path required to be used to vent the Containment and the reason why, in accordance with N1-EOP-4, Primary Containment Control?

Vent from the…

A. Torus to minimize cycling of the Torus-to-Drywell vacuum breakers.

B. Drywell because Torus water level is too high for venting from the Torus.

C. Torus to better scrub fission products from the Containment atmosphere before release.

D. Drywell to more quickly reduce the risk of hydrogen ignition by electrical equipment operation.

Proposed Answer: C

Explanation: N1-EOP-4 Detail Z1 requires venting from the Torus as long as Torus water level is below 27 feet. This is to scrub the Containment atmosphere through the Torus water volume prior to release to lower the radioactive release.

A. Incorrect – Torus venting is required. That part is correct. However, the reason is to scrub the atmosphere. Plausible because lowering Torus pressure first does also impact the vacuum breakers in that it prevents the need for vacuum breakers to cycle.
B. Incorrect – Torus venting is required. Plausible because if Torus water level were ≥27', then Drywell venting would be required.
D. Incorrect – Torus venting is required. Plausible because if Torus water level were ≥27', then Drywell venting would be required. Also plausible because there is greater risk of hydrogen ignition in the Drywell due to presence of electrical equipment.

Technical Reference(s): N1-EOP-4, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank – 2017 NRC #61

Question History: 2017 NRC #61

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.8
	Importance Rating	3.4

Ability to coordinate personnel activities outside the control room.

Question: #66

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

- Only two Reactor Operators are available on shift.
- Both Reactor Operators are currently in the Main Control Room.
- The Unit Supervisor is also in the Main Control Room.
- A task needs to be performed in the Auxiliary Control Room
- Another task needs to be performed in the Reactor Building.

Which one of the following describes the ability of these Reactor Operators to perform these tasks, in accordance with OP-NM-103-102, Watch-Standing Practices at Nine Mile Point?

A. Neither of the available Reactor Operators may leave the Main Control Room to perform these tasks.

B. One of the available Reactor Operators may leave the Main Control Room to perform the task in the Auxiliary Control Room. Neither of the available Reactor Operators may perform the task in the Reactor Building.

C. One of the available Reactor Operators may leave the Main Control Room to perform either of these tasks, as long as one Reactor Operator always remains in the Main Control Room.

D. The available Reactor Operators may simultaneously leave the Main Control Room to perform these tasks, as long as the Unit Supervisor remains in the Main Control Room.

Proposed Answer: C

Explanation: Two Reactor Operators are required to be on shift. One of these Reactor Operator is required to be in the Main Control Room in the At-The-Controls area at all times. The second Reactor Operator may go anywhere in the Protected Area, including the Auxiliary Control Room and the Reactor Building.

A. Incorrect – The second Reactor Operator may go anywhere in the Protected Area, including the Auxiliary Control Room and the Reactor Building. Plausible because shift staffing does require two Reactor Operators.
B. Incorrect – The second Reactor Operator may also go into the Reactor Building. Plausible that the second Reactor Operator would be able to go only into the Auxiliary Control Room since it is still inside the Control Room complex and close to the Main Control Room.
D. Incorrect – One of these Reactor Operator is required to be in the Main Control Room in the At-The-Controls area at all times. Plausible that this requirement would allow short term access to the Auxiliary Control Room due to proximity to the Main Control Room.

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

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

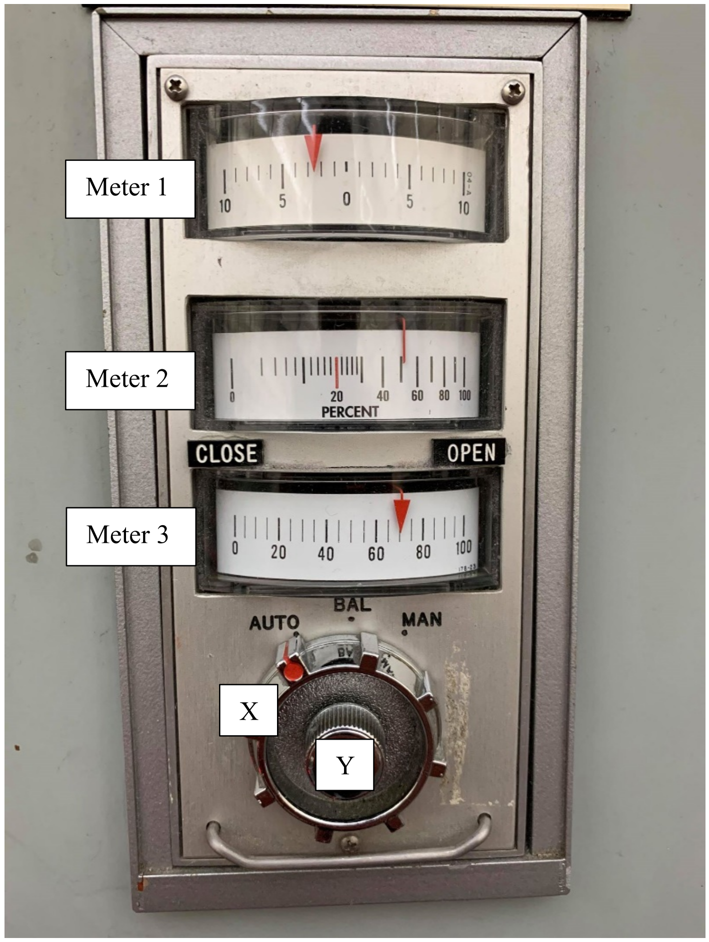
Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.28
	Importance Rating	4.1

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

Question: #67

Given the following GEMAC controller with meters labeled 1, 2, and 3 and knobs labeled X and Y:



This controller must be placed in manual for a system evolution. It is desired to make the transfer with as little system transient as possible.

Which one of the following describes the required operation of the controller to accomplish this transfer with minimum system transient?

First, …

A. rotate knob Y until the meter 1 indication is centered. Second, place knob X in MAN.

B. rotate knob Y until the meter 1 indication is downscale. Second, place knob X in MAN.

C. place knob X in BAL. Second, rotate knob Y until the meter 1 indication is centered. Third, place knob X in MAN.

D. place knob X in BAL. Second, rotate knob Y until the meter 1 indication is downscale. Third, place knob X in MAN.

Proposed Answer: C

Explanation: The controller is in Automatic, as indicated by the red dot on knob X at the AUTO position. In this mode, operation of knob Y will NOT null the controller. Therefore, the first required operation is to place knob X in BAL in order to activate the ability of knob Y to null the controller. Second, knob Y must be operated to null the controller. The controller is nulled when the meter 1 indication is centered. Only then can knob X be placed in MAN with minimum system transient.

Note: The question satisfies the K/A by testing knowledge of the function of a control that is used in multiple major systems at Nine Mile Point Unit 1. The question requires specific knowledge of how this controller works. This knowledge is beyond that required for the GFE.

A. Incorrect – First the controller must be placed in BAL. Plausible because this would be the correct answer if the controller were initially in BAL.
B. Incorrect – First the controller must be placed in BAL. Plausible because this would be the correct answer if the controller were initially in BAL. Also, the controller is nulled by getting the meter 1 indication centered, not downscale. Plausible that downscale would indicate 0 error.
D. Incorrect – The controller is nulled by getting the meter 1 indication centered, not downscale. Plausible that downscale would indicate 0 error.

Technical Reference(s): N1-202001-RBO-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-5

Question Source: Bank – 2015 NRC #66

Question History: 2015 NRC #66

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.2
	Importance Rating	4.1

Knowledge of operator responsibilities during all modes of plant operation.

Question: #68

Given the following:

- You are a licensed Reactor Operator.
- You have been on vacation for the previous 9 days.
- Today is your first shift back on watch.

Which one of the following describes the Control Room Logs that should be reviewed before assuming the shift, in accordance with OP-AA-112-101, Shift Turnover and Relief?

A. All since last watch

B. Only those from the past 7 days

C. Only those from the past 4 days

D. Only those from the past 48 hours

Proposed Answer: C

Explanation: OP-AA-112-101 Section 4.8.3 contains the following guidance:

4.8.3. Prior to relief, the on-coming Reactor Operators should **PERFORM** the following:

- **READ** the Control Room logs through the last previous date on shift, or the preceding four days logs, whichever is less.

A. Incorrect – Only the Control Room logs from the past 4 days need be reviewed. Plausible that all logs to the last shift would need to be reviewed since it has only been 9 days and this would ensure complete understanding of plant operation history during the time off.
B. Incorrect – Only the Control Room logs from the past 4 days need be reviewed. Plausible that all logs for the past week need to be reviewed this would ensure a more thorough understanding of plant operation history during the time off.
D. Incorrect – More logs than just those from the past 48 hours need to be reviewed. Plausible because this would likely cover the most pertinent information affecting current operation.

Technical Reference(s): OP-AA-112-101

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-ADMROJ15

Question Source: Bank - NMP1 2017 Cert #66

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

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

Ability to determine Technical Specification Mode of Operation.

Question: #69

The plant is completing a refueling outage with the following:

- Reactor coolant temperature is 180°F.
- The Reactor Mode Switch is in STARTUP.
- The first control rod is being withdrawn to commence the Reactor startup.

Which one of the following is the current Reactor Operating Condition, in accordance with Technical Specifications?

A. Shutdown Condition - Cold

B. Shutdown Condition - Hot

C. Major Maintenance Condition

D. Power Operating Condition

Proposed Answer: D

Explanation: The plant is in the Power Operating Condition because the Reactor Mode Switch is in STARTUP and control rod withdrawal is in progress for a Reactor startup.

A. Incorrect – The plant is in the Power Operating Condition. Plausible because Reactor coolant temperature is below 212ºF and the Reactor is not critical.
B. Incorrect – The plant is in the Power Operating Condition. Plausible because Reactor coolant temperature is close to 212ºF and the Reactor is not critical.
C. Incorrect – The plant is in the Power Operating Condition. Plausible because a refueling outage has just been completed.

Technical Reference(s): Technical Specification 1.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-101001-RBO-14

Question Source: Bank - 2015 NRC #68

Question History: 2015 NRC #68

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(5)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	3
Group #	
K/A #	2.2.2
Importance Rating	4.6

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

Question: #70

A plant startup is in progress.

Which one of the following describes a condition that must be met prior to placing the Mode Switch in RUN, in accordance with N1-OP-43A, Plant Startup?

A. All IRMs must be on Range 10.

B. Reactor pressure must be above 850 psig.

C. Both Feedwater isolation valves must be open.

D. Containment oxygen concentration must be less than 4%.

Proposed Answer: B

Explanation: Per N1-OP-43A Precaution and Limitation #1.3:

“Failure to maintain reactor pressure greater than 850 psig, with mode switch in RUN, or STARTUP with IRMs on range 10, will result in an MSIV isolation and reactor scram.”

A. Incorrect – Not all IRMs must be on Range 10 prior to placing the Mode Switch in RUN. Plausible because N1-OP-43A does require at least one IRM in each RPS channel to be on Range 10 prior to placing the Mode Switch in RUN.
C. Incorrect – The Feedwater IVs do not both need to be open prior to placing the Mode Switch in RUN. Plausible because they are required to both be open prior to exceeding 25% power.
D. Incorrect – Plausible because containment atmosphere oxygen concentration shall be reduced to less than 4% by volume within 24 hours of placing the Mode Switch in RUN, but is not required prior.

Technical Reference(s): N1-OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-10

Question Source: Bank - 2013 NRC #69

Question History: 2013 NRC #69

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

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

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

Question: #71

The plant is operating at power with the following:

- Personnel are preparing to enter the Drywell at power to investigate a problem.
- The personnel are going to need access to all Drywell elevations.
- The Drywell has been de-inerted.

Which one of the following describes the requirements of RP-AB-462, Access Controls During Drywell Power Entries?

The upper limit for Reactor power is ___(1)___ and the lower limit for oxygen concentration is ___(2)___.

	(1)	(2)
A.	5%	15%
B.	5%	19.5%
C.	40%	15%
D.	40%	19.5%

Proposed Answer: B

Explanation: The maximum allowed Reactor power for personnel entry into all elevations of the Drywell is 5%. The minimum oxygen concentration is 19.5%.

A. Incorrect – The minimum oxygen concentration is 19.5%. Plausible because 10% is well above the concentration present while the Drywell is inerted and a value included in RP-AB-462.
C. Incorrect – The maximum allowed Reactor power for personnel entry into all elevations of the Drywell is 5%. Plausible because 40% Reactor power the value for entry into only the lower elevations of the Drywell. The minimum oxygen concentration is 19.5%. Plausible because 10% is well above the concentration present while the Drywell is inerted and a value included in RP-AB-462.
D. Incorrect – The maximum allowed Reactor power for personnel entry into all elevations of the Drywell is 5%. Plausible because 40% Reactor power the value for entry into only the lower elevations of the Drywell.

Technical Reference(s): RP-AB-462, RP-AB-462-100-F-04

Proposed references to be provided to applicants during examination: None

Learning Objective: S-RPIP-10.4-TO01

Question Source: Bank - 2015 Cert #70

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.3.4
	Importance Rating	3.2

Knowledge of radiation exposure limits under normal or emergency conditions.

Question: #72

The plant is shutdown with the following:

- A 31 year old Operator is entering the Drywell for a job.
- General area dose rate is 900 mRem/hr.
- The Operator's TEDE for the year is 1610 mRem.
- The Operator's lifetime exposure is 25 Rem.
- No dose extensions have been obtained.
- It will take 75 minutes to complete the job.

Which one of the following describes the dose gained in order to complete the job, in accordance with RP-AA-203, Exposure Control and Authorization?

The operator's dose…

A. remains within the normal dose control level and does NOT require an extension.

B. requires an extension beyond the normal dose control level, but does NOT exceed the Exelon annual administrative limit.

C. exceeds the Exelon annual administrative limit, but NOT the Federal annual limit.

D. exceeds both the Exelon annual administrative limit and the Federal annual limit.

Proposed Answer: B

Explanation: The normal administrative dose control level without any extensions is 2000 mRem/yr TEDE. The Exelon annual administrative limit and the Federal annual limit are both 5000 mRem. The total dose the worker will have received after completing the job will be 1610+900(75/60)=2735 mRem. This exceeds the Exelon normal administrative dose control level, but not the Exelon annual administrative limit or the Federal limit.

A. Incorrect – The dose exceeds the normal control level of 2000 mRem/yr. Plausible because the Operator’s initial dose is below this level.
C. Incorrect – The dose does not exceed the Exelon annual administrative limit. Plausible because the dose does exceed the Exelon normal administrative dose control level.
D. Incorrect – The dose does not exceed the Exelon annual administrative limit or the Federal limit. Plausible because the dose does exceed the Exelon normal administrative dose control level.

Technical Reference(s): RP-AA-203

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Modified Bank - NMP2 2017 NRC #71

Question History: NMP2 2017 NRC #71

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(12)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.8
	Importance Rating	3.8

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

Question: #73

The plant has scrammed with the following:

- Reactor water level is 70” and stable.
- N1-SOP-1, Reactor Scram, is being implemented.
- N1-EOP-2, RPV Control, is being implemented.

Then, Reactor water level lowers to 50” due to a Feedwater malfunction.

Which one of the following describes the required use of N1-SOP-1 and N1-EOP-2?

A. Re-enter both procedures from the beginning.

B. Re-enter N1-EOP-2 from the beginning. Exit N1-SOP-1.

C. Re-enter N1-EOP-2 from the beginning. Remain in N1-SOP-1.

D. Remain at the current location in both procedures and re-execute just the applicable steps.

Proposed Answer: C

Explanation: When Reactor water level goes below 53”, an entry condition for N1-EOP-2 is met. This requires re-entering the procedure from the beginning. There is no requirement to do the same for N1-SOP-1. Additionally, there is no requirement to exit SOPs when an EOP is entered. Therefore, remaining in N1-SOP-1 is correct.

A. Incorrect – There is no requirement to re-enter N1-SOP-1 from the beginning. Plausible because this is required when new or additional EOP entry conditions are met.
B. Incorrect – There is no requirement to exit SOPs when an EOP is entered. Plausible because the EOP is the higher tiered document and will override the SOP in the event of a conflict.
D. Incorrect – N1-EOP-2 must be re-entered from the beginning. Plausible because this would be an alternate way of ensuring the applicable steps are re-performed.

Technical Reference(s): NER-1M-095, N1-EOP-2, N1-SOP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: S101-EOP00C01 TO #1

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.35
	Importance Rating	3.8

Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Question: #74

N1-SOP-21.2, Control Room Evacuation, is being performed.

Which one of the following describes an action performed per N1-SOP-21.2?

A. Air is isolated to ensure the outboard MSIVs are closed.

B. Breakers are opened to ensure the inboard MSIVs are closed.

C. Jumpers are installed to prevent MSIVs from closing on lo-lo Reactor water level.

D. Mechanical Vacuum pump is started to prevent MSIVs from closing on low vacuum.

Proposed Answer: A

Explanation: Per N1-SOP-21.2 Attachment 1, instrument air is isolated to the outboard MSIVs (01-03 and 01-04). This field action ensures these valves are closed and remain closed.

B. Incorrect – Breakers to the inboard MSIVs are not opened. Plausible because action is taken to ensure MSIVs are closed.
C. Incorrect – MSIV logic jumpers are not installed. Plausible because field action is taken to ensure proper control of the MSIVs. Also plausible because jumpers are installed to defeat MSIV lo-lo Reactor water level closure in other emergency situations.
D. Incorrect – The Mechanical Vacuum pump is not started. Plausible because field action is taken to ensure proper control of the MSIVs. Also plausible because the Mechanical Vacuum pump is started to prevent MSIV closure on low vacuum in other emergency situations.

Technical Reference(s): N1-SOP-21.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP21.2C01

Question Source: Bank – 2015 NRC #73

Question History: 2015 NRC #73

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

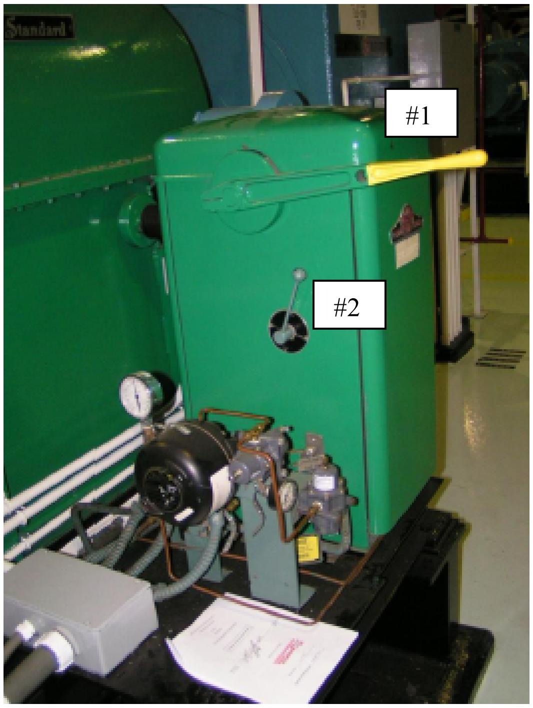
Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.34
	Importance Rating	4.2

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Question: #75

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

- Reactor Recirculation pump 11 is in local-lock due to controller oscillations.
- Reactor Recirculation pump 11 speed must be lowered due to high generator slot temperatures.
- Refer to the generic picture below of Reactor Recirculation pump 11 scoop tube positioner, with two handles labeled #1 and #2:



Which one of the following describes the operation(s) required to adjust Reactor Recirculation pump 11 speed?

A. Operate handle #1, only.

B. Operate handle #2, only.

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

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

Proposed Answer: D

Explanation: Per N1-OP-1, sect. H.5.2, the scoop tube positioner must first be unlocked by operating handle #2 (OPERATING LEVER), then the scoop tube must be repositioned with handle #1 (HAND LEVER).

A. Incorrect – Both handles must be operated. Plausible because this is an infrequently performed task and there are no labels provided.
B. Incorrect – Both handles must be operated. Plausible because this is an infrequently performed task and there are no labels provided.
C. Incorrect – Handle #1 will not function until handle #2 is moved from LOCK to MANUAL. Plausible because this is an infrequently performed task and there are no labels provided.

Technical Reference(s): N1-OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO010

Question Source: Bank - 2013 NRC #68

Question History: 2013 NRC #68

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295001 2.1.19
	Importance Rating	3.8

Partial or Complete Loss of Forced Core Flow Circulation

Ability to use plant computers to evaluate system or component status.

Question: #76

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

- Two Reactor Recirculation pumps (RRPs) trip.
- N1-SOP-1.3, Recirc Pump Trip at Power, is entered
- The associated RRP discharge valves are closed in accordance with N1-SOP-1.3, Recirc Pump Trip at Power.
- Reactor power is now approximately 85%.
- Plant parameters are now as shown on the computer screen on the next page.
- Reactor Engineering has determined that it is NOT desired to raise the rod line any further.

Which one of the following describes the control of Reactor power, in accordance with N-OP-1, Nuclear Steam Supply System (NSSS), and Technical Specifications?

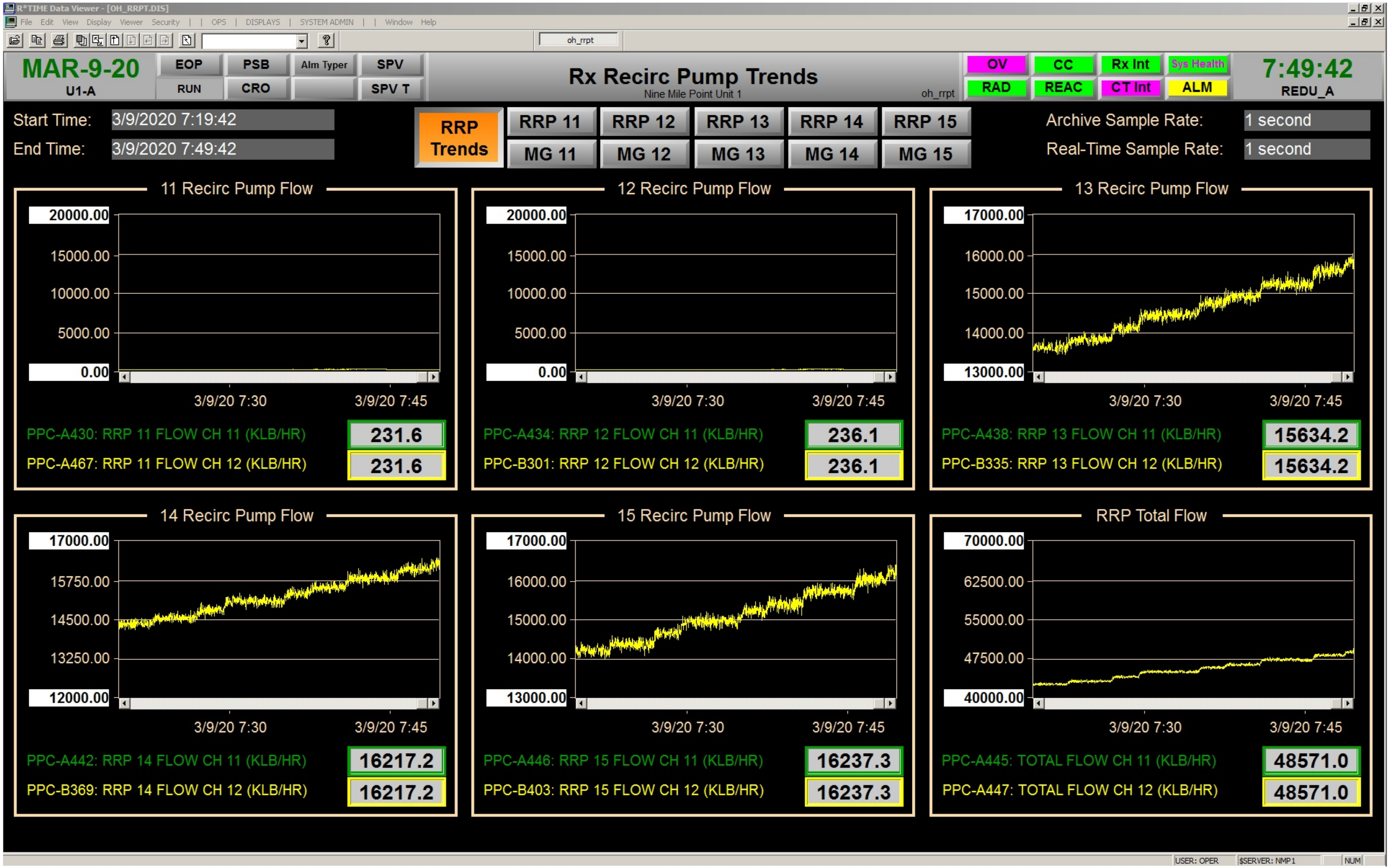
Reactor power…

A. must be lowered due to exceeding a Reactor power limit.

B. must be lowered due to exceeding a Recirculation flow limit.

C. may be maintained at the current level, but may NOT be raised further.

D. may be raised further.


R*TIME Data Viewer - [OH_RRPT.DIS]
File Edit View Display Viewer Security | | OPS | DISPLAYS | SYSTEM ADMIN | | Window Help
oh_rrpt
MAR-9-20
U1-A
EOP
PSB
Alm Typer
SPV
RUN
CRO
SPV T
Rx Recirc Pump Trends
Nine Mile Point Unit 1
oh_rrpt
OV
CC
Rx Int
Sys Health
RAD
REAC
CT Int
ALM
7:49:42
REDU_A
Start Time: 3/9/2020 7:19:42
End Time: 3/9/2020 7:49:42
RRP Trends
RRP 11
RRP 12
RRP 13
RRP 14
RRP 15
MG 11
MG 12
MG 13
MG 14
MG 15
Archive Sample Rate: 1 second
Real-Time Sample Rate: 1 second
11 Recirc Pump Flow
20000.00
15000.00
10000.00
5000.00
0.00
3/9/20 7:30
3/9/20 7:45
PPC-A430: RRP 11 FLOW CH 11 (KLB/HR) 231.6
PPC-A467: RRP 11 FLOW CH 12 (KLB/HR) 231.6
12 Recirc Pump Flow
20000.00
15000.00
10000.00
5000.00
0.00
3/9/20 7:30
3/9/20 7:45
PPC-A434: RRP 12 FLOW CH 11 (KLB/HR) 236.1
PPC-B301: RRP 12 FLOW CH 12 (KLB/HR) 236.1
13 Recirc Pump Flow
17000.00
16000.00
15000.00
14000.00
13000.00
3/9/20 7:30
3/9/20 7:45
PPC-A438: RRP 13 FLOW CH 11 (KLB/HR) 15634.2
PPC-B335: RRP 13 FLOW CH 12 (KLB/HR) 15634.2
14 Recirc Pump Flow
17000.00
15750.00
14500.00
13250.00
12000.00
3/9/20 7:30
3/9/20 7:45
PPC-A442: RRP 14 FLOW CH 11 (KLB/HR) 16217.2
PPC-B369: RRP 14 FLOW CH 12 (KLB/HR) 16217.2
15 Recirc Pump Flow
17000.00
16000.00
15000.00
14000.00
13000.00
3/9/20 7:30
3/9/20 7:45
PPC-A446: RRP 15 FLOW CH 11 (KLB/HR) 16237.3
PPC-B403: RRP 15 FLOW CH 12 (KLB/HR) 16237.3
RRP Total Flow
70000.00
62500.00
55000.00
47500.00
40000.00
3/9/20 7:30
3/9/20 7:45
PPC-A445: TOTAL FLOW CH 11 (KLB/HR) 48571.0
PPC-A447: TOTAL FLOW CH 12 (KLB/HR) 48571.0
USER: OPER
$SERVER: NMP1
NUM

Proposed Answer: D

Explanation: Technical Specification 3.1.7.e requires Reactor power to be limited to 90% when operating with only three Reactor Recirculation pumps in service. N1-OP-1 limits the Reactor Recirculation pumps to 16.8 Mlbm/hr each for a total of 50.2 Mlbm/hr. The given indications show operation with three operating RRP loops, Reactor power below 90%, and Recirculation loop flows less than 16.8 Mlbm/hr. Therefore, Reactor may be raised using Recirculation flow.

A. Incorrect – Reactor power may be raised. Plausible because this would be correct if Reactor power was >90%.
B. Incorrect – Reactor power may be raised. Plausible because this would be correct if any Recirc pump flow was >16.875 Mlbm/hr.
C. Incorrect – Reactor power may be raised. Plausible because this would be correct if Reactor power was 90% and/or Recirc pump flows were at the 16.875 Mlbm/hr limit.

Technical Reference(s): N1-OP-1, Technical Specification 3.1.7

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-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 SRO
Tier # 1
Group # 1
K/A # 295003 AA2.01
Importance Rating 3.7

Partial or Complete Loss of AC Power

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

Question: #77

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

Date	Time (hh:mm)	Event
11/21	08:00	Core Spray pump 111 is declared inoperable.
11/21	12:00	Breaker R10 trips and CANNOT be re-closed.
11/21	16:00	Emergency Diesel Generator (EDG) 103 is declared inoperable.

Which one of the following describes the latest time that a plant shutdown can be initiated while still complying with Technical Specifications?

A. 11/21 at 13:00

B. 11/21 at 17:00

C. 11/22 at 17:00

D. 11/28 at 09:00

Proposed Answer: B

Explanation: When EDG 103 becomes inoperable at time 1600, Technical Specification 3.6.3.g is no longer met because Core Spray pump 111 (powered by EDG 102) is inoperable. With Technical Specification 3.6.3.g not met, the required action is to initiate a normal orderly shutdown within one hour (by 1700 on 11/21).

Note: The question meets the K/A by presenting two partial losses of AC power sources (Line 1, EDG 103) and requiring interpretation of the cause of loss. Specifically, the candidate must interpret that Breaker R10 tripping (the cause) results in Line 1 being declared inoperable (the result and what must be applied in Tech Specs).

A. Incorrect – This is one hour after Line 1 became inoperable. Plausible because this would be correct if Technical Specifications included something analogous to TS 3.6.3.g for Lines 1 and 4, and not just EDGs 102 and 103. Additionally, at time 12:00, Core Spray pump 121 must be declared inoperable per TS 3.0.1, since it loses it's normal power source (Line 1) and has a redundant component inoperable (Core Spray pump 111). This requires entry into TS 3.1.4.c, since Core Spray pumps 111 and 121 are in separate Core Spray systems, but still allows 7 days to restore one of these pumps to operable.
C. Incorrect – This is 25 hours after EDG 103 became inoperable. When EDG 103 became inoperable, both TS 3.6.3.c and TS 3.6.3.g were required to be entered. TS 3.6.3.c allows 24 hours to restore EDG 103, before then requiring a shutdown to be initiated within 1 hour (25 hours total). However, TS 3.6.3.g has a more restrictive shutdown requirement.
D. Incorrect – Plausible because this is the time when TS 3.1.4.c would require a shutdown to be initiated by for the original Core Spray inoperability, if it were not compounded by the electrical losses.

Technical Reference(s): Technical Specifications 3.1.4 and 3.6.3, N1-OP-2, C-18007-C

Proposed references to be provided to applicants during examination: Technical Specifications 3.0.1, 3.1.4, and 3.6.3

Learning Objective: N1-262001-RBO-14

Question Source: Bank – 2015 NRC #79

Question History: 2015 NRC #79

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 #	295005 AA2.08
	Importance Rating	3.3

Main Turbine Generator Trip

Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Electrical distribution status

Question: #78

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

- A sustained electrical fault occurs on the Main Generator.
- A Main Generator lockout occurs.
- Two minutes later, the following breaker positions are observed:
    - R112, PB-11 Reserve Supply, is open.
    - R113, PB-11 Normal Supply, is open.
    - R122, PB-12 Reserve Supply, is open.
    - R123, PB-12 Normal Supply, is open.

Which one of the following describes the status of Powerboards 11 and 12 and the required procedure execution to control these Powerboards?

Powerboards 11 and 12 ___(1)___ responded as designed. Control these Powerboards using the guidance in ___(2)___.

	(1)	(2)
A.	have	N1-SOP-31.1, Turbine Trip
B.	have	N1-SOP-30.3, Loss of Power Board 11 and Power Board 12
C.	have NOT	N1-SOP-31.1, Turbine Trip
D.	have NOT	N1-SOP-30.3, Loss of Power Board 11 and Power Board 12

Proposed Answer: D

Explanation: On the Main Generator lockout, Powerboards 11 and 12 should have automatically transferred from the normal supply to reserve. Well before two minutes after the lockout, R112 and R122 should be closed. Therefore, the Power Boards have failed to respond as designed. N1-SOP-30.3 has the specific guidance to manually re-energize these Power Boards.

A. Incorrect – Power Boards 11 and 12 have failed to respond as designed. Plausible because the Main Generator lockout does result in the normal supply breakers opening and does prevent closure of some breakers. Also plausible because N1-SOP-31.1 also must be entered, does have a step to check on Power Board 11 and 12 alignment, and does give guidance to reset the Main Generator lockout.
B. Incorrect – Power Boards 11 and 12 have failed to respond as designed. Plausible because the Main Generator lockout does result in the normal supply breakers opening and does prevent closure of some breakers. N1-SOP-30.3 has the specific guidance to manually re-energize these Power Boards, not N1-SOP-31.1. Plausible because N1-SOP-31.1 also must be entered, does have a step to check on Power Board 11 and 12 alignment, and does give guidance to reset the Main Generator lockout.
C. Incorrect – N1-SOP-30.3 has the specific guidance to manually re-energize these Power Boards, not N1-SOP-31.1. Plausible because N1-SOP-31.1 also must be entered, does have a step to check on Power Board 11 and 12 alignment, and does give guidance to reset the Main Generator lockout.

Technical Reference(s): N1-OP-30, N1-SOP-31.1, N1-SOP-30.3

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP30.3C01 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 #	1
	K/A #	295016 AA2.03
	Importance Rating	4.4

Control Room Abandonment

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

Question: #79

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

- A Control Room evacuation has been performed.
- All immediate actions of N1-SOP-21.2, Control Room Evacuation, have been performed.
- Operators have just arrived at Remote Shutdown Panel 12.
- Emergency Condenser (EC) 12 is in service.
- Reactor pressure is 850 psig and lowering.

Which one of the following describes the required control of EC 12, in accordance with N1-SOP-21.2?

A. Maintain EC 12 continuously in service irrespective of cooldown rate.

B. Cycle EC 12 to maintain the Reactor in a stable hot shutdown condition.

C. Cycle EC 12 to maintain a cooldown. Maintain the cooldown rate ≤100°F/hr.

D. Cycle EC 12 to maintain a cooldown. The cooldown rate may exceed 100°F/hr.

Proposed Answer: B

Explanation: Once at the RSP, N1-SOP-21.2 and N2-EOP-2 require cycling EC 12 to maintain the Reactor in a stable hot standby condition until the Control Room becomes available again.

A. Incorrect – EC 12 must be cycled. Plausible because this is the required control of ECs during a Station Blackout.
C. Incorrect – A cooldown is not commenced. Plausible because the Reactor is scrammed and a cooldown is normally commenced thereafter to lower the energy state of the plant.
D. Incorrect – A cooldown is not commenced. Plausible because the Reactor is scrammed and a cooldown is normally commenced thereafter to lower the energy state of the plant. Also plausible that normal cooldown rates could be exceeded due to degraded plant control capabilities. Plausible the higher allowed cooldown rate may still be not as high as would be achieved by not cycling the EC in and out of service.

Technical Reference(s): N1-SOP-21.2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP21.2C01 EO-2

Question Source: Bank – JAF 17-2 NRC #77

Question History: JAF 17-2 NRC #77

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295019 2.4.9
	Importance Rating	4.2

Partial or Complete Loss of Instrument Air

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

Question: #80

A refueling outage is in progress with the following:

- The Spent Fuel Pool gates are removed.
- Shutdown Cooling is in service in accordance with N1-OP-4, Shutdown Cooling System.
- Spent Fuel Pool Cooling is in service in accordance with N1-OP-6, Fuel Pool Filtering and Cooling System.

Then, a complete loss of Instrument Air occurs.

- N1-SOP-20.1, Loss of Instrument Air, is being executed.
- N1-SOP-6.1, Loss of Rx Cavity/Decay Heat Removal, is being executed.

Which one of the following describes the status of decay heat removal?

Decay heat removal…

A. remains available due to previous actions taken in N1-OP-4.

B. remains available due to previous actions taken in N1-OP-6.

C. is lost, but can be restored with Shutdown Cooling using an attachment in N1-SOP-6.1.

D. is lost, but can be restored with Spent Fuel Pool Cooling using an attachment in N1-SOP-20.1.

Proposed Answer: A

Explanation: Decay heat removal would be lost due to loss of both Shutdown Cooling (RBCLC supply valve 70-53 fails closed) and Spent Fuel Pool Cooling (pumps trip). However, N1-OP-4 takes action to harden the SDC system by gagging open 70-53.

B. Incorrect – Decay heat removal from Spent Fuel Pool Cooling is lost due to pump trips. Plausible that action would have been taken in N1-OP-6 to harden the system to prevent such a failure due to the importance of decay heat removal. Action is taken with SDC to harden against certain failures when secured (SDC IV fuses and breakers).
C. Incorrect – Decay heat removal is not lost. Plausible because this would be correct except for hardening actions taken in N1-OP-4. Also plausible because there is SOP guidance for restoring DHR and SOP 6.1 is entered for a loss of DHR.
D. Incorrect – Decay heat removal is not lost. Plausible because this would be correct except for hardening actions taken in N1-OP-4.

Technical Reference(s): N1-SOP-20.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP20.1C01 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 #	1
	K/A #	295037 2.4.41
	Importance Rating	4.6

Scram Condition Present and Reactor Power Above APRM Downscale or Unknown

Knowledge of the emergency action level thresholds and classifications.

Question: #81

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

Time (minutes)	Conditions
0	• A coolant leak occurs inside the Drywell. • The Reactor fails to automatically scram on a high Drywell pressure signal. • Attempts to manually scram the Reactor and initiate ARI fail to insert control rods. • Reactor power remains at 100%.
5	• Boron has been injected. • Recirc pumps have been tripped. • Reactor water level has been intentionally lowered. • Reactor water level is -70” and stable. • Reactor pressure is 950 psig and stable. • Reactor power is downscale on all APRMs. • All required isolations have been verified SAT. • Torus water temperature is 112°F and rising. • Torus water level is 11.1 ft. • The Shift Manager is ready to classify the event.

Which one of the following describes the highest emergency action level that is met or exceeded at time 5 minutes, in accordance with the emergency plan?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer: B

Explanation: An Alert (MA3) is met. This is due to an automatic scram not shutting down the Reactor and manual/ARI actions taken at the Reactor Control Console also not shutting down the Reactor. These manual actions do not include the subsequent actions to lower Reactor power (Boron injection, lowering level, tripping Recirc pumps). Torus water temperature is elevated, but not above HCTL.

A. Incorrect – The highest EAL met is an Alert. Plausible because Unusual Event EAL MU3 is also relevant under these plant conditions.
C. Incorrect – The highest EAL met is an Alert. Plausible because Site Area Emergency MS3 is also challenged under these plant conditions and would be correct if Torus water temperature were higher (HCTL exceeded).
D. Incorrect – The highest EAL met is an Alert. Plausible because multiple fission product barriers are degraded or potentially challenged, which makes General Emergency FG1 relevant.

Technical Reference(s): Hot EAL Matrix

Proposed references to be provided to applicants during examination: Hot EAL Matrix, HCTL curve

Learning Objective: 1101-EOP3C01 EO-3

Question Source: Bank – NMP2 2019 NRC #82

Question History: NMP2 2019 NRC #82

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	700000 AA2.01
	Importance Rating	3.6

Generator Voltage and Electric Grid Disturbances

Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Operating point on the generator capability curve

Question: #82

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

- Grid conditions are degraded.
- The Main Generator is operating the following parameters:
    - Real load is 640 MWe.
    - Reactive load is 200 MVAR lagging.
    - Hydrogen pressure is 43 psig.

Then, the following occurs:

- Annunciator A1-4-1, GENERATOR H2 SEAL OIL PRESSURE LOW, alarms.
- Annunciator A1-4-2, GENERATOR HYDROGEN SYSTEM, alarms.
- Computer point D113, GEN H2 SYS SUP OIL PR _ LOW, has been received.
- Seal oil differential pressure at the Main Generator is 0.5 psid.
- Generator hydrogen pressure is 28 psig and lowering due to Hydrogen leaking out of the seals.
- The Main Seal Oil Pump (MSOP) is still in service.
- MSOP discharge pressure is 105 psig.

Note: The Generator Estimated Capability curve is provided on the following page.

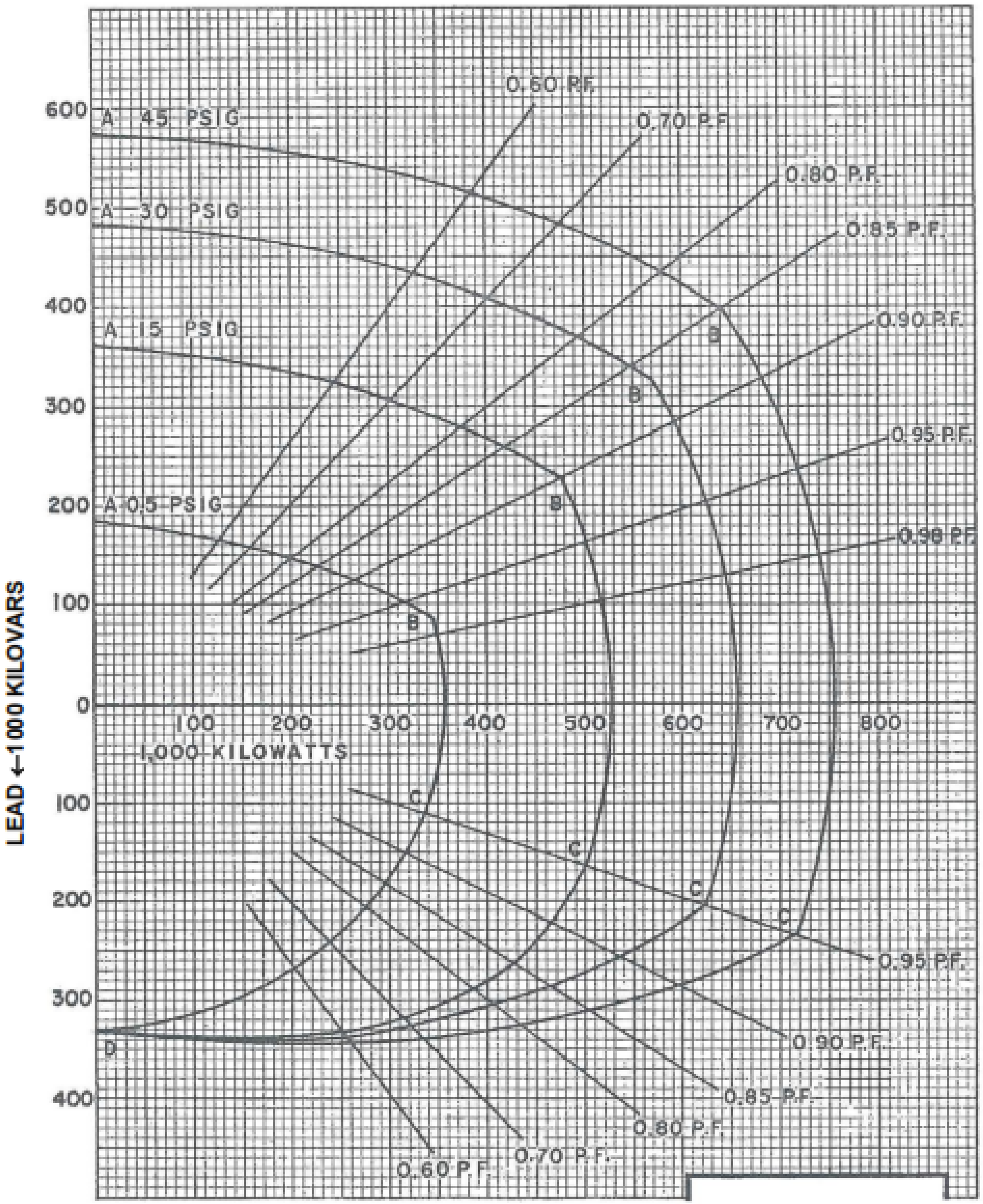
Which one of the following describes the **first** action required to be taken under these conditions, in accordance with plant procedures?

A. Add hydrogen to the Main Generator per N1-OP-7.

B. Commence an Emergency Power Reduction per N1-SOP-1.1.

C. Start the Emergency Seal Oil pump and secure the MSOP per N1-SOP-32.

D. Scram the Reactor per N1-SOP-1 and verify the Turbine trip per N1-SOP-31.1.



LEAD ←1000 KILOVARS
1,000 KILOWATTS
600
500
400
300
200
100
0
100
200
300
400
100
200
300
400
500
600
700
800
A 45 PSIG
A 30 PSIG
A 15 PSIG
A 0.5 PSIG
B
C
D
0.60 P.F.
0.70 P.F.
0.80 P.F.
0.85 P.F.
0.90 P.F.
0.95 P.F.
0.98 P.F.

Proposed Answer: B

Explanation: Under the conditions described, entry into N1-SOP-32, Generator Auxiliaries Failures, is required. In executing the conditional override steps in N1-SOP-32, the first applicable action requires evaluation of the current conditions against the Estimated Capability Curve in Attachment 1 of N1-SOP-32. Current plant conditions place the generator outside the Estimated Capability Curve for hydrogen pressure of 28 psig and require an emergency power reduction per N1-SOP-1.1 until operating within the curve.

A. Incorrect –Plausible because hydrogen pressure is low and adding hydrogen is directed from ARP A1-4-2 for low H2 pressure and is a follow-up action in N1-SOP-32. However, both annunciators in the stem are entry conditions for N1-SOP-32 which is a higher tiered document. Also, it would not be appropriate to add hydrogen when it is known to be leaking from the generator.
C. Incorrect – The given MSOP discharge pressure indicates the pump is operating properly, so securing the pump is not required. Plausible because this would be required at slightly lower pressures (ESOP should start if MSOP discharge pressure drops below 90 psig).
D. Incorrect – A scram and turbine trip is not required under the given conditions. Plausible because this would be correct if hydrogen pressure was still lowering rapidly and continued below 8 psig.

Technical Reference(s): N1-SOP-32

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP32C01 EO-2

Question Source: Bank – 2009 NRC #20

Question History: 2009 NRC #20

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(10)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295013 AA2.01
	Importance Rating	4.0

High Suppression Pool Temperature

Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Suppression pool temperature

Question: #83

A plant startup is in progress with the following:

- N1-ST-C2, Solenoid-Actuated Pressure Relief Valves Operability and Flow Verification Test, is being performed.
- Torus average water temperature is rising.

Which one of the following describes the Torus average water temperature limits during and after the test, in accordance with Technical Specifications?

Torus average water temperature shall NOT exceed ___(1)___ during performance of the test and must be reduced below the normal power operation limit within ___(2)___.

	(1)	(2)
A.	85ºF	12 hours
B.	85ºF	24 hours
C.	95ºF	12 hours
D.	95ºF	24 hours

Proposed Answer: D

Explanation: Per Technical Specification 3.3.2.d, during testing of relief valves that adds heat to the Torus, the operating limit of 85ºF is raised to 95ºF. 24 hours are permitted to return the temperature to less than 85ºF.

A. Incorrect – The limit is 95ºF during this testing. Plausible because the normal limit is 85ºF. 24 hours are permitted to return the temperature to less than 85ºF. Plausible because 12 hours is used frequently in Technical Specifications.
B. Incorrect – The limit is 95ºF during this testing. Plausible because the normal limit is 85ºF.
C. Incorrect – 24 hours are permitted to return the temperature to less than 85ºF. Plausible because 12 hours is used frequently in Technical Specifications.

Technical Reference(s): Technical Specification 3.3.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223001-RBO-14

Question Source: Bank - 2010 NRC #80

Question History: 2010 NRC #80

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295014 2.1.23
	Importance Rating	4.4

Inadvertent Reactivity Addition

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

Question: #84

A plant startup is in progress with the following:

- A transient occurs and results in an inadvertent reactivity addition.
- N1-SOP-1.5, Unplanned Power Change, has been entered.
- All five (5) Reactor Recirc pumps are operating.
- Reactor power is 50%.
- Core flow is 23 Mlbm/hr.

Which one of the following describes the required action(s), in accordance with N1-SOP-1.5?

A. Manually scram the Reactor.

B. Insert CRAM rods. Raising Recirc flow is NOT allowed. A scram is NOT required.

C. Raise Recirc flow. Inserting CRAM rods is NOT allowed. A scram is NOT required.

D. Insert CRAM rods or raise Recirc flow. A scram is NOT required.

Proposed Answer: D

Explanation: The given conditions place the plant in the Restricted Zone of the 5-loop power to flow map. An override in N1-SOP-1.5 requires exiting the Restricted Zone by either inserting CRAM rods or raising Recirc flow.

A. Incorrect – A scram is not required. Plausible because multiple other conditions in N1-SOP-1.5 require a scram (THI, entering other zones of P/F map, no Recirc pumps running).
B. Incorrect – Raising Recirc flow is allowed. Plausible because raising Recirc flow also raises Reactor power further and the initiating event was an inadvertent reactivity addition.
C. Incorrect – Inserting CRAM rods is allowed. Plausible because this would make conditions worse for another region of the P/F map (Reactor Internals Protection Region).

Technical Reference(s): N1-SOP-1.5

Proposed references to be provided to applicants during examination: 5-loop Power to Flow Map

Learning Objective: 1101-SOP1.5C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference: Level	SRO
Tier #	1
Group #	2
K/A #	295029 EA2.03
Importance Rating	3.5

High Suppression Pool Water Level

Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Drywell/containment water level

Question: #85

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

- Reactor water level is 30” and rising.
- Reactor pressure is 450 psig and lowering.
- Feedwater and CRD are injecting to the Reactor.
- Containment Sprays are in service.
- Drywell pressure is 5 psig and lowering.
- Torus water level is 13.4’ and slowly rising.
- Torus water temperature is 125°F and slowly rising.

Which one of the following is required, in accordance with the Emergency Operating Procedures?

A. Perform an RPV Blowdown.

B. Vent the Primary Containment.

C. Secure Containment Sprays and initiate Torus Cooling.

D. Stop injection from Feedwater and CRD not needed for adequate core cooling.

Proposed Answer: D

Explanation: Torus water level is approaching 13.5’. Per N1-EOP-4, this requires stopping injection into the RPV from sources outside the Primary Containment (which includes Feedwater and CRD) not needed for adequate core cooling. With Reactor water level at 30” and rising, adequate core cooling is currently assured and there is more injection than needed to maintain adequate core cooling, therefore at least some of the injection from Feedwater and CRD can be stopped to help mitigate the rise in Torus water level.

A. Incorrect – An RPV Blowdown is not required. Plausible because this would be correct if Torus water level could not be restored and maintained below 13.5’.
B. Incorrect – Venting the Primary Containment is not required. Plausible if the candidate believed the concern on Torus water level reaching 13.5’ had to do with operability of Primary Containment venting, which is of concern at higher containment water levels.
C. Incorrect – Securing Containment Sprays is not required. Plausible because Containment Sprays are required to be secured when Drywell pressure lowers further, and Torus temperature is elevated, which makes Torus Cooling appropriate.

Technical Reference(s): N1-EOP-2, N1-EOP-4

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP-4C01 EO-2

Question Source: Modified Bank – 2010 Cert #52

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Facility:	Nine Mile Point Unit 1
Vendor:	GE
Exam Date:	2010
Exam Type:	RO

Examination Outline Cross-reference:	Level	RO		SRO
	Tier #	1		
	Group #	1		
	K/A #	295024	EA2.06	
	Importance Rating	4.1		

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

Proposed Question: RO Question # 52

During a LOCA, the following conditions exist:

- Reactor water level is -80 inches and rising
- Reactor pressure is 600 psig and lowering
- Feedwater and CRD are injecting
- Containment Sprays are in service
- Drywell pressure is 3.5 psig and lowering
- Torus water level is 11.5 feet and slowly rising
- Torus water temperature is 125°F and slowly rising

Which one of the following identifies the NEXT required operator action?

A. Enter EOP-8, RPV Blowdown, and open 3 ERVs.

B. Secure Containment Sprays and place Containment Spray in torus cooling.

C. Enter EOP-8, RPV Blowdown and place Containment Spray in torus cooling.

D. Secure all RPV injection sources with suctions outside the Primary Containment.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Entering EOP-8 would be required if no actions could be taken to prevent exceeding HCTL, but torus cooling can be placed in service to prevent exceeding HCTL

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	206000 A2.17
	Importance Rating	4.3

High Pressure Coolant Injection

Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: HPCI inadvertent initiation:BWR-2,3,4

Question: #86

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

- A spurious HPCI initiation occurs.
- Attempts to reset the HPCI initiation signal fail.

Which one of the following describes:

(1) actions that will prevent injection from HPCI

and

(2) the most limiting Technical Specification impact if these actions are taken?

	(1) Actions That Will Prevent Injection From HPCI	(2) Most Limiting Technical Specification Impact
A.	Place Feedwater 11 and 12 flow controllers in MAN and dial to 0	Restore the system to operable status within a maximum of 15 days
B.	Place Feedwater 11 and 12 flow controllers in MAN and dial to 0	Initiate a normal orderly shutdown within one hour
C.	Place Feedwater pump 11 and 12 control switches in Pull-to-Lock	Restore the system to operable status within a maximum of 15 days
D.	Place Feedwater pump 11 and 12 control switches in Pull-to-Lock	Initiate a normal orderly shutdown within one hour

Proposed Answer: D

Explanation: With the HPCI initiation signal sealed in, the Feedwater 11 and 12 flow controllers will not function to prevent HPCI injection. Placing the Feedwater pump 11 and 12 control switches in Pull-to-Lock will stop the pumps and prevent HPCI injection. With both pumps in Pull-to-Lock, they are both inoperable for their HPCI function. Technical Specification 3.1.8.c requires initiating a normal orderly shutdown within one hour.

Note: Question meets requirements of the K/A due to meeting the second part of the K/A statement.

A. Incorrect – With the HPCI initiation signal sealed in, the Feedwater 11 and 12 flow controllers will not function to prevent HPCI injection. Plausible that the flow controllers would still work but the pump control switches would not. Technical Specification 3.1.8.c requires initiating a normal orderly shutdown within one hour. Plausible because Technical Specification 3.1.8.b is also applicable and requires restoring the pumps to operable within 15 days.
B. Incorrect – With the HPCI initiation signal sealed in, the Feedwater 11 and 12 flow controllers will not function to prevent HPCI injection. Plausible that the flow controllers would still work but the pump control switches would not.
C. Incorrect – Technical Specification 3.1.8.c requires initiating a normal orderly shutdown within one hour. Plausible because Technical Specification 3.1.8.b is also applicable and requires restoring the pumps to operable within 15 days.

Technical Reference(s): N1-OP-16, Technical Specification 3.1.8

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-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 #	209001 2.1.30
	Importance Rating	4.0

Low Pressure Core Spray

Ability to locate and operate components, including local controls.

Question: #87

The plant is operating at 100% power with Core Spray 11 valves aligned as shown on the following page.

Which one of the following describes:

(1) the action required to restore Core Spray to the normal standby lineup, in accordance with N1-OP-2, Core Spray System,

and

(2) the status of Technical Specification 3.1.4, Core Spray System, until this action is taken?

	(1) Action to Restore Core Spray to Normal Standby Lineup	(2) Status of Technical Specification 3.1.4
A.	Close 40-12	LCO entry required
B.	Close 40-12	LCO entry NOT required
C.	Open the breaker for 40-30	LCO entry required
D.	Open the breaker for 40-30	LCO entry NOT required


40-11
CORE SPRAY DISCHARGE
IV 111 (INSIDE)
40-10
CORE SPRAY DISCHARGE
IV 112 (INSIDE)
40-12
CORE SPRAY DISCHARGE
IV 11 (OUTSIDE)
11-CS
CORE SPRAY VENT
ISOLATION VALVE 111
40-30

Proposed Answer: D

Explanation: The given picture shows all Core Sprays properly aligned, except the green light for 40-30 should be off (due to circuit breaker open with valve closed). This requirement is to meet NFPA 805 requirements (fire protection). Even though this requirement is not met, it is not required by Technical Specification 3.1.4 or the associated bases, therefore TS 3.1.4 LCO entry is not required.

A. Incorrect – 40-12 is properly aligned. Plausible because this valve will auto open if closed and a Core Spray initiation signal is received and the inside IVs are normally closed. Also plausible because there is a special requirement in Technical Specifications for this valve (open with breaker locked in off and redundant control room position indication available. TS 3.1.4 LCO entry is not required. Plausible because an important administrative requirement is not being met.
B. Incorrect – 40-12 is properly aligned. Plausible because this valve will auto open if closed and a Core Spray initiation signal is received and the inside IVs are normally closed. Also plausible because there is a special requirement in Technical Specifications for this valve (open with breaker locked in off and redundant control room position indication available.
C. Incorrect – TS 3.1.4 LCO entry is not required. Plausible because an important administrative requirement is not being met.

Technical Reference(s): N1-OP-2, Technical Specification 3.1.4 and associated bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-209001-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 #	211000 A2.04
	Importance Rating	3.4

Standby Liquid Control

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

Question: #88

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

- Both trains of Liquid Poison are being tested due to recent industry OE.
- The following sequence occurs:

Date	Time (hh:mm)	Event
11/21	08:00	• N1-ST-M1A, Liquid Poison Pump 11 Operability Test, is completed. • Liquid Poison pump 11 flow rate has been calculated as 31.2 gpm
11/21	16:00	• N1-ST-M1B, Liquid Poison Pump 12 Operability Test, is completed. • Liquid Poison pump 12 flow rate has been calculated as 28.9 gpm.

Which one of the following describes the latest time that a plant shutdown can be initiated while still complying with Technical Specifications, if any?

A. 11/21 at 17:00

B. 11/28 at 09:00

C. 11/28 at 17:00

D. NO Technical Specification shutdown is required because both Liquid Poison pumps remain operable.

Proposed Answer: C

Explanation: Technical Specification 3.1.2 bases and N1-ST-M1A(B) require Liquid Poison pump flow rate to be ≥30 gpm for the pump to be operable. Liquid Poison pump 11 flow rate was 31.2 gpm and Liquid Poison pump 12 flow rate was 28.9 gpm. Therefore, Liquid Poison pump 11 is operable and pump 12 is inoperable. Technical Specification 3.1.2.b is entered at 16:00 on 11/21. This allows 7 days to restore the pump before entering Technical Specification 3.1.2.e, which gives one additional hour before starting a shutdown. This makes the latest time for initiating a shutdown 11/28 at 17:00 (11/21 at 16:00 + 7 days + 1 hour).

A. Incorrect – The latest time for initiating a shutdown is 11/28 at 17:00. Plausible because this would be the time if both Liquid Poison pumps were inoperable.
B. Incorrect – The latest time for initiating a shutdown is 11/28 at 17:00. Plausible because this would be the time if Liquid Poison pump 11 was inoperable and pump 12 was operable.
D. Incorrect – The latest time for initiating a shutdown is 11/28 at 17:00. Plausible because this would be correct if both pump remained operable.

Technical Reference(s): Technical Specification 3.1.2

Proposed references to be provided to applicants during examination: Technical Specification 3.1.2 with part c. blocked out

Learning Objective: N1-211000-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 #	261000 2.2.25
	Importance Rating	4.2

Standby Gas Treatment

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Question: #89

Which one of the following evolutions in the Reactor Building would require additional testing of the Reactor Building Emergency Ventilation System, in accordance with Technical Specification bases, and why?

	Evolution	Reason
A.	Significant painting	Possible degraded performance of HEPA filters
B.	Significant painting	Possible degradation of radiation detectors
C.	Significant welding	Possible degraded performance of HEPA filters
D.	Significant welding	Possible degradation of radiation detectors

Proposed Answer: A

Explanation: Technical Specification 3.4.4 Surveillance Requirements direct the tests and sample analysis of specification 3.4.4b, c, and d be performed following significant painting, fire, or chemical release in any ventilation zone communication with the system. TS 3.4.4 bases state, If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use.

B. Incorrect – The reason is possible degraded performance of HEPA filters. Plausible that the painting concern could be desensitization of detectors used in automatic initiation.
C. Incorrect – The evolution of concern is painting. Plausible that welding would be the concern because it also has the potential to put pollutants in the air.
D. Incorrect – The evolution of concern is painting. Plausible that welding would be the concern because it also has the potential to put pollutants in the air. The reason is possible degraded performance of HEPA filters. Plausible that the concern could be desensitization of detectors used in automatic initiation.

Technical Reference(s): Technical Specification 3.4.4 bases

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-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 #	1
	K/A #	218000 A2.06
	Importance Rating	4.3

Automatic Depressurization System

Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: ADS initiation signals present

Question: #90

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

- Reactor water level is -20" and lowering slowly.
- Reactor pressure is 650 psig and lowering slowly.
- Drywell pressure is 10 psig and rising slowly.
- All Core Spray pumps have failed to start.
- Condensate pumps are running.
- ADS has NOT been overridden.
- Annunciator F2-3-1, MAIN STM LINE AUTOMATIC DE-PRESS TIMING, came into alarm 30 seconds ago.

Which one of the following describes:

(1) the response of the ERVs if the current conditions continue until the ADS timers time out,

and

(2) the required control of ADS under these conditions, in accordance with N1-EOP-2, RPV Control?

	(1) If current conditions continue, ERVs will…		(2) N1-EOP-2 requires…
A.	open		overriding ADS
B.	open		allowing ERVs to open
C.	remain closed		overriding ADS
D.	remain closed		allowing ERVs to open

Proposed Answer: A

Explanation: A valid ADS initiation signal exists due to Reactor water level <-10" and Drywell pressure >3.5 psig. ADS is designed to lower Reactor pressure to allow Core Spray to inject. When the timers time out, ERVs will open even though no Core Spray pumps are running due to lack of Core Spray discharge pressure confirmation in the ADS logic. N1-EOP-2 requires overriding ADS in this situation. ERVs are only manually opened if Reactor water level continues to lower to -84".

B. Incorrect – N1-EOP-2 requires overriding ADS in this situation. Plausible because this is the normal design functioning of the ADS system and Condensate pumps are available to inject at lower Reactor pressure.
C. Incorrect – ERVs will open. Plausible because no Core Spray pumps are running and ADS is designed to lower Reactor pressure to allow Core Spray pumps to inject.
D. Incorrect – ERVs will open. Plausible because no Core Spray pumps are running and ADS is designed to lower Reactor pressure to allow Core Spray pumps to inject. N1-EOP-2 requires overriding ADS in this situation. Plausible because this is the normal design functioning of the ADS system and Condensate pumps are available to inject at lower Reactor pressure.

Technical Reference(s): N1-OP-2, ARP F2-3-1, N1-EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO-2

Question Source: Bank - JAF 16-1 NRC #87

Question History: JAF 16-1 NRC #87

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	214000 A2.01
	Importance Rating	3.3

Rod Position Information

Ability to (a) predict the impacts of the following on the ROD POSITION INFORMATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failed reed switches

Question: #91

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

- Control rod 06-39 is at position 48.
- Then, control rod 06-39 loses all position indication at all positions.
- Control rod 06-39 position CANNOT be determined by other means.

Which one of the following describes the required action to be directed, in accordance with N1-OP-5, Control Rod Drive System, and whether Technical Specification LCO entry is required?

Direct…

A. placing administrative controls to prevent movement of control rod 06-39 with RMCS except in an emergency. Technical Specification LCO entry is required.

B. placing administrative controls to prevent movement of control rod 06-39 with RMCS except in an emergency. Technical Specification LCO entry is NOT required.

C. fully inserting control rod 06-39, disarming the control rod, and isolating the HCU. Technical Specification LCO entry is required.

D. fully inserting control rod 06-39, disarming the control rod, and isolating the HCU. Technical Specification LCO entry is NOT required.

Proposed Answer: C

Explanation: N1-OP-5 section H.10 provides the direction for response to a loss of rod position indication. Since all control rod position indication is lost for this control rod and its position cannot be determined by other means, subsection H.10.4 requires fully inserting the control rod, disarming the control rod, and isolating the HCU. With no rod position indication, the full-in reed switch will not provide the required input to the Remote Shutdown Panel "all rods in" light. This makes the "all rods in" light inoperable, therefore Technical Specification 3.6.13 LCO entry is required.

A. Incorrect – N1-OP-5 section H.10.4 requires fully inserting the control rod, not administratively preventing movement. Plausible that since position cannot be determined during movement and it is already fully withdrawn, the control rod could be left in the original position without much risk.
B. Incorrect – N1-OP-5 section H.10.4 requires fully inserting the control rod, not administratively preventing movement. Plausible that since position cannot be determined during movement and it is already fully withdrawn, the control rod could be left in the original position without much risk. Technical Specification 3.6.13 LCO entry is required. Plausible because LCO entry in the more obvious Technical Specification 3.1.1, Control Rod System, is NOT required.
D. Incorrect – Technical Specification 3.6.13 LCO entry is required. Plausible because LCO entry in the more obvious Technical Specification 3.1.1, Control Rod System, is NOT required.

Technical Reference(s): N1-OP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201002-RBO-14

Question Source: Bank – 2017 Cert #93

Question History: 2017 Cert #93

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 2.4.31
	Importance Rating	4.1

Fuel Handling Equipment

Knowledge of annunciator alarms, indications, or response procedures.

Question: #92

Refueling is in progress with the following:

- An irradiated fuel bundle is grappled and is being removed from the Reactor core.
- L1-3-5, FUEL POOL LVL LOW – SURGE TANK MAKE-UP VALVE OPEN, alarms.
- The Refuel Bridge team observes that Spent Fuel Pool and Reactor cavity water level is lowering.
- Spent Fuel Pool and Reactor cavity water level has lowered approximately one foot below the normal level.
- The Refuel Bridge radiation monitors indicate 25 mr/hr and slowly rising.
- The Control Room enters N1-SOP-6.1, Loss of SFP/Rx Cavity Level/Decay Heat Removal.
- The irradiated fuel bundle is currently half way out of the Reactor core.

Which one of the following describes the required direction to be given in accordance with N1-SOP-6.1?

Direct evacuation of…

A. all personnel from the Refuel Floor, leaving the irradiated fuel bundle in the current position.

B. only non-essential personnel from the Refuel Floor. Direct the Refuel Bridge team to transfer the irradiated fuel bundle to the Spent Fuel Pool.

C. only non-essential personnel from the Refuel Floor. Direct the Refuel Bridge team to lower the irradiated fuel bundle back into the original Reactor core location.

D. only non-essential personnel from the Refuel Floor. Direct the Refuel Bridge team to halt fuel movement and leave the irradiated fuel bundle in the current position.

Proposed Answer: C

Explanation: N1-SOP-6.1 entry is required due to reported or observed loss of SFP/Reactor cavity inventory. A conditional override step requires evacuation of all personnel from the Refuel Floor if either irradiated fuel is uncovered or the Refuel Bridge high range radiation monitor alarms. Neither of these conditions are met, as evidenced by water level being one foot below normal level (therefore fuel bundle in SFP are covered by over 20 feet of water still) and Refuel Bridge radiation monitors indicating 20 mr/hr and rising (well below the 50 mr/hr and 1000 mr/hr nominal high alarm setpoints). A subsequent step in N1-SOP-6.1 requires evacuating non-essential personnel from the Refuel Floor and returning any core component being transferred to the nearest storage location in the SFP or Reactor core. Since the irradiated fuel bundle is only half way out of the Reactor core, the nearest storage location is back in the Reactor core.

A. Incorrect – Only non-essential personnel need to be evacuated. Plausible because if conditions were worse, such that either irradiated fuel was uncovered or the Refuel Bridge high radiation alarm sounded, immediate evacuation of the Refuel Floor would be the required action in N1-SOP-6.1.
B. Incorrect – N1-SOP-6.1 requires returning any core component being transferred to the nearest storage location in the SFP or Reactor core. Since the irradiated fuel bundle is only half way out of the Reactor core, the nearest storage location is back in the Reactor core. Plausible because if the irradiated fuel bundle had already been transferred closer to the SFP, that would be the correct storage location to continue to. Additionally, going to the SFP instead of the Reactor core does eliminate a positive reactivity addition, which is desirable in certain situations.
D. Incorrect – N1-SOP-6.1 requires returning any core component being transferred to the nearest storage location in the SFP or Reactor core. Since the irradiated fuel bundle is only half way out of the Reactor core, the nearest storage location is back in the Reactor core. Plausible because leaving the irradiated fuel bundle in the current position does keep it covered with more water than going to the SFP and eliminates a positive reactivity addition associated with re-inserting it into the Reactor core.

Technical Reference(s): N1-SOP-6.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP6.1CO1 EO-2

Question Source: Bank – 2015 NRC #92

Question History: 2015 NRC #92

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 #	245000 A2.03
	Importance Rating	3.6

Main Turbine Generator/Auxiliary

Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of condenser vacuum

Question: #93

A plant startup is in progress with the following:

- Reactor power is 28%.
- Main Generator load is 125 MWe.
- Air in-leakage is causing Main Condenser vacuum to degrade.
- Main Condenser vacuum is 24.7” Hgv and slowly lowering.
- Turbine exhaust pressure is 5.3” Hga and slowly rising.

Which one of the following describes the required control of the plant, in accordance with N1-SOP-25.1, Unplanned Loss of Condenser Vacuum?

A. Place the standby SJAE in service per N1-OP-25. NO emergency power reduction, Turbine trip, or Reactor scram is required.

B. Perform an emergency power reduction per N1-SOP-1.1, Emergency Power Reduction. A Turbine trip or Reactor scram is NOT required.

C. Trip the Turbine per N1-SOP-31.1, Turbine Trip. A Reactor scram is NOT required.

D. Scram the Reactor per N1-SOP-1, Reactor Scram, and trip the Turbine per N1-SOP-31.1, Turbine Trip.

Proposed Answer: C

Explanation: With Turbine exhaust pressure >5” Hga and Main Generator load <190 MWe, N1-SOP-25.1 requires tripping the Turbine per N1-SOP-31.1. With Reactor power at 28%, F3-4-6, First Stage Bowl Pressure, is in alarm, so no Reactor scram is required.

A. Incorrect – A Turbine trip is required. Plausible because Main Condenser vacuum is well above 22.1” Hgv, which is the more typical Turbine trip benchmark used in N1-SOP-25.1 and the option is given to place a standby SJAE in service prior to 22.1” Hgv. However, the override conditions are met, requiring a turbine trip.
B. Incorrect – A Turbine trip is required. Plausible because Main Condenser vacuum is well above 22.1” Hgv, which is the more typical Turbine trip benchmark used in N1-SOP-25.1 and an emergency power reduction is directed to stabilize vacuum. However, the override conditions are met, requiring a turbine trip.
D. Incorrect – A Reactor scram is not required. Plausible because this would be correct if power were slightly higher.

Technical Reference(s): N1-SOP-25.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP25.1C01 EO-2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

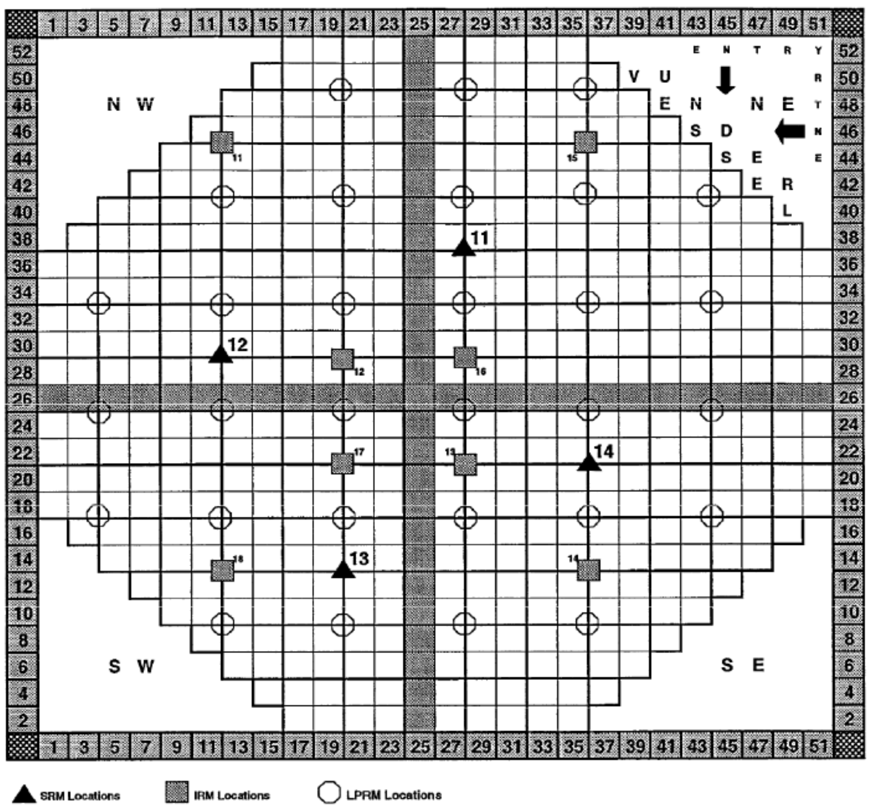
Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.40
	Importance Rating	3.9

Knowledge of refueling administrative requirements.

Question: #94

The plant is shutdown for a refueling outage with the following:

- The Reactor is fully loaded with fuel.
- Core shuffle phase 1 is about to begin.
- The SRMs indicate as follows:
  - SRM 11 - 2 cps
  - SRM 12 - 50 cps
  - SRM 13 - 0 cps, with the DN SCL OR INOP light ON
  - SRM 14 - 47 cps



Which one of the following describes the allowable fuel movements, if any, in accordance with Technical Specifications?

Fuel movements…

A. are NOT allowed.

B. may be performed in any of the core quadrants.

C. may be performed in two of the core quadrants, only.

D. may be performed in three of the core quadrants, only.

Proposed Answer: A

Explanation: SRMs 11 and 13 are both inoperable due to count rate being below 3 cps. A count rate less than 3 cps is allowed only if it occurs during a spiral unload and count rate was greater than 3 cps at the start. With SRMs 11 and 13 inoperable, core alterations are NOT allowed in either of their respective core quadrants (NE and SW). In addition, neither of the other two core quadrants meet the requirement to have an operable SRM in the quadrant and in an adjacent quadrant. Therefore, fuel moves are not allowed in any quadrant.

B. Incorrect – No fuel moves are allowed. Plausible because only one SRM is fully downscale and TS 3.5.1 only requires three SRMs operable.
C. Incorrect – No fuel moves are allowed. Plausible because two core quadrants have operable SRMs.
D. Incorrect – No fuel moves are allowed. Plausible because three of the SRMs are indicating some number of counts.

Technical Reference(s): Technical Specifications 3.5.1 and 3.5.3, N1-OP-34 P&L 14

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-234000-RBO-14

Question Source: Modified Bank – 2018 NRC #78

Question History:

Question Cognitive Level: Comprehension or Analysis

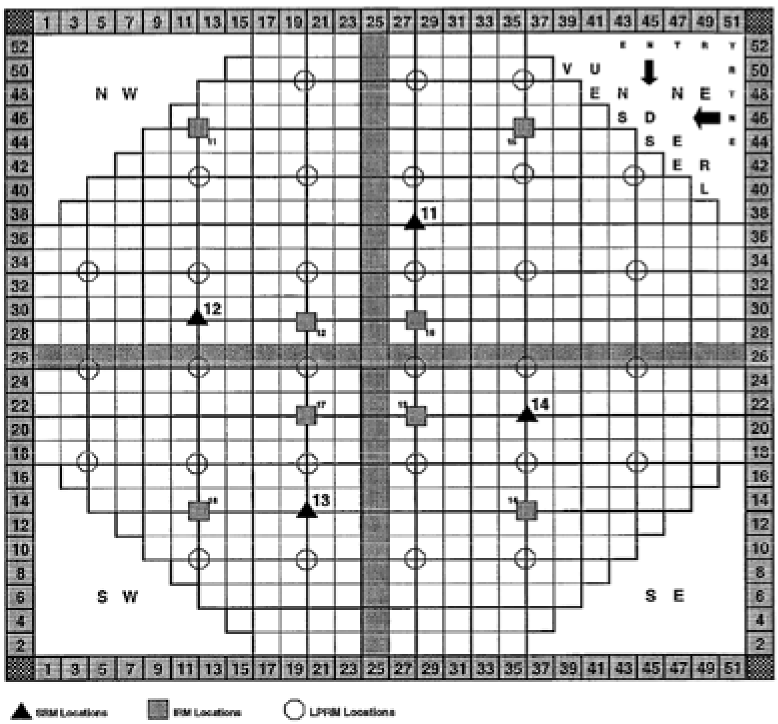
10 CFR Part 55 Content: 55.43(b)(6)

Comments:

Proposed Question: #78

The plant is shutdown for a refueling outage with the following:

- The Reactor is fully loaded with fuel.
- Core shuffle phase 1 is about to begin.
- The SRMs indicate as follows:
  - SRM 11 - 47 cps
  - SRM 12 - 50 cps
  - SRM 13 - 0 cps, with the DN SCL OR INOP light ON
  - SRM 14 - 2 cps



Which one of the following describes the allowable fuel movements, if any, in accordance with Technical Specifications?

Fuel movements…

A. are NOT allowed.

B. may be performed in any of the core quadrants.

C. may be performed in two of the core quadrants, only.

D. may be performed in three of the core quadrants, only.

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.35
	Importance Rating	3.9

Knowledge of the fuel-handling responsibilities of SROs.

Proposed Question: #94

Given the following evolutions:

(1) The plant is operating at 100% power and an irradiated fuel bundle is being moved within the Spent Fuel Pool in preparation for an outage.

(2) A refueling outage is in progress and an irradiated fuel bundle is being removed from the Reactor core and transferred to the Spent Fuel Pool.

(3) A refueling outage is in progress with fuel in the Reactor and a control rod blade is being removed from the Reactor core for replacement.

Which one of the following identifies the evolution(s) that must be directly supervised by a Senior Reactor Operator (SRO) or SRO Limited to Fuel Handling (LSRO) who has no other concurrent responsibilities, in accordance with S-ODP-NFM-0101, Refueling Operations, Fuel Handling Procedures (N1-FHPs), and the UFSAR?

A. (2) only

B. (1) and (2) only

C. (2) and (3) only

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

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.5
	Importance Rating	3.9

Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

Question: #95

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

- The shift is at minimum manning.
- Then, an Equipment Operator (EO) becomes injured while in a contaminated area.
- The EO is transported via ambulance to Oswego Hospital.
- The RP Tech accompanies the EO to the hospital.
- The ambulance leaves the site at 0300.
- The on-coming shift is expected to arrive at 0600.

Which one of the following describes the requirement to replace the EO and the RP Tech in accordance with Technical Specifications?

A. Both the EO and RP Tech must be replaced earlier than the expected arrival of the on-coming shift.

B. The EO must be replaced earlier than the expected arrival of the on-coming shift. The RP Tech does NOT need to be replaced earlier than the expected arrival of the on-coming shift.

C. The RP Tech must be replaced earlier than the expected arrival of the on-coming shift. The EO does NOT need to be replaced earlier than the expected arrival of the on-coming shift.

D. NEITHER the EO NOR the RP Tech must be replaced earlier than the expected arrival of the on-coming shift.

Proposed Answer: A

Explanation: Both EOs and an RP Tech are part of required for shift manning. Since the shift started at minimum manning, loss of both the EO and the RP Tech takes the shift below minimum manning in both of these positions. Technical Specifications allow up to 2 hours to return shift manning to at least the minimum requirements. Since the on-coming shift is not expected to arrive for another 3 hours, both the EO and the RP Tech must be replaced earlier than arrival of the on-coming shift.

B. Incorrect – The RP Tech is part of minimum shift manning and must be replaced within 2 hours (by 0500). Plausible that the RP Tech would not be part of minimum shift staffing since they are not in Operations.
C. Incorrect – The EO is part of minimum shift manning and must be replaced within 2 hours (by 0500). Plausible that the EO would be able to extend longer since they are not licensed.
D. Incorrect – Both the EO and the RP Tech are part of minimum shift manning and must be replaced within 2 hours (by 0500). Plausible that the RP Tech would not be part of minimum shift staffing since they are not in Operations. Plausible that the EO would be able to extend longer since they are not licensed.

Technical Reference(s): Technical Specification 6.2.2, OP-NM-103-102

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – JAF 14-1 NRC #67

Question History: JAF 14-1 NRC #67

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.17
	Importance Rating	3.8

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

Question: #96

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

- A job is being planned.
- The job has the potential to result in an unplanned 7 day shutdown LCO entry.
- The job has the potential to cause an unplanned load reduction of 50 MWe.

Which one of the following describes the need to classify this job as an Operational Risk Activity (ORA), in accordance with OP-AA-107, Integrated Risk Management?

This job…

A. does NOT need to be classified as an ORA.

B. needs to be classified as an ORA, based on the potential unplanned LCO entry, only.

C. needs to be classified as an ORA, based on the potential unplanned load reduction, only.

D. needs to be classified as an ORA, based on both the potential unplanned LCO entry and unplanned load reduction.

Proposed Answer: C

Explanation: OP-AA-107 requires classifying this job as an ORA because the potential for an unplanned power change exceeds 20 MWe.

A. Incorrect – OP-AA-107 requires classifying this job as an ORA. Plausible because this would be correct if the potential for an unplanned power change was less than 20 MWe.
B. Incorrect – The potential unplanned LCO entry time does not require classification of the job as an ORA. Plausible because it would if it was 12 hours or less.
D. Incorrect – The potential unplanned LCO entry time does not require classification of the job as an ORA. Plausible because it would if it was 12 hours or less.

Technical Reference(s): OP-AA-107

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.25
	Importance Rating	4.2

Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Question: #97

Technical Specification 3.2.4, Reactor Coolant Specific Activity, limits the allowable specific activity in the Reactor coolant.

Which one of the following describes the limitation and the basis behind this limitation, in accordance with Technical Specifications?

Limits specific activity of dose equivalent ___(1)___ in the Reactor coolant to prevent exceeding 10 CFR limits during a design basis ___(2)___.

	(1)	(2)
A.	Cesium 137	Recirc loop rupture
B.	Cesium 137	Main Steam line break
C.	Iodine 131	Recirc loop rupture
D.	Iodine 131	Main Steam line break

Proposed Answer: D

Explanation: Technical Specification 3.2.4 limits the dose equivalent Iodine 131. The basis is to prevent exceeding dose limits following a Main Steam line break.

A. Incorrect – Technical Specification 3.2.4 limits the dose equivalent Iodine 131. Plausible because Cesium 137 is a significant radioisotope referenced in the ODCM. The basis behind the limit is a Main Steam line break. Plausible because a Recirc loop rupture is the basis for many other requirements.
B. Incorrect – Technical Specification 3.2.4 limits the dose equivalent Iodine 131. Plausible because Cesium 137 is a significant radioisotope referenced in the ODCM.
C. Incorrect – The basis behind the limit is a Main Steam line break. Plausible because a Recirc loop rupture is the basis for many other requirements.

Technical Reference(s): Technical Specification 3.2.4 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: Bank – JAF 14-2 NRC #94

Question History: JAF 14-2 NRC #94

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.11
	Importance Rating	4.3

Ability to control radiation releases.

Question: #98

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

- Annunciator H1-4-5, Liquid Process Rad Monitor, alarms.
- An Operator reports that the EQUIP FAIL light on the Service Water Rad Monitor is ON.
- Chemistry has been notified and determines that Service Water release rates are normal.

Which one of the following actions is required to allow Service Water operation to continue, in accordance with the Offsite Dose Calculation Manual (ODCM)?

A. Verify the other Service Water Radiation Monitor is operable within 12 hours.

B. Collect and analyze Service Water effluent grab samples at least once per 12 hours.

C. Sample Reactor Building and Turbine Building Service Water return lines alternately every 15 minutes.

D. Collect and analyze two independent Service Water effluent grab samples and have two technically qualified individuals verify calculations and valving.

Proposed Answer: B

Explanation: ODCM Table 3.6.14-1 requires a Service Water effluent line radiation monitor to be operable. With the only installed Service Water effluent line radiation monitor inoperable, Note (d) applies and requires “With the number of channels functional less than required by the minimum channels functional requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed…”.

A. Incorrect – Only one Service Water radiation monitor is installed. Plausible because most rad monitors have a backup. However, NMP1 only has one service water rad monitor that automatically shifts sampling between the TB SW header and the RB SW header. ODCM Table 3.6.14-1 requires only one Service Water radiation monitor to be operable, so if there was a second one, the table would be met as long as it was operable.
C. Incorrect – The requirement is to analyze grab samples every 12 hours. Plausible because this is patterned after ODCM Table 3.6.14-1 Note (c), but this note applies to liquid radwaste radiation monitoring, not Service Water.
D. Incorrect – The requirement is to analyze grab samples every 12 hours. Plausible because ODCM Table 3.6.14-1 Note (i), "Monitoring will be conducted continuously by alternately sampling the reactor building and turbine building service water return lines for approximately 15-minute intervals", applies to the normal sampling during all modes of operation, but not this situation.

Technical Reference(s): ARP H1-4-5, ODCM 3.6.14, N1-OP-50B

Proposed references to be provided to applicants during examination: ODCM 3.6.14

Learning Objective: N1-272000-RBO-14

Question Source: Bank – 2015 NRC #96

Question History: 2015 NRC #96

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.19
	Importance Rating	4.1

Knowledge of EOP layout, symbols, and icons.

Question: #99

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

- The Reactor has been manually scrammed.
- N1-EOP-4, Primary Containment Control, has been entered due to high Drywell temperature and pressure, only.

Which one of the following describes how to implement N1-EOP-4, in accordance with the EOPs and bases?

A. All legs are entered. There is a pre-determined order for which leg to execute first.

B. All legs are entered. There is NOT a pre-determined order for which leg to execute first.

C. Only the DWT and PCP legs are entered. There is a pre-determined order for which leg to execute first.

D. Only the DWT and PCP legs are entered. There is NOT a pre-determined order for which leg to execute first.

Proposed Answer: B

Explanation: When an EOP entry condition is met, the EOP is entered and all legs are entered concurrently. There is no pre-determined prioritization between the legs. Prioritization is left up to the EOP director.

Note: Question meets the KA statement because the SRO must know what the filled circles in EOPs symbolize in order to answer the question.

A. Incorrect – There is no pre-determined prioritization between the legs. Prioritization is left up to the EOP director. Plausible that a pre-determined order would be given to ensure consistent implementation.
C. Incorrect – When an EOP entry condition is met, the EOP is entered and all legs are entered concurrently. Plausible because not all legs have associated entry conditions met. There is no pre-determined prioritization between the legs. Prioritization is left up to the EOP director. Plausible that a pre-determined order would be given to ensure consistent implementation.
D. Incorrect – When an EOP entry condition is met, the EOP is entered and all legs are entered concurrently. Plausible because not all legs have associated entry conditions met.

Technical Reference(s): N1-EOP-4, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO-2

Question Source: Bank – JAF 17-2 NRC #100

Question History: JAF 17-2 NRC #100

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.22
	Importance Rating	4.4

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

Question: #100

The plant has experienced an earthquake with the following:

- The Diesel Fire pump is the only available pump.
- Firefighting activities are in progress.
- Reactor water level is 50" and slowly lowering.
- Spent Fuel Pool water level is 3" below the level of the weirs and slowly lowering.

Which one of the following describes the prioritization for use of the Diesel Fire pump, in accordance with NER-1M-095, NMP1 Emergency Operating Procedures and Severe Accident Procedures Basis Document, and OP-NM-101-111-1001, Transient Mitigation Guidelines?

The EOP Director…

A. is allowed to set the priority for use of the Diesel Fire pump based on plant conditions.

B. must make lining up the Diesel Fire pump for Reactor injection the priority.

C. must make lining up the Diesel Fire pump for Spent Fuel Pool makeup the priority.

D. must make maintaining the Diesel Fire pump lined up to the Fire Water header the priority.

Proposed Answer: A

Explanation: NER-1M-095 and OP-NM-101-111-1001 both specify that there is no pre-determined prioritization for pump use or between EOPs. EOPs are to be implemented concurrently and priorities are set as determined by the EOP Director based on plant conditions.

B. Incorrect – Reactor injection does NOT always take priority over firefighting activities and Spent Fuel Pool makeup. Plausible because adequate core cooling, if challenged, is given priority over other concerns in the EOPs.
C. Incorrect – Spent Fuel Pool makeup does NOT always takes priority over Reactor injection and firefighting activities. Plausible because low Spent Fuel Pool level can lead to a direct release to the environment, whereas the Reactor is inside a Primary Containment.
D. Incorrect – Firefighting activities do NOT always take priority over Reactor injection and Spent Fuel Pool makeup. Plausible because firefighting is the design purpose of the Diesel Fire pump, whereas other systems can be made available for these other functions.

Technical Reference(s): NER-1M-095, OP-NM-101-111-1001

Proposed references to be provided to applicants during examination: None

Learning Objective:

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(5)

Comments: