Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295001 AK3.03
	Importance Rating	2.8

Partial or Complete Loss of Forced Core Flow Circulation

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

Proposed Question: #1

N1-SOP-1.3, Recirc Pump Trip at Power, contains the following step:

Notify SM that APRMs are inoperable.

Which one of the following describes the reason for this step?

Reverse flow through the idle Recirc loop results in __(1)__ until either the __(2)__.

	(1)	(2)
A.	non-conservative power-to-flow scram setpoints	pump discharge or suction valve is closed
B.	non-conservative power-to-flow scram setpoints	pump control switch is green-flagged or placed in pull-to-lock
C.	APRM indications reading erroneously high	pump discharge or suction valve is closed
D.	APRM indications reading erroneously high	pump control switch is green-flagged or placed in pull-to-lock

Proposed Answer: A

Explanation: From N1-SOP-1.3 – "A Recirc Pump trip at power results in non-conservative Recirc Flow-biased APRM scram and rod block trip setpoints due to the reverse-flow through the non-isolated Recirc Loop still being measured as part of total core flow. The Recirc Flow-biased APRM scram and rod block trip functions are inoperable until the tripped Recirc Pump's associated discharge valve is closed…" Additionally, if the discharge valve fails to close, SOP-1.3 directs closing the associated suction valve to stop the reverse flow.

B. Incorrect – The Recirc pump control switch position does not affect the Recirc flow summer.
C. Incorrect – The APRMs continue to read proper neutron flux following a Recirc pump trip.
D. Incorrect – The APRMs continue to read proper neutron flux following a Recirc pump trip. The Recirc pump control switch position does not affect the Recirc flow summer.

Technical Reference(s): N1-SOP-1.3

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP1.3C01 EO #2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295003 AA1.01
	Importance Rating	3.7

Partial or Complete Loss of AC Power

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

Proposed Question: #2

The plant is operating at 100% when the following occur:

- An electrical fault develops between breaker R10 and disconnect SW 168.
- Breakers R10 and R40 open.
- R10 and R40 Auto-Reclosure switches are in ON.

Which one of the following describes the electrical distribution system lineup five minutes later?

	Breaker R40	MOD 8106
A.	Closed	Closed
B.	Closed	Open
C.	Open	Closed
D.	Open	Open

Proposed Answer: B

Explanation: If a 115 kV bus fault occurs, breakers R10 and R40 open. If 115 kV lines are energized (with 115 kV bus de-energized), breakers R10 and R40 attempt sequential reclosure. Reclosure will fail due to the location of the fault downstream of R10. Since reclosure fails, bus sectionalizing disconnect switch MOD 8106 opens and then R10 and R40 attempt another reclosure to re-energize the un-faulted section of the 115 kV bus. R10 will fail to close due to the location of the fault, but with MOD 8106 open, R40 will be isolated from the fault and stay closed. This entire sequence completes within five minutes.

A. Incorrect – MOD 8106 will open to isolate the fault.
C. Incorrect – R40 will close and remain closed once MOD 8106 opens.
D. Incorrect – R40 will close and remain closed once MOD 8106 opens.

Technical Reference(s): N1-OP-33A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262000-RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295004 AK2.02
	Importance Rating	3.0

Partial or Complete Loss of DC Power

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

Proposed Question: #3

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

- A3-4-4, BAT. BD. 12 BATTERY BREAKER TRIP

Which one of the following describes the extent of the DC power loss, if any?

A. Static Battery Charger 171 maintains DC power to Battery Board 12 without a loss of power.

B. DC Valve Board 12 automatically transfers to Battery Board 11. All other DC power from Battery Board 12 is lost.

C. MG Set 167 automatically swaps to “BATTERY CHARGE” mode and supplies Battery Board 12 after a momentary loss of power.

D. A complete loss of DC power to Battery Board 12 occurs.

Proposed Answer: D

Explanation: With the trip of #12 Battery Breaker, the battery charger DC output breaker trips as well due to an electrical interlock. This results in a total loss of DC to Battery Board 12. There are no automatic features enabled to re-power the battery board or the panels it feeds.

A. Incorrect – Trip of the battery breaker also results in automatic trip of the Static Battery Charger output breaker.
B. Incorrect – DC Valve Board 12 does not automatically transfer to Battery Board 11, but may be manually transferred later.
C. Incorrect – MG Set 167 does not automatically transfer to battery charge mode, but may be manually aligned later.

Technical Reference(s): ARP-A3, OP-47A P&L

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-05

Question Source: Bank – 2009 NRC #49

Question History: 2009 NRC #49

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

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

Main Turbine Generator Trip

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

Proposed Question: #4

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

- Powerboards 11 and 12 are energized from House Service Transformer T-10.
- Main Turbine vibrations are rising.
- An Operator manually trips the Main Turbine.

Which one of the following describes the electrical distribution system response?

Powerboards 11 and 12 __(1)__ transfer to Reserve power based on a(n) __(2)__ signal.

	(1)	(2)
A.	slow	undervoltage
B.	slow	Generator trip
C.	fast	undervoltage
D.	fast	Generator trip

Proposed Answer: D

Explanation: At 25% power, a Main Turbine trip does NOT result in a Reactor scram, however it still does result in a Generator trip. When the Generator trips, Powerboard 11 and 12 fast transfer is initiated. Powerboard 11 and 12 also have a slow transfer mechanism in the event of degraded voltage on the board. The fast transfer mechanism is preferable when normal voltage is present so that power is not interrupted to loads. The slow transfer mechanism is only necessary with degraded voltage conditions to ensure Powerboard voltage decays before connecting to the Reserve power source. Nothing in the stem of the question indicates degraded voltage, therefore the fast transfer will occur.

A. Incorrect – Slow transfer will NOT occur since no degraded voltage conditions are present. Undervoltage will NOT occur since the Generator trip will directly cause fast transfer, which happens quick enough to preserve Powerboard voltage.
B. Incorrect – Slow transfer will NOT occur since no degraded voltage conditions are present.
C. Incorrect – Undervoltage will NOT occur since the Generator trip will directly cause fast transfer, which happens quick enough to preserve Powerboard voltage.

Technical Reference(s): OP-30

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262001-RBO-05
N1-245001-RBO-05

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:

Level	RO
Tier #	1
Group #	1
K/A #	295006 2.4.6
Importance Rating	3.7

SCRAM

Knowledge of EOP mitigation strategies

Proposed Question: #5

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

- Feedwater pump 13 clutch dis-engages.
- An Operator places the mode switch in Shutdown.
- One control rod sticks at position 36.
- All other control rods fully insert.
- Reactor water level reaches a low of 15 inches and then begins rising.

Which one of the following describes the correct strategies for Reactor water level and pressure control, in accordance with the Emergency Operating Procedures?

	Reactor Water Level Control	Reactor Pressure Control
A.	Restore and maintain Reactor water level 53-95".	Reactor cooldown is allowed.
B.	Restore and maintain Reactor water level 53-95".	Reactor cooldown is NOT allowed.
C.	Terminate and prevent injection and lower Reactor water level to at least -41".	Reactor cooldown is allowed.
D.	Terminate and prevent injection and lower Reactor water level to at least -41".	Reactor cooldown is NOT allowed.

Proposed Answer: A

Explanation: Upon entry to EOP-2, diagnostic questions related to the success of the scram determine which EOP mitigation strategy will be followed. If the Reactor will stay shutdown under all conditions without boron, EOP-2 strategies are used and EOP-3 is NOT entered. In this case, the Reactor will stay shutdown under all conditions without boron based on shutdown margin criteria. Only in EOP-3 would Reactor water level be intentionally lowered and a cooldown NOT be permitted. EOP-2 directs normal Reactor water level control in a band of 53-95". EOP-2 also permits Reactor cooldown under these conditions, even with one control rod partially withdrawn.

B. Incorrect – Reactor cooldown is allowed.
C. Incorrect – Reactor water level is to be restored and maintained 53-95".
D. Incorrect – Reactor water level is to be restored and maintained 53-95". Reactor cooldown is allowed.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO #2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295016 AA1.08
	Importance Rating	4.0

Control Room Abandonment

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

Proposed Question: #6

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

- A fire starts in the Control Room.
- N1-SOP-21.2, Control Room Evacuation, is entered.
- The immediate actions of N1-SOP-21.2 are performed.
- All Operators exit the Control Room.

Which one of the following describes the required Reactor pressure control action per N1-SOP-21.2?

A. Commence a Reactor cooldown using Turbine Bypass Valves.

B. Commence a Reactor cooldown using Emergency Condensers.

C. Maintain Reactor pressure 800-1000 psig using Turbine Bypass Valves.

D. Maintain Reactor pressure 800-1000 psig using Emergency Condensers.

Proposed Answer: B

Explanation: The immediate actions of SOP-21.2 require closure of the MSIVs, therefore Turbine Bypass Valves are unavailable for pressure control. SOP-21.2 directs a cooldown be established using the Emergency Condensers from the Remote Shutdown Panels.

A. Incorrect – Turbine Bypass Valves are unavailable because the MSIVs are closed per the immediate actions of SOP-21.2.
C. Incorrect – Turbine Bypass Valves are unavailable because the MSIVs are closed per the immediate actions of SOP-21.2. SOP-21.2 directs a cooldown be commenced.
D. Incorrect – SOP-21.2 directs a cooldown be commenced.

Technical Reference(s): SOP-21.2

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	1
	Group #	1
	K/A #	295018 AA2.02
	Importance Rating	3.1

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: Cooling water temperature

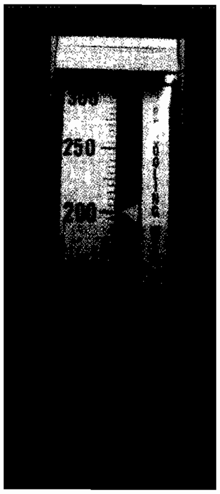
Proposed Question: #7

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

- Annunciator H1-4-1, R BUILDING COOLING WATER PRESS TEMP MAKEUP FLOW, alarms.
- The Plant Process Computer (PPC) is unavailable.
- The RBCLC indications on the next page are available in the Control Room.

Which one of the following describes the parameter causing the alarm and the required Operator action to correct the condition?

	Parameter	Action
A.	Pressure	Start an additional RBCLC pump.
B.	Pressure	Initiate leak identification and isolation.
C.	Temperature	Take manual control and open the RBCLC TCV.
D.	Temperature	Take manual control and close the RBCLC TCV.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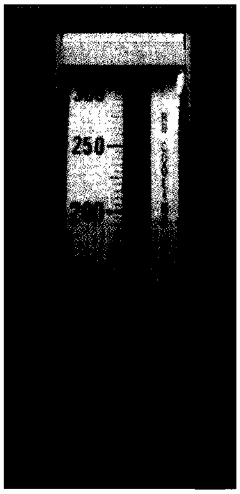
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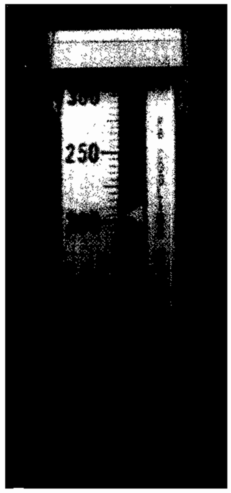
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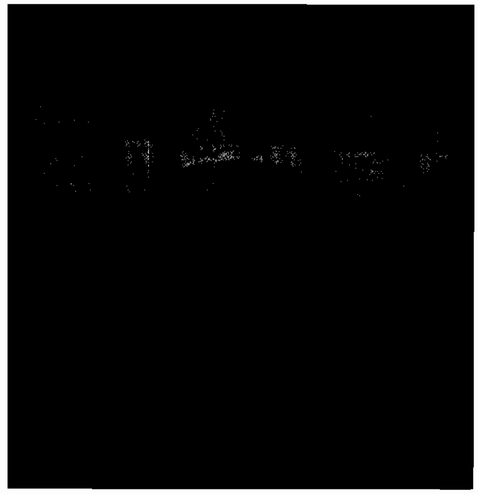
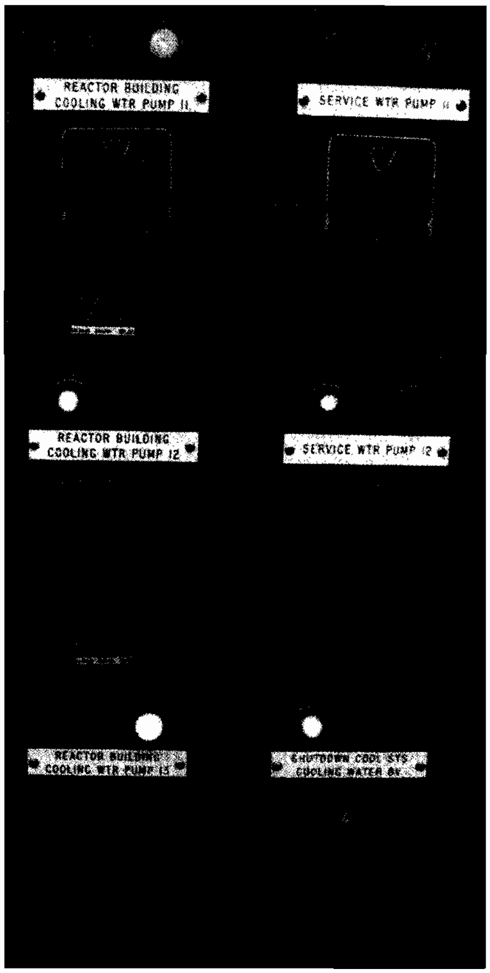
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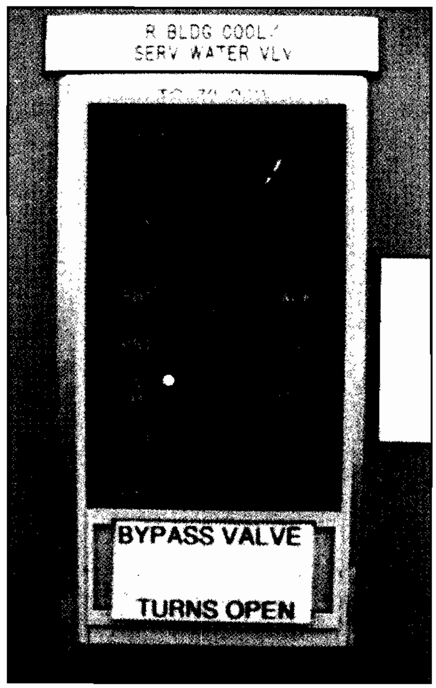

R BLDG COOL/
SERV WATER VLV
BYPASS VALVE
TURNS OPEN

Proposed Answer: C

Explanation: The indications show RBCLC supply temperature is approximately 95°F, above the alarm setpoint of 92.5°F. RBCLC pressure is at the normal value of approximately 94 psig. The indications also show the RBCLC temperature controller is at minimum demand. The controller should be demanding further TCV opening with a high temperature condition, indicating controller or valve failure. The ARP directs taking manual control of the RBCLC TCV (either manual on controller or manual in field). The valve needs to be further opened to cause more cooling to lower temperature.

A. Incorrect – RBCLC pressure is at a normal value.
B. Incorrect – RBCLC pressure is at a normal value.
D. Incorrect – The RBCLC TCV must be further opened to cause more cooling to lower temperature.

Technical Reference(s): ARP H1-4-1, C-18022-C Sheet 2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-208000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295019 AA1.01
	Importance Rating	3.5

Partial or Complete Loss of Instrument Air

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Backup air supply

Proposed Question: #8

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

- Instrument Air Compressor (IAC) 13 is running.
- IAC 11 is red-flagged with Loading Selector Switches in Position 1.
- IAC 12 is green-flagged with Loading Selector Switches in Position 2.

Then, an Instrument Air (IA) line break results in the following sequence of events:

Time (mm:ss)	Event
00:10	Instrument Air Compressor (IAC) 13 loads.
00:20	IAC 11 loads.
00:30	IAC 12 starts and loads.
00:45	Annunciator L1-4-7, INST AIR BACK – UP VALVE OPEN, alarms.
00:57	Breathing Air Blocking Valve 114-02 closes; Automatic Dryer Bypass valves 94-164 and 94-206 open.

IA pressure has still been continuously lowering throughout this transient.

Which one of the following describes Instrument Air pressure at time 00:25?

At time 00:25, Instrument Air pressure is between…

A. 98 and 101 psig.

B. 93 and 98 psig.

C. 90 and 93 psig.

D. 85 and 90 psig.

Proposed Answer: B

Explanation: IAC 11 is red-flagged as the backup compressor with Load Selector Switches in Position 1, therefore it loads on an IA pressure of 98#. IAC 12 is green-flagged as the standby compressor, therefore it starts and loads at an IA pressure of 93#, then attempts to control pressure 96#-102#.

A. Incorrect – IA pressure was ~101# at time 00:10 and ~98# at time 00:20 based on IAC 13 and 11 loading, respectively.
C. Incorrect – IA pressure was ~93# at time 00:30 and ~ 90# at time 00:45 based on IAC 12 start and Annunciator L1-4-7, respectively.
D. Incorrect – IA pressure was ~90# at time 00:45 and ~85# at time 00:57 based on Annunciator L1-4-7 and valve operations, respectively.

Technical Reference(s): OP-20, ARP L1-4-7

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278001-RBO-05

Question Source: Modified 2010 NMP1 NRC #52

Question History: 2010 NMP1 NRC #52

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(4)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295021 AA2.06
	Importance Rating	3.2

Loss of Shutdown Cooling

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

Proposed Question: #9

A plant shutdown is in progress after 437 days of continuous operation with the following:

- Shutdown Cooling (SDC) Loop 12 is in operation.
- Reactor water level is 90 inches and stable.
- Reactor coolant temperature is 338°F and slowly lowering.
- Reactor pressure is 116 psig and slowly lowering.

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

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

Proposed Answer: C

Explanation: The SDC heat exchangers are cooled by water from the Reactor Building Closed Loop Cooling (RBCLC) Water System. Loss of cooling will cause Reactor temperature and pressure to rise. SDC isolation valves are interlocked to prevent placing the system in service above 120 psig. However, once open, the SDC isolation valves will NOT automatically close if pressure rises above 120 psig. SDC pumps have a start-permissive requiring coolant temperature below 350°F. Additionally, SDC pumps will trip if coolant temperatures rise to 350°F.

A. Incorrect - There are no valve closures on pressure. Pressure is only an open-permissive for these valves. The isolation valves close on:
    - Reactor Vessel Level Low-Low (>+5")
    - High Area Temperature (170°F T.S. Limit)
    - Manual Isolation
B. Incorrect - Valves close on high space temperature indicative of a leak, not on high coolant temperature.
D. Incorrect - SDC pump does not trip on high pressure.

Technical Reference(s): OP-4, C-19859-C Sheet 12, C-19436-C Sheet 3, C-19436-C Sheet 2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-08

Question Source: Bank – 2008 NMP1 NRC #9

Question History: 2008 NMP1 NRC #9

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

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

Refueling Accidents

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

Proposed Question: #10

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

- Fuel movement is in progress in the Spent Fuel Pool in preparation for a refueling outage.
- An irradiated fuel assembly has just been loaded on the Refuel Bridge main hoist and raised to the full up position.

Then, a seismic event results in the following:

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

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

Spent Fuel Pool water level ___(1)___. Spent Fuel Pool makeup water is ___(2)___.

	(1)	(2)
A.	maintains the fuel assembly fully covered	still available from the normal makeup source
B.	maintains the fuel assembly fully covered	NOT available from the normal makeup source
C.	lowers and partially uncovers the fuel assembly	available using hoses from Fire Water only
D.	lowers and partially uncovers the fuel assembly	available using hoses from Condensate Transfer, Demineralized Water, and Fire Water

Proposed Answer: A

Explanation: A complete rupture of the SFPC pumps common discharge line results in loss of water circulation back to the SFP. This will result in SFP water level dropping about 2" due to the height of the weirs. However, siphoning of SFP water back through the rupture is prevented by vacuum breakers installed in the discharge piping. The design of the Refuel Bridge main hoist also ensures this SFP water level will maintain the elevated fuel assembly covered with water. While the backup source of makeup water (from Condensate Transfer to the SFPC skimmer surge tanks) will be unavailable for SFP makeup due to the pipe break, the normal source of makeup water (from Condensate Transfer directly to the SFP) will still be available.

B. Incorrect – Normal makeup from Condensate Transfer to the SFP is still available through LCV 57-25.
C. Incorrect – SFP water level will lower about 2" to the top of the weirs, which will still maintain the fuel assembly fully covered with water. Normal makeup water is also available from Condensate Transfer and Demineralized Water.
D. Incorrect – SFP water level will lower about 2" to the top of the weirs, which will still maintain the fuel assembly fully covered with water.

Technical Reference(s): C-18008-C, SOP-6.1, OP-6, 1101-233000C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-233000-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(13)

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

High Drywell Pressure

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

Proposed Question: #11

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

- Drywell pressure is 8 psig and rising slowly.
- Torus pressure is 6 psig and rising slowly.
- Drywell temperature is 310°F and rising slowly.

Which one of the following describes how Containment Spray is required to be controlled and the associated reason, in accordance with N1-EOP-4, Primary Containment Control?

A. Initiate Containment Spray because Drywell pressure is above 3.5 psig.

B. Initiate Containment Spray because Drywell temperature is above 300°F.

C. Do NOT initiate Containment Spray because Torus pressure is below 13 psig.

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

Proposed Answer: D

Explanation: With Drywell temperature above 150°F and Drywell pressure above 3.5 psig, EOP-4 is entered. EOP-4 requires Containment Spray initiation before Drywell temperature exceeds 300°F or once Torus pressure exceeds 13 psig. However, even if these parameters are exceeded, Containment Spray is NOT initiated if the Containment Spray Initiation Limit (CSIL) curve is exceeded. The combination of Drywell temperature and pressure given exceed CSIL, therefore Containment Spray must NOT be initiated.

A. Incorrect – Drywell pressure above 3.5 psig is an EOP-4 entry condition and part of the Containment Spray auto-initiation logic, but not a condition requiring sprays in EOP-4.
B. Incorrect – Containment Spray is normally required before (or above) 300°F Drywell temperature, but not when CSIL is exceeded.
C. Incorrect – While Containment Spray is required when Torus pressure is above 13 psig, they are also required before (or above) 300°F Drywell temperature. Torus pressure being below 13 psig is not the reason associated with the current requirement to not initiate Containment Spray

Technical Reference(s): EOP-4

Proposed references to be provided to applicants during examination: CSIL curve

Learning Objective: 1101-EOP4C01 EO #2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295025 2.1.27
	Importance Rating	3.9

High Reactor Pressure

Knowledge of system purpose and/or function

Proposed Question: #12

Which one of the following describes the basis for the Electromatic Relief Valve (ERV) opening setpoints?

The ERV opening setpoints are based on preventing...

A. lifting of the Reactor Safety valves in the event of a Main Turbine trip with failure of the Turbine Bypass Valves to open.

B. lifting of the Reactor Safety valves in the event of MSIV closure with failure of the MSIV closure scram.

C. exceeding the Reactor high pressure safety limit in the event of a Main Turbine trip with failure of the Turbine Bypass Valves to open.

D. exceeding the Reactor high pressure safety limit in the event of MSIV closure with failure of the MSIV closure scram.

Proposed Answer: A

Explanation: The setpoints for the ERVs are set to ensure the Reactor Safety valves will not open following a rapid isolation (Turbine trip) with failure of the Turbine Bypass Valves to relieve pressure.

B. Incorrect – MSIV closure with failure of MSIV closure scram is part of the basis for Reactor Safety valve setpoints.
C. Incorrect – The Reactor Safety valves are set to prevent exceeding the Reactor high pressure safety limit.
D. Incorrect – The Reactor Safety valves are set to prevent exceeding the Reactor high pressure safety limit. MSIV closure with failure of MSIV closure scram is part of the basis for Reactor Safety valve setpoints.

Technical Reference(s): Safety Limit 2.2.2 Bases, Tech Spec 3.2.9 Objective

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(3)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295026 EA1.03
	Importance Rating	3.9

Suppression Pool High Water Temperature

Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Temperature monitoring

Proposed Question: #13

The plant is operating at 100% power when an ERV inadvertently lifts.

Which one of the following describes the threshold for Torus water temperature that (1) requires entry into Emergency Operating Procedures (EOPs) and (2) requires a manual Reactor scram?

	(1)	(2)
A.	80°F	95°F
B.	80°F	110°F
C.	85°F	95°F
D.	85°F	110°F

Proposed Answer: D

Explanation: EOP-4, Primary Containment Control, is required to be entered when Torus water temperature exceeds 85°F. A manual Reactor scram is required before Torus water temperature exceeds 110°F. This is requirement of SOP-1.4, EOP-4, and Technical Specifications.

A. Incorrect – 80°F is the setpoint for annunciator F1-2-8, however EOP entry is not required until Torus water temperature exceeds 85°F. 95°F is a limit for Torus water temperature during testing, however a manual Reactor scram is not required until 110°F.
B. Incorrect – 80°F is the setpoint for annunciator F1-2-8, however EOP entry is not required until Torus water temperature exceeds 85°F.
C. Incorrect – 95°F is a limit for Torus water temperature during testing, however a manual Reactor scram is not required until 110°F.

Technical Reference(s): EOP-4, SOP-1.4, Technical Specifications, ARP F1-2-8

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO #2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295028 EK1.02
	Importance Rating	2.9

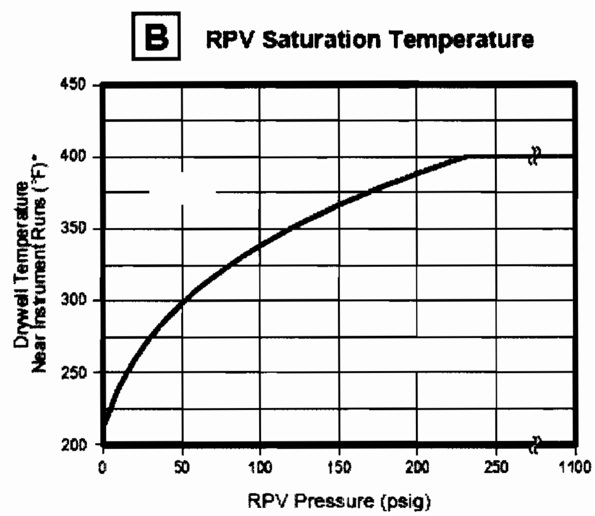
High Drywell Temperature

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Equipment environmental qualification

Proposed Question: #14

The plant is shutdown due to a small coolant leak with the following:

- Drywell temperature at elevation 319' is 325°F and slowly rising.
- Reactor pressure is 60 psig and slowly lowering.
- Reactor water level is 72" and stable.



Which one of the following describes the implication of these parameters on the validity of the GEMAC Reactor water level indicators?

GEMAC Reactor water level indicators…

A. are valid regardless of these parameters due to density compensation.

B. may be indicating lower than actual level due to these parameters.

C. may be indicating higher than actual level due to these parameters.

D. are NOT allowed to be used due to the effects of these parameters.

Proposed Answer: C

Explanation: The “RPV Saturation Temperature” (Figure B of Detail A) is a plot of the saturation temperature of water as a function of pressure. If the temperature of the water in a Reactor water level instrument run exceeds this temperature, the water may start to boil, resulting in unreliable level indication. Boil-off from the reference leg reduces the height of water in the leg. This decreases the pressure on the reference leg side of the dP cell and increases the indicated level. Boiling in the variable leg increases the pressure on the variable leg side of the dP cell, likewise increasing the indicated level.

A. Incorrect – The given values of Drywell temperature and Reactor pressure are in the BAD region of the RPV Saturation Temperature curve. This curve does apply to GEMACs. Fuel zone indicators are compensated and flash if invalid, such that this curve need not be evaluated.
B. Incorrect – GEMACs may be indicating higher than actual level due to boiling.
D. Incorrect – The wording of Detail A permits continued use of a level instrument until boiling is actually observed.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-216000-RBO-12

Question Source: Modified NMP1 2008 Audit #53

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Comments:
TRH 1/15/13 – Ran K/A match past NRC Chief Examiner and he is good with it. Based on discussion, covered “Good” and “Bad” labels on provided graph.

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295030 2.4.1
	Importance Rating	4.6

Low Suppression Pool Water Level

Knowledge of EOP entry conditions and immediate action steps

Proposed Question: #15

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

- A leak has developed from the Core Spray suction piping into the Reactor Building Southeast corner room.
- Torus water level is 10.4 feet and lowering slowly.
- An Operator in the field reports area water level is approximately 3" in the Southeast corner room.

Which one of the following describes the Emergency Operating Procedures that require entry?

A. N1-EOP-4, Primary Containment Control, only.

B. N1-EOP-5, Secondary Containment Control, only.

C. N1-EOP-4, Primary Containment Control, and N1-EOP-5, Secondary Containment Control, only.

D. N1-EOP-4, Primary Containment Control, N1-EOP-5, Secondary Containment Control, and N1-EOP-2, RPV Control.

Proposed Answer: C

Explanation: EOP-4 must be entered due to Torus water level below 10.5 feet. EOP-5 must be entered due to area water level above 0 inches. EOP-2 entry may be eventually required, but not until driven by EOP-4 (before Torus water level reaches 8.0 feet). With Torus water level just below the EOP-4 entry condition, no information about location of leak (isolable vs. un-isolable), and no information on Torus makeup capability, there is no immediate indication that a scram and EOP-2 entry is required.

A. Incorrect – EOP-5 entry is also required due to area water level above 0 inches.
B. Incorrect – EOP-4 must also be entered due to Torus water level below 10.5 feet.
D. Incorrect – EOP-2 entry is only required before Torus water level reaches 8.0 feet, and after Torus makeup / leak isolation has been attempted.

Technical Reference(s): EOP-4, EOP-5

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 EO #2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295031 EK1.01
	Importance Rating	4.6

Reactor Low Water Level

Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: Adequate core cooling

Proposed Question: #16

Which one of the following sets of post-LOCA conditions ensures adequate core cooling?

	Reactor Pressure	Reactor Water Level	Pressure Control	Injection Source
A.	50 psig	-115"	ECs in-service	Only CRD and HPCI
B.	70 psig	Unknown	3 ERVs open	Only CRD
C.	150 psig	-118"	No ERVs open	None available
D.	175 psig	-120"	2 ERVs open	Only CRD

Proposed Answer: C

Explanation: Adequate core cooling is assured through steam cooling per N1-EOP-9 while RPV level is maintained above -121 inches with no injection. The Minimum Zero-Injection RPV Water Level (-121") is the lowest RPV water level at which the covered part of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1800°F. The level is used to preclude significant fuel damage and hydrogen generation for as long as possible.

A. Incorrect – With injection, RPV water level must be above -109" or adequate Core Spray flow must exist to have adequate core cooling.
B. Incorrect – With RPV water level unknown and no Core Spray flow, adequate core cooling is not assured.
D. Incorrect – With injection, RPV water level must be above -109" or adequate Core Spray flow must exist to have adequate core cooling.

Technical Reference(s): EOP Bases

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP2C01 EO #2

Question Source: Bank NMP1 2010 Audit #72

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

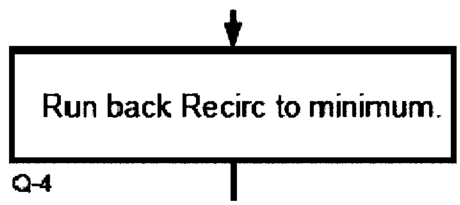
Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	295037 EK3.01
	Importance Rating	4.1

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: Recirculation pump trip/runback: Plant-Specific

Proposed Question: #17

N1-EOP-3, Failure to Scram, contains the following step:



Which one of the following describes the reason for this step?

A. Raise voiding in the core to lower Reactor power while attempting to avoid tripping the Main Turbine.

B. Raise voiding in the core to lower Reactor power while maintaining adequate coolant flow to avoid Thermal Hydraulic Instability (THI).

C. Lower power-to-flow scram setpoints to provide a redundant Reactor scram signal while attempting to avoid tripping the Main Turbine.

D. Lower power-to-flow scram setpoints to provide a redundant Reactor scram signal while maintaining adequate flow to avoid Thermal Hydraulic Instability (THI).

Proposed Answer: A

Explanation: If the reactor remains critical following initial attempts to insert control rods, power is reduced by decreasing recirculation flow. While power can be reduced most rapidly by tripping the recirculation pumps, tripping the pumps from a high power level may cause a turbine trip due to changes in steam flow, RPV pressure, and RPV water level. If the main turbine is on-line, recirculation flow is run back to minimum before the recirculation pumps are tripped to effect a more controlled power reduction. If the turbine-generator is not on-line, a turbine trip is of no concern and the runback need not be performed.

B. Incorrect – Recirculation flow is reduced despite the fact that it may place the plant in a high power-to-flow condition where thermal-hydraulic instabilities are possible.
C. Incorrect – Recirculation flow is reduced to lower Reactor power, not lower scram setpoints.
D. Incorrect – Recirculation flow is reduced to lower Reactor power, not lower scram setpoints. Recirculation flow is reduced despite the fact that it may place the plant in a high power-to-flow condition where thermal-hydraulic instabilities are possible.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP3C01 EO #2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

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

High Off-site Release Rate

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

Proposed Question: #18

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

	Condition Requiring Manual CRVS Isolation and CREVS Initiation	Associated Reason
A.	Unit 2 enters N2-EOP-RR, Radioactivity Release Control.	Minimize the station's total off-site radioactivity release rate.
B.	Unit 2 enters N2-EOP-RR, Radioactivity Release Control.	Ensure continued habitability for Control Room operators.
C.	Unit 1 enters N1-EOP-5, Secondary Containment Control.	Minimize the station's total off-site radioactivity release rate.
D.	Unit 1 enters N1-EOP-5, Secondary Containment Control.	Ensure continued habitability for Control Room operators.

Proposed Answer: B

Explanation: OP-49 requires manual isolation of the normal Control Room Ventilation System (CRVS) and manual initiation of the Control Room Emergency Ventilation System (CREVS) in the following conditions:

- Total iodine radioactivity concentration greater than 9.47 µCi/gm (T.S. 3.2.4)
- Unit 2 enters RR (Radioactivity Release) EOP

The reason for isolation of the normal Control Room Ventilation System (CRVS) and manual initiation of the Control Room Emergency Ventilation System in this situation is to maintain habitability of the Control Room to meet requirements of 10CFR Appendix A GDC 19.

A. Incorrect – While CREVS will filter out radioactivity in the air it handles, CREVS initiation is not designed to lower off-site release rate.
C. Incorrect – While some EOP-5 entry conditions could result in elevated radioactivity release and dose to Control Room personnel, EOP-5 entry does not directly require CRVS isolation and CREVS initiation. While CREVS will filter out radioactivity in the air it handles, CREVS initiation is not designed to lower off-site release rate.
D. Incorrect – While some EOP-5 entry conditions could result in elevated radioactivity release and dose to Control Room personnel, EOP-5 entry does not directly require CRVS isolation and CREVS initiation.

Technical Reference(s): OP-49, 1101-288003C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288003-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

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

Plant Fire On Site

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

Proposed Question: #19

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

Time (minutes)	Event
0	A fire alarm is received for Reactor Building 281 East. A Fire Brigade member in the area immediately confirms the presence of a fire. N1-SOP-21.1, Fire in Plant, is entered.
15	The Fire Brigade reports that the fire is still in progress and NOT under control.
20	80-118, CONTAINMENT SPRAY TEST TO TORUS F.C.V., spuriously opens.
25	Attempts to close 80-118 from the Control Room are unsuccessful.
30	The Shift Manager determines that the fire endangers Safe Shutdown capability.

Which one of the following describes the earliest time a manual Reactor scram is required, in accordance with N1-SOP-21.1?

A. 15 minutes

B. 20 minutes

C. 25 minutes

D. 30 minutes

Proposed Answer: A

Explanation: SOP-21.1 requires a manual Reactor scram if any of the following conditions exist due to a fire in an area included in table 21.1-1:

- Spurious valve operation
- Loss of equipment control
- Fire NOT under control within 15 minutes
- Fire endangers Safe Shutdown capability

At time 15 minutes, the fire is NOT under control and has been in progress for 15 minutes, therefore a manual Reactor scram is required.

B. Incorrect – At time 20 minutes, there is indication of spurious valve control, which would require a manual Reactor scram. However, a scram was already required at time 15 minutes.
C. Incorrect – At time 25 minutes, there is indication of loss of equipment control, which would require a manual Reactor scram. However, a scram was already required at time 15 minutes.
D. Incorrect – At time 30 minutes, the fire is determined to be endangering Safe Shutdown capability, which would require a manual Reactor scram. However, a scram was already required at time 15 minutes.

Technical Reference(s): SOP-21.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP21.1C01 EO #2

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	1
	K/A #	700000 2.4.45
	Importance Rating	4.1

Generator Voltage and Electric Grid Disturbances

Ability to prioritize and interpret the significance of each annunciator or alarm

Proposed Question: #20

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

- A6-3-3, 345 KV SYS FREQUENCY HIGH-LOW
- A2-3-2, GENERATOR CORE MONITOR

An operator observes the following Control Room indications:

- Generator Frequency is 58 Hz and slowly lowering.
- Generator MVAR indication is 125 MVARS TO BUS and slowly rising.

Which one of the following describes the required operator action?

A. Reduce Generator load as required to maintain stator temperatures < 125°C.

B. Lower Generator load to the self-cooled rating of 9000 bus amperes continuous.

C. Insert a manual Reactor Scram and verify the Main Turbine trips per N1-SOP-31.1.

D. Initiate a Generator load reduction to < 45% load while attempting to clear the core monitor alarm.

Proposed Answer: C

Explanation: Annunciator A6-3-3, validated by the low frequency, requires entry into SOP-33B.1. Since grid frequency is greater than 1.9 Hz from normal, SOP-33B.1 requires a Turbine trip per SOP-31.1. Since Reactor power is above 45%, SOP-31.1 directs a manual Reactor scram, which will in turn cause a Turbine trip.

A. Incorrect – A Reactor scram is required. This load reduction is based on actions for loss of bus duct cooling, not Generator overheating.
B. Incorrect – A Reactor scram is required. This load reduction is based on actions for loss of bus duct cooling, not Generator overheating.
D. Incorrect – A Reactor scram is required. This action is based on ARP A2-3-2, but the need for an immediate Reactor scram and Turbine trip takes priority.

Technical Reference(s): ARP A6-3-3, ARP A2-3-2, SOP-33B.1, SOP-31.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP-33B.1C01 EO #2

Question Source: Bank # NMP1 2008 NRC #52

Question History: NMP1 2008 NRC #52

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295010 2.1.31
	Importance Rating	4.6

High Drywell Pressure

Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup

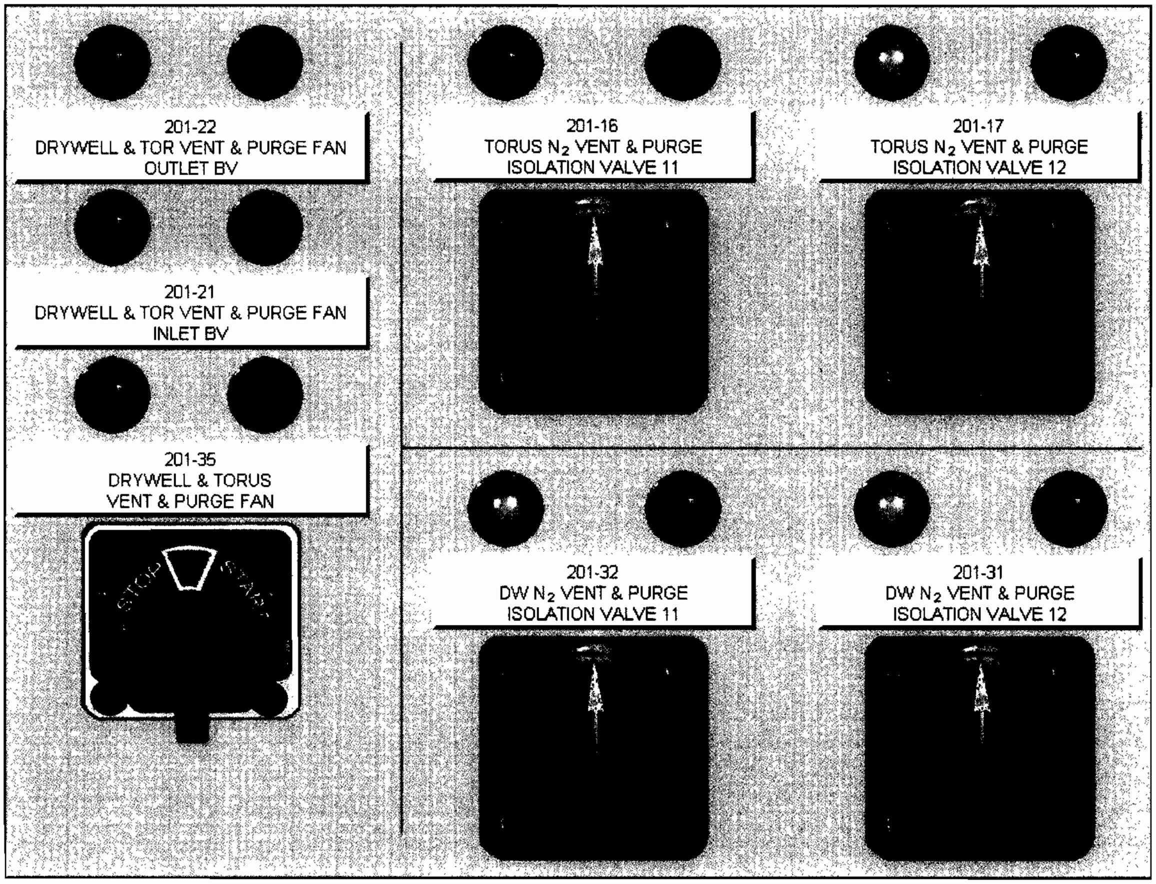
Proposed Question: #21

The plant is operating at 85% power with the following:

- A shutdown is in progress for a planned refueling outage.
- Torus venting is in progress.

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

- The mode switch is placed in SHUTDOWN.
- Drywell pressure is 3.7 psig and slowly rising.
- Reactor water level is 72" and stable after reaching a low of 30" following the scram.
- The following indications are present:



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

A. These indications are correct for the given plant conditions.

B. 201-16, TORUS N2 VENT & PURGE ISOLATION VALVE 11, should be closed.

C. 201-17, TORUS N2 VENT & PURGE ISOLATION VALVE 12, should be open.

D. 201-35, DRYWELL & TORUS VENT & PURGE FAN, should be tripped with inlet and outlet dampers closed.

Proposed Answer: B

Explanation: With Torus venting in progress, 201-16 and 201-17 would initially be open or throttled open, with 201-35 running and its inlet and outlet dampers open. However, when Drywell pressure exceeds 3.5 psig, a Containment isolation signal is received. Both 201-16 and 201-17 should close on this signal. 201-35 and its associated dampers are unaffected by this signal. The picture shows 201-16 still open, while it should be closed.

A. Incorrect – 201-16 should be closed.
C. Incorrect – 201-17 is in the correct position for a Containment isolation.
D. Incorrect – 201-35 and its associated dampers are unaffected by a Containment isolation signal and should still be running from the original Torus venting lineup.

Technical Reference(s): SOP-40.2, OP-9, C-19432-C sheet 2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223003-RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295012 AK1.01
	Importance Rating	3.3

High Drywell Temperature

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Pressure/temperature relationship

Proposed Question: #22

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

- Five Drywell cooling fans are running.
- Drywell temperature is 115°F and stable.
- Drywell pressure is 1.5 psig and stable.
- Drywell relative humidity is 25% and stable.

Then, one Drywell cooling fan trips.

Which one of the following describes the expected response of Drywell pressure and relative humidity?

	Drywell Pressure	Drywell Relative Humidity
A.	Remains the same	Rises
B.	Remains the same	Lowers
C.	Rises	Rises
D.	Rises	Lowers

Proposed Answer: D

Explanation: Drywell temperature will rise due to the loss of one Drywell cooling fan (~5^{o}F). While one Drywell cooling fan was originally out of service, there is no auto-start feature on trip of another fan. At 100% power, the Drywell is closed. Since the Drywell is a fixed volume of gas, the rise in temperature causes a rise in pressure. No design feature prevents or corrects for this pressure change. Since temperature rises with the same overall moisture content in the Drywell, the relative humidity lowers.

A. Incorrect – Drywell pressure rises. Drywell relative humidity lowers.
B. Incorrect – Drywell pressure rises.
C. Incorrect – Drywell relative humidity lowers.

Technical Reference(s): OP-8, 1101-223001C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223001-RBO-08

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(5)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295013 AK3.02
	Importance Rating	3.6

High Suppression Pool Temperature

Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Limiting heat additions

Proposed Question: #23

The plant is operating at 100% power when a transient results in the following:

- The mode switch is placed in SHUTDOWN.
- Multiple control rods fail to insert.
- Reactor power is 55%.
- ERVs are cycling.
- Emergency Condensers have been placed in service.

Which one of the following describes how to operate the ERVs, in accordance with N1-EOP-3, Failure to Scram, and the associated reason?

Manually open ERVs until RPV pressure drops to __(1)__. This target pressure is based on ensuring __(2)__.

	(1)	(2)
A.	800 psig	Turbine Bypass Valve use is maximized in order to minimize heat addition to the Torus
B.	800 psig	the Reactor scram can be reset in order to attempt repeated manual scrams
C.	965 psig	Turbine Bypass Valve use is maximized in order to minimize heat addition to the Torus
D.	965 psig	the Reactor scram can be reset in order to attempt repeated manual scrams

Proposed Answer: C

Explanation: EOP-3 steps P-1 and P-2 required ERVs to be manually open until RPV pressure drop to 965 psig if ERVs are cycling. This pressure is based on not going so low that Turbine Bypass Valves begin closing, such that as much heat can be rejected to the Main Condenser as possible. This limits the heat addition to the Torus, which is a priority in failure to scram conditions.

A. Incorrect – 965 psig is the correct pressure.
B. Incorrect – 965 psig is the correct pressure. The basis for 965 psig is based on Turbine Bypass Valve opening/closing pressure.
D. Incorrect – The basis for 965 psig is based on Turbine Bypass Valve opening/closing pressure.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP3C01 EO #2

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295014 AK3.02
	Importance Rating	3.7

Inadvertent Reactivity Addition

Knowledge of the reasons for the following responses as they apply to INADVERTENT REACTIVITY ADDITION: Control rod blocks

Proposed Question: #24

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

- Turbine Stop Valve 14 spuriously closes.
- Reactor pressure rises to 1040 psig.
- Reactor power rises to 117%.
- Reactor water level lowers to 65".
- Annunciator F3-4-4, ROD BLOCK, alarms.

Which one of the following describes the reason for the control rod block?

A. Reactor power signal

B. Reactor pressure signal

C. Reactor water level signal

D. Turbine Stop Valve position signal

Proposed Answer: A

Explanation: Reactor power causes both control rod blocks and scrams. At rated flow conditions, APRMs cause a control rod block at a maximum of 117% and a Reactor scram at a maximum of 122%. TSV position, Reactor pressure, and Reactor water level can all cause Reactor scrams, but not control rod blocks. The given values of Reactor pressure and water level cause alarms, but are not severe enough to cause a Reactor scram.

B. Incorrect – Reactor pressure of 1040 psig will cause Annunciator F2-3-4 to alarm, but not a control rod block. Reactor pressure of 1080 psig will cause a Reactor scram.
C. Incorrect – Reactor water level of 65" will cause Annunciator F2-3-3 to alarm, but not a control rod block. Reactor water level of 53" will cause a Reactor scram.
D. Incorrect – TSV position indication can cause a Reactor scram, but not a control rod block.

Technical Reference(s): Tech Spec 3.6.2g, LSSS 2.1.2.a, ARP F1-2-5, ARP F2-1-6, ARP F2-3-3, ARP F2-3-4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295020 AK1.02
	Importance Rating	3.5

Inadvertent Containment Isolation

Knowledge of the operational implications of the following concepts as they apply to INADVERTENT CONTAINMENT ISOLATION: Power/reactivity control

Proposed Question: #25

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

- I&C is troubleshooting Main Steam Isolation Valve (MSIV) position indication problem.
- MSIV 111 receives an inadvertent isolation signal and closes.

Which one of the following describes the required operator response to this MSIV closure?

A. Commence a plant shutdown within one hour per N1-OP-43C Section H.1.0, 10 Hour Shutdown and Cooldown.

B. Perform an emergency power reduction to approximately 50% power per N1-SOP-1.1, Emergency Power Reduction.

C. Perform an emergency power reduction to restore power to 100% or below per N1-SOP-1.1, Emergency Power Reduction.

D. Respond to an automatic Reactor scram per N1-SOP-1, Reactor Scram.

Proposed Answer: D

Explanation: Closure of MSIV 111 isolates one of the two main steam lines and causes a half scram. Since Reactor power was originally at 100%, the rise in steam flow through the other main steam line will cause a high flow condition and subsequent isolation of the second main steam line. This will cause a full Reactor scram.

A. Incorrect – A Reactor scram will occur immediately.
B. Incorrect – Although 50% power operation may be possible with only one main steam line open, a Reactor scram will occur due to the initial power level.
C. Incorrect – Although Reactor power will initially rise due to rising pressure and void collapse, isolation of the second main steam line on high flow will cause an automatic Reactor scram.

Technical Reference(s): 1101-239001C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	295033 EA1.02
	Importance Rating	3.7

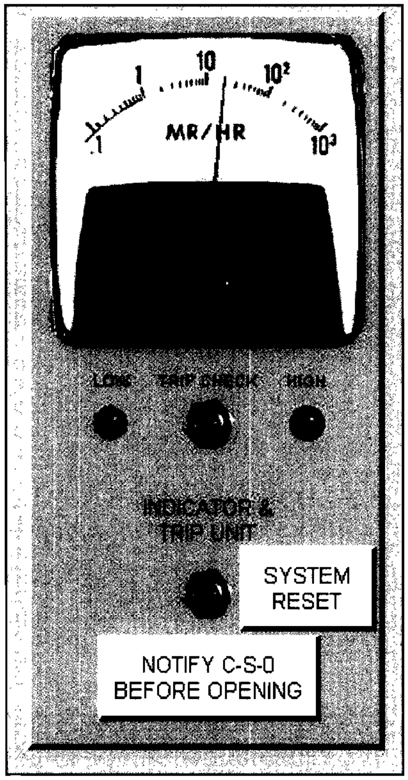
High Secondary Containment Area Radiation Levels

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

Proposed Question: #26

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

- A leak has developed from an Emergency Cooling steam line into the Reactor Building.
- Reactor Building ventilation exhaust radiation monitors both indicate as shown below:



Which one of the following describes the approximate value of Reactor Building ventilation exhaust radiation and the expected status of the Reactor Building Emergency Ventilation System?

	Radiation Value	Status of RBEVS
A.	11 mR/hr	Running
B.	11 mR/hr	In standby
C.	20 mR/hr	Running
D.	20 mR/hr	In standby

Proposed Answer: C

Explanation: The given Reactor Building ventilation exhaust radiation monitor is reading approximately 20 mR/hr. When Reactor Building ventilation exhaust radiation monitors reach 5 mR/hr, Reactor Building Ventilation isolates and RBEVS initiates.

A. Incorrect – The rad monitor reads approximately 20 mR/hr.
B. Incorrect – The rad monitor reads approximately 20mR/hr. Above 5 mR/hr, RBEVS is running due to an automatic initiation signal.
D. Incorrect –Above 5 mR/hr, RBEVS is running due to an automatic initiation signal.

Technical Reference(s): ARP L1-4-3, OP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(11)

Examination Outline Cross-Reference:	Level	RO
	Tier #	1
	Group #	2
	K/A #	500000 EA2.01
	Importance Rating	3.1

High Containment Hydrogen Concentration

Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Hydrogen monitoring system availability

Proposed Question: #27

A loss of coolant accident has resulted in the following:

- Reactor water level is -60 inches and rising slowly.
- Drywell pressure is 12 psig and rising slowly.

Which one of the following describes the availability of the H_2O_2 monitors and what operator actions are required?

The H_2O_2 monitors are...

A. available and lined up. Verify proper operation at H_2O_2 Monitor Control Panel 11 (TB 291).

B. isolated. Bypass the isolation using CAD CHANNEL 11 and 12 RPS BY-PASS switches.

C. isolated. Bypass the isolation using CAD CHANNEL 11 and 12 RPS BY-PASS switches AND AUTO VESSEL ISOL CH 11 and CH 12 switches.

D. NOT available. Direct Chemistry to manually sample the Containment using N1-ECP-209 OR N1-ECP-210.

Proposed Answer: B

Explanation: The H_2O_2 monitors isolate with Drywell pressure above 3.5 psig. Per N1-OP-9, Sect. H.3, and EOP-4, to recover the H2 and O2 sampling systems it is necessary to un-isolate the system by bypassing the Containment Isolation using CAD CHANNEL 11 and 12 RPS BY-PASS switches.

A. Incorrect – The H_2O_2 monitors isolate with Drywell pressure above 3.5 psig.
C. Incorrect – AUTO VESSEL ISOL CH 11 and Ch 12 switches do not need to be manipulated to restore H_2O_2 monitors.
D. Incorrect – EOP-4 and OP-9 provide direction to restore H_2O_2 monitors.

Technical Reference(s): EOP-4, OP-9, SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223001-RBO-05

Question Source: Bank NMP1 2008 NRC #65

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 A1.01
	Importance Rating	3.3

Shutdown Cooling

Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) controls including: Heat exchanger cooling flow

Proposed Question: #28

A plant cooldown is in progress with the following:

- Shutdown Cooling (SDC) loop 11 is in service.
- 70-49, BV - 11 SDC HX RBCLC INLET, is throttled open 4 turns.
- 38-09, SD COOLING TCV 11, is open 50%.
- SDC loops 12 and 13 are in standby.
- The CRS has directed a cooldown rate of 75°F/hr.
- The following Reactor coolant temperature data is observed:

Time	Reactor Coolant Temperature (°F)
0800	330
0815	310
0830	290

Which one of the following describes the appropriate Operator action to achieve the desired cooldown rate, in accordance with N1-OP-4, Shutdown Cooling System?

A. Lower flow through the SDC heat exchanger by further closing either 70-49 or 38-09.

B. Raise flow through the SDC heat exchanger by further opening either 70-49 or 38-09.

C. Lower flow through the SDC heat exchanger by further closing 38-09. Further closing of 70-49 is NOT allowed.

D. Raise flow through the SDC heat exchanger by further opening 38-09. Further opening of 70-49 is NOT allowed.

Proposed Answer: A

Explanation: The given Reactor coolant temperature data results in a calculated cooldown rate of 80°F/hr (40°F/0.5hr), which is above the desired 75°F/hr rate. In order to lower the cooldown rate, SDC heat exchanger flow rate must be lowered, either through the SDC or RBCLC sides of the heat exchanger. OP-4 section F.1 allows throttling of either the RBCLC inlet valve (70-49) or the SDC TCV (38-09) to control cooldown rate.

B. Incorrect – SDC heat exchanger flow must be lowered to lower the cooldown rate.
C. Incorrect – 70-49 may be closed further to control cooldown rate.
D. Incorrect – SDC heat exchanger flow must be lowered to lower the cooldown rate.

Technical Reference(s): OP-4, C-18018-C sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(3)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	205000 K4.01
	Importance Rating	3.4

Shutdown Cooling

Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: High temperature isolation: Plant-Specific

Proposed Question: #29

The plant is shutdown with the following:

- Shutdown Cooling (SDC) loops 11 and 12 are in service.
- Reactor coolant temperature is 250°F and lowering slowly.

Then, a significant leak from SDC pump 11 results in the following:

- Annunciator K3-3-2, SD COOLING SYSTEM STEAM LEAK AREA T HI, alarms.
- Computer point D006, SHTDN AREA TEMP DET 1, alarms HIGH.

Which one of the following describes the response of the SDC system?

A. SDC loops 11 and 12 remain in service until a second area temperature detector alarms HIGH.

B. SDC loop 11 isolates and SDC loop 12 remains in service.

C. SDC pump 11 trips, but NO isolations occur.

D. The entire SDC system isolates.

Proposed Answer: D

Explanation: Annunciator K3-3-2 alarms when any one of four SDC area temperature detectors exceeds 165°F. These same four temperature detectors actuate logic to isolate all of SDC (isolation valve 38-01, 38-02, and 38-13). This logic is 1 out of 4 taken once. Therefore, the single detector that is in alarm would cause an isolation of the entire SDC system. The SDC pumps are in a common room, thus there is no ability for the logic to distinguish which pump is causing the high temperature condition.

A. Incorrect – The SDC high area temperature logic only requires one temperature detector to exceed its setpoint to cause an isolation.
B. Incorrect – The isolation closes valves 38-01, 38-02, and 38-13, which isolate the entire SDC system.
C. Incorrect – SDC isolates. SDC pump 11 would trip and not isolate on high suction temperature of 350°F.

Technical Reference(s): C-18018-C sheet 1, C-19859-C sheet 12, C-22017-C sheet 3, OP-4, ARP K3-3-2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-205000-RBO-05

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	206000 2.4.46
	Importance Rating	4.2

HPCI

Ability to verify that the alarms are consistent with the plant conditions

Proposed Question: #30

A plant startup is in progress with the following:

- Reactor power is 18%.
- Turbine startup is in progress.

Then, a Turbine overspeed signal results in the following:

- Annunciator F1-2-5, RPS CH 11 TURBINE TRIP, alarms.
- Annunciator F4-2-4, RPS CH 12 TURBINE TRIP, alarms.
- Reactor water level is 72" and stable.
- Reactor pressure is 918 psig and stable.

Which of the following annunciator(s) would also be in alarm?

(1) F1-2-1, RPS CH 11 AUTO REACTOR TRIP
(2) F4-2-8, RPS CH 12 AUTO REACTOR TRIP
(3) F4-4-1, HPCI MODE AUTO INITIATE
(4) A1-4-5, TURBINE STOP VALVES CLOSED

A. (4) only

B. (1) and (2) only

C. (3) and (4) only

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

Proposed Answer: C

Explanation: Turbine overspeed is a mechanical automatic Turbine trip signal, as evidenced by annunciators F1-2-5 and F4-2-4 being in alarm. Below approximately 40-45% power, a Turbine trip does not cause an automatic Reactor scram. Therefore, annunciators F1-2-1 and F4-2-8 would not be in alarm. The Turbine trip signal does cause Turbine stop valves to close (Annunciator A1-4-5). The subsequent generator trip will cause an electrical turbine trip signal and cause HPCI to auto-initiate (Annunciator F4-4-1) (3T-1, 3T-2). These functions are unaffected by the low power level.

A. Incorrect – F4-4-1 would also be in alarm.
B. Incorrect – F1-2-1 and F4-2-8 would not be in alarm. F4-4-1 and A1-4-5 would be in alarm.
D. Incorrect – F1-2-1 and F4-2-8 would not be in alarm.

Technical Reference(s): ARP F4-4-1, ARP A1-4-5, OP-40, OP-16

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	207000 A1.05
	Importance Rating	4.0

Isolation (Emergency) Condenser

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

Proposed Question: #31

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

- Main steam line isolation occurs due to steam leak detection.
- Reactor power stabilizes at 17%.

Which one of the following describes the appropriate operator action(s) to stabilize and control Reactor pressure for these conditions?

A. Control steam flow through the Turbine Bypass Valves.

B. Cycle only one Emergency Condenser loop as needed.

C. Place one Emergency Condenser loop in service and cycle the other.

D. Place both Emergency Condensers loops in service and cycle ERVs.

Proposed Answer: D

Explanation: 17% Reactor power with MSIV isolation indicates an ATWS and EOP-3 entry. With a main steam isolation due to steam leak detection, the Turbine bypass valves are not available. Startup testing found that the ECs had a heat removal capability of up to ~8.0x10^8 BTU/hr, which equates to ~235MW or 12% of rated power, but were only designed for about halfof that capability, or 6% power. With 17% power in this question, the ECs alone will not be enough to control pressure. Therefore ERVs will be cycled to control pressure, in addition to use of ECs. 1.9EE8 = ~55.7MW = ~3% per EC

A. Incorrect – Turbine Bypass Valves are unavailable due to main steam line isolation with steam leak detection.
B. Incorrect – One EC loop is only designed to control Reactor pressure associated with approximately 3% Reactor power.
C. Incorrect – Just two EC loops are only designed to control Reactor pressure associated with approximately 6% Reactor power.

Technical Reference(s): EOP-3, OP-13, 1101-207000C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-02

Question Source: Bank NMP1 2008 NRC #48

Question History: NMP1 2008 NRC #48

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	209001 K2.01
	Importance Rating	3.0

LPCS

Knowledge of electrical power supplies to the following: Pump power

Proposed Question: #32

The plant is operating at 50% power when the following occur:

Time (mm:ss)	Event
00:00	Lines 1 and 4 de-energize.
00:10	Powerboard 102 fails to re-energize due to an electrical fault.
04:00	A coolant leak develops in the Drywell.
06:00	The mode switch is placed in SHUTDOWN.
10:00	Annunciators F1-1-5, RPS CH 11 DRYWELL PRESS HIGH, and F4-1-4, RPS CH 12 DRYWELL PRESS HIGH, alarm

Which one of the following describes the response of the Core Spray pumps?

Core Spray pump ___(1)___ starts at time 10:00. Core Spray pump 122 starts at time ___(2)___.

	(1)	(2)
A.	112	10:07
B.	112	10:13
C.	121	10:07
D.	121	10:13

Proposed Answer: B

Explanation: With Powerboard 102 de-energized due to a fault, Core Spray pumps 111 and 121 are unavailable. Core Spray pumps 112 and 122 are still available from Powerboard 103. These pumps start sequentially upon receipt of an automatic start signal (in this case, Drywell pressure of 3.5 psig). Core Spray pump 112 starts with no time delay. Core Spray Topping pump 112 starts after a 7 second time delay. Core Spray pump 122 starts after a 13 second time delay. Core Spray Topping pump 122 starts after a 20 second time delay.

A. Incorrect – Core Spray pump 122 does not start until 10:13.
C. Incorrect – Core Spray pump 121 will not start due to loss of power to Powerboard 102. Core Spray pump 122 does not start until 10:13.
D. Incorrect – Core Spray pump 121 will not start due to loss of power to Powerboard 102.

Technical Reference(s): OP-2, 1101-209001C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-209001-RBO-04 and N1-209001-RBO-05

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	211000 A2.02
	Importance Rating	3.6

SLC

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: Failure of explosive valve to fire

Proposed Question: #33

A failure to scram has occurred with the following:

- RPS Bus 11 is de-energized due to an electrical fault.
- The Liquid Poison system is started by placing the key lock selector switch to SYS 12.
- Liquid Poison explosive valve 12 fails to open due to defective squibs.

Which one of the following describes the impact of these conditions on boron injection and the required operator action?

A. Reactor Water Cleanup fails to isolate and will filter out injected boron. Isolate Reactor Water Cleanup per N1-OP-3.

B. Boron flow to the Reactor is blocked. Place the key lock selector switch to SYS 11 to fire explosive valve 11 per EOP-HC.

C. Boron flow to the Reactor is blocked. Line-up boron injection using the hydro pump per N1-EOP-3.2.

D. Boron flow to the Reactor is blocked. Line-up boron injection using Reactor Water Cleanup per N1-EOP-3.2.

Proposed Answer: C

Explanation: RPS Bus 11 is the power supply to RWCU isolation logic. However, loss of RPS Bus 11 causes RWCU to automatically isolate, not fail to isolate. Loss of RPS Bus 11 also causes loss of firing power to Liquid Poison explosive valve 11. Since Liquid Poison explosive valve 12 also failed to open, Liquid Poison flow to the Reactor is blocked, regardless of which pump is running. RWCU cannot be used for boron injection per EOP-3.2 due to loss of isolation logic power, regardless of EOP jumper installation. Injection of boron can only be accomplished using the hydro pump per EOP-3.2. The hydro pump injects boron into the Reactor from a connection downstream of the explosive valves.

A. Incorrect – Reactor Water Cleanup is already isolated due to the loss of RPS Bus 11.
B. Incorrect – With no power to explosive valve 11 and failure of explosive valve 12, starting the alternate Liquid Poison pump will have no effect.
D. Incorrect – Reactor Water Cleanup isolates due to the loss of RPS Bus 11 and will not be available for restoration.

Technical Reference(s): C-18019-C, C-19859-C sheet 12, EOP-3.2, F-45114-C and F-45115-C (to show squib power)

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-211000-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

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

RPS

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

Proposed Question: #34

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

- N1-ST-W15, MANUAL SCRAM INSTRUMENT CHANNEL TEST, is in progress.
- An operator depresses the REACTOR TRIP 11 pushbutton.
- Simultaneously, Reactor water level transmitter 36-03A fails downscale.
- This transmitter inputs to the Reactor Protection System (RPS) scram circuitry.

Which one of the following describes the response of the Reactor Protection System (RPS) and HPCI System?

	RPS	HPCI...
A.	A half scram occurs only.	does NOT initiate.
B.	A half scram occurs only.	initiates.
C.	A full scram occurs.	does NOT initiate.
D.	A full scram occurs.	initiates.

Proposed Answer: A

Explanation: Depressing the REACTOR TRIP 11 pushbutton causes a half scram on RPS channel 11. Reactor water level transmitter 36-03A failing downscale also causes a half scram on RPS channel 11. Neither have any effect on RPS channel 12, therefore a full scram does not occur. HPCI logic is one out of two taken twice, therefore no initiation occurs due to only a single failed level transmitter.

B. Incorrect – HPCI logic is one out of two taken twice, therefore no initiation occurs due to only a single failed level transmitter.
C. Incorrect – RPS scram channel 12 is unaffected by these conditions, therefore a full scram does not occur.
D. Incorrect – RPS scram channel 12 is unaffected by these conditions, therefore a full scram does not occur. HPCI logic is one out of two taken twice, therefore no initiation occurs due to only a single failed level transmitter.

Technical Reference(s): , C-19859-C sheets 2, 3 & 4, C-23077-C sheets 5 (for proof of HPCI logic)

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-212000-RBO-05

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215003 A3.02
	Importance Rating	3.3

IRM

Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including: Annunciator and alarm signals

Proposed Question: #35

The plant is shutdown with the following:

- The mode switch is in REFUEL.
- Preparations are underway for Shutdown Margin Testing.
- The REFUEL INST TRIP BYPASS 11 and 12 switches are in the NON-COINCIDENT position.
- All Intermediate Range Monitors (IRMs) are on range 2.

Then, Annunciator F1-1-1, RPS CH 11 REACT NEUTRON MONITOR, alarms.

Which one of the following describes the IRMs that input to this annunciator and the status of the Reactor Protection System (RPS)?

	IRM Inputs to F1-1-1	Status of RPS
A.	11, 12, 13, 14	A half scram has occurred, only.
B.	11, 12, 13, 14	A full scram has occurred.
C.	11, 13, 15, 17	A half scram has occurred, only.
D.	11, 13, 15, 17	A full scram has occurred.

Proposed Answer: B

Explanation: IRMs 11, 12, 13, and 14 input into RPS channel 11, and thus Annunciator F1-1-1. With the REFUEL INST TRIP BYPASS 11 and 12 switches in the NON-COINCIDENT position, only RPS channel 11 or 12 needs to experience a neutron monitoring scram signal to cause a full scram.

A. Incorrect – With the REFUEL INST TRIP BYPASS 11 and 12 switches in the NON-COINCIDENT position, only RPS channel 11 or 12 needs to experience a neutron monitoring scram signal to cause a full scram.
C. Incorrect – IRMs 15 and 17 input into RPS channel 12, and thus Annunciator F4-1-8, not F1-1-1. With the REFUEL INST TRIP BYPASS 11 and 12 switches in the NON-COINCIDENT position, only RPS channel 11 or 12 needs to experience a neutron monitoring scram signal to cause a full scram.
D. Incorrect – IRMs 15 and 17 input into RPS channel 12, and thus Annunciator F4-1-8, not F1-1-1.

Technical Reference(s): ARP F1-1-1, ARP F4-1-8, OP-38B

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-05

Question Source: Bank – 2008 Audit #3

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 K1.02
	Importance Rating	3.4

SRM

Knowledge of the physical connections and/or cause-effect relationships between SOURCE RANGE MONITOR (SRM) SYSTEM and the following: Reactor manual control

Proposed Question: #36

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: SRM downscale rod blocks are bypassed when the SRM is fully inserted to allow startup with less than 100 cps. All SRM rod blocks are bypassed once the associated IRM are on Range 8 or above.

A. Incorrect – Just withdrawing an SRM does not cause bypass of rod blocks.
C. Incorrect – SRM downscale rod block is not bypassed by IRMs being on Range 2. IRM Range 2 bypasses the associated IRM downscale rod block. Just withdrawing an SRM does not cause bypass of rod blocks.
D. Incorrect – SRM downscale rod block is not bypassed by IRMs being on Range 2. IRM Range 2 bypasses the associated IRM downscale rod block.

Technical Reference(s): OP-43A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215004 A3.01
	Importance Rating	3.2

SRM

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

Proposed Question: #37

A plant startup is in progress with the following:

- The Reactor is critical.
- SRMs are partially withdrawn.
- SRMs are indicating $5x10^3$ cps.
- IRMs are mid-scale on Range 4.

Then, SRM 11 DETECTOR FULL IN pushbutton is depressed on E panel.

Which one of the following describes the expected movement of SRM 11 meter indication and when a rod block will first be received?

SRM 11 meter indication ___(1)___. A rod block will first be received when SRM counts reach ___(2)___.

	(1)	(2)
A.	lowers	3 cps
B.	lowers	100 cps
C.	rises	$1x10^5$ cps
D.	rises	$5x10^5$ cps

Proposed Answer: C

Explanation: Depressing the DETECTOR FULL IN pushbutton inserts the partially withdrawn SRM further into the core. Inserting an SRM further into the core exposes it to more neutron flux, which causes meter indication to rise. The SRM upscale rod block is received at $1x10^5$ cps.

A. Incorrect – SRM meter indication rises as the detector goes further into the core.
B. Incorrect – SRM meter indication rises as the detector goes further into the core.
D. Incorrect – The SRM upscale rod block is received at $1x10^5$ cps.

Technical Reference(s): OP-38A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-215000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	215005 K3.01
	Importance Rating	4.0

APRM / LPRM

Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: RPS

Proposed Question: #38

The plant is operating at 65% power with the following:

- Rod line is 100%.
- All Reactor Recirculation pumps are controlled by the master controller.
- APRM FLOW UNIT 12 signal drifts down to 30% and then remains steady.
- Actual core flow remains constant.

Which one of the following describes the plant response?

A. No rod block or half scram is received.

B. A rod block is received, but no half scram occurs.

C. A rod block and a half scram are received. A full scram does NOT occur.

D. A rod block and full scram are received.

Proposed Answer: C

Explanation: The power/flow operating map for 5 loop operation shows the combination of 65% power and 30% sensed flow to be above the rod block and scram lines. APRM 12 flow unit provides the flow input to ARPM 15-18, so only these APRMs will generate a trip signal (rod block and half scram) in RPS channel 12. A flow comparator rod block also occurs because the difference between flow channels 11 and 12 exceeds 6.8%.

Technical Reference(s): 5 Loop Power-Flow Map

Proposed references to be provided to applicants during examination: 3, 4, and 5 Loop Power-Flow Maps

Learning Objective: N1-215000-RBO-11

Question Source: Bank – 2009 Audit #6

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 #	218000 K5.01
	Importance Rating	3.8

ADS

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

Proposed Question: #39

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

- I&C reports that both RPV Lo-Lo-Lo Water Level Relays from RPS Channel 11 (11K21 & 11K22) will not function on a valid signal.
- No other ADS circuitry problems exist.

Then, a transient occurs and the conditions required to initiate ADS are met. The ADS signals are valid and sustained.

Which one of the following describes the operation of the ADS system?

A. The primary and secondary valves will remain closed; the 111 second and 115.5 second timers will never start timing.

B. The primary and secondary valves will remain closed after both the 111 second and 115.5 second timers have timed out.

C. The secondary valves will be opened when the initiation signal has been present for 115.5 seconds; the primary valves remain closed.

D. The primary valves will be opened when the initiation signal has been present for 111 seconds; the secondary valves remain closed.

Proposed Answer: A

Explanation: Both one (out of two) Hi DW Press and one (out of two) Lo Lo Lo level trip signals from both the Ch 11 and Ch 12 logic are required to start the ADS timers. Since Ch 11 will not process a Lo Lo Lo signal, the logic will never be met to start the timers or open the valves.

B. Incorrect – The timers are started by energizing the 2-1A and 2-1B relays. Since the 2-1A and 2-1B relays are never energized (because 11K21 & 11K22 never close) the timers are never started.
C. Incorrect – No valves will open because the logic is never initiated.
D. Incorrect – No valves will open because the logic is never initiated.

Technical Reference(s): C-19859-C Sheets 2, 18 and 18A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-5

Question Source: Bank 2008 Audit #1

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

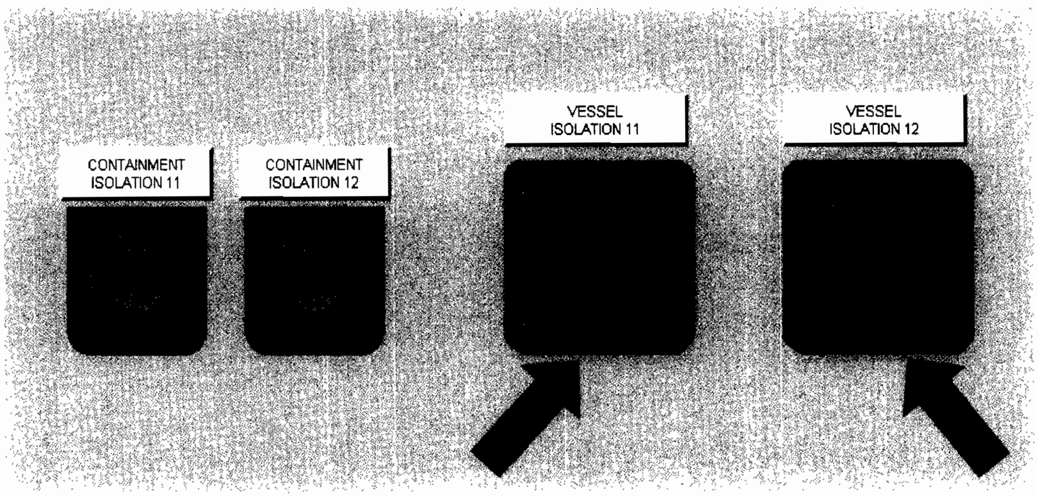
Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 K4.03
	Importance Rating	3.5

PCIS/Nuclear Steam Supply Shutoff

Knowledge of PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF design feature(s) and/or interlocks which provide for the following: Manual initiation capability: Plant-Specific

Proposed Question: #40

The switches indicated by red arrows in the picture below are taken simultaneously to the ISOLATION position.



Given the following sets of valves:

(1) Main Steam Isolation Valves
(2) Reactor Water Cleanup Isolation Valves
(3) H_2O_2 Monitor Isolation Valves
(4) TIP Ball Valves

Which one of the following identifies the set(s) of valves listed above that go closed?

A. (1) only

B. (1) and (2) only

C. (3) and (4) only

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

Proposed Answer: B

Explanation: The indicated switches are the manual vessel isolation switches. Taking both switches simultaneously to ISOLATION causes a vessel isolation, but not a containment isolation. MSIVs and RWCU isolation valves close on a vessel isolation. H2O2 Monitor Isolation Valves and TIP Ball Valves close on a containment isolation, but not a vessel isolation.

A. Incorrect – RWCU isolation valves also closed.
C. Incorrect – MSIVs and RWCU IVs close, H2O2 IVs and TIP Ball valves do not close.
D. Incorrect – H2O2 IVs and TIP Ball valves do not close.

Technical Reference(s): SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	223002 K1.14
	Importance Rating	2.8

PCIS/Nuclear Steam Supply Shutoff

Knowledge of the physical connections and/or cause-effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: Containment drainage system

Proposed Question: #41

Which one of the following conditions will cause the Drywell Floor Drain Isolation Valves to close?

A. Reactor water level at 0"

B. Drywell pressure at 2.7 psig

C. Drywell temperature at 160°F

D. Drywell floor drain leakage at 20 gpm

Proposed Answer: A

Explanation: PCIS will close the Drywell Floor Drain Isolation Valves on either Reactor water level below 5" or Drywell pressure above 3.5 psig.

B. Incorrect – High Drywell pressure will cause Drywell Floor Drain IVs to close, but only above 3.5 psig.
C. Incorrect – High Drywell temperature above 150°F is an EOP entry condition, but not a PCIS signal.
D. Incorrect – Drywell Floor Drain IV closure is designed to prevent excess Drywell leakage from exiting Primary Containment, however high sensed Drywell leakage does not cause an isolation.

Technical Reference(s): SOP-40.2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-223002-RBO-05

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

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

SRVs

Knowledge of electrical power supplies to the following: SRV solenoids

Proposed Question: #42

Which one of the following identifies an electrical power supply to Electromatic Relief Valve (ERV) solenoids and the associated number of ERV solenoids powered from this source?

	Power Supply To ERV Solenoids	Number Of ERV Solenoids Powered From This Source
A.	Powerboard 102	3
B.	Battery Board 11	3
C.	Powerboard 102	6
D.	Battery Board 11	6

Proposed Answer: B

Explanation: There are six ERVs, each with a single solenoid. Three of the ERV solenoids are powered from Battery Board 11. Three of the ERV solenoids are powered from Battery Board 12.

A. Incorrect – Powerboard 102 supplies power to the ADS confirmatory logic, but not to the actual ERV solenoids.
C. Incorrect – Powerboard 102 supplies power to the ADS confirmatory logic, but not to the actual ERV solenoids.
D. Incorrect – Battery Board 11 only supplies solenoid power to three ERVs. Each ERV has only one solenoid. The other three solenoids are powered from Battery Board 12.

Technical Reference(s): C-19859-C Sheets 24, 24a, 25, 25a, 26, 26a

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-218000-RBO-4

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 #	1
	K/A #	259002 K5.01
	Importance Rating	3.1

Reactor Water Level Control

Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM: GEMAC/Foxboro/Bailey controller operation: Plant-Specific

Proposed Question: #43

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

- Reactor water level is 72" and stable.
- Feedwater pump 13 is operating in automatic on the master controller.
- Feedwater pump 13 individual controller is in BAL and nulled.

Then, Reactor power is lowered to 95% and plant conditions stabilize.

Which one of the following describes the effect of now placing Feedwater pump 13 individual controller in MAN without adjusting the manual setpoint knob?

Reactor water level will...

A. stay at the current value unless the Feedwater master controller is also placed in MAN.

B. stay at the current value due to automatic adjustment of the controller manual setpoint while in AUTO.

C. lower because the manual demand signal will be less than the current automatic demand signal.

D. rise because the manual demand signal will be greater than the current automatic demand signal.

Proposed Answer: D

Explanation: Since the Feedwater pump 13 individual controller (GEMAC) was nulled at 100% power, the manual demand signal is equivalent to the flow needed at 100% power. The flow needed at 95% power is less than this value. When the controller is placed in MAN at the lower power level, the controller will cause more flow to the Reactor, which will raise Reactor water level.

A. Incorrect – The Feedwater master controller is only a factor if the individual controller is in AUTO or BAL.
B. Incorrect – The controller does not have an auto tracking feature for the manual setpoint.
C. Incorrect – The manual demand signal will be equivalent to the flow at 100% power, which is more than the flow at 95% power. With more flow, Reactor water level will rise.

Technical Reference(s): OP-16, 1101-259001C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-259001-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 #	2
	Group #	1
	K/A #	261000 K3.02
	Importance Rating	3.6

SGTS

Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: Off-site release rate

Proposed Question: #44

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

- RBEVS fan 11 is tagged out for maintenance.
- An un-isolable steam leak develops from Reactor Water Cleanup into the Reactor Building.
- Reactor Building Ventilation exhaust radiation monitors are reading 30 mR/hr and slowly rising.
- Reactor Building differential pressure is -0.05 psig and stable.

Then, RBEVS fan 12 trips and CANNOT be restarted.

Which one of the following describes the consequences of the RBEVS fan 12 trip on Reactor Building differential pressure (D/P) and off-site release rate?

	Reactor Building D/P	Off-site Release Rate
A.	Maintained negative by normal Reactor Building Ventilation	Remains unchanged
B.	Maintained negative by normal Reactor Building Ventilation	Rises
C.	Becomes positive	Remains unchanged
D.	Becomes positive	Rises

Proposed Answer: D

Explanation: Since Reactor Building Ventilation exhaust radiation monitors are greater than 5 mR/hr, normal Reactor Building Ventilation has tripped and RBEVS has auto-started. Since RBEVS fan 11 is tagged out, only RBEVS fan 12 is initially maintaining RB D/P. Once RBEVS fan 12 trips, nothing is available to maintain RB D/P negative. The un-isolable steam leak into the building will therefore make D/P go positive. With positive RB D/P and elevated building airborne rad levels from the steam leak, this leads to a ground level release and a rise in off-site release rate.

A. Incorrect – Since Reactor Building Ventilation exhaust radiation monitors are greater than 5 mR/hr, normal Reactor Building Ventilation has tripped.
B. Incorrect – Since Reactor Building Ventilation exhaust radiation monitors are greater than 5 mR/hr, normal Reactor Building Ventilation has tripped. Off-site release rate rises due to an uncontrolled ground level release from the Reactor Building because the Reactor Building is NOT designed to be a zero-leakage containment without ventilation.
C. Incorrect – Off-site release rate rises due to an uncontrolled ground level release from the Reactor Building because the Reactor Building is NOT designed to be a zero-leakage containment without ventilation.

Technical Reference(s): OP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-261000-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262001 A4.01
	Importance Rating	3.4

AC Electrical Distribution

Ability to manually operate and/or monitor in the control room: All breakers and disconnects (including available switch yard): Plant-Specific

Proposed Question: #45

A plant startup is in progress and the Main Generator is about to be synchronized to the grid by closing breaker R915.

Which one of the following describes voltage and frequency requirements for closing R915, in accordance with N1-OP-32, Generator?

	Voltage	Frequency
A.	Generator and grid voltages matched	Synchroscope rotating slowly in the SLOW direction (counterclockwise)
B.	Generator and grid voltages matched	Synchroscope rotating slowly in the FAST direction (clockwise)
C.	Generator voltage at least 3-5 kV above grid voltage	Synchroscope rotating slowly in the SLOW direction (counterclockwise)
D.	Generator voltage at least 3-5 kV above grid voltage	Synchroscope rotating slowly in the FAST direction (clockwise)

Proposed Answer: B

Explanation: Incoming (Generator) and running (grid) voltages must be matched prior to closing R915. Incoming (Generator) frequency must be slightly higher than running (grid) frequency prior to closing R915. This is indicated by the synchroscope rotating slowly in the FAST direction (clockwise).

A. Incorrect – The synchroscope must be rotating slowly in the FAST direction.
C. Incorrect – Generator and grid voltages must be matched. The synchroscope must be rotating slowly in the FAST direction.
D. Incorrect – Generator and grid voltage must be matched.

Technical Reference(s): OP-32

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-245001-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	262002 K3.14
	Importance Rating	2.8

UPS (AC/DC)

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

Proposed Question: #46

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

- Motor-Generator (MG) Set 141 trips.
- Powerboard 16B trips.

Which one of the following describes the response of Reactor power to these electrical losses, if any?

Reactor power…

A. remains at 100%.

B. rapidly lowers due to a Reactor scram.

C. rises and stabilizes slightly above 100%.

D. lowers and stabilizes slightly below 100%.

Proposed Answer: C

Explanation: The loss of MG Set 141 causes the loss of an entire string of Feedwater heating. This leads to lowering Feedwater inlet temperature, colder water entering the core region, increased moderation of neutrons, and rising Reactor power.

A. Incorrect – Reactor power rises due to a loss of Feedwater heating.
B. Incorrect – Although Powerboard 16B is normally supplying power to RPS Bus 11 through UPS 162, on the loss of Powerboard 16B, UPS 162 continues to power RPS Bus 11 from Battery 11 with no half scram on RPS 11. A half scram does exist on RPS 12 due to loss of RPS trip bus 141.
D. Incorrect – Reactor power rises due to a loss of Feedwater heating. Powerboard 16B normally powers MG Set 167, which is related to Feedwater Level Control and Turbine Controls, which can affect Reactor power. However, MG Set 167 automatically transfers to Battery Board 11 on loss of Powerboard 16B.

Technical Reference(s): C-19409-C Sheet 1B, OP-48, OP-40, SOP-1.5, SOP-16.1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262002-RBO-08

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(7)

Comments:

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

UPS (AC/DC)

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

Proposed Question: #47

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

- Lines 1 and 4 de-energize.
- EDGs 102 and 103 re-energize their respective powerboards.

Which one of the following describes the response of UPS 172 to this electrical transient?

UPS 172...

A. loads are automatically transferred to the bypass transformer. UPS 172 loads remain on this source until manually re-transferred.

B. loads are automatically transferred to the bypass transformer. UPS 172 loads automatically transfer back after power restoration.

C. inverter is automatically supplied power from Battery Board 12. UPS 172 inverter remains supplied from this source until manually re-transferred.

D. inverter is automatically supplied power from Battery Board 12. UPS 172 inverter power supply automatically transfers back after power restoration.

Proposed Answer: D

Explanation: Loss of Lines 1 and 4 causes Powerboard 17B to de-energize. Powerboard 17B is the normal power supply for UPS 172, and also the power supply for the UPS 172 bypass transformer. With the loss of the normal AC input to UPS 172, Battery Board 12 automatically supplies power to the inverter. When normal AC power is restored (automatically after EDG start and synchronization), the UPS 172 inverter automatically swaps back to the normal power source.

A. Incorrect – UPS 172 inverter automatically receives power from Battery Board 12 and the bypass transformer does not have power due to the loss of PB 17B. UPS 172 automatically transfers back to normal power supply.
B. Incorrect – UPS 172 inverter automatically receives power from Battery Board 12 and the bypass transformer does not have power due to the loss of PB 17B.
C. Incorrect – UPS 172 automatically transfers back to normal power supply.

Technical Reference(s): OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-262002-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

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

DC Electrical Distribution

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

Proposed Question: #48

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

- A3-4-3, BATTERY BOARD 11 GROUND, alarms.
- The ground detection voltmeter for Battery 11 indicates +100 V.

Then, a fuse from a recently energized circuit is pulled on Battery Board 11 as part of ground isolation activities, in accordance with N1-OP-47A, 125 VDC Power System.

- The ground detection voltmeter for Battery 11 now indicates 0 V.

Which one of the following describes (1) the current status of grounds on Battery 11 and (2) the required action with the pulled fuse, in accordance with N1-OP-47A?

	Current Status Of Grounds	Required Action With Pulled Fuse
A.	Battery 11 is free of grounds.	Re-install the fuse.
B.	Battery 11 is free of grounds.	Maintain the fuse un-installed.
C.	There is still a ground connected to Battery 11.	Re-install the fuse.
D.	There is still a ground connected to Battery 11.	Maintain the fuse un-installed.

Proposed Answer: B

Explanation: The current ground indication shows 0 V, which indicates the ground cleared and no further grounds are present. The given indications show the removal of the fuse isolated a significant ground, therefore the fuse should NOT be re-installed.

A. Incorrect – The given indications show the removal of the fuse isolated a significant ground, therefore the fuse should NOT be re-installed.
C. Incorrect – The 0 V ground indication indicates that no further grounds exist on Battery 11. The given indications show the removal of the fuse isolated a significant ground, therefore the fuse should NOT be re-installed.
D. Incorrect – The 0 V ground indication indicates that no further grounds exist on Battery 11.

Technical Reference(s): OP-47A

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-10

Question Source: Modified – JAF 2012 NRC #50

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 #	263000 2.2.36
	Importance Rating	3.1

DC Electrical Distribution

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

Proposed Question: #49

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

- Electrical Maintenance reports that they have detected a hot spot on the output of Static Battery Charger 161.
- Static Battery Chargers 161A and 161B must be immediately removed from service per N1-OP-47A, 125 VDC Power System, to repair the issue.

Which one of the following describes the effect of removing Static Battery Chargers 161A and 161B from service on Battery System 11 operability, in accordance with N1-OP-47A?

Battery System 11...

A. is operable as long as voltage remains 106 volts or greater. If voltage lowers below 106 volts, Battery System 11 operability can be maintained by aligning MG Set 167 to charge Battery 11.

B. is operable as long as voltage remains 106 volts or greater. If voltage lowers below 106 volts, MG Set 167 can be aligned to charge Battery 11, but Battery System 11 will remain inoperable.

C. must immediately be declared inoperable. Battery System 11 may be declared operable again if MG Set 167 is aligned to charge Battery 11.

D. must immediately be declared inoperable. MG Set 167 may be aligned to charge Battery 11, but Battery System 11 will remain inoperable.

Proposed Answer: D

Explanation: OP-47A P&L 9.0 states that removal of SBC 161A/B from service is treated as a loss of the associated Battery System and Technical Specification 3.6.3.h applies. Use of MG Set 167 as a battery charger is allowed, but does not result in declaring the Battery System operable.

A. Incorrect – Per OP-47A P&L 9.0, TS 3.6.3.h must be entered for an inoperable Battery System as soon as both Static Battery Chargers are removed from service.
B. Incorrect – Per OP-47A P&L 9.0, TS 3.6.3.h must be entered for an inoperable Battery System as soon as both Static Battery Chargers are removed from service.
C. Incorrect – Per OP-47A P&L 9.0, TS 3.6.3.h must be entered for an inoperable Battery System as soon as both Static Battery Chargers are removed from service.

Technical Reference(s): OP-47A, Technical Specification 3.6.3.h

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-263000-RBO-10

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 A2.09
	Importance Rating	3.7

EDGs

Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of A.C. power

Proposed Question: #50

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

- An electrical fault develops on Powerboard 102.
- Breaker R1012, Reserve Supply to PB-102, trips due to overcurrent.

Which one of the following describes the response of EDG 102 and the required operator action?

	EDG 102...	Required Operator Action
A.	starts.	Manually close EDG 102 output breaker.
B.	starts.	Perform an emergency shutdown of EDG 102.
C.	remains in standby.	Manually start EDG 102.
D.	remains in standby.	Place EDG 102 output breaker in pull-to-lock.

Proposed Answer: B

Explanation: EDG 102 starts due to undervoltage on Powerboard 102, however EDG 102 output breaker will not close on the bus due to the presence of a fault. Powerboard 102 and 16B will remain de-energized, which leaves EDG 102 running without cooling water. SOP-33A.4 requires emergency shutdown of EDG 102 in this situation.

A. Incorrect – EDG 102 output breaker will not close due to the fault on Powerboard 102.
C. Incorrect – EDG 102 starts due to undervoltage on Powerboard 102.
D. Incorrect – EDG 102 starts due to undervoltage on Powerboard 102.

Technical Reference(s): SOP-33A.4, 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(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	264000 K1.06
	Importance Rating	3.2

EDGs

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

Proposed Question: #51

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

- Annunciator A5-4-5, DSL. GEN. 103 STAND BY OFF NORMAL, alarms.
- Computer point F035, DSL GEN 103 START AIR PR, is alarming LOW.
- A Plant Operator reports EDG 103 starting air pressure is 180 psig.

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

The alarm is...

A. invalid because EDG 103 starting air pressure is normal.

B. valid. EDG 103 still has enough starting air pressure for at least five start attempts.

C. valid. EDG 103 may NOT have enough starting air pressure for five start attempts, but does have enough starting air pressure for EDG 103 operability.

D. valid. EDG 103 does NOT have enough starting air pressure to ensure operability.

Proposed Answer: D

Explanation: 210 psig is the low starting air pressure alarm setpoint, therefore the alarm is valid. The EDG starting air system is designed to allow for at least 5 start attempts with normal starting air pressure. EDG operability is not ensured with starting air pressure below 185 psig.

A. Incorrect – 210 psig is the low starting air pressure alarm setpoint, therefore the alarm is valid.
B. Incorrect – EDG starting air pressure is too low to ensure operability.
C. Incorrect – EDG starting air pressure is too low to ensure operability.

Technical Reference(s): ARP A5-4-5, N1-OP-45

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-264001-RBO-3

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(8)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	300000 K4.02
	Importance Rating	3.0

Instrument Air

Knowledge of INSTRUMENT AIR SYSTEM design feature(s) and or interlocks which provide for the following: Cross-over to other air systems

Proposed Question: #52

Which one of the following describes the operation of valve 94-19, BV - HSA RECEIVER TO IA SEPARATOR, ABSORBER A.F.?

94-19 automatically opens on low ___(1)___ air pressure. If air pressure recovers, 94-19 ___(2)___.

	(1)	(2)
A.	service	automatically re-closes
B.	service	must be manually re-closed
C.	instrument	automatically re-closes
D.	instrument	must be manually re-closed

Proposed Answer: D

Explanation: 94-19 is a normally closed valve between service and instrument air. If air pressure lowers to 90 psig in instrument air receiver 11, 94-19 opens to allow service air to backup instrument air. Once open, 94-19 must be manually re-closed.

A. Incorrect – 94-19 opens on low instrument air pressure. 94-19 does not automatically re-close.
B. Incorrect – 94-19 opens on low instrument air pressure.
D. Incorrect – 94-19 does not automatically re-close.

Technical Reference(s): N1-OP-20

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-278001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(4)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	1
	K/A #	400000 A3.01
	Importance Rating	3.0

Component Cooling Water

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

Proposed Question: #53

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

- Annunciator H1-4-4, CLOSED LOOP COOL MAKEUP TANK LEVEL HIGH-LOW, is in alarm.
- Closed Loop Cooling Make-Up Tank level indicates 3 feet.

Which one of the following describes an automatic action that results from these conditions?

A. Operating RBCLC pumps will trip on a low tank level signal.

B. Operating RBCLC pumps will trip on a low suction pressure signal.

C. 71-127, CLC MU Tank LCV, will be open due to low tank level.

D. 71-127, CLC MU Tank LCV, will be closed due to high tank level.

Proposed Answer: C

Explanation: 3 feet is the low alarm setpoint for annunciator H1-4-4. At this level, the CLC makeup tank level control valve should be open to admit more water from Condensate Transfer to the CLC makeup tank.

A. Incorrect – RBCLC pumps do not trip based on low tank level. Low tank level may cause a low suction pressure alarm or overcurrent trip.
B. Incorrect – RBCLC pumps do not trip based on low suction pressure. Low suction pressure is an alarm and start-permissive only.
D. Incorrect – The indicated tank level is low.

Technical Reference(s): ARP H1-4-4, N1-OP-11

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-208000-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 #	201001 A4.06
	Importance Rating	2.8

CRD Hydraulic

Ability to manually operate and/or monitor in the control room: SDV isolation valve test switch

Proposed Question: #54

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

- N1-ST-Q4, Reactor Coolant System Isolation Valves Operability Test, is in progress.
- The following step is about to be performed at Control Room panel F4:

8.6.2 Place 113-277A SDV VENT & DRAIN VALVES control switch to CLOSE.

Which one of the following describes the expected response of the Scram Discharge Volume (SDV) vent and drain valves when this step is performed?

A. One vent valve and one drain valve close and remain closed.

B. Both vent valves and both drain valves close and remain closed.

C. One vent valve and one drain valve close and then automatically re-open.

D. Both vent valves and both drain valves close and then automatically re-open.

Proposed Answer: B

Explanation: There is a single switch to control the two SDV vent valves and the two SDV drain valves. When taken to CLOSE and released, the switch spring-returns to normal. However, the valves remain closed until re-opened.

A. Incorrect – Both vent valves and both drain valves close.
C. Incorrect – Both vent valves and both drain valves close. The valves remain closed until the switch is taken to OPEN.
D. Incorrect – The valves remain closed until the switch is taken to OPEN.

Technical Reference(s): N1-ST-Q4

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201002 2.1.20
	Importance Rating	4.6

RMCS

Ability to interpret and execute procedure steps.

Proposed Question: #55

The plant was operating at 100% power when an ATWS occurred with the following:

- The Mode Switch is in SHUTDOWN.
- The RPS pushbuttons have been depressed.
- ARI has been manually initiated.
- All Group Scram Solenoid white lights are de-energized.
- All Backup Scram Valve red lights are de-energized.
- The Individual Control Rod Scram blue lights are de-energized.
- Drive water differential pressure indicates 260 psig.
- Scram solenoid air header pressure indicates 70 psig and stable.
- No control rod motion has been achieved.

Given the following procedure sections from N1-EOP-3.1, Alternate Control Rod Insertion:

1. Scram Control Rods Electrically
2. Scram Control Rods by Venting the Scram Air Header
3. Drive Control Rods Using Reactor Manual Control System

Which one of the following identifies the methods above that will insert the control rods?

A. 2 only

B. 1 and 3 only

C. 2 and 3 only

D. 1, 2, and 3

Proposed Answer: C

Explanation: The given indications show that actions have resulted in de-energizing the scram channels, however the scram air header has not depressurized and scram inlet and outlet valves have not opened. Since the group scram solenoid white lights and backup scram valve red lights are de-energized, pulling the scram fuses will have no effect. Since scram solenoid air header pressure is still at the pre-scram normal value of 70 psig, venting the scram air header is a viable option. Since drive water pressure is available and no indications are given with an RMCS problem, rods may also be driven in with RMCS.

A. Incorrect – Method 3 is also available.
B. Incorrect – Method 1 will not work since scram lights are de-energized. Method 2 is also available.
D. Incorrect – Method 1 will not work since scram lights are de-energized.

Technical Reference(s): N1-EOP-3.1, N1-OP-5, C-18016-C, C-19859-C sheet

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP3.1C01 EO #2

Question Source: Modified Bank 2010 Audit #62

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	201006 K4.02
	Importance Rating	3.5

RWM

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

Proposed Question: #56

Which one of the following describes the number of (1) insert errors and (2) withdraw errors that first result in the Rod Worth Minimizer enforcing a rod block?

	Insert Errors	Withdraw Errors
A.	Two	One
B.	Two	Two
C.	Three	One
D.	Three	Two

Proposed Answer: C

Explanation: The Rod Worth Minimizer enforces a rod block based on either three insert errors or one withdraw error.

A. Incorrect – The Rod Worth Minimizer first enforces a rod block when three insert errors are present.
B. Incorrect – The Rod Worth Minimizer first enforces a rod block when three insert errors or one withdraw error are present.
D. Incorrect – The Rod Worth Minimizer first enforces a rod block when one withdraw error is present.

Technical Reference(s): N1101201003C01, N1-OP-37

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-201003-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	202001 A1.04
	Importance Rating	3.3

Recirculation

Ability to predict and/or monitor changes in parameters associated with operating the RECIRCULATION SYSTEM controls including: Reactor water level

Proposed Question: #57

The plant is operating at 100% with the following:

- Operators are executing N1-SOP-1.1, Emergency Power Reduction, due to elevated Turbine vibrations.
- Recirculation flow is rapidly lowered by 5 Mlbm/hr using the master controller.

Which one of the following describes the initial response of Reactor pressure and water level as Recirculation flow is lowered?

	Reactor Pressure	Reactor Water Level
A.	Lowers	Lowers
B.	Lowers	Rises
C.	Rises	Lowers
D.	Rises	Rises

Proposed Answer: B

Explanation: The reduction in recirculation flow lowers Reactor power and main steam flow. Due to lower head losses in the main steam lines and action by the Turbine controls to maintain constant Turbine inlet pressure, Reactor pressure lowers. As recirculation flow lowers, less water is removed from the Reactor annulus region by the recirc pumps, while Feedwater inlet flow initially remains high. Additionally, increased core voiding tends to push water out of the core region and into the annulus. Therefore, Reactor water level (measured in the annulus region) rises initially until Feedwater level control adjusts.

A. Incorrect – Reactor water level rises.
C. Incorrect – Reactor pressure lowers. Reactor water level rises.
D. Incorrect – Reactor pressure lowers.

Technical Reference(s): 1101-202001C01, 1101-248000C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-11

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	216000 K6.01
	Importance Rating	3.1

Nuclear Boiler Instrumentation

Knowledge of the effect that a loss or malfunction of the following will have on the NUCLEAR BOILER INSTRUMENTATION: A.C. electrical distribution

Proposed Question: #58

The plant is operating at 100% power when RPS Bus 12 de-energizes.

Which one of the following identifies the status of Reactor pressure transmitters, 36-07(A-D)?

	36-07A	36-07B	36-07C	36-07D
A.	Energized	De-energized	Energized	De-energized
B.	Energized	Energized	De-energized	De-energized
C.	De-energized	De-energized	Energized	Energized
D.	De-energized	Energized	De-energized	Energized

Proposed Answer: A

Explanation: 120 VAC RPS Bus 11 supplies power to Reactor pressure transmitters 36-07A and 36-07C. 120 VAC RPS Bus 12 supplies power to Reactor pressure transmitters 36-07B and 36-07D. Therefore on a loss of RPS Bus 12 only, 36-07A and 36-07C remain energized and 36-07B and 36-07D de-energize.

B. Incorrect – 36-07B is de-energized. 36-07C is energized.
C. Incorrect – 36-07A is energized. 36-07D is de-energized.
D. Incorrect – 36-07A and 36-07C are energized. 36-07B and 36-07D are de-energized.

Technical Reference(s): C-19957-C Sheet 1, 1101-216000C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-216000-RBO-4

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(7)

Comments:

Examination Outline Cross-Reference:

Level	RO
Tier #	2
Group #	2
K/A #	219000 A4.02
Importance Rating	3.7

RHR/LPCI: Torus/Pool Cooling Mode

Ability to manually operate and/or monitor in the control room: Valve lineup

Proposed Question: #59

Which one of the following describes how the Containment Spray Bypass valves (80-40, 80-41, 80-44, 80-45) and 80-118, CONT SPRAY TEST TO TORUS FCV, are aligned to place Containment Spray in the Torus Cooling lineup?

	Containment Spray Bypass Valves	80-118
A.	All closed	Closed
B.	All closed	Open
C.	At least one open	Closed
D.	At least one open	Open

Proposed Answer: D

Explanation: The Torus Cooling lineup requires 80-118 open, as well as at least one Containment Spray Bypass valve.

A. Incorrect – At least one Containment Spray Bypass valve and 80-118 must be open to achieve the Torus Cooling lineup.
B. Incorrect – At least one Containment Spray Bypass valve and 80-118 must be open to achieve the Torus Cooling lineup.
C. Incorrect – 80-118 is opened to achieve the Torus Cooling lineup.

Technical Reference(s): N1-EOP-1 Attachment 16, C-18012-C Sheet 2

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-226000-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 #	226001 2.4.20
	Importance Rating	3.8

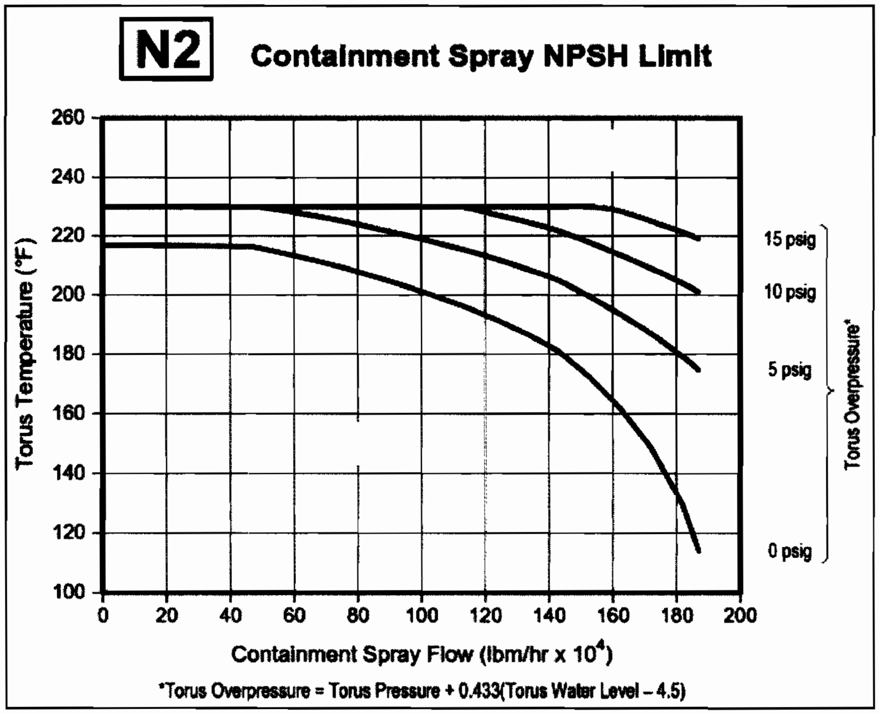
RHR/LPCI: Containment Spray Mode

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

Proposed Question: #60

A loss of coolant accident has resulted in the following:

- Containment Spray is in service.
- Torus pressure is 6 psig.
- Torus water level is 12.0 feet.
- Torus temperature is 180°F.
- Containment Spray flow is 160 x 10^4 lbm/hr.



Which one of the following describes (1) the status of the Containment Spray NPSH Limit and (2) the ability to use Containment Spray if the Containment Spray NPSH Limit is exceeded?

Containment Spray operation is currently in the __(1)__ region of the Containment Spray NPSH Limit. If the Containment Spray NPSH Limit is exceeded, Containment Spray operation __(2)__.

	(1)	(2)
A.	BAD	must be immediately stopped
B.	BAD	may continue, but operation in the BAD region should be minimized
C.	GOOD	must be immediately stopped
D.	GOOD	may continue, but operation in the BAD region should be minimized

Proposed Answer: D

Explanation: The given plant parameters result in a Torus overpressure calculation of 9.25 psig. This allows the 5 psig curve to be used, which results in the given Torus temperature and Containment Spray flow being in the GOOD region of the NPSH curve. An EOP-4 caution states:

⚠ **Operating Containment Spray with suction from the torus while above the NPSH Limit (Fig N2) or with torus water level below 6.3 ft may cause system damage.**

The EOP bases states "Containment Spray operation with torus temperature above the NPSH limit or torus water level below 6.3 ft. may result in system damage and should be avoided… The operating limits are specified in a caution to provide event-specific flexibility and because it is difficult to define in advance exactly when the limits should be observed. Although the caution does not expressly prohibit operation beyond the limits, challenges to the limits should be considered only if the risk of equipment damage is warranted by the nature of the event… Operation beyond the NPSH limit is not expected to result in immediate or catastrophic pump failure."

A. Incorrect – The given parameters are in the GOOD region of the NPSH limit. If in the BAD region of the NPSH limit, Containment Spray operation should be avoided/minimized, but is not expressly prohibited.
B. Incorrect – The given parameters are in the GOOD region of the NPSH limit.
C. Incorrect – If in the BAD region of the NPSH limit, Containment Spray operation should be avoided/minimized, but is not expressly prohibited.

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: 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 #	239001 A2.11
	Importance Rating	4.1

Main and Reheat Steam

Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Steam line break

Proposed Question: #61

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

- A Main Steam leak develops inside the steam tunnel.
- Operators place the mode switch in SHUTDOWN.
- Main Steam tunnel temperature reaches a peak of 205°F.

Which one of the following describes the available systems for controlling Reactor pressure, in accordance with N1-EOP-2, RPV Control?

A. Turbine Bypass Valves, Emergency Condensers, and ERVs remain available.

B. Turbine Bypass Valves are unavailable. Emergency Condensers and ERVs may be used.

C. Turbine Bypass Valves and Emergency Condensers are unavailable. ERVs may be used.

D. Turbine Bypass Valves, Emergency Condensers, and ERVs are unavailable. Alternate Pressure Control systems must be used.

Proposed Answer: B

Explanation: Main Steam tunnel temperature of 205oF causes a vessel isolation signal. MSIVs close on the vessel isolation, making Turbine Bypass Valves unavailable for pressure control. Emergency Condenser vents and drains also close on the vessel isolation, however the Emergency Condensers are still available for pressure control. ERVs tap off the Main Steam lines inside of the MSIVs, and therefore also remain available for pressure control.

A. Incorrect – Turbine Bypass Valves are unavailable due to MSIV closure.
C. Incorrect – Emergency Condensers remain available.
D. Incorrect – Emergency Condensers and ERVs remain available.

Technical Reference(s): ARP F1-2-2, N1-SOP-40.2, C-18002-C Sheet 1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-5

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 #	2
	K/A #	271000 K6.11
	Importance Rating	3.2

Offgas

Knowledge of the effect that a loss or malfunction of the following will have on the OFFGAS SYSTEM: Condenser vacuum

Proposed Question: #62

The plant is operating at 100% power when a through-wall crack develops in the low-pressure Turbine exhaust hood.

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

A. Main Condenser vacuum lowers and Offgas flow-rate rises.

B. Main Condenser vacuum lowers and Offgas flow-rate lowers.

C. Steam leaks into the Turbine Building and Offgas flow-rate rises.

D. Steam leaks into the Turbine Building and Offgas flow-rate lowers.

Proposed Answer: A

Explanation: The pressure at the low-pressure Turbine exhaust is below atmospheric pressure. Therefore, leakage is from the Turbine Building atmosphere into the Main Condenser through the crack. The introduction of air into the Main Condenser causes vacuum to lower. Some of the additional air will be drawn out of the Main Condenser through the SJAEs and into the Offgas system. This will cause Offgas flow-rate to rise.

B. Incorrect – Offgas flow-rate rises, as additional air is removed from the Main Condenser.
C. Incorrect – Leakage is from the Turbine Building atmosphere into the Main Condenser.
D. Incorrect – Leakage is from the Turbine Building atmosphere into the Main Condenser. Offgas flow-rate rises, as additional air is removed from the Main Condenser.

Technical Reference(s): N1-OP-25, ARP H1-2-6

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-271000-RBO-11

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 K4.03
	Importance Rating	2.8

Plant Ventilation

Knowledge of PLANT VENTILATION SYSTEMS design feature(s) and/or interlocks which provide for the following: Automatic starting and stopping of fans

Proposed Question: #63

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

- Outside air temperature is 34°F.
- Reactor Building Ventilation Supply fan 11 is operating.

Then, 202-15, REACTOR BLDG SUPPLY ISOLATION VALVE 11, spuriously closes.

Which one of the following describes the response of the Reactor Building Ventilation system?

	Supply Air Heater	Supply Fan 11
A.	Trips	Trips
B.	Trips	Remains running
C.	Remains energized	Trips
D.	Remains energized	Remains running

Proposed Answer: A

Explanation: The RBVS supply air flow path contains two IVs (202-15 and 202-16) in series. Closure of a single IV isolates the line. The RBVS supply air heater would normally be energized with outside air temperature of 34°F, however it automatically trips on low flow once 202-15 closes. Supply fan 11 receives a trip signal based on position of 202-15.

B. Incorrect – Supply fan 11 trips based on position of 202-15.
C. Incorrect – The supply air heater trips on low flow.
D. Incorrect – The supply air heater trips on low flow. Supply fan 11 trips based on position of 202-15.

Technical Reference(s): C-18013-C, C-19432-C Sheet 2, C-19859-C Sheet 15

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288001-RBO-5

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290001 K6.08
	Importance Rating	2.7

Secondary Containment

Knowledge of the effect that a loss or malfunction of the following will have on the SECONDARY CONTAINMENT: Plant air systems

Proposed Question: #64

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

- Reactor Building Ventilation is in service.
- Reactor Building Emergency Ventilation is in standby.
- Both Reactor Building Track Bay doors are closed.
- Reactor Building differential pressure is -0.25" H_2O.

Then, a total loss of Instrument Air occurs.

Which one of the following describes the effect of the Instrument Air loss on Reactor Building differential pressure?

Reactor Building differential pressure is…

A. maintained by normal Reactor Building Ventilation.

B. lost due to Reactor Building Track Bay door seal failure.

C. lost due to isolation of normal Reactor Building Ventilation.

D. maintained by auto-start of Reactor Building Emergency Ventilation.

Proposed Answer: C

Explanation: Reactor Building Ventilation isolation dampers (202-15, 16, 31, 32) fail closed on loss of Instrument Air. This isolates normal Reactor Building Ventilation. Without Reactor Building Ventilation, the Reactor Building differential pressure will lower as the building pressure equalizes to atmospheric pressure through leakage paths.

A. Incorrect – Normal Reactor Building Ventilation isolates.
B. Incorrect – Only the inner Reactor Building Track Bay door relies on Instrument Air for sealing. With the outer Reactor Building Track Bay door closed, its seal would hold Reactor Building differential pressure if fans remained in service.
D. Incorrect – Reactor Building Emergency Ventilation dampers do open on loss of Instrument Air, but the fans do not auto-start.

Technical Reference(s): N1-SOP-20.1, C-18013-C, N1-OP-52, N1-OP-10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-290001-RBO-8

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(9)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	2
	Group #	2
	K/A #	290003 K5.03
	Importance Rating	2.6

Control Room HVAC

Knowledge of the operational implications of the following concepts as they apply to CONTROL ROOM HVAC: Temperature control

Proposed Question: #65

Which one of the following describes how the Control Room Ventilation Chillers are aligned for proper temperature control based on the season?

During winter months,...

A. only Chiller compressors 111 and 121 (60% compressors) are operated. Chiller compressors 112 and 122 (40% compressors) are secured.

B. only Chiller compressors 112 and 122 (40% compressors) are operated. Chiller compressors 111 and 121 (60% compressors) are secured.

C. Chiller compressors 111 and 121 (60% compressors) are operated in Lead, with Chiller compressors 112 and 122 (40% compressors) operating in standby.

D. Chiller compressors 112 and 122 (40% compressors) are operated in Lead, with Chiller compressors 111 and 121 (60% compressors) operating in standby.

Proposed Answer: D

Explanation: Per N1-OP-49 Precaution and Limitation #4.0:

"The Control Room Ventilation System chillers are operated with the 60% compressors (111 and 121) in lead during warm months (May - September), and the 40% (112 and 122) compressors in lead during the cool months (November - March). This alignment minimizes unnecessary cycling of the compressors."

A. Incorrect – Chiller compressors 112 and 122 are also operated, just as the non-Lead/standby compressors.
B. Incorrect – Chiller compressors 111 and 121 are also operated, just as the non-Lead/standby compressors.
C. Incorrect – Chiller compressors 112 and 122 are operated in Lead during winter months.

Technical Reference(s): N1-OP-49 P&L

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-288003-RBO-9

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

Ability to explain and apply system limits and precautions.

Proposed Question: #66

The plant is returning to rated power after Turbine valve testing with the following:

- Reactor power is approximately 90%.
- Power ascension is in progress with control rods and recirculation flow.

Which one of the following describes (1) the predicted upper power limit for control rod withdrawal and (2) the minimum amount of time required between positive reactivity additions when above 98% power, in accordance with N1-OP-43B, Normal Power Operations?

	(1)	(2)
A.	<1840 MWth	1 minute
B.	<1840 MWth	3 minutes
C.	<1846 MWth	1 minute
D.	<1846 MWth	3 minutes

Proposed Answer: B

Explanation: N1-OP-43B Precaution and Limitation #24.0 contains the following reactivity management expectations:

- When operating at 98% of rated power or greater, positive reactivity additions whether with rods or recirculation flow will be followed by a minimum 3 minute wait period to ensure reactor conditions stabilize prior to making another positive reactivity addition.

- When withdrawing control rods to raise power or rod line, the predicted power achieved with control rods shall be maintained less than 1840 MWth. The final approach to rated at 1840 MWth and above will be done with recirculation flow.

A. Incorrect – When above 98% power, a minimum of 3 minutes is required between positive reactivity additions.
C. Incorrect – Control rod withdrawal is not allowed above 1840 MWth. 1846 MWth is the lower limit of the normal power maintenance band. When above 98% power, a minimum of 3 minutes is required between positive reactivity additions.
D. Incorrect – Control rod withdrawal is not allowed above 1840 MWth. 1846 MWth is the lower limit of the normal power maintenance band.

Technical Reference(s): N1-OP-43B P&L

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-9

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.14
	Importance Rating	3.1

Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc.

Proposed Question: #67

Given the following events:

Event 1. A Reactor Recirculation pump trips.
Event 2. A Reactor Building sump overflows onto the floor.
Event 3. The Reactor reaches criticality during a startup.

Which one of the following identifies (1) the events that must be announced to the plant and (2) the event(s) that should also be accompanied by the station alarm?

	(1)	(2)
A.	1 and 2, only	1 only
B.	1 and 2, only	1 and 2, only
C.	1, 2, and 3	1 only
D.	1, 2, and 3	1 and 2, only

Proposed Answer: D

Explanation: Event 1 must be announced with the station alarm because it requires SOP entry. Event 2 must be announced with the station alarm because it requires EOP-5 entry. Event 3 must be announced, but does not warrant the station alarm, for reaching criticality.

A. Incorrect – Event 3 must be announced. Event 2 should be accompanied by the station alarm.
B. Incorrect – Event 3 must be announced.
C. Incorrect – Event 2 should be accompanied by the station alarm.

Technical Reference(s): CNG-OP-1.01-2001, ARP H2-2-1, N1-EOP-5, N1-SOP-1.3

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-COO-OBJ-03

Question Source: Modified Bank – 2009 Audit #94

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

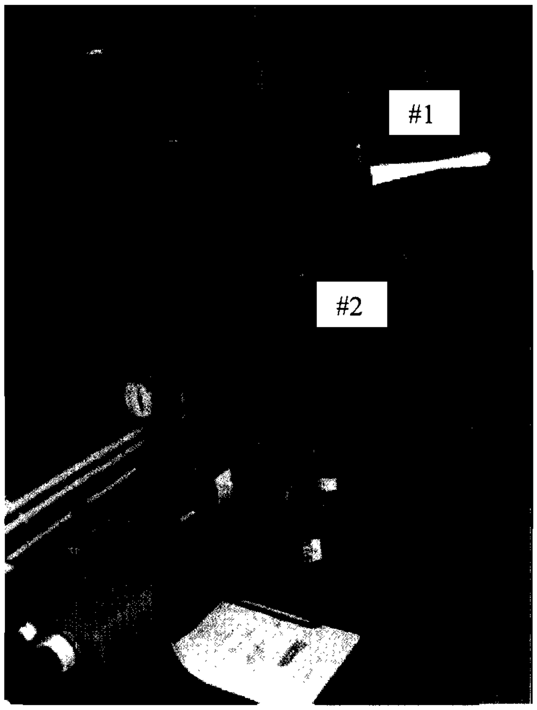
Examination Outline Cross-Reference:	Level	RO
	Tier #	3
	Group #	
	K/A #	2.1.30
	Importance Rating	4.4

Ability to locate and operate components, including local controls.

Proposed Question: #68

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: The scoop tube positioner must first be unlocked by operating handle #1 (HAND LEVER), then the scoop tube must be repositioned with handle #2 (OPERATING LEVER).

A. Incorrect – Both handles must be operated.
B. Incorrect – Both handles must be operated.
C. Incorrect – Handle #1 will not function until handle #2 is moved from LOCK to MANUAL.

Technical Reference(s): N1-OP-1

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-202001-RBO-10

Question Source: Modified Bank – 2002 Duane Arnold NRC

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(6)

Comments:

Examination Outline Cross-Reference:	Level	RO
	Tier #	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.

Proposed Question: #69

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 – N1-OP-43A does place at least one IRM in each RPS channel in Range 10 prior to taking the Mode Switch to RUN, but not all IRMs must be in Range 10.
C. Incorrect – Both Feedwater isolation valves must be open prior to exceeding 25% power for HPCI operability requirements, but not before taking the Mode Switch to RUN.
D. Incorrect – Containment atmosphere oxygen concentration shall be reduced to less than 4% by volume within 24 hours of placing the Mode Switch in RUN, but not 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: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(10)

Comments:

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

Knowledge of the process for managing troubleshooting activities.

Proposed Question: #70

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

- The Control Room is performing N1-ST-C25, Liquid Poison Pump Operability Test, as a post maintenance test.
- After Liquid Poison Pump 11 is locally started, the discharge relief valve begins opening at a pump discharge pressure of approximately 950 psig.
- Liquid Poison Pump 11 is secured and troubleshooting is initiated.

Which one of the following describes how to control Liquid Poison System 11, in accordance with CNG-MN-1.01-1002, Troubleshooting, and CNG-OP-1.01-1000, Conduct of Operations?

A. Quarantine Liquid Poison System 11. If an emergency event occurs requiring use of the system, attempted restart is allowed.

B. Quarantine Liquid Poison System 11. If an emergency event occurs requiring use of the pump, attempted restart is NOT allowed.

C. In order to allow troubleshooting to continue, do NOT quarantine Liquid Poison System 11. If an emergency event occurs requiring use of the system, attempted restart is allowed.

D. In order to allow troubleshooting to continue, do NOT quarantine Liquid Poison System 11. If an emergency event occurs requiring use of the pump, attempted restart is NOT allowed.

Proposed Answer: A

Explanation: Operators are responsible for quarantining malfunctioned equipment to preserve evidence for troubleshooting activities. CNG-OP-1.01-1000 and CNG-MN-1.01-1002 direct limiting access to and operation of malfunctioned equipment. Therefore the system should not be manipulated until further inspection/troubleshooting. Quarantine does not prevent access to equipment for inspection/troubleshooting. The quarantine guidance does allow further operation of the malfunctioned equipment if specifically needed for emergency event response.

B. Incorrect – The quarantine guidance does allow further operation of the malfunctioned equipment if specifically needed for emergency event response.
C. Incorrect – Quarantine should be performed to preserve evidence. Quarantine does not prevent access to equipment for inspection/troubleshooting.
D. Incorrect – Quarantine should be performed to preserve evidence. Quarantine does not prevent access to equipment for inspection/troubleshooting. The quarantine guidance does allow further operation of the malfunctioned equipment if specifically needed for emergency event response.

Technical Reference(s): N1-ST-C25, CNG-OP-1.01-1000, CNG-MN-1.01-1002

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-ADM037-11-TO-01

Question Source: Modified Bank – 2009 Audit #68

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

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

Knowledge of tagging and clearance procedures.

Proposed Question: #71

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

- Electrical Maintenance is preparing to perform an inspection on a lighting panel.
- During the inspection, it is desired for the Electricians to be able to both:
  - Open the panel's disconnect switch for personnel protection, and
  - Close the panel's disconnect switch for periodic verifications.
- It is also desired for the Tagout to be continuously hung during the activity, such that repeated tag clearing and re-hanging is NOT required.

Which one of the following describes a tagging arrangement that will allow this maintenance activity, in accordance with CNG-OP-1.01-1007, Clearance and Safety Tagging?

Tag the panel's disconnect switch with...

A. a danger tag, only.

B. a caution tag, only.

C. an operating permit tag and lock, only.

D. both a danger tag and an operating permit tag.

Proposed Answer: C

Explanation: Components tagged with an operating permit tag may be manipulated by the Tagout Holder, or a person under the direction of the Tagout Holder, without removing the Operating Permit tag. These components may be used for personnel protection. A device tagged with an Operating Permit tag shall not be tagged with a Danger tag at the same time.

A. Incorrect – A component tagged with a danger tag may not be manipulated without removing the tag.
B. Incorrect – A component tagged with a caution tag only may not be used for personnel protection.
D. Incorrect – A component may not be simultaneously tagged with a danger tag and an operating permit tag.

Technical Reference(s): CNG-OP-1.01-1007

Proposed references to be provided to applicants during examination: None

Learning Objective: HAZNRG-CO-00011

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 #	2.3.7
	Importance Rating	3.5

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

Proposed Question: #72

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

- TIP runs are in progress.
- The detector position indication on TIP machine 1 is not working properly.
- TIP machine 1 has been given a withdraw signal.
- An operator is being sent into the TIP Room to verify the detector position.
- The TIP Room is currently posted as a Very High Radiation Area.

Which one of the following describes the type of Radiation Work Permit (RWP) required and the permissions required for TIP room entry?

A. A Specific RWP with permission given by the General Supervisor - Rad Protection only.

B. A Specific RWP with permission given by the General Supervisor - Rad Protection, Shift Manager, and Plant Manager.

C. An Emergency Response RWP with permission given by the Shift Manager only.

D. An Emergency Response RWP with permission given by the General Supervisor - Rad Protection, Shift Manager, and Plant Manager.

Proposed Answer: B

Explanation: Areas with the potential for being Very High Radiation Areas include: TIP Rooms, Upper elevations of the drywell during fuel moves, Spent Fuel Pool during diving operations. A specific RWP is required for entry into Very High Radiation Area.

If the activity is within a Very High Radiation Area, obtain permission before any entry to the area from:

1. General Supervisor - Rad Protection
2. Shift Manager
3. Plant Manager

A. Incorrect – Permission must be given by the Shift Manager, General Supervisor - Rad Protection and Plant General Manager.
C. Incorrect – A specific RWP is required for entry into Very High Radiation Area. An Emergency RWP would be used during an emergency; there is nothing in the stem to indicate an emergency exists. General Supervisor - Rad Protection and Plant Manager must also approve of the entry.
D. Incorrect – A specific RWP is required for entry into Very High Radiation Area. An Emergency RWP would be used during an emergency; there is nothing in the stem to indicate an emergency exists.

Technical Reference(s): GAP-RPP-02, GAP-RPP-08

Proposed references to be provided to applicants during examination: None

Learning Objective: GAP-RPP-02-CT-01

Question Source: Bank – 2010 Audit #71

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41(12)

Comments:

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

Ability to control radiation releases.

Proposed Question: #73

An un-isolable primary system rupture has occurred, causing rising Offsite release rates.

Which one of the following lists the condition that requires entry into N1-EOP-6, Radioactivity Release Control, AND the condition that requires entry into N1-EOP-8, RPV Blowdown, to ensure protection of the general public?

Entry into N1-EOP-6, Radioactivity Release Control, is required when Offsite release rate exceeds the ___(1)___ Emergency Action Level value. Entry into N1-EOP-8, RPV Blowdown, is required before Offsite release rate exceeds the ___(2)___ Emergency Action Level value.

	(1)	(2)
A.	Unusual Event	Site Area Emergency
B.	Unusual Event	General Emergency
C.	Alert	Site Area Emergency
D.	Alert	General Emergency

Proposed Answer: D

Explanation: Entry into N1-EOP-6 is required at the Alert level. Entry into N1-EOP-8 is required before the General Emergency level.

A. Incorrect – Entry into N1-EOP-6 is required at the Alert level. Entry into N1-EOP-8 is required before the General Emergency level.
B. Incorrect – Entry into N1-EOP-6 is required at the Alert level.
C. Incorrect – Entry into N1-EOP-8 is required before the General Emergency level.

Technical Reference(s): N1-EOP-6

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO #2

Question Source: Bank – 2010 NRC #59

Question History: 2010 NRC #59

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 #	2.4.14
	Importance Rating	3.8

Knowledge of general guidelines for EOP usage.

Proposed Question: #74

A plant shutdown is in progress with the following:

- All control rods are inserted.
- Reactor pressure is 50 psig and lowering slowly.

Then, a coolant leak inside the Containment results in the following:

Time (hh:mm)	Event
00:00	Drywell temperature is 150°F and rising.
00:05	Drywell pressure is 3.5 psig and rising.

Which one of the following describes the EOP entry requirements for this event?

	N1-EOP-4, Primary Containment Control	N1-EOP-2, RPV Control
A.	Must be entered at time 00:00 and re-entered at time 00:05.	Must be entered at time 00:05.
B.	Must be entered at time 00:00 and re-entered at time 00:05.	Does NOT need to be entered.
C.	Must be entered at time 00:00. Does NOT need to be re-entered at time 00:05.	Must be entered at time 00:05.
D.	Must be entered at time 00:00. Does NOT need to be re-entered at time 00:05.	Does NOT need to be entered.

Proposed Answer: A

Explanation: A general rule of EOP usage is that each EOP is entered upon receipt of an entry condition, and re-entered if another entry condition is later received. This rule ensures that all steps in the EOP are re-addressed in light of the new condition, which could result in different mitigating actions. Drywell temperature of 150°F is an entry condition to N1-EOP-4. Drywell pressure of 3.5 psig is an entry condition to N1-EOP-4 and N1-EOP-2.

B. Incorrect – N1-EOP-2 must be entered due to high Drywell pressure at time 00:05, even though all control rods are in and Reactor pressure is substantially below normal operating pressure.
C. Incorrect – Even though N1-EOP-4 was previously entered, it must be re-entered at time 00:05.
D. Incorrect – Even though N1-EOP-4 was previously entered, it must be re-entered at time 00:05. N1-EOP-2 must be entered due to high Drywell pressure at time 00:05, even though all control rods are in and Reactor pressure is substantially below normal operating pressure.

Technical Reference(s): N1-EOP-2, N1-EOP-4, NER-1M-095, GAI-OPS-20

Proposed references to be provided to applicants during examination: None

Learning Objective: S101-EOP00C01 TO #1

Question Source: Modified Bank – 2003 Riverbend NRC Exam

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41(10)

Comments:

Examination Outline Cross-Reference:		
	Level	RO
	Tier #	3
	Group #	
	K/A #	2.4.30
	Importance Rating	2.7

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

Proposed Question: #75

Which one of the following describes the notification required following initiation of an Emergency Condenser (EC), in accordance with N1-OP-13, Emergency Cooling System?

Notify…

A. Chemistry to assess EC effluent dose.

B. Radiation Protection to assess EC effluent dose.

C. Chemistry to assess Reactor Building radiation levels.

D. Radiation Protection to assess Reactor Building radiation levels.

Proposed Answer: A

Explanation: N1-OP-13 requires notification to Chemistry after either automatic or manual initiation of Emergency Condensers to assess effluent dose in accordance with Tech Specs and the ODCM.

B. Incorrect – Chemistry, not Radiation Protection, is responsible for assessing EC effluent dose.
C. Incorrect – The reason is to assess EC effluent dose.
D. Incorrect – Chemistry, not Radiation Protection, is responsible for assessing EC effluent dose.

Technical Reference(s): N1-OP-13

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-207000-RBO-10

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	SRO
	Tier #	1
	Group #	1
	K/A #	295026 2.1.25
	Importance Rating	4.2

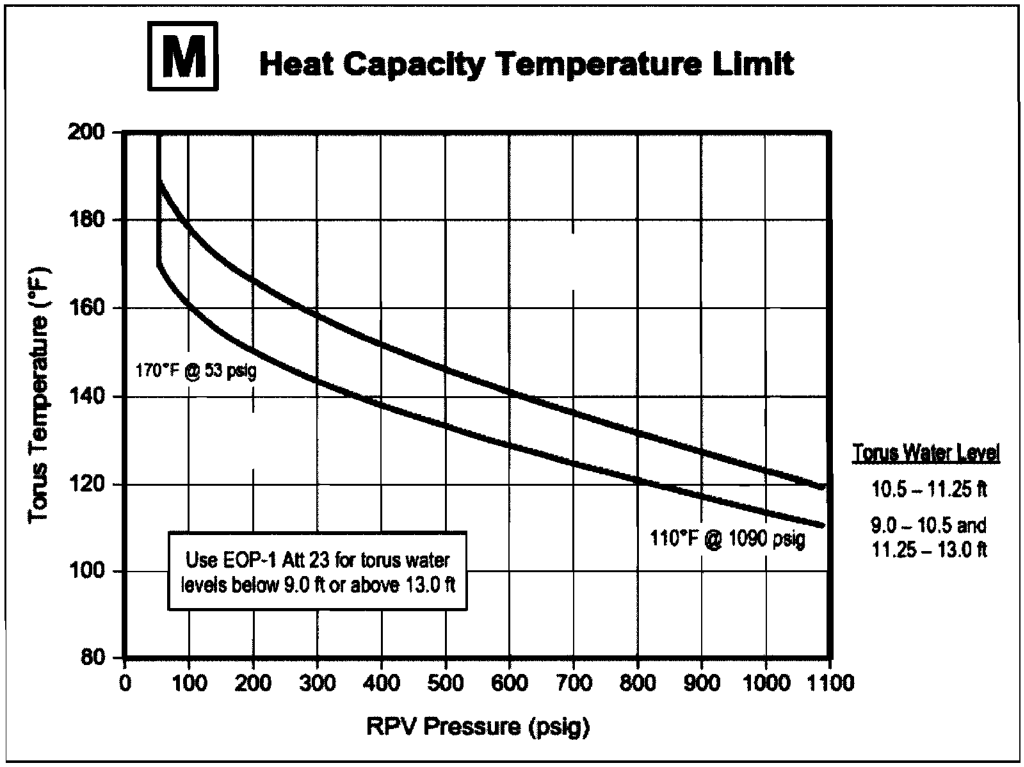
Suppression Pool High Water Temperature

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

Proposed Question: #76

An ATWS has occurred with the following conditions:

- Reactor power is 20%.
- Reactor water level is -50" and lowering.
- Turbine Bypass Valves have failed closed.
- Both Emergency Condensers are in service.
- ERVs are being manually opened to control Reactor pressure.
- Reactor pressure is 700 psig and stable.
- Torus water temperature is 130°F and rising.
- Torus water level is 11.1 ft and rising.
- Liquid Poison is being injected.
- Initial Liquid Poison tank level was 1450 gallons.
- Current Liquid Poison tank level is 1000 gallons.



Which one of the following indicates the NEXT action that is required to be directed by the Control Room Supervisor?

A. Perform an RPV Blowdown due to HCTL violation

B. Raise Reactor water level to 53" to 95" to mix boron

C. Reduce Reactor pressure to stay below the applicable HCTL limit

D. Inject with Condensate/FW to maintain Reactor water level -50" to -109"

Proposed Answer: C

Explanation: For a Torus water level of 11.1 ft, the top curve on the HCTL graph is used. When RPV pressure is 700 psig, the torus temperature limit is ~137°F. Reactor is at 20% power. Emergency Condensers take away ~6% of Reactor power. An excess of ~14% Reactor power is being sent to the Torus. The CRS should direct Reactor pressure to be lowered to stay below HCTL.

A. Incorrect – A blowdown is not required unless HCTL has been violated. This answer would be the correct answer if the lower curve is used on the HCTL graph. The lower curve is only used if Torus level is 9-10.5 ft or 11.25-13.0 ft.
B. Incorrect – Level is only raised in N1-EOP-3 when hot shutdown boron is injected. Hot shutdown boron is 600 gallons and only 450 gallons have been injected.
D. Incorrect – With Torus temp above 110°F, power above 6% and an ERV open, RPV water level must continue to be lowered.

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 – 2010 Audit #80

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295030 2.1.23
	Importance Rating	4.4

Low Suppression Pool Water Level

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

Proposed Question: #77

A steam leak in the Drywell, coincident with a small Torus water leak, has resulted in the following:

- Drywell pressure is 3.6 psig and rising slowly.
- Torus pressure is 2.0 psig and rising slowly.
- Drywell average temperature is 145°F and rising slowly.
- Torus water level is 10.2 feet and lowering slowly.

Which one of the following describes a direction that must be given in accordance with N1-EOP-4, Primary Containment Control?

A. Perform a blowdown in accordance with N1-EOP-8, RPV Blowdown.

B. Raise Torus water level using N1-EOP-1 Attachment 6, Torus Makeup from Keep-Fill.

C. Raise Torus water level using N1-EOP-1 Attachment 18, Raw Water to Torus Makeup.

D. Spray the Containment using N1-EOP-1 Attachment 17, Auto or Manual Initiation of Containment Spray.

Proposed Answer: C

Explanation: The given conditions require N1- EOP-4 entry based on both high Drywell pressure and low Torus water level. The N1-EOP-4 pressure leg does not require Containment Spray until Torus pressure exceeds 13 psig. The N1-EOP-4 Drywell temperature leg does not require Containment Spray until before 300°F, and Drywell average temperature has not yet exceeded even the N1-EOP-4 entry value of 150°F. The N1-EOP-4 Torus level leg requires initiating Torus makeup from either N1-EOP-1 Att 6 or Att 18. However, N1-EOP-1 Att 6 is not allowed in this situation, because Core Spray is running on a high Drywell pressure signal. Therefore, N1-EOP-1 Att 18 must be directed.

A. Incorrect – Containment conditions have not yet degraded to levels that threaten Blowdown benchmarks (Torus pressure of ~18 psig, Drywell temperature of 300°F, Torus level of 8 feet), and the preceding N1-EOP-4 actions have not yet been performed as required prior to proceeding to Blowdown.
B. Incorrect – N1-EOP-1 Att 6 cannot be used if Core Spray pumps are running. Core Spray pumps have started on high Drywell pressure (3.5 psig).
D. Incorrect – The N1-EOP-4 pressure leg does not required Containment Spray until Torus pressure exceeds 13 psig. The N1-EOP-4 Drywell temperature leg does not required Containment Spray until before 300°F, and Drywell average temperature has not yet exceeded even the N1-EOP-4 entry value of 150°F.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP4C01 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 #	295003 AA2.04
	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: System lineups

Proposed Question: #78

The plant is operating at 100% power when Containment Spray pump 112 is tagged out for maintenance.

Then, Powerboard 103 de-energizes due to an electrical fault.

Which one of the following describes the limiting Technical Specification action statement if neither Containment Spray pump 112 nor Powerboard 103 are returned to service?

A. Shutdown shall begin within one hour and the Reactor coolant shall be below 215°F within ten hours.

B. Reactor coolant shall be reduced to 110 psig or less and saturation temperature or less, respectively, within ten hours.

C. Reactor operation may continue for 7 days.

D. Reactor operation may continue for 15 days.

Proposed Answer: A

Explanation: The loss of Powerboard 103 makes Containment Spray pumps 121 and 122 inoperable. With Containment Spray pump 112 already inoperable, Technical Specification 3.3.7 is not met due to loss of both pumps in one system and loss of one pump in the second system. This requires shutdown to be initiated within one hour, and coolant temperature to be less than 215°F within 10 hours.

B. Incorrect – This is the action statement based on Tech Spec 3.1.5 (ADS). This action statement is also applicable due to loss of 103, but is not as limiting as the action statement from Tech Spec 3.3.7. Saturation temperature at 110 psig is approximately 344 °F.
C. Incorrect – This is the action statement based on Tech Spec 3.3.7 for loss of only two Containment Spray pumps.
D. Incorrect – This is the action statement based on Tech Spec 3.3.7 for loss of one Containment Spray pump.

Technical Reference(s): N1-OP-14, Technical Specifications, C-19409-C Sheet 1b, C-18012 Sheet 2

Proposed references to be provided to applicants during examination: Technical Specifications 3.1.4, 3.1.5, 3.3.7, 3.6.3

Learning Objective: N1-226001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295005 AA2.05
	Importance Rating	3.9

Main Turbine Generator Trip

Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Reactor power

Proposed Question: #79

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

- The Reactor scrams.
- Multiple control rods fail to insert.
- All APRM indications are lost.
- The Main Turbine trips.
- Reactor pressure stabilizes at approximately 925 psig with two Turbine Bypass Valve full open and all other Turbine Bypass Valves full closed.
- Reactor water level is 72" and stable with Feedwater injecting.
- N1-EOP-3, Failure to Scram, is being executed.

Which one of the following describes the approximate value of Reactor power based on Turbine Bypass Valve position and how Feedwater injection is to be controlled per N1-EOP-3?

	Reactor power is…	Feedwater injection…
A.	<6%	must be terminated.
B.	<6%	is NOT required to be terminated.
C.	≥6%	must be terminated.
D.	≥6%	is NOT required to be terminated.

Proposed Answer: C

Explanation: Total Turbine Bypass Valve capacity is approximately 34-40% of rated steam flow for all nine Turbine Bypass Valves. This represents a capacity of approximately 3-5% of rated steam flow for one Turbine Bypass Valve. With two Turbine Bypass Valves open (plus some steam flow to auxiliary steam loads), this correlates to approximately 6-12% Reactor power under the given conditions. N1-EOP-3 requires Feedwater injection to be terminated if Reactor water level is above 6% or unknown with Reactor water level above -41". NER-1M-095 states that even with direct Reactor power indications lost (APRMs), indirect methods of determining Reactor power can be used (such as steam flow through Turbine Bypass Valves).

A. Incorrect – Reactor power is greater than or equal to 6% if two Turbine Bypass Valves are full open.
B. Incorrect – Reactor power is greater than or equal to 6% if two Turbine Bypass Valves are full open.
D. Incorrect – With Reactor power greater than or equal to 6% and Reactor water level above -41", N1-EOP-3 requires terminating and preventing Feedwater injection.

Technical Reference(s): N1-OP-31, N1-EOP-3, NER-1M-095, 1101-239001C01

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-239001-RBO-2, 1101-EOP3C01 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 #	295023 AA2.03
	Importance Rating	3.8

Refueling Accidents

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Airborne contamination levels

Proposed Question: #80

The plant is shutdown for a refueling outage with the following:

- An irradiated fuel bundle has been dropped in the Spent Fuel Pool.
- Stack radiation monitors have indicated 4000-6000 cps for the last hour.
- Chemistry has projected the expected dose from the event at the site boundary to be 15 mRem TEDE.

Which one of the following describes the proper emergency classification level?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Proposed Answer: B

Explanation: The projected dose at the site boundary exceeds the value for Alert 5.2.3, but is below the SAE or GE levels.

A. Incorrect – The projected dose at the site boundary exceeds the value for Alert 5.2.3.
C. Incorrect – The projected dose at the site boundary is below the SAE or GE levels.
D. Incorrect – The projected dose at the site boundary is below the SAE or GE levels.

Technical Reference(s): EPIP-EPP-01, EPMP-EPP-0101

Proposed references to be provided to applicants during examination: EAL flowchart

Learning Objective: NS-EPLREF-EO-01

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(4)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	1
	K/A #	295016 2.1.7
	Importance Rating	4.7

Control Room Abandonment

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

Proposed Question: #81

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.
- Actions have been taken to establish control at Remote Shutdown Panel (RSP) 12.
- The following indications are available at RSP 12:
  - Reactor pressure is 800 psig and slowly lowering.
  - Reactor water level is 0 inches and stable.
  - Torus water temperature is 75°F and stable.
  - Drywell pressure is 1.8 psig and stable.
  - Drywell temperature is 135°F and stable.
  - The Control Rods In white light is lit.
  - Emergency Condenser 12 is in service.
  - Emergency Condenser 12 water level is 6.5 feet and slowly lowering with the controller in AUTO.

Which one of the following describes a required action, in accordance with N1-SOP-21.2?

A. Enter N1-EOP-3.1, Alternate Rod Insertion, to insert control rods.

B. Place Emergency Condenser 12 level controller in MAN and raise the makeup rate.

C. Close Emergency Condenser 12 steam isolation valves and transition to RSP 11 section of N1-SOP-21.2.

D. Refer to N1-SOP-29.1, EOP Key Parameter – Alternate Instrumentation, to locate alternate Reactor water level indication.

Proposed Answer: D

Explanation: N1-SOP-21.2 contains an override that states if instrumentation is offscale, use N1-SOP-29.1 for alternate instrumentation. The RSP RPV water level indication is 0-100", therefore it is offscale low. Alternate indications exist on the ATS cabinets that extend down to -34".

A. Incorrect – The Control Rods In lights on the RSPs are de-energized when control rods are withdrawn, and illuminated when all control rods are inserted. Therefore in this case, all control rods successfully inserted on the scram. N1-SOP-21.2 does allow use of N1-EOP-3.1 in the event of failure of all rods to insert on the scram.
B. Incorrect – N1-SOP-21.2 contains a conditional step to take manual control of EC makeup, however the given value is in the normal control range of 6-7'. Lowering EC level is an expected condition with the EC in service due to boil off.
C. Incorrect – N1-SOP-21.2 contains an override to close steam IVs and transition to the other RSP if an EC is malfunctioning. However, the given values of Reactor pressure and EC water level indicate no issues with EC 12.

Technical Reference(s): N1-SOP-21.2, N1-SOP-29.1, N1-EOP-3.1

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-SOP21.2C01 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 #	600000 2.4.11
	Importance Rating	4.2

Plant Fire On Site

Knowledge of abnormal condition procedures.

Proposed Question: #82

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

Time (hh:mm)	Event
00:00	The following Main Fire Panel annunciators alarm: • 2-1-1-1, TURB. BLDG. 261 LOCAL PNL NO. 1 FIRE • 2-2-1-2, DIESEL FIRE PUMP #1 RUNNING • 2-2-2-2, ELECTRIC FIRE PUMP #1 STARTED. Fire Zone WD-8131, T1 XFMR Water Deluge System, is in alarm and indicates discharge is in progress.
00:03	The Fire Brigade leader reports that there are visible flames coming from Transformer 1.

Given the following sections of EPIP-EPP-01-EAL, Emergency Action Level Matrix / Unit 1 **(see next page)**.

Which one of the following describes the earliest time at which the fire can be considered "confirmed" and the time at which EAL 8.2.1 is met, assuming the fire is still in progress, in accordance with EPIP-EPP-28, Firefighting, and EPIP-EPP-01, Classification of Emergency Conditions at Unit 1?

	Earliest Time Fire Can Be Considered "Confirmed"	Time EAL 8.2.1 Is Met
A.	00:00	00:00
B.	00:00	00:15
C.	00:03	00:03
D.	00:03	00:18

8.2.1

1	2	3	4	5

Confirmed fire in or contiguous to any plant area, Table 5 or Table 6, not extinguished in ≤ 15 min. of Control Room notification

Table 5 Plant Areas
• Radwaste Solidification and Storage Building • 115 KV Switchyard • 345 KV Switchyard • Security West Bldg.
Table 6 Plant Vital Areas
• Reactor Building • Control Room • Turbine Building • Diesel Generator Engine and Board Rooms • Battery Rooms • Battery Board Rooms • Cable Spreading Room • Central Alarm Station • Secondary Alarm Station • Security Uninterruptible Power Supply Room • Telephone Rooms • Main Steam Isolation Valve Room

Proposed Answer: B

Explanation: Fire Zone WD-8131 is a water deluge system for Transformer 1 with automatic initiation capability. The given indications show that this zone has both detected a fire and starting deluge flow. EPIP-EPP-28 defines a confirmed fire as either (1) fire alarm/annunciator and suppression system activation accompanied by actual flow or discharge or (2) reported actual fire with flames showing. The first set of conditions for a confirmed fire is met at time 00:00. EAL 8.2.1 includes a 15 minute fire duration to ensure only fires of a certain magnitude/extent are declared as emergencies. Although the fire is confirmed at 00:00, EAL 8.2.1 is not met until 15 minutes later at 00:15.

A. Incorrect – EAL 8.2.1 includes a 15 minute fire duration to ensure only fires of a certain magnitude/extent are declared as emergencies. Although the fire is confirmed at 00:00, EAL 8.2.1 is not met until 15 minutes later at 00:15.
C. Incorrect – The first set of conditions for a confirmed fire is met at time 00:00.
D. Incorrect – The first set of conditions for a confirmed fire is met at time 00:00.

Technical Reference(s): EPIP-EPP-28, EPIP-EPP-01, EAL Matrix, ARP 2-1, ARP 2-2, OP-21A

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-EPIP28-CE-01

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295016 2.4.8
	Importance Rating	4.5

Incomplete SCRAM

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

Proposed Question: #83

The plant has just scrammed with the following conditions:

- N1-SOP-1, Reactor Scram, is being performed.
- Reactor water level is 25" and rising.
- One control rod is still at position 48.
- All other control rods are full-in.

Which one of the following sets of actions is correct?

A. Enter N1-EOP-2, RPV Control. Continue performing N1-SOP-1. In the event of a conflict between the procedures, N1-EOP-2 is the overriding document.

B. Enter N1-EOP-2, RPV Control. Continue performing N1-SOP-1. In the event of a conflict between the procedures, N1-SOP-1 is the overriding document.

C. Exit N1-SOP-1 and enter N1-EOP-2, RPV Control. N1-SOP-1 is re-entered at the step in-progress after exiting N1-EOP-2.

D. Exit N1-SOP-1 and enter N1-EOP-2, RPV Control. N1-SOP-1 entry conditions are re-evaluated after exiting N1-EOP-2.

Proposed Answer: A

Explanation: N1-SOP-1 is entered based on the occurrence of a Reactor scram. N1-EOP-2 is entered in this case based on Reactor water level less than 53". There is no requirement to exit SOPs when EOPs are entered. In fact, both procedures are executed concurrently. In the case of an incomplete scram without N1-EOP-3 entry, this is particularly important since the guidance for control rod insertion is found in N1-SOP-1, not N1-EOP-2. The Emergency Operating Procedures are higher-tiered documents than the Special Operating Procedures, therefore in the event of a conflict, the EOP must be followed.

B. Incorrect – N1-EOP-2 is the higher-tiered document.
C. Incorrect – N1-SOP-1 is not exited.
D. Incorrect – N1-SOP-1 is not exited.

Technical Reference(s): NER-1M-095, GAI-OPS-20, N1-SOP-1, N1-EOP-2

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP00C01 TO #1

Question Source: Modified – 2008 NRC #97

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	1
	Group #	2
	K/A #	295036 EA2.01
	Importance Rating	3.2

Secondary Containment High Sump/Area Water Level

Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Operability of components within the affected area

Proposed Question: #84

The plant is operating at 80% power when a seismic event occurs. A Torus water leak results in the following:

- Annunciator H2-2-1, R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH, is in alarm.
- Computer point F188 NE RB CORNER RM WTR LVL HIGH is in alarm.
- An operator reports water level in the Northeast Corner Room is 5 feet and slowly rising.

Which one of the following describes the operability of the safety-related pumps in this area?

A. Core Spray Pumps 121 and 122 are inoperable at this time.

B. Containment Spray Pumps 112 and 122 are inoperable at this time.

C. Core Spray Pumps 121 and 122 remain operable until level in this area rises an additional 2 feet.

D. Containment Spray Pumps 112 and 122 remain operable until level in this area rises an additional 2 feet.

Proposed Answer: B

Explanation: Containment Spray Pumps 112 and 122 are in the NE corner room and are the components affected by the water level in the room. The alarm is actuated at a water level of 5 feet in the room, which is the maximum safe value. The max safe value is defined to be the highest value at which equipment necessary for the safe shutdown of the plant will operate. Therefore the components in the area are inoperable.

A. Incorrect – Core Spray Pumps 121 and 122 are in the SE corner room and are not affected.
C. Incorrect – The maximum safe value is already reached. Core Spray Pumps 121 and 122 are in the SE corner room and are not affected.
D. Incorrect – The maximum safe value is already reached.

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

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 EO #2

Question Source: Bank – 2009 NRC #83

Question History: 2009 NRC #83

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 #	295007 AA2.03
	Importance Rating	3.7

High Reactor Pressure

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

Proposed Question: #85

A LOCA has occurred and the following conditions exist:

- All control rods are in.
- Core Spray pumps automatically started and are running.
- ADS is bypassed.
- RPV water level is -85" and lowering.
- RPV pressure is 700 psig and lowering slowly.

Which one of the following describes the appropriate action to be taken?

A. Enter N1-EOP-9, Steam Cooling.

B. Enter N1-EOP-8, RPV Blowdown.

C. Install Core Spray jumpers per N1-EOP-1 Attachment 4.

D. Rapidly depressurize the RPV using Emergency Condensers and Turbine Bypass Valves per N1-EOP-2.

Proposed Answer: B

Explanation: When RPV water level has fallen below -84" and an injection subsystem is lined up enter N1-EOP-8 to blowdown the RPV, lower pressure and allow injection prior to level reaching 109".

A. Incorrect – Steam cooling would only be entered if no injection source was available.
C. Incorrect – This would be done to prevent Core Spray injection not needed for core cooling. With RPV water level below the top of active fuel, Core Spray is needed for core cooling.
D. Incorrect – MSIV have isolated and a blowdown is directed by the EOPs therefore alternate depressurization cannot be used. Additionally, alternate depressurization is contrary to the level control strategy of N1-EOP-2, which is to preserve inventory as long as possible while attempting to lineup high pressure systems before a blowdown is required.

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: Bank - 2008 Audit #60

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	207000 2.4.18
	Importance Rating	4.0

Isolation (Emergency) Condenser

Knowledge of the specific bases for EOPs.

Proposed Question: #86

A plant transient has resulted in the following:

- The Mode Switch is in SHUTDOWN.
- All control rods are fully inserted.
- Reactor water level is unknown.
- N1-EOP-7, RPV Flooding, is in progress.
- Four ERVs have just been opened.

Which one of the following describes how to control the Emergency Condensers and the associated reason, in accordance with N1-EOP-7, RPV Flooding?

A. Place the Emergency Condensers in service to augment Reactor depressurization.

B. Place the Emergency Condensers in service to inject additional water from the condensate return piping.

C. Close the Emergency Condenser steam isolation valves to prevent damage to the Emergency Condensers.

D. Close the Emergency Condenser steam isolation valves to reduce the volume of water required to flood to the Main Steam lines.

Proposed Answer: C

Explanation: N1-EOP-7 contains a series of diagnostic steps surrounding control of ERVs, MSIVs and ECs. All rods are inserted and 4 ERVs have just been opened, therefore N1-EOP-7 step 22 is now in progress. This step asks, "Can any ERV be opened?" Since the answer is yes, N1- EOP-7 step 25 directs closing EC steam IVs. If the answer were no, N1-EOP-7 step 26 may direct placing ECs in service. NER-1M-095 states the basis for N1-EOP-7 step 25 is:

> "Isolating the steam lines avoids:
> - Damage resulting from cold water coming in contact with hot metal as the steam lines are flooded.
> - Excessive loading of lines or hangers not designed to accommodate the weight of water.
> - Flooding of steam-driven equipment."

A. Incorrect – ECs should be isolated. This is the basis for initiating ECs per N1-EOP-8 step 12.
B. Incorrect – ECs should be isolated. This is the basis for initiating ECs per N1-EOP-2 step L-6, or N1-EOP-7 step 26.
D. Incorrect – While isolating ECs does slightly reduce the volume of water required to flood to the Main Steam lines, this is not the correct basis for the EOP step.

Technical Reference(s): N1-EOP-7, NER-1M-095

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP7C01 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 #	2
	Group #	1
	K/A #	215005 2.4.47
	Importance Rating	4.2

APRM / LPRM

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

Proposed Question: #87

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

- APRM 14 is inoperable and bypassed due to a malfunctioning circuit card.
- APRM operability is being reviewed.
- An Operator has recorded inoperable LPRMs on N1-OP-38C Attachment 5 (see next page).

Which one of the following describes the status of APRMs, in accordance with N1-OP-38C and Technical Specifications?

A. No LCO entry is required for APRMs.

B. An LCO entry is required for the APRM scram function, only.

C. An LCO entry is required for the APRM rod block function, only.

D. An LCO entry is required for both the APRM scram and rod block functions.

ATTACHMENT 5: APRM OPERABILITY

CHANNEL 11

		1. Check or (X) inoperable detectors for EACH APRM				2. > than 4 Detectors marked		3. > than 2 C level Detectors marked		4. A & C Detector marked in one RADIAL location	
		28-33	28-49	36-41	44-33	YES	NO	YES	NO	YES	NO
APRM 11	A	X		X							
	C		X		X						
		04-33	12-41	20-33	20-49						
APRM 12	A										
	C										
		04-25	12-17	20-09	20-25						
APRM 13	A			X							
	C			X							
		28-09	28-25	36-17	44-25						
APRM 14	A										
	C										

CHANNEL 12

		1. Check or (X) inoperable detectors for EACH APRM				2. > than 4 Detectors marked		3. > than 2 D level Detectors marked		4. B & D Detector marked in one RADIAL location	
		28-33	28-49	36-41	44-33	YES	NO	YES	NO	YES	NO
APRM 15	B										
	D										
		04-33	12-41	20-33	20-49						
APRM 16	B										
	D										
		04-25	12-17	20-09	20-25						
APRM 17	B										
	D										
		28-09	28-25	36-17	44-25						
APRM 18	B										
	D										

5. A YES answer constitutes a OPERABILITY CONCERN						

Proposed Answer: C

Explanation: APRMs are allowed to have up to 4 LPRMs inoperable and 2 C/D level LPRMs inoperable. Therefore APRM 11 is still operable. Tech Spec Table 3.6.2g Note (c) states, "In the Run mode of operation, bypass of two chambers from one radial core location in any one APRM shall cause that APRM to be considered inoperative." This Table and Note applies to the rod block function only. No similar requirement exists for the scram function. Since APRM 13 has both A and C LPRMs inoperable at radial location 20-09, APRM 13 is inoperable for the rod block function. Tech Spec 3.6.2 would still be satisfied, except that APRM 14 is also inoperable for the rod block function. Since two APRMs in system 11 are inoperable for the rod block function, Tech Spec 3.6.2 is NOT satisfied for the rod block function, and LCO entry is required (LCO 3.6.2.a(7)).

A. Incorrect – Tech Spec 3.6.2 is NOT satisfied for the rod block function.
B. Incorrect – Tech Spec 3.6.2 is NOT satisfied for the rod block function, but is met for the scram function.
D. Incorrect – Tech Spec 3.6.2 is met for the scram function.

Technical Reference(s): N1-OP-38C, Technical Specification 3.6.2

Proposed references to be provided to applicants during examination: Technical Specification 3.6.2 and associated tables

Learning Objective: N1-215000-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 #	206000 A2.04
	Importance Rating	3.0

HPCI

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: A.C. failures: BWR-2,3,4

Proposed Question: #88

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

- Feedwater pump 13 is operating.
- Feedwater pumps 11 and 12 are in standby.

Then, Powerboard 1671C de-energizes due to an electrical fault.

Which one of the following describes the impact of this electrical failure on the High Pressure Coolant Injection (HPCI) system and the associated Technical Specifications?

	Impact on HPCI	Impact on Technical Specifications
A.	One Feedwater pump is inoperable.	Enter a 15 day LCO.
B.	One Feedwater pump is inoperable.	Initiate a normal orderly shutdown.
C.	One Feedwater flow control valve is inoperable.	Enter a 15 day LCO.
D.	One Feedwater flow control valve is inoperable.	Initiate a normal orderly shutdown.

Proposed Answer: A

Explanation: Powerboard 1671C supplies power to the Auxiliary Oil pump for Feedwater pump 12. Feedwater pump 12 will not start unless the Auxiliary Oil pump first starts and provides adequate bearing oil pressure, therefore Feedwater pump 12 is inoperable. Feedwater pump 12 is a redundant component in the HPCI system, therefore TS 3.1.8.b requires entry into a 15 day LCO.

B. Incorrect – The Auxiliary Oil pump for Feedwater pump 11 is powered from Powerboard 1671A, which is separated from Powerboard 1671C by a normally open breaker. Since only one redundant component is inop, TS 3.1.8 does not require an immediate shutdown.
C. Incorrect – Feedwater flow control valves are air-operated, with control power coming normally from RPS 11/12 and alternately from MG Set 167.
D. Incorrect – Feedwater flow control valves are air-operated, with control power coming normally from RPS 11/12 and alternately from MG Set 167.

Technical Reference(s): OP-16, C-19409-C Sheet 1b, C-19424-C Sheet 7, Technical Specification 3.1.8, C-23077-C, C-19844-C Sheet 1

Proposed references to be provided to applicants during examination: Technical Specification 3.1.8

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 #	212000 A2.03
	Importance Rating	3.5

RPS

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

Proposed Question: #89

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

- N1-ST-W15, Manual and Automatic Scram Instrument Channel Test, is in progress.
- An Operator places APRM 15 S-10 switch in STANDBY and then in OPERATE.
- An Operator places APRM 16 S-10 switch in STANDBY and then in OPERATE.
- An Operator places APRM 17 S-10 switch in STANDBY and then in OPERATE.
- An Operator places APRM 18 S-10 switch in STANDBY and then in OPERATE.
- RPS 11 and 12 scram solenoid white lights stay lit through these operations.
- No annunciators or computer points have been received.

Which one of the following describes the significance of these indications and required action?

A. These are the expected indications. Continue N1-ST-W15.

B. A half-scram has failed to occur as expected. Up to 12 hours are allowed before a half scram must be inserted on RPS channel 12.

C. A half-scram has failed to occur as expected. Insert a half scram on RPS channel 12 within 1 hour.

D. A half-scram has failed to occur as expected. Commence a Reactor shutdown per N1-OP-43C, Plant Shutdown.

Proposed Answer: C

Explanation: A half-scram should have been generated when each S-10 switch was taken out of OPERATE. With all four APRM channels to RPS channel 12 not working correctly, Technical Specification 3.6.2 Table A Note (o) requires "Within one hour, verify sufficient channels remain Operable or tripped* to maintain trip capability for the Parameter." Since the stem conditions prove sufficient channels do NOT remain Operable, the RPS channel must be verified tripped within one hour.

A. Incorrect – A half-scram should have been generated when each S-10 switch was taken out of OPERATE.
B. Incorrect – Only one hour is allowed before a half scram must be inserted on RPS channel 12.
D. Incorrect – A Reactor shutdown is NOT required.

Technical Reference(s): N1-ST-W15, Technical Specification 3.6.2

Proposed references to be provided to applicants during examination: Technical Specification 3.6.2 and Table 3.6.2.a

Learning Objective: N1-212000-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(5)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	1
	K/A #	211000 2.2.38
	Importance Rating	4.5

SLC

Knowledge of conditions and limitations in the facility license.

Proposed Question: #90

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

- Liquid Poison (LP) tank volume is 1341 gallons.
- LP tank concentration is 17.9 weight %, with a Boron-10 enrichment of 65.95 atom %.
- LP tank temperature is currently reading 68°F.

Which one of the following describes the operability of the Liquid Poison system and the required action?

A. Inoperable. Restore system operability within seven days.

B. Inoperable. Commence a normal orderly shutdown within one hour.

C. Operable. Direct Chemistry to raise the LP tank volume to 1385 gallons.

D. Operable. Restore tank temperature per Section H.3.0 of N1-OP-12, Liquid Poison.

Proposed Answer: B

Explanation: Per TS 3.1.2.d, and figure 3.1.2.b LP is inop and an orderly shutdown must be commenced within one hour.

A. Incorrect – The LP tank is not a redundant component, therefore when solution chemistry is out of spec, the entire system is inop, requiring the more restrictive S/D spec to apply.
C. Incorrect – Liquid Poison is inop.
D. Incorrect – Liquid Poison is inop.

Technical Reference(s): Technical Specification 3.1.2

Proposed references to be provided to applicants during examination: TS 3.1.2, including Fig 3.1.2.b

Learning Objective: N1-211000-RBO-14

Question Source: Bank – 2008 Audit #89

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	202002 2.2.22
	Importance Rating	4.7

Recirculation Flow Control

Knowledge of limiting conditions for operations and safety limits.

Proposed Question: #91

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

- Four Reactor Recirculation pumps (RRPs) are operating.
- Total Reactor Recirculation flow is 60.7 Mlbm/hr.

Then, one RRP trips, resulting in the following:

- The RRP discharge valve is closed in accordance with N1-SOP-1.3, Recirc Pump Trip at Power.
- Total Reactor Recirculation flow is 46.7 Mlbm/hr.
- Reactor power is 85%.

It is now desired to return the Reactor to the maximum allowed power level.

Which one of the following describes the approximate maximum total Reactor Recirculation flow that is allowed in this configuration, in accordance with Technical Specifications and N1-OP-1?

A. 67.5 Mlbm/hr

B. 60.7 Mlbm/hr

C. 50.4 Mlbm/hr

D. 45.0 Mlbm/hr

Proposed Answer: C

Explanation: Both the pre-trip and post-trip data show Reactor operation approximately on the 108% rod line. Technical Specification 3.1.7.e requires Reactor power to be limited to 90% when operating with only three Reactor Recirculation pumps in service. Using the three loop power-to-flow map and the 108% rod line, 90% power equates to approximately 52.0 Mlbm/hr total Reactor Recirculation flow. However, N1-OP-1 limits the Reactor Recirculation pumps to16.8 mlbm/hr each for a total of 50.4 mlbm/hr.

A. Incorrect – Total Reactor Recirculation flow is limited to approximately 52.0 Mlbm/hr.
B. Incorrect – Total Reactor Recirculation flow is limited to approximately 52.0 Mlbm/hr.
D. Incorrect – Total Reactor Recirculation flow can go higher to approximately 52.0 Mlbm/hr.

Technical Reference(s): Technical Specification 3.1.7; 3, 4, and 5 loop power-to-flow maps

Proposed references to be provided to applicants during examination: 3, 4, and 5 loop power-to-flow maps, Technical Specification 3.1.7

Learning Objective: N1-202001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	201001 A2.14
	Importance Rating	2.8

CRD Hydraulic

Ability to (a) predict the impacts of the following on the CONTROL ROD DRIVE HYDRAULIC SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low drive header pressure

Proposed Question: #92

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

- The following annunciators are received:
  - F3-1-5, CRD CHARGING WTR PRESSURE HI/LO
  - F3-3-3, CRD FILTER DIFF PRESS
- CRD pump 11 is running.
- CRD Charging Water pressure indicates 0 psig.
- CRD Drive Water differential pressure indicates 0 psid.
- Field investigation indicates that the in-service CRD Filter is severely plugged.
- N1-SOP-5.1, Loss of CRD, has been entered.
- Operators are being briefed to swap CRD Filters per N1-OP-5, Control Rod Drive System.

Which one of the following describes (1) a consequence of this failure and (2) the implication for Technical Specification 3.1.6, Control Rod Drive Coolant Injection, until the standby CRD Filter is placed in service?

	(1) Consequence	(2) Technical Specification 3.1.6...
A.	The control rod scram function is non-functional.	is satisfied as long as a CRD pump remains running.
B.	The control rod scram function is non-functional.	requires entry into a shutdown LCO.
C.	Control rod movement with RMCS is unavailable.	is satisfied as long as a CRD pump remains running.
D.	Control rod movement with RMCS is unavailable.	requires entry into a shutdown LCO.

Proposed Answer: D

Explanation: The given pressures indicate that the plugged CRD Filter is so severely plugged that it is restricting flow to approximately 0. The HCU accumulators, as well as Reactor pressure, maintain the motive force for control rod insertion by a scram. However, with no Drive Water differential pressure, there is no motive force for control rod insertion with RMCS. TS 3.1.6 requires the CRD system to be able to provide flow to the Reactor, either through the Charging Water header or the direct return to the Reactor. Since the CRD Filter is blocking both flow paths, TS 3.1.6.c must be entered. This requires Reactor coolant temperature to be reduced to 212°F or less within 10 hours.

A. Incorrect – The HCU accumulators, as well as Reactor pressure, maintain the motive force for control rod insertion by a scram. Since the CRD Filter is blocking both flow paths from the CRD pumps to the Reactor, TS 3.1.6.c must be entered. This requires Reactor coolant temperature to be reduced to 212°F or less within 10 hours.
B. Incorrect – The HCU accumulators, as well as Reactor pressure, maintain the motive force for control rod insertion by a scram.
C. Incorrect – Since the CRD Filter is blocking both flow paths from the CRD pumps to the Reactor, TS 3.1.6.c must be entered. This requires Reactor coolant temperature to be reduced to 212°F or less within 10 hours.

Technical Reference(s): C-18016-C Sheet 1, N1-SOP-5.1, N1-OP-5, TS 3.1.6 and bases

Proposed references to be provided to applicants during examination: TS 3.1.6

Learning Objective: N1-201001-RBO-14

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43(2)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	2
	Group #	2
	K/A #	233000 2.4.31
	Importance Rating	4.1

Fuel Pool Cooling/Cleanup

Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: #93

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

- Annunciator L1-4-5, FUEL POOL ANNUNCIATOR, alarms.
- An Operator in the field reports annunciator SFP-1-2, FUEL POOL PUMPS INLET TEMP, is in alarm.
- The Operator also reports Spent Fuel Pool temperature is 100°F and rising slowly.

Which one of the following describes the most restrictive Spent Fuel Pool temperature limitation that must be observed to avoid violation of license condition?

Spent Fuel Pool temperature must be maintained...

A. 110°F based on Secondary Containment licensing basis.

B. 110°F based on Spent Fuel Pool licensing basis.

C. 140°F based on Secondary Containment licensing basis.

D. 140°F based on Spent Fuel Pool licensing basis.

Proposed Answer: A

Explanation: During power operation, Spent Fuel Pool temperature must be maintained less than 110°F per the licensing basis for the Secondary Containment.

B. Incorrect – The 110°F limit is per the licensing basis for the Secondary Containment.
C. Incorrect – The limit is 110°F since the plant is in the power operating condition.
D. Incorrect – The limit is 110°F since the plant is in the power operating condition.

Technical Reference(s): ARP L1-4-5, ARP SFP-1-2, OP-6, UFSAR Chapter 10

Proposed references to be provided to applicants during examination: None

Learning Objective: N1-233000-RBO-7

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:

Level	SRO
Tier #	3
Group #	
K/A #	2.1.6
Importance Rating	4.8

Ability to manage the control room crew during plant transients.

Proposed Question: #94

A plant transient is in progress with the following:

- An Operator is performing transient response actions in the Auxiliary Control Room.
- The Operator requests another individual come to the Auxiliary Control Room to assist.
- The following individuals are in the Main Control Room:

Individual	Position/Duties
#1	Reactor Operator
#2	Reactor Operator At-The-Controls
#3	Control Room Supervisor
#4	Shift Manager

Which one of the following identifies the individual(s) that may go to the Auxiliary Control room to help without turning over their position/duties, in accordance with CNG-OP-1.01-1000, Conduct of Operations?

A. #1, only.

B. #1 or #4, only.

C. #1, #2, or #4, only.

D. #1, #2, #3, or #4.

Proposed Answer: B

Explanation: CNG-OP-1.01-1000 Attachment 2 defines the areas in which the Reactor Operator At-The-Controls and the Control Room Supervisor must remain. Both individuals are prohibited from going into the Auxiliary Control Room without turning over their duties. There is no restriction keeping the Shift Manager from going to the Auxiliary Control Room.

A. Incorrect – The Shift Manager may also go to the Auxiliary Control Room.
C. Incorrect – The Reactor Operator At-The-Controls may not go to the Auxiliary Control Room.
D. Incorrect – Neither the Reactor Operator At-The-Controls nor the Control Room Supervisor may not go to the Auxiliary Control Room.

Technical Reference(s): CNG-OP-1.01-1000, 10CFR50.54(k)

Proposed references to be provided to applicants during examination: None

Learning Objective: CNG-OP-1.01-1000-CE-01

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(1)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.1.20
	Importance Rating	4.6

Ability to interpret and execute procedure steps.

Proposed Question: #95

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

- A Reactor Operator is performing an Operating Procedure section to restore a piece of equipment following a maintenance activity.
- Due to an abnormal lineup from the maintenance activity, one step in the procedure does not apply and CANNOT be performed.
- The procedure step does not contain a specific allowance to mark the step "N/A".
- The Reactor Operator requests to move on without performing the step.

Which one of the following describes the required direction to the Reactor Operator, in accordance with CNG-PR-1.01-1009, Procedure Use and Adherence Requirements?

A. Process an Editorial Change, including approval by two Senior Reactor Operators.

B. Obtain technical concurrence from a second Reactor Operator, then "N/A" the step and move on in the procedure.

C. Obtain technical concurrence from a Senior Reactor Operator, then "N/A" the step and move on in the procedure.

D. Process a Technical Procedure Step Deletion Screening Form, including approval by a Senior Reactor Operator.

Proposed Answer: D

Explanation: CNG-PR-1.01-1009 provides the guidance for this situation in section 5.5.C (equipment out of service, equipment is not required to be operated, and the procedure does NOT contain a statement authorizing the partial performance). The procedure directs completion of Attachment 1, Technical Procedure Step Deletion Screening Form. This form requires two SROs to approve of the step deletion (First Line Supervisor, which would be an SRO for an Ops procedure, and another SRO).

A. Incorrect – This change does not meet the requirements for use of the Editorial Change process in CNG-PR-1.01-1011 Attachment 4.
B. Incorrect – This situation does not meet the requirements for use of "N/A" per CNG-PR-1.01-1009 section 5.4
C. Incorrect – This situation does not meet the requirements for use of "N/A" per CNG-PR-1.01-1009 section 5.4

Technical Reference(s): CNG-PR-1.01-1009, CNG-PR-1.01-1011

Proposed references to be provided to applicants during examination: None

Learning Objective: CNG-PR-1.01-1009-CT-01

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.2.6
	Importance Rating	3.6

Knowledge of the process for making changes to procedures.

Proposed Question: #96

A change is being processed for N1-OP-16, Feedwater System Booster Pump to Reactor, to alter technical steps in the section for restoring an electric Feedwater pump following maintenance. The procedure section does NOT introduce any temporary plant configuration changes.

Which one of the following describes required reviews for this procedure change, in accordance with CNG-PR-1.01-1011, Station-Specific Procedure Process?

A 10 CFR 50.59 review is __(1)__. Plant Operations Review Committee (PORC) review is __(2)__.

	(1)	(2)
A.	required	required
B.	required	NOT required
C.	NOT required	required
D.	NOT required	NOT required

Proposed Answer: B

Explanation: CNG-PR-1.01-1011 step 5.7.D.2.a(1) requires a 10 CFR 50.59 review be performed since this procedure does NOT have a 10 CFR 50.59 exemption. CNG-PR-1.01-1011 Attachment 12 contains guidance on which changes must receive PORC review. This procedure does not require PORC review because it is not one of the following:

- Emergency Planning Technical Procedures
- Security Technical Procedures
- Any procedure that introduces a temporary plant configuration change that affects nuclear safety

A. Incorrect – PORC review is NOT required.
C. Incorrect – A 10 CFR 50.59 Applicability Determination is required. PORC review is NOT required.
D. Incorrect – A 10 CFR 50.59 Applicability Determination is required.

Technical Reference(s): CNG-PR-1.01-1011

Proposed references to be provided to applicants during examination: None

Learning Objective: CNG-PR-1.01-100-CT-01

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(3)

Comments:

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.3.5
	Importance Rating	2.9

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: #97

An un-isolable leak has developed from Reactor Water Cleanup into the Reactor Building requiring entry into N1-EOP-5, Secondary Containment Control.

Given the following separate sets of conditions:

Condition Number	Condition Description
#1	One Reactor Building Area Radiation Monitor (ARM) indicates upscale.
#2	Two Reactor Building Area Radiation Monitors (ARMs) indicate upscale.
#3	A Radiation Protection Technician reports one side of a Reactor Building elevation has radiation readings up to 10 R/hr.
#4	A Radiation Protection Technician reports both sides of a Reactor Building elevation have radiation readings up to 10 R/hr.

Which one of the following lists the conditions that would require entering N1-EOP-8, RPV Blowdown, in accordance with N1-EOP-5, Secondary Containment Control?

A. #2, only

B. #4, only

C. #2 and #4, only

D. #1, #2, #3, and #4

Proposed Answer: B

Explanation: With an un-isolable system discharging in the Reactor Building, EOP-5 directs transition to EOP-8 when the Maximum Safe radiation level of 8 R/hr is exceeded in two or more General Areas. A General Area is described has one half of a Reactor Building Elevation, as divided by the fire break zones. The Reactor Building ARMs only read to 1000 mR/hr, and are therefore CANNOT be used to determine if 8 R/hr has been exceeded. This determination can only be made based on evidence from a portable survey instrument.

A. Incorrect – The Reactor Building ARMs only read to 1000 mR/hr, and are therefore CANNOT be used to determine if 8 R/hr has been exceeded.
C. Incorrect – The Reactor Building ARMs only read to 1000 mR/hr, and are therefore CANNOT be used to determine if 8 R/hr has been exceeded
D. Incorrect – The Reactor Building ARMs only read to 1000 mR/hr, and are therefore CANNOT be used to determine if 8 R/hr has been exceeded

Technical Reference(s): EOP-5, GAI-OPS-20

Proposed references to be provided to applicants during examination: None

Learning Objective: 1101-EOP5C01 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 #	3
	Group #	
	K/A #	2.3.4
	Importance Rating	3.7

Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: #98

An emergency is in progress. A worker has volunteered to receive an emergency exposure to SAVE A LIFE.

Which one of the following is the emergency dose that you can authorize as the SM/ED, in accordance with EPIP-EPP-15, Emergency Health Physics Procedure?

A. Up to 10 Rem, MINUS the worker's current occupational exposure.

B. Up to 10 Rem, EXCLUSIVE of the worker's current occupational exposure.

C. Up to or above 25 Rem, MINUS the worker's current occupational exposure.

D. Up to or above 25 Rem, EXCLUSIVE of the worker's current occupational exposure.

Proposed Answer: D

Explanation: Emergency exposure limits are EXCLUSIVE of the workers current occupational exposure. The limit for protecting valuable property is 10 Rem and for lifesaving is 25 Rem. The 25 Rem limit may be exceeded in this case since the worker is a volunteer.

A. Incorrect – Emergency exposure limits are EXCLUSIVE of the workers current occupational exposure. The limit for lifesaving is 25 Rem, but this may even be exceeded since the worker is a volunteer.
B. Incorrect – The limit for lifesaving is 25 Rem, but this may even be exceeded since the worker is a volunteer.
C. Incorrect – Emergency exposure limits are EXCLUSIVE of the workers current occupational exposure.

Technical Reference(s): EPIP-EPP-15

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-EPL001-EPP09-TO-01

Question Source: Bank - 2009 Audit #96

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments:

Q #99 Redacted;

contains Sensitive Information

-Not for Public Disclosure -

Examination Outline Cross-Reference:	Level	SRO
	Tier #	3
	Group #	
	K/A #	2.4.44
	Importance Rating	4.4

Knowledge of emergency plan protective action recommendations.

Proposed Question: #100

Which one of the following describes when Protective Action Recommendations (PARs) are made to the County and State, and the size of the area that is recommended for evacuation?

PARs are made to the County and State when the emergency classification is ___(1)___.
PARs are made for the area two miles around and ___(2)___ miles downwind from the plant.

	(1)	(2)
A.	General Emergency, only	five
B.	General Emergency, only	ten
C.	Site Area Emergency and General Emergency	five
D.	Site Area Emergency and General Emergency	ten

Proposed Answer: A

Explanation: PARs are only made at the General Emergency classification level. PARs are made for 2 miles around and 5 miles downwind from the plant.

B. Incorrect – PARs are only made for 5 miles downwind.
C. Incorrect – PARs are not made at the Site Area Emergency level.
D. Incorrect – PARs are not made at the Site Area Emergency level. PARs are only made for 5 miles downwind.

Technical Reference(s): EPIP-EPP-08

Proposed references to be provided to applicants during examination: None

Learning Objective: NS-EPIP08-CE-01

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43(4)

Comments: