EXAMINATION ANSWER KEY

15-1 NRC RO EXAM

1	ID: 1263894	Points: 1.00

Unit 1 was operating at 65% power when the 1A Reactor Recirc Pump tripped.

- Reactor Power is now at 45%.

The IMMEDIATE drop in reactor power is the result of …

A. core void fraction RISING.

B. total Steam Flow LOWERING.

C. Feedwater Heater outlet temperature RISING.

D. reactor pressure LOWERING before STABILIZING.

Answer: A

Answer Explanation
Explanation: As total core flow lowers, the coolant's ability to cool the core lowers and void fraction rises. When void fraction rises, moderator density lowers and neutron moderation becomes less efficient. This results in more resonance capture and in greater neutron leakage from the core. Thus, void coefficient adds negative reactivity and reactor power lowers. **Distractor 1** is incorrect: Steam flow lowers due to the drop in reactor power as the steam cycle responds to the transient. Plausible because total steam flow is expected to lower; however, lowering steam flow is the result of reactor power lowering and not its cause. **Distractor 2** is incorrect: Void coefficient adds negative reactivity and reactor power lowers causing steam flow to lower and Feedwater temperature to drop rather than rise. Plausible because rising Feedwater temperature causes reactor power to lower. Candidate must recognize that Feedwater temperature would lower rather than rise. **Distractor 3** is incorrect: Void coefficient adds negative reactivity and reactor power lowers causing reactor pressure to drop; however, lowering reactor pressure is the result of reactor power lowering and not its cause. DEHC then responds to stabilize reactor pressure. Plausible because if it were the initiating condition, lowering reactor pressure would raise void fraction adding negative reactivity causing reactor power to drop. **Reference:** UFSAR 4.3.2.3, 4.3.2.3.1 **Reference provided during examination:** N/A **Cognitive level:** Memory **Level (RO/SRO):** RO **Tier:** 1 **Group:** 1 **295001 Partial or Complete Loss of Forced Core Flow Circulation** **K3.02 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:** Reactor power response **10 CFR Part 55 Content: 41.5** ***SRO Justification:*** *N/A* **Question Source:** LaSalle Bank **Question History:** 2009 LORT Annual Exam **Comments:** **Associated objective(s):**

ID: 1433052 **Points: 1.00**

Unit 1 is operating at 100 % power:

- Annunciator 1PM01J-A214, 4KV BUS 141X/Y OVERCURRENT is in alarm.
- Breaker indicating lights at 1PM01J indicate that breakers 1411 and 1415 are OPEN.
- 4 KV BUS 141X is Locked Out (86 device tripped).
- Relay targets indicate a ground fault on the Bus.

Which of the following actions allow 4 KV Bus 141X to be reenergized while protecting equipment and ensuring personal safety?

A. Reset the lockout relay ONLY.

B. Verify all 3 phases of 141X voltage read approximately the same AND the lockout relay has been reset.

C. Verify annunciator 1PM01J-A214, 4KV BUS 141X/Y OVERCURRENT has cleared AND the lockout relay has been reset.

D. Verify the ground fault has been isolated AND the lockout relay has been reset.

Answer: D

Answer Explanation

Explanation: The relay targets provided indicate that there could be an existing ground fault. To protect equipment and ensure personal safety, the Bus should not be reenergized until the cause has been corrected or isolated from the Bus; additionally, the lockout relay must be reset.

Distractor 1 is incorrect: This would potentially restore power to a faulted Bus. Plausible because the lockout relay must be reset to reenergize the Bus.

Distractor 2 is incorrect: The ground fault must be isolated prior to reenergizing the Bus. Distractor is plausible because per LOA-AP-101, all 3 phases of 141X voltage are verified to read approximately the same to rule out blown Bus pot fuses.

Distractor 3 is incorrect: Because annunciator 1PM01J-A214 will CLEAR when the over current relay targets have been reset, there could be a ground fault existing on the bus. Per LOA-AP-101, to protect equipment and ensure personal safety the Bus should not be reenergized until the cause has been corrected or isolated from the Bus. Distractor is plausible because annunciator 1PM01J-A214 will CLEAR when the over current relay targets have been reset and the lockout relay must be reset to reenergize the Bus.

Reference: LOR-1PM01J-A214, Revision 1 and LOA-AP-101, Revision 53, Section B.8
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 1 **Group:** 1
K/A: 295003 Partial or Complete Loss of A.C. Power
K3.04 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Ground isolation
10 CFR Part 55 Content: 41.5
SRO Justification: *N/A*
Question Source: New
Question History: N/A

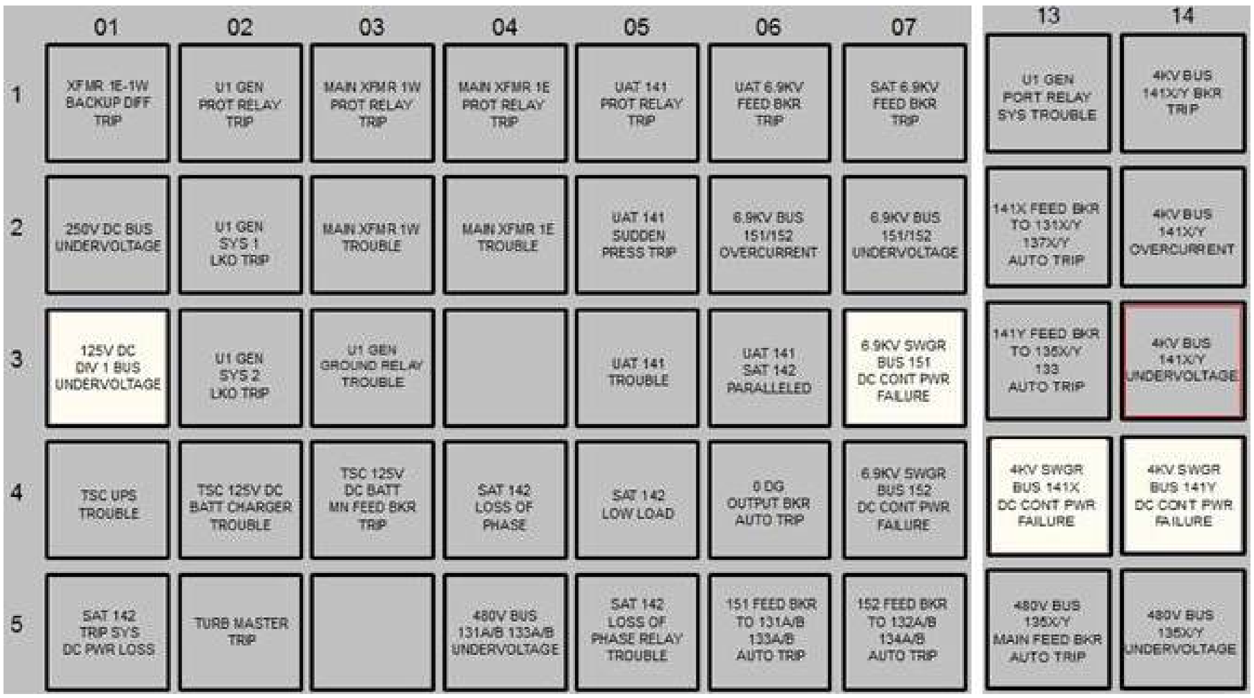
Comments:

Associated objective(s):

3	ID: 1455126	Points: 1.00

Unit 1 and 2 are operating at 100% power:

- The following alarms are received at 1PM01J:



- NO additional alarms are received at 1PM01J.
- IMMEDIATELY following the alarms, 125 V Division One (1) BATTERY voltage at 1PM01J indicates ZERO (0) Volts.

What is the condition of the Division 1 125V DC electrical distribution system **AND** the correct operator action?

A. 125 V Division 1 BUS 1A has been lost. Manually SCRAM the reactor.

B. The on-line Division 1 Battery Charger has been lost. Manually SCRAM the reactor.

C. 125 V Division 1 BUS 1A has been lost. Place the alternate Division 1 Battery Charger in service.

D. The on-line Division 1 Battery Charger has been lost. Place the alternate Division 1 Battery Charger in service.

Answer: A

Answer Explanation
Explanation: Per LOA-DC-101, the alarms provided coupled with a loss of 125 V Division One (1) BATTERY voltage are indicative of a loss of 125V DC Division 1 Bus 1A. When 125V DC Division 1 Bus 1A is lost, a manual reactor Scram is required. **Distractor 1** is incorrect: Additional alarms would be expected at 1PM01J AND at 1H13-P601 for a loss of division 1 battery charger, including battery charger trouble alarms. Plausible because a prolonged loss of the division 1 battery charger would eventually result in a loss of 125V DC Division 1 Bus 1A for which a manual SCRAM is required. **Distractor 2** is incorrect. Plausible because the alarms provided coupled with a loss of 125 V Division One (1) BATTERY voltage are indicative of a loss of 125V DC Division 1 Bus 1A, and a prolonged loss of the 125 V Division One (1) BATTERY Charger would eventually result in a loss of 125V DC Division 1 Bus 1A. Also, a loss of the division 1 battery charger could require the crew to place the alternate Division 1 Battery Charger in service. **Distractor 3** is incorrect: Additional alarms would be expected at 1PM01J AND at 1H13-P601 for a loss of division 1 battery charger, including battery charger trouble alarms. Distractor is plausible because a loss of the division 1 battery charger could require the crew to place the alternate Division 1 Battery Charger in service. **Reference:** LOA-DC-101, Revision 20, UNIT 1 DC Power System Failure **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** RO **Tier:** 1 **Group:** 1 **K/A: 295004 Partial or Complete Loss of D.C. Power** **A1.01 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:** D.C. electrical distribution systems **10 CFR Part 55 Content: 41.7** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

4	ID: 1271596	Points: 1.00

Unit 1 is operating at 15% of rated power:

- Main Turbine Generator is synched to the grid.
- DEHC is in dome pressure mode and reactor pressure is stable at 935 psig.

A Generator Differential Current condition results in a Main Turbine Generator trip signal.

- Reactor power is at 15% and stable.
- RPV level is 36 inches and steady.
- All Turbine Stop Valves indicate CLOSED.
- All Turbine Control Valves indicate CLOSED.
- All Turbine Intermediate Stop Valves indicate CLOSED.
- All Turbine Intercept Valves indicate CLOSED.
- Four turbine bypass valves indicate full OPEN.

RPV pressure is 900 psig and lowering.

Interpret the current reactor pressure response and whether or not a manual SCRAM is required.

A. Reactor pressure will continue lowering. Insert a manual reactor scram.

B. Reactor pressure will soon stabilize. Do NOT insert a manual reactor scram.

C. Reactor pressure will STOP lowering and begin to RISE. Insert a manual reactor scram.

D. Reactor pressure will STOP lowering and begin to RISE. Do NOT insert a manual reactor scram.

Answer: A

Answer Explanation

Explanation: The turbine bypass valves are not controlling reactor pressure; therefore, a reactor scram is required in accordance with LOA-TG-101, Unit 1 Turbine Trip. At 15% reactor power and the Main Turbine tripped with a minimal reactor water level transient, only 3 Turbine Bypass Valves should be open passing approximately 15% of rated steam flow at approximately 935 psig as in the initial conditions; however, greater than three bypass valves are open, and reactor pressure is lowering and will continue to lower. When the bypass valves are not maintaining reactor pressure, LOA-TG-101 directs the crew to insert cram rods or manually scram the reactor, as required. In this case, inserting control rods would only serve to increase the reactor power to bypass valve position mismatch; specifically, with RPV pressure lowering due to the bypass valve failure, if reactor power is also lowered due to rod insertion, RPV pressure would lower at an accelerating rate leading to a Group 1 isolation and an automatic reactor scram; therefore, the only acceptable operator response is to insert a manual reactor scram.

Distractor 1 is incorrect: pressure will continue to lower and would lower at an accelerating rate if control rods were inserted; however, the distractor is plausible because if the candidate concludes RPV pressure will stabilize, it is reasonable to also conclude that a scram is not required.

Distractor 2 is incorrect: pressure will continue to lower and would lower at an accelerating rate if control rods were inserted; however, the distractor is plausible because the candidate may conclude a scram is required as a result of the turbine trip. Also, if the candidate concludes that the turbine has not properly tripped; a manual scram would be required.

Distractor 3 is incorrect: pressure will continue to lower and would lower at an accelerating rate if control rods were inserted; however, the distractor is plausible because a manual scram is required when the main turbine trips above 25% power regardless of reactor pressure response; therefore, the candidate may conclude that with power below 25%, a scram is not required.

Reference: LOA-TG-101(201), Rev. 17, Unit 1 Turbine Trip and LOA-EH-101, Rev 35. UNIT 1 EHC ABNORMAL

Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): RO

Tier: 1 **Group:** 1

K/A: 295005 Main Turbine Generator Trip

AA2.04 Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Reactor pressure

10 CFR Part 55 Content: 41.10

SRO Justification: *N/A*

Question Source: New

Question History: N/A

Comments:

Associated objective(s):

5 **ID: 1454978** **Points: 1.00**

Unit 2 was operating at rated conditions when a Generator Load Reject Turbine Trip caused a Reactor Scram.

IMMEDIATELY following the scram:

- 2 control rods remain full out at position 48.
- ALL scram solenoid lights are EXTINGUISHED.
- ALL scram pilot valves are open.
- APRM Downscale lights are LIT.

What immediate action will the NSO take first?

A. Initiate ARI.

B. Initiate SBLC.

C. TRIP one TDRFP and CLOSE its discharge valve.

D. Insert Control Rods per LGA-NB-01, Alternate Rod Insertion.

Answer: A

Answer Explanation
Explanation: Per LGP-3-2, Attachment E-1 and OP-LA-103-102-1002, Initiate ARI Distractor 1 is incorrect: SLC is not initiated because reactor power is below 3%. The distractor is plausible because it is an immediate action Per LGP-3-2 Attachment E-1. **Distractor 2** is plausible because tripping one TDRFP and closing its discharge valve is an immediate action Per LGP-3-2 Attachment E-1. **Distractor 3** is plausible because inserting Control Rods per LGA-NB-01 is an immediate action Per LGP-3-2 Attachment E-1. **References:** LGP-3-2, Rev 72, Reactor Scram and OP-LA-103-102-1002, Rev 16, Strategies for Successful Transient Mitigation **Reference provided during examination:** N/A **Cognitive level:** Memory **Level (RO/SRO):** RO **Tier:** 1 **Group:** 1 **K/A: 295006 SCRAM** **Generic 2.4.11: Knowledge of abnormal condition procedures.** **10 CFR Part 55 Content: 41.10** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

6	ID: 1252810	Points: 1.00

A severe fire forced an immediate evacuation of the control room.

What is the method of scramming the reactor from outside of the control room per LOA-RX-101, UNIT 1 Control Room Evacuation Abnormal?

A. Vent the scram air header in the Reactor Building.

B. Activate the ARI system in the Auxiliary Equipment Room.

C. Open the RPS output breakers at the RPS distribution panel.

D. Open and reclose the APRM supply breakers at the RPS distribution panel.

Answer: C

Answer Explanation

Explanation: "Open the RPS output breakers at the RPS distribution panel." is correct. Distracters are credible but are not in accordance with the referenced procedure.

Distractor 1 is plausible because venting the scram air would result in control rod insertion.

Distractor 2 is plausible because activating ARI in the Aux Equipment room would result in control rod insertion.

Distractor 3 is plausible because opening and reclosing the APRM supply breakers at the RPS distribution panel would result in control rod insertion.

Reference: LOA-RX-101, Revision 10, UNIT 1 UNIT 1 Control Room Evacuation Abnormal
Reference provided during examination: N/A

Cognitive level: Memory
Level (RO/SRO): RO
Tier: 1 **Group:** 1

K/A: 295016 Control Room Abandonment
K2.02 Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Local control stations: Plant-Specific

10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*

Question source: New
Question History: N/A

Comments:

Associated objective(s):

7	ID: 1455128	Points: 1.00

Unit 1 is operating at 100% reactor Power.
Blockage in the Drywell RBCCW piping has resulted in a complete stoppage of all RBCCW flow to the 1A Recirc pump ONLY.

What is the effect on the 1A Recirc pump, and if RBCCW flow is NOT restored, what is/are the required operator action(s)?

Damage to the Recirc pump motor…

A. bearings ONLY will occur.
Scram the reactor and trip the 1A Recirc pump.

B. bearings ONLY will occur.
Trip the 1A Recirc pump; a reactor scram is NOT required.

C. bearings AND windings will occur.
Scram the reactor and trip the 1A Recirc pump.

D. bearings AND windings will occur.
Trip the 1A Recirc pump; a reactor scram is NOT required.

Answer: D

Answer Explanation

Explanation: Damage to the Recirc Pump motor bearings and motor windings will result due to rising temperatures; therefore, a Recirc Pump trip is required. If any Recirc Pump motor winding temperatures limits are reached (248° continuous or 266° Intermittent), the Recirc pump must be tripped; additionally, if Recirc Pump bearing temperature limits are exceeded (220F/210F), the affected Recirc Pump is tripped. A reactor scram is not required in the event of a single Recirc Pump trip.

Distractor 1 is incorrect: Damage to the pump motor windings will also occur and a scram is NOT required. The distractor is plausible because damage to the motor bearings will occur and because Recirc Pump trip will be required due to rising bearing temperatures if RBCCW cooling is not restored. Additionally, the distractor is plausible because high Recirc Pump motor bearing and motor winding temperatures are addressed in separate procedures. High winding temperature is addressed in LOR-1H13-P602-A304, and high bearing temperature is addressed in LOR-1H13-P602-B507; furthermore, neither high winding temperature nor high bearing temperature are addressed directly in LOA-WR-101, Loss of Reactor Building Closed Cooling Water. Additionally, the distractor is plausible because on a loss of WR cooling to the windings, forced air cooling remains available.

Distractor 2 is incorrect: Damage to the pump motor windings will also. The distractor is plausible because damage to motor bearings will occur if RBCCW cooling is not restored and because a reactor scram is not required in the event of a single Recirc Pump trip. Additionally, the distractor is plausible because high Recirc Pump motor bearing and motor winding temperatures are addressed in separate procedures. High winding temperature is addressed in LOR-1H13-P602-A304, and high bearing temperature is addressed in LOR-1H13-P602-B507; furthermore, neither high winding temperature nor high bearing temperature are addressed directly in LOA-WR-101, Loss of Reactor Building Closed Cooling Water. Additionally, the distractor is plausible because on a loss of WR cooling to the windings, forced air cooling remains available.

Distractor 3 is incorrect: A reactor scram is NOT required. Distractor is plausible because damage to motor bearings and motor windings will occur due to rising bearing and winding temperatures if RBCCW cooling is not restored.

Reference: LOA-WR-101, Rev. 14, Loss of Reactor Building Closed Cooling Water, LOR-1H13-P602-B507, Rev 19, Reactor Recirc Pump1A/1B Temperature High and LOR-1H13-P602-A304, Rev 7, 1A Recirc Pump Motor Cooling Water Flow Low

Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): RO

Tier: 1 **Group**: 1

K/A: 295018 Partial or Total Loss of Component Cooling Water

AK1.01 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Effects on component/system operations

10 CFR Part 55 Content: 41.7

SRO Justification: N/A

Question Source: New

Question History: N/A

Comments:

Associated objective(s):

8	ID: 1433120	Points: 1.00

Which of the following describes a correct lineup for Reactor Water Cleanup in the decay heat removal mode?

A. One Reactor Water Cleanup pump, “A” Regenerative Heat Exchanger, “A” Nonregenerative Heat Exchanger, "B" Regenerative Heat Exchanger, Reactor Pressure Vessel

B. One Reactor Water Cleanup pump, “A” Regenerative Heat Exchanger, “A” Nonregenerative Heat Exchanger, "A" Regenerative Heat Exchanger, Reactor Pressure Vessel

C. Both Reactor Water Cleanup pumps, “A” Regenerative Heat Exchanger, “B” Nonregenerative Heat Exchanger, "A" Regenerative Heat Exchanger, Reactor Pressure Vessel

D. Both Reactor Water Cleanup pumps, “A” Regenerative Heat Exchanger, “B” Nonregenerative Heat Exchanger, "B" Regenerative Heat Exchanger, Reactor Pressure Vessel

Answer: A

Answer Explanation
Explanation: Shutdown cooling is a mode of operation of the A RHR and B RHR systems. When shutdown cooling is lost and cannot be restored, alternate decay heat removal systems must be used per LOA-RH-101. The Reactor Water Cleanup (RWCU) system can be utilized in the decay heat removal mode as an alternate method of shutdown cooling per LOP-RT-13. This mode of operation of RWCU is only used to support alternate decay heat removal and uses a system lineup that is not used in any other mode of operation of the Reactor Water Cleanup system; specifically, flow is through the RHX tube side, the NRHX, bypassing the demins, and then through the RHX shell side of the other RWCU train. **Distractor 1:** RT pump, “A” RHX, “A” NRHX, “A” RHX, reactor - This flowpath will send flow through the same RHX. Plausible because it is also the normal system lineup, but it does not maximize decay heat removal. **Distractor 2:** Both RT pumps, “A” RHX, “B” NRHX, “A” RHX, reactor - Cannot use A RHX HX with B NRHX. Use only one RT pump. Plausible because LOP-RT-13 directs flow through both trains of heat exchangers; additionally, running both pumps could result in a greater flowrate maximizing heat removal. **Distractor 3:** Both RT pumps, “A” RHX, “B” NRHX, “B” RHX, reactor - Cannot use A RHX HX with B NRHX. Use only one RT pump. Plausible because running both pumps could result in a greater flowrate maximizing heat removal. **Reference:** LOA-RH-101, Rev 19, UNIT 1 RHR ABNORMAL and LOP-RT-13, Rev 17, RWCU LINEUP FOR HEAT REMOVAL **Reference provided during examination:** N/A **Cognitive level:** Memory **Level (RO/SRO):** RO **Tier:** 1 **Group:** 1 **KA: 295021 Loss of Shutdown Cooling** **A3.01 - Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING : A1.01 Reactor water cleanup system** **10 CFR Part 55 Content: 41.7** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

9	ID: 1455150	Points: 1.00

Unit 1 is in a refueling outage with the Refueling Cavity flooded and core alterations in progress:

- Annunciator 1PM13J-B303, DW Equipment Drain Sump Trouble is in alarm.
- R0539, RX Vessel Drywell Leak is in alarm on the SER.
- Skimmer Surge Tank is lowering.
- 1A Drywell Equipment Drain Sump Pump is RUNNING.

Which of the following is the cause for the given conditions?

A leak in the…

A. Fuel Pool Gate

B. Refueling Bellows Seal

C. Transfer Canal leakage

D. Cask Well Annulus Drain

Answer: B

Answer Explanation

Explanation: Loss of Refueling Cavity inventory is a potentially serious refueling accident, as evidenced by the first step of LOA-FC-101 Section B.4, which is to evacuate the Drywell. The indications provided are symptomatic of a leak into the Drywell, and the Refueling Bellows Seal is the only potential leak source that indicates drywell leakage.

Distractor 1: The FP Gate leakage flow switch is only functional when the gates are installed and must be isolated when they are removed for core alterations. Also, the area between the gates and the leakage detection both drain to the RBEDT. Plausible because the FP Gates leaking would cause Skimmer Tank level to lower.

Distractor 2: Transfer canal leakage would result in annunciator 1H13-P601-C207 and lowering Skimmer Surge Tank level but not the leakage into the DWEDS or computer alarm point R0539. Plausible because the Refueling Bellows leaking would give these indications.

Distractor 3: Plausible because the Cask Well Annulus Drain is monitored for leakage.

Reference: LOR-1H13-P601-C207; LOA-FC-101; LOR-1PM13J-B303
P&ID M-98 Sh 1&4; P&ID M-91 Sh 4
Reference provided: N/A

Cognitive level: Higher (2-DR)

Level (RO/SRO): RO Tier: Group: Tier 1 Group 1
K/A Number and Statement: 295023 Refueling Accidents, AA2.02, 3.4/3.7
Ability to determine and/or interpret the following as they apply to Refueling Accidents: Fuel Pool level

Question Source: Bank
Question History: 13-1 Cert

10 CFR Part 55 Content: N/A
SRO Justification: N/A

Comments:
Associated objective(s):

10	ID: 1454980	Points: 1.00

- Unit 1 has scrammed
- The crew has entered LGA-003, Primary Containment Control, on High Drywell pressure.
- It becomes necessary to perform LGA-VQ-102, Emergency Containment Vent.

What is the purpose of venting the containment under these conditions?

To maintain primary containment pressure…

A. less than the SRV Tail Pipe Level Limit.

B. within the Drywell Spray Initiation Limit.

C. less than the Pressure Suppression Pressure.

D. less than the Primary Containment Pressure Limit.

Answer: D

Answer Explanation

Explanation: Maintain containment below 60 psig, which is the upper limit of Primary Containment Pressure Limit (PCP), is the reason for using LGA-VQ-102. To limit off-site release rates, venting is limited to only as necessary to stay below the PCPL.

Distractor 1 is plausible because the basis of the SRV Tail Pipe Limit is to limit SRV actuation dynamic loads to within the Suppression Chamber wall load limits preserving primary containment integrity. The strategy to stay below the SRV tail pipe limit includes lowering pressure. The distractor is incorrect because reactor pressure is lowered rather than drywell pressure.

Distractor 2 is plausible because the Drywell Spray Initiation Limit is expressed as a function of Drywell temperature and Drywell pressure. The conditions of the curve must be met in order to safely spray the drywell which is a key strategy to limit drywell pressure in an event. This distractor is incorrect because the Drywell Spray Initiation Limit is limiting in a high temperature and low pressure condition and would be satisfied at high drywell pressures as described in the stem.

Distractor 3 is plausible because the PSP limit is expressed as a function of Suppression Pool water level and Suppression Chamber pressure. This curve is based on the design limits of the Suppression Chamber wall and preserving the pressure suppression function of Primary Containment. The distractor is incorrect because the strategy to stay below the PSP is to blowdown. Venting per LGA-VQ-102 is not appropriate at this pressure.

Reference: LGA-VQ-102, Rev 0, Unit 1 Emergency Containment Vent

Reference provided during examination: N/A

Cognitive level: memory

Level (RO/SRO): RO

Tier: 1 **Group:** 1

295024 High Drywell Pressure:

EK1.01 Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE: Drywell integrity: Plant-Specific

10 CFR Part 55 Content: 41.8

SRO Justification: *N/A*

Question Source: Bank

Question History: 11-1 LaSalle NRC ILT Exam

Comments:

Associated objective(s):

11 **ID: 1262265** **Points: 1.00**

Unit 1 has scrammed after an extended full power run:

- Reactor level is 35 inches and being maintained by Feedwater.
- Reactor pressure is 935 psig and being maintained by the bypass valves.
- 1A and 1C Circulating Water (CW) pumps have tripped on the reactor SCRAM.
- Bus 142X-142Y Tie ACB 1425 TRIPS on Overcurrent.

An NSO has taken the appropriate actions per LOA-CW-101, Unit 1 Circulating Water Systems Abnormal.

If NO additional operator action is taken, what will be the initial response of RPV pressure?

A. Remain steady at 935 psig

B. Lower to a value below 935 psig

C. Rise UNTIL RPV pressure turns at 1076 psig

D. Rise UNTIL RPV pressure turns at 1096 psig

Answer: C

Answer Explanation
Explanation: This pressure is the setpoint at which "S" and "U" SRVs will open due to the MSIVs being closed from the complete loss of CW Pumps. When BKR 1425 Trips, the 1B CW pump loses power, the NSO also reports that 1A and 1C CW pumps have tripped. This is a complete loss of circulating water and the Condenser is no longer available for Decay heat removal. The MSIV's were closed Circulating water pumps were lost. With the reactor SCRAM and the MSIVs closed following an extended full power run, decay heat causes reactor temperature and pressure to rise until "S" and "U" SRVs open. **Distractor 1** is incorrect: Following an extended full power run, decay heat causes reactor temperature and pressure to rise with no CW pumps running and the MSIVs closed. The answer is plausible because if the candidate fails to recognize that the B CW pump does not have power and that the MSIVs are closed, then DEHC would continue to maintain pressure at 925 psig utilizing the bypass valves. **Distractor 2** is incorrect: Following an extended full power run, decay heat causes reactor temperature and pressure to rise with no CW pumps running and the MSIVs closed. The answer is plausible because if the candidate fails to recognize that the B CW pump does not have power and that the MSIVs are closed, reactor pressure would eventually lower as energy is removed in the main condenser. **Distractor 3** is incorrect: Following an extended full power run, decay heat causes reactor temperature and pressure to rise with no CW pumps running and the MSIVs closed until SRVs lift at 1076 psig. The answer is plausible because if the candidate recognizes that the SRVs will cycle because the B CW pump does not have power and that the MSIVs are closed, 1096 is a valid SRV Lift setpoint. **Reference:** LOA-SRV-101, Unit 1 Stuck open Safety Relief Valve, Rev 8 LOA-CW-101, Unit 1 Circulating Water Systems Abnormal, Rev 20 **Reference provided during examination**: None **Cognitive level:** High **Level (RO/SRO):** RO **Tier: 1 Group: 1** **K/A: 295025High Reactor Pressure** **EK1.04 - Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: Decay heat generation.** **10 CFR Part 55 Content: 41.7** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** None **Associated objective(s):**

12 ID: 1454888 Points: 1.00

LGA-003, Primary Containment Control, has been entered on Unit 1.

- Suppression pool is – 2 feet and lowering

Which of the following parameter combinations meets the Heat Capacity Temperature Limit (HCTL)?

A. 600 psig
210°F

B. 700 psig
195°F

C. 800 psig
200°F

D. 900 psig
170°F

Answer: D

Note
Applicant Question: does meet mean at the limit, above the limit, or higher than the limit?
Response Given: would be evaluated as acceptable.

Answer Explanation

Explanation: This question requires the candidate to understand the relationship between RPV pressure, Suppression Pool temperature and Suppression Pool water level to determine which HCTL curve to utilize and having determined the correct curve, to determine the safe area of the HCTL graph for the conditions provided. Then, the candidate must to accurately plot points on the graph as determined by reactor pressure and Suppression Pool temperature to determine which parameter set satisfies the requirements of the HCTL curve. Specifically, with a Suppression Pool water level of – 2 feet, 900 psig RPV pressure at 170°F Suppression Pool temperature is the only parameter set that maintains the HCTL. This combination of RPV pressure and Suppression Pool temperature are below the HCTL curve for abnormal Suppression Pool water level that is in effect when Suppression Pool level outside the normal band of + 5 feet to -1 feet. The combination of 900 psig RPV pressure and 170°F Suppression Pool temperature meets this criterion.

Distractor 1 is incorrect but plausible because 600 psig and 210°F is located in the region just above the HCTL curve for a Suppression Pool level in the normal band of + 5 feet to -1 feet.

Distractor 2 is incorrect but plausible because but plausible because 700 psig and 195°F is located in the region below the HCTL curve for a Suppression Pool level in the normal band of + 5 feet to -1 feet.

Distractor 3 is incorrect but plausible because 800 psig and 200°F is on the HCTL curve for abnormal Suppression Pool level outside the normal band of + 5 feet to -1 feet.

Reference: LGA-003, Primary Containment Control, Rev. 16, Figure H, HCTL and LPGP-PSTG-01S05A, Attachment A, G.5 Page 61
Reference provided: LGA-003, Figure H, HCTL
Cognitive level: High
Level (RO/SRO): RO
Tier: 1 **Group:** 1
K/A: 295026 Suppression Pool High Water Temperature
A2.03 Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE: Reactor pressure: Mark-II.
10 CFR Part 55 Content: 41.10
SRO Justification: N/A
Question Source: New
Question History: N/A
History: N/A
Comments:

Associated objective(s):

13 **ID: 1454935** **Points: 1.00**

In the event of a total loss of Instrument Air header pressure, which of the following describes the effect on the TBCCW and RBCCW systems?

RBCCW heat exchanger outlet temperature will __(1)__.
TBCCW heat exchanger outlet temperature will __(2)__.

A. (1) RISE
(2) LOWER

B. (1) RISE
(2) RISE

C. (1) LOWER
(2) RISE

D. (1) LOWER
(2) LOWER

Answer: D

Answer Explanation

Explanation: TBCCW and RBCCW heat exchanger outlet temperatures would lower. Service water flow to the TBCCW and RBCCW heat exchangers would rise lowering heat exchanger outlet temperature, because the Temperature Control Valves (WS029, WS087 A, B & C) fail open.

Distractor 1 is plausible because RBCCW outlet temperature would rise as a result of its Service Water TCV failing closed. Also plausible because the majority of the essential and nonessential air operated valves identified in LOA-IA-101, Attachment A and B fail closed. Also, the TBCCW TCVs does fail open, lowering TBCCW heat exchanger outlet temperature.

Distractor 2 is plausible because the majority of the essential and nonessential air operated vales identified in LOA-IA-101, Attachment A and B fail closed, and if the Service Water TCVs for RBCCW and TBCCW were to fail closed, heat exchanger outlet temperatures would rise.

Distractor 3 is plausible because RBCCW outlet temperature will lower as a result of its Service Water TCV failing open. The TBCCW TCVs fails open; however, if theTBCCW TCV were to fail closed, the TBCCW, heat exchanger outlet temperature would rise; this makes the TBCCW portion of the distractor plausible. Also plausible because the majority of the essential and nonessential air operated vales identified in LOA-IA-101, Attachment A and B fail closed.

Reference: LOA-IA-101, Attachment A and B, Rev 12 and P&ID M-69, sh 1 and 2.
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 1 **Group:** 1
K/A: 295019 Partial or Complete Loss of Instrument Air
K2. 02 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Component cooling water
10 CFR Part 55 Content: 41.7
SRO Justification: N/A
Question Source: New
Question History: N/A
Comments:

Associated objective(s):

14	ID: 1263726	Points: 1.00

In accident conditions, IAW LGA-003, Primary Containment Control, WHY IS ACTION REQUIRED if drywell temperature cannot be restored and held below 340°F.

A. At this temperature, closure of the MSIVs, if required, could not be assured because the MSIV Solenoids have reached their environmental qualification temperature limit.

B. Implementation of Drywell Spray above this temperature will exceed the drywell analytical withstand temperature.

C. To provide margin to the temperature where the ADS SRVs and ADS Solenoids may not function if required to depressurize to RPV.

D. Suppression Chamber to Drywell Vacuum Breakers is not designed to operate at this temperature and may not be able to function and minimize a Suppression Chamber pressure spike under LOCA conditions.

Answer: C

Answer Explanation

Explanation: The EQ rating of equipment in the drywell, specifically the ADS valves and ADS solenoids is 340 °F for a significant time. If drywell temperature cannot be restored and maintained below the ADS qualification temperature, emergency RPV depressurization is performed to ensure that the SRVs are opened while still operable. Consistent with the definition of "restore," emergency RPV depressurization is not required until it has been determined that drywell sprays (initiated in Step DW/T-2) are ineffective in reducing drywell temperature. It is not expected that SRV operability will be immediately challenged when the ADS qualification temperature is reached. If drywell temperature is already above the specified value when Steps DW/T-2 and DW/T 3 are reached, drywell sprays may still be used, if available, in preference to emergency depressurization. If sprays are effective in reducing drywell temperature, emergency depressurization need not be performed. Extended operation above the ADS qualification temperature is not permitted, however. A determination that drywell temperature cannot be restored and maintained below the ADS qualification temperature may be made when, before, or after temperature actually reaches the specified value.

Distractor 1 is incorrect: The MSIVS and their solenoids are not a concern at this point in the EOPs; furthermore, they are likely already closed due to a LOCA condition. Plausible because the MSIV solenoids are EQ rated equipment.

Distractor 2 is incorrect: Drywell Spray, if not already initiated, may prevent exceeding the drywell analytical withstand temperature. Plausible because 340F is the design temperature of the Drywell.

Distractor 3 is incorrect: the design temperature of the Drywell is 340F. Plausible because the Suppression Chamber to Drywell Vacuum Breakers are essential for maintaining containment integrity: If the vacuum breaker were to fail, the rate of return of non-condensable gasses to the drywell as drywell pressure lowers during Drywell spray would be inadequate, and the rate of decrease in drywell pressure could be too fast for the operators to respond. .

Reference: LPGP-PSTG-01S05A, Revision 3, PLANT SPECIFIC TECHNICAL GUIDELINES SECTION 5A - PRIMARY CONTAINMENT CONTROL ENTRY, SUPPRESSION POO L TEMPERATURE, AND DRYWELL TEMPERATURE CONTROL, pages 79 - 82.

Reference provided during examination: N/A

Cognitive level: memory

Level (RO/SRO): RO

Tier: 1 **Group:** 1

K/A: 295028 High Drywell Temperature

K1.02 Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Equipment environmental qualification

10 CFR Part 55 Content:41.8

SRO Justification: N/A

Question Source: Bank

Question History: N/A

Comments:

Associated objective(s):

15	ID: 1259802	Points: 1.00

Given the following post transient conditions:

- Suppression Pool temperature is 197.5°F
- Suppression Pool level is -12 feet
- Suppression Chamber pressure is 2 psig
- Drywell pressure is 7 psig

Which one of the following parameter changes could result in damage to an RHR pump being used to cool the Suppression Pool?

A. Drywell pressure drops 1 psig.

B. Suppression Pool level drops an additional 2 feet.

C. Suppression Pool temperature rises an additional 2.5°F.

D. Suppression Chamber pressure drops an additional 1 psig

Answer: B

Answer Explanation

Explanation: Dropping Suppression Pool level an additional 2 feet requires use of the > -18 feet line on the NPSH curve (Figure NR). When above the line, as indicated by the pressure and temperature provided, cavitation could occur resulting in RHR pump damage.

Distractor 1 is incorrect: Lowering drywell pressure 1 psig would slightly lower the margin to pump cavitation; however, with Suppression Pool level at -12 feet, the >-13 feet line on the NPSH curve (Figure NR) indicates that cavitation would not occur. Plausible because lowering drywell pressure lowers Suppression Chamber pressure and lower pressure reduces RHR Pump NPSH.

Distractor 2 is incorrect: Raising suppression pool temperature an additional 2.5°F would lower the margin to pump cavitation; however, with Suppression Pool level at -12 feet, the >-13 feet line on the NPSH curve (Figure NR) indicates that cavitation would not occur. Plausible because raising Suppression Pool temperature reduces RHR Pump NPSH.

Distractor 3 is incorrect: Lowering suppression pool pressure an additional 1 psig would lower the margin to pump cavitation; however, with Suppression Pool level at -12 feet, the >-13 feet line on the NPSH curve (Figure NR) indicates that cavitation would not occur. Plausible because lowering Suppression Chamber pressure reduces RHR Pump NPSH.

Reference: LGA-003, Revision 16
Reference provided during examination: LGA-003, Detail NR
Cognitive level: High
Level (RO/SRO): RO
Tier: 1 **Group:** 1
K/A: 295030 Low Suppression Pool Water Level
A1.01 Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: ECCS systems
10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*
Question Source: New
Question History: N/A
Comments:
Associated objective(s):

16	ID: 1271648	Points: 1.00

- Suppression Pool Level is +3 inches
- Suppression Chamber Pressure is 13 psig
- Drywell Pressure is 17 psig
- RPV Pressure is 95 psig and lowering at 10 psig per minute
- Suppression Pool Temperature is 225°F and rising at 2°F per minute
- RPV Level is -150 inches on Wide Range
- All control rods are full in
- All available ECCS is injecting
- B RHR pump has just been returned to service

What is the required use of the B RHR pump?

A. Establish injection with the B RHR pump in the LPCI mode.

B. Establish suppression chamber spray with the B RHR Pump.

C. Establish suppression pool cooling with the B RHR Pump.

D. Establish drywell spray using B RHR pump.

Answer: A

<u>Note</u>

Applicant question: is RPV level being held at -150 inches or is it trending?

Applicant question: RPV level -150 on Wide Range meaning downscale, or are you positive level is at -150 Wide Range not lowering?

Response Given: level is lowering.

Answer Explanation
Explanation: Level control, per LGA-001, directs the use of preferred injection systems, which includes both the A and B RHR pumps, to control RPV water level between 11 in. and 59.5 inches. If level cannot be restored and maintained in that band, the operator is directed to hold level above –150 inches on WR. In this case, reactor water level is approaching –150 inches on WR and the RCIC system is approaching its isolation setpoint; therefore, injection with the A and B RHR pumps in the LPCI mode is required. **Distractor 1** is incorrect: The B RHR pump should not be used for suppression chamber spray because level is approaching – 150 on WR and adequate core cooling may not be assured without B RHR injecting in the LPCI mode. Plausible because suppression chamber spray would have been directed if adequate core cooling had been assured. **Distractor 2** is incorrect: The B RHR pump should not be used for suppression pool cooling because level is approaching – 150 on WR and adequate core cooling may not be assured without B RHR injecting in the LPCI mode. Plausible because suppression pool cooling would have been directed if adequate core cooling had been assured. **Distractor 3** is incorrect: The B RHR pump should not be used for drywell spray because level is approaching – 150 on WR and adequate core cooling will not be assured without B RHR injecting in the LPCI mode. Plausible because drywell spray would have been directed if adequate core cooling had been assured. **Reference:** LGA-001, Revision 16 and LGA-003, Revision 16 **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** RO **Tier:** 1 **Group:** 1 **K/A: 295031: Reactor Low Water Level** **EA1.01 Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL:** Low pressure coolant injection (RHR): Plant-Specific **Question Source:** Bank **Question History: N/A** **Comments:** **Associated objective(s):**

17 | **ID: 1272141** | **Points: 1.00**

The following conditions exist on Unit 2 following a transient:

- Reactor power is stable at 18% following a reactor scram.
- Reactor water level is -30 inches and lowering.
- MSIVs are open and reactor pressure is stable.
- The Motor Driven Feedwater Pump is NOT available.

LGA-010, Failure to Scram requires the crew to bypass MSIV isolations per LGA-MS-01, Using Main Condenser as Heat Sink.

Which of the following describes the reason this action must be taken?

A. Avoid adding heat to the Suppression Pool.

B. Maintain Feedwater as a preferred ATWS injection system.

C. Avoid the LGA-010 requirement for Standby Liquid Control (SBLC) injection.

D. Maintain Turbine Bypass Valves available for depressurization when Cold Shutdown Boron has been injected.

Answer: A

Answer Explanation

Explanation: LGA-MS-01 supports LGA-010, Failure to Scram by keeping the main steam lines open so steam can be sent to the Main Condenser. In this way the Main Condenser remains the primary heat sink for the reactor in an ATWS condition. Without this, RPV steam would all have to go to the Suppression Pool potentially challenging containment.

Distractor 1 is incorrect. Plausible because Feedwater is a preferred injection system and the motor driven Feedwater pump is unavailable.

Distractor 2 is incorrect. Plausible if the candidate assumes that boron injection is NOT required if heat is rejected outside containment.

Distractor 3 is incorrect. Plausible because the RPV is depressurized using bypass valves per the Pressure leg of LGA-010 after the Cold Shutdown Boron Weight has been injected.

Reference: LGA-010, Failure to Scram, Rev 15 and LGA-MS-01, Using Main Condenser as Heat Sink in ATWS, Rev 14.
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 1, **Group:** 1
K/A 295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown
EK3.06 Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Maintaining heat sinks external to the containment.
10 CFR Part 55 Content: 41.5
SRO Justification: *N/A*
Question Source: New
History: N/A
Comments:

Associated objective(s):

18	ID: 1455130	Points: 1.00

Unit 1 is at 100% power.

The NSO starts 1A CD/CB pump and secures 1B CD/CB pump.

One minute later, Annunciator 1H13-P601-F402, MSL A/B Radiation Monitor HI alarms.

(1) Is the MSL A/B Radiation Monitor High alarm expected or unexpected?

(2) What is the correct operator action, if any?

A. (1) Unexpected.
(2) Commence power reduction per LGP 3-1.

B. (1) Unexpected.
(2) Direct all nonessential personal to stay clear of Turbine Building Elevation 768.

C. (1) Expected.
(2) No additional action required. Monitor parameters and trends; annunciator 1N62-P600-B502, OFF GAS PRE-TREATMENT RADIATION HI may alarm.

D. (1) Expected.
(2) No additional action required. Monitor parameters and trends; annunciator 1N62-P600-B304, STATION VENT STACK RAD HI may alarm.

Answer: C

Note: this question was deleted based upon post-exam comment resolution. See exam report for further details.

Answer Explanation

Explanation: May also expect 1H13-P601-F402, MSL A/B Radiation Monitor HI alarm. – During CP changes and CD/CB Pump swaps MSL Rad monitor HI alarms have spuriously annunciated in the past when HWC is online. This phenomenon is an actual radiation level change induced by N-16 production, which is a normal byproduct of H2 gas injection into the reactor. The suspect cause is a release of H2 gas within the CD/CB piping from a pocketed location. When this finite amount of gas reaches the reactor it results in the formation of N-16 and is detected as a spike on the MSL and OG Pretreatment Rad monitors

Distractor 1 is incorrect: The alarm is expected. Plausible because if actual fuel damage had occurred, a power reduction would be the correct response.

Distractor 2 is incorrect: Plausible because if actual fuel damage had occurred, it would be reasonable action to maintain radiation exposure ALARA.

Distractor 3 is incorrect: N-16 is too short lived to cause a station vent stack rad alarm. Plausible because a station vent stack rad alarm would be a significant indication if actual fuel damage had occurred.

Reference: LOR-1H13-P601-F402, MSL A/B Radiation Monitor Downscale/INOP/HI and LOR-1N62-P600-B502, Rev 008, OFF GAS PRE-TREATMENT RADIATION MONITOR HIGH RADIATION
Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): RO
Tier: 1 **Group:** 1

K/A: 295038 High Off-Site Release Rate
G04.45 – Ability to prioritize and interpret the significance of each annunciator or alarm.

10 CFR Part 55 Content: 41.1
SRO Justification: *N/A*

Question Source: New
Question history: N/A

Comments:

Associated objective(s):

19	ID: 1455149	Points: 1.00

Unit 1 is operating at 100% of rated power.

- A fire is reported in the vicinity of the Turbine Driven Reactor Feed Pumps.
- Service Water system alignment and parameters are normal.

Which of the following would require a manual scram and entry into LGA-001, RPV Control?

A. ONE diesel fire pump ONLY is UNAVAILABLE, and fire header pressure is 119 psig and RISING.

B. ONE diesel fire pump ONLY is UNAVAILABLE, and fire header pressure is 121 psig and LOWERING.

C. TWO diesel fire pumps are UNAVAILABLE, and fire header pressure is 123 psig and RISING.

D. TWO diesel fire pumps are UNAVAILABLE, and fire header pressure is 125 psig and LOWERING.

Answer: D

Answer Explanation
Explanation: From LOA-FP-101, UNIT 1 Fire Protection System Abnormal, if a fire is currently in progress at 710' elevation or above while both Diesel Fire Pumps are NOT available; restore header pressure to 125 PSIG or shutdown/scram the reactor. With two fire pumps unavailable and a pressure of 125 psig and lowering, Service Water is not able to maintain pressure greater than 125 psig, and a manual scram and entry into LGA-001, RPV Control, is required. **Distractor 1** is incorrect: Only one diesel fire pump is unavailable and fire header pressure is rising. The distractor is plausible because fire header pressure is less than 125 psig. **Distractor 2** is incorrect: Only one diesel fire pump is unavailable; also, the second diesel fire pump starts at 120 psig, so this indicates that fire header pressure can be restored when the 0B Diesel Fire Pump starts. Distractor is plausible because fire header pressure is less than 125 psig and lowering **Distractor 3** is incorrect: Fire header pressure at 123 psig and rising with TWO diesel fire pumps UNAVAILABLE indicates that Service Water s restoring Fire header pressure. Distractor is plausible because both diesel fire pumps are unavailable and fire header pressure is less than 125 psig. **Reference:** LOA-FP-101 UNIT 1 Fire Protection System Abnormal, LOP-FP-02 Fire Pump Diesel Startup and Shutdown **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** RO **Tier:** 1 **Group:** 1 **K/A: 600000 Plant Fire on Site** **2.4.1 Knowledge of EOP entry conditions and immediate action steps.** **10 CFR Part 55 Content: 41.10** **SRO Justification:** N/A **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

20	ID: 1271649	Points: 1.00

Units 1 and 2 are operating at rated power supplying VARS to the grid:

An electrical grid disturbance has OCCURRED and both Units are stable.

The crew receives notification from Transmission Switching Operations (TSO) that LaSalle switchyard voltage is predicted to fall below 353 kV if either LaSalle main generator were to trip.

Under these conditions, what is the relationship between generator output voltage, VARS, and switchyard voltage?

A. Raising generator output voltage will RAISE VARS and may RAISE post generator trip switchyard voltage.

B. Lowering generator output voltage will RAISE VARS and may RAISE post generator trip switchyard voltage.

C. Raising generator output voltage will LOWER VARS and may RAISE post generator trip switchyard voltage.

D. Lowering generator output voltage will LOWER VARS and may RAISE post generator trip switchyard voltage.

Answer: A

Answer Explanation

Explanation: Under these conditions, VARS are raised by raising generator output voltage. Per LOA-GRID-001, Low Grid Voltage, with two units in operation, raising generator voltage to raise VARS may help to restore switchyard voltage.

Distractor 1 is incorrect: Lowering voltage will lower VARs under these conditions and will not raise post trip switchyard voltage. The distractor is plausible because if VARS were initially negative, lowering generator voltage would raise VARs.

Distractor 2 is incorrect: Under these conditions, VARS are raised by raising generator output voltage. Distractor is plausible because if VARS were initially negative, raising generator voltage would lower VARs.

Distractor 3 is plausible because under these conditions, lowering voltage will lower VARS. If the candidate does not understand the relationship between generator voltage, switchyard voltage and the electrical grid during disturbances that cause switchyard voltage to lower, they may conclude that switchyard voltage will rise when lower generator voltage during an electrical grid disturbance and switchyard voltage is lowering.

Reference: LOA-GRID-001, Low Grid Voltage, Rev 14
Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): RO
Tier: 1 **Group:** 1

K/A Number: 700000 Generator Voltage and Electric Grid Disturbances
AK2.03 Knowledge of the interrelations between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following: Sensors, detectors, indicators

10 CFR Part 55 Content: 4.4
SRO Justification: *N/A*

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

21	ID: 1454914	Points: 1.00

Shutdown Cooling (SDC) is in operation on Unit 2

- An inadvertent isolation signal causes the outboard SDC Suction Valve, 2E12-F008, to close.

Identify the cause of the inadvertent isolation.

A. 2B21-N703A and 2B21-N703B, Reactor Vessel Water Level 3 trip units, failed downscale.

B. 2B21-N703A and 2B21-N703D, Reactor Vessel Water Level 3 trip units, failed downscale.

C. 2B21-N704A and 2B21-N704B, Reactor Vessel Water Level 2 trip units, failed downscale.

D. 2B21- N704A and 2B21-N704D, Reactor Vessel Water Level 2 trip units, failed downscale.

Answer: A

Answer Explanation
Explanation: The SDC Suction Valve isolates on Level 3. The isolation logic is of 2 out of 2 taken once with transmitters A and B for the outboard logic and C and D for the inboard logic; therefore, 2B21-N703A and 2B21-N703B would need to actuate. **Distractor 1** is incorrect because the 2 out of 2 taken once isolation logic has not been satisfied; transmitter A inputs to the outboard isolation logic and transmitter D inputs to the inboard isolation logic. Distractor is plausible because MSIV isolation logic utilizes channels A and D in a one out of two taken twice logic. **Distractor 2** is incorrect but plausible because 2B21-N704A and 2B21-N704B input to the Level 2 isolation logic and would result in isolation signals to their affected valves if actuated and because the isolation logic operates similarly to the Level 3 isolation logic; it is 2 out of 2 taken once with transmitters A and B for the outboard isolation logic. **Distractor 3** is incorrect but plausible because 2B21-N704A and 2B21-N704D input to the Level 2 isolation logic and because the MSIV isolation logic utilizes channels A and D in a one out of two taken twice logic. **Reference: TRM Appendix G, page 53 and 68** **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** RO **Tier:** 1 **Group:** 2 **K/A: 295020 Inadvertent Containment Isolation** **A2.06 Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION:** Cause of isolation **10 CFR Part 55 Content: 41.10** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

22	ID: 1433114	Points: 1.00

Unit 1 is operating at rated power.

Determine the operability requirement(s) of the 1B21-R884B, Post Accident Monitoring (PAM) RPV Level/Pressure Instrument on 1H13-P601 in accordance with Tech Spec 3.3.3.1.

Table 3.3.3.1-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS
1. Reactor Steam Dome Pressure	2
2. Reactor Vessel Water Level	
a. Fuel Zone	2
b. Wide Range	2

The 1B21-R884B PAM Level function is ___1___ AND the pressure function is ___2___.

A. (1) required to be operable
(2) required to be operable

B. (1) required to be operable
(2) NOT required to be operable

C. (1) NOT required to be operable
(2) required to be operable

D. (1) NOT required to be operable
(2) NOT required to be operable

Answer: A

Answer Explanation

Explanation: Tech Spec 3.3.3.1 requires two channels of RPV Steam Dome Pressure and Rx Vessel Wide Range Water Level. The Tech Spec states that the PAM instrumentation for each Function in Table 3.3.3.1-1 shall be OPERABLE in Mode 1 and 2. The table lists RPV Steam Dome Pressure and Rx Vessel Wide Range Water Level as two different functions. Therefore, both the Rx Vessel Water Level and the pressure functions are required to be operable in the current Mode 1 condition. The specific instruments in the Main Control Room that meet the Tech Spec 3.3.3.1 requirements are not provided in the question stem. This requires knowledge of the indicators and recorders in the Main Control Room that satisfy the channel requirements for both the RPV Steam Dome Pressure function and the Rx Vessel Wide Range Water Level function. For both Division 1 and 2, there is one recorder for each division that satisfies the RPV Steam Dome Pressure function; however, for the Rx Vessel Wide Range Water Level function, division 1 and division 2 differ; specifically, for division 2, which is utilized in the question stem, only one level recorder (1B21-R884B) is available to satisfy the level function while for division 1, there is a level recorder (1B21-R884A) AND a level indicator (1B21-R615) that are able to satisfy the level function; therefore, because division 2 is referenced in the stem, both the pressure and the level functions must be operable for 1B21-R884B.

Distractor 1 is incorrect: The PAM instrumentation for each Function in Table 3.3.3.1-1 shall be OPERABLE in Mode 1 and 2. The distractor is plausible because the specific instruments in the Main Control Room that meet the Tech Spec 3.3.3.1 requirements are not provided in the question stem. This requires knowledge of the indicators and recorders in the Main Control Room that satisfy the channel requirements for both the RPV Steam Dome Pressure function and the Rx Vessel Wide Range Water Level function, and there are differences between the two divisions as outlined in the explanation above. The distractor is also plausible because for division 2, which is utilized in the question stem, only one level recorder (1B21-R884B) is available to satisfy the level function.

Distractor 2 is incorrect: The PAM instrumentation for each Function in Table 3.3.3.1-1 shall be OPERABLE in Mode 1 and 2. The distractor is plausible because the specific instruments in the Main Control Room that meet the Tech Spec 3.3.3.1 requirements are not provided in the question stem. This requires knowledge of the indicators and recorders in the Main Control Room that satisfy the channel requirements for both the RPV Steam Dome Pressure function and the Rx Vessel Wide Range Water Level function, and there are differences between the two divisions as outlined in the explanation above. Furthermore, the distractor is plausible because if the recorder EIN in the question stem were to be changed to 1B21-R884A, this answer would be correct because the level function would be satisfied by 1B21-R615.

Distractor 3 is incorrect: The PAM instrumentation for each Function in Table 3.3.3.1-1 shall be OPERABLE in Mode 1 and 2. The distractor is plausible because the specific instruments in the Main Control Room that meet the Tech Spec 3.3.3.1 requirements are not provided in the question stem. This requires knowledge of the indicators and recorders in the Main Control Room that satisfy the channel requirements for both the RPV Steam Dome Pressure function and the Rx Vessel Wide Range Water Level function, and there are differences between the two divisions as outlined in the explanation above.

Reference: Tech Spec LCO 3.3.3.1

Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): RO

Tier: 1 **Group:** 2

K/A: 295009 Low Reactor Water Level

Generic 02.37 Ability to determine operability and/or availability of safety related equipment.

10 CFR Part 55 Content: 41.7

SRO Justification: *N/A*

Question source: Modified

Question history: Bank, ID number 1263885

Comments:

Associated objective(s):

23	ID: 1271917	Points: 1.00

Unit 2 was operating at 100% power when a Reactor Water Cleanup (RWCU) leak required entry into LGA-002, Secondary Containment Control, and a manual scram was initiated.

- RWCU cannot be isolated.
- Primary Containment parameters are normal.
- RPV pressure is being maintained 450 to 650 psig.
- SBGT inlet temp is 160 degrees.
- RPV water level is 40 inches on 2C34-R606B and lowering at a rate of 2 inches per minute.

If RPV water level continues to lower, which of the following level instruments will be the FIRST to become unreliable?

A. Fuel Zone

B. Wide Range

C. Upset Range

D. Narrow Range

Answer: D

Answer Explanation

Explanation: With normal drywell temperature (approximately 110°F) and Reactor Building temperatures greater than 150 degrees, Narrow Range level instruments must be ≥ + 25 inches if they are to be used. For the given conditions, below + 25 inches, Narrow Range level instruments become unreliable.

Distractor 1 is plausible because the Fuel Zone is affected by elevated Reactor Building temperatures. Distractor is incorrect because the Fuel Zone instrument is reliable if level is ≥ - 311 inches.

Distractor 2 is plausible because the Wide Range instrument is affected by elevated Reactor Building temperatures. Distractor is incorrect because the Wide Range instrument is reliable if level is ≥ - 90 inches.

Distractor 3 is plausible because the Upset Range instrument is affected by elevated Reactor Building temperatures. Distractor is incorrect because the Upset Range instrument is reliable if level is ≥ + 9 inches.

Reference: LGA-001, RPV Control, Table K, Rev 17
Reference provided during examination: Table K of LGA-001, RPV Control

Level (RO/SRO): RO

Cognitive level: High
Tier: 1 **Group:** 2

K/A: 295023 High Secondary Containment Area Temperature
K2.05 Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following: Temperature sensitive instrumentation

10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

24	ID: 1272149	Points: 1.00

Unit 1 is in Mode 5 for a planned refueling outage with fuel movements in progress.
Unit 2 is operating at 100% power.

Unit 1 receives the following alarms:

- LOR-1N62-P600-B401, Reactor Building Ventilation Exhaust Radiation High reading 6.0 mrem/hr and rising slowly.
- LOR-1H13-P601-E306, Fuel Pool Ventilation Radiation High reading 5.0 mrem/hr and rising slowly.

What is the correct operator ACTION and the REASON(S) for that action?

A. Evacuate the entire Refuel Floor to minimize the possible spread of contamination ONLY.

B. Evacuate the Unit 1 Refuel Floor ONLY to minimize the possible spread of contamination ONLY.

C. Evacuate the entire Refuel Floor to minimize personnel radiation exposure AND the possible spread of contamination.

D. Evacuate the Unit 1 Refuel Floor ONLY to minimize personnel radiation exposure AND the possible spread of contamination

Answer: C

Answer Explanation

Explanation: Evacuate the entire Refuel Floor to minimize personnel radiation exposure AND the spread of contamination – as directed per step B.1 of LOA-AR-101. Abnormally high area radiation is an indication of many possible conditions. High radiation in several areas is a symptom of major leaks, major equipment failure, and spreading contamination. Action is taken to minimize personnel radiation exposure, stop the source of the radiation, stop the spread of contamination, and clean up the contamination.

Distractor 1 is plausible because the ventilation radiation monitors are designed to monitor for elevated radiation levels inside the ventilation duct which would likely be from particulate and noble gas activity. The VR rad levels given would cause a trip of the VR system. VR system flow is designed to move air from areas of lowest to highest contamination and without ventilation flow spread of contamination would be affected. Distractor is incorrect because with the FC rad alarm in, personal rad exposure from possibly damaged fuel bundle is also a concern.

Distractor 2 is plausible because minimizing the spread of loose surface contamination is a valid reason for evacuating the refuel floor and the refueling operations are taking place on Unit 1. Distractor is incorrect because the Refuel Floor is one space encompassing both Unit 1 and Unit 2 areas. All individuals in this area must be evacuated.

Distractor 3 is plausible because the refuel floor and the refueling operations are taking place on Unit 1.

Reference: LOA-AR-101, Rev 04, Unit 1 Area Radiation Monitoring System Abnormal
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 1 **Group:** 2
KA:295034 Secondary Containment Ventilation High Radiation
K3.03 Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Personnel evacuation
10 CFR Part 55 Content: 41.5
SRO Justification: *N/A*
Question Source: New
Question History: N/A

Comments:

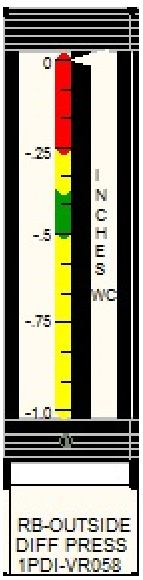
Associated objective(s):

25 **ID: 1419527** **Points: 1.00**

Unit 1 is operating at 100% power:
Unit 2 is in Mode 5 for a planned refueling outage with fuel movements in progress.

- A fuel handling accident occurs on Unit 2.
- Standby Gas Treatment is manually initiated.
- On 1PL27JC and 2PL27JC, Alarm 2-2, Reactor Building Ventilation System High Differential Pressure is received and does not clear.

Below is the indication on 1PM06J:



What is the PRIMARY operational concern as a result of this alarm?

A. Unmonitored release from secondary containment.

B. Excessive flow rate through the Standby Gas Treatment filters.

C. Airborne contamination in the Reactor Building due to VR Supply Fans NOT running.

D. Airborne contamination in the Turbine Building when entering or exiting either Reactor Building.

Answer: A

Answer Explanation
Correct because positive differential pressure raises the chance of unmonitored, unfiltered leakage. **Distractor 1** is incorrect: Any excess leakage from secondary containment would be unfiltered. Plausible if the candidate believes that a High D/P in secondary containment would cause a high flow rate through SBGT. **Distractor 2** is incorrect. Plausible because the VR system would have been manually shutdown when SBGT was started and because the alarms on 1(2)PL27JC would cause the VR fans to trip if they had been running. **Distractor 3** is incorrect: Plausible because the alarm is indicative of a higher pressure in secondary containment (i.e. positive pressure relative to outside secondary containment). **Reference:** 1(2)PL27JC, Alarm 2-2, Reactor Building Ventilation System High Differential Pressure, Rev 3 **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** RO **Tier:** 1 **Group:** 2 **KA: 295035 Secondary Containment High Differential Pressure** **K1.02 Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE:** Radiation release **10 CFR Part 55 Content: 41.8** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

26 **ID: 1430888** **Points: 1.00**

A LOCA has occurred in the Drywell on Unit 1:

- Drywell pressure is 5 psig.
- Drywell temperature is 340°F.
- Suppression Pool level is + 10 feet.

What is the operational impact, if any, of initiating Drywell Sprays?

A. Drywell parameters will return to allowable values upon initiation of Drywell Sprays with no additional operational impact.

B. High differential pressure between the Suppression Chamber and the Drywell may result in exceeding the Primary Containment Pressure Limit.

C. The evaporative cooling pressure drop may result in Primary Containment pressure dropping at a rate too fast to be compensated for by operator action.

D. The evaporative cooling pressure drop may result in high differential pressure between the Suppression Chamber and the Drywell, lowering the margin to the Pressure Suppression Pressure limit.

Answer: C

Answer Explanation

Correct answer: The Drywell Spray Initiation Limit (DSL) is the highest drywell temperature at which initiation of drywell sprays will not result in an evaporative cooling pressure drop to below the high drywell pressure scram setpoint. The final pressure following evaporative cooling is limited to the scram setpoint to ensure that the operator has time to terminate sprays before convective cooling reduces pressure below 0 psig. Drywell spray operation must be terminated by the time drywell pressure drops to 0 psig to ensure that primary containment pressure is not reduced below atmospheric. Maintaining a positive pressure precludes air intake through the vacuum relief system to de inert the primary containment and also provides a positive margin to the negative design pressure of the primary containment.

Distractor 1 is incorrect: Drywell spray is not allowed outside of the DSIL curve. Plausible because this answer would be correct if drywell pressure was greater than 8 psig.

Distractor 2 is incorrect: There is a delta-P between the suppression chamber and the Drywell. If sprays were initiated with suppression pool level > 722', the pressures could not equalize between the chamber and drywell. Plausible if the student confuses the level limit with the DSIL. Also, this answer would be correct if Suppression Pool level were greater than 22 feet.

Distractor 3 is incorrect: Plausible because the differential pressure does rise when drywell sprays are initiated; however, the margin to the Pressure Suppression Pressure limit is improved when vacuum breakers open during drywell spray.

References: LGA-003, Revision 16, Drywell Spray Initiation Limit Curve, Plant Specific Technical Guidelines Section 5A – Primary Containment Control – Entry, Suppression Pool Temperature, And Drywell Temperature Control, LPGP-PSTG-01S05A, Revision 3
Reference provided during examination: LGA-003, Figure D
Cognitive level: High
Level (RO/SRO): RO
Tier: 1 **Group:** 2
K/A: 295010 High Drywell Pressure
K1.03 Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE : Temperature increases
10 CFR Part 55 Content: 41.10
SRO Justification: *N/A*
Question Source: New
Question History: N/A

Comments:

Associated objective(s):

27	ID: 1432823	Points: 1.00

Unit 1 is in a LOCA

- Drywell pressure is 10 psig and rising slowly
- Drywell and Suppression Chamber Hydrogen is 1%
- Drywell and Suppression Chamber Oxygen is 2%

What action is required?

The Hydrogen Recombiner must be…..

A. STOPPED manually

B. STARTED manually

C. Verified to have AUTO-TRIPPED

D. Verified to have AUTO-STARTED

Answer: B

Answer Explanation

Explanation: Manually start the Hydrogen Recombiner is correct. Detectable H2 requires the H2 Recombiner to be started as mixing systems to provide necessary drywell circulation.

Distractor 1: Manually stop the Hydrogen Recombiner. The distractor is plausible because Drywell pressure is rising and it is a sound operational practice to trip equipment rather than relying on automatic protective features.

Distractor 2 is incorrect: The setpoint for auto trip is 15.3 psig DW pressure. Plausible because Drywell pressure is rising, and it is a sound operational practice to verify equipment functions as designed when the setpoint for an automatic protective feature has been met.

Distractor 3 is incorrect: The Hydrogen Recombiners do not have an auto start feature. Distractor is plausible because other related equipment - specifically, the Post-LOCA Hydrogen Monitors - does have an auto start feature. Also plausible because it is a sound operational practice to verify equipment functions as designed when the setpoint for automatic initiation has been met.

Reference: LGA-011, Rev 011, HYDROGEN CONTROL
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 1 **Group:** 2
K/A: 500000 High Containment Hydrogen Concentration
A1.04 Ability to operate and monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: Drywell recirculating fans
10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*
Question Source: New
Question History: N/A

Comments:
Associated objective(s):

28	ID: 1454974	Points: 1.00

What is the power supply to the 1E12-F042B, LPCI Injection Valve?

A. 111Y

B. 135Y

C. 112Y

D. 136Y

Answer: D

Answer Explanation
Explanation: 1E12-F042B is powered by 136Y **Distractor 1** is incorrect: Plausible because 111Y powers Division 1 LPCI relay logic and control power. **Distractor 2** is incorrect: Plausible because 135Y powers multiple RHR valves, including: 1E12-F040A/B, 1E12-F052A, 1E12-F064A and 1E12-F087A **Distractor 3** is incorrect: Plausible because 112Y powers Division 2 LPCI relay logic and control power. **Reference:** 1E-1-4000CX, Key Drawing 480V MCC 136-Y1 **Reference provided during examination:** N/A **Cognitive level:** Memory **Level (RO/SRO):** RO **Tier:** 2 **Group:** 1 **K/A 203000 RHR/LPCI: Injection Mode** **K2.02 Knowledge of electrical power supplies to the following:** Valves **10 CFR Part 55 Content:** 41.7 ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** **Comments:** **Associated objective(s):**

29	ID: 1260222	Points: 1.00

Unit is 1 in Hot Shutdown with a leak in the Reactor Recirculation piping:

- Drywell Pressure is 1.1 psig
- Reactor Pressure is 280 psig
- RPV water level is - 60 inches

Which of the following PCIS group isolation(s) are expected for the present conditions?

A. Group 1, 2 and 3 ONLY

B. Group 2, 3 and 6 ONLY

C. Group 1, 2, 3, 4, and 5

D. Groups 2, 3, 4, 5, and 6

Answer: D

Answer Explanation
Explanation: Answer is correct because PCIS Group 6 isolates at 135 psig reactor pressure and Groups 2, 3 and 4, 5 isolate on Level 2 (-48”). **Distractor 1** is plausible because Group 2 and 3 isolate at Level 2 and Group 1 also isolates on lowering level (Level 1). **Distractor 2** is plausible because Group 2 and 3 isolate at Level 2 and Group 6 will also isolate on high RPV pressure. **Distractor 3** is plausible because Groups 2, 3 4, and 5 isolate at Level 2 and Group 1 also isolates on lowering level (Level 1). **Reference:** LOP-CX-06, Rev 008, Primary Containment Isolation Status Display **Reference provided during examination:** N/A **Cognitive level:** Low **Level (RO/SRO):** RO **Tier:** 2 **Group:** 1 **K/A: 205000 Shutdown Cooling System (RHR Shutdown Cooling Mode)** **K4.02 Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: High pressure isolation: Plant-Specific** **10 CFR Part 55 Content: 417** ***SRO Justification:*** *N/A* **Question Source:** Bank **Question History:** LGS NRC-05, OYS CERT-04 **Comments:** **Associated objective(s):**

30	ID: 1271693	Points: 1.00

Unit 1 was operating at rated power:

The reactor SCRAMMED coincident with a loss of all offsite power.
All AC sources supplying the station batteries are lost.

With the system operating as designed and no action taken per LOA-AP-101 to shed DC loads, the discharge rate of the station batteries is such that essential DC loads will be supplied at adequate voltage for a MAXIMUM of ______ hour(s).

A. THREE

B. FOUR

C. EIGHT

D. TWENTY-FOUR

Answer: B

Answer Explanation
Explanation: The ampere-hour capacity of each battery is adequate to supply expected essential loads for four hours following station trip and loss of all a-c power without battery terminal voltage falling below 105-Vdc/210 VDC terminal voltage. **Distractor 1** is plausible because the crew must shed 250 VDC loads within three hours following a loss of AC power to preserve the RCIC function for the duration of the coping strategy. **Distractor 2** is plausible because following a loss of AC power the station coping time is eight hours per EC 391 795 with load shedding and is also the time per LOA-FSG-011 to provide FLEX EDG power to the 125 VDC and 250 VDC battery chargers. **Distractor 3** is plausible because there is a 24 hours coping time following a loss of offsite power before offsite equipment is expected to be onsite and ready for use. **Reference:** UFSAR 8.3.2.1.1, UFSAR 15.9.3.2, LOA-FSG-011, Rev 001, FLEX BEYOND DESIGN BASIS EXTERNAL EVENT GUIDANCE **Reference provided during examination:** N/A **Cognitive level:** Memory **Level:** (RO/SRO): RO **Tier:** 2 Group:1 **K/A: 263000 DC Electrical Distribution** **K/A: A1.01 Ability to predict and/or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including:** Battery charging/discharging rate **10 CFR Part 55 Content:** 41.7 **SRO Justification:** N/A **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

31	ID: 1260758	Points: 1.00

Unit 1 has scrammed from rated conditions:

- Reactor pressure is 1000 psig and rising slowly.
- All SRVs are CLOSED.

What color is the SPDS SRV Module status box for the given conditions?

A. RED

B. BLUE

C. GREEN

D. YELLOW

Answer: C

Answer Explanation

Explanation: The PPC has multiple functions such as indication and alarm, ESF display and SPDS display. Color is used along with graphical display and data to aid in operator understanding of plant conditions. Green is used for condition normal and data within the expected range. Yellow is used on many displays when a valve is within 2% of a limit. Red is used when a condition is not as expected or a limit has been exceeded. Blue is used for insufficient data or data out of range. Therefore, if the SPDS reactor pressure is above the setpoint (1076 psig), regardless of ADS demand, and all SRVs are closed, the status box color is changed to red. If an ADS demand exists and any of the ADS-SRV valves are closed, the status box color is also changed to red. In this case, the SRVs should be closed, so conditions are normal, and the status box is green.

Distractor 1 is plausible because Red is used when a condition is not as expected or a limit has been exceeded.

Distractor 2 is plausible because Blue is used for insufficient data or data out of range.

Distractor 3 is plausible because Yellow is used on many displays when a valve is within 2% of a limit.

Reference: LOP-CX-93, Rev 3, PLANT PROCESS COMPUTER ALARM MESSAGE INTERPRETATION, LOP-CX-93, REV 16, PLANT PROCESS COMPUTER MISCELLANEOUS FUNCTIONS and LOP-CX-02, Rev 8, SAFETY PARAMETER DISPLAY SYSTEM
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 2 **Group:** 1
K/A: 239002 Relief/Safety Valves
K1.02 Knowledge of the physical connections and/or cause-effect relationships between RELIEF/SAFETY VALVES and the following: SPDS/ERIS/CRIDS/GDS: Plant-Specific
10 CFR Part 55 Content: 41.2
SRO Justification: *N/A*
Question source: Bank
Question History: N/A
Comments:

Associated objective(s):

32	ID: 1254783	Points: 1.00

Unit 1 is operating at 75% power:

Low Pressure Core Spray (LPCS) pump surveillance is in progress with the system maintaining rated flow.

Which of the following will be observed FIRST if the LPCS system receives a high drywell pressure signal with a simultaneous Sudden Pressure signal in the Unit 1 Unit Auxiliary Transformer (UAT)?

A. The LPCS pump will trip.

B. Injection valve (1E21-F005) starts to open.

C. Minimum Flow valve (1E21-F011) starts to open.

D. Full Flow Test valve (1E21-F012) starts to close.

Answer: D

Answer Explanation

Explanation: "Test return valve automatically starts to close." is correct. The following events automatically occur upon the initiation of the Low Pressure Core Spray System:
1. LPCS Pump 1(2)E21-C001 starts.
2. DG 0 and LPCI A start (initiates by same logic as LPCS).
3. DG 0 Cooling Water Pump and LPCS/RCIC Room Cooler Fan start.
4. Full Flow Test Valve (F012) closes
5. Enables Injection Valve opening if the two pressure interlocks are met.

In this case, the pump is already running and will not trip. The injection valve will not open until the RPV depressurizes. The minimum flow valve will stay closed until the test return closes and flow drops below the minimum flow setpoint.

Distractor 1 is plausible because the LPCS pump is running at the start of the event. The 0 DG will auto start but will not close in to 141Y since the SAT remains available. The UAT sudden pressure will cause a generator trip and lockout of the UAT. If the sudden pressure trip had been on the SAT, the LPCS pump would have tripped as part of the 141Y load shed and would restart after the 0 DG powers 141Y.

Distractor 2 is plausible because with the LPCS pump running and an initiation signal present, the system is designed to inject to the RPV when the injection valve opens. The injection valve will receive an open signal, but it is interlocked closed due to high RPV pressure > 505 psig (normal for post scram). The distractor is incorrect because the injection valve will not open first; it will open at a later time.

Distractor 3 is plausible because with a LPCS initiation signal present and RPV pressure at normal operating pressure, the LPCS pump starts and operates on minimum flow until the injection valve low pressure interlock clears. The distractor is incorrect because the LPCS pump was already running at full flow at the start of the event and E21-F011 will not auto open until LPCS flow lowers to < 1670 gpm setpoint as the full flow test valve closes.

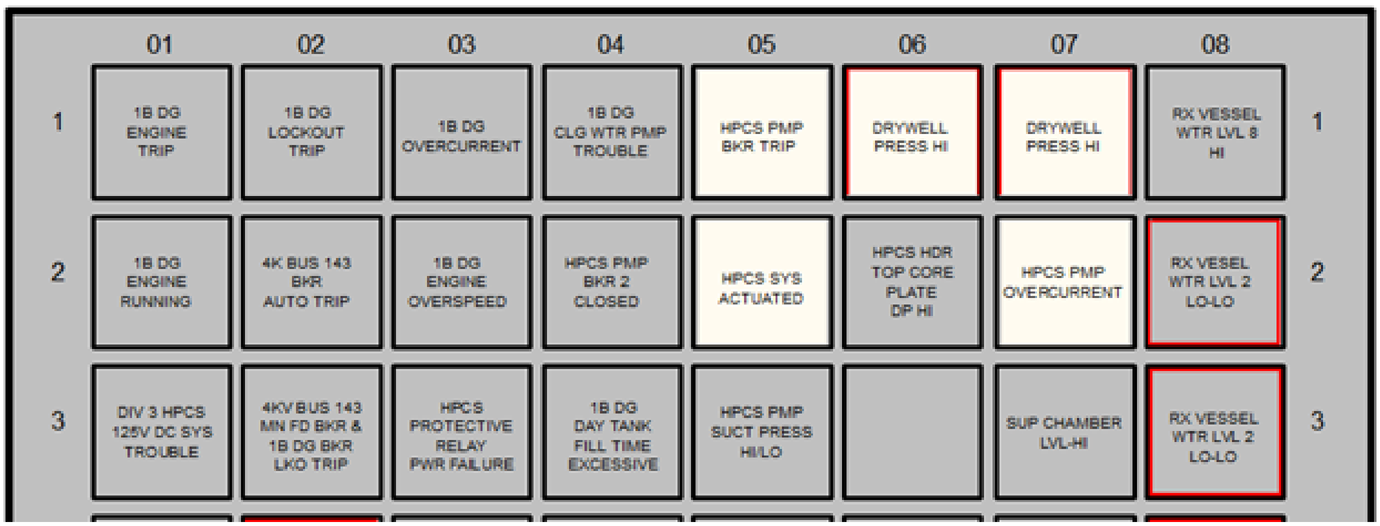
Reference: LOR-1H13-P601-C210, Rev 4, LPCI Injection Valve Low Pressure Permissive and LOS-LP-Q1, Rev 58, LPCS System Inservice Test
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 2 **Group:** 1
K/A: 209001 Low Pressure Core Spray System
A3.01 Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: Valve operation
10 CFR Part 55 Content: 41.7
SRO Justification: N/A
Question Source: LaSalle Bank
Question History: 11-01 ILT NRC Exam LaSalle
Comments:
Associated objective(s):

33	ID: 1430869	Points: 1.00

Unit 1 has experienced a LOCA in the drywell:

- Drywell Pressure reached 2.0 psig.
- HPCS has initiated and is injecting into the RPV.
- CRD, Condensate, Feedwater and RCIC are unavailable for injection.
- RPV water level is 30 inches and slowly lowering.

The following annunciators are observed at 1H13-P601:



What is the status of the HPCS Pump, and what operator action is required in accordance with the annunciator procedure(s)?

A. HPCS pump is TRIPPED. Dispatch an operator to SWGR 143 and to the HPCS Corner Room to determine the cause of the trip.

B. HPCS pump is TRIPPED. Dispatch an operator to SWGR 141Y and to the HPCS Corner Room to determine the cause of the trip.

C. HPCS pump is TRIPPED. The NSO must depress HI DRYWELL PRESSURE/LO WATER LEVEL pushbutton reset to restore injection.

D. HPCS pump is RUNNING. Dispatch an operator to SWGR 142Y and to the HPCS Corner Room to determine the cause of the alarms.

Answer: A

Answer Explanation

Explanation: HPCS pump is TRIPPED on overcurrent. Dispatch an operator to SWGR MCC 143 and to the HPCS Corner Room to determine the cause of the trip.

Distractor 1 is incorrect. Plausible because SWGR 141Y is the power supply for the LPCS pump.

Distractor 2 is incorrect: HPCS pump is tripped on OVERCURRENT. Plausible because the HPCS injection valve closes automatically at level 8 and requires manual reset if level is above the level low initiation setpoint.

Distractor 3 is incorrect. Plausible because HPCS MOV thermal overload protection IS BYPASSED for the Injection Valve, Full Flow Test Valve and Minimum Flow Valve in the event of a HPCS auto initiation. Also plausible because SWGR 142Y is the power supply for the C RHR pump.

Reference: 1H13-P601-A105, HPCS Pump Breaker Trip and 1H13-P601-A207, HPCS Pump Overcurrent
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 2 **Group:** 1
K/A 209002 High Pressure Core Spray System (HPCS)
A2.02 Ability to (a) predict the impacts of the following on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
Pump trips: BWR-5,6
10 CFR Part 55 Content: 41.5
SRO Justification: *N/A*
Question Source: New
Question History: N/A

Comments:

Associated objective(s):

34	ID: 1430889	Points: 1.00

Unit 2 is in an ATWS condition following a trip of the Main Turbine:

- The feeder breaker for MCC 236X-2 has tripped on overcurrent.
- The keylocked switch for the 2B SBLC Pump is placed in "SYS B".

How will the SBLC system respond to this condition?

A. ONLY the System B Squib valve will fire.

B. ONLY the System A Squib valve will fire.

C. Squib valves for BOTH System A and System B will fire.

D. NEITHER the System A nor the System B Squib valves will fire.

Answer: B

Answer Explanation
Explanation: The 2B SBLC pump is powered by MCC 236Y-2 and will start. The “B” squib vale is powered by MCC 236X-2 and will not fire. When the key switch is taken to the "B" position the "A" squib will fire because bus 235X-1 powering the “A” squib valve has not lost power. **Distractor 1** is incorrect: This is plausible because the 2B SBLC pump and the "B" squib have the same power supply, 242Y, but are powered from different sub-busses. **Distractor 2** is incorrect: This is plausible because both squib valves would normally fire. **Distractor 3** is incorrect: This is plausible because MCC 236X-2 powers SBLC components including the B Squib valve. **Reference:** LOR-2H13-P603-A105, Revision 4 and LGA-SC-201, Revision 2 **Reference provided during examination:** None **Cognitive level:** High **Level:** RO **Tier:** 2 **Group:** 1 **K/A: 211000 Standby Liquid Control System** **K6.03 Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY LIQUID CONTROL SYSTEM:** A.C. power **10 CFR Part 55 Content: 41.7** ***SRO Justification:*** *N/A* **Question Source:** New **Question History: N/A** **Comments:** **Associated objective(s):**

35	ID: 1454982	Points: 1.00

Unit 1 is operating at 100% power:

- RPS Surveillance test LOS-RP-M6, Manual Scram Instrumentation, is in progress for RPS Channel ‘A’.
- A Manual Scram was inserted using the RPS pushbutton for Trip system A1.
- The following indications are observed FOLLOWING an attempted reset of the tripped RPS channel:

  A1 and A4 Scram Group lights for the A RPS bus remain out (not illuminated).
  A2 and A3 Scram Group lights are illuminated.

Which of the following is the correct course of action per LOA-RP-101, Loss of RPS Power?

A. Suspend half-scram testing and consult Tech Specs. No additional action is required.

B. Suspend half-scram testing. VERIFY there are no blown Scram Group fuses; then, RESET the HALF SCRAM.

C. Complete half-scram testing for trip System A2; then, consult Tech Specs. No additional action is required.

D. Complete half-scram testing for trip System A2; then, VERIFY there are no blown Scram Group fuses; then, RESET the HALF SCRAM.

Answer: B

Answer Explanation

Explanation: Per LOA-RP-101, the crew must suspend any HALF SCRAM testing in progress; verify the affected 1C71-F18 fuse(s) not blown and reset the HALF SCRAM.

Distractor 1 is incorrect. Plausible because Tech Specs are normally referenced for a loss of safety related instrumentation; therefore, the candidate may consider the Technical Specification impact a priority.

Distractor 2 is incorrect. Plausible because surveillance test LOS-RP-M6 directs the performance of RPS testing for trip System A2 immediately after the performance of RPS testing for trip System A1 and because the candidate may consider the Technical Specification impact a priority.

Distractor 3 is incorrect. Plausible because surveillance test LOS-RP-M6 directs the performance of RPS testing for trip System A2 immediately after the performance of RPS testing for trip System A1.

Reference: LOS-RP-M6, Revision 1 and LOA-RP-101, Revision 15, Section B.4, Steps 1 through 5
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 2 **Group:** 1
K/A: 212000 Reactor Protection System
A2.03 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
10 CFR Part 55 Content: 41.5
SRO Justification: *N/A*
Question Source: New
Question History: N/A

Comments:

Associated objective(s):

36	ID: 1253217	Points: 1.00

Unit 2 Reactor Mode Switch was in STARTUP when an IRM detector spiked, causing a momentary upscale alarm.

What design feature allows the RO to determine which detector spiked?

A. Core monitoring computer printout.

B. Annunciator remains lit on front panel.

C. 2H13-P603, IRM upscale light seals in.

D. Back panel alarm seals in on the drawer.

Answer: D

Answer Explanation

Explanation: "back panel alarm seals in on the drawer." is correct.

Distractor 1: The Core monitoring computer printout does not use IRM input. Plausible because the Core Monitoring program does look at power.

Distractor 2: The Upscale annunciator comes in for multiple IRMs, does not discern which one. Plausible because an IRM upscale does cause an IRM upscale.

Distractor 3: The IRM upscale light does not seal in. Plausible because the light does come in on an alarm, but does not seal in.

Reference: LIS-NR-102A, Rev 7, IRM Channels A&E Rod Block and Reactor Scram Calibration.
Reference provided during examination: N/A
Cognitive level: Memory
Level (RO/SRO): RO
Tier: 2 **Group:** 1
K/A: 215003 Intermediate Range Monitor (IRM) System
K4.06 Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Alarm seal-in
10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*
Question Source: Bank
Question History: 2001-01 NRC License Exam
Comments:
Associated objective(s):

37	ID: 1454873	Points: 1.00

Unit 1 is in Mode 2 performing a normal startup from a mid-cycle maintenance outage.

- Rod pulls are in progress
- SRM A is ~ 6.0 cps
- SRM B is ~ 4.0 cps
- SRM C is ~ 5.0 cps
- SRM D is ~ 5.0 cps but is intermittently spiking to ~ 1 x 10E6 cps.

What is the impact on the SRM system and what operator action(s) is/are required?

A. SRM Rod Blocks occur; BYPASS D SRM and continue the startup.

B. SRM Rod Blocks occur; continue the startup. No other action is required.

C. SRM Rod Blocks AND a ½ scram occur; reset the ½ scram and continue the startup.

D. A full reactor scram occurs; perform reactor scram actions per LGP 3-2, Reactor Scram.

Answer: A

Answer Explanation

Explanation: With the SRM shorting links installed, as they would be for a mid-cycle startup, rod blocks occur, and LOA-NR-101 allows for bypassing an SRM that is operating erratically in the event of spurious trips. A rod block is received at 2 x 10E5 cps. With the reactor In Mode 2, 3 SRMs are sufficient to perform a startup; therefore, bypass SRM D and continue the startup.

Distractor 1 is plausible because the rod block setpoint has been exceeded and because, according to LOA-NR-101, short duration spikes and excursions on individual channels do not make the channel inoperable and do NOT require bypassing the channel except in the case of spurious trips (SRM channel noise normally adds to the SRM signal biasing the channel in the conservative - closer to trip – direction).

Distractor 2 is incorrect because a HI HI SRM trip with the shorting links installed would NOT result in a ½ scram. Distractor is plausible because the rod block and scram setpoints have been exceeded. Also plausible because short duration spikes and excursions on individual channels do not make the channel inoperable and do NOT require bypassing the channel except in the case of spurious trips (SRM channel noise normally adds to the SRM signal biasing the channel in the conservative - closer to trip – direction).

Distractor 3 is plausible because a HI HI SRM trip with the shorting links removed would result in a full reactor scram requiring the performance of reactor scram actions per LGP 3-2.

Reference: LOA-NR-101, Rev 19, Neutron Monitoring Trouble, LOR-1H13-P603-B404, Rev 4, Channel A SRM Hi-Hi, LGP-1-S1, Rev 079, MASTER STARTUP CHECKLIST, LIS-NR-301, Rev 021, UNIT 1 SOURCE RANGE MONITOR ROD BLOCK FUNCTIONAL TEST and 1E-1-4215A6G, RPS System schematic, part 7

Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): RO

Tier: 2 **Group:** 2

K/A: 215004 Source Range Monitor (SRM) System

A2. Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.05 Faulty or erratic operation of detectors/system RO 3.3 SRO 3.5

10 CFR Part 55 Content: 41.5 / 45.6

SRO Justification: *N/A*

Question Source: New

Question History: N/A

Comments:

LGP-1-S1 ensures LIS-NR-301 is complete prior to a normal reactor startup. LIS-NR-301 requires the SRM shorting links to be installed.

Associated objective(s):

041.00.21 Given a System lineup and various plant conditions, predict the Source Range Monitor System response to various system component failures while operating the system or on an exam in accordance with student text.

38	ID: 1433169	Points: 1.00

The Unit 1 ‘A’ SRM will generate a High-High count rate non-coincident full SCRAM only when the SRM shorting links are _______ in Backpanel 1H13-P609 AND _______ in Backpanel 1H13-P611.

A. INSTALLED AND INSTALLED

B. INSTALLED AND REMOVED

C. REMOVED AND INSTALLED

D. REMOVED AND REMOVED

Answer: D

Answer Explanation

Explanation: 1H13-P609 and 1H13-P611are located in the Main Control Room and are part of the NSO watch station. Recognizing the shorting links, their location and their normal configuration is part of the NSO general area and cyber security checks. A non-coincident neutron monitoring reactor SCRAM can be generated when at least one SRM shorting link associated with the 'A' SRM is removed in panel 1H13-P609 (RPS A) and one SRM shorting link associated with the 'A' SRM is removed in panel 1H13-P611 (RPS B).

Distractor 1 is incorrect. Plausible because many logic bypasses involve the insertion of jumpers; therefore, the candidate believe the shorting links are installed.

Distractor 2 is incorrect. Plausible because the SRM scram is non-coincident; therefore, the candidate may conclude that only one set of shorting links need to be removed in the 'A' RPS cabinet.

Distractor 3 is incorrect. Plausible because the SRM scram is non-coincident; therefore, the candidate may conclude that only one set of shorting links need to be removed in the 'A' RPS cabinet.

Reference: 1E-1-4215A6G, RPS RPS System schematic, part 7 and LIP-NR-505, Revision 5
Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 2 **Group:** 1

K/A: 215004 Source Range Monitor (SRM) System
2.1.30 Ability to locate and operate components, including local controls.

10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*

Question source: New
Question History: N/A

Comments:

Associated objective(s):
Lesson Plan number 41, Source Range Monitoring System. Objectives 041.00.18.a and 041.00.02
Lesson Plan number 49, Reactor Protection System. Objectives 049.00.04, 049.00.10 and 049.00.16

39 **ID: 1433113** **Points: 1.00**

Unit 2 is at rated conditions indicating 100% of RTP on all APRMs:

- Multiple LPRM High Voltage Power Supplies inside APRM ‘A’ fail de-energizing their associated LPRMs.

What are the expected Average Power Range Monitor (APRM) readings at 2H13-P603?

A. APRM ‘A’, APRM ‘C’ <u>and</u> APRM ‘E’ <u>all</u> indicate 100%.

B. APRM ‘A’, APRM ‘C’ <u>and</u> APRM ‘E’ <u>all</u> indicate LESS THAN 100%.

C. APRM ‘A’ indicates LESS THAN 100%. APRM ‘C’ <u>and</u> APRM ‘E’ indicate 100%.

D. APRM ‘A’ <u>and</u> APRM ‘C’ indicate LESS THAN 100%. APRM ‘E’ indicates 100%.

Answer: C

Answer Explanation
Explanation: Answering this question requires an understanding of the LPRM power supplies and how the LPRM outputs are distributed. An LPRM is an individual instrument. There are 172 LPRM instruments, each of which possesses its own high voltage power supply, physically located inside 8 different cabinets. Four of the cabinets containing LPRMs are powered from RPS A: APRM A, C and E and LPRM group A, and four cabinets are powered from RPS B: APRM B, D and F and LPRM group B. LPRM outputs are distributed to multiple instruments: their respective APRMs, the companion APRM's OPRM module, and the Rod Block Monitor. De-energizing individual LPRM power supplies associated with the LPRMs located inside APRM 'A' affects the 'A' APRM and the 'A' OPRM; therefore, APRM 'A' would indicate less than 100%. **Distractor 1** is incorrect. Distractor is plausible because this would be correct if all the failed LPRM power supplies were in LPRM Group 'A'. **Distractor 2** is incorrect. Distractor is plausible because LPRM outputs are distributed to their respective APRMs, the companion APRM's OPRM module, and the Rod Block Monitor. This answer would be correct if the LPRMs were also distributed across all 3 APRMs. **Distractor 3** is incorrect. Distractor is plausible because LPRM outputs are distributed to their respective APRMs, the companion APRM's OPRM module, and the Rod Block Monitor. This would be correct if the LPRM outputs were distributed across APRMs 'A' and 'C', similar to the RBM inputs. **Reference:** LOA-NR-201, Revision 20, Neutron Monitoring System Trouble and System Description 43 **Reference provided during examination:** N/A **Cognitive level:** memory **Level (RO/SRO):** RO **Tier:** 2 **Group**: 1 **K/A: 215005 Average Power Range Monitor/Local Power Range Monitor System** **K3.05 Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following:** Reactor power indication **10 CFR Part 55 Content: 41.7** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

40 | **ID: 1432832** | **Points: 1.00**

RCIC is operating in the PRESSURE CONTROL MODE with the RCIC Pump Discharge Flow Controller in AUTO set to 600 GPM.

Which of the following set of RCIC system control manipulations would result in the FASTEST RATE of RISE in Suppression Pool water temperature?

Throttle 1E51-F022, Full Flow Test Upstream Valve, ____(1)____, in order to ____(2)____.

A. (1) Open
(2) Maximize pump flowrate

B. (1) Closed
(2) Maximize pump flowrate

C. (1) Open
(2) Maximize pump discharge pressure

D. (1) Closed
(2) Maximize pump discharge pressure

Answer: D

Answer Explanation

Explanation: The RCIC flow controller is normally in auto and set at 600 GPM. With the controller in auto, it acts to raise or lower RCIC turbine speed to maintain the desired flowrate as 1E51-F022 is throttled is to maintain RCIC discharge pressure greater than reactor pressure. With the controller set to 600 GPM and 1E51-F022 throttled closed, RCIC turbine speed will rise to maintain the desired flowrate resulting in the maximum use of steam energy by the RCIC turbine and the fastest rise in Suppression Pool temperature.

Distractor 1 is incorrect because opening 1E51-F022 will not raise flowrate; instead, the RCIC flow controller will respond to lower turbine speed resulting in a slower rate of rise in Suppression Pool temperature. Plausible because the operator may assume that lowering system head loss will result in greater system flowrate and a faster rate of heat addition to the Suppression Pool.

Distractor 2 is incorrect. Plausible because throttling 1E51-F022 closed will force the RCIC Flow Controller to raise turbine speed to maintain the desired flowrate as system head loss changes.

Distractor 3 is incorrect: Opening 1E51-F022 will not raise flowrate; instead, the RCIC flow controller will respond to lower turbine speed resulting in a slower rate of rise in Suppression Pool temperature. Plausible because the operator may conclude that lowering system head loss will result in greater system flowrate and a faster rate of heat addition to the Suppression Pool.

References: LOP-RI-05, Revision 33 and LOP-RI-09, Revision 11 and UFSAR 5.4.6.3
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 2 **Group:** 1
K/A: 217000 Reactor Core Isolation Cooling System (RCIC)
A1.08 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including:
Suppression pool temperature
10 CFR Part 55 Content: 41.5
SRO Justification: *N/A*
Question Source: New
Question History: N/A

Comments:

Associated objective(s):

41	ID: 1432859	Points: 1.00

Unit 1 Drywell is experiencing a LOCA:

- Reactor Pressure is 900 psig and lowering slowly
- Reactor Water Level is -80 inches and lowering slowly
- Drywell Pressure is 0.5 psig and STABLE
- 1C RHR Pump is RUNNING
- LPCS, 1A RHR Pump AND 1B RHR Pump are NOT running

Based solely on the above conditions, which division(s) of ADS logic can be utilized to MANUALLY ACTUATE ADS valves, if any?

MANUAL ACTUATION of ADS valves via the DIVISION 1 ADS logic _______; AND MANUAL ACTUATION of ADS valves via the DIVISION 2 ADS logic _______.

A. IS possible;
IS possible

B. IS possible;
is NOT possible

C. is NOT possible;
IS possible

D. is NOT possible;
is NOT possible

Answer: A

Note

Applicant Question: do you want to know if they will function if manually initiated or if we should actuate them given the parameters?
Response Given: this question is asking how the equipment functions.

Answer Explanation
Explanation: Both divisions are available. Low pressure ECCS pumps are NOT required for manual ADS initiation. **Distractor 1** is incorrect. Plausible because low pressure ECCS pumps <u>are</u> required for AUTOMATIC initiation of the ADS logic and only one of four pumps is running. **Distractor 2** is incorrect. Plausible because low pressure ECCS pumps <u>are</u> required for AUTOMATIC initiation of the ADS logic and only one of four pumps is running. **Distractor 3** is incorrect. Plausible because the candidate may believe that two low pressure ECCS pumps <u>are</u> required ADS initiation and only one of four pumps is running. **Reference:** LOP-MS-03, Rev 011, Preparation For Standby Operation Of The Automatic Depressurization System **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** RO **Tier: 2 Group:** 1 **K/A: 218000 Automatic Depressurization System** **K4.02 Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following: Allows manual initiation of ADS logic** **10 CFR Part 55 Content: 41.7** **SRO Justification:** N/A **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

42	ID: 1259310	Points: 1.00

Unit 1 is operating at rated conditions:

- 1B RPS has lost power

Which of the following describes the expected indications on the 1H13-P601 PCIS Display?

A. 1IN031, TIP Purge Isolation Valve, - WHITE

B. 1WR040, RBCCW Outboard Isolation Valve, - GREEN

C. 1VQ047, Outboard Supply Isolation Damper, - GREEN

D. 1VP053A, Outboard Chilled Water Isolation Valve, - WHITE

Answer: D

Answer Explanation
Explanation: On loss of RPS B Inboard and Outboard PCIS valves for groups 1-3, 5-7- and 10 close EXCEPT FOR MSIV's, PCCW and RBCCW (which are DC powered.) When the isolation signal is received and the valves go closed they will appear GREEN on the 1H13-P601 PCIS Display. If the valves receive no isolation signal they appear white. **Distractor 1** is plausible because on a loss of RPS A, only the outboard PCIS group 1-3, 5-7, and 10 close. Since 1IN031 is a Division 2 inboard PCIS isolation valve, it would remain open on a loss of RPS A and would indicate white on the PCIS display. The distractor is incorrect because the valve is affected by this event and would close. **Distractor 2** is plausible because a loss of RPS B causes both an inboard and an outboard isolation for PCIS groups 1-3, 5-7- and 10. 1WR040 is a group 2 inboard PCIV. The distractor is incorrect because the valve is DC powered and will not close. **Distractor 3** is plausible because a loss of RPS B causes both an inboard and an outboard isolation for PCIS groups 1-3, 5-7- and 10. The distractor is incorrect because 1VQ047 is a Group 4 PCIV which will not close. **Reference**: LOP-CX-06, Rev 008, Primary Containment Isolation Status Display **Reference provided during examination:** N/A **Cognitive level:** Memory **Level (RO/SRO):** RO **Tier:** 2 **Group:** 1 **K/A: 223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off A4.01 Ability to manually operate and/or monitor in the control room:** Valve closures **10 CFR Part 55 Content**:41.7 **SRO Justification**: N/A **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

43	ID: 1455131	Points: 1.00

Unit 1 is in an ATWS.

- Reactor Power is 15% of rated.
- Drywell pressure is 3 psig.
- Suppression pool cooling is in service.
- MSIVs are open.
- All required LGA-010 procedural steps to intentionally lower RPV water level have been COMPLETED.

RPV water level rapidly lowers to – 150 inches and continues lowering at 2 inches per minute for 3 minutes AND is NOT immediately recoverable.

(1) How many ADS SRVs ARE open, if any?
(2) Procedurally, what action is required?

A. (1) SEVEN
(2) Slowly inject to the RPV with available Preferred ATWS Systems ONLY.

B. (1) ZERO
(2) Manually initiate ADS per LGA-006.

C. (1) ZERO
(2) Place SEVEN SRV handswitches to open per LGA-006.

D. (1) SEVEN
(2) Slowly inject to the RPV with Preferred AND Alternate ATWS Systems.

Answer: B

Note
Applicant Question: were steps prior to intentionally lower RPV level in LGA-010 completed?
Response Given: all prior steps were correctly completed.

Answer Explanation

Explanation: Zero SRVs will open because to mitigate the possibility of a Blowdown when intentionally lowering RPV water level during an ATWS, ADS is inhibited per LGA-010, Failure to Scram. Following the rapid drop of RPV water level to < - 150 inches with the inability to immediately recover RPV level, LGA-010 requires an RPV blowdown per LGA-010.

Distractor 1 is incorrect: 7 ADS valves will not open because ADS has been inhibited per LGA-010. The distractor is plausible because the conditions for auto ADS actuation have been met, and ADS would have auto initiated, opening 7 SRVs, if all the procedural actions for lowering level intentionally per LGA-010 had not already been completed, which includes inhibiting auto ADS actuation.

Distractor 2 is incorrect: Manually initiating ADS per LGA-006 is the correct operator action. The distractor is plausible because ZERO SRVs open is correct and because placing seven SRV handswitches to open is the required procedural action per LGA-006 in the event that manual ADS actuation failed to open seven SRVs.

Distractor 3 is incorrect: 7 ADS valves will not open because ADS has been inhibited per LGA-010. The distractor is plausible because the conditions for auto ADS actuation have been met, and ADS would have auto initiated, opening 7 SRVs, if all the procedural actions for lowering level intentionally per LGA-010 had not already been completed, which includes inhibiting auto ADS actuation. Additionally, in the event that injection with preferred ATWS systems failed to recover RPV level, injection with Alternate ATWS Systems is required per LGA-010.

Reference: LOR-1H13-P601-E202 (F202), Rev 3, LOR-1H13-P601-F202, Rev 1, Automatic Depressurization Logic B/A Initiated, and LGA-010, Rev 15, Failure to Scram

Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): RO

Tier: 2 **Group:** 1

KA: 239002 Relief/Safety Valves

A2.04 **Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** ADS actuation

10 CFR Part 55 Content: 41.5

Question Source: New

Question History: N/A

Comments:

Associated objective(s):

44 ID: 1432833 Points: 1.00

Unit 2 is operating at rated conditions:

- ONE speed sensing magnetic pickup on the 2A TDRFP fails to ZERO output.

What is the operational impact of this failure on the RWLC system, if any?

A. RWLC transfers 2A TDRFP to DEMAND Substitution.

B. The TDRFP low Speed Limit interlock is automatically reset.

C. No Impact on RWLC; TDRFP speed control will NOT be impacted.

D. The 2A TDRFP will trip, and it's M/A station output is automatically set to 0%.

Answer: C

Answer Explanation
Explanation: Three magnetic speed probes and one proximity probe are used for speed feedback. All speed probes are input to each of the 3 kernels so speed control will not be affected with the loss of any one of the three speed probes. **Distractor 1** is incorrect. Plausible because if ALL speed setpoint signals were lost, then RWLC would transfer the TDRFP to Demand Substitution and allow the operator to maintain level control. **Distractor 2** is incorrect. Plausible because the TDRFP low Speed Limit interlock is reset when actual TDRFP speed falls below 2000 RPM. **Distractor 3** is incorrect. Plausible because if all speed probes were lost, the affect TDRFP would trip and its M/A station output would automatically set to 0%. **Reference:** LOA-FW-201, Rev 008, Reactor Level Feedwater Pump Control Trouble and LOP-FW-16, Rev 035, 1(2)DS001 Operator Station Alarm Message Interpretation **Reference provided during examination:** N/A **Cognitive level:** Memory **Level (RO/SRO):** RO **Tier:** 2 **Group:** 1 **K/A: 259002 Reactor Water Level Control System** **K5.07 Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM:** Turbine speed control mechanisms: TDRFP **10 CFR Part 55 Content**:41.5 **SRO Justification**: N/A **Question Source**: New **Question History: N/A** **Comments:** **Associated objective(s):**

45	ID: 1432908	Points: 1.00

Unit 2 reactor power is 100%

- Annunciator 2H13-P603-B401, DIV 1 RX VESSEL WTR LVL 2 LO-LO, is in alarm.
- It is determined that 2B21-N402A, RPV Water Level 2 Channel A level sensor, has failed low.
- NO other 2H13-P603 panel annunciators have alarmed.

There is NO change in RPV water level.

Which of the following describes the effects of this failure, if any, on the plant?

A. Unit 1 and Unit 2 SBGT trains remain in STANDBY

B. Unit 1 SBGT is running AND an Inboard Group IV isolation on Unit 2

C. Unit 2 SBGT is running AND an Outboard Group IV isolation on Unit 2

D. Unit 1 and Unit 2 SBGT are running AND a FULL Group IV isolation on BOTH Units

Answer: A

Answer Explanation

Explanation: SBGT will not initiate because coincident RPV low level initiation signals are required; both divisional logics require 2 of 2 taken once in order to initiate.

Distractor 1: The Unit 1 SBGT initiation logic consists of Division 1 and Division 2 with Division 1 also initiating the Unit 2 SBGT train. The Unit 2 SBGT initiation logic consists of Division 1 and Division 2 with Division 1 also initiating the Unit 1 SBGT train. There are switches in the SBGT panel that will cause only the opposite unit VG train to start. This distractor is plausible because a partial Unit 2 Division 1 signal is present. The distractor is incorrect because the conditions in the question stem do not satisfy the 2 of 2 initiation logic needed for actuation.

Distractor 2 is plausible because the initiation signal is from Unit 2 and is the correct setpoint. This signal, if complete, would cause the Unit 2 SBGT train to start and a Group 4 isolation on Unit 2. The distractor is incorrect because the conditions in the question stem do not satisfy the 2 of 2 initiation logic needed for actuation.

Distractor 3 is plausible because this represents the expected system response in the event of a valid initiation signal: a full Group 4 isolation causing both Unit 1 and Unit 2 SBGT trains to start and a Group 4 isolation on both units. The distractor is incorrect because the conditions in the question stem do not satisfy the 2 of 2 initiation logic needed for actuation.

Reference: LOR-2H13-P603-B401, Revision 4 and LOR-2H13-P603-B412, Revision 4
Reference provided during examination: N/A
Cognitive Level: High
Level (RO/SRO): RO
Tier: 2 **Group:** 1
261000 Standby Gas Treatment System
K6.08 Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM: Reactor vessel level: Plant-Specific
10 CFR Part 55 Content: 41.7
SRO Justification: *N/A)*
Question Source: New
Question History: N/A

Comments:

Associated objective(s):

46	ID: 1454989	Points: 1.00

Which of the following is correct when paralleling AC sources?

When Paralleling an unloaded AC generator to an energized AC BUS, incoming (generator) frequency must be slightly ______ than running (BUS) frequency, and the generator synchroscope is turning _______.

A. lower; in the FAST (clockwise) direction

B. higher; in the FAST (clockwise) direction

C. lower; in the SLOW (counterclockwise) direction

D. higher; in the SLOW (counterclockwise) direction

Answer: B

Answer Explanation

Explanation: The diesel generator should be paralleled with the synchroscope rotating in the fast direction when unloaded, paralleling with an energized BUS.

Distractor 1 is incorrect: Frequency is higher. Plausible because the synchroscope should be turning in the FAST direction.

Distractor 2 is plausible because the frequency would be lower and the synchroscope should be turning in the SLOW direction if the BUS were being paralleled to a running generator.

Distractor 3 is plausible because frequency would be higher and because the synchroscope should be turning in the SLOW direction if the BUS were being paralleled to a running generator.

Reference: LOS-DG-02, REV 59, DG Startup and Operation
References provided during the examination: N/A

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 2 **Group:** 1

KA: 262001 A.C. Electrical Distribution
K5.01 Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: Principle involved with paralleling two A.C. sources

10 CFR Part 55 Content: 41.5
SRO Justification: *N/A*

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

47	ID: 1263940	Points: 1.00

Unit 1 is at full power performing LOS-DG-M3, 1B Diesel Generator Idle Start Surveillance.

The 'INCOMING Division III VOLTS' meter is reading 121 Volts AC.
The 'RUNNING BOP/Division III VOLTS' meter is reading 120 Volts AC.

Which of the following describes the indications immediately after ACB 1433, 1B DG Feed to Bus 143, is closed?

A. Bus 143 Voltage will rise. Diesel KVARS will rise.

B. Bus 143 Voltage will NOT rise. Diesel KVARS will rise.

C. Bus 143 Voltage will rise. Diesel KVARS will NOT rise.

D. Bus 143 Voltage will NOT rise. Diesel KVARS will NOT rise.

Answer: B

Answer Explanation

The 'INCOMING VOLTS' meter reading higher than the 'RUNNING VOLTS' meter indicates that the EDG output voltage is higher than the voltage on Bus 143. When the EDG breaker is closed with a higher output voltage than Bus 143, KVARS will be positive. Bus voltage will remain unchanged.

Distractor 1 is plausible because load (KW) will rise when diesel frequency is higher than BUS frequency when paralleling with a previously energized BUS and generator frequency is higher than BUS frequency; therefore, because “real” load rises when paralleling, some candidates may conclude that reactive load (KVARs) will rise as well.

Distractor 2 is plausible because if a diesel were supplying an isolated BUS, raising diesel voltage would raise BUS voltage.

Distractor 3 is plausible because BUS voltage will not rise and because KVARs can lower with raising generator voltage will lower the reactive load a generator is carrying in the case of operation with a lagging power factor.

Reference: LOS-DG-M3, REV 94, DG Operability Test
Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): RO
Tier: 2 **Group:** 1

K/A: 262001
A1.03 Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including: Bus voltage

10 CFR Part 55 Content: 41.5
SRO Justification: *N/A*

Question Source: Bank
Question history: From Quad Cities edited to make plant specific to LaSalle

Comments:

Associated objective(s):

48	ID: 1263941	Points: 1.00

Unit 1 is operating at 100% power with 250 VDC MCC 121Y de-energized, when the supply breaker to MCC 135X-3 trips.

What is the status of the Plant Process Computer (PPC) Uninterruptible Power Supply (UPS) **AND** what must occur in order for it to be powered from its preferred, normal source?

A. De-energized; restore MCC 135X-3 **OR** 250 VDC MCC 121Y

B. De-energized; restore MCC 135X-3 **AND** 250 VDC MCC 121Y

C. On alternate power supply; restore MCC 135X-3 **OR** 250 VDC MCC 121Y

D. On alternate power supply; restore MCC 135X-3 **AND** 250 VDC MCC 121Y

Answer: C

Note
Applicant Question: for the second part of the question, neither answer appears to be correct. The preferred, normal source of power is normal AC. Answer C if you use the OR statement sounds like either DC or normal AC is the normal source. Answer D implies that both AC and DC is required to be operating to have normal power which is untrue as a blocking diode prevents battery charging from the AC. Clarification on what are they considering "prefered, normal source"?
Response Given: remove word "normal."

Answer Explanation

Correct: On alternate power supply; restore MCC 135X-3 **OR** 250 VDC MCC 121Y – A loss of MCC 135X-3 AND 250 VDC MCC 121Y will result in the PPC UPS swapping to the alternate AC power – Restoration of either source will bring power back to the preferred source.

Distractor 1: A loss of MCC 135X-3 AND 250 VDC MCC 121Y will result in the PPC UPS swapping to the alternate AC power. Plausible if the candidate believes that a loss of the 2 normal power supplies de-energizes the PPC UPS.

Distractor 2: A loss of MCC 135X-3 AND 250 VDC MCC 121Y will result in the PPC UPS swapping to the alternate AC power. Plausible if the candidate believes that a loss of the 2 normal power supplies de-energizes the PPC UPS.

Distractor 3: Restoration of either source will bring power back to the preferred source. Plausible is the candidate believes that both sources must be restored in order to return the PPC UPS to its preferred source.

Reference: LOP-CX-08, Condition Report #: 1529395 Assignment #: 102
Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): RO
Tier: 2 **Group:** 1

KA: 262002 Uninterruptable Power Supply (A.C./D.C.)
A4.01 Ability to manually operate and/or monitor in the control room: Transfer from alternative source to preferred source

10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

49 **ID: 1433168** **Points: 1.00**

Unit 1 is operating at 100% power:

- Annunciator 1PM01J-A210, 250V DC BUS 121X/Y BATT GND DET ALARM, has alarmed.

What AUTOMATIC action(s), if any, will occur?

A. Selective TRIPPING of the affected DC loads.

B. A Ground Fault LOCKOUT of 250 VDC BUS 1.

C. The Battery Charger Output breaker trips on the affected bus.

D. The alarming Annunciator (1PM01J-A210) is the ONLY automatic action.

Answer: D

Answer Explanation
Explanation: The alarming Annunciator (1PM01J-A210) is the ONLY automatic action because the DC electrical distribution system is designed to be highly reliable to support the automatic operation of engineered safety features. **Distractor 1** is plausible because selective tripping is utilized for fault isolation on AC and DC power systems. DC systems are ungrounded and can tolerate a single ground and not cause an overcurrent condition. The distractor is incorrect because other alarms would be received if individual loads were lost. **Distractor 2** is plausible because BUS LOCKOUT devices (86) operate when other relays detect a fault. They automatically protect equipment by maintaining the system deenergized until the fault can be cleared and the 86 device reset. 86 devices are extensively used at LaSalle in AC distribution. A lockout device is also used for the Feed Regulating valve. LOA-DC-101 section B.9 indicates that a fault is a likely reason for the loss of 250 VDC Bus 1. The distractor is incorrect because a lockout device is not used for this system. **Distractor 3** is plausible because battery chargers do have automatic trips associated with their output breakers. The distractor is incorrect because the trips are for overvoltage. The charger is designed to limit current to within its capacity, and DC systems are ungrounded and can tolerate a single ground and not cause an overcurrent condition; therefore, a ground would not cause a charger to trip. Also, additional alarms would be received if a charger breaker were to trip open. **Reference:** LOP-DC-03, 250 VDC SYSTEM GROUND LOCATION AND ISOLATION, Rev 16 **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** RO **Tier:** 2 **Group:** 1 **KA: 263000 D.C. Electrical Distribution** **A3.01 Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including:** Meters, dials, recorders, alarms, and indicating lights **10 CFR Part 55 Content: 41.7** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

50 **ID: 1455132** **Points: 1.00**

Unit 1 is operating at 100% power:

The 1A DG is in STANDBY.
1A DG, "A" Starting Air Compressor supply breaker is OOS for maintenance.

1A DG, "B" Starting Air Compressor supply breaker has tripped on overcurrent.

- The 1A DG 'A' Air Receiver pressure is 150 psig.
- The 1A DG 'B' Air Receiver pressure is 225 psig.

What is the operability status of the 1A Diesel Generator, and what is the reason for that determination?

A. Operable; ONLY ONE air receiver is adequately pressurized

B. Operable; BOTH air receivers are adequately pressurized

C. Inoperable; BOTH air receivers are NOT adequately pressurized

D. Inoperable; ONLY ONE air receiver is adequately pressurized

Answer: D

Answer Explanation

Explanation: Inoperable; ONE air receivers is NOT adequately pressurized. The diesel is inoperable because one air receivers is < 165 psig. – Per LOP-DG-01, Limitation D.1.1.
Distractor 1 is incorrect. Plausible because one air bank with adequate pressure makes the DG available.
Distractor 2 is incorrect. Plausible because two air banks with adequate pressure makes the DG operable.
Distractor 3 is incorrect. Plausible because one air bank with adequate pressure makes the DG available.
Reference: LOP-DG-01, Rev 36, Preparation for Standby Operation of Diesel generator
Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): RO
Tier: 2 **Group:** 1

KA: 264000
2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

10 CFR Part 55 Content: 41.10
SRO Justification: *N/A*

Question source: Modified
Question History: 2008 LaSalle NRC Exam, modified

Comments:

D.1 Specific DG Limitations.

DG will remain operable if both air start subsystems are operable and air start receivers are pressurized to, at least, 200 psig. With starting air receivers pressure <200 psig sufficient capacity for five successive starts for the Division 1 or 2 DG or three successive starts for the Division 3 DG, as applicable, does not exist. However, as long as the receiver pressure is >165 psig, there is adequate air receiver pressure for, at least, one start, and the DG can be considered OPERABLE while the air receiver pressure is restored to the required limit (TS Bases B.3.8.3 D.1).
Associated objective(s):

51	ID: 1432910	Points: 1.00

Both Units were at full power with the Unit 1 and Unit 0 Station Air Compressors running when Bus 141X TRIPPED on overcurrent.

What is the status of Station Air Compressors 0SA01C AND 1SA01C?

0SA01C is ___(1)___.
1SA01C is ___(2)___.

A. (1) RUNNING;
(2) RUNNING

B. (1) NOT running;
(2) RUNNING

C. (1) RUNNING;
(2) NOT running

D. (1) NOT running
(2) NOT running

Answer: C

Answer Explanation
Explanation: The 1SA01C Air Compressor Motor is powered from 4KV Switchgear 141X, Cub # 5, which is de-energized; therefore, 1SA01C is off. **Distractor 1** is incorrect: The 0SA01C, Air Compressor Motor is powered from 4KV Switchgear 142X, Cub # 5. Plausible because 1SA01C Air Compressor Motor is powered from 4KV Switchgear 141X. **Distractor 2** is incorrect. The 0SA01C, Air Compressor Motor is powered from 4KV Switchgear 142X, Cub # 5. Plausible because 1SA01C Air Compressor Motor is powered from 4KV Switchgear 141X. **Distractor 3** is incorrect: The 0SA01C, Air Compressor Motor is powered from 4KV Switchgear 142X, Cub # 5. Plausible because 1SA01C Air Compressor Motor is powered from 4KV Switchgear 141X. **References:** LOR-1PM01J-A214, Revision 1 and LOR-1PM10J-B204, Revision 1 **Reference provided during examination:** N/A **Cognitive level:** memory **Level (RO/SRO):** RO **Tier:** 2 **Group:** 1 **K/A: 300000 Instrument Air System (IAS)** **K2.01 Knowledge of electrical power supplies to the following:** Instrument air compressor **10 CFR Part 55 Content: 41.7** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

52 **ID: 1432948** **Points: 1.00**

Both Units are operating at 100% power:

- Station Air Compressors 2SA01C and 0SA01C are running in a normal lineup.
- Unit 2 Turbine Building Closed Cooling Water (TBCCW) flow is lost in the Turbine Building.

How will the Station Air compressors respond if TBCCW flow is NOT restored in the Unit 2 Turbine Building?

A. 0SA01C Air Compressor ONLY will TRIP on high discharge AIR TEMPERATURE.

B. 2SA01C Air Compressor ONLY will TRIP on high discharge AIR TEMPERATURE.

C. 2SA01C AND 0SA01C Air Compressors are unaffected and will continue operating.

D. 2SA01C AND 0SA01C Air Compressors will TRIP on high discharge AIR TEMPERATURE.

Answer: B

Answer Explanation

Explanation: LaSalle Station has three station air compressors: Unit 0, Unit 1 and Unit 2. Two of the three air compressors are normally running, and they are swapped to equalize runtime. The air compressors are cooled by the TBCCW systems. The Unit 0 and Unit 1 Station Air Compressors are normally cooled by Unit 1 TBCCW but can be cross-tied to Unit 2 TBCCW. The Unit 2 Station Air Compressors are normally cooled by Unit 2 TBCCW but can be cross-tied to Unit 2 TBCCW. Air compressors trip on discharge High Air Temperature at 145°F. Because the question stem specifies the air compressors are running in a normal lineup, and only the 2SA01C Air Compressor is normally cooled by Unit 2 TBCCW, only the 2SA01C Air Compressor will trip on discharge High Air Temperature.

Distractor 1 is incorrect: The 0SA01C Air Compressor is normally cooled by Unit 1 TBCCW and will not trip. The distractor is plausible because the 0SA01C Air Compressor does automatically trip on discharge High Air Temperature and can be aligned to Unit 2 TBCCW.

Distractor 2 is incorrect: The 2SA01C Air Compressor is normally cooled by Unit 2 TBCCW and will trip. Distractor is plausible because the 0SA01C Air Compressor is normally cooled by Unit 1 TBCCW and will not trip. Also, the 2SA01C Air Compressor can be aligned to Unit 1 TBCCW.

Distractor 3 is incorrect: The 0SA01C Air Compressor is normally cooled by Unit 1 TBCCW and will not trip. Distractor is plausible because the 2SA01C Air Compressor will automatically trip on discharge High Air Temperature given the current, normal plant alignment and because the 0SA01C Air Compressor can be aligned to Unit 2 TBCCW.

References: LOR-1PM10J-B104, Revision 4 and LOR-2PM10J-B104, Revision 4
Reference provided during examination: N/A
Cognitive level: Memory
Level (RO/SRO): RO
Tier: 2 **Group:** 1
K/A 300000 Instrument Air System (IAS)
A3.02 Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including: Air temperature
10 CFR Part 55 Content: 417
SRO Justification: *N/A*
Question Source: New
Question History: N/A

Comments:

Associated objective(s):

53	ID: 1454981	Points: 1.00

Unit 1 is operating at rated power

- A trip of the 1A Service Water Pump results in a Service Water low pressure alarm.
- Shortly thereafter, 1H13-P601-B301, SERV WTR EFFLUENT RAD HI alarms.
- NO other alarms have been received on 1H13-P601.

What is the source of the rising radiation levels?

A. RBCCW Heat Exchangers

B. TBCCW Heat Exchangers

C. Fuel Pool Cooling Heat Exchangers

D. Primary Containment Ventilation Chiller Condensers

Answer: C

Answer Explanation
Explanation: Fuel Pool Cooling Heat Exchangers are a possible source of rising radiation levels. **Distractor 1** is incorrect: RBCCW is wrong due to there being no RBCCW rad alarms. Plausible because RBCCW is cooled by Service Water which is monitored by the Service Water PRM. **Distractor 2:** TBCCW does not cool any loads that would cause rising rad levels, and this portion of the Service Water header is not monitored by any PRMs. Plausible because TBCCW is cooled by Service Water. **Distractor 3:** Primary Containment Ventilation Chiller Condensers is wrong because the refrigerant is what is being cooled. Plausible because Primary Containment Ventilation Chiller Condensers are cooled by Service Water which is monitored by the Service Water PRM. **Reference:** LOR-1H13-P601-B301, Revision 2 **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** RO **Tier:** 2 **Group:** 1 **K/A: 400000 Component Cooling Water System (CCWS)** **K1.03 Knowledge of the physical connections and / or cause-effect relationships between CCWS and the following:** Radiation monitoring systems **10 CFR Part 55 Content: 41.2** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

54 **ID: 1252990** **Points: 1.00**

The plant is operating at rated power.

Which one of the following statements describes the operation of the CRD Drive Water Pressure Control Valve (PCV)?

_____(1)_______ the CRD Drive Water PCV will cause _______(2)________ water pressure to lower.

A. (1) Opening;
(2) Cooling

B. (1) Opening;
(2) Drive

C. (1) Closing;
(2) Drive

D. (1) Closing;
(2) Charging

Answer: B

Answer Explanation
Explanation: "(1) Opening; (2) Drive" is correct. Opening the drive water PCV will cause drive water pressure to drop because the drive water header branches off UPSTREAM of the drive water PCV, which acts as a backpressure regulator, normally, PCVs are throttled closed to lower pressure downstream, the opposite of the drive water PCV. **Distractor 1** is incorrect because the cooling water header is downstream of the drive water PCV. The distractor is plausible because the candidate may confuse the respective locations of the drive and cooling water headers. **Distractor 2** is plausible because PCVs are normally throttled closed to lower pressure downstream, the opposite of the drive water PCV. **Distractor 3** is incorrect because the closing the drive water PCV will cause charging water pressure to rise because the charging water header branches off UPSTREAM of the drive water PCV. Distractor is plausible because the candidate may confuse the respective locations of the charging and cooling water headers. **References:** LOP-RD-01, Rev 33, LOA-RD-101, Revision 20 and P&ID 100 and 146. **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** RO **Tier:** 2 **Group:** 2 **K/A: 201001 Control Rod Drive Hydraulic System** **A1.09 Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD DRIVE HYDRAULIC SYSTEM controls including:** CRD drive water flow **10 CFR Part 55 Content: 41.5** ***SRO Justification:*** *N/A* **Question Source:** Modified **Question History:** 2002-01 ILT Certification Exam **Comments:** **Associated objective(s):**

55	ID: 1259357	Points: 1.00

What RWCU interlock protects the LOW PRESSURE PIPING to Radwaste and the Main Condenser when draining reactor water to those locations?

The Blowdown Flow Control Valve 1G33-F033 will…

A. close on upstream pressure of < 5psig.

B. close on downstream pressure of < 5psig.

C. close on upstream pressure of > 140 psig.

D. close on downstream pressure of > 140 psig.

Answer: D

Answer Explanation

Explanation: The Blowdown Flow Control Valve will automatically close on either low pressure upstream of <5 psig or High Pressure Downstream of >140 psig. The high pressure downstream interlock protects the low pressure drain piping. The low pressure upstream interlock prevent condenser vacuum from draining RWCU.

Distractor 1 is plausible because the Blowdown Flow Control Valve does close on low upstream pressure of 5 psig.

Distractor 2 is incorrect: closes at 5 psig upstream; plausible because the Blowdown Flow Control Valve does close at 5 psig.

Distractor 3 is incorrect: closes at 140 psig downstream; plausible because the Blowdown Flow Control Valve does close at 140 psig.

References: LOR-1H13-P602-A208, Rev 2, Reactor Water Cleanup Recirculation Pumps Reject Discharge Pressure HIGH-LOW and LOP-RT-09, Rev 017, Reactor Water Clean-Up System (RWCU) - Coolant Rejection
Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 2 **Group:** 2

K/A: 204000 Reactor Water Cleanup System
K4.07 - Knowledge of REACTOR WATER CLEANUP SYSTEM design feature(s) and/or interlocks which provide for the following: Draining of reactor water to various locations

10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

56	ID: 1454435	Points: 1.00

The reactor was scrammed from full power several minutes ago.

The following are indicated on the Full Core Display

12 rods indicate:	**??**
3 rods indicate:	**XX**
168 rods indicate:	**00**
2 Rods indicate:	**FI**

What will the RCMS display indicate for the number of rods at unknown positions?

A. 3

B. 12

C. 15

D. 17

Answer: C

Answer Explanation
Explanation: 15 RCMS indicates that the 12 unknown rods (??) and the 3 rods with data faults (XX) are all at unknown positions. **Distractor 1:** Plausible because 3 (xx) rods indicate on the Full Core Display. These rods do not have a valid position indication and will be counted as unknown. The distractor is incorrect because the question stem includes other rods with unknown position (12 - ??). **Distractor 2:** Incorrect because 12 rods indicate (??) on the full core display. These rods do not have valid data from RPIS and are counted as unknown. The distractor is incorrect because the question stem includes other rods with unknown position (3 - XX). **Distractor 3:** Plausible because 17 rods are not indicating full in. **Reference:** LOP-RM-02 **Reference provided during examination:** N/A **Cognitive level:** Memory **Level (RO/SRO):** RO **Tier:** 2 **Group:** 2 **K/A: 214000 Rod Position Information System** **K3.01 Knowledge of the effect that a loss or malfunction of the ROD POSITION INFORMATION SYSTEM will have on following:** RWM: Plant-Specific **10 CFR Part 55 Content: 41.7** ***SRO Justification:*** *N/A* **Question Source:** New **Comments:** Note: At LaSalle Station, the RWM does not exist as a stand-alone system; rather, the RWM function is built into the RCMS system. **Associated objective(s):**

57 **ID: 1432953** **Points: 1.00**

Unit 1 is at full power with Primary Containment Chillers "B" and “C” in service.

- Bus 141Y trips on overcurrent.

IMMEDIATELY following the loss of 141Y, which of the following describes the status of the Primary Containment Chillers?

A. ONLY the “B” Chiller is running.

B. ONLY the “C” Chiller is running.

C. The “B” and “C” Chillers ARE running.

D. The “B” and “C” Chillers are NOT running.

Answer: C

Answer Explanation

Explanation: 141Y powers Chiller Unit 1VP01CA; therefore, Chiller Unit 1VP01CB, which is powered by 142Y, and Chiller Unit 1VP14C, which is powered by 138X, are both running and full containment cooling is maintained.

Distractor 2 is incorrect. Plausible because Chiller Unit 1VP14C is powered by 138X and is running.

Distractor 1 is incorrect. Plausible because Chiller Unit 1VP01CB is powered by 142Y and is running.

Distractor 3 is incorrect. Distractor is plausible because 141Y also powers ‘A’ RPS resulting in a half-scram and various containment isolations. The candidate may incorrectly conclude that a loss of ‘A’ RPS caused a PCCW isolation and no cooling water is circulating in the drywell.

Reference: LOA-AP-101, Revision 53 and LOP-VP-02, Revision 50
Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 2 **Group:** 2

K/A: 223001 Primary Containment System and Auxiliaries
K2.10 Knowledge of electrical power supplies to the following: Drywell chillers: Plant-Specific

10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*

Question Source: New
Question History: N/A

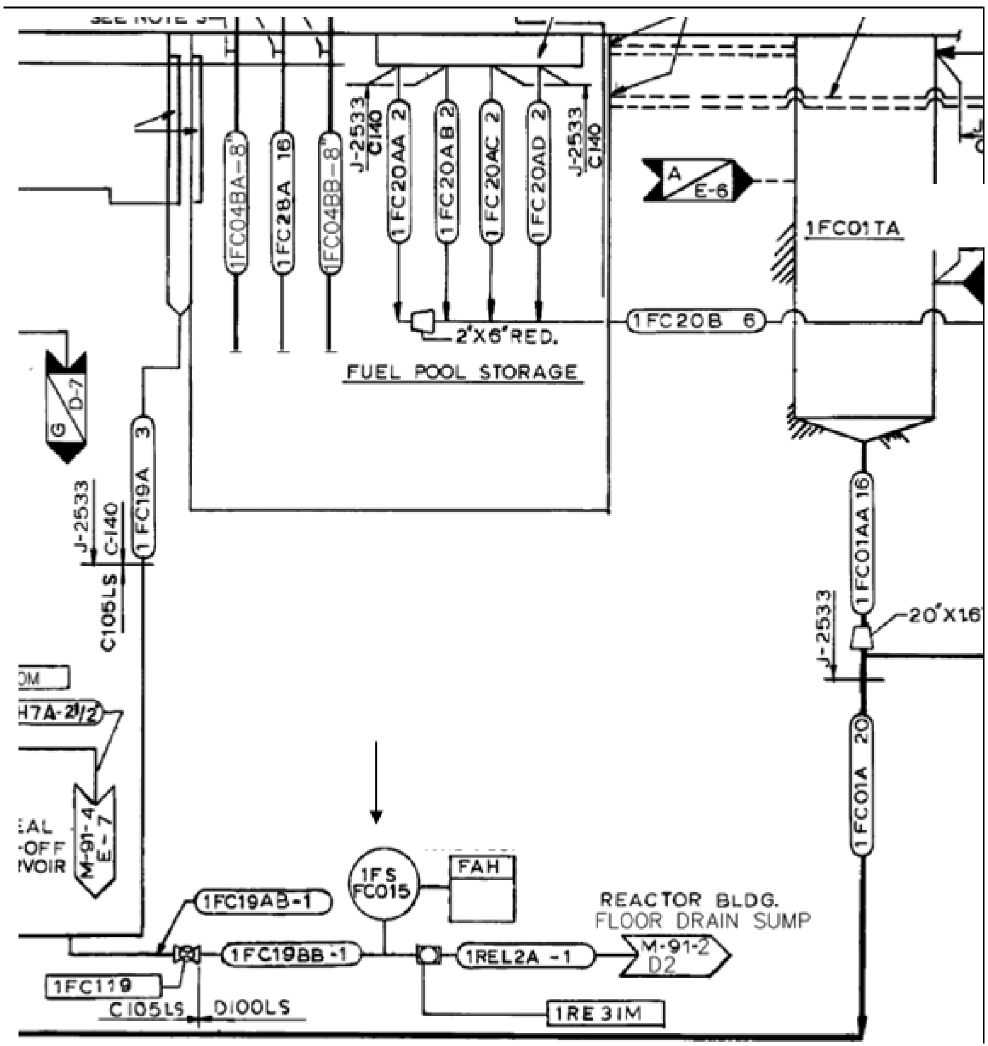
Comments:

Associated objective(s):

58 | **ID: 1271734** | **Points: 1.00**

Unit 1 is at 100% power.

Leakage flow has become just high enough to activate the Flow Switch 1FS-FC015 shown below.



Identify the location(s) where this Flow switch initiates an alarm.

A. In the MCR ONLY on panel 1PM13J

B. In the MCR ONLY on panel 1H13-P601.

C. In the Turbine Building at the Fuel Pool Cooling Panel ONLY

D. In the MCR AND in the Turbine Building at the Fuel Pool Cooling Panel

Answer: B

NOTE:
Question 58 was deleted from the exam based upon the outcome of a post-exam appeal review.

Answer Explanation

Explanation: Picture shows flow switch 1FS-FC015 which monitors Fuel Pool Gate Leakage. This switch provides an alarm as SER R-Point 0540 and on panel 1H13-P601-C207.

Distractor 1 is incorrect: Flow switch 1FS-FC015 monitors Fuel Pool Gate Leakage NOT Floor Drain Sump level. Plausible because the Reactor Building South Floor Drain Sump Trouble alarms on 1PM13J-A in the MCR, and the P&ID indicates that flow is to a Reactor Building Floor Drain Sump.

Distractor 2 is incorrect: Incorrect location but plausible because the primary Fuel Pool Cooling control panel is located 663 TB basement including the controls for the system and associated alarms.

Distractor 3 is incorrect: Alarms in the MCR; however, the Fuel Pool Cooling control panel in the Turbine Building is the incorrect location. Plausible because the primary control Fuel Pool Cooling control panel is located in the 663 TB basement including the controls for the system and associated alarms.

Reference: Drawing M-98 Sheet 1, LOR-1H13-P601-C207
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): RO
Tier: 2 **Group:** 2

K/A: 233000 Fuel Pool Cooling and Clean-up
A3.03 Ability to monitor automatic operations of the FUEL POOL COOLING AND CLEAN-UP including: System indicating lights and alarms

10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*

Question Source: Modified
Question History: LaSalle 13-1 NRC RO exam

Comments:

Associated objective(s):

59	ID: 1433051	Points: 1.00

Given the following:

- Refueling is in progress on Unit 2.
- All control rods are fully inserted.

While moving a fuel bundle to the core in the fuel transfer canal, a LOSS of Rod Control Management System (RCMS) power occurs.

Which of the following describes the expected PLANT RESPONSE and required OPERATOR ACTION in accordance with LOA-RM-201, Unit 2 RCMS Abnormal Situations?

A. Fuel movements are BLOCKED toward the core. RETURN the fuel bundle to the fuel pool.

B. Fuel movement is BLOCKED in either direction. LOWER the fuel bundle as LOW as possible in the transfer canal.

C. Fuel movements are BLOCKED from the core. CONTINUE to move the fuel bundle to its designated location.

D. Fuel movement is ALLOWED in either direction. CONTINUE to move the fuel bundle to its designated location.

Answer: A

Answer Explanation

Correct Answer: Fuel movements are prohibited toward the core; return the fuel bundle to the fuel pool. - A loss of RCMS causes the fuel interlocks to no longer see all rods in, stopping movement toward the core.

Distractor 1 is incorrect. Plausible if the candidate believes that fuel movements are prohibited in both directions on a loss of RCMS.

Distractor 2 is incorrect. Plausible if the candidate believes that fuel movements are prohibited in from the core on a loss of RCMS.

Distractor 3 is incorrect. Plausible if the candidate does not realize that even though all rods are in, the refueling interlocks cannot see this with a loss of RCMS.

Reference: LOA-RM-201, Unit 2 RCMS Abnormal Situations Rev. 027
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 2 **Group:** 2
KA: 234000 Fuel Handling Equipment
A2.03 Ability to (a) predict the impacts of the following on the FUEL HANDLING EQUIPMENT ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of electrical power
10 CFR Part 55 Content: 41.5
SRO Justification: *N/A*
Question Source: New
Question History: N/A

Comments:

Associated objective(s):

60	ID: 1260015	Points: 1.00

Unit 1 was operating at rated power when a complete loss of Main Generator Stator Cooling occurred:

- Reactor Recirc Flow was lowered to a minimum.
- Reactor Power is 40%.
- One Main Turbine Bypass valve is 30% open.

Given these conditions:

(1) What is the status of the Main Generator?
(2) What are the correct operator actions?

A. (1) A turbine runback is occurring
(2) Scram and carry out actions per LGP-3-2, Reactor Scram.

B. (1) A turbine runback is occurring
(2) Reduce reactor power with CRAM arrays and then per LGP-2-1, Normal Unit Shutdown.

C. (1) Stator water conductivity is rising
(2) Dispatch an operator to monitor stator water conductivity.

D. (1) Stator water conductivity is rising
(2) Trip the main turbine if conductivity exceeds 0.5 µmho/cm.

Answer: A

Answer Explanation

Explanation: Since power is greater than 20% with a loss of Generator Stator Cooling, a manual scram is required.

Distractor 1 is plausible because bypass valves opening initiates a reactivity transient due to reduced Feedwater heating and CRAM rods are inserted for various reactivity transients (e.g. a Recirc Flow Control valve failed open, or power reductions with Recirc at its lower flow limit), and a normal shutdown is a conservative action with the generator operating without stator cooling.

Distractor 2 is plausible because high conductivity is symptomatic of stator water cooling degradation and the amount of time the generator can remain online following a loss of stator water cooling is determined by the conductivity immediately preceding the event.

Distractor 3 is plausible because the main turbine is manually tripped within three minutes on a loss of stator cooling if conductivity immediately PRECEEDING the event was greater than 0.5 □mhos
runback, the turbine is tripped if conductivity rises to 9.9 □mh
scram is required prior to tripping the main turbine.

Reference: LOA-GC-101, UNIT 1 GENERATOR STATOR COOLING ABNORMAL, Revision 10, Page 4, Step B.1.2.

Reference provided during examination: N/A

Cognitive level: High

K/A: 245000 Main Turbine Generator and Auxiliary Systems

K1.06 Knowledge of the physical connections and/or cause-effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following: Component cooling water systems

10 CFR Part 55 Content: 41.2

SRO Justification: *N/A*

Question Source: Bank

Question History: N/A

Comments:

Associated objective(s):

61	ID: 1433002	Points: 1.00

Unit 1 is operating at 100% power:

- Offgas Train "A" is in service.
- A leak develops in the "A" SJAE Condenser tubes.

What will be the effect of the leak on plant conditions, if any?

A. Offgas flow will rise.

B. Main Condenser backpressure will degrade

C. Main Condenser backpressure will remain the same

D. An Offgas fire upstream of the Recombiner will occur

Answer: B

Note
Applicant Question: how big is leak? Pinhole? Large failure?
Response Given: significant leak.

Answer Explanation

Correct answer: Main Condenser backpressure will rise. – Loss of cooling water to the SJAE condenser and OG results in reduced SJAE efficiency and lowering condenser vacuum.

Distractor 1 is incorrect: Offgas flow will rise. – Offgas flow will lower due to the higher pressure condensate restricting the Offgas flow path.

Distractor 2: Main Condenser backpressure will remain the same. – Incorrect because loss of cooling water to the SJAE condenser and OG results in reduced SJAE efficiency, degrading condenser vacuum. Distractor is plausible since a leak in the SJAE condenser tubes does not directly affect the air removal capability of the SJAE, but will instead reduce the cooling efficiency and subsequent overall air removal efficiency.

Distractor 3 is incorrect: An Off Gas fire upstream of the Recombiner. – The probable cause of the fire is leakage through the isolation valve on the cross-over piping of the standby train. This would allow a flammable mixture to diffuse into the standby train until reaching the Recombiner.

References: P&ID M-88 Off Gas System and P&ID M-58 Condensate System. LOA-OG-101, Unit 1 Off Gas System Abnormal, Rev. 19.
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 2 **Group:** 2
KA: 256000 Reactor Condensate System
K3.08 Knowledge of the effect that a loss or malfunction of the REACTOR CONDENSATE SYSTEM will have on following: SJAE
10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*
Question Source: New
Question History: N/A

Comments:

Associated objective(s):

62	ID: 1271742	Points: 1.00

Which one of the following will cause the Feedwater Regulating Valve to transfer from automatic to manual?

A. If total steam line flow is <19%.

B. If 3 out of 4 level signals have failed.

C. If deviation occurs in 3 out of 4 steam line flow signals.

D. If a hardware failure of any of the FW pump flow signals occur coincident with a Total FW Header Flow Error.

Answer: B

Answer Explanation

Explanation: If a level signal failure exists (3 out of 4 level signals have failed) the FRV will transfer from automatic to manual without operator action.

Distractor 1 is incorrect: If total Steam line Flow is <19%, the final control elements are automatically transferred from Three-Element to Single-Element control. This is a plausible distractor because it is a signal that initiates automatic action to prevent a level transient.

Distractor 2 is incorrect: If deviation occurs in 3 out of 4 steam line flow signals, the final control elements are automatically transferred from Three-Element to Single-Element control. This is a plausible distractor because it is a signal that initiates automatic action to prevent a level transient.

Distractor 3 is incorrect: If a hardware failure of any of the FW pump flow signals occurs coincident with a Total FW Header Flow Error, the final control elements are automatically transferred from Three-Element to Single-Element control. This is a plausible distractor because it is a signal that initiates automatic action to prevent a level transient.

Reference: LOP-FW-16, Rev 35, 1(2)DS001 Operator Station Alarm Message Interpretation, page 18

Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): RO

Tier: 2 **Group:** 2

K/A: 259001 Reactor Feedwater System

K6.07 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR FEEDWATER SYSTEM: Reactor water level control system

10 CFR Part 55 Content: 41.7

SRO Justification: *N/A*

Question Source: New

Question History: N/A

Comments:

Associated objective(s):

63 **ID: 1432969** **Points: 1.00**

Main Condenser Offgas Technical Specification 3.7.6 limits the gross gamma activity rate of the noble gasses measured prior to the holdup line.

When is this Technical Specification applicable?

A. MODE 1 ONLY.

B. MODES 1 and 2 ONLY.

C. MODE 1, MODES 2 and 3 with ANY main steam line NOT isolated and the steam jet air ejector in operation.

D. MODE 1, MODES 2 and 3 with ANY main steam line NOT isolated and the Off Gas Mechanical Vacuum Pump in operation.

Answer: C

Answer Explanation
Correct answer: MODE 1, MODES 2 and 3 with ANY main steam line not isolated and steam jet air ejector (SJAE) in operation. **Distractor 1** is incorrect: The Tech Spec is applicable if there is a possibility of producing steam that will reach the Offgas system. **Distractor 2:** Also applicable if there is a possibility of producing steam and there is any possibility of it getting to the Offgas system. **Distractor 3:** Not applicable due to the Mechanical Vacuum pump in operation and not the SJAE. **Reference:** Technical Specification 3.7.6 **Reference provided during examination:** N/A **Cognitive level:** memory **Level (RO/SRO):** RO **Tier:** 2 **Group:** 2 **K/A: 271000 Offgas System** **Generic 2.40 Ability to apply Technical Specifications for a system.** **10 CFR Part 55 Content: 41.10** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

64	ID: 1455169	Points: 1.00

Both units are operating at 100% power:

LOR-2PM10J-B303, Fire Protection Alarm Panel Trouble annunciates
Fire Detection Display (FDD) indicates the following:
- 2A Diesel Generator Room CO_2 Trouble

Two minutes later, the EO dispatched to investigate the alarms reports:
- Local Panel 0CO02JB in the Diesel Corridor has a Zone 1 Alarm
- The Diesel Corridor red beacon is light is lit
- There is smoke in the vicinity of the 2A DG room.

Complete the following two statements.

_____(1)_____ have sensed a fire. The DG room CO_2 system ________(2)________.

A. (1) Smoke detectors
(2) must be MANUALLY actuated.

B. (1) Heat detectors
(2) must be MANUALLY actuated.

C. (1) Heat detectors
(2) should have AUTOMATICALLY actuated

D. (1) Smoke detectors
(2) should have AUTOMATICALLY actuated.

Answer: C

Answer Explanation

Explanation: Heat detectors have sensed a fire. The CO_2 system should have automatically actuated.

Distractor 1 is plausible because there is smoke in the vicinity of the 2A DG room. Also, the system can be manually actuated.

Distractor 2 is plausible because the system can be manually actuated.

Distractor 3 is plausible because there is smoke in the vicinity of the 2A DG room.

Reference: LOA-FP-201, Rev 35, UNIT 2 Fire Protection System Abnormal

Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): RO

Tier: 2 **Group:** 2

K/A: 286000 Fire Protection System

K.5.1 Knowledge of the operational implications of the following concepts as they apply to FIRE ROTECTION SYSTEM: Heat detection

10 CFR Part 55 Content: 41.5

SRO Justification: *N/A*

Question Source: Bank

Question History: N/A

Comments:

Associated objective(s):

65	ID: 1432951	Points: 1.00

Unit 2 is operating at 100% power.

- A steam leak develops on the body-to bonnet joint of 2B21-F028A, 2A Outboard MSIV.

Which of the following alarms is expected?

2H13-P601-...

A. E407, Leak Detection Pipe Tunnel Main Steam Line Diff. Temperature High

B. C507, Leak Detection MSIV Inboard Valve Stem Leakage Temperature High

C. E401, Reactor Core Isolation Cooling Pipe Route Equipment Area Temperature High

D. C211, Leak Detection Reactor Water Cleanup System ‘A’ Ambient Temperature High

Answer: A

Answer Explanation

Explanation: Leak Detection Pipe Tunnel Main Steam Line Diff. Temperature High is correct. Outboard MSIVs are in the steam tunnel.

Distractor 1: The MSIVs are located on either side of the Drywell wall. The inboard MSIVs are located in the drywell, and the outboard MSIVs are located in the upper steam tunnel in the outboard MSIV Room with only the Drywell wall between them. The distractor is plausible because there is little physical distance between the valves, and the inboard MSIVs do have a stem leak-off high temperature alarm. The distractor is incorrect because the inboard MSIV would not normally detect leakage on the outboard MSIV.

Distractor 2 is plausible because the Main Steam Line Tunnel and the RCIC pipe tunnel both have high temperature alarms that are designed to detect steam leakage, and both tunnels start at the same elevation. The distractor is incorrect because leakage in the Main Steam Line Tunnel would not normally result in a RCIC pipe tunnel temperature rise.

Distractor 3 is plausible because the Reactor Water Cleanup System is located above the outboard MSIV Room, and RT piping goes into the outboard MSIV Room where it connects to the Feedwater header that also passes through the Steam Tunnel. Also, ventilation flows from the RT equipment and valve rooms to the upper steam tunnel (outboard MSIV Room). The distractor is incorrect because normal ventilation flow is from the RT areas to the steam tunnel.

Reference: 2H13-P601-E407, Leak Detection Pipe Tunnel Main Steam Line Diff. Temperature High (Div II), Rev. 5; M-155-2, PCIS Leak Detection System
Reference provided during examination: N/A
Cognitive level: memory
Level (RO/SRO): RO
Tier: 2 **Group:** 2
K/A: 290001 Secondary Containment
A4.02 Ability to manually operate and/or monitor in the control room: Reactor building area temperatures: Plant-Specific
10 CFR Part 55 Content: 41.7
SRO Justification: *N/A*
Question Source: New
Question History: N/A

Comments:

Associated objective(s):

66	ID: 1433170	Points: 1.00

When faced with reactor behavior that is clearly abnormal, NSOs MUST

A. request a peer check before taking any further action.

B. take conservative action, including reducing power or initiating a reactor scram.

C. take conservative action, reactivity should only be changed in a slow, carefully controlled manner.

D. request guidance from an SRO or Qualified Nuclear Engineer (QNE) before taking any further action.

Answer: B

Answer Explanation

Explanation: When faced with unexpected or anomalous behavior of the reactor, or its control
systems, take conservative action, including reducing power or a initiating a reactor scram without hesitation.

Distractor 1 is plausible because peer checking is a regularly utilized human performance tool but is not required prior to taking conservative action.

Distractor 2 is plausible because OP-AA-101-111-1001 states that plant procedures require reactivity be changed only in a deliberate, carefully controlled manner; however, this is an abnormal situation and prompt conservative action is required.

Distractor 3 is incorrect because unexpected or anomalous reactor response requires the NSO to take conservative action without hesitation. The distractor is plausible because NQE guidance is normally available during startups and other significant reactivity manipulations; also, OP-AA-101-111-1001 states that operations personnel do not proceed in the face of uncertainty, but instead place the plant in a safe condition and then obtain the appropriate guidance before proceeding.

Reference: OP-AA-101-111-1001, Rev 17, Operations Standards and Expectations, Section 4.11. Conservative Decision-Making (CM-1)
Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 3

KA: 2.1.39 Knowledge of conservative decision making practices.

10 CFR Part 55 Content: 41.1
SRO Justification: *N/A*

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

67	ID: 1433171	Points: 1.00

Unit 1 was operating at 100% power when a reactor scram occurred:

- Following the SCRAM, the PCIS Status Display for Isolation Group V indicates RED.

What is the status of the RWCU isolation valves?

A. An isolation signal WAS received; all RWCU isolation valves are CLOSED.

B. An isolation signal was NOT received; all RWCU isolation valves are CLOSED.

C. An isolation signal WAS received; at least one RWCU isolation valve is OPEN.

D. An isolation signal was NOT received; at least one RWCU isolation valve is OPEN.

Answer: C

Answer Explanation

Explanation: Red is the PCIS Status Display color if an isolation signal is received and a valve remains open.

Distractor 1 is plausible because an isolation signal was received and for many valves in the plant, red position indicating lights are used to indicate a valve is closed.

Distractor 2 is plausible because an isolation signal was received and for many valves in the plant red lamps indicate a valve is closed.

Distractor 3 is plausible because a red lamp indicates that at least one RWCU isolation valve is OPEN when an isolation signal was received.

Reference: LOP-CX-06, Primary Containment Isolation Status Display
Reference given during the exam: N/A

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 3

KA: 2.1.19 Ability to use plant computers to evaluate system or component status.

10 CFR Part 55 Content: 41.10
SRO Justification: *N/A*

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

68	ID: 1433172	Points: 1.00

A reactor startup following a 21 day refueling outage is in progress on Unit 2:

- The NSO determines the reactor is critical and stops rod withdrawal.
- Reactor power continues to rise with no rod motion.
- One hour later, reactor power turns and begins to lower.

In accordance with LGP 1-1 Attachment L, Precautions to Be Used Before, During, and After Pulling Critical

(1) Why did reactor power start lowering?
(2) Which of the following would indicate the reactor has returned to subcritical?

A. (1) Moderator temperature lowered.
(2) Multiple SRM or IRM readings continuously lowering without rod insertions ONLY.

B. (1) The reactor reached the point of adding heat.
(2) Multiple SRM or IRM readings continuously lowering without rod insertions ONLY.

C. (1) Moderator temperature lowered.
(2) Multiple SRM or IRM readings continuously lowering without rod insertions
AND
Multiple IRM downscale readings
AND
Multiple IRMs are below the levels established when initially at the Point of Adding Heat.

D. (1) The reactor reached the point of adding heat.
(2) Multiple SRM or IRM readings continuously lowering without rod insertions
AND
Multiple IRMs have been down ranged through at least two ranges
AND
Multiple IRMs are below the levels established when initially at the Point of Adding Heat.

Answer: D

Answer Explanation

Explanation:
(1) The reactor reached the point of adding heat.
(2) Multiple SRM or IRM readings continuously lowering without rod insertions
AND
Multiple IRMs have been down ranged through at least two ranges
AND
Multiple IRMs are below the levels established when initially at the Point of Adding Heat.

The POAH is shown by reactor power rising and then starting to lower with no rod motion.

Distractor 1:
(1) Moderator temperature lowered.
(2) Multiple SRM or IRM readings continuously lowering without rod insertions only**.**

Approaching the POAH and Recirc pump heat is adding reactivity. If moderator temperature did lower, reactor power would rise, not lower. This is the indication for subcriticality prior to the POAH. The stem indicates that the POAH was reached.

Distractor 2:
(1) The reactor reached the point of adding heat.
(2) Multiple SRM or IRM readings continuously lowering without rod insertions only.

This is the indication for subcriticality prior to the POAH. The stem indicates that the POAH was reached. Plausible if the candidate does not understand this.

Distractor 3:
(1) Moderator temperature lowered.
(2) Multiple SRM or IRM readings continuously lowering without rod insertions
AND
Multiple IRM downscale readings
AND
Multiple IRMs are below the levels established when initially at the Point of Adding Heat.

Approaching the POAH and Recirc pump heat is adding reactivity. If moderator temperature did lower, reactor power would rise, not lower. Plausible if the candidate does not understand these concepts.

Reference: LGP 1-1, Rev 115, Attachment L, Precautions To Be Used Before, During, and After Pulling Critical.
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO
Tier: 3
KA: 2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc. RO 4.1 SRO 4.3
10 CFR Part 55 Content: 41.10
SRO Justification: *N/A*
Question Source: New
Question History: N/A
Comments:
Associated objective(s):

69	ID: 1455170	Points: 1.00

Unit 2 is operating at rated power when a Loss of Feedwater Heating occurred.

The QNE reports Minimum Critical Power Ratio (MCPR) is 1.13.

This value of MCPR is...

A. within the Safety Limit; NO operator actions are required.

B. at the Safety Limit; it is required to insert in-sequence rods to obtain adequate margin.

C. in violation of the Safety Limit; it is required to insert all insertable control rods within two hours.

D. in violation of the Safety Limit; it is required to reduce reactor power to less than 25% RTP within four hours.

Answer: C

Answer Explanation
Explanation: A loss of feedwater heating is an inadvertent reactivity addition, the operational implications of which are exceeding thermal limits and the required actions to address them. MCPR limit on Unit 2 with both Recirc loops is >1.14. Required actions are to restore compliance with all safety limits and insert all insertable control rods within two hours (T.S. 2.0) **Distractor 1** is incorrect: Below the required MCPR limit for Unit 2. Plausible because Unit 1's MCPR limit is 1.13. **Distractor 2** is incorrect: Below the required MCPR limit for Unit 2 however this is at the safety limit for Unit 1 with both Recirc loops. Plausible because Unit 1's MCPR limit is 1.13. **Distractor 3** is incorrect: Must insert all insertable control rods within 2 hours; not reduce power less than 25%. Plausible because this is an action from LCO 3.2.2 **Reference:** T.S. 2.0 **Reference provided during examination:** None **Cognitive level:** Memory **Level (RO/SRO):** RO **K/A: 2.2.22 Knowledge of limiting conditions for operations and safety limits.** **10 CFR Part 55 Content: 41.5** **SRO Justification**: N/A **Question source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

70	ID: 1454708	Points: 1.00

During the last performance of a weekly surveillance test, the surveillance was started at 1300 and was completed at 1700 on Tuesday, October 4th.

What is the LATEST this test can be completed without the surveillance exceeding its allowed completion time?

A. 1300 on Wednesday, October 12th.

B. 0100 on Thursday, October 13th.

C. 0700 on Thursday, October 13th.

D. 1300 on Tuesday, October 18th.

Answer: C

Answer Explanation
Explanation: Correct – Started at 0700 on Thursday, October 13th. – SR 3.0.2 – The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. Since the previous surveillance was completed at 1300 on Tuesday, October 4th and it is a weekly surveillance, the 1.25 times is 42 hrs (168 x 0.25), so the surveillance is due by 0700 on Thursday, October 13th. **Distractor 1:** 1300 on Wednesday, October 12th. Plausible because there are surveillance that must be completed within 24 hrs of their due time; this is correct for surveillances of 24 hrs frequency or less. **Distractor 2:** 0100 on Thursday, October 13th. Plausible because there are Tech Spec LCO completion times that must be completed within 36 hrs. **Distractor 3:** 1300 on Tuesday, October 18th. Plausible if the candidate gets the completion time confused with the extension time for a missed surveillance IAW SR 3.0.3, which would be 7 days for a weekly surveillance. **Reference:** SR 3.0.2 – The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** RO **Tier:** 3 **KA: 2.2.23 Ability to track Technical Specification limiting conditions for operations.** **RO 3.1 SRO 4.6** **10 CFR Part 55 Content: 41.10** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):** 201.01.e Given a set of plant conditions, identify completion times.

71	ID: 1262106	Points: 1.00

Unit 1 is operating at rated power.

Which of the following components could be eligible to have independent verification waived?

A Danger tag to be hung on the...

A. 1AP57E-F4, breaker for 1WS113 at MCC 131Y-1.

B. 1FC095A, Fuel Pool Filter Demineralizer Hold Pump Suction valve.

C. 0CO01C CO_2 compressor pressure switch isolation valve, 0CO039.

D. 1CD035A, condensate polishing 1A inlet valve control switch at 1PL05J, TB 687.

Answer: B

Answer Explanation

Explanation: A Danger tag to be hung on 1FC095A, Fuel Pool Filter Demineralizer Hold Pump Suction valve handwheel is correct. The Shift manager may waive verification requirements for ALARA concerns.

A, B and D are incorrect. The components in these choices are in areas where radiation levels are not a factor. The candidate must understand that these components are in very low, or no, radiation areas and independent verification would not be waived.
Reference: HU-AA-101, Rev 9, Human Performance Tools and Verification Procedures
Reference provided during examination: N/A
Cognitive level: Memory
Level (RO/SRO): RO
Tier: 3 Group:
K/A: 2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.
10 CFR Part 55 Content:41.12
SRO Justification: N/A
Question Source: LaSalle Bank
Question History: 08-1 NRC RO Exam
Comments:
Associated objective(s):

72	ID: 1433230	Points: 1.00

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

- 1H13-P601-F205, Div I Fuel Pool Radiation High High.
- 1H13-P601-E205, Div II Fuel Pool Radiation High High.

Which of the following describes the cause of these alarms?

A. Fuel Element Failure

B. Reactor Water Cleanup leak

C. Recirculation Pump seal failure

D. Airborne contamination in the vicinity of the Fuel Pool

Answer: D

<u>Note</u>
Applicant Question: fuel element failure where?
Response Given: fuel element failure in the RPV.

Answer Explanation

Explanation: Airborne contamination in the vicinity of the Fuel Pool. The Reactor Building Ventilation System isolates and the Standby Gas Treatment System initiates on high radiation in the exhaust of the Fuel Pool Ventilation System or Reactor Building Ventilation System.

Distractor 1 is incorrect: A fuel failure would cause off gas rad levels to rise.

Distractor 2 is incorrect: This leak is in the drywell and would only cause containment rad level changes.

Distractor 3 is incorrect: This leak is in the drywell and would only cause containment rad level changes.

Reference: LOA-AR-101, UNIT 1 Area Radiation Monitoring System Abnormal
Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): RO
Tier: 3

KA: 2.3.5 – Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring |equipment, etc. IMPORTANCE RO 2.9 SRO 2.9

10 CFR Part 55 Content: 41.11
SRO Justification: *N/A*

Question Source: New
Question History: N/A

Comments:

Associated objective(s):

73 **ID: 1433231** **Points: 1.00**

The Unit is operating at rated power.

While performing a fire protection system mechanical checklist for valves in the reactor building, the operator must participate in an HRA Pre-Job Brief from Radiation Protection prior to entering the…

A. Refuel Floor.

B. HPCS Pump Room.

C. RCIC Pump Room.

D. RHR ‘C’ Pump Room.

Answer: D

Answer Explanation
Explanation: 'C' RHR Pump Room is a High Rad Area during operation and as such would require a high rad brief from RP prior to entering. Of the 4 areas among the answer choices, only the 'C' RHR Pump Room is a Locked HRA. **Distractor 1** is incorrect: The Refuel Floor is a radiation area, so no pre-job brief is required prior to entry. Plausible because it is located in the Reactor Building. **Distractor 2** is incorrect: The HPCS pump room is a radiation area, so no pre-job brief is required prior to entry. Plausible because it is located in the Reactor Building. **Distractor 3** is incorrect: RCIC is a radiation area, so no pre-job brief is required prior to entry. Plausible because RCIC is located in the Reactor Building and because RCIC has been a high radiation area in the past. **Reference:** RP-AA-403, Rev 8, Administration of the Radiation Work Permit Program **Reference provided during examination:** N/A **Cognitive level:** Memory **Level (RO/SRO):** RO **Tier:** 3 **KA: 2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions**. **10 CFR Part 55 Content: 41.12** ***SRO Justification:*** *N/A* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

74	ID: 1454977	Points: 1.00

Unit 2 is operating at 100% power

The following conditions are noted after a transient:

- Reactor pressure is slightly lower.
- Generator MWe is slightly lower.
- Feedwater temperature is lower.
- Indicated Feedwater Flow is greater than indicated steam flow.

Which of the following caused this transient?

A. A bypass valve opening.

B. A stuck open Safety Relief Valve.

C. A Turbine Control Valve drifting closed.

D. Isolation of Extraction Steam to a Feedwater Heater.

Answer: B

Answer Explanation

Explanation: A stuck open relief valve will drop pressure slightly as EHC compensates for the open relief valve by closing TCVs, but not enough to return pressure to the previous value.
A stuck open relief valve bypasses the Main Turbine, which lowers Generator MWe.
A stuck open relief valve lowers steam flow to the Main Turbine and extraction steam flow from the Main Turbine, which lowers Feedwater Temperature.
A stuck open relief valve reduces steam flow past the Main Steam Flow instruments.

From LOA-SRV-101, UNIT 1 Stuck Open Safety Relief Valve
A. SYMPTOMS/ENTRY CONDITIONS
A.6 Generator load drop with no change in Reactor power.
A.7 Drop in steam flow.

Distractor 1 is incorrect: A bypass valve opening would not affect steam flow past the Main Steam Flow instruments. Plausible because it lowers Feedwater temperature, bypasses the Main Turbine, which lowers Generator MWe and may cause a slight pressure transient.
Distractor 2 is incorrect: A TCV drifting closed would result in a slight pressure rise and would not result in a drop in steam flow past the Main Stem flow detectors because EHC would compensate for any pressure rise as a result of the valve failing closed.
Distractor 3 is incorrect: Isolation of a Feedwater Heater raises power, which raises MWe. Plausible because a loss of extraction steam lowers Feedwater temperature.
Reference: LOA-SRV-101, UNIT 1 Stuck Open Safety Relief Valve
Reference provided during examination: N/A
Cognitive level: High
Level (RO/SRO): RO or SRO
Tier: 3 **Group:**
K/A: 2.4.47 **Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.**
10 CFR Part 55 Content: 41.10 / 43.5 / 45.12
Question Source: New
Question History: N/A
Comments:
Associated objective(s)

75	ID: 1262779	Points: 1.00

While executing a LaSalle Emergency Operating Procedures (LGA), what must be done if ANOTHER entry condition occurs for the same procedure OR the initial entry condition re-occurs?

A. Immediately return to the start of the procedure and execute the procedure.

B. Immediately return to the start of the current procedure leg(s) ONLY and execute the procedure.

C. Continue with the current procedure leg(s) ONLY; returning to the start of the procedure is NOT required.

D. When you have exited the current procedure leg(s), THEN return to the start of the procedure and execute the procedure.

Answer: A

Answer Explanation
Explanation: LGAs developed from these PSTGs shall be entered whenever a defined entry condition occurs or an explicit direction to do so is encountered, even if the procedure has already been entered. **Distractor 1** is plausible because if an operator returns to the start of the procedure due to a recurrence of the same initial entry conditions, it is likely that the operator will re-execute the original procedure legs. **Distractor 2** is plausible because for most procedures, one proceeds through the procedure from beginning to end until the procedure is complete. **Distractor 3** plausible because for most procedures, one proceeds through the procedure from beginning to end until the procedure is complete. **Reference:** LPGP-PSTG-01S01, Revision 2, PLANT SPECIFIC TECHNICAL GUIDELINES SECTION 1-INTRODUCTION **Reference provided during examination:** N/A **Cognitive level:** Memory **Level (RO/SRO):** RO **Tier:** 3 **K/A: 2.4.14 Knowledge of general guidelines for EOP usage.** **10 CFR Part 55 Content: 41.10** ***SRO Justification:*** *N/A* **Question Source:** Bank **Question history:** Dresden exam bank **Comments:** **Associated objective(s):**

EXAMINATION ANSWER KEY

15-1 NRC SRO EXAM

1	ID: 1262120	Points: 1.00

Unit 1 was at 20% power when the Main Turbine tripped.

- The Main Condenser is at 4 inches of backpressure Hg.
- At 1300 RPM, annunciator 1PM02J-A401, TURB GEN VIBR HI is in alarm, and PPC Bearing High Vibration alarms.
- All main turbine bearings are between 5 and 6 mils except bearings 9 and 10 which are indicating 11mils by the PPC indications.

The alarms are _______(1)_______ with the bearing indications.
The Unit Supervisor will direct the operators to _________(2)

A. (1) CONSISTENT
(2) THROTTLE OPEN 1TE111, CONDENSER VACUUM BREAKER, per LOA-TG-101, Unit 1 Turbine Generator

B. (1) CONSISTENT
(2) VERIFY 1TE111, CONDENSER VACUUM BREAKER, CLOSED per LGP-3-2, Reactor Scram Procedure

C. (1) NOT CONSISTENT
(2) VERIFY 1TE111, CONDENSER VACUUM BREAKER,CLOSED per LOA-TG-101, Unit 1 Turbine Generator

D. (1) NOT CONSISTENT
(2) THROTTLE OPEN 1TE111, CONDENSER VACUUM BREAKER, per LGP-3-2, Reactor Scram Procedure

Answer: A

Answer Explanation

Answer explanation

The alarm setpoint for LOR-1PM02J-A401 is currently set at 7 mils. When the turbine is tripped, the entry condition for LOA-TG-101 are met and per LOA-TG-101 step B.1.20 the crew should throttle open the vacuum breaker until back pressure reaches 5 inches of HG back pressure or until the turbine is past critical speed range of 900 to 1300 RPM. This will cause the turbine to slow at a faster rate.

Distractor 1 is plausible because the alarm is valid and because LGP 3-2, Reactor Scram, also directs the Vacuum breaker to be opened on high vibes when at 1300 RPM.

Distractor 2 is incorrect: The alarm is valid and the correct action per the LOA-TG-101 is to throttle the 1TE111 until past critical speeds. The distractor is plausible because the Unit Supervisor will enter and direct actions per LOA-TG-101.

Distractor 3 is plausible because the alarm is valid: The alarm is valid and because LGP 3-2, Reactor Scram, provides guidance to operate the 1TE111 on a turbine trip, and the correct action from LGP 3-2 is directed.

Reference: LOA-TG-101, Rev. 17, UNIT 1 Turbine Generator; LOR-1PM02J-A401, Rev. 002, U1 Turbine Generator Vibration High; LGP 3-2, Rev 072, Reactor Scram
Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): SRO
Tier: 1 **Group:** 1

K/A: 295005 Main Turbine Generator Trip
2.4.46 Ability to verify that the alarms are consistent with the plant conditions.

10 CFR Part 55 Content: 43.5
SRO Justification: 10 CFR 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Question Source: Bank
Question History: LaSalle ILT NRC 11-1

Comments:
Associated objective(s):

2	ID: 1454735	Points: 1.00

Both Units were at 100% power when a fire in the main control room required its evacuation.

- The crew has entered LOA-FX-101(201), Unit 1(2) Safe Shutdown with a Fire in the Control Room OR AEER.
- The initial actions in the control room for the Unit NSOs were completed prior to evacuation.
- It has been 20 minutes since the evacuation and RPV pressure control has yet to be fully established. The fire has just been reported extinguished.

(1) Which of the following is the correct method of RPV pressure control?
(2) What is the HIGHEST emergency action level that has been met?

A. (1) Operators will manually cycle the Relief Valves to establish a cooldown.
(2) A Site Area Emergency must be declared.

B. (1) Operators will manually initiate a cooldown using DEHC.
(2) A Site Area Emergency must be declared.

C. (1) Operators will manually initiate a cooldown using DEHC.
(2) An Alert must be declared.

D. (1) Operators will manually cycle the Relief Valves to establish a cooldown.
(2) An Alert must be declared.

Answer: A

Answer Explanation

Explanation: LOA-FX-101(201) directs the operators to insert a manual reactor scram before leaving the Main Control room and establishing a cooldown by manually actuating relief valve solenoids locally. Because control was not established within 15 minutes, the threshold values for HS2, Site Area Emergency (SAE) have been satisfied and a SAE must be declared.

Distractor 1 is incorrect because the control room has been evacuated making the normal DEHC controls unavailable. Distractor is plausible because DEHC is an automatic system located in the AER that does possess an automatic cooldown function that can be selected from the HMI under the Control, Reactor Cooldown screen. The initial actions of LOA-FX-101(201) does not have steps to align the DEHC system prior to evacuating the MCR; however, it does have actions to align other systems for local control. Also plausible because an Alert would have been declared due to the control room evacuation if control had been established within 15 minutes.

Distractor 2 is incorrect: A SAE must be declared because control of the plant has not been established within 15 minutes of control room evacuation. Distractor is plausible because the crew will cycle relief valves to establish a cooldown and because an Alert would have been declared due to the control room evacuation alone if control had been established within 15 minutes.

Distractor 3 is incorrect because the control room has been evacuated making the normal DEHC controls unavailable. Distractor is plausible because DEHC is an automatic system located in the AER that does possess an automatic cooldown function that can be selected from the HMI under the Control, Reactor Cooldown screen. The initial actions of LOA-FX-101(201) does not have steps to align the DEHC system prior to evacuating the MCR; however, it does have actions to align other systems for local control.

Reference: LOA-FX-101, Rev 027, UNIT 1 Safe Shutdown With A Fire In The Control Room or AEER and EP-AA-1005, Rev 1, Emergency Action Levels for LaSalle Station.

Reference provided during examination: Hot Matrix - EP-AA-1005, Addendum 3, Emergency Action Levels for LaSalle Station

Cognitive level: High

Level (RO/SRO): SRO

Tier: 1 Group: 1

KA: 295016 Control Room Abandonment

AA2.03 Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Reactor pressure

10 CFR Part 55 Content: 43.5

SRO Justification: *10 CFR 55.43(b)(5)*

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This action is unique to the SRO position. The question is linked to a learning objective that is specifically labeled in the lesson plan as being SRO-only

Question Source: New

Question History: N/A

Comments:

Associated objective(s):

3	ID: 1454733	Points: 1.00

If an irradiated fuel bundle is damaged in the fuel pool during refueling operations, the PRIMARY concern is a higher level of ________(1)__________ , and this condition will be made worse if _________(2)_________.

A. (1) External radiation exposure
(2) Fuel pool temperature is at the LOWER limit

B. (1) External radiation exposure
(2) Fuel pool temperature is at the UPPER limit

C. (1) Exposure to gaseous fission products
(2) Fuel pool level is at the LOWER limit

D. (1) Exposure to gaseous fission products
(2) Fuel pool level is at the UPPER limit

Answer: C

Answer Explanation

Explanation

If an irradiated fuel bundle is damaged, the primary concern is the release of gaseous fission products resulting in rising airborne contamination levels, and the condition is made worse with low fuel pool level because more of the iodine coming out of solution will make it to the surface of the fuel pool raising airborne contamination levels.

Distractor 1 is plausible because fuel pool temperatures at the lower limit would increase the solubility of fission product gasses allowing the gasses to accumulate in the fuel pool water raising external exposure for those nearest to the fuel pool.

Distractor 2 is plausible because high fuel pool temperature could lead to a rise in airborne contamination levels caused by increased evaporation from the fuel pool.

Distractor 3 is plausible because high fuel pool level could result in the fuel pool overflowing into the ducts of the Reactor Building ventilation system leading to the spread of contamination.

Reference: Tech Spec Basis 3.7.8 and 3.9.6; UFSAR 15.7.4; UFSAR 12.2.2.7, and UFSAR 9.1.1.3.2.1.1.b
Reference provided during examination: N/A

Cognitive level: Memory

Level (RO/SRO): SRO
Tier: 1 **Group:** 1

K/A: 295023 Refueling Accidents
A2:03 Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: Airborne Contamination Levels

10 CFR Part 55 Content: 43.4
SRO Justification: *10 CFR 55.43(b)(4)*
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Question source: New
Question history: N/A

Comments:

Associated objective(s):

4 **ID: 1433308** **Points: 1.00**

Unit 1 was operating at 100% power when:

- All running Circulating Water Pumps have tripped.
- A manual reactor scram is inserted.
- 3 control rods remain at position 48.

LGA-010, Failure to Scram, directs the Unit Supervisor to stabilize RPV pressure below 1059 psig using main turbine bypass valves.

What order will the Unit Supervisor give concerning pressure control?

A. Control reactor pressure 800 – 1000 psig with SRVs.

B. Control reactor pressure 800 – 1000 psig with bypass valves.

C. Commence a cooldown not to exceed 100 ^{o}F per hour with SRVs.

D. Commence a cooldown not to exceed 100 ^{o}F per hour with bypass valves.

Answer: A

Answer Explanation
Explanation: Control pressure 800 – 1000 psig with SRVs. With all Circulating Water pumps tripped, the main condenser is not available, so the crew must use the SRVs for pressure control. For the given conditions, more than 1 control rod is out, so the unit is in an ATWS; therefore, a cooldown cannot be performed and pressure must be stabilized in an appropriate band. **Distractor 1** is incorrect: With all Circulating Water pumps tripped, the main condenser is not available, so the crew cannot use bypass valves for pressure control; rather, they must use the SRVs. Distractor is plausible because if the Circulating Water pumps in service, pressure control using bypass valves would be allowed. **Distractor 2** is incorrect: For the given conditions, more than 1 control rod is out, so the unit is in an ATWS; therefore, a cooldown cannot be performed and pressure must be stabilized in an appropriate band. Distractor is plausible because under non-ATWS conditions with the Main Condenser unavailable, directing a cooldown utilizing SRVs would be correct. **Distractor 3** is plausible because under non-ATWS conditions with the Main Condenser unavailable, directing a cooldown utilizing bypass valves would be correct. **Reference:** LGA-010, REV 15, Failure to Scram and LPGP-PSTG-01S04B, Rev 4, Plant Specific Technical Guidelines Section 4B – RPV Control – Pressure Control **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** SRO **Tier:** 1 **Group:** 1 **KA: 295025 High Reactor Pressure** **G2.1.20 Ability to interpret and execute procedure steps.** **10 CFR Part 55 Content: 43.5** ***SRO Justification:*** *10 CFR 55.43(b)(5)* Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations including knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures. **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

5	ID: 1259804	Points: 1.00

Unit-2 was operating at 100% power:

- A rupture in the 2C RHR suction line has flooded the Reactor Building Raceway.
- Manual scram was inserted due to low Suppression Pool level.
- 14 Control Rods are at position 04.
- The MDRFP is maintaining level per the LGAs.
- All ECCS Pumps are in PTL.
- The Turbine Bypass Valves are maintaining reactor pressure at 920 psig.
- Suppression Pool Level has rapidly dropped to -20 feet.

What is the correct operator action?

A. Initiate ADS per LGA-006, ATWS Blowdown

B. Depressurize using Other Blowdown Systems per LGA-004, RPV Blowdown

C. Depressurize using Other Blowdown Systems per LGA-006, ATWS Blowdown

D. Depressurize using Alternate Pressure Control Systems per LGA-010, Failure to Scram

Answer: C

Answer Explanation
Explanation: Depressurize using Other Blowdown Systems per LGA-006, ATWS Blowdown. LGA-003 Pool Level leg requires a reactor scram and a Blowdown if Suppression Pool Level cannot be held above -12 feet. Following the scram, the Unit is in an ATWS condition; therefore, LGA-006, ATWS Blowdown is the correct procedure to utilize, and LGA-006 states that if level is less than -18 feet then do not initiate ADS, instead use other Blowdown Systems per Detail B2. Detail B2 states that SRVs are only to be used if SP Level is greater than -18 feet. **Distractor 1** is plausible because with Suppression Pool level NOT below -18 feet, Initiating ADS per LGA-006, ATWS Blowdown, would be the correct operator action. The distractor is incorrect because LGA-006 states that if level is less than -18 feet then do not initiate ADS, instead use other Blowdown Systems per Detail B2. Detail B2 states that SRVs are only to be used if SP Level is greater than -18 feet. **Distractor 2** is plausible because under non-ATWS conditions, depressurize using Other Blowdown Systems per LGA-004, RPV Blowdown, would be the correct operator action. The distractor is incorrect because LGA-006, ATWS Blowdown, is the correct procedure to utilize to blowdown during an ATWS. **Distractor 3** is plausible because LGA-010 is entered due to the ATWS condition and because under non-ATWS conditions depressurizing using Alternate Pressure Control Systems would be the correct operator action. The distractor is incorrect because LGA-006, ATWS Blowdown, is the correct procedure to utilize to blowdown during an ATWS. **Reference:** LGA-006, Revision 09, ATWS Blowdown, LGA-003, Revision 16 Primary Containment Control, and LPGP-PSTG-01S06, Rev 6, Plant Specific Technical Guidelines Section 6 - Secondary Containment Control. **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** SRO **Tier:** 1 **Group:** 1 **K/A: 295030 Low Suppression Pool Water Level** **A2.03 Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL:** Reactor pressure **10 CFR Part 55 Content: 43.5** ***SRO Justification:*** *10 CFR 55.43(b)(5)* Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

6	ID: 1433351	Points: 1.00

Unit 1 is in Mode 5, and irradiated fuel assemblies are being moved within the Reactor Pressure Vessel (RPV).

In accordance with LCO 3.9.6 Reactor Pressure Vessel (RPV) Water Level—Irradiated Fuel,

RPV cavity water level must be greater than or equal to _______(1)_________ above the RPV flange; otherwise, movement of irradiated fuel assemblies in the RPV must be IMMEDIATELY SUSPENDED to maintain sufficient water level to __________(2)_____________.

A. (1) 22 feet
(2) ensure 99.5% of the total iodine released from a damaged fuel assembly is retained in the water

B. (1) 23 feet
(2) ensure 99.5% of the total iodine released from a damaged fuel assembly is retained in the water

C. (1) 23 feet
(2) retain iodine fission product activity in the event of a fuel handling accident, keeping offsite doses within limits

D. (1) 22 feet
(2) retain iodine fission product activity in the event of a fuel handling accident, keeping offsite doses within limits

Answer: D

Answer Explanation
Explanation: RPV cavity water level must be greater than or equal to 22 feet above the RPV flange; otherwise, per Tech Spec 3.9.6, movement of irradiated fuel assemblies in the RPV must be IMMEDIATELY SUSPENDED to maintain sufficient water level to retain iodine fission product activity in the event of a fuel handling accident, keeping offsite doses within 10 CFR 50.67 limits, as modified by Regulatory Guide 1.183, July 2000. **Distractor 1** is plausible because the normal RPV water level of 23 feet allows a decontamination factor of 200 to be used in the accident analysis for iodine which is related to the assumption that 99.5% of the total iodine released from the pellet to the cladding gap of all the damaged fuel assembly rods is retained by the reactor cavity water. The distractor is incorrect because the basis for the Tech Spec minimum limit of 22 feet above the RPV flange is to maintain sufficient water level to retain iodine fission product activity in the event of a fuel handling accident, keeping offsite doses within 10 CFR 50.67 limits. **Distractor 2** is plausible because normal RPV water level during refueling is 23 feet above the RPV flange, and that level allows a decontamination factor of 200 to be used in the accident analysis for iodine which is related to the assumption that 99.5% of the total iodine released from the pellet to the cladding gap of all the damaged fuel assembly rods is retained by the reactor cavity water. The distractor is incorrect because 22 feet above the RPV flange is the Tech Spec minimum limit, and the basis for that limit is to maintain sufficient water level to retain iodine fission product activity in the event of a fuel handling accident, keeping offsite doses within 10 CFR 50.67 limits. **Distractor 3** is plausible because normal RPV water level during refueling is 23 feet above the RPV flange and because the Tech Spec limit of 22 feet is to maintain sufficient water level to retain iodine fission product activity in the event of a fuel handling accident, keeping offsite doses within 10 CFR 50.67 limits. **Reference:** Tech Spec 3.9.6 and 3.9.6 Tech Spec bases. **Reference provided during examination**: N/A **Cog:** Memory **Level (RO/SRO):** SRO **Tier:** 1 **Group:** 1 **KA: 295031 Reactor Low Water Level** **G 2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.** **SRO Justification:**10CFR55.43(b)(2) Facility operating limitations in Technical Specifications and their bases **10 CFR Part 55 Content:** 41.7 / 41.10 / 43.2 / 43.3 / 45.3 ***SRO Justification:*** *10 CFR 55.43(b)(5)* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

7	ID: 1454934	Points: 1.00

A reactor scram was inserted on Unit 2 due to an unisolable steam leak in the RCIC room:

- ONE (1) control rod is at position 36.
- Feed and condensate are maintaining RPV water level.
- The Unit 2 VR Exhaust line failed to isolate.
- The sum of readings on the Vent Stack and SBGT WRGMs is 1.0 E+09 uCi/sec and continues rising at the same rate.

What action will the Unit Supervisor direct NEXT?

A. Initiate ADS ONLY per LGA-004, RPV BLOWDOWN.

B. Open the Main Turbine Bypass Valves ONLY per LGA-004, RPV BLOWDOWN.

C. Terminate and prevent injection except Born, CRD and RCIC, and initiate ADS per LGA-006, ATWS. BLOWDOWN.

D. Terminate and prevent injection except Born, CRD and RCIC; open the Main Turbine Bypass Valves AND initiate ADS per LGA-006, ATWS BLOWDOWN.

Answer: A

Answer Explanation
Explanation: Initiate ADS only. With a release in progress and off site levels greater than those for a Site Emergency and rising, the blowdown criteria of LGA-009 have been reached. Only one rod is stuck out, so blowdown is per LGA-004, Rev 009, RPV Blowdown. Blowdown must be inside containment due to the potential to raise exposure offsite. **Distractor 1** is plausible because blowdown via the Main Turbine Bypass Valves is a correct operator action to direct if there were not an offsite release in progress. **Distractor 2** is incorrect: Only 1 control rod is out, so the unit is not in an ATWS, so it is not necessary to terminate and prevent injection. The distractor is plausible because terminate and prevent injection would be the correct action to direct if the unit were in an ATWS condition. **Distractor 3** is incorrect: Only 1 control rod is out, so the unit is not in an ATWS and it is not necessary to terminate and prevent injection. Distractor is plausible because terminate and prevent injection would be the correct action to direct if the unit were in an ATWS condition and because blowing down to the main condenser is an appropriate action to direct with no offsite release in progress. **References:** LGA-009, Rev 007, Radioactivity Release Control, LGA-006, Rev 009, ATWS Blowdown, LGA-004, Rev 009, RPV Blowdown, and EAL Hot Matrix - EP-AA-1005 Addendum 3, Rev 001, Emergency Action Levels for LaSalle Station **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** SRO **Tier:** 1 **Group:** 1 **KA: 295038 A2.01 Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: EA2.01 Off-site.** **10 CFR Part 55 Content: 43.5** **SRO Justification: 10CFR55.43(b)(5)** Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

8	ID: 1455174	Points: 1.00

Unit 1 is at full power with RPV pressure at 1025 psig.

What is the MAXIMUM allowable time to complete the required action in accordance with Tech Specs, and what is the basis of that action?

Reduce reactor steam dome pressure to < 1020 psig within...

A. 15 MINUTES, because the reactor coolant system pressure/temperature limit for brittle fracture is exceeded.

B. 15 MINUTES, because the initial steam dome pressure assumed in transient and accident analysis is exceeded.

C. 2 HOURS, because the reactor coolant system pressure/temperature limit for brittle fracture is exceeded.

D. 2 HOURS, because the initial steam dome pressure assumed in transient and accident analysis is exceeded.

Answer: B

Answer Explanation

Explanation: LCO 3.4.12, Reactor Steam Dome Pressure, Condition A requires steam dome pressure to be restored to within limit (< 1020 psig) within 15 minutes. This is because it is an initial condition for the RCS over-pressurization analysis and the transient and accident analysis used to determine thermal limits.

Distractor 1: Plausible because a RPV high pressure condition can result in exceeded P/T limits for brittle fracture per LCO 3.4.11 which is limiting at high pressure and low temperature conditions.

Distractor 2: Combination of distractor 1 and 3.

Distractor 3: Plausible because exceeding the RCS pressure safety limit requires restoration within 2 hours. The pressure given in the question stem does exceed LCO 3.4.11 but does not exceed the safety limit.

Reference: LCO 3.4.12 Amendment No. 200/187, TS B 3.4.12 Rev 0
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): SRO
Tier: 1 **Group:** 2

Question Source: New
Question History: None

10 CFR Part 55 Content: 41.10 / 43.5 / 45.13
SRO Justification: 10 CFR 55.43(b)(2) Facility operating limitations in the TS and their bases. The second portion of the question requires knowledge of the Technical Specification bases for the required action.

Comments:

Associated objective(s):

9 **ID: 1454979** **Points: 1.00**

Unit 1 is operating at rated Power.

- The CY tank was inadvertently cross-tied to the Suppression Pool via the RCIC suction piping resulting in Suppression Pool level exceeding + 3 inches.
- TWO hours later, the needed repairs have been identified, and they are expected to take an additional TWELVE hours.

(1) What action is required?
(2) What is the MAXIMUM allowable time to notify the NRC per LS-AA-1110, Reportability Reference Manual?

A. (1) Initiate a plant shutdown.
(2) within a MAXIMUM of FOUR HOURS.

B. (1) Initiate a plant shutdown.
(2) IMMEDIATELY.

C. (1) Declare an Unusual Event.
(1) within a MAXIMUM of ONE HOUR.

D. (1) Declare an Unusual Event.
(2) within a MAXIMUM of FIFTEEN MINUTES.

Answer: A

Note
Applicant Question: SP level has exceeded +3". From the question, I gather no attempt to lower pool level was made; however, did pool level continue to go up? Rate? Did it stop at say +10" or somewhere? If I lose my pressure suppression function due to high pool level, I may change my answer.
Response Given: level is currently +4".

Answer Explanation
Explanation: A plant shutdown is initiated per Tech Spec 3.6.2.2, Condition B, to be in Mode 3 within 12 hours and Mode 4 within 36 hours, and the event is reportable within FOUR hours per LS-AA-1110, SAF 1.2 due to the initiation of a plant shutdown as required by Tech. Specs. **Distractor 1** is plausible because a plant shutdown must be initiated and because the NRC is notified immediately during the course of an event of the effectiveness of response or protective measures taken (though in this case an event has not been declared). The distractor is incorrect because the only NRC notification required is a four hour notification due to the Tech Spec directed Shutdown. **Distractor 2** is plausible because if an Unusual Event had been declared, a one hour NRC notification would normally be required. Distractor is incorrect because the only NRC notification required is a four hour notification due to the Tech Spec directed Shutdown. **Distractor 3** is plausible because if an Unusual Event had been declared, state and local agencies would be notified within 15 minutes. Distractor is incorrect because an Unusual Event declaration is not required and because the only notification required is a four hour NRC notification due to the Tech Spec directed Shutdown. **Reference:** Hot Matrix - EP-AA-1005 Emergency Action Levels For LaSalle Station, Addendum 3; Tech Spec 3.6.2.2, and LS-AA-1110, REV 23, Reportability Reference Manual **Reference provided during exam:** Hot Matrix, Tech Spec 3.6.2.2, LS-AA-1110, SAFETY **Cognitive level:** High **Level (RO/SRO): SRO** **Tier:** 1 **Group:** 2 **K/A: 295029 High Suppression Pool Water Level** G. 2.4.30 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. **10 CFR Part 55 Content:41.5** **SRO Justification:** 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

10	ID: 1454549	Points: 1.00

Unit 2 is at 100% Power.

Annunciator 2PM13J-B304,REACTOR BUILDING SOUTHEAST OR SOUTHWEST EQUIPMENT DRAIN SUMPS EXCESSIVE PUMP OUT TIME, PUMP EXCESSIVE START FREQ. OR HI-HI LEVEL, is in alarm.

The Process Computer shows R1438 RB SE Equip Drn Sump Trouble.

The SE corner room level is 22 inches.
The SW corner room has normal level.

What is the source of the leak and what is the NEXT action required?

The leak is from ___________(1)________ and the NEXT required action is to ___________ (2)___________.

A. (1) CRD
(2) Isolate the leak

B. (1) CRD
(2) Insert a reactor scram

C. (1) “B” RHRSW
(2) Isolate the leak

D. (1) “B” RHRSW
(2) Insert a reactor scram

Answer: C

Note
Applicant Question: does 'B' refer to 'B' RHR SW, meaning Div 2 RHR SW and not 'B' RHR SW pump (Div 1)?
Response Given: 'B' RHR Heat Exchanger.

Answer Explanation

Explanation: B and C RHR are in the SE corner room with "B" RHRSW. "B" RHRSW is not a primary system, and LGA-002 does not direct a scram based on non-primary system leakage causing areas to exceed max safe values.. Additionally, action to isolate the leak is correct per LGA-002.
Distractor 1 is plausible because the CRD room is located in the south side of the Reactor Building in a location that could cause alarm 2PM13J-B304 if it were leaking. Additionally, action to isolate the leak is correct per LGA-002. The distractor is incorrect because the CRD system is located in the southwest area of the reactor building.
Distractor 2 is plausible because the CRD room is located in the south side of the Reactor Building in a location that could cause alarm 2PM13J-B304 if it were leaking. Additionally, action to isolate the leak per LGA-002 would cause a loss of CRD HCU accumulator charging line which may lead to multiple HCU accumulators becoming inoperable, which would, per Tech Spec 3.1.5 require a reactor scram in 20 minutes if charging water header pressure cannot be restored. The distractor is incorrect because the CRD system is located in the southwest area of the reactor building.
Distractor 3 is plausible because B RHR is located in the south side of the Reactor Building in a location that could cause alarm 2PM13J-B304 if it were leaking. The distractor is incorrect because B RHR is not primary system leakage and LGA-002 does not direct a scram based on non-primary system leakage causing areas to exceed max safe values.
References: LGA-002, 1PM13J-B304 REACTOR BUILDING SOUTHEAST OR SOUTHWEST EQUIPMENT DRAIN SUMPS HI-HI LEVEL and 1PM13J-A304 REACTOR BUILDING NORTHEAST OR NORTHWEST EQUIPMENT DRAIN SUMPS HI-HI LEVEL rev 2
Reference provided during examination: LGA-002, Table W
Cognitive level: High
Level (RO/SRO): SRO
Tier: 1 **Group:** 2
K/A: 295036 Secondary Containment High Sump/Area Water Level
A2.03 Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Cause of the high water level
10 CFR Part 55 Content: 43.5
SRO Justification: *10 CFR 55.43(b)(5)*
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
OP-AA-101-111 Conduct of operations procedure describes the role of directing EOP activities as an SRO only task.
Question Source: New
Question History: N/A
Comments:
Objectives:

11	ID: 1454628	Points: 1.00

Unit 1 is operating at 100% power.
Unit 2 is in a refueling outage, and fuel shuffles are in progress.

There is an inadvertent Unit 1 Reactor Building Ventilation isolation and SBGT initiation.

The assist NSO reports that 1VG001, Inlet Isolation Damper, opened, reclosed and CANNOT be manually reopened.

The Unit 2 Supervisor will direct _______(1)_______, and Tech Spec 3.6.4.3 requires entry into an LCO to __________(2)_________.

A. (1) Unit 1 SBGT train to be secured
(2) restore SBGT to operable status, ONLY

B. (1) Unit 1 SBGT train to be secured
(2) restore SBGT to operable status, AND immediately suspend core alterations and movement of irradiated fuel

C. (1) Unit 2 SBGT train to be secured
(2) restore SBGT to operable status, ONLY

D. (1) Unit 2 SBGT train to be secured
(2) restore SBGT to operable status, AND immediately suspend core alterations and movement of irradiated fuel

Answer: A

Answer Explanation
Explanation: The inlet damper has failed closed, so must secure even though the initiation signal came from Unit 1. 7 day LCO due to Unit 1 at full power. Fuel movements can continue with one SBGT train available. It would have to be started after 7 days. **Distractor 1:** Not required to suspend fuel movements. The second train must be manually started after 7 days. Plausible because a 7 day LCO is entered. **Distractor 2:** Required to secure the Unit 1 SBGT train. Plausible because a SBGT train must be secured and the initiation signal came from Unit 2. **Distractor 3:** Required to secure the Unit 1 SBGT train. Plausible because a SBGT train must be secured and the initiation signal came from Unit 2 and a 7 day LCO is entered. **References:** TS 3.6.4.3, LOP-VG-02 Shutdown Of The Standby Gas Treatment System (SBGT), Rev. 17 **Reference provided to examinee:** TS 3.6.4.3 **KA: 261000 Standby Gas Treatment System** **A2.06 Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** **Cog:** Hi **Level (RO/SRO):** SRO **Tier:** 2 **Group:** 1 **10 CFR Part 55 Content**: 41.5 / 45.6 **SRO Justification:** 10CFR55.43(b)(2) Facility operating limitations in the TS and their bases. **Question Source:** New **Question History:** N/A **Comments:** **Objectives:**

12	ID: 1454995	Points: 1.00

Unit 2 has experienced a large break LOCA coincident with a Loss of Offsite Power and a hydraulic ATWS:

- SBLC was initiated.
- RPV injection was maximized with RCIC.
- All other ECCS pumps are unavailable for injection.
- Due to lowering RPV level, LGA-006, ATWS Blowdown, has been executed.

RPV level is determined to be UNKNOWN due to reference leg boiling:

- ONE (1) control rod remains partially withdrawn at notch 40.
- The LPCS pump has been returned to service.

(1) What action will the Unit Supervisor direct regarding RPV injection systems?
(2) Injection will be TERMINATED when the .....

A. (1) Inject with CRD and SBLC ONLY per LGA-010.
(2) RPV level is above – 150 on Wide Range

B. (1) Inject with CRD, SBLC, and LPCS per LGA-005.
(2) RPV level is above – 150 on Wide Range

C. (1) Inject with CRD and SBLC ONLY per LGA-010.
(2) RPV is flooded to the Main Steam lines

D. (1) Inject with CRD, SBLC, and LPCS per LGA-005.
(2) RPV is flooded to the Main Steam lines

Answer: D

Answer Explanation
Explanation: RPV level is unknown; therefore, the crew must enter LGA-005, RPV Flooding. With only one control rod withdrawn, the unit is not in an ATWS condition, so Injection with CRD, SBLC, and LPCS is appropriate until the RPV is flooded to the Main Steam lines. **Distractor 1** is plausible because injecting with CRD and SBLC until level is above – 150 inches would be the correct action in an ATWS condition per LGA-10. **Distractor 2** is plausible because injecting with CRD, SBLC and LPCS until level is above – 150 inches would be the correct action in a non-ATWS condition if RPV level were known. The distractor is incorrect because level is unknown and because direction to inject with CRD, SBLC and LPCS until level is above – 150 is contained in LGA-001, not LGA-005. **Distractor 3** is plausible because injecting with CRD and SBLC until the Main Steam lines are flooded would be the correct action in an ATWS condition with RPV water level unknown. **Reference:** LGA-001, Rev 16, RPV Control, LGA-005, Rev 14, RPV Flooding, LGA-010, Rev 15, Failure to Scram **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** SRO **Tier:** 2 **Group:** 1 **K/A: 209001 Low Pressure Core Spray System** **2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.** ***SRO Justification:*** *10 CFR 55.43(b)(5)* Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

13	ID: 1454933	Points: 1.00

A reactor startup is in progress on Unit 1:

- Reactor Mode Switch is in RUN.
- Annunciator 1H13-P603-A107, IRM HI, is in alarm
- Annunciator 1H13-P603-B304, Channel ‘A’ IRM HI-HI/INOP, is in alarm

It is determined that the ‘A' IRM is NOT fully withdrawn, and the detector is STUCK.

(1) What is the expected IRM system response, if any?
(2) What action(s) is/are required, if any?

A. (1) There are NO IRM generated trips.
(2) Continue with the startup. No additional action is required.

B. (1) An IRM UPSCALE rod block and a ½ scram.
(2) Bypass IRM ‘A’ per 1H13-P603-B304, Channel ‘A’ IRM HI-HI/INOP, and continue with the startup. No additional action is required.

C. (1) There are NO IRM generated trips.
(2) Momentarily STOP the startup, bypass IRM ‘A’ and troubleshoot the stuck IRM detector per LOA-NR-101, Neutron Monitoring Trouble.

D. (1) An IRM UPSCALE rod block and a ½ scram.
(2) Momentarily STOP the startup; bypass IRM ‘A’ per 1H13-P603-B304, Channel ‘A’ IRM HI-HI/INOP, and troubleshoot the stuck IRM detector per LOA-NR-101, Neutron Monitoring Trouble.

Answer: C

Answer Explanation
Explanation: All IRM system trips are bypassed with the Reactor Mode Switch in RUN; therefore, there are NO IRM generated trips. Per LOA-NR-101, Neutron Monitoring Trouble, STOP the startup; bypass IRM 'A'; check/replace IRM 'A' Detector Drive fuses and attempt to move the detector. **Distractor 1** is plausible because all IRM system trips are bypassed with the Reactor Mode Switch in RUN, so there are NO IRM generated trips. **Distractor 2** is plausible because the IRMs generate UPSCALE rod blocks and scram signals with the mode switch in Startup. **Distractor 3** is plausible because the IRMs generate UPSCALE rod blocks and scram signals with the mode switch in Startup. **Reference:** Per LOA-NR-101, Rev 19, Neutron Monitoring Trouble **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** SRO **Tier:** 2 **Group:** 1 **K/A: 215003 - Intermediate Range Monitor System** **A2.04 - Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Upscale or down scale trips.** **10 CFR Part 55 Content:** 10CFR55.43(b)(5) ***SRO Justification:*** Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

14	ID: 1454688	Points: 1.00

Unit 2 has scrammed from 100% reactor power:

- Unit 2 RPV water level is STABLE at -80 inches with RCIC and HPCS injection maximized.
- Annunciator 2H13-P601-D102, RCIC VACUUM TANK PRESSURE HIGH, is in alarm.
- The Barometric Condenser Vacuum Pump is not running.

(1) What would be the effect of continued RCIC operation?
(2) What action will the Unit Supervisor direct?

A. (1) RCIC turbine shaft seal damage.
(2) Secure RCIC.

B. (1) RCIC turbine shaft seal damage.
(2) Continue to operate RCIC.

C. (1) Airborne contamination in the RCIC room.
(2) Secure RCIC.

D. (1) Airborne contamination in the RCIC room.
(2) Continue to operate RCIC.

Answer: D

<u>Note</u>
Question: is FW available?
Response: feedwater is not available.

Answer Explanation

Explanation: With a loss of the RCIC barometric condenser vacuum pump, vacuum will be lost in the barometric condenser, leading to steam in the RCIC room and airborne contamination. Per LOR-2H13-P601-D102, if an emergency condition requires continued operation of RCIC with a high vacuum tank pressure, RCIC can operate without the barometric condenser, and per the conditions in the stem, RCIC is required for adequate core cooling; therefore, the Unit Supervisor will direct that RCIC operation continue.

Distractor 1 is plausible because there is shaft seal leakage as a result of losing vacuum in the barometric condenser.

Distractor 2 is plausible because there is shaft seal leakage as a result of losing vacuum in the barometric condenser and because, per LOR-2H13-P601-D102, if an emergency condition did not require continued operation of RCIC without the barometric condenser, RCIC would be shutdown.

Distractor 3 is plausible because there is airborne contamination in the RCIC room and because, per LOR-2H13-P601-D102, if an emergency condition did not require continued operation of RCIC without the barometric condenser, RCIC would be shutdown.

Reference: 2H13-P601-D202, Rev 5, RCIC Barometric Condenser Vacuum Tank Level Low- High

Reference provided during examination: N/A

Cognitive level: High

Level (RO/SRO): SRO

Tier: 2 **Group:** 1

KA: 217000 Reactor Core Isolation Cooling System (RCIC)

A2.09 Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of vacuum pump

10 CFR Part 55 Content: 43.5

SRO Justification: *10 CFR 55.43(b)(5)*

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Source: New

Question Source: New

Question History: N/A

Comments:

Associated objective(s):

15	ID: 1454691	Points: 1.00

Refer to the provided RWLC_01 process overview.

Based on the indications, the Unit Supervisor should direct?

A. LGP-3-2, Reactor Scram and direct a manual reactor scram

B. LOP-FW-16, 1(2)DS001 Operator Station Alarm Message Interpretation

C. LOA-FW-101, Reactor Level/Feedwater Pump Control Trouble - B.1 Failure of Automatic RWLCS

D. LOA-FW-101, Reactor Level/Feedwater Pump Control Trouble - B.5 Emergency Recovery of Turbine Driven Reactor Feed Pump

Answer: B

Answer Explanation
Explanation: LOP-FW-16, 1(2)DS001 Operator Station Alarm Message Interpretation. The printout shows a failure of the "B" TDRFP flow indication, which will allow RFWLC to remain in auto. Enter LOP-FW-16, 1(2)DS001 Operator Station Alarm Message Interpretation to address the failed instrument. **Distractor 1:** No current scram signals and no trends in progress that would scram the reactor. Plausible because the indicated "B" TDRFP flow is 0. This would lead to a scram due to low level if this were actual flow. **Distractor 2:** RWLC is still in auto. Plausible because steam flow inputs failing will take RWLC out of Auto. **Distractor 3:** Both TDRFPs are running. Plausible because the "B" TDRP flow indicates 0, the flowrate for a tripped TDRFP. **Reference:** LOR-1H13-P603-A511, Rev. 2, RWLCS Trouble LOP-FW-16, 1(2)DS001 Operator Station Alarm Message Interpretation, Rev. 35 Printout of RWLC_01 Process Overview with "B" TDRFP flow at 0 **Reference provided to examinees:** Printout of RWLC_01 Process Overview with "B" TDRFP flow at 0 **KA: 259002 Reactor Water Level Control System** **2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.** **Cog**: High **Level (RO/SRO):** SRO **Tier:** 2 **Group:** 1 **10 CFR Part 55 Content: 41.10 / 43.5 / 45.12** ***SRO Justification:*** *10 CFR 55.43(b)(5)* **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

16	ID: 1454613	Points: 1.00

Unit 1 is at 70% power performing a Control Rod Sequence Exchange

- Control rod 18-19 is being continually withdrawn from position 00 to position 12.
- The operator does not release either the withdraw or continuous withdraw pushbuttons.

(1) The RWM:

(2) The Unit supervisor will direct:

A. (1) Stops control rod withdrawal at position 12
(2) The RWM is to be declared inoperable ONLY.

B. (1) Stops control rod withdrawal at position 14
(2) The control rod is to be inserted to position 00.

C. (1) Stops control rod withdrawal at position 14
(2) The control rod is to be inserted to position 12.

D. (1) Stops control rod withdrawal at position 12
(2) The RWM is to be declared INOP and the control rod is to be inserted to position 00.

Answer: C

Answer Explanation
Explanation: (1) Stop control rod withdrawal at position 14 (2) Direct the control rod inserted to position 12. The RWM should stop the control rod at position 14, which is one notch beyond the target position. Control rod moved to a position inconsistent with the Reactivity Plan or associated predictions requires entry into LOA-RD-101. This will direct the rod inserted one notch to its required position. **Distractor 1 is incorrect:** The RWM should not stop the control rod at position 12, and the RWM should not be declared inoperable. The distractor is plausible because notch 12 is the target position; therefore, if the RWM were to stop the control rod at position 12, it would not be functioning properly and should be declared inoperable. **Distractor 2 is incorrect:** The RWM should stop the control rod at position 14, which is one notch beyond the target position: however, the control rod is not required to be inserted to position 00. The distractor is plausible because the RWM should stop the control rod at position 14 and because inserting the control rod to position 00 would be the correct action if the control rod had been withdrawn one notch beyond its target position. **Distractor 3 is incorrect:** The RWM should not stop the control rod at position 12, and the RWM should not be declared inoperable. Also incorrect because the control rod is not required to be inserted to position 00. The distractor is plausible because notch 12 is the target position; therefore, if the RWM were to stop the control rod at position 12, it would not be functioning properly and should be declared inoperable. **References:** LOA-RD-101, Rev. 20 CONTROL ROD DRIVE ABNORMAL, page 10 **References given during the exam:** None **Cognitive level:** High **Level (RO/SRO):** SRO **Tier:** 2 **Group:** 2 **KA: 201006 Rod Worth Minimizer System.** **A2.05 Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWH) (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Out of sequence rod movement; P-Spec(Not-BWR6)** **10 CFR Part 55 Content:** 41.5 / 45.6 **SRO Justification:** 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

17	ID: 1454976	Points: 1.00

Unit 1 is at 100% power:

- The assist NSO is completing a TIP set per LOP-NR-06, Traversing Incore Probe Operation.
- TIP Drive Control Unit 1D is taken to Manual Reverse.
- The assist NSO reports the Unit 1D TIP indicates stuck at the indexer.

Two minutes later, Unit 1 scrams.

What action will the Unit Supervisor direct and why?

A. De-energize the Ball Valve in NOT more than 72 hours to close the ball valve.

B. De-energize the Ball Valve in NOT more than 4 hours to close the ball valve.

C. Fire the Shear Valve in NOT more than 72 hours to isolate the penetration.

D. Fire the Shear Valve in NOT more than 4 hours to isolate the penetration.

Answer: D

Answer Explanation
Explanation: With the TIP stuck in the core, you have 4 hours to isolate the penetration per TS 3.6.1.3 for an open system. Additionally, LOP-NR-06, directs that the Shear Valve be fired in the event of Group 7 isolation with the TIP beyond the Ball Valve and not retracting. **Distractor 1** is plausible because there is a 72 hour completion time for isolation valves in 3.6.1.3 for penetrations with a closed system. **Distractor 2** is plausible because the Ball Valve would close if the TIP were not inside the mechanism. **Distractor 3** is plausible because there is a 72 hour completion time for isolation valves in 3.6.1.3 for penetrations with a closed system and because the Ball Valve would close if the TIP were not inside the mechanism. . **References:** LOP-NR-06, Rev 31, Traversing Incore Probe (TIP) Operation, TS 3.6.1.3, TRM Appendix A **References provided during the exam:** TS 3.6.1.3, TRM Appendix A **Cognitive level:** High **Level (RO/SRO):** SRO **Tier:** 2 **Group:** 2 **KA: 215001 Traversing In-Core Probe** **2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.** **SRO Justification:** 10CFR55.43(b)(2) Facility operating limitations in Technical Specifications and their bases **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

18	ID: 1454972	Points: 1.00

Unit 2 is operating at 100% power on THROTTLE pressure control:

- One of three (3) throttle pressure transmitters has drifted low with a pressure output 20 psig lower than the others.
- A second pressure transmitter fails, and the drifted low throttle pressure transmitter is auto selected as the controlling pressure transmitter.

For the given conditions, Turbine Control Valves will ___(1)___, and the Unit Supervisor will direct action in accordance with ___(2)___ .

A. (1) Close
(2) LOA-EH-201, UNIT 2 EHC Abnormal, B.3 Turbine Control Valve/Turbine Stop Valve Failed/Failing Full Closed

B. (1) Close
(2) LOA-EH-201, UNIT 2 EHC Abnormal, B.1 Loss of a Pressure Regulator

C. (1) Open
(2) LOA-EH-201, UNIT 2 EHC Abnormal, B.2 Turbine Control Valve Failed/Failing Full Open

D. (1) Open
(2) LOA-EH-201, UNIT 2 EHC Abnormal, B.1 Loss of Pressure Regulator

Answer: B

Answer Explanation

Explanation:

(3) Close

(4) LOA-EH-201, UNIT 2 EHC Abnormal, B.1 Loss of a Pressure Regulator

When the second pressure transmitter fails, the drifted low throttle pressure transmitter is auto selected as the controlling pressure transmitter; therefore, sensed throttle pressure is 20 psig lower than before the failure and control valves will close to return pressure to the normal setpoint. LOA-EH-201, UNIT 2 EHC Abnormal, B.1 Loss of a Pressure Regulator is entered as a result of this failure.

Distractor 1 is plausible because the control valves will close and LOA-EH-201, UNIT 2 EHC Abnormal, B.3 Turbine Control Valve/Turbine Stop Valve Failed/Failing Full Closed would be entered if the valves closed due to Control Valve failure.

Distractor 2 is plausible because if the setpoint were reduced by 20 psig the valves would open; also, if control valves themselves had failed so as to open, UNIT 2 EHC Abnormal, B.2 Turbine Control Valve Failed/Failing Full Open would be the appropriate procedure section to execute.

Distractor 3 is because if the setpoint were reduced by 20 psig the valves would open; also, LOA-EH-201, UNIT 2 EHC Abnormal, B.1 Loss of a Pressure Regulator is the appropriate section to execute as a result of this failure.

Reference: LOA-EH-201, Rev 033, UNIT 2 EHC ABNORMAL

Reference provided during the exam: N/A

Cognitive level: High

Level (RO/SRO): SRO

Tier: 2 **Group:** 2

KA: 241000 Reactor/Turbine Pressure Regulating System

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

A2.16 Low turbine inlet pressure (loss of pressure signal)

10 CFR Part 55 Content: 41.5 / 41.6

SRO Justification: *10 CFR 55.43(b)(5)*

Question Source: New

Question History: N/A

Comments:

Associated objective(s):

19	ID: 1263886	Points: 1.00

Core Alterations have been stopped by the Refuel SRO due to a Refuel Bridge equipment failure.

Who has the authority to grant permission to resume fuel movement?

A. Dedicated Refueling NSO

B. Operations Shift Manager

C. Qualified Nuclear Engineer

D. Outage Services Director

Answer: B

Note: both 'A' or 'B' were accepted as correct answers for this question based upon post-exam comment resolution. See exam report for further details.

Note

Applicant Question: is this referring to permission in regards to the NCTL or permission regarding failed refueling equipment to be used after it has been repaired?
Response Given: repairs have been completed.

Answer Explanation
Explanation: Per LFP-100-1 Master Refuel Procedure Section D.3.1, page 19, the Operations Shift Manager's permission is required to resume fuel movement. **Distractor 1** is incorrect: The Refueling Dedicated NSO has responsibility during core alterations but does not possess the authority to grant permission to resume fuel movement. Distractor is plausible because the Refueling Dedicated NSO has responsibility to verify requirements necessary for core alterations are satisfied prior to core alterations. **Distractor 2** is incorrect: The Qualified Nuclear Engineer does not have the authority to grant permission to move fuel. Plausible because the QNE must grant permission to bypass refueling interlocks which affect refuel bridge operability. **Distractor 3** is incorrect: The Outage Services Director does not have the authority to grant permission to move fuel. Plausible because the OSD must approve the Abnormal Obstacle Avoidance plan. . **Reference:** LFP-100-1, Rev 61, 1 Master Refuel Procedure. **Reference provided during examination:** N/A **Cognitive level:** Memory **Level (RO/SRO):** SRO **Tier:** 3 **K/A: 2.1.35 Knowledge of the fuel-handling responsibilities of SROs.** **10 CFR Part 55 Content: 43.7** ***SRO Justification:*** *10 CFR 55.43(b)(7)* Fuel handling facilities and procedures **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

20	ID: 1454948	Points: 1.00

Unit 1 is at 100% power:

- The 1A RHR Pump was taken out of service two hours ago.

Identify the protected paths that MUST be in place per OP-AA-108-117, Protected Equipment Program, and the provided Paragon Remain in Service List.

1. B RHR
2. HPCS
3. RCIC
4. MDRFP
5. B RHR SW
6. 1A DG CWP
7. B/C RHR Room Fan
8. Div 2 CSCS Room Fan

A. 2,3,5,6,7 and 8

B. 1,2,5,6,7 and 8

C. 1,2,4,5,6 and 8

D. 2,3,4,5,6 and 8

Answer: B

Answer Explanation

Explanation: Per the Paragon Remain in Service List, items 1, 5, 6, 7 and 8 are identified as protected equipment due to a their loss changing the online risk assessment red, and per OP-AA-108-117, Protected Equipment Program, HPCS must be protected due to causing entry into a 12 hour or less shutdown time clock.

Distractor 1 is incorrect because B RHR is not included and RCIC is included. The distractor is plausible because all of the other specified items are protected equipment.

Distractor 2 is incorrect because B/C RHR Room Fan is not included and the MDRFP is included. The distractor is plausible because all of the other specified items are protected equipment.

Distractor 3 is incorrect because B/C RHR Room Fan and B RHR are not included and the MDRFP is included. The distractor is plausible because all of the other specified items are protected equipment.

Reference: TS 3.5.1 and OP-AA-108-117, Rev 4, Protected Equipment Program

Reference provided during examination: Paragon Remain in Service List

Cognitive level: High

Level (RO/SRO): SRO

Tier: 3 **Group:**

K/A: 2.2.15 **Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.**

10 CFR Part 55 Content: 41.10 / 43.3 / 45.13

SRO Justification: *10 CFR 55.43(b)(5)*

Question Source: New

Question History: N/A

Comments:

OP-AA-108-117, Protected Equipment Program: When to Protect Equipment

4.2.1. **When** SSCs are planned to or become unavailable **then PROTECT** redundant equipment if plant configuration is such that redundant equipment unavailability or manipulation would cause:

1. An overall online or outage risk assessment change to red risk **(CM-2, CM-3, CM-4),**
2. A loss of generation capability of > 20 MWe, or
3. An entry into Tech Spec 3.0.3 (3.0.1 for TMI) or a shutdown Tech Spec LCO of 12 hrs or less (i.e. be in hot shutdown in 12 hrs or less).

Associated objective(s):

21	ID: 1433248	Points: 1.00

Unit 2 is operating at 100% power:

Several I&C jobs are currently active.
One crew of I&C personnel is available to perform work.

Given the status of I&C resources, utilize the Work Screening and Processing checklist provided to determine which of the following activities must receive the highest work priority?

A. Drywell high pressure input to RPS is faulted and requires repair.

B. Calibration of Unit 2 Waste Sample Tank level transmitter is due today.

C. Surveillance test on low level ECCS initiation that will go past its critical date in three days.

D. Unit 2 Station Air Compressor failure has occurred due to a failed control board, and the trailer mounted air compressor is in standby.

Answer: A

Answer Explanation

Explanation: Drywell high pressure input to RPS requires repair. Per Attachment 1, the drywell pressure input to RPS being failed requires that question 5, 14 day or less S/D LCO be answered yes, pushing this activity into a B1 priority.

Distractor 1 is incorrect: Not a Tech Spec required calibration, so would not drive the unit into a required S/D. The distractor is plausible because this is a required calibration.

Distractor 2 is incorrect: The low level ECCS would also answer yes to this question, but not until the ST expires in 3 days, so a lower priority. Plausible because the surveillance test on low level ECCS initiation is required by tech specs.

Distractor 3 is plausible because the Station Air system is required to keep the unit on line.

Reference: WC-AA-106, Rev 16, Work Screening and Processing, Attachment 1, Priority Screening Decision Flow

Reference provided during examination: WC-AA-106, Attachment 1,

Cognitive level (Memory/High): High

Level (RO/SRO): SRO

Tier: 3

K/A: 2.2.17 Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

10 CFR Part 55 Content: 43.5

SRO Justification: *10 CFR 55.43(b)(5)*

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

SRO Justification: 10CFR55.43(b)(5)

"Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

Question Source: New

Question History: N/A

Comments:

Associated objective(s):

22 **ID: 1264292** **Points: 1.00**

The Unit 1 is in Mode 1 and the Drywell must be purged as part of a shutdown for a maintenance outage.

Which of the following actions MUST be verified complete prior to authorizing the NSO to commence purging the drywell?

1.) Verify Noble Gas Channels 1PL15J and 1PL75J are clear
2.) Verify one SBGT Train has been placed in service
3.) Obtain sample results for Principal Gamma Emitters and Tritium from within 30 hours
4.) Verify Containment is aligned for purge through the Containment vent and Purge System

A. 1 and 4 ONLY

B. 1 and 2 ONLY

C. 2 and 3 ONLY

D. 3 and 4 ONLY

Answer: D

Answer Explanation
Explanation: Obtain sample results for Principal Gamma Emitters and Tritium from within 30 hours, and verify Containment is aligned for purge through the Containment vent and Purge System. **Distractor 1** is incorrect: LOP-VQ-04 Noble Gas Channels 1PL15J and 1PL75J to be clear prior to Primary Containment venting, but they are not required for Primary Containment purge operations. The distractor is plausible because there is a requirement to verify Containment is aligned for purge through the Containment vent and Purge System and because Noble Gas Channels 1PL15J and 1PL75J to be clear prior to Primary Containment venting, and it is a good practice to monitor the Noble Gas Channels prior to purging even though it is not required. **Distractor 2** is incorrect: The Primary Containment is not purged through SBGT. The distractor is plausible because there is a requirement to verify Containment is aligned for purge through the Containment vent and Purge System and because Noble Gas Channels 1PL15J and 1PL75J to be clear prior to Primary Containment venting, and it is a good practice to monitor the Noble Gas Channels prior to purging even though it is not required. **Distractor 3:** is plausible because there is a requirement to obtain sample results for Principal Gamma Emitters and Tritium from within 30 hours. **Reference:** LOP-VQ-04, Rev 038, Startup, Shutdown, and Operations of the Primary Containment Vent and Purge System **Reference provided during examination:** N/A **Cognitive level:** High **Level (RO/SRO):** SRO **Tier: Group:** **K/A: 2.3.6** Ability to Approve Release permits **10 CFR Part 55 Content: 55.41** **SRO Justification:** 10CFR55.41(b)(13) Procedures and equipment available for handling and disposal of radioactive materials and effluents. **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

23	ID: 1264260	Points: 1.00

A Unit startup is in progress with the following conditions:

- The reactor is critical, below the point of adding heat.
- The reactor is on an infinite period.
- Reactor power is on the Intermediate Range Monitors (IRMs) Range 2.

The unit NSO withdraws several control rod notches per the sequence package and takes the following readings to determine reactor period:

- Initial power 20 on IRM Range 2.
- Final power 40 on IRM Range 2.
- Time to change from initial to final power was 30 seconds.
- Annunciator 1H13-P603-A306, SRM Short Period, alarms.

Based on the above information and calculated reactor period, the SRM SHORT PERIOD annunciator ____________(1)__________ be alarming, and the Unit Supervisor shall direct the NSO to ____________(2)__________.

A. (1) should NOT
(2) insert control rods using RCMS

B. (1) should NOT
(2) continue control rod withdrawal

C. (1) should
(2) insert control rods using RCMS

D. (1) should
(2) manually scram the reactor

Answer: C

Answer Explanation
Explanation: (1) should (2) insert control rods using RCMS. Reactor period = (1.443) x (doubling time) = (1.443) x (30) = 43.3 seconds. The target period should be 150 to 200 seconds and if <50 seconds the RO is required to insert control rods. **Distractor 1:** The SRM period alarm comes in at ≤50 seconds. Plausible if the applicant improperly calculates the period. The applicant would fail to understand the Alarm should be alarming. **Distractor 2:** The SRM period alarm comes in at ≤50 seconds. Plausible if the applicant improperly calculates the period. The applicant would fail to understand the Alarm should be alarming. Also plausible because control rod withdrawal is appropriate with no period alarm. **Distractor 3:** Control rods are required to be inserted to turn the short reactor period. The applicant properly calculated the period. Plausible because a manual scram is required if a valid scram signal exists or did exist per LOA-RD-101. **Reference:** LOR-1H13-P603-A306 Rev. 003 and LOA-RD-101, Rev 20, CONTROL ROD DRIVE ABNORMAL **Reference provided during examination:** None **Cognitive level:** High **Level (RO/SRO): SRO** **Tier:** 3 **Group:** **K/A: 2.1.7** **Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.** **10CFR Part 55:** CFR: 41.5 / 43.5 / 45.12 / 45.13 **SRO Justification**: 10CFR55.43(b)(1) Knowledge of "Conditions and limitations in the facility license **Question Source**: Bank **Question History:** N/A **Comments:** **Associated objective(s):**

24	ID: 1433249	Points: 1.00

10CFR50.54(X) Conditions of licenses and HU-AA-104-101, Procedure Use and Adherence, state, in part:

"....... reasonable action that departs from a license condition or a Technical Specification in an emergency when this action is immediately needed to protect the public health and safety is permitted.

These actions shall be:

A. Approved by a licensed SRO prior to taking the action.

B. Reported to the NRC within 15 minutes of the action being taken.

C. Approved by the Operations Director prior to the action being taken.

D. Performed immediately and approved by a licensed SRO when possible.

Answer: A

Answer Explanation
Explanation: Per HU-AA-104-101, Procedure Use and Adherence, step 4.8.4.3 and 10CFR50.54(Y) Conditions of licenses. – Approved by a licensed SRO prior to taking the action. **Distractor 1** is incorrect: The 1 hour reporting requirement is to the NRC. Distractor is plausible because there is a 15 minutes reporting requirement to the state for EAL declaration. **Distractor 2** is plausible if the candidate thinks the senior person in Ops should approve this action. **Distractor 3** is incorrect: The procedure and 10CFR50.54 both say to get approval first. Plausible it is an emergency action to protect the health and safety of the public. **Reference:** HU-AA-104-101, Rev 5, Procedure Use and Adherence **Reference provided during examination:** N/A **Cognitive level:** Memory **Level (RO/SRO):** SRO **Tier:** 3 **KA: 2.4.5 – Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.** **10 CFR Part 55 Content: 43.5** ***SRO Justification:*** *10 CFR 55.43(b)(5)* Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations **Question Source:** New **Question History:** N/A **Comments:** **Associated objective(s):**

25	ID: 1455171	Points: 1.00

Unit-1 is operating at 75% power:

- Annunciator LOR 1H13-P603-B106, CHAN A1/B1 TCV FAST CLOSURE is in alarm.
- All Scram Group Solenoid Lights are illuminated
- The #1 Turbine Control Valve indicates closed
- Reactor power, pressure, and level are steady

What actions, if any, should be directed?

A. No actions are required.

B. Manually insert a full scram IMMEDIATLY.

C. Manually insert a 1/2 scram on RPS A within 12 hours.

D. Manually insert a 1/2 scram on RPS B within 12 hours.

Answer: C

Answer Explanation

Explanation: Receipt of LOR 1H13-B106, CHAN A1/B1 TCV FAST CLOSURE, indicates that at least one TCV has closed. Additional information in the question stem indicates that the number 1 TCV is closed informing the candidate that an RPS ‘A’ half-scram should have occurred and that a full reactor scram should not have occurred. All scram group solenoid lights being lit informs the candidate that the expected half-scram did not occur. Additionally, indications that power, pressure and level are steady inform the candidate that a full reactor scram has not occurred. Tech Spec 3.3.1.1, Table 3.3.1.1-1, Function 9, TCV FAST CLOSURE trip applies when ≥ 25% RTP, and with one or more required channels inoperable, as indicated by the absence of an A RPS half-scram, the Tech Spec directs that the channel or its associated trip system be placed in trip within 12 hours; therefore, manually inserting a 1/2 scram on RPS A within 12 hours is the correct action to direct.

Distractor 1: LOA-EH-101 directs a manual scram if reactor pressure is not stable or if more than one TCV has closed, and neither of these conditions are present in the question stem; LOA-EH-101 also directs reactor power to be reduced to less than 85%, but reactor power is already below that threshold; therefore, because none of these conditions exist, the distractor is plausible. The distractor is incorrect because manually inserting a 1/2 scram on RPS A within 12 hours is required per Tech Spec 3.3.1.1.

Distractor 2 is plausible because expected automatic actions did not occur and because LOA-EH-101 requires a manual reactor scram if reactor pressure is not stable or if more than one TCV is closed. Also plausible because LOR 1H13-P603-B106 directs a reactor scram if reactor power is greater than 25% and both RPS channels are tripped. The distractor is incorrect because none of these scram conditions exist in the question stem.

Distractor 3 is plausible because expected automatic actions did not occur, a failed TCV should cause a half-cram, and because Turbine Stop Valve #1 inputs to RPS A. Additionally, providing RPS B as a distractor requires the candidate to diagnose the specific RPS failure by determining which RPS system the affected TCV should have actuated. Furthermore, with reactor power ≥ 25% and one or more required channels inoperable, as indicated by the failure to half-scram, a manual 1/2 scram must be inserted within 12 hours. The distractor is incorrect because the half-scram is required on RPS A.

Reference: Tech Spec 3.3.1.1 and Tech Spec Table 3.3.1.1-1; also, LOR 1H13-P603-B106, Rev. 7, Channel A1/B1 Turbine Control Valve Fast Closure

Reference provided during examination: Tech Spec 3.3.1.1 and Tech Spec Table 3.3.1.1-1

Cognitive level: High

Level (RO/SRO): SRO

Tier: 3 **Group:**

KA: 2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

(CFR: 41.10 / 43.5 / 45.3)

10 CFR Part 55 Content: 41.10 / 43.5 / 45.3

SRO Justification: *10 CFR 55.43(b)(5)*

Question Source: Bank

Question History: 2002-01 LaSalle NRC License Exam

Comments:

Associated objective(s):