EXAMINATION ANSWER KEY

17-1 NRCEXAM

1 **ID: 2058680** **Points: 1.00**

Unit 1 is at 100% RTP, when the following occurs:

- Core Thermal power drops 2%
- MWE has a corresponding drop
- MSL flow drops
- Core Plate Differential Pressure drops
- Total jet pump flow rises

What is causing these indications?

A. Leaking SRV

B. A Broken Jet Pump

C. Bi-stable flow oscillations

D. Reactor Recirculation Flow Control Valve RVDT failure

Answer: B

Answer Explanation

A is incorrect: This is plausible because the stem has elements of indications from a leaking SRV, i.e..MWE decrease, MSL flow decrease but is incorrect because reactor power would rise and coreplate DP would not drop.

B is correct: Answer Explanation: These are all indications of a Jet pump failure, per LOA-RR-101 REACTOR RECIRCULATION SYSTEM ABNORMAL Discussion Section C.10:

Cracking of jet pump hold-down beams has occurred at several operating BWRs. Beam failures due to these cracks have occurred during plant operation causing jet pump mixer displacement. In all cases, significant recirculation system performance degradation occurred prior to beam failure. Plant Responses to jet pump failures can be as follows:

Failure of a hold-down assembly will cause a displaced mixer, so the jet pumps will lose drive flow and the displaced jet pump will start reverse flow. The plant responses are:

- Core Power and Plant Electrical Output will drop due to loss in core flow.
- Core Plate Differential Pressure indication will drop corresponding to the core flow decrease.
- Displaced Jet Pump: The flow direction will change from forward to reverse. Since the diffuser pressure tap is still at a lower pressure than the lower plenum, the flow indication is positive. The flow signal will be less noisy due to reverse flow.
- Other Jet Pump on Same Riser - The flow will drop due to the preferential flow out the displaced jet pump. However, drive flow is still sufficient to maintain forward flow in the jet pump.
- Remaining Jet Pumps - The jet pumps in the sound loop will increase their flow due to the decreased internal vessel jet pump flow resistance offered by the inactive jet pump. The flow of the other jet pumps in the loop with the displaced jet pump will decrease due to preferential drive flow distribution to the riser with the inactive jet pump
- Jet Pump Flow Relationship - The jet pump flow D/P pattern for the sound loop will remain unchanged. The pattern for the other loop changes significantly since the indicated flow of the displaced jet pumps is high and the flow of its partner is very low.
- Jet Pump Loop Flow - The loop flow of the sound loop will increase 5% or more and that of the loop with the displaced jet pump will show an increase or slight decrease since the reverse flow through the displaced jet pump is added rather than subtracted by the core flow measurement system.
- Core Flow indication will increase due to addition of the reverse flow.
- Recirculation Pump Flow - Since the 180 degree bend to nozzle sections comprise 80-85% of the resistance in the external loop, loss of one jet pump removes significant flow resistance. Consequently, the recirculation pump flow should increase.

C is incorrect: This is plausible because bi-stable flow is a common occurrence at LaSalle station and is described in LGP-3-1 POWER CHANGES precaution section, the stem contains elements of bistable flow, i.e. MWE decrease, Core power decrease, and momentary drop in core plate DP but is incorrect because total jet pump flow would also decrease and not rise.

D is incorrect: This is plausible because a failure of a RVDT would cause a change in reactor power but the power change would increase around 2 to 3% reactor power and also would be self correcting when the Reactor Recirculation Flow Control Valve responds with a swap to the LVDT

Reference: LOA-RR-101 UNIT 1, REACTOR RECIRCULATION SYSTEM ABNORMAL rev 38, LGP-3-1 POWER CHANGES rev 66, 1(2) DS001 OPERATOR STATION ALARM MESSAGE INTERPRETATION
LOP-FW-16 rev 36, LOA-SRV-101 UNIT 1 STUCK OPEN SAFETY RELIEF VALVE Rev 8

Reference provided during examination: none

Cognitive level: High
10CFR Part 55: CFR: 41.10

Level (RO/SRO): 3.0/3.1
Tier: 1 Group: 1

PRA: No
K/A: 295001 AA2.04 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Individual jet pump flows

SRO Justification: N/A

Question Source: New

Question History: N/A

Comments:
Objective: 022.00.14

2	ID: 2058712	Points: 1.00

Unit 1 is at 100% RTP.

- 135X-2 AC feed breaker to 1A RPS MG set trips on overcurrent.

How will the plant respond in the first 5 minutes?

A. VR fans trip

B. Reactor scram

C. IN Compressors trip

D. Div 1 backup scram valves ENERGIZE

Answer: C

Answer Explanation
A is incorrect: but plausible because there is a group 4 signal generated to the outboard valves generated but the logic for the VR valves is now DC and a loss of RPS will not affect VR valves. B is incorrect: but plausible because a loss of 135X-2 would be a major electrical transient and have an effect on the RPS system but the design of the plant only a 1/2 scram trip signal would occur. The 1/2 scram would cause the IN compressors to trip and a loss of IN to the inboard MSIVs. When IN bleeds off the inboard MSIVs would close and cause a high pressure scram. However, this would take longer than 5 minutes to occur. C is correct: On a loss of 135X-2, the A RPS bus would de-energize, this loss would cause a PCIS Group 10 isolation and the IN compressors would trip on low suction pressure. D is incorrect: but plausible because other RPS components reposition, i.e. on a loss of Bus 135X-2 there will be a scram signal generated in the 1A RPS (scram discharge level Hi-HI) and the scram pilot (117) valves will reposition but it requires both buses of RPS to de-energize to energize the circuitry for the back-up scram valves, this allows the valves to energize and open. Reference: LOA-RP-101 UNIT 1 LOSS OF REACTOR PROTECTION SYSTEM POWER Rev 16, LOA-AP-101 UNIT 1 AC POWER SYSTEM ABNORMAL Rev 58, LOP-IN-01 Rev 36, DRYWELL PNEUMATIC SYSTEM STARTUP AND OPERATION. Electrical Schematic 1E-1-4215AK Rev AA Reference provided during examination: none Cognitive level: Memory 10CFR Part 55: 41.5 Level (RO/SRO): 4.4/4.7 Tier: 1 Group: 1 PRA: No K/A: 295003 Partial or Complete Loss of A.C. Power G 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. SRO Justification: N/A Question Source: Bank Question History: 2010 River Bend Comments: Objective: 005.00.14

3	ID: 2058721	Points: 1.00

Unit 1 is in Mode 2 with IRMs on ranges 3 and 4.

- A loss of 1A 24/48 VDC battery bus and charger occurs.

What are the effects, if any, on the RCMS (Rod Control Management System) and RPS (Reactor Protection System)?

A. A Rod Block and a full scram

B. A Rod Block and a half scram

C. A Rod Block but NO RPS actuation.

D. NEITHER Rod Block NOR an RPS actuation

Answer: B

Answer Explanation

A is incorrect: plausible because a rod block will be generated on the loss of nuclear instrumentation and failure of nuclear instrumentation on a single channel can cause a scram, if the shorting links are installed.

B is correct: With the loss of the 1A 24/48 volt system the following occurs Per LOA-DC-101 UNIT 1 DC POWER SYSTEM FAILURE
1. The following are NOT operable due to loss of 24/48 VDC Distr Pnl 1A:

- SRMs A/C.
- IRMs A/C/E/G.
- A OffGas Post Treatment Rad Monitor.
- OffGas PreTreatment Linear Rad Monitor.

2. If Mode Switch is NOT in RUN, a Half SCRAM will occur.

C is incorrect: plausible because a rod block will be generated on the loss of nuclear instrumentation.

D is incorrect: plausible because this would be the correct answer in MODE 1.

Reference: LOA-DC-101 UNIT 1 DC POWER SYSTEM FAILURE rev 24, LOR-1H13-P603-A106 Source Range Monitor High or Inoperative Rev 3
Reference provided during examination: none

Cognitive level: Memory
10CFR Part 55: 41.8

Level (RO/SRO): 2.9/2.9
Tier: 1 Group: 1

PRA: No
K/A: 295004 AK1.03 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Electrical bus divisional separation

SRO Justification: N/A

Question Source: Bank

Question History: Cooper 2008 NRC exam

Comments:

Objective: 006.00.18

EXAMINATION ANSWER KEY

17-1 NRCEXAM

4	ID: 2059013	Points: 1.00

Unit 2 is at 23% RTP.

A Spurious Power Load Unbalance signal is received, and the following is observed:

- All Main Stop Valves (MSVs) remain open
- All Control Valves (CVs) are closed
- The Combined Intermediate Valves (CIVs) are throttled open

What is the status of reactor pressure control?

A. 1 to 2 Bypass valves are controlling reactor pressure.

B. 4 to 5 Bypass valves are controlling reactor pressure.

C. All Main Steam Isolation Valves (MSIVs) are closed, Safety Relief Valves (SRVs) are controlling Reactor pressure.

D. Turbine remains on line and controlling reactor pressure with no Bypass valves open.

Answer: B

Answer Explanation
A is incorrect: but plausible because this would be the status of the bypass valves if the Unit scrams. B is correct: When the power load unbalance trip is received, the main turbine trips, the turbine control valves will fast close due to low RETS pressure, the turbine main stop valves will remain open, and the intermediate valves will initially close and then throttle open to prevent turbine overspeed, with reactor power at 23%, the RPS trip will remain bypassed and no scram will occur. C is incorrect: but plausible because this would be the status for a failure of the turbine valves to close fully, as described in the stem the main stop valves are open and the Intermediate valves are open, if the CVs were also open then the main stem line pressure would drop below the low pressure group 1 setpoint isolation to occur thus requiring SRV operation. D is incorrect: because the stator cooling runback circuit is bypassed at this power limit and is controlled by the Electrical EHC system similar to the power load unbalance circuit. Reference: LOR 2PM02J-B110 U2 Power Load Unbalance rev 2, 1E-2-4029AQ rev F, LOP-EH-11 EHC WORKSTATION ALARM RESPONSE AND OTHER INFORMATION, Rev 23 Reference provided during examination: none Cognitive level: High 10CFR Part 55:41.7 Level (RO/SRO): 3.6/3.7 Tier: 1 Group: 1 PRA: No K/A: 295005 AK2.07 Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: Reactor Pressure Control SRO Justification: N/A Question Source: New Question History: None Comments: Objective: 074.00.05

5	ID: 2059039	Points: 1.00

Which RWLC interlock works in conjunction with the scram profile and what function does it provide for?

A. Total Flow Control Setback, which will prevent reaching the Level 8 trip setpoint.

B. Total Flow Control Setback, which will prevent exceeding the maximum runout flow of the FW pumps.

C. Level Control System Setpoint Setdown, which will prevent reaching the Level 8 trip setpoint.

D. Level Control System Setpoint Setdown, which will prevent exceeding the maximum runout flow of the FW pumps.

Answer: C

Answer Explanation

A is incorrect: but plausible because during a scram the feed pump speed will increase to compensate for the level shrink during a scram, CP Dp can momentarily reach up to 100# due this flow increase. The total flow setback feature does prevent exceeding max pump runout but is not specifically designed as part of setpoint setdown feature.

B is incorrect: but plausible because during a scram the feed pump speed will increase to compensate for the level shrink during a scram, CP Dp can momentarily reach up to 100# due this flow increase. The total flow setback feature does prevent exceeding max pump runout but is not specifically designed as part of setpoint setdown feature.

C is correct: Setpoint Setdown works in conjunction with the scram profile to prevent exceeding the level 8 trip setpoints. Normally, after the initial level shrink, level will increase to above 20 inches. At this point the post scram profile signal is reset and a level setpoint setdown to an approximate 20 inch level setpoint will occur. The level setpoint will then ramp from 20 inches to the level set at the RWLC setpoint station (normally 36 inches). Total feedwater flow at this point is approximately 10% and an automatic bumpless transfer to single-element control will occur. If in single-element control at high reactor power, the post scram profile signal will automatically transfer RWLC system to three-element control unless the system is in single-element because of a total feedwater flow error. Total steam flow error, which will normally cause an automatic transfer to single-element control, will not prevent an automatic transfer to three-element control when the post scram profile signal is activated. Total feedwater flow error will prevent an automatic transfer to three element control in this situation. Although in this condition, the feedwater flow profile is not available, a level setpoint setdown to 20” will occur when level first increases above 20”. This action coupled with normal reactor operator response should prevent over filling the reactor.

D is incorrect: but plausible because the level control setpoint setdown is the correct interlock but setpoint setdown will slow the increase in reactor level by slowing down the feedwater pumps versus Total Flow Control Setback which just limits feed flow of the feed pumps to less than 17.2 million pounds per hour.

Reference: LOP-RL-01 OPERATION OF THE REACTOR LEVEL CONTROL SYSTEM Rev 026, LOP-FW-16 1(2)DS001 OPERATOR STATION ALARM MESSAGE INTERPRETATION rev 37

Reference provided during examination: none

Cognitive level: Memory
10CFR Part 55:41.5

Level (RO/SRO): 3.1/3.3
Tier: 1 Group: 1

PRA: No
K/A: 295006 AK3. Knowledge of the reasons for the following responses as they apply to SCRAM: Reactor water level setpoint setdown: Plant-Specific.

SRO Justification: N/A

Question Source: New

Question History: N/A

Comments:

Objective: 31.00.05

6	ID: 2059072	Points: 1.00

- RCIC is injecting due to an initiation signal
- CY tank level is at 3'
- RCIC suction is from the suppression pool

A control room evacuation was necessary, and the 1E51-F010/31/68 component transfer switch was placed into the Emergency position.

What is the status of RCIC operation 2 minutes after this action?

A. RCIC is not injecting, with the CY tank suction open only.

B. RCIC injecting into vessel, with suction from suppression pool only.

C. RCIC is not injecting, both suppression pool and CY tank suctions open.

D. RCIC injecting into vessel, with suction from suppression pool and CY tank.

Answer: D

Answer Explanation
A is incorrect: plausible because RCIC is initially aligned to the CY tank and with the CY tank low level interlock bypassed it would cause a Low Suction Pressure Trip of the RCIC Turbine. B is incorrect: plausible because the low CY tank interlock which caused the RCIC system to swap its suction flow path to the suppression pool, while operating in the control room this interlock will remain enforced and the CY suction will not re-open, but this interlock is bypassed when the transfer switch is taken to the emergency position with the CY suction switch in the open position. C is incorrect: plausible because with multiple suctions aligned to RCIC it will alter the suction pressure, however, the suction pressure will not lower to the Low Suction Pressure trip setpoint. D is correct: When the transfer switch is placed into the emergency position, the RCIC valves controlled by the transfer switch will re-position to whatever position that the control switches located on the remote shutdown panel are in, the 1E31-F010 RCIC PMP Suction from CY tank will go open despite the low CY tank level, (this interlock will be bypassed when the transfer switch is taken to emergency position), the 1E51-F031 RCIC pump suct from suppression pool will remain open, the 1E51-F068 RCIC TURB Exhaust Isolation Valve will remain open and RCIC will continue to operate. Reference: 1E-1-4226AP/AT/AY/AU, Reactor Core Isolation Cooling RI (E51). Reference provided during examination: none Cognitive level: High 10CFR Part 55: 41.5 Level (RO/SRO): 4.2/4.3 Tier: 1 Group: 1 PRA: No K/A: 295016 AA1.07 Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: Control room/local control transfer mechanisms........ SRO Justification: N/A Question Source: New Question History: N/A Comments: Objective: 54.00.05

EXAMINATION ANSWER KEY

17-1 NRCEXAM

7	ID: 2059097	Points: 1.00

Unit 1 is in Mode 4:

- Reactor Coolant temperature is 150°F
- Alternate shutdown cooling is in operation with heat removal provided by the reactor water cleanup system
- 2 RT filter demineralizers are in service
- The B RWCU Non-Regenerative Heat Exchanger is in use to maintain vessel temperature, with the local D/P gauge 1PDI-WR028, reading of 10.0 psid

Two minutes later:

- The local RBCCW D/P gauge 1PDI-WR028, reads 2 psid
- The WR head tank level is steady

What is the significance in the change of this D/P reading and the consequence of this change?

A. Reduced WR flow to the Non-Regenerative heat exchanger and a high motor temperature trip of the reactor water cleanup pumps.

B. Reduced WR flow to the Non-Regenerative heat exchanger and potential isolation of the RT filters.

C. Increased WR flow to the Non-Regenerative heat exchanger and could violate temperature limits on the reactor vessel.

D. Increased WR flow to the Non-Regenerative heat exchanger and potential heat exchange cavitation.

Answer: B

Answer Explanation

A is plausible because WR flow would be reduced and this could cause the RWCU pump motor temperatures to rise and the pump does have a trip setpoint of 140°F but the pump will still get cooling from CRD purge flow and the temperature rise would be slower than the RWCU system temperature rise, and thus the filter isolation will occur first and the pump will trip on low flow.

B is correct: In LOP-RT-13 RWCU LINEUP FOR HEAT REMOVAL temperature is controlled by setting WR FLOW through the heat exchanger by adjusting WR DP across the heat exchanger, normally DP is set at 8 - 10 psid, thus as DP is raised, this would provide increased cooling.

Per LOP-RT-13 Section D Limitations
D.5. The following actions will increase the heat removal capacity of the system:
D.5.1. Increasing system flow.
D.5.2. Increasing RBCCW flow to Non-Regenerative heat exchanger - max is 10 psid.

A drop in DP indication would indicate a loss of WR flow and thus a decreased cooling capacity, if WR cooling flow is not restored then the filter demineralizers would isolate on RT high coolant temperature of 135°F
Per LOP-RT-13 Step E.6

Ten (10) PSID across the RT Non-Regenerative Heat Exchanger corresponds to an optimum flow of 754 gpm. Flows less than 8.0 psid may cause a RT Isolation due to High Heat Exchanger Outlet Temperature, and flows greater than 10.0 psid may cause Heat Exchanger and/or valve cavitation.

C is incorrect but plausible because the difference between low DP and high DP has to be analyzed and its relationship to system flow, and if cooldown did increase, coolant temperature would drop and if not arrested this condition could violate temperature limits of the vessel.

D is incorrect but plausible because the difference between low DP and high DP has to be analyzed and its relationship to system flow, and per LOP-RT-13 step E.6 high system flow will cause cavitation in the heat exchanger.

Reference: LOP-RT-13 RWCU LINEUP FOR HEAT REMOVAL Rev 19
Reference provided during examination: none

Cognitive level: High
10CFR Part 55: 41.10

Level (RO/SRO): 2.9/2.9
Tier: 1 Group: 1

PRA: No
K/A: 295018 AA2.04 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: System flow

SRO Justification: N/A

Question Source: New Question History: N/A Comments: Objective: 27.00.14

8	ID: 2059119	Points: 1.00

Which of the following Instrument Air/Instrument Nitrogen restoration activities REQUIRE operations to be performed outside of the Main Control Room?

A. Starting a standby SA compressor.

B. Aligning ADS nitrogen supply after a group 10 isolation.

C. Raising pressure for the 100 psig air supply to the drywell.

D. Resetting station air compressor after the second air surge event.

Answer: C

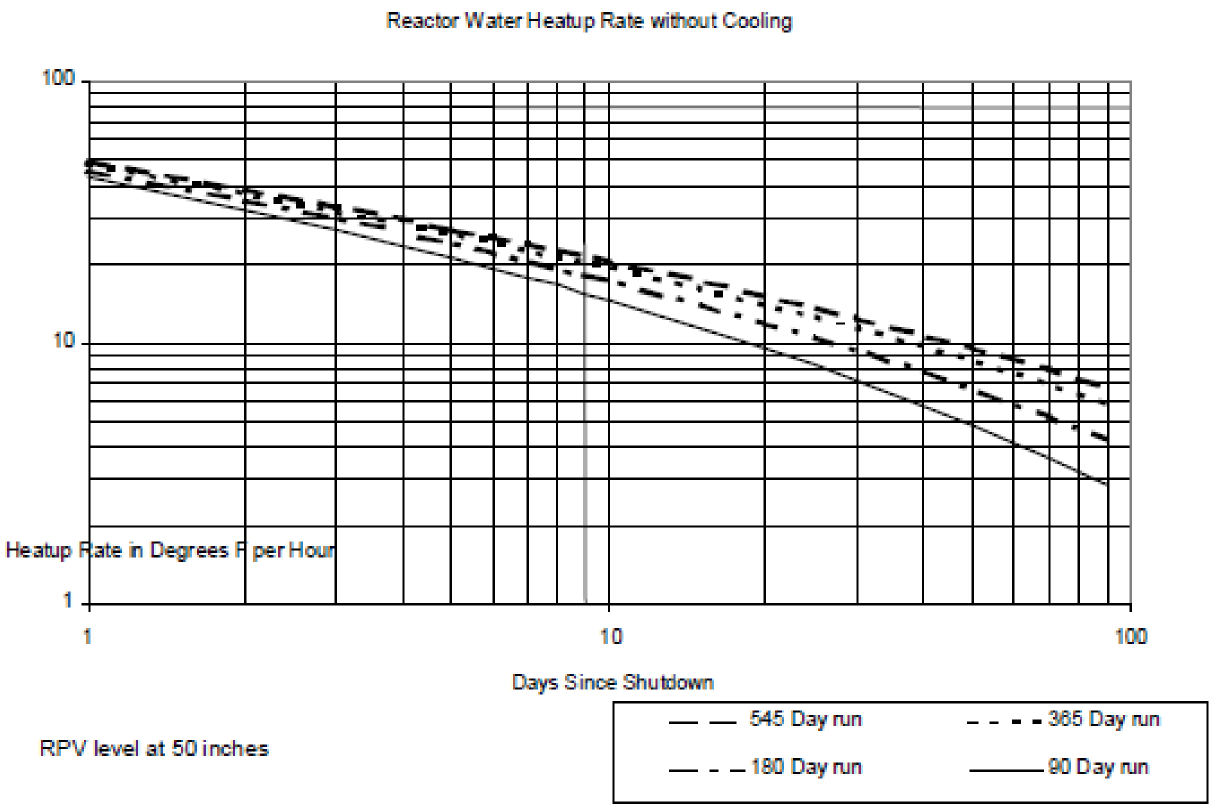
Answer Explanation
A is incorrect: Starting a standby compressor can be performed at the 1PM09J panel B is incorrect: When IN is lost to the drywell, the ADS has its own bottle supply and when pressure drops below 160# the bottle check valve will open and supply the ADS valves. C is correct: Raising pressure for the 100# air supply to the drywell is the only activity that requires an Equipment Operator to perform actions in the field, this activity is performed per LOA-IN-101 LOSS OF DRYWELL PNEUMATIC AIR SUPPLY section B.2 Loss of 100 lb IN D is incorrect: The air compressor now are equipped with an automatic surge reset, the compressor will attempt to reset itself up three times before requiring manual reset Reference: LOA-IN-101 LOSS OF DRYWELL PNEUMATIC AIR SUPPLY rev 11 Reference provided during examination: none Cognitive level: Memory 10CFR Part 55:41.10 Level (RO/SRO): 3.8/4.0 Tier: 1 Group: 1 PRA: No K/A: 295019 Partial or Complete Loss of Instrument Air 2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects SRO Justification: N/A Question Source: New Question History: N/A Comments: Objective: 97.00.21

9	ID: 2059131	Points: 1.00

The reactor has been shutdown for 40 days after a 180 day run.

- RPV level is 50" and steady
- 1E12-F009 Shutdown cooling suction spuriously isolated and will not open
- Reactor coolant temperature is currently 140°F and rising
- Reactor recirculation pumps are not available

How long will it take to FIRST reach MODE 3?



A. 6 hours

B. 7.5 hours

C. 9 hours

D. 10.5 hours

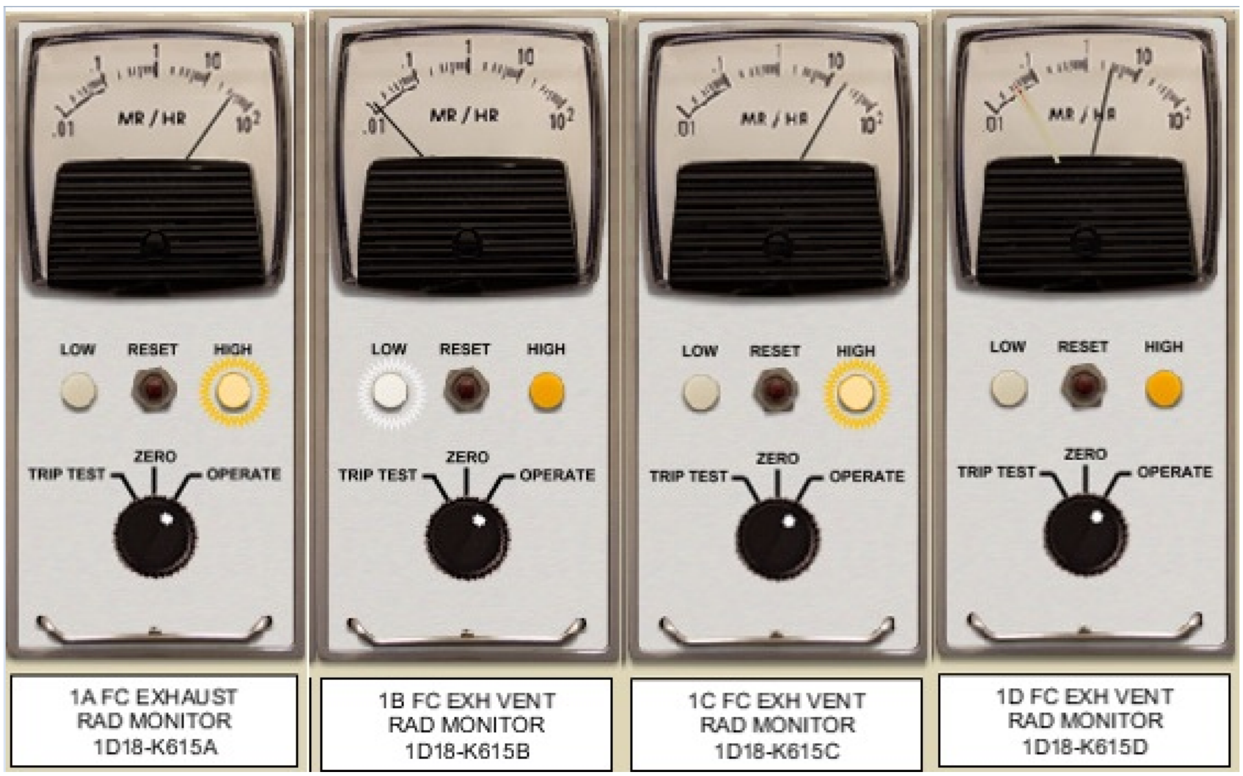
Answer: B

Answer Explanation
A is incorrect: plausible because the 365 day run curve after 40 days is Approximately 10 °F/ hour heat up rate, Mode 3 Temperature is 200 °F(200-140=60), 60°F until reaching Mode 3, 60/10 approximately 6 hours. The actual result would be just over 6 hours since the 365 day line falls just short of the 10 °F/ hour line making 6 hours plausible. B is correct: On a 180 day run the heatup after 40 days is Approximately 8 °F/ hour heat up rate, Mode 3 Temperature is 200°F (200-140=60), 60°F until reaching Mode 3, 60/8 = 7.5 hours C is incorrect: plausible if 212°F is used for entry into MODE 3, the 180 day run the heatup after 40 days is Approximately 8 °F/ hour heat up rate, Mode 3 Temperature is 200°F (212-140=72), 72°F until reaching Mode 3, 72/8 = 9 hours D is incorrect: plausible on a 90 day run the heatup after 30 days is Approximately 6 degrees / hour heat up rate, If using a Temperature of 200 °F(200-140=60), 60°F until reaching 200°F, 60/6 = 10 hours. Kept 10.5 hours for uniformity of distractors. Reference: LOA-RH-101 UNIT 1 RHR ABNORMAL, rev 21 Reference provided during examination: None Cognitive level: High 10CFR Part 55: 41.8 to 41.10 Level (RO/SRO): 3.6/3.8 Tier: 1 Group: 1 PRA: No K/A: 295021 AK1.01 Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING : Decay heat SRO Justification: N/A Question Source: New Question History: N/A Comments: Objective: 64.00.20

10	ID: 2059170	Points: 1.00

Unit 1 is in MODE 5:

A fuel bundle was accidently dropped into the Unit 1 Spent Fuel Pool



What is the status of **Unit 2** reactor building ventilation fans and release status?

A. Running and effluent monitored by VG WRGM

B. Running and effluent monitored by Stack WRGM

C. NOT running and effluent monitored by VG WRGM

D. NOT running and effluent monitored by Stack WRGM

Answer: B

Answer Explanation
A is incorrect: Plausible because the fans will remain running but the SBGT will not have an initiation signal. B is correct: The status of the Rad monitors indicate that a group 4 isolation has not been received. No group 4 dampers would close all the Fans would remain in operation with the release monitored by the Stack WRGM. Per Table 2 from Appendix G, a group 4 isolation requires A and B, or C and D trip channels to cause group isolation. With just the 1A, and the 1C rad monitors in alarm, a group 4 isolation does not exist, the 1B rad monitor is in a downscale condition will NOT generate a trip signal. C is incorrect: plausible because isolation logic of (A or C) and (B or D) is common for other PCIS systems and this describes the status of a group 4 isolation. D is incorrect: plausible because isolation logic of (A or C) and (B or D) is common for other PCIS systems and this describes the status of the VR fans from a group 4 signal. Reference: TRM Appendix G Table 2 Rev 5, LOR-2H13-P601-F305 Div I Fuel Pool Radiation Monitor Downscale Rev 4 Reference provided during examination: None Cognitive level: High 10CFR Part 55: 41.7 Level (RO/SRO): 3.5/3.7 Tier: 1 Group: 1 PRA: No K/A: 295023 AK2.05 Knowledge of the interrelations between REFUELING ACCIDENTS and the following: Secondary containment ventilation.................... SRO Justification: N/A Question Source: New Question History: N/A Comments: Objective: 118.00.10

11	ID: 2086439	Points: 1.00

Which of the following describes the purpose of the Reactor Scram from High Drywell Pressure?

A. The scram limits the amount of energy transmitted to the drywell during a LOCA and limits fuel damage during LOCA.

B. The scram limits the amount of energy transmitted to the drywell during a LOCA and limits decreased core inlet subcooling and steam formation on clad.

C. The scram counteracts the pressure rise by reducing heat generation from the core, which limits drywell pressure from impeding operation of the VQ valves and SRV operation.

D. The scram counteracts the pressure rise by reducing heat generation from the core. High drywell pressure directly affects the life span of EQ components located in the drywell.

Answer: A

Answer Explanation
A is correct: UFSAR chapter 7: High pressure inside the drywell may indicate a break in the reactor coolant pressure boundary. It is prudent to scram the reactor in such a situation to minimize the possibility of fuel damage and to reduce energy transfer from the core to the coolant. The drywell high-pressure scram setting is selected to be as low as possible without inducing spurious scrams. TS Bases 3.3.1.1 function 6: High pressure in the drywell could indicate a break in the RCPB. A reactor scram is initiated to minimize the possibility of fuel damage and to reduce the amount of energy being added to the coolant and the drywell. B is incorrect: Plausible because this is the purpose of the low water level scram. C is incorrect: Plausible because if drywell pressure gets too high it can affect operation of the VQ valves and SRV solenoids D is incorrect: Plausible because pressure in the drywell will affect the operability of drywell equipment and a scram will counteract the high pressure condition. Reference: TS B 3.3.1.1 rev 68, UFSAR Chapter 7.2 rev14 Reference provided during examination: none Cognitive level: Memory 10CFR Part 55: 41.5 Level (RO/SRO): 4.0/4.1 Tier: 1 Group: 1 K/A: EK3.06 Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Reactor SCRAM. SRO Justification: N/A Question Source: Bank Question History: Browns Ferry 2007 NRC Exam Comments: Objective 400.00.14

12	ID: 2059565	Points: 1.00

Unit 1 was at 100% RTP when all CW pumps tripped and were unable to be restarted.

Assuming no operator actions, after 60-minutes you would expect to observe __________ to automatically control reactor pressure.

A. Turbine Bypass Valves throttling at 920 psig

B. Turbine Bypass Valves throttling at 1006 psig

C. One SRV cycling between 896 and 1006 psig

D. One SRV cycling between 976 and 1076 psig

Answer: C

Answer Explanation

A is incorrect: Plausible because 60 minutes after the SCRAM, the Turbine Bypass Valves will be unavailable to control Reactor Pressure since the Main Condenser has lost vacuum due to loss of CW pumps.

B is incorrect: Plausible because 60 minutes after the SCRAM, the Turbine Bypass Valves will be unavailable to control Reactor Pressure since the Main Condenser has lost vacuum due to loss of CW pumps.

C is correct: From scram at 100% power, initially thermal output of reactor will be about 100% power, 7% supplied by decay heat. Eight to ten seconds after scram, thermal output is due mainly to decay heat and drops to 7% of rated thermal output. After approximately one minute, thermal output is 3 to 5% of rated and drops to about 2% after about ten minutes. One hour after scram, decay heat is about 1% rated thermal output.

The decay heat generated after a SCRAM will cause Reactor Pressure in increase. With the Main Condenser unavailable to act as a heat sink, the pressure will be controlled using the SRV's to remove steam from the Reactor Vessel and exhaust into the Suppression Pool.

Without CW pumps, Main Condenser back pressure will rise until the Turbine Bypass Valve trip setpoint is reached, preventing their use. The SRV's will cycle to control Rx pressure. Once two SRVs open (simultaneously), Low-Low Set seals in. Two SRVs, 1B21-F013U and S, will remain open until pressure decreases to 896 psig. After that, 1B21-F013U will cycle between 896 and 1006 psig due to decay heat.
One SRV is adequate to control pressure because with an operating history of 90% power, after 60 minutes decay heat will be approximately 1% and flow through one SRV is sufficient to handle pressure rise due to 1% decay heat. With LLS sealed in SRV "U" will cycle between 896 and 1006 psig.

D is incorrect: Plausible because this is the normal relief setpoint for an SRV, however, initially two SRVs would have opened, activating the LLS logic which changes the setpoint at which the SRV is controlled.

References LGP-2-1, Attachment A rev. 115, Normal Unit Shutdown, TS Bases 3.4.4, rev. 0, LOR-1PM03J-B511, rev. 6

References used on exam: None

Cog Level: High

License Level: RO

KA 295025 High Reactor Pressure A1.03 Ability to operate and/or monitor the following as they apply to High Reactor Pressure: Safety/relief valves: Plant Specific

CFR: 41.5 / 45.3

Question Source: Bank

Question History: None

Objective: 070.00.14

13	ID: 2061662	Points: 1.00

Unit 1 is at 100% RTP:

- A LOCA occurs
- Suppression pool temperature is 101˚F and rising 2˚F/min
- Reactor Pressure is 800 psig and lowering 10 psig/min
- All MSIVs are shut
- Drywell pressure is 8 psig and rising 0.2 psig/min
- Level is 20” and stable on Narrow Range with feedwater

Per LGA-003, PRIMARY CONTAINMENT CONTROL, if suppression pool temperature cannot be held below __(1)__ all available pool cooling should be started.

Given the plant conditions above, suppression pool cooling should be started __(2)__.

A. (1) 105 ˚F
(2) After Exceeding 105 ˚F

B. (1) 110 ˚F
(2) After Exceeding 110 ˚F

C. (1) 105 ˚F
(2) Immediately

D. (1) 110 ˚F
(2) Immediately

Answer: C

Answer Explanation
A is incorrect: Plausible because LGA-003 directs to hold temperature below 105 ˚F. If you can not hold temperature below 105 ˚F to go to step 18 which has you start all available pool cooling per LGA-RH-103. Incorrect because with the current heatup rate of the suppression pool, 105 ˚F will be exceeded. B is incorrect: Plausible because 110 ˚F is a before step in LGA-003 under the pool temperature leg. Incorrect because the action is to scram not start pool cooling. C is correct: When it is determined that suppression pool temperature cannot be maintained below the value of the most limiting suppression pool temperature LCO, a conclusion that may be reached in advance of suppression pool temperature actually reaching this value, the general direction of Step SP/T is supplemented with the explicit instruction to place into operation all available methods by which suppression pool cooling can be effected. D is incorrect: Plausible because the determination can be made that 110 ˚F would be exceeded with current plant trends. Incorrect because 110 ˚F is the criteria to scram on high suppression pool temperature. References: LGA-003 rev17, Primary Containment Control, LPGP-PSTG-01S05A rev 4 PLANT SPECIFIC TECHNICAL GUIDELINES SECTION 5A – PRIMARY CONTAINMENT CONTROL – ENTRY, SUPPRESSION POOL TEMPERATURE, AND DRYWELL TEMPERATURE CONTROL Reference provided during examination: LGA-003 Upper Pool Temp Leg Cognitive level: High Level (RO/SRO): RO Tier: 1 Group: 1 KA: 295026 EA2.01 Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature 10 CFR Part 55 Content: 41.10 / 43.5 / 45.13 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 421.010

14	ID: 2061635	Points: 1.00

Unit 1 was manually scrammed due to a small break LOCA in the primary containment.

- Drywell pressure is 2.0 psig and slowly rising
- Drywell temperature is 215°F and slowly rising

Which of the following identifies the operational implications of using LGA-VP-01, PRIMARY CONTAINMENT TEMPERATURE REDUCTION?

A. Use of LGA-VP-01 is NOT acceptable; its use could cause possible breach of the primary containment from water hammer.

B. Use of LGA-VP-01 is NOT acceptable; its use could cause possible breach of the primary containment due to evaporative cooling effect.

C. Use of LGA-VP-01 IS acceptable; it is required to reduce containment temperatures to prevent level reference leg boiling.

D. Use of LGA-VP-01 IS acceptable; it is required to reduce containment temperatures to preserve SRV solenoid operability.

Answer: A

Answer Explanation
A is correct: Use of VP under these conditions is prohibited. The reason is with high drywell pressure the VP system has isolated with temperatures above 212 degrees F. The water in the piping can boil lifting the reliefs on the piping. The piping has the potential for voiding and if system is un-isolated and restarted these actions may cause water hammer, VP piping damage, and potentially a leak in the drywell or a release path outside the primary containment. B is incorrect: This distractor is plausible because in a small break LOCA scenario, drywell temperature is expected to get very high without a large rise in containment pressure. With these conditions evaporative cooling is a concern. This issue is addressed further down the temperature leg of LGA-003 with the use of drywell sprays and using the "Drywell Spray Initiation Limit" curve as a guideline, but is incorrect answer based on directions from the LGA-003 and the precautions from LGA-VP-01. C is incorrect: This distractor is plausible because reference leg flashing is a concern for elevated temperatures in the drywell, but is incorrect answer based on LGA-003 PRIMARY CONTAINMENT CONTROL and the precautions from LGA-VP-01 PRIMARY CONTAINMENT TEMPERATURE REDUCTION. D is incorrect: This distractor is plausible because in a small break LOCA scenario, drywell temperature is expected to get very high and challenge SRV solenoid operability. Actions to address this concern is included further down the temperature control leg, but is incorrect answer based on the directions from the LGA-003 and the precautions from LGA-VP-01. Reference: LGA-003 PRIMARY CONTAINMENT CONTROL rev 17, LGA-VP-01 PRIMARY CONTAINMENT TEMPERATURE REDUCTION rev 8 Reference provided during examination: None K/A: 295028 High Drywell Temperature (Mark I/Mark II only) G.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 Cognitive level: High Level: RO Tier: 1 Group: 1 Question Source: 13-1 NRC Exam 1262345 Question History: Bank 10 CFR Part 55 Content: 41.5 / 43.5 / 45.12 SRO Justification: N/A Comments: None Associated objective(s): 096.00.20

15	ID: 2062114	Points: 1.00

What is the reason that -18 feet was chosen for the Suppression Pool Level at which ADS may not be initiated?

A. -18 feet is the lowest suppression pool level that can be read in the Control Room.

B. At -18 feet insufficient volume of water exists in the suppression pool to absorb the energy from an ADS blowdown.

C. At -18 feet the SRV T-quenchers are uncovered and steam suppression during an ADS blowdown can no longer be assured.

D. -18 feet is the lowest suppression pool level which can be maintained without exceeding the suppression pool boundary design load if SRVs are opened.

Answer: A

Answer Explanation
A is correct: -22.6 feet is the top of SRV discharge device. If the SRVs were opened with the discharge devices exposed, steam would pass directly into the suppression chamber airspace, bypassing the suppression pool. The resulting pressure increase could exceed the maximum pressure capability of the primary containment. -18 feet is the lowest level that can be read in the Control Room and was chosen as the value at which SRVs are not used per LGA-004 because it is not assured that the SRV discharge device is covered. The SRV discharge device is located at a suppression pool level of -22.6 feet. B is incorrect: Plausible because low suppression pool levels lower the inventory of the quenching volume and at -12 ft the abnormal suppression pool level curve limits how much energy can be placed in the suppression pool. C is incorrect: T-quenchers are located at -22.6 ft, so with suppression pool level at -18 ft they are still submerged. Plausible because if SRVs were opened with the T-quenchers exposed, steam would bypass directly into the suppression chamber airspace and steam suppression would not be assumed. D is incorrect: Plausible because suppression pool level is a factor in the PSP limit and with suppression pool level at -18ft plant conditions are outside the PSP limit. The portion of PSP associated with exceeding suppression pool boundary design load is associated with suppression pool levels >0". The low suppression pool level limit is associated with uncovering the drywell downcomers at -12 ft. References: LPGP-PSTG-01S13 rev 3, Figures, LPGP-PSTG-01S05A rev 4, PRIMARY CONTAINMENT CONTROL – ENTRY, SUPPRESSION POOL TEMPERATURE, AND DRYWELL TEMPERATURE CONTROL, LPGP-PSTG-01S09 CONTINGENCY #2 EMERGENCY RPV DEPRESSURIZATION rev. 3 Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 1 Group: 1 KA: 295030 Low Suppression Pool Water Level K1.01 Knowledge of the operational implications of the following concepts as they apply to Low Suppression Pool Water Level: Steam Condensation 10 CFR Part 55 Content: 41.8 - 41.10 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 638.010

16	ID: 2059706	Points: 1.00

Unit 2 was at 100% RTP when a LOCA occurred in the drywell:

- Reactor Pressure is 825 psig and lowering slowly
- Drywell Pressure peaked and stabilized at 1.1 psig
- Reactor water level is -130" and lowering slowly
- All low pressure ECCS pumps are running

What is the status of ADS?

A. Initiated when the first low pressure ECCS pressure switch picked up.

B. Not Initiated. ADS will initiate 100 seconds after reactor water level dropped below level 1.

C. Not Initiated, ADS will initiate 510 seconds after reactor water level dropped below level 1.

D. Not Initiated, ADS will initiate 610 seconds after reactor water level dropped below level 1.

Answer: D

Answer Explanation
A is incorrect: Plausible because the low pressure ECCS contact picks up downstream of both 100 (nominal field setting) second timer and 510 (nominal field setting) second timer. This is the last contact that needs to be closed in the flowpath for ADS to energize ADS solenoids. B is incorrect: Plausible because the 100 second (nominal field setting) or '105 second' timer is associated with level 1. Incorrect because the high drywell pressure signal was not made up. C is incorrect: Plausible because the 510 second (nominal field setting) timer would bypass the high drywell pressure signal. Incorrect because the 105 second timer is in series with the high drywell pressure signal and would not start timing out until the 100 second (nominal field setting) timer elapsed. D is correct: An ADS requires the following signals: Low level 1 -127", High Drywell Pressure 1.69 psig, Level 3 confirmatory, a low pressure ECCS pump running, and the 105 (100 seconds nominal field setting) second timer to time out. Alternately, low level 1 for 9 minute (510 second nominal field setting) timer bypasses the requirement for high drywell pressure. In this instance, ADS will occur 610 seconds (nominal field setting) after level dropped below level 1. References: LOP-MS-03, PREPARATION FOR STANDBY OPERATION OF THE AUTOMATIC DEPRESSURIZATION SYSTEM rev 11, LES-NB-202A rev 14, Unit 2 Division ADS Timers Calibration and Functional Test Reference provided during examination: 1E-2-4201AB Cognitive level: High Level (RO/SRO): RO Tier: 1 Group: 1 K/A: 295031 Reactor Low Water Level K2.08 Knowledge of the interrelations between Reactor Low Water Level and the following: Automatic depressurization system. 10 CFR Part 55 Content: 41.7 45.9) SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 062.00.05

17	**ID: 2059717**	**Points: 1.00**

When Cold Shutdown Boron Weight has been added and mixed uniformly, the reactor will remain shut down under which of the following conditions:

1. All control rods are fully withdrawn.
2. The reactor core is at its most reactive exposure.
3. No xenon is present in the reactor core.
4. No voids are present in the reactor core.
5. RPV water level is at the high level trip setpoint.

A. 1, 2, 3, 4, 5

B. 2, 3, 4, 5 ONLY

C. 1, 3, 4, ONLY

D. 1, 2, 5 ONLY

Answer: A

Answer Explanation
A is correct: The system is designed to bring the reactor from rated power to a cold shutdown condition at any time in core life. The negative reactivity reduces reactor power from rated to zero level and allows cooling the nuclear system to room temperature, with the control rods remaining withdrawn in the rated power pattern. It includes the reactivity gains that result from complete decay of the rated power xenon inventory. It also includes the positive reactivity effects from eliminating steam voids, changing water density from hot to cold, reduced Doppler effect in uranium, reducing neutron leakage from boiling to cold, and decreasing control rod worth as the moderator cools. UFSAR 9.3-14 The Cold Shutdown Boron Weight (CSBW) is that amount of soluble boron which, if injected into the RPV and mixed uniformly, will maintain the reactor shutdown under all conditions. 1. All control rods are fully withdrawn. 2. The reactor core is at its most reactive exposure. 3. No xenon is present in the reactor core. 4. No voids are present in the reactor core. 5. RPV water is at its most reactive temperature. 6. RPV water level is at the high level trip setpoint. 7. All shutdown cooling is in service and RWCU is operating in the recirculation mode. LPGP-PSTG-01S15B B is incorrect: Plausible because 2,3,4,5 are correct answers. 1 is also a correct answer. No voids in the core is an assumption in Hot Shutdown Boron Weight. C is incorrect: Plausible because 1,3,4 are correct answers. 2 and 5 are also correct answers. No voids in the core is an assumption in Hot Shutdown Boron Weight. D is incorrect: 1,2,5 are all correct answers. 2 and 3 are also correct answers. No voids in the core is an assumption in Hot Shutdown Boron Weight References: LPGP-PSTG-01S15B, Rev 0, PLANT SPECIFIC TECHNICAL GUIDELINES SECTION 15B – TABLES PART 2, UFSAR 9.3-14 rev 19, Process Auxiliaries Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 1 Group: 1 KA: 295037 SCRAM Condition Present and Power above APRM Downscale or Unknown K3.05 10 CFR Part 55 Content: 41.5, 45.6 SRO Justification: N/A Question Source: New Question History: N/A

18	ID: 2059854	Points: 1.00

Effluent radiation release rate is 5.30 e +8 uCi/sec from the Standby Gas Treatment Wide Range Gas Monitor.

What color is the SPDS RAD RELEASE NORM (HIGH) status box?

A. RED

B. CYAN

C. GREEN

D. YELLOW

Answer: A

Answer Explanation
A is correct: 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. CYAN is used for insufficient data or data out of range. The sum of the SBGT and station vent stack release rates is compared to the high alarm limit. If the value for any effluent path exceeds the alarm point or has no valid inputs, the status box color will change to RED or CYAN, respectively. 1.22e7 uCi/sec is the setpoint for both stack effluent, vg wrgm effluent, and effluent release (stack and vg effluent) B is incorrect: plausible because Cyan is used for invalid inputs or data out of range. C is incorrect: plausible because Green is used when a condition is not as expected or a limit has been exceeded. This value is below the required value to in LGA-009 to initiate a blowdown. D is incorrect: plausible because Yellow is used on many displays when a valve is within 2% of a limit. Reference: LOP-CX-93, Rev 4, PLANT PROCESS COMPUTER ALARM MESSAGE INTERPRETATION, LOP-CX-07, REV 17, PLANT PROCESS COMPUTER MISCELLANEOUS FUNCTIONS and LOP-CX-02, Rev 8, SAFETY PARAMETER DISPLAY SYSTEM UFSAR 7.8.2.2.2.5 rev 16 Reference provided during examination: N/A Cognitive level: Memory Level (RO/SRO): RO Tier: 1 Group: 1 K/A: 295038 High Off-site Release Rate A1.04 Ability to operate and/or monitor the following as they apply to High Off-site Release Rate: SPDS/ERIS/CRIDS/GDS: Plant Specific 41.7 / 45.6 *SRO Justification: N/A* Question source: New Question History: N/A Comments: Associated objective(s): 50.00.05

19 **ID: 2061636** **Points: 1.00**

LOA-FX-101, UNIT 1 Safe Shutdown with a Fire in the MCR or AEER procedure directs you to OPEN RPS System A and B feed breakers (CB2A and CB2B) located at the ______(1)______ on the 749 foot elevation of the Auxiliary Building.

These breakers are left open until recovery to ______(2)_______.

A. (1) RPS Distribution Panel
(2) close PCIVs and prevent valves from opening from electrical shorts

B. (1) RPS MG Sets
(2) close PCIVs and prevent valves from opening from electrical shorts

C. (1) RPS Distribution Panel
(2) prevent resetting the reactor scram

D. (1) RPS MG Sets
(2) prevent resetting the reactor scram

Answer: A

Answer Explanation
A is correct: The RPS System A and B feed breakers are opened as a backup means of scramming the reactor and to close the containment and reactor vessel isolation valves. They are left open until recovery to ensure against inadvertent reset caused by shorts. B is incorrect: The breakers are located at the RPS distribution panel. Plausible because the RPS MG set is accessible when exiting the MCR and the RPS MG sets are capable of being tripped locally at the RPS MG sets C is incorrect: Plausible because this would prevent the operator from resetting the SCRAM, however the SCRAM is not able to be reset from the remote shutdown panel. D is incorrect: The breakers are located at the RPS distribution panel. The RPS MG set is accessible when exiting the MCR and the RPS MG sets are capable of being tripped locally at the RPS MG sets. Plausible because this would prevent the operator from resetting the SCRAM, however the SCRAM is not able to be reset from the remote shutdown panel. References: LOA-FX-101 rev 30, UNIT 1 Safe Shutdown with a Fire in the MCR or AEER Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 1 Group: 1 KA: 600000 A2.16 Ability to determine and/or interpret the following as they apply to Plant Fire On Site: Vital equipment and control systems to be maintained and operated during a fire. IR 3.0 10 CFR Part 55 Content: 41.7 SRO Justification: N/A Question Source: Bank Question History: None Comments: Associated objective(s): 054.00.20

20	ID: 2086256	Points: 1.00

Unit 2 is operating at 100% RTP.

- HPCS is running in full flow test for pump surveillance testing.
- A transient on the grid causes a grid voltage disturbance.
- Bus 243 voltage drops to 3800 volts.

What will be the Division 3 system response to these conditions?

The Bus will load shed ___1____, to prevent damaging HPCS pump motor _____2_____

A. 1) Immediately
2) on an Auto start condition

B. 1) Immediately
2) from over current during the pump surveillance

C. 1) In 5 minutes
2) on an Auto start condition

D. 1) In 5 minutes
2) from over current during the pump surveillance

Answer: D

Answer Explanation
A is incorrect: plausible because this could be a condition if the HPCS pump was running and the bus trips could trip on an overcurrent since the lower voltage causes the bus to draw more current and the DG will auto start on a degraded condition but only after five minutes has elapsed. B is incorrect: plausible because if actions were taken from LOA-AP-201 Unit 2 AC Power System Abnormal, which directs manually tripping ACB 2432. The 2B DG would start immediately and close in to supply bus 243. C is incorrect: plausible because it is a homogeneous distractor, a mixture of possible conditions of the Bus and DG. D is correct: With voltage at 3800 volts, this represents a degraded voltage condition, per LOR-1H13-P601-A402 4KV Bus 143/143-1 Undervoltage 4KV Bus 143 Degraded Voltage, this condition will initiate a 5 minute timer, after five minutes the SAT feed breaker will trip and the DG will start. If the elapsed time has only been 4 minutes and with NO operator action, voltage will remain at 3700 VAC and the DG will not have started. This degraded voltage protection is provided to ESF systems to protect ESF components from degrading grid conditions. References: LOA-AP-201 Unit 2 AC Power System Abnormal rev 52 LOR-1H13-P601-A402 4KV Bus 143/143-1 Undervoltage 4KV Bus 143 Degraded Voltage Rev 8 Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 1 Group: 1 KA: 700000 Generator Voltage and Electric Grid Disturbances G01.28 Knowledge of the purpose and function of major system components and controls 10 CFR Part 55 Content: 41.7 SRO Justification: N/A Question Source: Bank Question History: N/A Comments: Associated objective(s): 11.00.14

21 **ID: 2058930** **Points: 1.00**

Per LGA-003, PRIMARY CONTAINMENT CONTROL, Low Suppression Pool Level leg, a level of ______(1)______ in the suppression pool, is the first level where ______(2)_______.

A. (1) -10 Feet
(2) damage may occur to RHR pumps due to exceeding the ECCS Vortex limit

B. (1) -10 Feet
(2) steam may not be adequately condensed and primary containment pressure could exceed allowable limits.

C. (1) -12 Feet
(2) damage may occur to RHR pumps due to exceeding the ECCS Vortex limit

D. (1) -12 Feet
(2) steam may not be adequately condensed and primary containment pressure could exceed allowable limits.

Answer: D

Answer Explanation
A is incorrect: Plausible since -10 Feet is the RCIC NPSH concern for suppression pool level. B is incorrect: Damage may occur if the NPSH Limit is approached, however the ECCS Vortex limit is -18 Feet. Plausible since the RCIC vortex limit is at -11.4 feet. C is incorrect: Damage may occur if the NPSH Limit is approached, however the ECCS Vortex limit is -18 Feet. Plausible since the RCIC vortex limit is at -11.4 feet. D is correct: If suppression pool water level cannot be maintained above the specified minimum value, steam may not be adequately condensed and primary containment pressure could exceed allowable limits. References: LPGP-PSTG-01S05B Primary Containment Control Sect 5B rev2, LGA-003 rev17 Primary Containment Control Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 1 Group: 2 KA: 295010 High Drywell Pressure K1.01 Knowledge of the operational implications of the following concepts as they apply to High Drywell Pressure: Downcomer submergence: Mark-I&II 10 CFR Part 55 Content: 41.8 – 41.10 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 423.00.02

22 **ID: 2059544** **Points: 1.00**

Per LGA-010, FAILURE TO SCRAM:

You have _____(1)_____ minutes to place _____(2)_____ RHR Heat Exchanger(s) online during a power ATWS with all heat going to the primary containment.

A. (1) 15
(2) two

B. (1) 10
(2) two

C. (1) 15
(2) one

D. (1) 10
(2) one

Answer: B

Answer Explanation
A is incorrect: Plausible because this is the time a heat exchanger is required to be put on in a Station Black Out. Plausible because both RHR heat exchangers are required to placed online in 10 minutes with an ATWS with all heat going to the suppression pool. B is correct: Both RHR heat exchangers are required to placed online in 10 minutes with an ATWS with all heat going to the suppression pool. C is incorrect: Plausible because this is the time a heat exchanger is required to be put on in a Station Black Out. D is incorrect: Caution 8 of LGA-003: At least one RHR Heat Exchanger must be in service within 10 minutes in a DBA LOCA. Design analyses assumes that the heat exchanger will be placed online in this timeframe to remove heat from the primary containment. Placing suppression pool cooling online is directed by LGA-003 after suppression pool temperature reaches 105F but before reaching 110F. References: LPGP-PSTG-01S03 rev 3, Plant Specific Technical Guidelines Section 3 – Cautions, LPGP-PSTG-01S05A rev 4, PLANT SPECIFIC TECHNICAL GUIDELINES SECTION 5A – PRIMARY CONTAINMENT CONTROL – ENTRY, SUPPRESSION POOL TEMPERATURE, AND DRYWELL. LGA-010 rev 18 TEMPERATURE CONTROL, LGA-003 rev17 Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 1 Group: 2 KA: 295013 High Suppression Pool Temperature K2.01 Knowledge of the interrelations between High Suppression Pool Temp. and the following: Suppression pool cooling 10 CFR Part 55 Content: 41.7 / 45.8 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 064.00.22

23	ID: 2059020	Points: 1.00

A 20 rod ATWS occurs on Unit 1.

Per LGA-NB-01, ALTERNATE ROD INSERTION, which of the following steps must be completed to defeat any rod block(s) and insert control rods using the ‘INSERT’ pushbutton?

1) Bypass RWM
2) Bypass CRD drive flow trip circuit switch
3) Reset ARI
4) Reset RPS

A. 1 & 2 ONLY

B. 1 & 3 ONLY

C. 2 & 4 ONLY

D. 1, 2, 3 & 4

Answer: A

Answer Explanation
A is correct: Bypassing the RWM removes insert and withdraw blocks. Bypassing the CRD drive flow trip circuit switch allows for manual insertion of control rods by disabling the 1C11-N900 and 1C11-N901 trip units. This allows signals to be sent from RCMS to the CRD system. B is incorrect: Plausible because resetting the SCRAM allows SDV vent and drain valves to open and drain the SDV but is not necessary for inserting control rods with Method 3. Resetting ARI will close valves that will let the SCRAM air header repressurize and open the SDV vent and drain valves. However this is not necessary for insertion of control rods with LGA-NB-01 Method 3. C is incorrect: Plausible because resetting the SCRAM allows SDV vent and drain valves to open and drain the SDV but is not necessary for inserting control rods with Method 3. D is incorrect: Plausible because resetting the SCRAM allows SDV vent and drain valves to open and drain the SDV but is not necessary for inserting control rods with Method 3. Resetting ARI will close valves that will let the SCRAM air header repressurize and open the SDV vent and drain valves. However this is not necessary for insertion of control rods with LGA-NB-01 Method 3. References: LOR-1H13-P603-A501 CRD DRIVE FLOW TRIP CIRCUIT ABNORMAL rev2, LGA-NB-01 Alternate Rod Insertion rev 19 Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 1 Group: 2 KA: 295015 Incomplete Scram K3.01 Knowledge of the reasons for the following responses as they apply to Incomplete SCRAM: Bypassing rod insertion blocks. 10 CFR Part 55 Content: 41.5, 45.6 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 431.010

24 **ID: 2061710** **Points: 1.00**

A leak in the RWCU system has occurred causing a high temperature isolation.

This isolation occurred at a temperature of ____(1)____ measured in the ___(2)___.

A. (1) 120°F
(2) RT Pump Room

B. (1) 146°F
(2) RT Pump Room

C. (1) 120°F
(2) RT HX Room

D. (1) 146°F
(2) RT HX Room

Answer: D

Answer Explanation
A is incorrect: The RT Pump Room Isolation occurs at 198°F. Plausible because this is the correct setpoint for RCIC Leak Detection Equipment Ambient Temp Alarm. B is incorrect: The RT Pump Room Isolation occurs at 198°F. Plausible because a leak detection isolation will occur on at 146°F in the RT HX Room C is incorrect: The RT HX Room Isolation occurs at 146°F. Plausible because this is the correct setpoint for RCIC Leak Detection Equipment Ambient Temp Alarm. D is correct: A leak detection isolation will occur on at 146°F in the RT HX Room References: LOR-1H13-P601-C211 rev 8, Leak Detection Reactor Water Cleanup System A Ambient Temperature High (Div I) , LOR-1H13-P601-B506 rev 6, Leak Detection Reactor Water Cleanup System B Differential Temperature High (Div II), LOR-1H13-P601-D507, Rev. 6 RCIC Pipe Routing Equipment Area Temperature High Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 1 Group: 2 KA: 295032 High Secondary Containment Area Temperature A1.02 Ability to operate and/or monitor the following as they apply to High Secondary Containment Area Temperature: Leak detection system concept: Plant-Specific 10 CFR Part 55 Content: 41.7 / 41.10 / 43.1 / 45.13 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 027.00.05

25	ID: 2062245	Points: 1.00

Which of the following will cause a Reactor Building Ventilation Hi Rad condition?

A. A leak on the suction side of LPCS during a surveillance.

B. Swapping CD/CB Pumps with hydrogen water chemistry on-line.

C. Installed CY Reactor Cavity Sprays fail during vessel drain down.

D. Drywell Inerting in progress and a Reactor Recirc Pump Seal Fails.

Answer: C

Answer Explanation
A is incorrect: The contamination levels of the Suppression Pool are not high enough to allow for a VR High Rad condition. Plausible because Reactor Building Ventilation supports LPCS and it takes a suction form the Suppression Pool. B is incorrect: Swapping CD/CB pumps does not release contamination to the secondary containment ventilation system. Plausible because Main Steam Line High Monitors rad Hi alarm have spuriously alarmed from CD/CB swaps with HWC on-line. This phenomenon is an actual radiation level change induced by N-16 production, which is a normal byproduct of H-2 gas injection into the reactor. This has also caused Offgas Pretreatment Hi Rad Alarms. C is correct: During drain down cavity sprays are used to minimize the radiation levels and airborne activity on the cavity walls. This would cause a high rad condition on the refuel floor and would be seen throughout the Reactor Building with ventilation aligned to the refuel floor. D is incorrect: When inerting is performed it is either by VQ only or VQ and Standby Gas Treatment and would not go through the portion of VR instrumentation is located. Plausible because the Recirc Leak would cause a High Rad condition in the Drywell and that condition would be seen by any rad monitor in the ventilation flowpath. Reference: LOP-FC-16, Reactor Vessel/Cavity Draindown Via RHR SDC Rev. 26; LOR-1H13-P601-F204, Div 1 Reactor Building Ventilation Radiation High-High Rev. 4, LOR-1H13-P601-F403, MSL A/B Radiation Monitor Hi-Hi, Rev. 9; LOR-1N62-P600-B502, Off Gas Pre-Treatment Radiation Monitor High Radiation, Rev. 8 Reference provided during examination: None Cognitive level: High 10CFR Part 55: 41(b)(10) Level (RO/SRO): 3.7/4.2 Tier: 1 Group: 2 PRA: No K/A: 295034 EA2.02 Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Cause of high radiation levels SRO Justification: N/A Question Source: New Question History: N/A Comments: Objective: 51.00.21

EXAMINATION ANSWER KEY

17-1 NRCEXAM

26	ID: 2059508	Points: 1.00

Unit 2 is in Mode 5

- VR isolated and shutdown.
- Refueling operations are in progress.

Unit 1 is in Mode 1

- A steam leak in the Unit 1 main steam tunnel causes a high differential pressure condition in the steam tunnel.

Reactor Building Differential pressure will challenge the operability of secondary containment on ____(1)____.

Standby Gas Treatment ____(2)____.

A. (1) Neither Unit
(2) starts on Unit 1 ONLY

B. (1) Neither Unit
(2) starts on Unit 1 & Unit 2

C. (1) Unit 1 and Unit 2
(2) starts on Unit 1 ONLY

D. (1) Unit 1 and Unit 2
(2) does not start on either unit

Answer: D

Answer Explanation
A is incorrect: VG does not start on either unit. Plausible that VG would start on unit 1 only since the VR isolation from the high differential pressure condition affects Unit 1 only. B is incorrect: Main steam tunnel differential pressure isolates VR for the affected unit only and does not start VG on either unit. Plausible because all group 4 isolation signals isolate VR dampers on both units and start VG on both units. Plausible because if VG was running secondary containment per LCO 3.6.4.1 would not be challenged per the design basis of the VG system. C is incorrect: VG on starts from a valid group 4 PCIS signal not steam tunnel high differential pressure. Plausible because secondary containment will be challenged for both Units since VR is not in operating and VG will not be running. Plausible that VG would start on unit 1 only since the VR isolation from the high differential pressure condition affects Unit 1 only. D is correct: Main steam tunnel differential pressure isolates VR for the affected unit only, but since VR is shutdown on Unit 2, secondary containment will be challenged for both units. VG will not start on either unit since high steam tunnel differential pressure is not a Group IV isolation signal. References: LGA-VR-01 rev 9, 1E-1-4083AG rev L, 1E-1-4083AF rev C Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 1 Group: 2 KA: 295035 Secondary Containment High Differential Pressure G02.38 Knowledge of conditions and limitations in the facility license. 10 CFR Part 55 Content: 41.7, 41.10 ,43.1, 45.13 SRO Justification: N/A Question Source: New Question History: N/A Comments: RO justification - operability of the secondary containment in systems level knowledge. Associated objective(s): 118.00.05

27	ID: 2078599	Points: 1.00

Unit 1 is at 100% RTP.

Fire Protection is leaking into the reactor building 694' elevation, southwest corner room.

What are the operational implications to the HPCS system, due to rising water level?

A. HPCS low water level transmitter failure.

B. HPCS full flow test valve trip overcurrent.

C. HPCS discharge valve trip on overcurrent.

D. HPCS Water Leg Pump trip on overcurrent.

Answer: D

Answer Explanation
A is incorrect: plausible because it is a component associated with HPCS and if in a moist condition this could cause a ground situation, but it is located at Bus 143-1 B is incorrect: plausible because this is a component associated with the HPCS system and if submerged its AC motor could short to ground but both locations are not directly in the SW corner room. C is incorrect: plausible because this is a component associated with the HPCS system and if submerged its AC motor could short to ground but both locations are not directly in the SW corner room. D is correct: When Water level in the HPCS rises above the floor level, this is greater than MAX safe level and entry into LGA-002 would be required. The first component affected by rising levels in the southwest corner room would be the HPCS water leg pump, rising water levels would cause a ground on the AC motor causing a malfunction and tripping of the pump motor. With the loss of the HPCS water leg pump, the system discharge piping would start voiding. The basis for Max safe values for water level from LGA-002 is the location of the water leg pumps located in the 694' RB elevations. Reference: LGA-002 Secondary Containment Control rev 9, M-95 Rev AQ, M-11 General Arrangement Basement Floor Plan Rev Q Reference provided during examination: None Cognitive level: Memory 10CFR Part 55:41.(b)(7) Level (RO/SRO): 2.6/2.8 Tier: 1 Group: 2 PRA: No K/A:295036 K1.02 Secondary Containment High Sump/Area Water Level Knowledge of the Operational Implications of the following concepts as they apply to Secondary Containment High Sump/Area Water Level: Electrical ground/ circuit malfunction SRO Justification: N/A Question Source: New Question History: N/A Comments: Objective: 61.00.21

EXAMINATION ANSWER KEY

17-1 NRCEXAM

28	**ID: 2062304**	**Points: 1.00**

What are the normal and emergency power supplies to the 1A RHR Water Leg Pump?

A. 133, 0 DG

B. 135Y, 0 DG

C. 134, 1A DG

D. 136Y, 1A DG

Answer: B

Answer Explanation
A is plausible because Bus 133 is energized by division 1 4 KV Bus 141Y, with the emergency power supply from the 0 DG. Bus 133 also supplies power to risk significant equipment for PRA but is incorrect because power for the waterleg pump comes from 135Y. B is correct: 135Y is the normal power supply to the water leg pump for ‘A’ RHR. Keeping the discharge piping of the RHR system filled prevents water hammer during system initiation. The 0 DG provides power to division 1 safety related components. C is plausible because Bus 134 is energized by the division 2 4KV Bus 142Y, with the emergency power supply from the 1A DG. The 1A DG also provides emergency power to the water leg pump for the ‘B’ & ‘C’ RHR pumps. Bus 134 also supplies power to risk significant equipment for PRA but is incorrect because power for the waterleg pump comes from 135Y. D is incorrect: Plausible because the 1A DG provides emergency power to the water leg pump for the ‘B’ & ‘C’ RHR pumps and 136Y provides the normal power to the water leg pump for 'B' & 'C' RHR pumps. References: Electrical Schematic 1E-1-4000BP Rev F, 1E-1-4000AK Rev D, 1E-1-4000AM Rev E Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 2 Group: 1 KA: 203000 RHR: Injection Mode K2.01 Knowledge of electrical power supplies to the following: Pumps 10 CFR Part 55 Content: 41.7 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 064.00.05

29	ID: 2078933	Points: 1.00

Unit 1 was at 100% power when the following occurs:

- Drywell pressure is 2 psig.
- 2 minutes later, Off-site power is lost on Unit 1.

When the diesel generator output breakers close......

A. A and C RHR pumps start immediately, Low Pressure Core Spray and B RHR pumps start five seconds later.

B. Low Pressure Core Spray and B RHR pumps start immediately, A and C RHR pumps start five seconds later.

C. Low Pressure Core Spray and C RHR pumps start immediately, A and B RHR pumps start five seconds later.

D. A and B RHR pumps start immediately, Low Pressure Core Spray and C RHR pumps start five seconds later.

Answer: C

Answer Explanation
A is incorrect: Plausible because the 'C' RHR pump does close in immediately on the '1A' DG. Incorrect because the LPCS pump closes in on '0' DG. The 'A' RHR pump starts 5 seconds later not the LPCS Pump. B is incorrect: Plausible because the LPCS pump does close in immediately on the '0' DG. Incorrect because the 'C' RHR pump closes in on '1A' DG. The 'A' and 'B' RHR Pump start 5 seconds later not the 'A' and 'C' RHR Pump. C is correct: To prevent overloading the diesel Generators, the 'C' RHR pump closes onto the division 2 bus supplied by the '1A' Diesel generator, and the LPCS pump breaker closes onto the division 1 bus supplied by '0' DG. The 'A' and 'B' RHR pump breakers close five seconds later once the diesel is at rated speed. This load shedding occurs because a LOCA signal is present from High Drywell pressure signal. This ensures that the diesels will be able to get up to rated speed and provide power to the low pressure ECCS systems for injection. D is incorrect: Plausible because this is the reverse order of the logic sequence. Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 2 Group: 1 KA: 203000 RHR Injection Mode G2.1.27 Knowledge of system purpose and/or function. 10 CFR Part 55 Content: 41.7 SRO Justification: N/A Question Source: Bank Question History: 13-1 NRC Exam 1262346 Comments: Associated objective(s): 064.00.05

30	ID: 2061888	Points: 1.00

Unit 1 is in MODE 4:

- The running reactor recirc pump has just TRIPPED.
- Reactor Level is 20” on SHUTDOWN range.
- Reactor Coolant Temperature is 170°F.
- Shutdown cooling flow is 5000 GPM.

Under these conditions, thermal stratification:

A. Is a concern, and
RPV level must be raised to greater than 50” on NARROW range to make thermal stratification not a concern

B. Is a concern, and
SDC flow must be raised to greater than 6000 GPM to make thermal stratification not a concern

C. Is not a concern, and
RPV level on shutdown range may be lowered to 0 inches before thermal stratification is a concern

D. Is not a concern, and
SDC flow may be lowered to 4000 GPM before thermal stratification is a concern

Answer: B

Answer Explanation
A is incorrect: RPV level needs only to be raised to +50" on shutdown range not 50" on narrow range. Plausible because 50" is on scale for narrow range and the correct value is used just not the right instrumentation. B is correct: With no RR pumps running, SDC flow must be maintained greater than 6000 gpm or vessel level must be above +50" on Shutdown range. C is incorrect: RPV level must be above +50" on shutdown range. Plausible because +0" is the bottom of the scale of shutdown range. D is incorrect: SDC flow is not sufficient enough since RPV water level is <50" on shutdown range. Plausible because 4000 gpm is the bottom value called out for SDC flow for a Quick Recovery of SDC per LOA-RH-101. References: LOP-RH-20 rev 12, Shutdown Cooling System Operation Utilizing LPCI Flowpath Return, LOA-RH-101 rev21, Unit 1 RHR Abnormal Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 2 Group: 1 KA: 205000 Shutdown Cooling K1.14 Knowledge of physical connections and/or cause-effect relationships between Shutdown Cooling and the following: Reactor temperatures (moderator, vessel, flange) 10 CFR Part 55 Content: 41.9 / 45.7 / 45.8 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 064.00.20

31 **ID: 2061885** **Points: 1.00**

Unit 1 is in MODE 5.

'B' RHR pump is operating in Shutdown Cooling mode.

Considering BOTH loops of RHR, if the ‘B’ RHR pump had a shaft shear, what function of RHR is unavailable?

A. Alternate Shutdown Cooling

B. Fuel Pool Cooling Assist

C. Shutdown Cooling

D. Head Spray

Answer: B

Answer Explanation
A is incorrect: Alternate shutdown cooling is available on the 'A' RHR loop. Plausible because it is lost for the 'B' loop, and Alternate Shutdown cooling is not available to 'C' RHR. B is correct: Fuel Pool Cooling Assist is only available to the 'B' loop. Unlike Fuel Pool Emergency makeup, the 'B' RHR pump is the motive force for fuel pool cooling assist. C is incorrect: Shutdown cooling is available on the 'A' RHR loop. Plausible because it is lost for the 'B' loop, and Shutdown cooling is not available to 'C' RHR. D is incorrect: Head Spray is available on the 'A' RHR loop. Plausible because head spray is not available to 'C' RHR. References: M-96 Sheet 1,2,3 Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 2 Group: 1 KA: 205000 Shutdown Cooling K3.05 Knowledge of the effect that a loss or malfunction of Shutdown Cooling will have on the following: Fuel pool cooling assist: Plant-Specfic 10 CFR Part 55 Content: 41.7 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 064.00.14

32	ID: 2062110	Points: 1.00

What conditions are REQUIRED for the LPCS Injection Valve (1E21-F005) to Automatically Open?

1) ECCS Signal
2) Power to LPCS Pump
3) LPCS Pump Discharge Pressure
4) Pressure between the Injection Line check valve and Injection Valve is less than 500 psig
5) Reactor Pressure less than 500 psig

A. 1, 2, 3, & 5

B. 1, 2, 3, & 4

C. 1, 2, 4, & 5

D. 3, 4, & 5

Answer: C

Answer Explanation
A is incorrect: A low pressure signal from between the check valve and injection valve is required. Plausible because LPCS discharge pressure is required for an automatic ADS initiation signal which is an ECCS System. B is incorrect: Reactor pressure less than 500 psig is required. Plausible because LPCS would not be able to inject from 500 psig to 440 psig due to discharge head of the pump. C is correct: In order for the LPCS Injection Valve to Automatically open, an ECCS initiation signal to ensure the Reactor requires a ECCS injection, Power to the LPCS Pump to ensure that the Pump is available for Injection, and pressure between the injection valve and the check valve as well reactor pressure less than 500 psig to ensure the injection line pressure and reactor pressure have fallen to a value below the LPCS System's maximum design pressure. D is incorrect: An ECCS signal and power to the LPCS Pump are required. Plausible because LPCS discharge pressure is required for an automatic ADS initiation signal which is an ECCS System. Reference: 1E-1-4221AB, 1E-1-4221AD, Tech Spec Basis 3.3.5.1, LOR-1H13-P601-C210, Rev. 5 Reference provided during examination: none Cognitive level: Memory 10CFR Part 55: 41.7 Level (RO/SRO): RO Tier: 2 Group: 1 PRA: No K/A: 209001 K4.01 Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: Prevention of over pressurization of core spray piping. SRO Justification: N/A Question Source: New Question History: none Comments: Objective: 63.00.05

33	**ID: 2061951**	**Points: 1.00**

HPCS is injecting to the Unit 1 RPV.

- HPCS Initiation Logic 125 VDC Control Power fuse F1 blows
- Reactor water level reaches 45” on wide range and is rising at 5”/min

Assuming no operator action, after five minutes, the HPCS Pump will be ______________ and the HPCS Injection Valve 1E22-F004 will be _______________.

A. Running; Closed

B. Running; Open

C. Not Running; Closed

D. Not Running; Open

Answer: B

Answer Explanation
A is incorrect: is plausible since the level 8 trip would close the HPCS Injection valve if F1 were not blown. B is correct: F1 powers the initiation and level 8 trip logic for HPCS. F1 also powers control power for bus 143. If HPCS is injecting, the pump is currently running. Bus 143 will remain closed when control power is lost and the HPCS pump will continue to run. Since F1 has blown, a trip of the HPCS pump will not occur. C is incorrect: plausible because the level 8 trip would close the HPCS injection valve if F1 were not blown. D is incorrect: plausible because the control power to the HPCS pump is lost. References: 1E-1-4222AB, 1E-1-4222AD Reference provided during examination: 1E-1-4222AB, 1E-1-4222AD Cognitive level: High Level (RO/SRO): RO Tier: 2 Group: 1 KA: 209002 HPCS K2.03 Knowledge of electrical power supplies to the following: Initiation logic 10 CFR Part 55 Content: 41.5 SRO Justification: N/A Question Source: New 2017-1 ILT NRC RO Exam Question History: N/A Comments: Associated objective(s): 061.00.08

34	ID: 2062109	Points: 1.00

HPCS is operating in full flow test for a surveillance.

HPCS indicates 2200 GPM on the 1H13-P601.

The HPCS Injection Valve breaker is OOS for maintenance.

A LOCA brings in a high drywell pressure signal.

What is the status of the HPCS MIN FLOW VLV, 1E22-F012, PRIOR to the LOCA signal?

What is the status of the HPCS MIN FLOW VLV, 1E22-F012, ONE MINUTE AFTER the LOCA signal?

	BEFORE	AFTER
A.	open	open
B.	open	closed
C.	closed	open
D.	closed	closed

Answer: C

Answer Explanation
A is incorrect: The min flow valve would close at the reset point 2047 gpm. 2200 gpm is still above the maximum LCO value for min flow valve reset. Plausible because this flow value is near the setpoint. Also plausible because the full flow test valve goes closed on manual initiation with the pushbutton or automatic initiation which would open the min flow valve when the low flow setpoint is reached and the HPCS pump running. B is incorrect: The min flow valve would close at the reset point 2047 gpm. 2200 gpm is still above the maximum LCO value for min flow valve reset. Plausible because this flow value is near the setpoint. The full flow test valve goes closed on manual initiation with the pushbutton or automatic initiation which would open the min flow valve when the low flow setpoint is reached and the HPCS pump running. C is correct: When a HPCS initiation signal is received, a signal is sent to start the HPCS pump, open the injection valve, and close the full flow test valve. HPCS system flow rate. Initiates opening of HPCS Pump Min Flow Isol Valve, 1E22-F012, if flow drops below setpoint and pressure sensed by 1E22-N012A is greater than setpoint. Initiates closing of 1E22-F012, if flow rises to reset point. Switch calibrated to actual setpoint of 26.5 inWC (1694 gpm) and reset point of < 38.7 inWC (2047 gpm) to allow for calibration and instrument inaccuracies and a maximum reset value of 44.5 in WC (2194 gpm) D is incorrect: Plausible because the min flow valve is initially closed. Plausible because the full flow test valve goes closed and the injection valve opens automatically on system initiation. This is the correct answer if the HPCS pump breaker was not racked out for maintenance. References: LIS-HP-105 rev32, Unit 1 High Pressure Core Spray Minimum Flow Bypass Calibration Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 2 Group: 1 KA: 209002 HPCS A1.08 Ability to predict and/or monitor changes in parameters associated with operating the HPCS controls including: System lineup 10 CFR Part 55 Content: 41.5 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 061.00.05

35	ID: 2062091	Points: 1.00

The plant is operating at 100% RTP.

- Instrument air was inadvertently isolated to the Standby Liquid Control (SBLC) system.
- Reactor Building temperature is 84°F and steady.
- No other evolutions are being performed on the SBLC system.

Which of the following Control Room Annunciators, if any, initially alert the Control Room of this isolation?

A. No Control Room Alarms will annunciate.

B. LOR-1H13-P603-A205 SBLC TANK LVL HI/LO for LOW level.

C. LOR-1H13-P603-A205 SBLC TANK LVL HI/LO for HIGH level.

D. LOR-1H13-P603-B502 Standby Liquid Tank Temp HI/LO for low temperature.

Answer: B

Answer Explanation

A is incorrect: plausible because other systems use a D/P switch using a reference and variable leg for level indication. A loss of instrument air would not affect level and thus no alarms will annunciate.

B is correct: The SBLC level instrumentation requires instrument air for operation. The level instrument fails low on a loss of Instrument Air.

C is incorrect: plausible because this assumes level will go high on a loss of instrument air.

D is incorrect: plausible because the low SBLC tank level will trip the heater, but with reactor building temperatures at 84°F SBLC tank temperature will not reach the alarm setpoint

Reference:
LOR-1H13-P603-A205 SBLC TANK LVL HI/LO Rev 4
LOR-1H13-P603-B502 Standby Liquid Tank Temp HI/LO Rev 3
Reference provided during examination: None

Cognitive level: High
10CFR Part 55:41(b)(7)

Level (RO/SRO): 2.5/2.6
Tier: 2 Group: 1

PRA: Yes
K/A: 211000 K1.03 Knowledge of the physical connections and/or cause/effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Plant air systems: Plant-Specific

SRO Justification: N/A

Question Source: Modified from question on Vermont Yankee 2009 NRC Exam

Question History: N/A

Objective 028.00.16

Comment: Copy of Vermont Yankee Question 2009

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	211000, K1.03	
Importance Rating	2.5	

(K&A Statement) Knowledge of the physical connections and/or cause- effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Plant air systems: Plant-Specific

Proposed Question: Common 1

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

- Instrument air was inadvertently isolated to the Standby Liquid Control (SLC) system
  Reactor Building temperature is 84°F and steady
- No other evolutions are being performed on the SLC system

Which of the following Control Room Annunciators initially alert the Control Room of this isolation?

- 9-5 A-1, SLC SQUIB VLV CONTINUITY LOSS
- 9-5 A-3, SLC TANK LVL HI/LO
- 9-5 A-4, SLC TANK/SUCT LINE TEMP HI/LO

Annunciator(s):

A. 9-5-A-1 only

B. 9-5-A-3 only

C. 9-5-A-1 and 9-5-A-4

D. 9-5-A-3 and 9-5-A-4

Proposed Answer: B.

VY 09

211000 K1.03

36	ID: 2059171	Points: 1.00

A 20% RTP ATWS occurs on Unit 1 and SBLC is manually initiated.

Which of the following is confirmation that the 1C41-F004A, Standby Liquid Control Squib Valve, is open AND SBLC is injecting into the RPV?

A. Reactor Water Cleanup is isolated

B. Discharge Pressure of the SBLC pump is 985 psig

C. Discharge Pressure of the SBLC pump is 1340 psig

D. SBLC INJ SQUIB VLV 1C41-F004A Continuity Light Lit

Answer: B

Answer Explanation

A is incorrect: The RT-004 valve goes closed when the keylock switch the 603 is taken to SYS A or SYS B. It does not prove that SBLC is injecting.

B is correct: The SBLC pump is pumping at 985 psig which is what reactor pressure is at for rated conditions since the unit is still at power. This shows that there is a flow path through the exploded squib valve and SBLC is injecting to the vessel.

C is incorrect: This is the operating pressure of the SBLC relief valve – This indicates that the water is being sent to the pump suction rather than to the vessel. Plausible because the SBLC pump is running which is tied to the same logic as firing the SQUIB valve.

D is incorrect: This light should go out after the squib valve fires. Plausible because this light means that the SBLC continuity circuit is made up and will fire.

References: DCP S01-1-94-031 1340# relief setpoint for 1C41-F029A, LOR-1H13-P603-A105 rev 4, SBLC SQUIB VLV CONTINUITY LOSS
Reference provided during examination: None
Cognitive level: High
Level (RO/SRO): RO
Tier: 2 Group: 1
KA: 211000 SBLC K5.04 Knowledge of operational implications of the following concepts as they apply to SLC SYSTEM: Explosive valve operation
10 CFR Part 55 Content: 41.5, 45.3
SRO Justification: N/A
Question Source: Modified
Question History: Fitzpatrick 2010

Comments:

QUESTION 32.

The plant was operating at 100% power when MSIV

- Reactor Power is 15%.
- SRVs are cycling to control reactor pressure.
- 'A' SLC pump has been manually started from

Which one of the following describes indications that open?

A. White SQUIB VLVS READY light is ON.

B. Discharge pressure of the SLC pump is 1150

C. White SQUIB VLVS READY light is OFF.

D. Discharge pressure of the SLC pump is 1400

Associated objective(s): 29.00.05

37	ID: 2062063	Points: 1.00

Unit 1 is at 100% RTP with LIS-NB-105A, UNIT 1 REACTOR HIGH PRESSURE SCRAM CHANNELS A AND C CALIBRATION, in progress.

IMD has discovered the High Pressure Scram Channel 1B21-N023AA will not trip.

1) What effect will this condition have on RPS if RPV pressure reaches a RPS scram setpoint?
2) What is the status of LCO 3.3.1.1 RPS Instrumentation?

A. 1) RPS Channels A2 and (B1 or B2) will actuate, causing a FULL reactor scram.
2) LCO 3.3.1.1 is met

B. 1) RPS Channels A2 and (B1 or B2) will actuate, causing a FULL reactor scram.
2) LCO 3.3.1.1 is NOT met.

C. 1) The “A” RPS trip system will NOT actuate, preventing a scram from high pressure.
2) LCO 3.3.1.1 is NOT met.

D. 1) The “A” RPS trip system will NOT actuate, preventing a scram from high pressure.
2) LCO 3.3.1.1 is NOT applicable.

Answer: B

Answer Explanation
A is incorrect: plausible because RPS will actuated and some protective actuation logic are (A and B) channels, or (C and D) channels. B is correct: The reactor protection system (RPS) is divided into two trip systems, A and B, which are physically and electrically independent. The design of his system is such that the loss of power to one of these trip systems neither prevents nor causes a reactor scram. Normal power to RPS buses And B are supplied by two motor-generator sets. Alternate power for either RPS bus is from the Alternate Instrument and RPS bus transformer, fed from bus 132B-1 (UNIT 1) or 232B-1 (Unit 2). Each trip system is further divided into two RPS logic Trip Channels A1 & A2 or B1 & B2. With one channel of the RPS high pressure scram function Inoperable LCO 3,.3.1.1 is applicable and considered NOT met. C is incorrect: plausible some actuation logic requires both channels to actuate. D is incorrect: plausible some actuation logic requires both channels to actuate Reference: LIS-NB-105A UNIT 1 REACTOR HIGH PRESSURE SCRAM CHANNELS A AND C CALIBRATION Rev 18, TS 3.3.1.1 Basis, UFSAR RPS Design Basis 7.2.1.1 G.7 page 7.2-2 Reference provided during examination: Tech Spec Table 3.3.1.1 RPS Instrumentation Cognitive level: High 10CFR Part 55:41(b)(7) Level (RO/SRO): 3.5/3.7 Tier: 2 Group: 1 PRA: No K/A: 212000 K6.03 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM: Nuclear boiler instrumentation SRO Justification: N/A Question Source: New Question History: N/A Comments: This questions requires the examinee to determine whether an LCO statement is met or not which is above the line information and RO knowledge.

38	ID: 2059105	Points: 1.00

A reactor startup is in progress.

- IRM A reads 25 on Range 2.
- Reactor mode switch is in STARTUP.

While selecting SRM detectors for withdrawal to maintain SRM Count rate between
1 X 10e3 to 1 X 10e5 CPS, IRM A is also selected. The drive out pushbutton is then depressed and held depressed.

The "A" IRM detector not full in limit switch will open causing...

A. a ½ scram only.

B. a rod block only.

C. no protective actions.

D. a rod block and a ½ scram.

Answer: B

Answer Explanation
A is incorrect: no ½ SCRAM occurs. There is a 1/2 scram trip on Hi-Hi level. Withdrawing the IRM detector may also cause a downscale trip which causes a rod block not a 1/2 scram. B is correct: A rod block will be generated when the IRM not full in limit switch is no longer made up since the mode switch is not in RUN and the IRM in question is not bypassed. C is incorrect: a rod block occurs. Plausible because the mode switch in RUN would allow the IRM to be withdrawn without a rod block being generated. Also if the IRM were bypassed no protective action would occur. D is incorrect: no ½ SCRAM occurs. Plausible because a 1/2 scram occurs for an INOP trip. Plausible because the IRM generates both rod block and 1/2 scram trips. There is a 1/2 scram trip on Hi-Hi level. Withdrawing the IRM detector may also cause a downscale trip which causes a rod block. References: LOS-NR-W2 rev 13, IRM DETECTOR NOT FULL IN ROD BLOCK FUNCTIONAL TEST Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 2 Group: 1 KA: 215003 Intermediate-Range Monitor K5.01 Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGER MONTIRO (IRM) SYSTEM: Detector Operation 10 CFR Part 55 Content: 41.5, 45.3 SRO Justification: N/A Question Source: Bank Question History: Susquehanna Comments: Associated objective(s): 042.00.05

39	ID: 2059120	Points: 1.00

The mode switch is in ‘STARTUP’.

IRMs are all on range 5 or 6.

During performance of a surveillance for the mode switch, SRM ‘A’ is moved out of ‘OPERATE’ and placed in the ‘STANDBY’ position.

Which of the following alarm response procedures must be entered?

A. LOR-1H13-P603-A308, ROD OUT BLOCK &
LOR-1H13-P603-A106, SRM INOPERATIVE OR HI

B. LOR-1H13-P603-A308, ROD OUT BLOCK &
LOR-1H13-P603-A206, SRM DOWNSCALE

C. LOR-1H13-P603-B203, CHANNEL A1 REACTOR AUTO SCRAM &
LOR-1H13-P603-A206, SRM DOWNSCALE

D. LOR-1H13-P603-B203, CHANNEL A1 REACTOR AUTO SCRAM &
LOR-1H13-P603-A106, SRM INOPERATIVE OR HI

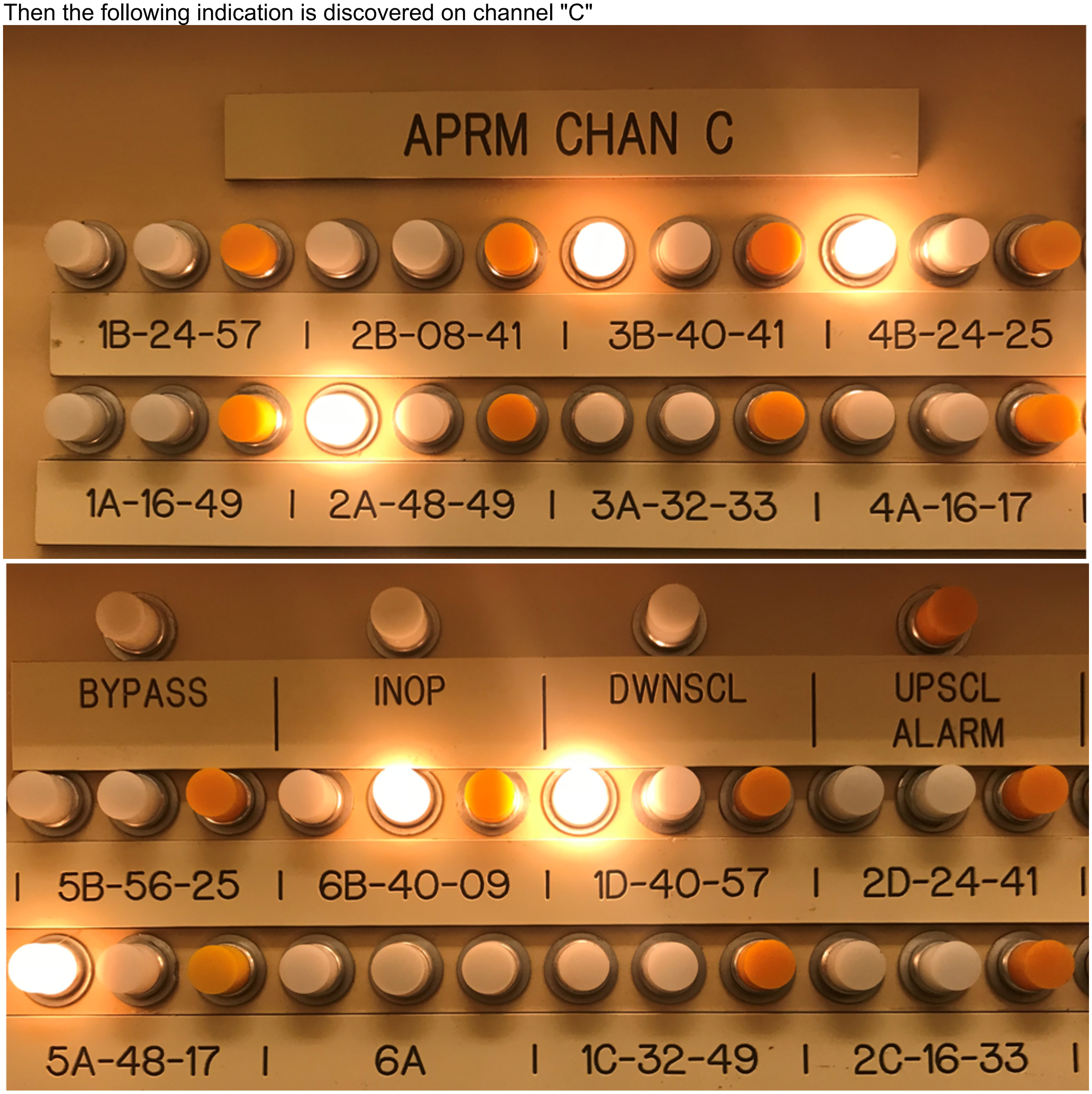
Answer: A

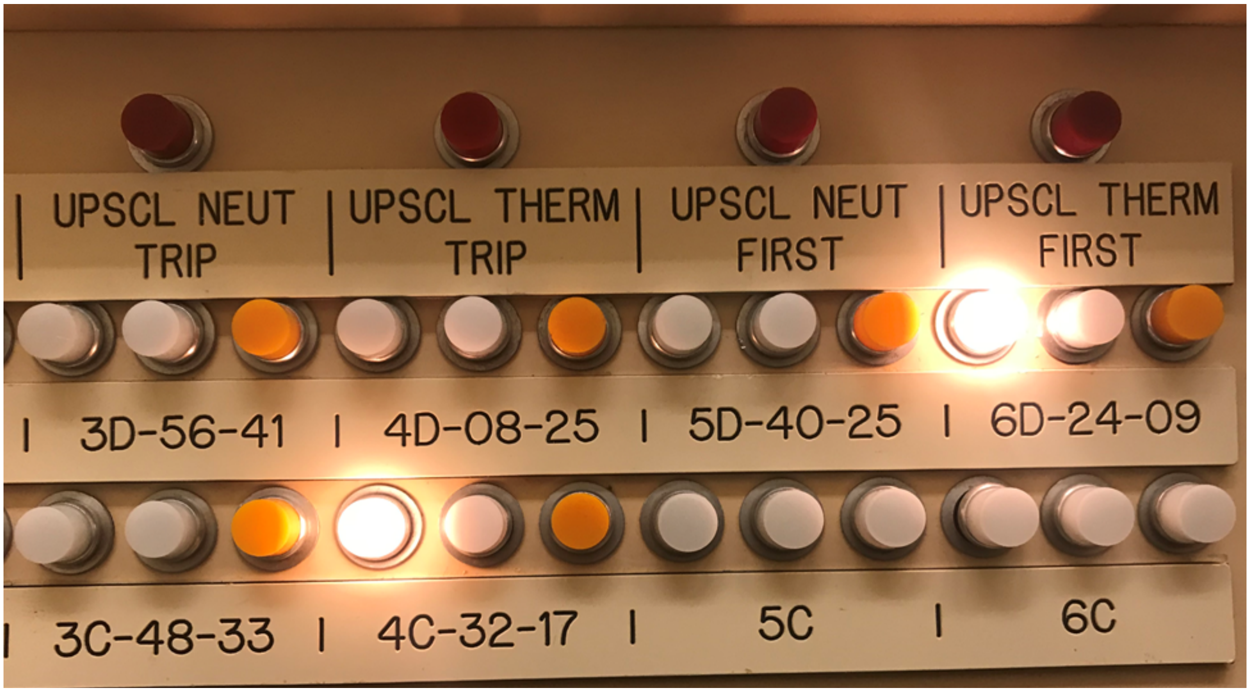
Answer Explanation
A is correct: Per LOP-NR-02: A SRM Inop rod block will be initiated on low voltage, module unplugged, or Function Switch NOT in OPERATE. B is incorrect: SRM Inop will come in. Plausible because there is a rod block is generated C is incorrect: An SRM HI-HI signal would cause a half SCRAM. Plausible because a downscale detector below range 3 issues a rod block as a protective action. D is incorrect: An SRM HI-HI signal would cause a half SCRAM. Plausible because the SRM INOPERATIVE OR HI annunciator would be in. References: LOR-1H13-P603-A106 rev 3, SRM INOPERATIVE OR HI, LOR-1H13-P603-B203 rev2, CHANNEL A1 REACTOR AUTO SCRAM, LOR-1H13-P603-B211 rev2, CHANNEL A2 REACTOR AUTO SCRAM, LOR-1H13-P603-A308 rev5, ROD OUT BLOCK, LOR-1H13-P603-A206 rev3, SRM DOWNSCALE, LOP-NR-01 rev15, SRM OPERATION Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 2 Group: 1 KA: 215004 Source-Range Monitors A2.02: Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM; and based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: SRM inop condition. 10 CFR Part 55 Content: 41.5, 45.6 SRO Justification: N/A Question Source: Bank Question History: Columbia 2009 Comments: Associated objective(s): 041.00.05

40 **ID: 2086093** **Points: 1.00**

Unit 1 is at 100% RTP.

Then the following indication is discovered on channel "C"





What effect will this have if the LPRM 6B-40-09 is placed into the bypass position on Unit 1?

A. APRM rod block only.

B. 1/2 scram on A RPS, and rod block.

C. 1/2 scram on B RPS, and rod block.

D. The Rod Block Monitor is generating a rod block.

Answer: B

Answer Explanation

A is incorrect: Plausible because there is an APRM rod block generated with the mode switch in run.

B is correct: With the downscale failure of 40-09B LPRM, the total number of operable LPRMs providing input to APRM Channel C is 13, but until bypassed the APRM will still function as normal. The Rod Block Monitoring system uses the LPRM signal level to determine the available inputs for the rod selected When the downscale LPRM is bypassed the automatic APRM INOP trip will occur with less than 14 LPRM inputs causing a 1/2 SCRAM on A RPS and a Rod Block.

C is incorrect: Plausible because this would cause a 1/2 scram on 'B' RPS and a rod block if the APRM was associated with 'B' RPS.

D is incorrect: Plausible because the rod block monitor can enforce a rod block on too few inputs, but that occurs when there are less than 50% of LPRM inputs available to its associated LPRM.

Reference: LOA-NR-101 NEUTRON MONITORING TROUBLE., Revision 20 , Section B.2, Step 4.3 and Attachment B
LOR-1H13-P603-A407 Local Power Range Monitor Downscale Rev 2, TS Bases 3.3.1.1, function 2a, page 3.3.1.1-8, Revision 0

Reference provided during examination: None

Cognitive level:High
10CFR Part 55:41.(b)(7)

Importance (RO/SRO):
Tier: 2 Group: 1

PRA: No
K/A: 215005 K4.01 Average Power Range Monitor (APRM) / Local Power Range Monitor: knowledge of ARPM / LPRM design feature(s) and/or interlocks which provide for the following: Rod withdrawal blocks.

SRO Justification: N/A

Question Source: New

Question History:

Comments:
References: LOP-NR-05 rev 14, Rod Block Monitor Operation (RBM), LOP-NR-08 rev 5, Bypassing LPRMs, LOR-1H13-P603-A208 rev3, APRM Downscale
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO):3.7/3.7
Tier: 2 Group: 1

KA: 215005 K4.01 Average Power Range Monitor (APRM) / Local Power Range Monitor: knowledge of ARPM / LPRM design feature(s) and/or interlocks which provide for the following: Rod withdrawal blocks.

10 CFR Part 55 Content: 41.7

Objective: 44.00.21

41 **ID: 2061965** **Points: 1.00**

What is the trip setpoint for maximum disagreement between APRM flow comparator channels, and what signal is generated when it is reached?

A. 10%
Half scram

B. 10%
Rod withdrawal block

C. 5%
Half scram

D. 5%
Rod withdrawal block

Answer: B

Answer Explanation
A is incorrect: Plausible because this is the trip setpoint and APRMs do issue half scram signals. B is correct: At a 10% difference in output flow signals between flow comparator channels, a rod withdrawal block is generated. C is incorrect: Plausible because APRMs due have a 5% trip for downscale while the mode switch is in run. This is for APRM power, not flow comparator disagreement. D is incorrect: Plausible because APRMs due have a 5% trip for downscale while the mode switch is in run. This is for APRM power, not flow comparator disagreement. References: LOR-1H13-P603-A209 rev3, APRM Flow Bias Off Norm Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 2 Group: 1 KA: 215005 A3.06 Average Power Range Monitor (APRM) Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR SYSTEM including: Maximum disagreement between flow comparator channels. 10 CFR Part 55 Content: 41.7 / 45.7 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 044.00.05

42	**ID: 2058925**	**Points: 1.00**

Unit 1 is at 100% RTP:

- The Reactor Core Isolation Cooling (RCIC) turbine was started manually per LOS-RI-Q4 COLD QUICK START IN MODES 1, 2, AND 3
- While raising turbine speed, a local operator reports steam leakage near the RCIC governor valve

How will the RCIC system respond, if at all, to the operator depressing the RCIC isolation pushbutton on the 1H13-P601 panel?

A. Neither of the RCIC system isolation valves will close.

B. ONLY the inboard isolation valve 1E51-F063 will close.

C. ONLY the outboard isolation valve 1E51-F008 will close.

D. The inboard and the outboard RCIC isolation valves 1E51-F063 and 1E51-F008 will close.

Answer: A

Answer Explanation
A is correct: This pushbutton only works if there is an AUTOMATIC RCIC system initiation signal present (the red initiation signal present indicating light lit). This light is lit when an initiation signal is present, either automatic (-50 inches) or Manual (manual pushbutton). Per Schematic 1E-1-4226AD The pushbutton is shown on this print and is interlocked with the K15 relay, but to get the K15 relay to energize, the K5 contact needs to be in the closed state. Looking at schematic 1E-1-4226AC the K5 relay is energized by an initiation signal. Depressing the pushbutton and with an initiation signal this will energize the K15 relay. The K15 relay is interlocked with the closing coil on the 1E51-F008 (1E-1-4226AN) Per Schematic 1E-1-4226AC with an initiation signal present the K5 relay will energize and associated contacts 11/12 will close. Looking at the K15 relay it is interlocked with the 1E51-F008 valve. Per 1E-1-4226AN Once the K15 relay is energized, associated contact 3/4 will close and which causes the valve to go closed. B is incorrect: Plausible because 1E51-F063 is an isolation valve for this system but is the wrong isolation valve and an initiation signal is required. C is incorrect: Plausible because it is the correct isolation valve but with no initiation signal the valve will not close. D is incorrect: Plausible because both are isolation valves for the system and both valves close on a valid PCIS actuation signal. Reference: 1E-1-4226AC/AD/AN Reference provided during examination: None Cognitive level: High Level: RO 10CFR Part 55 Content: (CFR: 41.7, 45.5-45.8) Tier: 2 Group: 1 K/A: 217000 REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) A4.04 Ability to manually operate and/or monitor in the control room: Manually initiated controls SRO Justification: N/A Question Source: Bank Question History: N/A Comments: Associated objective(s): 091.00.05

43 **ID: 2062144** **Points: 1.00**

Which of the following conditions require the ADS Inhibit Switches (S26A/S26B) to be placed in 'INHIBIT'?

1) A spurious ADS initiation occurs during a non ATWS scram IAW LGA-001, RPV CONTROL
2) An ATWS is occurring with RPV water level being maintained -60” to -100” IAW LGA-010, FAILURE TO SCRAM
3) A fire in the Main Control Room IAW LOA-FX-101, UNIT 1 SAFE SHUTDOWN WITH A FIRE IN THE CONTROL ROOM

A. 1 ONLY

B. 2 & 3

C. 3 ONLY

D. 1 & 2

Answer: D

Answer Explanation
A is incorrect: 2 is also a condition in which the inhibit switches should be repositioned. Plausible because 1 is a correct choice and directed per LGA-001. B is incorrect: 1 is also a condition in which the inhibit switches should be repositioned. Plausible because LGA-010 requires inhibiting ADS and ECCS prior to proceeding to the individual legs of the LGA. C is incorrect: 1 & 2 are conditions in which the inhibit switches should be positioned to inhibit. Plausible because the ADS defeat switches do prevent a manual initiation of ADS and IAW LOA-FX-101, during fire in the main control room the operator places the ADS DEFEAT switches to defeat. D is correct: LGA-001 has an override in the level leg that states if ADS actuates spuriously to reset and inhibit ADS. LGA-010 requires inhibiting ADS and ECCS prior to proceeding to the individual legs of the LGA these switches prevent an automatic ADS from occurring. References: LGA-001 rev 18, RPV Control, LGA-010 rev 18 Failure to Scram, LOA-FX-101 rev 30 Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 2 Group: 1 KA: 218000 ADS 2.4.6 Knowledge of EOP mitigation strategies. 10 CFR Part 55 Content: 41.10 / 43.5 / 45.13 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 062.00.06

44	ID: 2061890	Points: 1.00

Unit 1 was operating at 100% RTP when a reactor level transient occurred:

- HPCS and RCIC automatically initiated
- LGA-001 was entered and RPV level is now being maintained between +20" and +50"
- Drywell pressure is 0.1 psig

You have been directed to reset Group 2 isolations per LOA-PC-101 and have depressed the INBOARD and OUTBOARD isolation reset pushbuttons on the 1H13-P601 panel.

Which of the following systems requires additional logic reset before the affected valves can be opened?

A. Drywell Floor Drains

B. Containment Monitoring

C. Primary Containment Cooling

D. Reactor Building Closed Cooling Water

Answer: A

Answer Explanation
A is correct: Since HP and RI initiated, Level 2 PCIS actuations would also occur. Per LOA-PC-101 Att. C at panel 1PM16J, the Div 1 and 2 RE/RF Sys Cont Isol Reset PBs must also be depressed to allow DWFDS valves to be opened. All other distracters contain systems that need no additional reset. B is incorrect: Plausible because Containment Monitoring requires valves to be opened and verification checks per LOP-CM-01, but no logic resets. C is incorrect: Plausible because Primary Containment Cooling requires valves to be opened and the system restarted per LOP-VP-02, but no logic resets. D is incorrect: Plausible because WR has valves to supply DW equipment with cooling water that are required to be opened, but no logic resets. References: LOA-PC-101 rev 21, Primary/Secondary Containment Trouble Reference provided during examination: none Cognitive level: High Level (RO/SRO): RO Tier: 2 Group: 1 KA: 223002 PCIS Knowledge of the physical connections and/or cause-effect relationships between PCIS and the following: Reactor building drainage system: Plant-Specific 10 CFR Part 55 Content: 42.1-41.9 / 45.7 - 45.8 SRO Justification: N/A Question Source: Bank Question History: N/A Comments: Associated objective(s): 091.00.14

45	ID: 2058899	Points: 1.00

What is the power supply to the SRV solenoids that are operated by the 13 hand switches in the MCR?

A. 111Y

B. 112Y

C. 136X-2

D. 135Y-1

Answer: A

Answer Explanation
A is correct: 111Y powers the 'C' solenoid for the SRVs. The 'C' solenoid provides normal relief function and also power for the 13 hand switches in the MCR. B is incorrect: Plausible because this powers the 'B' SRV solenoids which can be energized from the MCR. C is incorrect: Plausible because this powers the solenoids for the outboard main steam line drains from the MCR. D is incorrect: Plausible because this powers the solenoids for the inboard MSIVs. References: 1E-1-4201AA rev M, 1E-1-4201AC rev Z Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 2 Group: 1 KA: 239002 K2.01 Knowledge of electrical power supplies to the following: SRV solenoids 10 CFR Part 55 Content: 41.7 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 70.00.16

46	ID: 2059073	Points: 1.00

Unit 1 is at 100% RTP:

- A TDRFP tripped offline
- The MDRFP failed to auto start

Assuming no operator action…

A Reactor Recirculation runback will occur _____________.

A. immediately and terminate once steam flow is less than 62% ONLY

B. immediately and terminate once reactor water level rises above 31.5" ONLY

C. once reactor water level drops below 20” and terminate when the FCV reaches its minimum position

D. once reactor water level drops below 31.5” and terminate when steam flow is less than 62% or level exceeds 31.5”

Answer: D

Answer Explanation
A is incorrect: Plausible because this will terminate the runback. Incorrect because the runback is tied with an 'AND' logic with feed flow steam flow mismatch which will also terminate the Runback. B is incorrect: Plausible because this will terminate the runback. Incorrect because the runback is tied with an 'AND' logic with reactor water Level 4 which will also terminate the Runback when level is higher than the level 4 setpoint (31.5"). C is incorrect: Plausible because the RR FCV has a minimum position and would terminate the RR FCV runback. However, reactor water level would recover prior reaching the minimum position. 20" is the setpoint for setpoint setdown and terminating the post scram profile. D is correct: Since the MDRFP will not auto start, RWLC only sees 67% as available feed flow. When reactor water level drops below 31.5" a Reactor Recirculation FCV runback will occur until level is > L4 or steam flow is less than 62%. In this instance level will recover above 31.5" before the FCV reaches the minimum position. References: LOP-RL-01 rev 26, Operation of the Reactor Level Control System LOP-FW-16, 1(2) DS001 OPERATOR STATION ALARM MESSAGE INTERPRETATION Reference provided during examination: None Cognitive level: High (Candidate has to bring memory level info but also must assess the plant conditions to determine the FCV will reach a runback termination criteria prior to the FCV reaching the minimum position) Level (RO/SRO): RO Tier: 2 Group: 1 KA: 259002 K3.05 Knowledge of the effect that a loss or malfunction of the RWLCS will have on following: Recirculation flow control system 10 CFR Part 55 Content: (41.7, 45.5, 45.8) SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 031.00.18

47	ID: 2058919	Points: 1.00

The Standby Gas Treatment Cooling Fan starts ____(1)_____,
and the charcoal bed is designed to capture radioactive _____(2)_____.

A. (1) on a Standby Gas Treatment Primary Fan stop signal
(2) nitrogen

B. (1) on a Standby Gas Treatment Primary Fan stop signal
(2) iodine

C. (1) when a low flow condition is sensed
(2) nitrogen

D. (1) when a low flow condition is sensed
(2) iodine

Answer: B

Answer Explanation
A is incorrect: The SGTS charcoal adsorber is designed to remove 99% of radioactive and nonradioactive forms of iodine. Plausible because nitrogen is also a radioactive gas that is created by the reactor. B is correct: UFSAR 6.5.1.1.J: The standby cooling air fan is conservatively sized to remove approximately 7700 Btu/hr of heat (generated by instantaneous deposition of iodine, on a HEPA filter bank and charcoal adsorbers) with less than a 50° F rise in cooling air temperature. This will limit the air temperature in the SGTS to 200° F maximum to prevent possible desorption and fire. Per LOP-VG-02 when the standby gas treatment fan is stopped the next step in the procedure is to verify the cooling fan automatically starts. Nitrogen and Iodine are both radioactive gasses produced by the reactor. 6.5.1.2.1.7: A charcoal adsorber capable of removing not less than 99% of radioactive and nonradioactive forms of iodine. C is incorrect: Plausible because the electric heater automatically turns off when a low flow condition is sensed, and nitrogen is also a radioactive gas that is created by the reactor. D is incorrect: Plausible because the electric heater automatically turns off when a low flow condition is sensed, and iodine is the radioactive gas that charcoal bed is holding to allow it to decay. References: UFSAR 6.5.1.1.1.j, 6.5.1.2.7, LOP-VG-02 rev 17 Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 2 Group: 1 KA: 261000 K4.05 Knowledge of SGTS design feature(s) and/or interlocks which provide for the following: Fission product gas removal 10 CFR Part 55 Content: 41.7 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 95.00.05

48 **ID: 2059236** **Points: 1.00**

With both Division 1 unit-tie breakers (1414 & 2414) open, which of the following breakers must be closed to allow closing breaker 1414 from the Main Control Room?

A. SAT Feed to 141Y (1412)

B. SAT Feed to 241Y (2412)

C. Common DG Feed to 141Y (1413)

D. 141Y to 141X Bus Tie Breaker (1415)

Answer: A

Answer Explanation
A is correct: In order to close the first unit-tie breaker, that unit's SAT supply breaker must be closed. B is incorrect: ACB 1414 does not tie into the unit tie closure logic for ACB 2412. Plausible because this would be true for the other unit if ACB 2414 was being closed. C is incorrect: ACB 1414 does not tie into the unit tie closure logic for ACB 1413. Plausible because this would make the 141Y energized from the diesel generator, which is also what the SAT feed breaker being closed provides. D is incorrect: ACB 1414 does not tie into the unit tie closure logic for ACB 1415. Plausible because ACB 1415 must be open to prevent supplying the opposite units non-safety busses with power from the unit tie. Reference: 1E-1-4005AK Reference provided during examination: none Cognitive level: Memory 10CFR Part 55: CFR: 41.5 / 45.3   Level (RO/SRO): 2.6 / 2.9 Tier: 2 Group: 1 PRA: No K/A: 262001 K5.02 Knowledge of the operational implications of the following concepts as they apply to AC Electrical Distribution: Breaker control SRO Justification: N/A   Question Source: Bank Question History: N/A Comments: Objective 005.00.05

49	ID: 2061947	Points: 1.00

Unit 1 is at 100% RTP.

- The Unit 1 PPC UPS inverter has failed.

What is currently powering the Unit 1 PPC UPS loads?

A. 135X

B. 235X

C. 121Y

D. 221Y

Answer: B

Answer Explanation
A is incorrect: Plausible because this is the alternate power supply for the Unit 2 PPC UPS loads. B is correct: On a loss of the Inverter, the loads will be powered by 235X. Cis incorrect: Plausible because this is a power source for the U1 PPC UPS. Incorrect because with a failed inverter this power supply is not available to power the loads. D is incorrect: Plausible because this is a power source for the U2 PPC UPS also the plausible because the current power source is being supplied from U2. References: LOR-1PM01J-A111 UPS Trouble rev 5, LOP-CX-08UNINTERRUPTIBLE POWER SUPPLY STARTUP, OPERATION AND SHUTDOWN, 1E-1-4000FA, 1E-1-4007AA Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 2 Group: 1 KA: 262002 UPS (AC/DC) K6.03 Knowledge of the effect that a loss or malfunction of the following will have on the UPS (AC/DC) Static Inverter 10 CFR Part 55 Content: 41.7 / 45.7 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 012.00.05

50	ID: 2059054	Points: 1.00

The plant is operating at 100% RTP.

A malfunction causes the 1A 24/48 VDC battery charger output voltage to change from the normal value to 22 VDC over one minute.

Which of the following lists the status of 24/48 VDC battery "1A” after the voltage change and the plant load that is affected by the change in voltage?

A.	Charging	Source Range Monitors A & C
B.	Discharging	Source Range Monitors A & C
C.	Charging	Local Power Range Monitors
D.	Discharging	Local Power Range Monitors

Answer: B

Answer Explanation

A is incorrect: The battery charger voltage has dropped below battery voltage. Plausible because A&C SRMs are powered from 24/48 VDC system.

B is correct: The candidates have to understand what is the value of terminal volts in relation to charger voltage. When the battery charger output drops below the batteries' voltage, the battery system will begin to discharge. The SRM A & C are powered by the 24/48 VDC system. Local Power Range Monitors are powered from RPS system.

C is incorrect: The battery charger voltage has dropped below battery voltage. Plausible because Local Power Range Monitors are a reactor power monitoring system.

D is incorrect: Plausible because the battery would be discharging, also Local Power Range Monitors are a reactor power monitoring system.

References: LOA-DC-101 Rev 23, Unit 1 DC Power System Failure, 1E-1-4000GA Key Diagram

Reference provided during examination: None
Cognitive level: High
Level (RO/SRO): RO
Tier: 2 Group: 1
KA: 263000 DC Electrical Distribution A1.01 Ability to predict and/or monitor changes in parameters associated with operating the DC Electrical Distribution controls, including: Battery charging/discharge rate.

10 CFR Part 55 Content: 41.5, 45.5
SRO Justification: N/A
Question Source: Bank
Question History: Nine Mile Point 2008 ILT NRC exam
Associated objective(s): 006.00.05
Comments: Nine Mile 2008 NRC Exam

Question Worksheet

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	263000, A1.01	
Importance Rating	2.5	

(K&A Statement) Ability to predict and/or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including: Battery charging/discharging rate

Proposed Question: Common 13

The plant is operating at 100% power when a malfunction causes 24 VDC battery charger 121 output voltage to change from the normal value to 22 VDC over one (1) minute.

Which one of the following lists the status of 24 VDC battery 121 after the voltage change and a plant load that is affected by the change in voltage?

	Battery 121 Status After Malfunction	Affected Plant Load
A.	Charging	Source Range Monitors (SRMs)
B.	Discharging	Source Range Monitors (SRMs)
C.	Charging	Electromatic Relief Valves (ERVs)
D.	Discharging	Electromatic Relief Valves (ERVs)

Proposed Answer: B.

Explanation (Optional):

B. Correct – The normal battery charger output voltage is approximately 27 VDC, which provides a constant float charge for the 24 VDC battery. With battery charger output voltage lowering to a value less than the battery's voltage, the battery will begin to discharge. The Source Range Monitors are powered by the 24 VDC system.

A. Incorrect – The battery will begin to discharge as the charger output voltage lowers below 24 VDC.

C. Incorrect – The battery will begin to discharge as the charger output voltage lowers below 24 VDC. The ERVs are powered by the 125 VDC system.

D. Incorrect – The ERVs are powered by the 125 VDC system.

Nine Mile 2008

51 **ID: 2059189** **Points: 1.00**

While performing a unit startup in Mode 2, the SAT feed breaker to bus 142Y spuriously tripped.

- The 1A Diesel Generator FAILED to automatically start
- Attempts to start the 1A DG from the control room have NOT been successful

1A DG was manually started locally and is running at 900 rpm. Attempts to raise DG output voltage above 3800 volts have been unsuccessful.

With regards to the 1A Diesel Generator, what is the next procedurally required action?

A. Trip the 1A Diesel Generator.

B. Synchronize and close ACB 1423 to re-energize 142Y.

C. Synchronize and close ACB 1425 to re-energize 142Y.

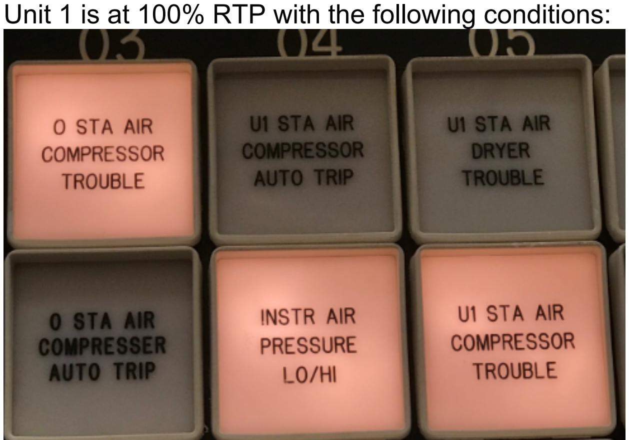
D. Install a jumper on the 1A DG HACR Relay per LOA-DG-101 Attachment A.

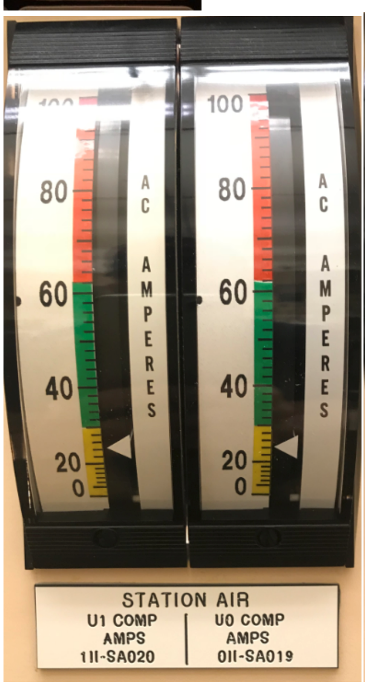
Answer: A

Answer Explanation
A is correct: With bus 142Y de-energized the diesel cooling water pump for the 1A Diesel Generator will not be powered. LOA-DG-101 27.3: If normal parameters cannot be established, Trip 1A DG. B is incorrect: This is the correct answer if the 1A Diesel Generator had come up to 4050 – 4300 Volts C is incorrect: This would bring power back to 142Y if the UAT was being powered. However, with the mode switch in STARTUP, the main generator is not synched to the grid. D is incorrect: The HACR relays are used in synchronization, plausible that installing this jumper would allow ACB 1423 to close. However, this is procedurally not allowed since the Diesel is not at the rated voltage. References: LOA-DG-101 rev 10 Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 2 Group: 1 KA: 264000 Ability to (a) predict the impacts of the following on the EDGs; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of A.C. power 10 CFR Part 55 Content: 41.7 SRO Justification: N/A Question Source: Bank Question History: None Comments: Associated objective(s): 011.00.10

52	ID: 2062112	Points: 1.00

Unit 1 is at 100% RTP with the following conditions:





Which of the following describes the plant status?

A. An air leak exists at the Instrument Air Receiver.

B. An air leak exists at the scram Instrument Header.

C. The Air compressors are operating in Modulate Mode.

D. The Air compressors are operating in a Surge condition.

Answer: D

Answer Explanation

A is incorrect: If a station air leak did exist, then amperage of the compressor would be rising, and not lowering. Plausible because lower system pressure and low pressure alarms are Indications of an air leak, but Compressor Amps would be higher.

B is incorrect: If a station air leak did exist, then amperage of the compressor would be rising, and not lowering. Plausible because lower system pressure and low pressure alarms are Indications of an air leak, but Compressor Amps would be higher.

C is incorrect: Plausible because this is a Mode of operation of the station air compressor prior to the event.

D is correct: With the following conditions present, both air compressor trouble alarms annunciating, station air compressor amperage lower than normal, low pressure conditions of station air, low pressure alarms at receiver and scram air header, this is an indication that the air compressor have both gone into the surge mode of operation.

Reference:
LOP-SA-02 STATION AIR COMPRESSOR STARTUP AND SHUTDOWN Rev 38
LOR-1PM10J-B205 U1 STA AIR COMPRESSOR TROUBLE Rev 2
LOR-1PM10J-B103 0 STA AIR COMPRESSOR TROUBLE Rev 2
LOR-1PM10J-B204 INSTR AIR PRESSURE LO/HI Rev 1
LOR-1H13-P603-A102 SCRAM PILOTVLV AIR HDR PRESS LO Rev 1
Reference provided during examination: none

Cognitive level: High
10CFR Part 55:41(b)(7)

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

PRA: Yes
K/A: 300000 A4.01 INSTRUMENT AIR SYSTEM: Ability to manually operate and / or monitor in the control room: Pressure gauges

SRO Justification: N/A

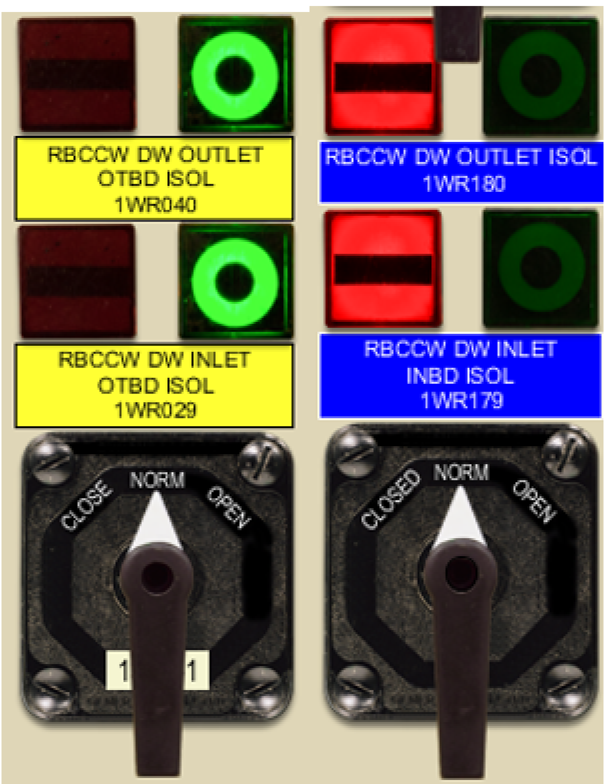
Question Source: New
Question History: None

Objective: 120.00.05
Recall the function, theory of operation, interlocks, trips, and characteristics of the following Plant Air Subsystem components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text:
a. Compressors
b. Air Receivers

Comments: Air Compressor surge events has been a LaSalle OPEX event

53 **ID: 2062671** **Points: 1.00**

The following indications are present following a loss of 112Y.



1) Are these indications expected?
2) If the 1WR179/180 DW Equip RBCCW Inlt/Otlt Inbd Isols Bypass Keylock Switch is taken to BYPASS and the 1WR179/180 Control Switch is taken to OPEN the valves will ___________.

A. (1) No
(2) Open

B. (1) No
(2) Remain Closed

C. (1) Yes
(2) Open

D. (1) Yes
(2) Remain Closed

Answer: C

Answer Explanation
A is incorrect: (1) Plausible because control power for 1WR179/180 logic is 136Y-2 which is not lost. Div 2 WR will also not close on a loss of B RPS as will most other Div 2 PCIVs. B is incorrect: (1) Plausible because control power for 1WR179/180 logic is 136Y-2 which is not lost. Div 2 WR will also not close on a loss of B RPS as will most other Div 2 PCIVs. (2) Plausible because the WR PCIS Bypass keylock Switch was designed for a spurious WR Isolation. If control power was from 112Y the valve would not be able to be reopened until 112Y is restored. C is correct: When 112Y is de-energized the 1B21H-K66X15 contact will close because the Division 2 PCIS relay for that contact lost power. This completes the close relay logic for 1WR0179/180 and causes both valves to close. When the PCIS bypass keylock switch for Div 2 is taken to Bypass it bypasses the PCIS contact in MOV control logic and allows the valves to open when the common hand switch is taken to open. The MOV control power is supplied from AC power to the MOV breaker from an MCC. D is incorrect: (2) Plausible because the WR PCIS Bypass keylock Switch was designed for a spurious WR Isolation. If control power was from 112Y the valve would not be able to be reopened until 112Y is restored. Reference: 1E-1-4096AE, 1E-1-4232AT, LOA-WR-101, LOSS OF REACTOR BUILDING CLOSED COOLING WATER, Rev. 16 Reference provided during examination: none Cognitive level: High 10CFR Part 55: 41.7 / 45.5 / 45.8 Level (RO/SRO): RO Tier: 2 Group: 1 PRA: No K/A: 400000 A4.01 Ability to operate and / or monitor in the control room. CCW indications and control. SRO Justification: N/A Question Source: New Question History: None Comments: Objective # 114.00.14

54	ID: 2061959	Points: 1.00

While performing a startup with reactor pressure at 300 psig, what will be the major driving force for the Scram if the Scram Solenoid Pilot Valves lost power?

A. Reactor Pressure

B. Cooling Header Pressure

C. Scram Accumulator Pressure

D. Control Rod Drive Pump Discharge Pressure

Answer: C

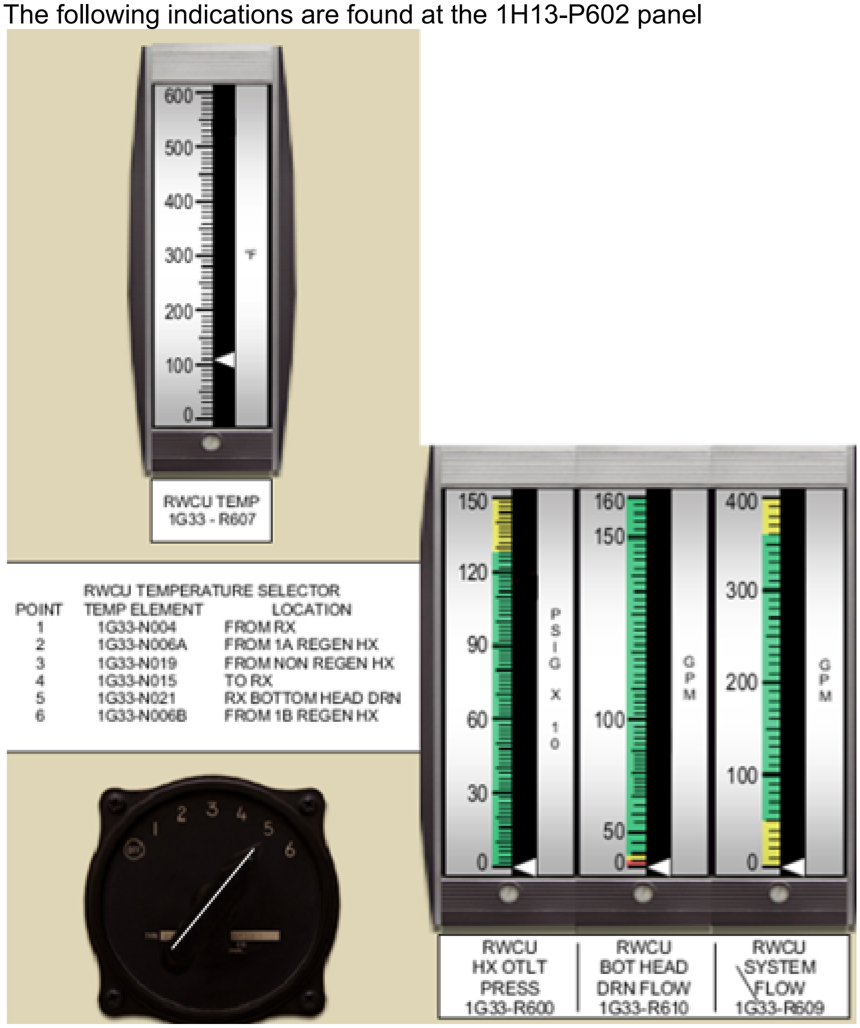
Answer Explanation
A is incorrect: With reactor pressure low and a scram condition present, accumulator pressure will be the major driving force to insert the control rods. Plausible because at high Reactor pressures, as the drive moves upward and accumulator pressure drops below the reactor pressure, a ball check valve opens, letting the reactor pressure complete the scram action. B is incorrect: With reactor pressure low and a scram condition present, accumulator pressure will be the major driving force to insert the control rods. Plausible because this pressure is felt on the insert port of the Control Rod Drive Mechanism which is where pressure is directed to insert the control rod. C is correct: If the reactor pressure is low, such as during startup, the accumulators will fully insert the control rods within the required time without assistance from reactor pressure. D is incorrect: With reactor pressure low and a scram condition present, accumulator pressure will be the major driving force to insert the control rods. Plausible because with the given conditions, this is the driving force that will be used to insert the rod had it failed to insert on a scram signal as well as inserting rods at normal rod speed. Reference: Tech Spec Bases 3.1.4 Reference provided during examination: none Cognitive level: High 10CFR Part 55: 41.7 / 45.7 Level (RO/SRO): 3.0 / 3.0 Tier: 2 Group: 2 PRA: No K/A: 201003 Control Rod and Drive Mechanism Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD AND DRIVE MECHANISM Reactor pressure SRO Justification: N/A Question Source: New Question History: none Comments: Objective # 025.00.21

55	ID: 2062745	Points: 1.00

Unit 1 is in MODE 3.

- Reactor coolant temperature is 220° F
- Both reactor recirculation pumps are secured.
- Rx Vessel Bottom Head metal temperature is 170° F as read from the CRD temperature recorder.

The following indications are found at the 1H13-P602 panel



If starting Reactor Recirculation under these conditions, what point of temperature measurement would you verify for the bottom head area and what issue does this address?

A. Rx Bottom Head Drain temperature, to prevent challenging the 100°F/hr heat up on the bottom head.

B. Rx Vessel Bottom Head Metal temperature to prevent challenging the 100 F/hr heat up on the bottom head.

C. Rx Bottom Head Drain temperature, to prevent dropping below the 10.1°F delta T limit.

D. Rx Vessel Bottom Head Metal temperature, to prevent dropping below the 10.1°F delta T limit.

Answer: B

Answer Explanation
A is incorrect: Because normally the bottom head drain temperature is the preferred choice for addressing SR 3.4.11.3 but with bottom head drain flow < 25 gpm this becomes unusable. B is correct: Per LOA-RR-101 25 gpm minimum RWCU flow through bottom head drain is required for accurate RVBHD temperature readings. As shown in stem with NO flow through the bottom head drain, temperature will be inaccurate and the best value to use will be the bottom head metal temperature as found on the CRD metal temperature recorder. C is incorrect: Plausible because the 10.1°F delta T limit is a temperature limit used for reactor recirculation pump for the difference between the steam dome and RR suction temperature. D is incorrect: Plausible because the 10.1°F delta T limit is a temperature limit used for reactor recirculation pump for the difference between the steam dome and RR suction temperature. Reference: UNIT 1, REACTOR RECIRCULATION SYSTEM ABNORMAL, LOA-RR-101 Rev 39 LOS-RR-SR2 RR PUMP STARTING TEMPERATURE SURVEILLANCE Rev 5, LOP-RR-04 PREPARATION AND STARTUP OF REACTOR RECIRC PUMPS IN SLOW SPEED Rev 59 Cognitive level: High 10CFR Part 55: 41.4 Level (RO/SRO): 2.8/2.9 Tier: 2 Group: 2 PRA: No K/A: 202001 A.1.11 Ability to predict and/or monitor changes in parameters associated with operating the RECIRCULATION SYSTEM controls including: Vessel bottom head drain temperature SRO Justification: N/A Question Source: New Question History: none Comments: Objective: 020.00.05 Recall the function, theory of operation, interlocks, trips, and characteristics of the following Reactor Pressure Vessel and Internals System components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text: c. Bottom Head Drain

56	ID: 2058817	Points: 1.00

Unit 1 is at 100% RTP with 1B RT Pump online.

Two minutes later…
- The Reactor was scrammed due to a feedwater leak in the Main Steam Tunnel
- All rods remained out
- SBLC is injecting
- Reactor water level is at -30” on Wide Range lowering at 2"/min

What is the status of the 1B RT Pump?

A. The 1B RT Pump is online

B. The 1B RT Pump tripped due to 1G33-F001 (RT inboard isolation valve) closing

C. The 1B RT Pump tripped due to 1G33-F004 (RT outboard isolation valve) closing

D. The 1B RT Pump tripped due to a Group 5 isolation caused by low reactor water level

Answer: C

Answer Explanation
A is incorrect: The 1B RT pump will trip off when the RT outboard isolation valve 1G33-F004 reaches full closed. Plausible because the student may that the Division I PCIS valve would only send a trip signal to the Division I powered RT Pump. B is incorrect: Low level for Group VI and VII are both +11.0” in NR which the stem of the question meets. SBLC initiation causes a group V isolation, however the level that causes a group V isolation is -50”. plausible because the 1G33-F004 valve closing trips the RT pump. C is correct: The 1B RT pump will trip on the closure of the outboard isolation valve when the SBLC pump is started. Both the RT inboard and outboard isolation valves send pump trip signals to each pump on valve closure. D is incorrect: Plausible since this is an RT pump trip and there is a high energy leak in the secondary containment. Incorrect because there is no indication that the main steam tunnel has failed. References: LOP-CX-06 rev 8, LOR-1H13-P602-A107, Reactor water cleanup recirculation pump 1G33-C001B Automatic Trip rev 4, Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 2 Group: 2 KA: 204000 A2.10, Ability to (a) predict the impacts of the following on the REACTOR WATER CLEANUP SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve closures 10 CFR Part 55 Content: 41.5 / 45.6 SRO Justification: 55.43(b)(5) Question Source: New Question History: N/A Comments: Unable to write a question for RO to both parts of this KA. The question is written to the higher order objective of (a) predict the impacts of the following on the REACTOR WATER CLEANUP SYSTEM: Associated objective(s): 27.00.05

57	ID: 2058816	Points: 1.00

Which of the following APRM and RBM Channel combinations will be affected by a downscale failure of FLOW UNIT ‘D’?

	APRM Channel	RBM Channel
A.	A	A
B.	D	A
C.	E	B
D.	F	B

Answer: D

Answer Explanation

A is incorrect: Flow units A and C provide inputs to APRM A and RBM A.

B is incorrect: Flow units A and C provide inputs to APRM D and RBM A.

C is incorrect: Flow units A and C provide inputs to APRM E and Flow units B & D provide inputs RBM B.

D is correct: Flow units B and D provide inputs to APRM F and RBM B.

References: LOR-1H13-P603-A209, Rev 3 APRM FLOW BIAS OFF NORM, LOA-NR-101 rev 20 Neutron Monitoring Trouble
Reference provided during examination: None
Cognitive level: Memory
Level (RO/SRO): RO
Tier: 2 Group: 2
KA: 215002 Rod Block Monitor, Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including: (CFR: 41.7 / 45.7) Verification of proper functioning/operability.
10 CFR Part 55 Content: 41.12 / 43.4 / 45.10
SRO Justification: 55.43(b)(5)
Question Source: Bank
Question History: Fitzpatrick 2010

Comments:

Associated objective(s): 045.00.16

58	ID: 2061952	Points: 1.00

Unit 1 was scrammed due to a small unisolable feedwater leak.

- ‘A’ RHR Pump is running in suppression pool cooling
- RCIC is injecting
- Suppression pool level is 1.5 feet below its normal operating level and stable

Which of the following are valid methods to monitor Suppression Pool Temperature from the Main Control Room?

1) Suppression Pool Temperature Monitors (Numac)
2) RCIC Discharge Temperature
3) A RHR Heat Exchanger Inlet
4) PPC

A. 1, 2 & 4

B. 1 ONLY

C. 2 & 3

D. 3 & 4

Answer: D

Answer Explanation
A is incorrect: Plausible because 4 is correct and 1 is correct greater than -1 foot below normal suppression pool operating level, 2 would give indication of suppression pool temperature, but it cannot be read from the main control room. B is incorrect: Plausible because if suppression pool were higher than -1ft below normal operating level 1 would be a correct answer. C is incorrect: Plausible because 2 would give indication of suppression pool temperature, but it cannot be read from the main control room, and 3 is one of the correct answers D is correct: Computer Points L122 and L123 are to be used when suppression pool level is greater than or equal to a foot less than its normal operating value. ‘A’ RHR HX inlet can be used when ‘A’ RHR pump is running and is available on the 1E12-R601 on the 1H13-P601. RCIC discharge temperature is called out by LOP-CM-03 as a valid means of measuring suppression pool temperature, however it must be obtained locally off temperature indicator 1E51-R005. References: LOP-CM-03 rev 13, Suppression chamber average water temperature determination Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 2 Group: 2 KA: 219000 RHR Pool Cooling Mode A4.12 Ability to manually operate or monitor in the control room: Suppression pool temperature 10 CFR Part 55 Content: 41.9 / 45.5-45.8 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 091.00.05

59	ID: 2062287	Points: 1.00

Unit 1 is at 100% Power.

- LOS-RH-Q1, RHR (LPCI) AND RHR SERVICE WATER PUMP AND VALVE INSERVICE TEST FOR MODES 1, 2, 3, 4 AND 5, is in progress.
- 1E12-F027A RHR, Chamber Spray Valve, failed its exercise test and is unable to be opened remotely or locally.

LCO 3.6.2.4 (RHR Suppression Pool Spray) __________ and applies in Modes ___________.

A. IS MET
1, 2, 3, 4, and 5

B. IS MET
1, 2, and 3 ONLY

C. IS NOT MET
1, 2, 3, 4, and 5

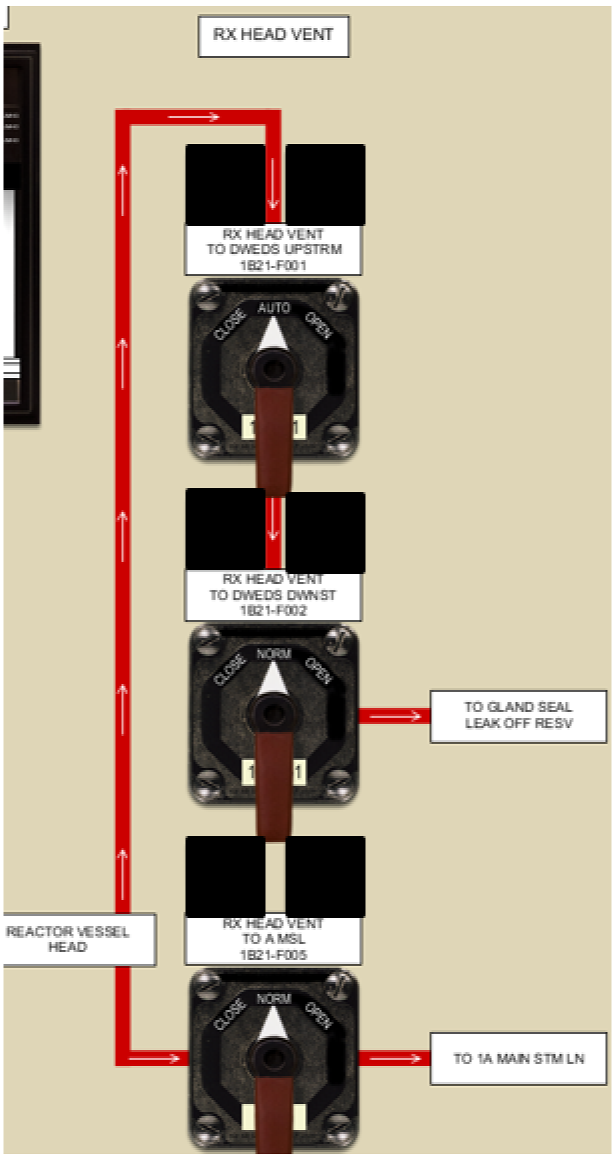
D. IS NOT MET
1, 2, and 3 ONLY

Answer: D

Answer Explanation
A is incorrect: Plausible that only one RHR subsystem be required because drywell and chamber spray can both be put on the same loop of RHR and perform its design function. Also plausible because RHR does have injection requirements for modes 4, 5 in LCO 3.5.2. B is incorrect: Plausible that only one RHR subsystem be required because drywell and chamber spray can both be put on the same loop of RHR and perform its design function. C is incorrect: Plausible because LCO 3.6.2.4 is not met and RHR does have injection requirements for in modes 4, 5 in LCO 3.5.2 D is correct: LCO 3.6.2.4 states ‘Two suppression pool spray subsystems shall be OPERABLE’ and applies in modes 1, 2, and 3. It is system level (RO) knowledge to know that with the 1E12-F027A closed, no flow path exists for chamber spray. References: TS 3.6.2.4 RHR Suppression Pool Spray 184/171 Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 2 Group: 2 KA: 226001 RHR: Containment Spray Mode G02.36 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 / 45.13 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 064.00.21

60	**ID: 2062285**	**Points: 1.00**

Unit 1 is in MODE 2 at 3% RTP:



At this power what will be the position of the Reactor Head Vents 1B21-F001 and 1B21-F002 and why?

A. Open; to prevent noble gas release in the drywell.

B. Open; to prevent non-condensables from accumulating in the reactor head area.

C. Closed; to prevent noble gas release in the drywell.

D. Closed; to prevent non-condensables from accumulating in the reactor head area.

Answer: C

Answer Explanation
A is incorrect: Plausible because this is the function of the head vents at power B is incorrect: Plausible because this is the function of the head vents at power C is correct: When the Unit is in MODE 2 with reactor power at 3%, the RPV would be pressurized. With the reactor pressurized the head vent valves must be re-aligned to the "A" main steam line. This requires the 1B21-F001 and 1B21-F002 to be closed, per LGP-1-1 Normal Unit Startup; "Prior to exceeding 212 ºF, 1(2) B21-F001 and 1(2) B21-F002 must be verified closed and power to one of the valves removed. [LSCS-FPR Table H.4-111]" D is incorrect: Plausible because this is the correct position of these valves at 3% power and this is a function of the head vents at power but with these valves closed it does prevent gases from accumulating in the reactor head area. Reference: Chapter P & ID M-55 Sheet 1 Rev Y, P & ID M-93 Sheet 4 Rev BA, LGP-1-1 NORMAL UNIT STARTUP Rev 122 Reference provided during examination: Closed Cognitive level: High 10CFR Part 55:41(b)(4) Level (RO/SRO): 2.7/2.7 Tier: 2 Group: 2 PRA: No K/A: K1.24 Knowledge of the physical connections and/or cause effect relationships between MAIN AND REHEAT STEAM SYSTEM and the following: Head vent SRO Justification: N/A Question Source: New Question History: none   Comments: Objective # 070.00.05 Recall the function, theory of operation, interlocks, trips and characteristics of the following Main Steam System components and relate these items to overall system operation while operating the system or on an exam in accordance with the student text: Reactor Head Vent

61	ID: 2058029	Points: 1.00

Unit 1 is 20% RTP.

The following is displayed on the 1PM02J:

With NO operator action, how will the plant respond?

1) Turbine Trip
2) Reactor Scram
3) Reactor Recirc Pump Downshift
4) ES Non-Return Check Valves Close

A. 1 and 3

B. 2 and 3

C. 1 and 4

D. 2 and 4

Answer: C

Answer Explanation
A is incorrect: The Turbine Trip is at 1100 psig EHC Pressure. Plausible because when RETS reaches 510 psig it will generate a TSV Not Full Open or TCV Fast Closure Scram signal, however this is bypassed when the Reactor is < 25% power. B is incorrect: Plausible because when RETS reaches 510 psig it will generate a TSV Not Full Open or TCV Fast Closure Scram signal, however this is bypassed when the Reactor is < 25% power. Plausible because when RETS reaches 510 psig RPS will generate an EOC-RPT signal to downshift RR Pumps to counteract the possible high pressure condition from the Turbine Trip, however this is bypassed when the Reactor is < 25% power. C is correct: FAS is supplied by the EHC Pumps and with the 1A EHC Pump OOS the given trip indicates that the only running EHC Pump has tripped and EHC Pressure is lowering. At 1100 psig FAS Pressure is an Automatic Turbine Trip. On a turbine trip the Air Extraction Relay Dump Valve closes causing a loss of Instrument Air to the ES NRCVs and causing them to close to prevent reverse ES flow causing a turbine overspeed condition following a turbine trip. D is incorrect: Plausible because when RETS reaches 510 psig RPS will generate an EOC-RPT signal to downshift RR Pumps to counteract the possible high pressure condition from the Turbine Trip, however this is bypassed when the Reactor is < 25% power. References: LOA-TG-101 Unit 1 Turbine Generator Rev. 19; LOR-1H13-P603-A211, Channel A1/B1 Turbine Control Valve and Fast Closure Turbine Stop Valve Trip Bypass Rev. 9; LOR-1H13-P603-B410, Reactor Recirculation Pump Trip System A Bypassed Rev. 4 Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 2 Group: 2 KA: 245000 K3.08 Knowledge of the effect that a loss or malfunction of the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS will have on following: Reactor/turbine pressure control system 10 CFR Part 55 Content: 41.7 / 45.4 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 73.00.05

62	ID: 2058040	Points: 1.00

What is the Power Supply to 1C Condensate/Condensate Booster Pump?

A. Bus 141X

B. Bus 142X

C. Bus 151

D. Bus 152

Answer: C

Answer Explanation

A is incorrect: The Power Supply to 1C CD/CB Pump is Bus 151. Plausible because Bus 141X Powers 1A and 1C Circulating Water Pumps which are also located on 1PM03J.

B is incorrect: The Power Supply to 1C CD/CB Pump is Bus 151. Plausible because Bus 142X Powers the 1B/1D Heater Drain Pumps which are also located on the 1PM03J. It also powers the 1C Primary Containment Ventilation (VP) Chiller.

C is correct: The Power Supply to 1C CD/CB Pump is Bus 151.

D is incorrect: The Power Supply to 1C CD/CB Pump is Bus 151. Plausible because Bus 152 Powers 1C Motor Driven Reactor Feed Pump as well as 1B/1D CD/CB Pumps which are also located on 1PM03J.

References: LOA-AP-101 Unit 1 AC Power System Abnormal
Reference provided during examination: None
Cognitive level: Memory
Level (RO/SRO): RO
Tier: 2 Group: 2
KA: 256000 Reactor Condensate System K2.01 Knowledge of the electrical power supplies to the following: System Pumps
10 CFR Part 55 Content: 41.7
SRO Justification: N/A
Question Source: New
Question History: none

Comments:

Associated objective(s): 75.00.16

63	ID: 2058840	Points: 1.00

What design feature of the Offgas system prevents the poisoning of the recombiner catalyst by the presence of water?

A. Offgas reheater

B. Offgas preheater

C. Offgas condenser

D. Offgas moisture separator

Answer: B

Answer Explanation

A is incorrect: Plausible because it removes moisture from the offgas system, incorrect because it downstream of the hydrogen recombiner.

B is correct: The off gas preheater

C is incorrect: Plausible because it removes moisture from the offgas system, incorrect because it downstream of the hydrogen recombiner.

D is incorrect: Plausible because it removes moisture from the offgas system, incorrect because it downstream of the hydrogen recombiner.

References: LOA-OG-101 rev22, LOR-1N62-P600-A104 rev2 OFF GAS 2A Catalytic Recom Inlet Temp Lo, M-88 Sheet 2, M-88 Sheet 3
Reference provided during examination: None
Cognitive level: Memory
Level (RO/SRO): RO
Tier: 2 Group: 2
KA: 271000 K4.02 Knowledge of OFFGAS SYSTEM design feature(s) and/or interlocks which provide for the following: Prevention of the poisoning of the recombiner catalysts by the presence of water.
10 CFR Part 55 Content: 41.7
SRO Justification: N/A
Question Source: New
Question History: N/A

Comments:

Associated objective(s): 80.00.05

64	ID: 2062062	Points: 1.00

Which of the following conditions could cause the VC Emergency Makeup Unit to automatically actuate?

A. Smoke in the outside air intake.

B. Smoke in the return air plenum.

C. Ammonia in the outside air intake.

D. Ammonia in return air plenum.

Answer: A

Answer Explanation

A is correct: If CR HVAC Outside Air Zone alarm 424 (local 1GJ2) is actuated (smoke in outside air) then the Emergency Makeup Train will automatically start making the only correct choice.

B is incorrect: Plausible as smoke in the return air plenum does have automatic actions associated with it. Incorrect because it aligns the charcoal filter, not start the EMU.

C is incorrect: Plausible because this does affect control habitability, and has an alarm in the MCR, and directs manually starting the EMU.

D is incorrect: Plausible because this does affect control habitability, and has an alarm in the MCR, and directs manually starting the EMU.

References: LOP-VC-01 rev52, Control Room HVAC Operation LOR-1PM06J-B204 rev 7, CR HVAC Ammonia Detection Alarm
Reference provided during examination: None
Cognitive level: Memory
Level (RO/SRO): RO
Tier: 2 Group: 1
KA: 286000 Fire Protection K5.07 Knowledge of the operational implications on the following concepts as they apply to Fire Protection: Smoke Detection
10 CFR Part 55 Content: 41.5 / 45.3
SRO Justification: N/A
Question Source: Bank
Question History: None

Comments:

Associated objective(s): 064.00.14

65	ID: 2062751	Points: 1.00

Units 1 and 2 are at 100% RTP.

- During a Fire Protection Surveillance, the "A" VC Charcoal train was accidently deluged.

What are the required actions?

Enter......

1). LCO 3.7.4 for Control Room Area Filtration Inoperable
2). LCO 3.7.5 for Control Room Area Ventilation Air Conditioning Inoperable
3). TRM 3.3.p for Fire Protection Instrumentation Inoperable
4). TRM 3.7.k for Deluge Spray and Sprinkler System Inoperable

A. 1 and 2.

B. 2 and 3.

C. 3 and 4.

D. 1 and 4.

Answer: D

Answer Explanation
A is incorrect: Plausible because LCO 3.7.4 for Control Room Area Filtration Inoperable is applicable and LCO 3.7.5 is a related LCO for "A" VC train but is not applicable as described in the stem B is incorrect: Plausible because both LCO 3.7.5 and TRM 3.3.p have some application to the VC system but as described in the stem neither are affected in this example. C is incorrect: Plausible because TRM 3.7.k Deluge Spray and Sprinkler System is applicable and some aspects of TRM 3.3.p Fire Protection Instrumentation have applicability to VC but is not applicable as described in the stem. D is correct: With an accidental deluge of the VC system the charcoal would not operate properly should be considered INOP and TS 3.7.4 entered. With the deluge not operating properly the deluge system for the VC charcoal should be considered INOP and TRM 3.7.k entered. Reference: LCO 3.7.4 for Control Room Area Filtration Inoperable, LCO 3.7.5 for Control Room Area Ventilation Air Conditioning Inoperable, TRM 3.3.p for Fire Protection Instrumentation Inoperable, TRM 3.7.k for Deluge Spray and Sprinkler System Inoperable Reference provided during examination: TRM 3.3.p, TRM 3.3.k Cognitive level: Memory 10CFR Part 55:41(b)(7) Level (RO/SRO): 2.6/2.8 Tier: 2 Group: 2 PRA: No K/A:290003 K6.04 Control Room HVAC: Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROOM HVAC: Fire protection SRO Justification: N/A Question Source: New Question History: N/A Comments: This question requires above the line information to answer and is RO knowledge. Objective #117.00.22 Given a copy of Technical Specifications, key system parameters, and various plant conditions, determine if Technical Specification LCOs are met, recall the basis for the LCO and identify the required actions in accordance with Technical Specifications, while operating the system or on an exam.

66	ID: 2057730	Points: 1.00

Unit 1 is at 100% RTP:

- LIS-MS-305A, UNIT 1 MAIN STEAM LINE PIPE TUNNEL HIGH DIFFERENTIAL TEMPERATURE ISOLATION (DIV 1) FUNCTIONAL TEST is in progress.
- Alarms have been flagged.

Subsequently:

- LOR-1H13-P601-F501, DIV 1 MSL PIPE TUNNEL DIFF TEMP HI Alarms. (The first annunciation of the shift).
- This alarm has no flag.

What action(s) is/are required?

Acknowledge the annunciator...

A. ONLY.

B. AND announce, "expected alarm"

C. AND announce, "DIV 1 MSL PIPE TUNNEL DIFF TEMP HI" ONLY.

D. AND announce, "DIV 1 MSL PIPE TUNNEL DIFF TEMP HI" AND reference the LOR.

Answer: D

Answer Explanation
A is incorrect: this is not an expected alarm related to the surveillance. It is plausible if the annunciator was discussed and flagged prior to receiving it. B is incorrect: this is not an expected alarm related to the surveillance. It is plausible if the annunciator was discussed prior to receiving it. C is incorrect: this the first time the alarm was received this shift, but is plausible because the LOR is related to the maintenance procedure. D is correct: This alarm must be treated as unexpected because it was not flagged. OP-AA-103-102 states that all annunciators must be reported to the unit supervisor unless the alarm is considered expected and appropriate actions have been taken. In this instance an alarm came in that was not flagged. This alarm would be considered unexpected. References: OP-AA-103-102 rev 18, LIS-MS-305A rev 14, LOR-1H13-P601-F501 rev 4 Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 3 Group: 3 KA: G.2.1.1 Knowledge of conduct of operations requirements 10 CFR Part 55 Content: 41.10 / 45.13 SRO Justification: 55.43(b)(5) Question Source: Bank Question History: Cooper 2008 Comments: Associated objective(s): 637.010

67	ID: 2057728	Points: 1.00

Unit 2 is at 100% RTP:

- 2A and 2C CW pumps are running
- A loss of 211Y occurs

The Assist NSO will dispatch the equipment operator to...

A. check condenser pit level at the Lake Screen House.

B. verify traveling screen differential pressure is less than 12 inches.

C. verify the slip guard relay for the 2B CW pump is clear and start the 2B CW pump from 2PM03J.

D. throttle discharge structure sluice gate 2 closed, after starting the 2B CW pump from the 2PM03J.

Answer: D

Answer Explanation
A is incorrect: This is plausible because no CW pumps are running and LOA-CW-101 has direction to perform this action. It is incorrect because the 1B CW pump is available and can be restarted right away. B incorrect: This is plausible because this would trip the running circ water pumps and is directed by LOA-CW-101. It is incorrect because The CW pumps were lost due to loss of power. C is incorrect: This is plausible because it is directed by both LOA-CW-101 for a loss of all circulating water pumps and LOR-2PM03J-B406 for CW pumps that tripped during a fast transfer. This is not correct because the B circ water pump was not running so it's slipguard relay would be clear. D is correct: 2A and 2C CW pumps will trip on loss of power (241x will not fast transfer due to loss of control power). 242x will still have power and the 2B CW pump will be able to be started right away from the 1PM03J. Per LOA-DC-101 the NSO should dispatch an operator to throttle closed the discharge structure sluice gates within an hour to prevent pump run out of the only running CW pump. References: LOA-CW-201 rev 22, LOR-2PM03J-B406 rev 6, LOA-DC-201 rev 19 Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 3 Group: 3 KA: G.2.1.8 Ability to coordinate personnel activities outside the control room. RO 3.4 10 CFR Part 55 Content: 41.10 / 45.5 / 45.12 / 45.13 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 111.00.05

68 **ID: 2057726** **Points: 1.00**

Which of the following plant conditions is allowed to be communicated during the Scram Choreography?

Updating the crew on a...

A. spurious Group 4 Isolation.

B. valid Group 7 Isolation on a Reactor Level signal.

C. valid Group 1 Isolation on Low Condenser Vacuum.

D. SINGLE Level 8 channel light illuminated with Reactor Level at 0" and slowly rising.

Answer: C

Answer Explanation
A is incorrect: A group 4 isolation will only cause a change in secondary containment parameters. Plausible because this is a PCIS isolation that is not expected during a scram. B is incorrect: A group 7 on reactor level (Level 3/+12.5")is expected due to shrink during a scram. Plausible because this is a valid PCIS signal. C is correct: Per OP-LA-103-102-1002, the only exception is when an immediate response is required to mitigate undesired effects to Power, Pressure, Level, or Containment. When a Group 1 happens, MSIVs Close, this will affect the Reactor Level, Pressure and Containment when SRVs lift to maintain Reactor Pressure. D is incorrect: The Level 8 (+60”) trip logic is a 2 of 3 logic and will cause no change in the any system or parameter. Plausible because if 2 channels were in a tripped condition then it would cause a loss of all Feedwater Pumps and thus affect Reactor level. References: OP-LA-103-102-1002 rev 17, STRATEGIES FOR SUCCESSFUL TRANSIENT MITIGATION Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 3 Group: 1 KA: G.2.1.17 Ability to make accurate, clear, and concise verbal reports. 10 CFR Part 55 Content: 41.10 / 45.12 / 45.13 SRO Justification: N/A Question Source: New Question History: none Comments: Associated objective(s):

69 **ID: 2057610** **Points: 1.00**

The reactor water cleanup system is tagged out for maintenance.

A valve that will have maintenance performed on it is discovered with an info tag placed on it.

What action by the work supervisor is acceptable and why?

A. Contact the Shift for permission to work on this valve because INFO cards are only used for Condition Dependent Clearance Orders to obtain the condition.

B. Inform the Outage Control Center no work can be done on this valve because INFO cards are signed off as placed and therefore are a part of the Clearance Order.

C. Contact the Shift for permission to work on this valve because INFO cards may NOT hold a boundary point, but only contain information.

D. Inform the Contract Supervisor no work can be done on this valve because this would violate the Clearance Order zone of protection.

Answer: C

Answer Explanation
A is incorrect: Plausible because information tags are used as part of a Condition Dependent Clearance Order. B is incorrect: Plausible because informing inform the Outage Control Center this would be the correct choice if NO work could not continue. C is correct: Section 5.3.1 Since Information Tags may not be used to hold a boundary point in a zone of protection, but merely identify a component that is associated with a C/O, then components with Information Tags may be operated, removed, or have maintenance performed on the component with Shift Management's permission. D is incorrect: Information Tags may not be used to hold a boundary point in a zone of protection with a clearance order. Plausible because other tags do hold boundaries. References: OP-AA-109-101 rev 12 section 5.3.1, Clearance and Tagging Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 3 Group: 4 KA: Emergency Procedures/Plan G. 2.2.13 GENERIC KNOWLEDGES AND ABILITIES, Equipment Control: Knowledge of tagging and clearance procedures 10 CFR Part 55 Content: 41.10 / 45.13 SRO Justification: N/A Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Question Source: Bank Question History: N/A Comments: Authorizing tagouts to be hung is an SRO function. Knowledge of which cards hold a boundary position is RO knowledge and is integral to the Clearance Order Writer's job. Associated objective(s): 801.00.03

70	ID: 2057609	Points: 1.00

Which of the following identifies a Tech Spec LCO condition which would require entry into the Required Action Statement and logged per OP-AA-111-101, OPERATING NARRATIVE LOGS AND RECORDS?

A. 'V' SRV INOP with Thermal Power at 30%

B. 'A' and 'D' SRMs INOP with Thermal Power at 20%

C. 'A' Rod Block Monitor (RBM) INOP with Thermal Power at 20%

D. 'A' and 'E' OPRM Modules are INOP with Thermal Power at 30%

Answer: D

Answer Explanation

A is incorrect Per Tech Spec 3.4.4, The safety function of 12 SRVs shall be operable while in mode 1, 2, and 3. With ‘V’ SRV INOP there are still 12 SRVs available. Plausible because if the 'U' or 'S' SRVs are required to operable in Mode 1 per TRM 3.5.b.

B is incorrect: SRMs are not required in Mode 1 (> 15% Power). Plausible because per Tech Spec 3.3.1.2 SRMs 3 SRMs are required to be operable in Mode 2.

C is incorrect: Per Tech Spec 3.3.2.1, 2 RBM Channels are required only when Thermal Power is $\geq$30% and no peripheral rod selected. Plausible because the Rod Worth Minimizer is required $\leq$10%.

D is correct: With both ‘A’ and ‘E’ OPRMs INOP it makes the A1 Channel of the OPRM System INOP and per Tech Spec 3.3.1.3, 4 Channels of OPRM Instrumentation are required to be operable when Reactor Power is greater than or equal to 25%.

References: Tech Specs 3.3.1.2 SRM , 3.3.1.3 OPRM, 3.4.4 SRV, and 3.3.2.1 Control Rod Block Instrumentation, OP-AA-111-101 rev 13, Operating Narrative Logs and Records
Reference provided during examination: none
Cognitive level: High

Importance(RO/SRO): 3.1/4.6
Tier: Group: 3
KA:G 2.2.23 Ability to track Technical Specification limiting conditions for operations.
10 CFR Part 55 Content: 41.(b)(10)
SRO Justification: N/A
Question Source: New
Question History: N/A

Comments: LCOs and 1 hour or less Technical Specifications are RO required knowledge at LaSalle Station.

Objective: 201.00.08

71	ID: 2057596	Points: 1.00

Unit 2 is at 100% RTP:

- Reactor water level is +36 inches and stable
- Drywell pressure is 0.2 psig and stable
- The Division 1 ADS Manual Initiation pushbuttons (2B21-S32A and 2B21-S33A) are armed, depressed, and released

Ten seconds later the Division 1 ADS Low Level Logic Reset Pushbutton (2B21-S31A) is depressed and released.

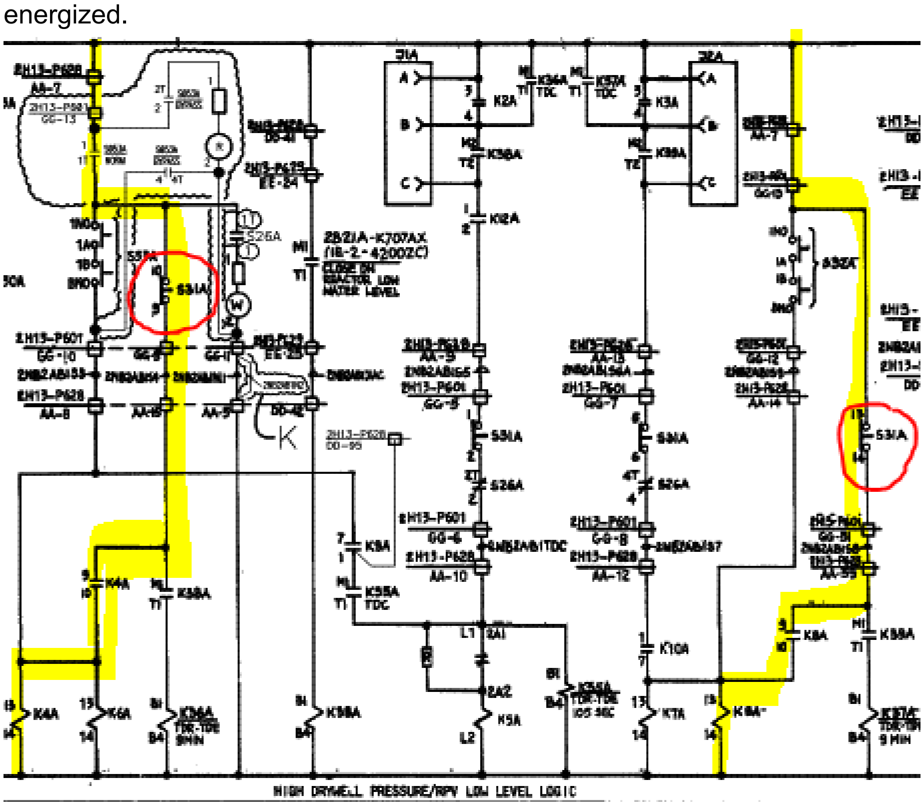
What is the status of the following relays?

	K4A	K8A
A.	De-Energized	De-Energized
B.	Energized	De-Energized
C.	De-Energized	Energized
D.	Energized	Energized

Answer: A

Answer Explanation

A is correct: When the Division 1 ADS Initiation pushbuttons (S32A and S33A) are armed and depressed and with the ADS bypass switch in normal it will energize the seal-in contacts (K4A and K8A). When the pushbuttons are released the seal-in circuit will remain energized and thus the K4A, K6A, K7A, and K8A will remain energized. When the ADS Low Level Reset Pushbuttons are depressed it opens the circuit and de-energizes the K4A and K8A relays, opening the K4A and K8A contacts, therefore maintaining the K4A and K8A relays de-energized.



B is incorrect: Plausible because only one reset pushbutton was depressed.

C is incorrect: Plausible because only one reset pushbutton was depressed.

D is incorrect: Plausible because ADS requires more than just a low-level signal to actuate automatically and energize the K4A and K8A relays. Therefore, would require more than just the low-level reset pushbuttons to reset the logic (i.e. the initiating signals would have to be clear)

References: 1E-2-4201AB
Reference provided during examination: 1E-2-4201AB
Cognitive level: High
Level (RO/SRO): RO
Tier: 3 Group: 2

KA: 2.2.41 Ability to obtain and interpret station electrical and mechanical drawings.
10 CFR Part 55 Content: 41.10 / 45.12 / 45.13
SRO Justification: N/A
Question Source: New
Question History: none

Comments:

Associated objective(s): 62.00.04

72 **ID: 2057589** **Points: 1.00**

Unit 1 is at rated power and annunciator 1H13-P601-B110, Reactor Building Radiation HI is in alarm

Based on the given conditions, which indication would require entry into Emergency Operating Procedures?

A. RB TIP ROOM

B. RB South HCU

C. RB TIP Drive Units

D. TB OFF GAS SAMPLE STATION

Answer: B

Answer Explanation
A is incorrect: Plausible because this ARM actuates the RB RAD annunciator on the 1H13-P601, but it is not listed on Detail R. It is indicating a failure low. B is correct: The Reactor Building HCU is a valid alarm in Detail R of LGA-002. Entry conditions state any Area Radiation above max normal (Detail R), which means any of the above ARMs that have a valid alarm. C is incorrect: This ARM is not in alarm. Plausible because this ARM is in Detail R. D is incorrect: This ARM is not listed on Detail R. Plausible because it is in in alarm and pegged high. Cog Level: High License Level: RO Reference: LGA-002, Secondary Containment Control, Revision 9 Reference provided: none 10CFR55.41(b)(10) 41.11 / 41.12 / 43.4 / 45.9 Objective: 051.00.05 KA:G2.3.5 - Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. 2.9/2.9 Question Source: Bank Question History: N/A

73 **ID: 2057328** **Points: 1.00**

Unit 2 is at 5% RTP:

- The Reactor Mode Switch is in 'STARTUP'.
- A Drywell entry is in progress with personnel currently investigating a suspected leak

Subsequently

- A high worth Control Rod begins to drift OUT

The crew will...

A. Scram Unit 2 AND evacuate the Drywell.

B. Scram Unit 2 AND continue to investigate the leak.

C. evacuate the Drywell AND follow-up with actions to insert ONLY the drifting control rod.

D. continue to investigate the leak AND follow-up with actions to insert ONLY the drifting control rod.

Answer: A

Answer Explanation
A is correct: Per LAP-900-45, with personnel inside the drywell at power, the Reactor Operator SHALL scram the reactor immediately upon observing any unexpected power increase AND evacuate all personnel from the Drywell. B is incorrect: While it is correct to scram the unit following a power spike, the personnel in the drywell are required to evacuate. However, this is plausible because following the power spike the rad levels should lower due to the lowering of reactor power from the scram. C is incorrect: In this situation a SCRAM is required. Plausible because you are supposed to evacuate the drywell. This is also plausible because during normal operations the crew would insert the drifting control rod per LOA-RD-101. D is incorrect: In this situation a SCRAM is required. Plausible because per LAP-900-45, certain areas of the drywell are accessible up to 40% reactor power. This is also plausible because during normal operations the crew would insert the drifting control rod per LOA-RD-101. References: LAP-900-45 rev16, Drywell Entry, LOA-RD-101 rev 21, Control Rod Drive Abnormal, LOP-DW-02 rev 15 Drywell Entry and Inspection Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 3 Group: 3 KA: G2.3.12 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 / 45.9 / 45.10 SRO Justification: N/A Question Source: New Question History: none Comments: Associated objective(s):

74	**ID: 2062224**	**Points: 1.00**

During an emergency condition, Reactor Operator actions that DEVIATE from plant Technical Specifications are needed to protect the health and safety of the public.

In accordance with HU-AA-104-101, Procedure Use and Adherence, these actions require...

A. approval of the Plant Manager (non-licensed).

B. approval of a licensed Senior Reactor Operator.

C. concurrence of a second licensed Reactor Operator.

D. approval of the Nuclear Regulatory Commission (NRC).

Answer: B

Answer Explanation
A is incorrect: Plausible because the Plant Manager is responsible for safe and reliable plant operation. 10CFR50.54: (x) A licensee may take reasonable action that departs from a license condition or a technical specification (contained in a license issued under this part) in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent. (y) Licensee action permitted by paragraph (x) of this section shall be approved, as a minimum, by a licensed senior operator, or, at a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted, by either a licensed senior operator or a certified fuel handler, prior to taking the action. B is correct: The licensee may take reasonable action that departs from a license condition or a Technical Specification in an emergency when: 1. The action is immediately needed to protect the public health and safety, and 2. No action consistent with license conditions and Technical Specifications that can provide adequate or equivalent protection is immediately apparent, and 3. As a minimum a licensed Senior Reactor Operator has approved the licensee action prior to taking the action. C is incorrect: Plausible because during non-transient conditions, any knowledge-based decision executed by Operations shall be peer checked. D is incorrect: Plausible because NRC notification is required. References: HU-AA-104-101, Procedure Use and Adherence Rev. 5 Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 3 KA: G2.4.12 Knowledge of general operating crew responsibilities during emergency operations 10 CFR Part 55 Content: 41.10 / 45.12 SRO Justification: N/A Question Source: Bank Question History: Perry 2001 ILT NRC Exam 2011 Quad City Cert Exam Comments: Associated objective(s): 638.010

EXAMINATION ANSWER KEY

17-1 NRCEXAM

75 **ID: 2057275** **Points: 1.00**

1) Which emergency facility assumes responsibility for communications with offsite agencies including the NRC once it is activated?
2) What is the lowest classification level that requires this facility's activation during a non-security event?

A. 1) Technical Support Center (TSC)
2) Alert

B. 1) Technical Support Center (TSC)
2) Unusual Event

C. 1) Operations Support Center (OSC)
2) Alert

D. 1) Operations Support Center (OSC)
2) Unusual Event

Answer: A

Answer Explanation
A is correct: EP-AA-112 rev 19 section 4.1.3, 'The Control Room shall initiate activation of the ERO at the Alert or higher classification level; or for MA/MW stations at the security Unusual Events (HU1 - Confirmed SECURITY CONDITION or threat, which indicates a potential degradation in the level of safety of the plant), or if directed by site specific activation guidance, which requires the staffing of emergency facilities.' There is no indication of a security event or degradation in the level of safety of the plant. Therefore, Alert is the lowest classification level that requires this facility's activation B is incorrect: This is the lowest classification if there was a security event or degradation in the level of safety at the plant, which is not present in the stem of the question. C is incorrect: The OSC provides support to the TSC but establishing communications with the NRC is not one of its duties. D is incorrect: The OSC provides support to the TSC but establishing communications with the NRC is not one of its duties. This is the lowest classification if there was a security event or degradation in the level of safety at the plant, which is not present in the stem of the question. References: EP-AA-112 rev 19 section 4.1.3, ERO/ERF Activation and Operation Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): RO Tier: 3 Group: 4 KA: Emergency Procedures/Plan A2.4.42 Knowledge of emergency response facilities. 10 CFR Part 55 Content: 41.10 / 45.10 SRO Justification: N/A Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Question Source: Bank Question History: Catawba 2009 ILT NRC Exam Comments: Associated objective(s): 711.00.00

76 **ID: 2058240** **Points: 1.00**

Unit 2 is at 100% RTP with the Division 1 Diesel Generator OOS:

- Loss of SAT occurs
- ACB 2415 FAILS to Fast Transfer
- Bus 242Y trips on overcurrent

(1) What Attachment of LOA-AP-201, Unit 2 AC Power System Trouble will be directed by the Unit Supervisor?
(2) What Reactor Water Level Instrument is available?

A. (1) Attachment K, Station Blackout Contingencies
(2) B Narrow Range Instrument

B. (1) Attachment J, Restore Power to 242Y Using Unit Tie Breakers, 142Y Powered from SAT 142 and 142Y - 142X Bus Tie Breaker Closed
(2) B Narrow Range Instrument

C. (1) Attachment K, Station Blackout Contingencies
(2) A Narrow Range Instrument

D. (1) Attachment J, Restore Power to 242Y Using Unit Tie Breakers, 142Y Powered from SAT 142 and 142Y - 142X Bus Tie Breaker Closed
(2) A Narrow Range Instrument

Answer: A

Answer Explanation

A is correct: With a loss of SAT, the Division 1 EDG OOS and ACB 2415 failing to fast transfer, Bus 241Y will de-energize. With an over-current on 242Y it will have a lockout and de-energize. When 241Y and 242Y de-energize it will cause a loss of power to Reactor Water Level Control and the Reactor to Scram on Low Level as well as a loss of both RPS Buses, which in turn would cause a loss of the UAT. A loss of the SAT/UAT and a loss of 241Y/242Y with the Division 1 EDG OOS are entry conditions for LOA-LOOP-201. LOA-LOOP-201 is entered and following the Scram hardcard actions it is exited to enter LOA-AP-201. When LOA-AP-201 is entered and the Transient Identification is performed which determines the SAT is Dead, Attachment K should be performed. 'A' Narrow Range Reactor Level Instrument is powered by 235X-3 which is powered by 241Y. It is also auctioneered by 236X-3 which is powered by 242Y. 'B' Narrow Range Reactor Level Instrument is powered by 212X (DC) which will still have power via the Div 2 125 vDC batteries.

B is incorrect: (1) Plausible because Attachment J is entered per LOA-AP-201 if Bus 242Y has lost power and is unable to be supplied with power from the SAT, UAT, Bus Cross-Tie, or Diesel Generator, which is true. However this is incorrect because there is a decision point in the "Loss of Bus 242Y" Section that if 1PM01J-B202 (4KV Bus Overcurrent on Bus 242Y or 242X) is not clear then go to Attachment F (Actions for a Loss of 242Y)

C is incorrect: (2) 'A' Narrow Range Level Indicator is powered from 241Y and auctioneered by 242Y and will neither will have power. Plausible because most plant equipment that has an 'B' component powered by 212X the 'B' component will be powered by 211X which will still have power.

D is incorrect: (1) Plausible because Attachment J is entered per LOA-AP-201 if Bus 242Y has lost power and is unable to be supplied with power from the SAT, UAT, Bus Cross-Tie, or Diesel Generator, which is true. However this is incorrect because there is a decision point in the "Loss of Bus 242Y" Section that if 1PM01J-B202 (4KV Bus Overcurrent on Bus 242Y or 242X) is not clear then go to Attachment F (Actions for a Loss of 242Y) (2) 'A' Narrow Range Level Indicator is powered from 241Y and auctioneered by 242Y and will neither will have power. Plausible because most plant equipment that has an 'B' component powered by 212X the 'B' component will be powered by 211X which will still have power.

References: 1E-2-4200AA, 1E-2-4208AN 1E-2-4208BP, 1E-2-4208BR, 1E-2-4208AK, LOA-AP-201 Unit 2 AC Power System Abnormal Rev.52, LOA-LOOP-201 Loss of Offsite Power Rev. 6
Reference provided during examination: None
Cognitive level: High
Level (RO/SRO): SRO
Tier: 1 Group: 1
KA: 295003 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: Reactor Power / pressure / and level
10 CFR Part 55 Content: 41.10 / 43.5 / 45.13
SRO Justification: Meets 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Question Source: New
Question History: none

Comments: Associated objective(s): 640.010

77	ID: 2078936	Points: 1.00

A loss of SAT occurs on Unit 1

- 141Y tripped on overcurrent
- 1A EDG has Failed
- 142Y failed to fast transfer causing a scram
- 142Y was manually transferred within 5 minutes and is being powered by Unit 2
- HPCS is maintaining Reactor Level automatically between its high and low level injection setpoints

As the Shift Manager what is the Emergency Classification Level that should be declared for Unit 1?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: B

Answer Explanation

A is incorrect: Plausible because the criteria for MU1(Unusual Event) is met “Loss of all offsite AC power capability to emergency buses for 15 minutes or longer, however with an ALERT condition met the alert should be declared.

B is correct: With a loss of power to 141Y, a scram will occur based on the logic failure of ACB 1425 which will result in loss of RPS. When power is restored to 142Y via the Unit crossties it meets the criteria for MA1, “Loss of all but one AC power source to emergency buses for 15 minutes or longer” as explained below. The 15 minutes is not required per the note in each classification, “The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded or will likely be exceeded.”

<u>MA1 Criteria</u>
AC power capability to unit ECCS busses (excluding Division 3) reduced to only one of the following power sources for $\geq$ 15 minutes. (Other SAT via crosstie breakers) AND ANY additional single power source failure (excluding Division 3) will result in a loss of ALL AC power to SAFETY SYSTEMS.

C is incorrect: Plausible because the criteria for MS1 (Site Area Emergency) for a failure of the Division 1 and Division 2 EDGs is met and the criteria for a loss of all offsite power for Unit 1 is met but is currently not met for Unit 2. However, the criteria for a failure to restore at least one ECCS bus is currently not being met because division 2 is being powered by Unit 2.

D is incorrect: Plausible because the criteria for MG1 (General Emergency) for a failure of the Division 1 and Division 2 EDGs is met and the criteria for a loss of all offsite power for Unit 1 is met but is currently not met for Unit 2. This last criteria for restoration of power or RPV level below -150 is not met because HPCS is maintaining level between -50" and +55".

Reference: EP-AA-1005 Addendum 3 Rev 4, Emergency Action Levels for LaSalle Station
Reference provided during examination: EP-AA-1005 Addendum 3 Rev 4 (Hot Matrix), Emergency Action Levels for LaSalle Station

Cognitive level: High
10CFR Part 55: 41.7 / 43.5 / 45.12

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

PRA: No
K/A: 295006 SCRAM G.4.21Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

SRO Justification: 10 CFR 55.43(b)(5) Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. Making emergency classifications as the Shift Manager is a SRO function.

Question Source: New

Question History: N/A

Comments: The initiating condition of MA1 from the basis is: This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more
than one, train of safety-related equipment. This IC provides an escalation path from IC MU1.

Objective # 708.00.00

78	ID: 2078205	Points: 1.00

Unit 2 is in MODE 3.

2A RHR is running in shutdown cooling.

- A level transient causes a HPCS initiation.
- The 2E12-F009, RHR shutdown cooling suction header inboard Isolation valve, closes and will not re-open.

1) What reactor level causes the trip of Shut Down Cooling?
2) Which alternate decay heat removal plan should the Unit Supervisor select from OU-LA-104 Shutdown safety management?

A. 1) SDC trips on reactor level 2
2) LOP-RH-17, Alternate SDC using SRVs

B. 1) SDC trips on reactor level 3
2) LOP-RH-17, Alternate SDC using SRVs

C. 1) SDC trips on reactor level 2
2) LOP-RT-13, RWCU Alternate Decay Removal

D. 1) SDC trips on reactor level 3
2) LOP-RT-13, RWCU Alternate Decay Removal

Answer: B

Answer Explanation
A is incorrect: Plausible because level 2 is a major PCIS system isolation setpoint and given the HPCS initiation signal, level 2 has been reached. B is correct: If HPCS has an initiation signal, reactor water level will have dropped near -50", this level will exceed the group 6 isolation setpoint for low level at level 3. Given conditions given in the stem per the Shutdown safety management program steps 3.5.2 use of SRVs per LOP-RH-17 would be the correct action. C is incorrect: Plausible because level 2 is a major PCIS system isolation setpoint and given the HPCS initiation signal, level 2 has been reached and RWCU is a viable alternate shutdown cooling system but it is not available due to the level 2 isolation signal. D is incorrect: Plausible because RWCU is a viable alternate shutdown cooling system but it is not available due to the level 2 isolation signal: Reference: LOP-RH-17 ALTERNATE SHUTDOWN COOLING Rev 34, LOP-RT-13 RWCU LINEUP FOR HEAT REMOVAL Rev 19, LOP-CX-06 PRIMARY CONTAINMENT ISOLATION STATUS DISPLAY Rev 8, OU-LA-104 SHUTDOWN SAFETY MANAGEMENT PROGRAM Rev 21 Reference provided during examination: None Cognitive level: High 10CFR Part 55: 41.10, 43.5, 45.13 Level (RO/SRO): 3.5 Tier: 1 Group: 1 PRA: No K/A: 295021 A2.03 Loss of Shutdown Cooling: Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: Reactor water level. SRO Justification: 55.43.(b)(2) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations. Question Source: New Question History: none Comments: Objective: 64.00.12

79 **ID: 2079022** **Points: 1.00**

A LOCA occurred on Unit 1:

- Drywell pressure is 67 PSIG
- Containment Flood Level is 700 Feet
- Hydrogen concentration is 3%
- Oxygen concentration is 2%
- Stack WRGM reads 8.56 E+09 uCi/sec
- VG WRGM reads 1.21 E+02 uCi/sec

(1) What is the required status of containment venting per LGA-VQ-102, UNIT 1 EMERGENCY CONTAINMENT VENT?
(2) Why?

A. (1) In progress
(2) to lower pressure within PCPL

B. (1) In progress
(2) to reduce hydrogen and oxygen concentrations below deflagration limits

C. (1) NOT in progress
(2) LGA-009 entry criteria is met

D. (1) NOT in progress
(2) The WRGM high-high rad release alarm is in

Answer: A

Answer Explanation
A is correct: When venting to stay below the 60 psig in LGA-003 or the PCPL in LSAMG-001 and when venting with deflagrable hydrogen/oxygen mix in the containment, there is no limit on radiation release. No ODCM samples are ever required prior to venting. B is incorrect: Plausible LGA-011 and LGA-VQ-102 direct venting to reduce hydrogen concentration. Both procedures have you stop venting if ODCM rad limits are exceeded UNLESS there is detonable concentrations of hydrogen and oxygen in the containment. (6% hydrogen & 5% oxygen) C is incorrect: There are no rad limits imposed when lowering the primary containment pressure below PCPL. Plausible because 8.26 E+07 µCi/sec is an entry condition into LGA-009 which will direct the operator to isolate all primary system discharges outside primary and secondary containment. D is incorrect: There are no rad limits imposed when lowering the primary containment pressure below PCPL. Plausible because LGA-VQ-102 suggests securing venting upon receipt of a WRGM high-high rad release alarm. References: LGA-VQ-102 rev 1, Unit 1 Emergency Containment Vent, LGA-003 rev 17, Primary Containment Control, LGA-HG-101 rev1, Operation of the Hydrogen Recombiner As A Mixing System, LGA-009, LGA-011 rev13, Hydrogen Control, Reference provided during examination: None Cognitive level: High Level (RO/SRO): RO Tier: 1 Group: 1 KA: KA: 295024 HIGH DRYWELL PRESSURE G.2.1.20 Ability to interpret and execute procedure steps. (4.6/4.6) 10 CFR Part 55 Content: 41.10 / 43.5 / 45.12 SRO Justification: N/A Question Source: New Question History: N/A Comments: Associated objective(s): 426.000

80	ID: 2058197	Points: 1.00

The Heat Capacity Temperature Limit (HCTL) is

A. the maximum primary containment temperature at which SRVs can be opened and will remain open.

B. the highest suppression chamber temperature which can occur without steam in the suppression chamber airspace.

C. the highest suppression pool temperature at which opening an SRV will not result in exceeding the code allowable stresses in the SRV tail pipe, tail pipe supports, quencher, or quencher supports.

D. the highest suppression pool temperature at which emergency RPV depressurization will not raise Suppression chamber temperature above maximum temperature capability of the suppression chamber.

Answer: D

Answer Explanation

A s incorrect: Plausible because there is a temperature limit for SRV operation, based solenoids EQ limits, but HCTL curves are not used in this case.
B is incorrect: This is the basis of the Pressure Suppression Pressure (PSP) except temperature terminology was added to raise difficulty of question. Plausible because per LGA-003 (Primary Containment Control) if unable to maintain Containment parameters below PSP a blowdown must be performed. This is also the basis for a different graph contained on LGA-003.

C is incorrect: This is the basis of SRV Tail Pipe Limit. Plausible because per LGA-003 (Primary Containment Control) if unable to restore suppression pool level and RPV pressure below SRV Tail Pipe Level Limit, and hold it there a blowdown must be performed with the exception that it is based on suppression pool level and not pool temperature. This is also the basis for a different graph contained on LGA-003.

D is correct: The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature at which emergency RPV depressurization will not raise:

- Suppression chamber temperature above maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized, or
- Suppression chamber pressure above the Primary Containment Pressure Limit, before the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent. Depressurizing the RPV when suppression pool temperature and RPV pressure cannot be maintained below the HCTL thus avoids failure of the containment and equipment necessary for the safe shutdown of the plant

In order to maintain Suppression Chamber Air Temperature below its limits, the Heat Capacity Temperature Limit (HCTL) compares Suppression Pool Temperature to RPV Pressure in the event an emergency depressurization is required. LGA-003 requires Drywell Sprays initiated based on drywell temperature and in turn meets the requirement to maximize pool cooling to ensure HCTL is maintained below its limits. At high suppression pool and chamber air space temperatures, suppression of steam from the drywell will be hindered and will cause an increase in drywell temperature. Suppression pool air temperature rises as suppression pool temperature rises.

References: LGA-003 Primary Containment Control, LPGP-PSTG-01S05A Plant Specific Technical Guidelines Section 5a – Primary Containment Control – Entry, Suppression Pool Temperature, And Drywell Temperature Control, LPGP-PSTG-01S05B, Plant Specific Technical Guidelines Section 5b – Primary Containment Control – Primary Containment Pressure And Suppression Pool Level Control
Reference provided during examination: None
Cognitive level: Memory
Level (RO/SRO): SRO
Tier: 1 Group: 1
KA: 295028 Ability to determine and/or interpret the following as they apply to High Drywell Temperature: Torus/suppression chamber air space temperature
10 CFR Part 55 Content: 41.10 / 43.5 / 45.13
SRO Justification: The Basis for the Limitation Curves in the Emergency Operating Procedures is SRO-only knowledge at LaSalle Station. Contains knowledge of diagnostic steps and decision

points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
Question Source: New
Question History: none

Comments: All Curves are located on LGA-003 Primary Containment Control. Suppression Chamber air temperature is not used to direct EOP actions. Actions are driven from Drywell Temperature and Suppression Pool Temperature, which correlate with suppression chamber air temperature.

Associated objective(s): 421.00.04 Given plant conditions and LGA entry, recall the basis for each portion of the Heat Capacity Temperature Limit curve and identify actions when limit is exceeded, while operating the plant or on an exam, IAW the LGA procedures.

81	ID: 2058313	Points: 1.00

The Technical Requirements Manual (TRM) Basis for the Automatic Depressurization System (ADS) Manual Inhibit Switches is to prevent...

A. core damage from low pressure ECCS systems injecting at low pressures from an automatic ADS during an ATWS.

B. an automatic ADS from reducing reactor pressure below 150 psig, ensuring RCIC is available for injection during an ATWS.

C. the ADS logic from being initiated from an external hot short by limiting the voltage across the relay coil with the shorting contact.

D. ADS SRVs from opening during an Anticipated Transient Without Scram (ATWS), ensuring all energy is diverted to the Main Condenser.

Answer: A

Answer Explanation
A is correct: During an ATWS, in order to affect a reduction in reactor power under extraordinary conditions, emergency procedures may direct actions to deliberately lower RPV water level to a level below the automatic initiation setpoint of ADS. Actuation of ADS imposes a severe thermal transient on the RPV and complicates the efforts to maintain RPV water level within the ranges specified in the SAG. Further, rapid and uncontrolled injection of large amounts of relatively cold, unborated water from low pressure injection systems may occur as RPV pressure decreases to and below the shutoff heads of these pumps. Such an occurrence would quickly dilute in-core boron concentration and reduce reactor coolant temperature. When the reactor is not shutdown, or when the shutdown margin is small, sufficient positive reactivity might be added in this way to cause a reactor power excursion large enough to severely damage the core. Therefore, the ADS Manual Inhibit Function is required to purposely prevent ADS initiation under these circumstances. B is incorrect: Plausible because the ADS inhibit Switches prevent an automatic ADS from initiating which would cause reactor pressure to lower, therefore causing RCIC to isolate and become unavailable for injection. LGA-010, Failure to Scram, an RPV depressurization should be terminated if it will result in a loss of RCIC injection required for core cooling. C is incorrect: This is when the ADS DEFEAT Switches are to be used. Plausible because the ADS Defeat Switches will also prevent ADS from Initiating. D is incorrect: Plausible because the preferred heat sink during an ATWS is the Main Condenser unless there is an indication of a Main Steam Line Rupture which there is not. References: TRM ECCS Instrumentation 3.3.i Basis Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): SRO Tier: 1 Group: 1 KA: 295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or unknown 2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. 10 CFR Part 55 Content: 41.5 / 41.7 / 43.2 SRO Justification: Meets 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases Question Source: New Question History: N/A Comments: Associated objective(s): 62.00.01

82	ID: 2057961	Points: 1.00

Unit 1 and Unit 2 are at 100% RTP:

- TSO Informs you Braidwood Unit 1 is going offline
- Grid Voltage changes from 362KV to 357KV

At LaSalle on Unit 1:

- MVAR changes from 200 MVARs to 600 MVARs
- Machine Gas pressure is 60 psig
- MWe changes from 1180 MWe to 1160 MWe

The main generator is currently operating __(1)__, and the unit supervisor should direct __(2)__.

A. (1) outside the Unit 1 generator capability curve
(2) lowering U1 main generator by 50 MVARs, IAW OP-AA-108-107-1002 (Interface Procedure Between Comed and Exelon Generation for Transmission Operations)

B. (1) outside the Unit 1 generator capability curve
(2) raising machine gas pressure 15 psig, IAW LOP-HY-07 (Increasing Generator Hydrogen Pressure)

C. (1) within the Unit 1 generator capability curve
(2) documenting minimum and projected switchyard voltage IAW attachment 'C' of LOA-GRID-001 (Low Grid Voltage)

D. (1) within the Unit 1 generator capability curve
(2) contacting Gen Dispatch for an unplanned change of >10 MWe, IAW OP-AA-102-101 (Management of Nuclear Generation)

Answer: B

Answer Explanation
A is incorrect: Plausible because the conditions do place the generator outside the capability curve. Lowering MVARs does move the right direction on the curve but does not place you within the capability curve. B is correct: LOP-HY-07 has gas operability curves for Hydrogen pressure and KVARs. This new shift has the main generator operating outside of its capability curve C is incorrect: Plausible because the conditions are within the 75 psig hydrogen pressure line If the student uses the wrong side of the curve or uses the 75 psig curve line the first part of this distractor is true. Also plausible because grid voltage has lowered, however it has not reached the 353KV threshold or entry into LOA-GRID-001. D is incorrect: Plausible because the conditions are within the 75 psig hydrogen pressure line If the student uses the wrong side of the curve or uses the 75 psig curve line the first part of this distractor is true. However, it is incorrect unplanned changes in load >30MWe or longer than 30 minutes require a prompt phone call to the NDO and Gen Dispatch, per OP-AA-102-101. References: LOP-HY-07 rev 12, LOA-GRID-001 rev 15, UFSAR 8.2 Offsite Power System, OP-LA-101-111-1002 rev75 LaSalle Operations Philosophy Handbook, OP-AA-102-101 rev 15 management of nuclear generation, OP-AA-108-107-1002 rev 11, Interface Procedure Between Comed and Exelon Generation for Transmission Operations Reference provided during examination: LOP-HY-07 Unit 1 Generator Capability Curve Cognitive level: High Level (RO/SRO): SRO Tier: 1 Group: 1 KA: 700000 Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: VARs outside capability curve 10 CFR Part 55 Content: 41.5 / 43.5 / 45.5 / 45.7 / 45.8 SRO Justification: 55.43(b)(5), Assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed. Question Source: New Question History: N/A Comments: Associated objective(s): 09.00.05

83	ID: 2058026	Points: 1.00

A LOCA has occurred on Unit 1:

- Drywell Temperature is 210 °F
- Reactor Building Temperature is 145 °F
- Reactor Pressure is 600 psig and stable

Which of the following RPV Level Instruments may be used AND is also Tech Spec required Post Accident Monitoring (PAM) Instrumentation?

K RPV Level Instrument Criteria

DRYWELL TEMPERATURE (°F)	FUEL ZONE (RB TEMP ≤ 200 °F)	FUEL ZONE (RB TEMP > 200 °F)	WIDE RANGE (RB TEMP ≤ 140°F)	WIDE RANGE (RB TEMP > 140°F)	NARROW RANGE (RB TEMP ≤ 150°F)	NARROW RANGE (RB TEMP >150°F)	UPSET RANGE	SHUTDOWN RANGE
≥ 500	≥ -311 in.	≥ -310 in.	≥ -150 in.	≥ -128 in.	≥ 0 in.	≥ 5 in.	≥ 183 in.	≥ 160 in.
400 – 499	≥ -311 in.	≥ -298 in.	≥ -150 in.	≥ -120 in.	≥ 0 in.	≥ 4 in.	≥ 141 in.	≥ 130 in.
300 – 399	≥ -311 in.	≥ -283 in.	≥ -150 in.	≥ -107 in.	≥ 0 in.	≥ 10 in.	≥ 84 in.	≥ 85 in.
200 – 299	≥ -311 in.	≥ -270 in.	≥ -150 in.	≥ -97 in.	≥ 0 in.	≥ 19 in.	≥ 41 in.	≥ 50 in.
100 – 199	≥ -311 in.	≥ -261 in.	≥ -150 in.	≥ -90 in.	≥ 0 in.	≥ 25 in.	≥ 9 in.	≥ 25 in.
38 – 99	≥ -311 in.	≥ -259 in.	≥ -150 in.	≥ -88 in.	≥ 0 in.	≥ 27 in.	≥ 0 in.	≥ 10 in.

A. Wide Range reading -135 inches

B. Upset Range reading +15 inches

C. Narrow Range reading +30 inches

D. Fuel Zone Range reading -300 inches

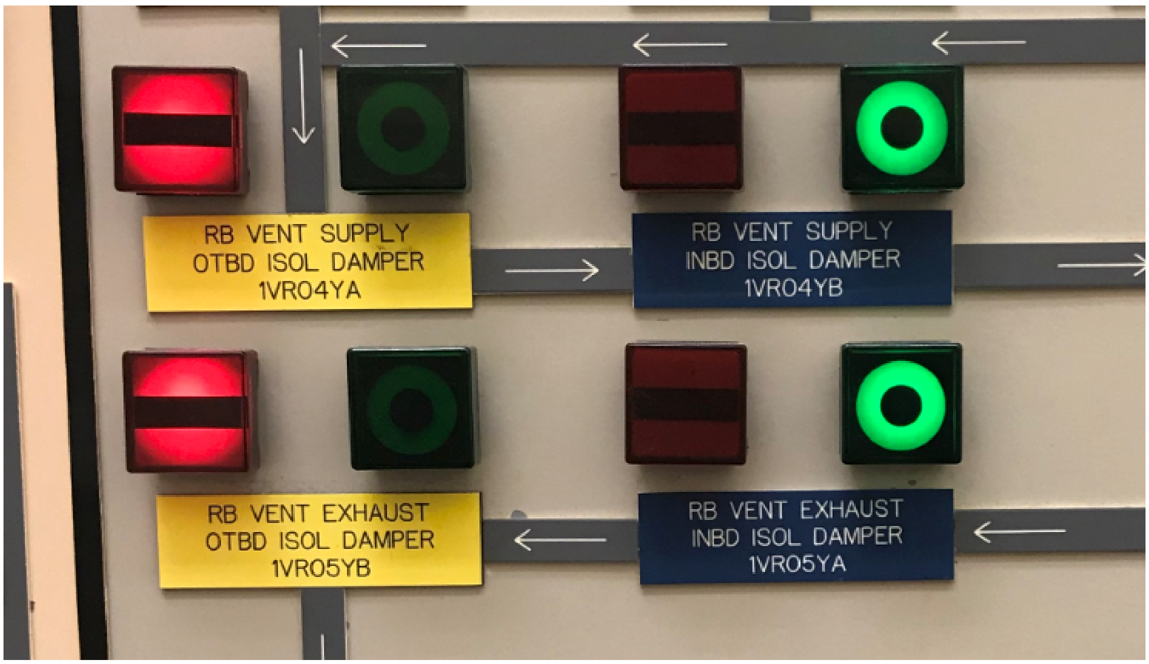
Answer: D

Answer Explanation
A is incorrect: For the given conditions Wide Range RPV Level indication is not accurate. Plausible because Wide Range Level Indication is part of Post-Accident Monitoring Instrumentation. B is incorrect: Upset Range is not part of Post-Accident Monitoring Instrumentation. Plausible because for the given conditions Upset range can be used for RPV Level Indication. C is incorrect: Narrow Range is not part of Post-Accident Monitoring Instrumentation. Plausible because for the given conditions Narrow range can be used for RPV Level Indication. D is correct: The given scenario has LGA-003 entry conditions on Drywell Temperature. The Drywell Temperature Leg gives direction to "Keep trying to lower drywell temperature below 135 F" with a finger note that says "Drywell temperature affects RPV water level indication. Check Detail I in LGA-001 or LGA-010". The given Figure K is contained within Figure I of LGA-001 and is used to determine which instruments are available. For the given conditions the only instrument that may be used AND is part of Post-Accident Monitoring Instrumentation is Fuel Zone Range. References: LGA-001 RPV Level Control, Tech Spec 3.3.3.1 and Tech Spec 3.3.3.1 Basis Reference provided during examination: None Cognitive level: High Level (RO/SRO): SRO Tier: 1 Group: 2 KA: 295012 High Drywell Temperature G2.4.3 Ability to identify post-accident instrumentation. 10 CFR Part 55 Content: 41.6 / 45.4 SRO Justification: 10 CFR 55.43(b)(5) knowledge of diagnostic steps and decision points in the emergency operating procedures (EOPs) that involve transitions to event-specific sub-procedures or emergency contingency procedures Question Source: New Question History: none Comments: Associated objective(s): 40.00.22

84	ID: 2058354	Points: 1.00

Unit 1 is at 100% RTP:

- Unit 2 SBGT is OOS



(1) What caused the above condition?
(2) Prior to restarting the VR fans, what action will be directed first by the Unit Supervisor?

A. (1) Loss of A RPS Bus
(2) Bypass Main Steam Tunnel delta T trips per LOA-VR-101, UNIT 1 RECOVERY FROM A GROUP 4 ISOLATION OR SPURIOUS TRIP OF REACTOR BUILDING VENTILATION

B. (1) Loss of 111Y
(2) Bypass Main Steam Tunnel delta T trips per LOA-VR-101 UNIT 1 RECOVERY FROM A GROUP 4 ISOLATION OR SPURIOUS TRIP OF REACTOR BUILDING VENTILATION

C. (1) Loss of A RPS Bus
(2) Secure Standby Gas Treatment per LOP-VG-02, SHUTDOWN OF THE STANDBY GAS TREATMENT SYSTEM

D. (1) Loss of 111Y
(2) Secure Standby Gas Treatment per LOP-VG-02, SHUTDOWN OF THE STANDBY GAS TREATMENT SYSTEM

Answer: B

Answer Explanation
A is incorrect: Plausible because losing A RPS is associated with DIV I and PCIS valve power. These actions are called out in the LOA-VR-101 and would restore ventilation, however these are not the next steps in the procedure. B is correct: When VR or VQ systems are started, or after a period of time with VR isolated, Main Steam Tunnel differential temperatures will cause a PCIS Group 1 Isolation and reactor scram. Differential temperature trips are bypassed on VR restart to prevent spurious automatic actions. LOA-VR-101 directs securing standby gas treatment after VR is restored and reactor building DP is confirmed more negative than -0.25” WC. LOA-DC-101 also has steps to secure VG on one of the units after evaluating proper operation of one of the systems. LOP-VG-02 has a note stating that both VG trains should not be operated at the same time do have greater than isometric flow and causes filter degradation. C is incorrect: Plausible because losing A RPS is associated with DIV I and PCIS valve power. LOP-VG-02 has a note stating that both VG trains should not be operated at the same time do have greater than isometric flow and causes filter degradation. Provides negative ventilation and rated flow for the reactor building when reactor building ventilation is not running. D is incorrect: Plausible because these actions are called out in the LOA-VR-101 and would restore ventilation, however these are not the next steps in the procedure. LOP-VG-02 has a note stating that both VG trains should not be operated at the same time do have greater than isometric flow and causes filter degradation. Provides negative ventilation and rated flow for the reactor building when reactor building ventilation is not running. References: LOP-CX-06 rev 8 1E-1(2)-4232AP, 1E-1-4232AQ, 1E-1-4441AA, LGA-VR-01, LOA-VR-101, LOP-VG-02, LOA-DC-101 Reference provided during examination: None Cognitive level: High Level (RO/SRO): SRO Tier: 1 Group: 2 KA: 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 / 43.5 / 45.13 SRO Justification: Assessment of plant conditions (normal, abnormal, emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed. Question Source: New Question History: N/A Comments: Associated objective(s): 91.00.16

85	ID: 2061969	Points: 1.00

There is a LOCA on Unit 1.

- Water level is -140” on WR and stable
- Hydrogen concentration is 3% and rising at 0.1%/min
- Oxygen concentration is 6% and rising at 0.1%/min
- Unit 1 Hydrogen Recombiner is in service on the U1 Drywell
- The primary containment is currently being vented per LGA-VQ-102, UNIT 1 EMERGENCY CONTAINMENT VENT
- Drywell pressure is 15.0 psig and continues to rise at 0.1 psig/min

The U1 Hydrogen Recombiner should be__(1)__ because the recombiner__(2)__.

A. (1) secured
(2) is a potential source for hydrogen deflagration or detonation

B. (1) secured
(2) discharge line loop seal will blow out at drywell pressures >15.3 psig

C. (1) left in service
(2) reduces hydrogen accumulation in the RX head and pedestal areas

D. (1) left in service
(2) is sized to maintain hydrogen concentration below 4% at the recombiner inlet

Answer: B

Answer Explanation
A is incorrect: plausible per LGA-011 when hydrogen concentration exceeds 6% this is the correct action and reason. B is correct: Per LGA-HG-101, The discharge line loop seal will blow out at Drywell pressures exceeding 15.3 PSIG, creating a pathway between the Drywell and the Reactor Building. C is incorrect: plausible per LGA-011 alone the hydrogen recombiner would be left running and this is the function of the recombiner per LGA-HG-101. D is incorrect: plausible per LGA-011 alone the hydrogen recombiner would be left running and this is the basis for the OG recombiner per the UFSAR. References: LGA-HG-101 rev 1, Operation Of The Hydrogen Recombiner As A Mixing System, LGA-011 rev 13, Hydrogen Control, UFSAR 11.3.2.1.1.14 rev 13 Reference provided during examination: None Cognitive level: High Level (RO/SRO): SRO Tier: 1 Group: 2 KA: 500000 G01.32 High CTMT Hydrogen Concentration: Ability to explain and apply system limits and precautions. SRO Justification:10 CFR 55.43(b)(5) Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures. Question Source: New Question History: N/A Comments: Associated objective(s): 094.00.20

86	ID: 1262298	Points: 1.00

Unit 2 is at 100% Power with NO active time clocks, when the following occurred:

- 2H13-P601-C404 RHR 2A and LPCS LINE INTEGRITY MONITOR Alarm was received.
- Local reading was confirmed to be reading 1 psid to the left of zero, indicating a problem with LPCS injection line.
- IMD was dispatched and the calibration was confirmed in tolerance for the DP switch per LOR actions.

(1) Based on the above conditions, a LPCS Core Spray piping leak/break may have occurred_____________.

(2) What is the status of the LPCS system?

A. (1) inside the shroud
(2) Requires an operability determination.

B. (1) inside the shroud
(2) Declare the system inoperable and IMMEDIATELY enter LCO 3.0.3.

C. (1) between the Reactor Pressure Vessel and the Core Shroud
(2) Requires an operability determination.

D. (1) between the Reactor Pressure Vessel and the Core Shroud
(2) Declare the system inoperable and IMMEDIATELY enter LCO 3.0.3.

Answer: C

Answer Explanation

A is incorrect: The first part of the question is incorrect, since cooling water would flow out of the break between the RPV wall and shroud and bypass the core region, part 2 of the question is correct the operability of the affected ECCS system should be evaluated for Operability to determine if the system is still capable of performing its specified function assumed in the safety analysis.

B is incorrect: Both parts of this question are wrong, the first part of the question is incorrect, since cooling water would flow out of the break between the RPV wall and shroud and bypass the core region, part 2 is wrong per basis an Operability determination is made to determine system operability.

C is correct: Per TRM 3.3.f Bases, the function of the ECCS header Differential Pressure Instrumentation is to provide an alarm to alert the Operator of a potential compromise of ECCS piping integrity internal to the RPV. The presence of this alarm may indicate that the system is not operable since cooling water would flow out of the break and bypass the core region, potentially invalidating the flow delivery assumptions in the safety analysis. If the alarm is determined to be valid, the operability of the affected ECCS system should be evaluated for Operability to determine if the system is still capable of performing its specified function assumed in the safety analysis.

D is incorrect: The first part of the question is correct, since cooling water would flow out of the break and bypass the core region, but part 2 is wrong per basis an Operability determination is made to determine system operability.

Reference: TRM 3.3.f
Reference provided during examination: none

Cognitive level: High
10CFR Part 55: 41.5 / 45.6

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

PRA: No
K/A: 209001 A2.05 Ability to (a) predict the impacts of the following on the LPCS system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequence of those abnormal conditions or operations: Core Spray Line Break

SRO Justification: 10 CFR 55.43(b)(2) (Tech Specs) Knowledge of TS bases that is required to analyze TS required actions and terminology

Question Source: Bank

Question History: LaSalle ILT 13-1 NRC Exam

Comments:

Objective # 063.00.21

87 **ID: 1259623** **Points: 1.00**

Both units are at 100% power.

The alternate RPS power supply is out of service for corrective maintenance.

- 0830 on June 1, the underfrequency protection for one of the EPMAs on the 1A RPS MG Set is declared INOPERABLE.

- 0930 on June 1, the undervoltage protection for one of the EPMAs on the 1B RPS MG Set is declared INOPERABLE.

- The RPS Alternate Power Supply cannot be restored to service

Including any extensions allowed by Technical Specifications, when must the 1A RPS MG Set EPMA be restored to OPERABLE status to prevent removing the EPMA from service OR entering a Condition requiring a shutdown?

A. 1030 on June 1

B. 0830 on June 4

C. 0930 on June 4

D. 0830 on June 5

Answer: B

Answer Explanation
A is incorrect: This starts a 72 hour clock to restore the electric power monitoring assembly. This is plausible because RA B.1 has you shutdown the inservice power supply if one or both inservice power supplies lose both EPMAs. B is correct: Condition A : One or Both in-service power supplies (1A and 1B RPS MGs) with one EPMA inoperable, therefore remove the associated MG from service in 72 hours. Separate condition Entries are not prohibited; therefore, Condition A is entered for both the A and B MG. Completion Time Extensions do not apply, therefore, the 1A EPMA must be repaired or removed from service by 0830 on June 4, or enter Condition C, which requires a shutdown. C is incorrect: At 0930 a separate condition A is entered for the 'B' EPMA. It is plausible because RA B.1 has you shutdown the inservice power supply if both EPMAs for a specific power supply are lost. 1030 would be 72 hours after the second EPMA function was lost. However you cannot reenter A.1 when the 'B' EPMA loses a function. D is incorrect: 0830 on June 5th would be 72 hours from the first entry in to RA A.1 plus a 24 hour extension for performing a surveillance requirement. This extension does not apply in this instance. References: TS 3.3.8.2, 1.3 Completion Times. Reference: TS 3.3.8.2 RPS Electric Power Monitoring with 3.3.8.2 not blanked Reference provided during examination: TS 3.3.8.2 Cognitive level: High Level (RO/SRO): SRO Tier: 2 Group: 1 KA: 212000 A2.40 Ability to apply Technical Specifications for a system. 10 CFR Part 55 Content: 41.10 / 43.2 / 43.5 / 45.3 SRO Justification: 55.43(b)(2) Question Source: Bank Question History: N/A Comments: Associated objective(s): 49.00.022

88 **ID: 2057918** **Points: 1.00**

A LOOP occurred on Unit 1 and Unit 2:

- The RCIC corner room fan trips on overcurrent.
- RCIC EQP ROOM temperature reads 185˚F and is rising.
- RCIC is online and maintaining reactor water level in band.

Unit 1 RCIC is....

A. operable but will be INOP if corner room temperature is not lowered below 150˚F within 4 hours.

B. inoperable because RB ventilation and room cooling are both lost to the RCIC corner room.

C. operable until an analysis is performed that demonstrates inoperability of the equipment.

D. inoperable until room temperature lowers below 150˚F.

Answer: B

Answer Explanation
A is incorrect: plausible because this exceeds TLCO 3.7.g. Incorrect because the bases states that exceeding these limits require analysis of operability but do not make the equipment inoperable. B is correct: The temperature limits specified in Table T3.7.g-1 do not represent the OPERABILITY limits for equipment in that area, but rather, represent the point at which the continued OPERABILITY of equipment must be reverified by analysis since the bounding assumptions of the original analysis may have been exceeded. However, with the fan trip, the corner room cooling will not be functioning as designed. With RCIC running, the room will continue to heat up, and there is no chance to get below the operability limits. C is incorrect: plausible because if room cooling and reactor building ventilation are lost to a corner room, that ECCS system is considered unavailable per UFSAR 9.2.1.1.1.g. D is incorrect: plausible because this exceeds TLCO 3.7.g. requires analysis to demonstrate continued operability of equipment. References: TRM 3.7.g, TRM B 3.7.g, UFSAR 9.2.1.1.1.g Reference provided during examination: none Cognitive level: High Level (RO/SRO): SRO Tier: 2 Group: 1 KA: 217000 A2.13 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 room cooling 10 CFR Part 55 Content: 41.5 / 45.6 SRO Justification: 55.43(b)(2) Question Source: New Question History: N/A Comments: Associated objective(s): 032.00.22

89	ID: 2058199	Points: 1.00

Unit 2 is at 100% RTP:

- 'A' Reactor Recirc Flow Control Valve Locked-up for Maintenance

Moments later

- On the DS001 Alarm Screen "TDRFP A/B Soft Trip" appears
- 2H13-P603-A409, Feedwater Control Reactor Water Level Low - Level 4 Annunciates
- 2C Motor Driven Reactor Feed Pump Starts
- Reactor Water Level is recovering

(1) What condition caused the above alarm on the DS001?
(2) What procedure will the Unit Supervisor direct as the highest PRIORITY?

A. (1) High Pressure Stop Valve and Low Pressure Stop Valve not full open and a Trip device in the tripped position
(2) LOA-FW-201, Reactor Level/Feedwater Pump Control Trouble

B. (1) Greater than 60% mismatch between expected discharge flow and actual discharge flow for 3 seconds, provided there is no mismatch between total pump flow and total header flow.
(2) LOA-FW-201, Reactor Level/Feedwater Pump Control Trouble

C. (1) High Pressure Stop Valve and Low Pressure Stop Valve not full open and a Trip device in the tripped position
(2) LOA-RR-201, Unit 2 Reactor Recirculation System Trouble

D. (1) Greater than 60% mismatch between expected discharge flow and actual discharge flow for 3 seconds, provided there is no mismatch between total pump flow and total header flow.
(2) LOA-RR-201, Unit 2 Reactor Recirculation System Trouble

Answer: D

Answer Explanation
A is incorrect: (1) These are the conditions for a TDRFP trip that is not a soft trip. Plausible because these conditions will indicate a TDRFP Trip to Reactor Water Level Control and cause the Motor Driven Reactor Feed Pump to start. (2) With Reactor Water Level Recovering and a Flow Mismatch LOA-FW-201 is not the priority. Plausible because the alarm response (LOP-FW-16) directs entry into LOA-FW-201 for the soft trip. B is incorrect: (2) With Reactor Water Level Recovering and a Flow Mismatch LOA-FW-201 is not the priority. Plausible because the alarm response (LOP-FW-16) directs entry into LOA-FW-201 for the soft trip. C is incorrect: (1) These are the conditions for a TDRFP trip that is not a soft trip. Plausible because these conditions will indicate a TDRFP Trip to Reactor Water Level Control and cause the Motor Driven Reactor Feed Pump to start. D is correct: A Soft Trip is Greater than 60% mismatch between expected discharge flow and actual discharge flow for 3 seconds, provided there is no mismatch between total pump flow and total header flow. With Level 4 and a loss of a TDRFP when Reactor Recirc Flow Control Valves (FCV) will runback. With a single FCV locked-up it will cause a loop mismatch and an entry into Tech Spec 3.4.1, therefor the priority is LOA-RR-101 to restore the mismatch. References: LOA-RR-101 Reactor Recirculation System Abnormal, LOA-FW-101 Reactor Level/Feedwater Pump Control Trouble, LOP-FW-16 1(2)DS001 Operator Station Alarm Message Interpretation, LOP-RL-01 Operation Of The Reactor Level Control System, Tech Spec 3.4.1 Recirculation Loops Operating Reference provided during examination: None Cognitive level: High Level (RO/SRO): SRO Tier: 2 Group: 1 KA: 259002 Reactor Water Level Control System 2.4.46 Ability to verify that the alarms are consistent with the plant conditions 10 CFR Part 55 Content: 41.10 / 43.5 / 45.3 / 45.12 SRO Justification: Meets 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Question Source: New Question History: none Comments: Associated objective(s):

90	ID: 2078179	Points: 1.00

Unit 1 is at 100% RTP with LOS-RI-Q5 REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM PUMP OPERABILITY, VALVE INSERVICE TESTS IN MODES 1,2,3 AND COLD QUICK START is in progress.

- When the RCIC system is started in full flow test, the following alarm is received.
- LOR-1PM01J-A409 125VDC Panel 111X/Y Panel Ground Detector Alarm.
- The Rounds Equipment Operator reports that the ground is at 115 Volts to the left.

1). What is the consequence of a large ground?
2). What actions should the Unit Supervisor direct next?

A. 1). Large ground will drain the battery and lower terminal voltage
2). Secure RCIC operation per LOS-RI-Q5

B. 1). Large ground will drain the battery and lower terminal voltage.
2). Place the backup battery charger online and isolate the on service battery charger per LOP-DC-04, 125 VDC DIV 1 GROUND LOCATION AND ISOLATION

C. 1). Large ground could energize or defeat protective relays
2). Secure RCIC operation per LOS-RI-Q5

D. 1). Large ground could energize or defeat protective relays
2). Place the backup battery charger online and isolate the on service battery charger per LOP-DC-04, 125 VDC DIV 1 GROUND LOCATION AND ISOLATION

Answer: C

Answer Explanation
A is incorrect: Plausible because LOP-DC-04 is a procedure that the Operator will direct in this scenario but isolating the on service battery charger will not be the first action taken. B is incorrect: Plausible because if the ground is large and there was high current flow to ground the battery float current may lower, but this answer is incorrect because the charger should pick up any additional load initiated by the ground condition C is correct: A large ground is a sign of potential degradation of the DC system, if a second ground were to occur then a current path would exist and the function of components of the DC system could cause system actuations or a defeat of safety functions. The SRO must direct what procedure and actions that are required to be performed next. When the alarm is received with the start of the RCIC system, a good association can be made that the ground originates in the RCIC DC logic. LOR-1PM01J-A409 directs stopping all testing in progress that can be associate with the Div. 1 DC system. D is incorrect: Plausible because if the ground is large and there was high current flow to ground the battery float current may lower, but this answer is incorrect because the charger should pick up any additional load initiated by the ground condition and because LOP-DC-04 is a procedure that the Operator will direct in this scenario but isolating the on service battery charger will not be the first action taken. References: LOR-1PM01J-A409 125VDC Panel 111X/Y Panel Ground Detector Alarm Rev 6, LOP-DC-04 125 VDC DIV 1 Ground Location and Isolation Procedure Rev 36, LOS-RI-Q5 rev 42 REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM PUMP OPERABILITY Reference provided during examination: None Cognitive level: High 10CFR Part 55:43(b)(5) Level (RO/SRO): 3.2 Tier: 2 Group: 1 KA: 263000 A2.01 Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Grounds SRO Justification: Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations 10CFR Part 55:43(b)(5) Question Source: New Question History: none Comments: Objective: 06.00.21

91 **ID: 1262134** **Points: 1.00**

Unit 1 is shutting down

- RCMS shows INSERT and WITHDRAW blocks
- RCMS shows power below the Low Power Setpoint (LPSP) and Low Power Alarm Point (LPAP) box
- 1H13-P603-A308, Rod Out Block is LIT

What is the Technical Specification Basis for the CURRENT Rod Out Block?

To prevent exceeding .....

A. 280 cal/gram during a control rod drop accident.

B. 1% plastic strain during a control rod drop accident.

C. 280 cal/gram during a control rod withdrawal accident.

D. 1% plastic strain during a control rod withdrawal accident.

Answer: A

Answer Explanation
A is correct: RWM being INOP would cause a rod withdrawal and insert block. The bases for RWM is to limit the potential amount and rate of reactivity during a CRDA to not exceed 280 cal/gm when <10% RTP. B is incorrect: Plausible because the bases of the rod out block is to protect during a rod drop accident, however the bases for RBM is to limit the control rod withdrawals if localized neutron flux exceeds a predetermined setpoint during control rod manipulations to preclude MCPR Safety Limits and the cladding 1% plastic strain fuel design limit not the RWM. C is incorrect: 280 cal/gram is correct and a rod withdrawal would add positive reactivity, but the bases of the rod out block protects against a control rod drop accident, not a rod withdrawal accident. D is incorrect: The bases for RBM is to limit the control rod withdrawals if localized neutron flux exceeds a predetermined setpoint during control rod manipulations to preclude MCPR Safety Limits and the cladding 1% plastic strain fuel design limit. Reference: B 3.3.2.1 Control Rod Block Information rev44 Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): SRO Tier: 2 Group: 2 K/A: 201002 Reactor Manual Control (RMCS) G.2.22 Knowledge of limiting conditions of operations and safety limits 10 CFR Part 55 Content: 43.2 / 41.5 / 45.2 SRO Justification: 10 CFR 55.43(b)(2) Knowledge of TS bases (3.3.3.2) that is required to analyze TS required actions and terminology. Question Source: Bank Question History: 11-01 NRC Exam LaSalle Comments: The RWM is part of the RCMS system at LaSalle.

92	ID: 2078713	Points: 1.00

Unit 1 is at 100% power:

LOR-1N62-P600-B208, OFF GAS POST-TREATMENT RADIATION TROUBLE is in alarm.

At the 1H13-P604 the ‘1A’ & ‘1B’ Off Gas Post-Treatment Rad Monitors (1D18-K601A & 1D18-K601B) fail downscale.

(1) What is the status of the 1N62-F057, Off Gas Vent Stack Discharge Valve?

(2) What is/are the ODCM required action(s)?

A. (1) OPEN
(2) Enter ODCM 12.2.2 RA D.1 and D.2 and D.3

B. (1) OPEN
(2) Enter ODCM 12.2.2 RA C.1

C. (1) CLOSED
(2) Enter ODCM 12.2.2 RA D.1 and D.2 and D.3

D. (1) CLOSED
(2) Enter ODCM 12.2.2 RA C.1

Answer: C

Answer Explanation
A is incorrect: Plausible because these are the ODCM required actions for having two channels downscale for the given plant conditions. If only one channel was inoperable, the 1N62-F057 would remain open. Also other rad monitors, such as the VC rad monitors, do not cause a trip on the downscale condition. B is incorrect: Plausible because these are the ODCM required actions for having one channel downscale for the given plant conditions. If only one channel was inoperable, the 1N62-F057 would remain open. C is correct: Two downscale signals will cause the 1N62-F057 to close. A downscale signal alone will not cause the 1N62-F057, Off Gas Vent Stack Discharge Valve to close. A downscale signal with a HI-HI-HI signal on the opposite channel would cause a closure of the 1N62-F057. The SRO is required to evaluate if this failure is a failure in the less than conservative direction and must determine that both channels are inoperable. D is incorrect: Plausible because this is the correct position for the 1N62-F057 valve for the given conditions. Also plausible because these are the ODCM required actions for having one channel downscale for the given plant conditions. References: LOR-1N62-P600-B208 rev 5, Off Gas Post-Treatment Radiation Trouble, CY-LA-170-301 rev 9, Offsite Dose Calculation Manual Reference provided during examination: ODCM 12.2.2 Cognitive level: High Level (RO/SRO): SRO Tier: 2 Group: 2 K/A:272000 A2.06 Ability to (d) predict the impacts of the following on the RADIATION MONITORING SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Downscale trips 10 CFR Part 55 Content: 41.10 / 45.13 SRO Justification: 55.43.(b)(2) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations. This is a tech spec determination which meets 10 CFR 55.43.(b)(2) Facility operating limitations in the TS and their basis. Question Source: New Question History: none Comments:

93 **ID: 2078902** **Points: 1.00**

A DBA LOCA is occuring on Unit 1:

- HPCS is injecting at 6500 gpm
- 141Y tripped on overcurrent
- Drywell and Chamber Sprays are on '1B' RHR
- '1C' RHR pump tripped on overcurrent
- RPV Pressure is 20 psig
- DW pressure is 20 psig
- RPV Water level is -220” on Fuel Zone and lowering 2"/min
- An RPV blowdown per LGA-004, RPV BLOWDOWN, has been initiated

(1) Adequate core cooling is not met because

(2) Per LGA-001, RPV CONTROL, the next action directed by the Unit Supervisor will be to

A. (1) spray flow is inadequate, and the top 1/3 of the fuel is not adequately cooled.
(2) maximize injection using preferred and alternate injection sources

B. (1) RPV water level is too low, and the top of jet pump risers are uncovered.
(2) maximize injection using preferred and alternate injection sources

C. (1) spray flow is inadequate, and the top 1/3 of the fuel is not adequately cooled.
(2) exit all LGAs and Enter all SAMGs

D. (1) RPV water level is too low, and the top of jet pump risers are uncovered.
(2) exit all LGAs and Enter all SAMGs

Answer: B

Answer Explanation
A is incorrect: Plausible because if flow was less than 6250 GPM is the flow required to ensure the top 1/3 of the fuel is cooled. Incorrect because adequate core cooling is not maintained by the having level <-210" on fuel zone, uncovering the jet pump risers which prevents the bottom of the core from being cooled. The correct action per LGA-001, following a blowdown on level, is to maximize injection with preferred and alternate systems. B is correct: For LaSalle the DBA LOCA analyses adequate core cooling means three conditions must exist. (1) First, at least one core spray pump (HPCS or LPCS) must be running at design core spray flow to cool the top one-third of the core. This design spray flow was given to LaSalle as 6250 gpm at 0 psid (RPV to DW) in GE-NE-0000-0022-8684-R2 and is also in UFSAR Section 6.3.2.2.16. (2) Second, RPV water level must be at or above the top of the jet pumps to cool the bottom two-thirds of the core (–210 in. per EC 341045 when rounded). (3) Lastly the differential pressure between the RPV and DW must be 0 psid depressurized per the UFSAR and per the GE Position Summary DRF-E22-00135-01, Long-Term Post-LOCA Adequate Cooling Requirements. Currently only prefferred water systems are being utilized, no systems from detail 'e' "alternate injection systems" are being utilized. Since level is lowering slowly, additional preferred systems such as condensate, 1B RHR, or alternate systems can be utilized to restore level. C is incorrect: Plausible because if flow was less than 6250 GPM is the flow required to ensure the top 1/3 of the fuel is cooled. Incorrect because adequate core cooling is not maintained by the having level <-210" on fuel zone, uncovering the jet pump risers which prevents the bottom of the core from being cooled. D is incorrect: Exit all LGAs and enter the SAMGs is directed by LGA-001, but only after attempting to maximize injection with alternate systems. Plausible because water level is too low and uncovering the jet pump risers which prevents the bottom of the core from being cooled. References: LGA-001 rev 18, LPGP-PSTG-01S04A rev 5, UFSAR 6.3.2.2.16 Reference provided during examination: None Cognitive level: High Level (RO/SRO): SRO Tier: 2 Group: 2 KA: 290002 Reactor Vessel Internals G4.18 Knowledge of the specific bases for EOPs. 10 CFR Part 55 Content: 41.10 / 43.1 / 45.13 SRO Justification: 55.43(b)(5) Requires assessment of plant conditions and selection of procedure with which to proceed. & Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency procedures. Question Source: New Question History: N/A

94 **ID: 2057849** **Points: 1.00**

In order to maintain an active Senior Reactor Operator (SRO) License, quarterly, the MINIMUM the License Holder must stand is...

A. five (5) twelve (12) hour shifts as the Unit Supervisor or Shift Manager.

B. four (4) twelve (12) hour shifts as the Unit Supervisor or Shift Manager.

C. five (5) twelve (12) hour shifts as Work Execution Control (WEC) Supervisor.

D. four (4) twelve (12) hour shifts as the Field Supervisor fulfilling the role of Shift Technical Advisor (STA).

Answer: A

Answer Explanation
A is correct: Per OP-AA-105-102, SRO licenses by performing the duties of Shift Manager or Unit Supervisor for a minimum of seven 8 hour or five 12 hour shifts per calendar quarter, including turnover to the next shift. The second Unit Supervisor can receive watchstanding credit because duties are analogous to the duties of the first Unit Supervisor (who is required by Technical Specifications). B is incorrect: This does not meet the 56 hour requirement. Plausible because either Unit Supervisor can be credited for License credit hours and to re-activate an SRO license 40 hours of shift duties is required. C is incorrect: WEC does not count towards SRO license hours. Plausible because this is a role normally fulfilled by a licensed SRO and it exceeds 56 hours. D is incorrect: This does not meet the 56 hours requirement and neither Field Supervisor or STA fulfill the requirements of OP-AA-105-102. Plausible because the STA position must be filled by a Licensed SRO and to re-activate an SRO license 40 hours of shift duties is required. References: OP-AA-105-102, NRC Active License Maintenance Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): SRO Tier: 3 Group: 1 KA: G.2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc. 10 CFR Part 55 Content: 43(b)(1) Conditions and limitations in the facility license. SRO Justification: Designated as a SRO Licensee responsibility at LaSalle Station to meet the requirements of 10CFR55.53(e) for SRO Licensee Holders. Question Source: New Question History: none Comments: Associated objective(s):

95 **ID: 2057673** **Points: 1.00**

A high power ATWS is in progress on Unit 2:

- Hardcard actions for ATWS are completed
- LGA-MS-201, USING MAIN CONDENSER AS A HEAT SINK IN ATWS, is in progress with the hardcard actions complete.
- MSL radiation monitors indicate a Hi/Hi radiation condition
- Off Gas post treatment rad monitor indicates Hi radiation
- Main Steam Tunnel temperature is 200 F and rising
- MSL flow is rising

What is the NEXT required action?

A. **OPEN** MSIVs to use the Main Condenser as a heat sink.

B. Verify MSIVs **REMAIN OPEN** to use the Main Condenser as a heat sink.

C. **CLOSE** MSIVs so that the Containment is used as a heat sink.

D. Verify MSIVs **REMAIN CLOSED** so that the Containment is used as a heat sink.

Answer: C

Answer Explanation

A is incorrect: The examinee may recognize that the Main Condenser is the preferred heat sink during an ATWS, but not understand that this lineup has already been established. Additionally, the examinee may not understand that with the leak into the Main Steam Tunnel, the Main Condenser is no longer the preferred heat sink and the MSIV's need to be closed to minimize the release into the Main Steam Tunnel.

B is incorrect: The examinee may recognize that the Main Condenser is the preferred heat sink during an ATWS, and understand that this lineup has already been established. However, the examinee may not understand that with the leak into the Main Steam Tunnel, the Main Condenser is no longer the preferred heat sink and the MSIV's need to be closed to minimize the release into the Main Steam Tunnel.

C is correct: During an ATWS, the Main Condenser is the preferred heat sink. This is true even with a fuel failure (as indicated with the annunciators in alarm) as the OG system processes the release better than the Standby Gas Treatment System UNLESS there is a MSL break outside of Primary Containment. However, if there is a leak in the steam tunnel (not the DW), even without any fuel failure, close the MSIVs (a) to protect the health and safety of the public and (b) to allow operators to enter and perform EOP actions in the RB & TB. With LGA-MS-201 completed, the Main Condenser is being used as a heat sink. With Main Steam Tunnel temperature at 200 F and rising, there is an indication of a leak in the Main Steam Tunnel. Therefore, the MSIV's should be closed to isolate the leak and the Primary Containment should be used as the heat sink. SBGT will be used to process the release into the Main Steam Tunnel prior to closing the MSIVs.

D is incorrect: The examinee may understand that with a leak into the Main Steam Tunnel, the Main Condenser is not the preferred heat sink, however, the stem of the question does not indicate that any isolation signals have occurred that would cause the MSIVs to be closed. Hardcard actions do not close the MSIVs.

References: LGA-MS-201 Rev 0

Reference provided during examination: None
Cognitive level: High
Level (RO/SRO): SRO
Tier: 3 Group: 3
KA: G.2.1.39 Knowledge of conservative decision making practices.
10 CFR Part 55 Content: 41.10 / 43.5 / 45.12

SRO Justification: 55.43(b)(5) Assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed. In this case, the SRO assesses plant conditions, weighing maintaining the main condenser as a heat sink during a high power ATWS against the steam rupture in the Main Steam Tunnel. The SRO must then select a different section of LGA-MS-201 (E.4) to close the MSIVs that is incongruent with the overall mitigative strategy of the procedure.

Question Source: Bank
Question History: N/A

Comments: Associated objective(s): 432.00.01

96	ID: 2062137	Points: 1.00

What weather condition would trigger an On-line risk re-evaluation in Paragon per the On-Line Work Control Process?

A. Tornado Watch

B. Wind gust of 45 mph

C. Sustained winds of 35 mph

D. Severe Thunderstorm warning

Answer: D

Answer Explanation
A is incorrect: Plausible because tornados is an example of a condition that could trigger a change in on-line risk, but the change would occur if it is a tornado warning. B is incorrect: Plausible because wind gust is an example of a condition that could trigger a change in on-line risk, but the change would occur if wind gust were ≥ 58 MPH. C is incorrect: Plausible because sustained winds is an example of a condition that could trigger a change in on-line risk, but the change would occur if ≥ 40 MPH. D is correct: Shift Operations responsibility includes reassess risk if emergent condition results in a plant configuration that has NOT been previously assessed. Per Step 4.1 of WC AA-101 ON-LINE WORK CONTROL PROCESS, a severe thunderstorm warning would be an example of an emergent condition that would warrant a risk reassessment. Reference: WC-AA-101 ON-LINE WORK CONTROL PROCESS Rev 28, LOA-TORN-001 HIGH WINDS / TORNADO Reference provided during examination: None Cognitive level: Memory 10CFR Part 55.43(b)(5) Level (RO/SRO): 3.8 Tier: Group: 3 PRA: Yes K/A: G 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. SRO Justification: Knowledge of the online work control process and making risk evaluations are SRO functions at LaSalle Station. Question Source: New Question History: none Objective # LO-TO-630 Based on your response to a variety of scenarios and related questions, Demonstrate a thorough understanding of Work Control Organization, IAW station procedures.

97	ID: 1455091	Points: 1.00

The Reactor Mode Switch is in Shutdown and Average Reactor Coolant Temperature is 165 F.

The Unit is currently in____(1)_____ and the ____(2)____ must be completed prior to Reactor Vessel Head closure bolt detensioning.

A. (1) Mode 3
(2) Mode Change Checklist From Mode 3 To Mode 4

B. (1) Mode 4
(2) Mode Change Checklist From Mode 4 To Mode 5

C. (1) Mode 4
(2) Mode Change Checklist From ‘At All Times’ To Mode 5

D. (1) Mode 5
(2) Mode Change Checklist From Mode 5 To Mode 4

Answer: B

Answer Explanation

A is incorrect: because the mode is wrong. This answer is plausible because it is a mode that is achieved with the vessel head bolts fully tensioned and the mode switch in shutdown, however reactor coolant temperature is not >200 F.

B is correct: LOP-AA-03 Attachment A is required to be done prior to entering Mode 5. Since no vessel head bolts are detensioned, with the mode switch in shutdown, coolant temperature being less than 200 F places the reactor in Mode 4. There is OPEX at Lasalle where operators did not make a mode change when required to do so because of detensioning of reactor head bolts.

C is incorrect: because the wrong checklist is used. It is plausible because the At All Times to Mode 5 checklist does have you verify the shutdown switch in the shutdown or refuel position.

D is incorrect: because the mode is wrong. The answer is plausible because the modes that are be transitioned are modes 4 and 5. Tensioning or detensioning also can change modes between 4 and 5.

References: LOP-AA-03 rev 37 Reactor Mode Changes, Tech Spec LCO 1.1 Definitions, Tech Spec LCO 1.3 Completion Times, Tech Spec LCO 1.4 Frequency
Reference provided during examination: None
Cognitive level: High
Level (RO/SRO): SRO
Tier: 3 Group: 3
KA: G2.2.35 Ability to determine Technical Specification Mode of Operation.
10 CFR Part 55 Content: 41.7 / 41.10 / 43.2 / 45.13
SRO Justification: 55.43(b)(2) TS Section 1 Use and Application Requirements This is a SRO only task at LaSalle to complete the administrative requirements (Mode Change Checklist) to change modes. The Mode change checklist consists of verifying operability with each applicable tech spec that would apply when changing modes. This is a tech spec determination which meets 10 CFR 55.43.(b)(2) Facility operating limitations in the TS and their basis.
Question Source: Significantly Modified 1262309
Question History: 13-1 NRC Exam

Comments:

1	ID: 1262309	Points

Given the following Conditions:

- Reactor Mode Switch is in Shutdown.
- Average Reactor Coolant Temperature is 205° F.
- Reactor Vessel Head Closure bolt tensioning will be complete in 2 hours.

The Unit is currently in____(1)_____ and the ____(2)____ must be completed prior to the completion of Rea
Vessel Head closure bolt tensioning.

A.	(1) MODE 5 (2) MODE 4 Checklist
B.	(1) MODE 5 (2) MODE 3 Checklist
C.	(1) MODE 4 (2) MODE 3 Checklist
D.	(1) MODE 3 (2) cooldown to less than or equal to 200 degrees F.

Answer:	B

Associated objective(s): 201.003

98	ID: 2057344	Points: 1.00

An electrical ATWS has occurred from rated conditions on Unit 1:

- ‘B’ RPS remains energized following the ATWS choreography
- ARI has FAILED to actuate
- LGA-NB-01 (Alternate Rod Insertion) Method 1 is in progress
- LGA-NB-01 Attachment 1B is complete (Scram Solenoid Fuses Removal)
- LGA-NB-01 Attachment 1E (SDV Vent and Drain Fuses Removal) has been ordered but has NOT yet been started

(1) What is the current status of the Scram Discharge Volume Vent and Drain Valves?

(2) What should be ordered NEXT if Attachment 1E is NOT successful in repositioning SDV Vent and Drain Valves?

A. (1) OPEN
(2) Evacuate unnecessary personnel from the Reactor Building

B. (1) OPEN
(2) Perform Method 6 (Single Rod Scram) of LGA-NB-01

C. (1) CLOSED
(2) Evacuate unnecessary personnel from the Reactor Building

D. (1) CLOSED
(2) Perform Method 6 (Single Rod Scram) of LGA-NB-01

Answer: A

Answer Explanation

A is correct: (1) With Attachment 1B complete the Pilot Scram Valve solenoids are de-energized in all 4 RPS groups, this repositions to the Scram Inlet and Outlet Valves for all HCU's and inserts all rods. With B RPS energized and only the Attachment 1B fuses removed, the Scram Discharge Volume Vent and Drain Pilot Valves will remain energized and open.
(2) When the SDV fuses fail to reposition the Vents and Drains this will create a path from the reactor to the RBEDT and RT Phase Separator causing Rad levels to increase. Per LGA-NB-01, if the SDV Vent and Drains remain open following Attachment 1E, the Reactor Building must be evacuated of unnecessary personnel.

Distractor B is incorrect: (1) With Attachment 1B complete the Pilot Scram Valve solenoids are de-energized in all 4 RPS groups, this repositions to the Scram inlet and Outlet Valves for all HCU's and inserts all rods. With 1B RPS energized and only the Attachment B fuses removed, the Scram Discharge Volume Vent and Drain Pilot Valves will remain energized and open.
(2) Plausible because Method 6 is an alternate method of performing a reactor scram using SRI test switches, however with Attachment 1B fuses removed the SRI test switches will not perform their function.

Distractor C is incorrect: (1) Plausible because with Attachment 1B complete a section of the scram air header is depressurized which allows the Scram Inlet and Outlet Valves to reposition. Incorrect because the SDV Vent and Drain Valves require a different section of the scram air header to depressurize in order to reposition.
(2) When a scram is inserted radiological conditions in the reactor building will be elevated, however LGA-NB-01 does not direct evacuation unless there is a failure causing a direct path from the reactor vessel to the RBEDT.

Distractor D is incorrect: (1) Plausible because with Attachment 1B complete a section of the scram air header is depressurized which allows the Scram Inlet and Outlet Valves to reposition. Incorrect because the SDV Vent and Drain Valves require a different section of the scram air header to depressurize in order to reposition.
(2) Plausible because Method 6 is an alternate method of performing a reactor scram using SRI test switches, however with Attachment 1B fuses removed the SRI test switches will not perform their function.

Reference: 1E-1-4215AB, 1E-1-4215AJ, 1E-1-4215AK, LGA-NB-01, M-100 Sheet 2
Reference provided during examination: LGA-NB-01 Attachment 1B and 1E
Cognitive level: High

Level (RO/SRO): SRO
Tier: 3 Group: 3
KA: G2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.
10 CFR Part 55 Content: 41.12 / 43.4 / 45.10
SRO Justification: 55.43(b)(5) Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps in accordance with LGA-RD-101.

Question Source: New
Question History: N/A

Comments: Associated objective(s): 413.010

EXAMINATION ANSWER KEY

17-1 NRCEXAM

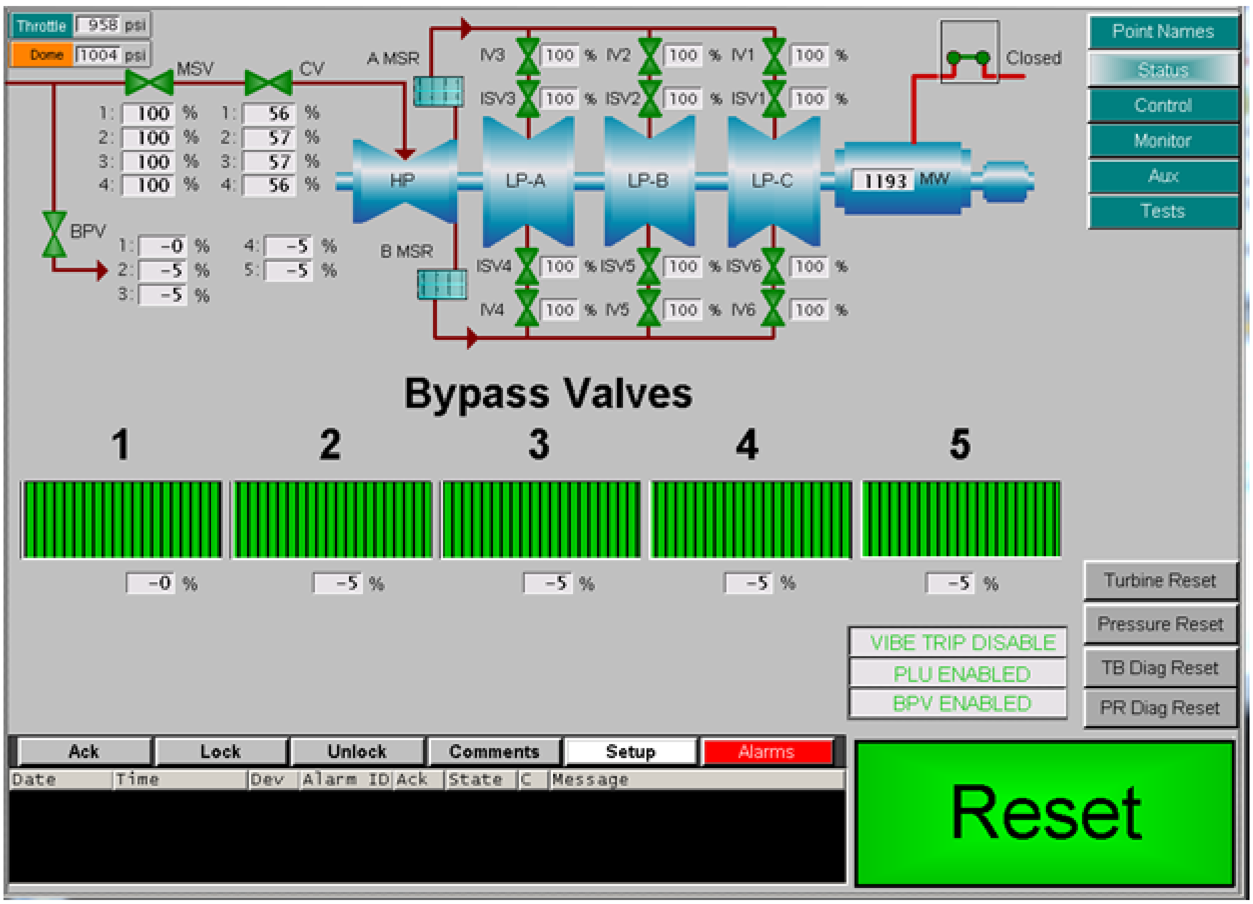
99	ID: 2057288	Points: 1.00

Unit 1 is at 98% RTP:

- Reactor Pressure is 1002 psig
- Turbine Generator is 1193 MWe

Moments later the following changes were noted...

- Average Power Range Monitors (APRM's) indicate 98.3% and are stable



The Unit Supervisor should direct the crew to enter...

A. LOA-EH-101 (Unit 1 EHC Abnormal) to reduce reactor power to less than 85% to maintain within the COLR assumptions

B. LOA-PWR-101 (Unit 1 Unplanned Reactivity Addition) to reduce Reactor power using Control Rods

C. LOA-EH-101 (Unit 1 EHC Abnormal) to check Reactor Pressure consistent with EHC Active Pressure Setpoint

D. LOA-PWR-101 (Unit 1 Unplanned Reactivity Addition) to reduce Reactor power using RR FCVs

Answer: C

Answer Explanation
A is incorrect: Plausible because LOA-EH-101 has entry conditions and a problem exists with the EHC pressure controller. However, the condition to lower power to less than 85% is only for a failed closed Control Valve/Stop Valve for COLR assumptions and the stem indicates that TCVs are ~57% and TSVs are 100% open. B is incorrect: Plausible because LOA-PWR-101 has entry conditions for the power spike and pressure increase and only control rods are used if FCL is greater than 113.2%. However, while in LOA-PWR-101, power may only be lowered if: 1) Reactor Power is greater than 100% 2) Reactor Pressure is not less than 1005 psig 3) Outside the allowed region of the Power to Flow Map 4) Reactor Power is still increasing C is correct: Reactor Pressure at full power operations is maintained at 1002 psig by the EHC System. For the given conditions the only way that Reactor Pressure would be above its setpoint would be a malfunction of EHC which in turn would cause an increase in Reactor Power. An entry condition for LOA-PWR-101 is an unplanned increase in Generator MW Electric and for LOA EH-101 is an unexpected change in Main Generator MW Electric loading. D is incorrect: Plausible because LOA-PWR-101 has entry conditions for the power spike and pressure increase and RR FCVs are the quickest method of power reduction. However, while in LOA-PWR-101, power may only be lowered if: 1) Reactor Power is greater than 100% 2) Reactor Pressure is not less than 1005 psig 3) Outside the allowed region of the Power to Flow Map 4) Reactor Power is still increasing References: LOA-EH-101, Unit 1 EHC Abnormal Rev. 35; LOA-PWR-101, Unit 1 Unplanned Reactivity Addition Rev. 14 Reference provided during examination: None Cognitive level: High Level (RO/SRO): SRO Tier: 3 Group: 4 KA: G2.4.11 Knowledge of abnormal condition procedures. 10 CFR Part 55 Content: 41.10 / 43.5 / 45.13 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: none Comments: None Associated objective(s): 614.010

100	ID: 2058269	Points: 1.00

Which of the following must the Work Execution Center perform when terminating a fire impairment?

1) Verify the condition causing the fire watch is corrected
2) Notify supervisor in charge of the fire watch that it may be terminated
3) Retain a copy of the completed fire watch log sheets in the WEC for 1 year
4) Notify NEIL of any fire pump or suppression system impairments that exceeded 48 hours

A. 1 & 2

B. 1 & 3

C. 2 & 4

D. 3 & 4

Answer: A

Answer Explanation
A is correct: OP-MW-201-007 states: 4.7.1 Shift management shall: 1. Verify the condition causing the fire watch has been corrected (i.e., returned to operable service). 2. Inform the supervisor in charge of the fire watch may be terminated. B is incorrect: Notify supervisor in charge of the fire watch that it may be terminated is required, however retain a copy of the completed fire watch log sheets in the WEC for 1 year is a duty of the fire marshal. C is incorrect: Plausible because 2 is correct, however notifying NEIL of any fire suppression impairments that exceed 48 hours is a Fire Marshal Duty. D is incorrect: Plausible because 3 and 4 are fire marshal responsibilities listed in OP-MW-201-007. References: OP-MW-201-107, Fire Protection Impairment Control Reference provided during examination: None Cognitive level: Memory Level (RO/SRO): SRO Tier: 3 Group: N/A KA: 2.4.42 Knowledge of fire protection procedures 10 CFR Part 55 Content: 41.10 / 43.5 / 45.13 SRO Justification: Closing fire impairments is an SRO Duty at LaSalle Station. Question Source: New Question History: N/A Comments: Associated objective(s): 677.010 Given the proper procedure and a set of conditions, terminate a fire watch, IAW station procedures.