Test Answer Key

2023 ILT NRC RO Exam Final Version
Test ID: 369039

1 **ID: 2474300** **Points: 1.00**

Unit 1 was operating at 93% thermal power when the following occurred:

- 1A RECIRC PUMP tripped
- Power has stabilized at approximately 60% thermal power

What is the Technical Specification SAFETY LIMIT for MCPR in this condition?

A. 1.08

B. 1.10

C. 1.18

D. 1.43

Answer: B

Answer Explanation

Answer Explanation: Reactor Core Safety Limit 2.1.1.2 states: 2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:
For two recirculation loop operation, MCPR shall be ≥ 1.08, or for single recirculation loop operation, MCPR shall be ≥ 1.10.

Distractor 1: Plausible because 1.08 is the Dual Loop MCPR Safety Limit. Incorrect because the a recirculation pump is tripped and the Single Loop MCPR Safety Limit now applies.

Distractor 2: Plausible because 1.18 is the ATRIUM 10XM Interior Fuel Bundles MCPRf Limit for Base Case 108% rated flow. Incorrect because it is an Operating Limit and not the MCPR Safety Limit.

Distractor 3: Plausible because 1.43 is the ATRIUM 10XM Interior Fuel Bundles Dual Loop MCPRp Operating Limit. Incorrect because it is the MCPR Operating Limit and not the MCPR Safety Limit.

Reference: TS Section 2.0 Safety Limits, Amendment 276/271
Reference provided during examination: None

Cognitive level: Memory

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295001AK1.03 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the (APE 1) PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Thermal limits.

Importance: 4.1
10 CFR Part 55 Content: 41.8 to 41.10

Question Source: New
Question History: N/A
Comments: None

2 **ID: 2474306** **Points: 1.00**

Unit 1 is at 100% power.

Which of the following Unit 1 transients would require OPERATOR ACTION to complete a RWCU isolation?

A. Sustained loss of the Essential Service Bus.

B. Failure of SBLC to initiate due to a Squib Valve failure.

C. Loss of Bus 13 with a failure of the 1/2 EDG to autostart.

D. Reactor scram with a concurrent loss of 250 VDC Bus 1B.

Answer: C

Answer Explanation

Answer Explanation: The following conditions cause an automatic initiation of a Group 3 isolation: (1) 0 inches RPV water level, (2) RWCU area high temperature 165°F, and (3) Steam Tunnel high temperature 165°F. The following conditions cause an automatic initiation of a RWCU
System isolation: (1) SLC System initiation and (2) RWCU non-regenerative heat exchanger high outlet temperature. The Group III isolation is an automatic closure of MO 1(2)-1201-2, MO 1(2)-1201-5, and MO 1(2)-1201-80 valves. A loss of Bus 13 results in a loss of Bus 18 and RPS Bus A. RPS A supplies power to PCI relays for a Group III isolation that when deenergized will result in a RWCU isolation. The concurrent 1/2 EDG failure prevents the MO 1-1201-2 valve from closing as it is powered from MCC 18-1A. Manually starting the 1/2 EDG will reenergize Bus 18 and MCC 18-1A allowing MO 1-1201-2 valve to close and complete the isolation.

Distractor 1: Plausible because the RWCU valve isolation logic is the powered by the ESS Bus. A sustained loss of the ESS Bus will cause a RWCU isolation. Incorrect because no operator action will be required to complete the isolation.

Distractor 2: Plausible because initiation of SBLC system will isolate the RWCU system. Incorrect because the failure was due to the Squib Valve circuit. The SBLC control switch functioned by supplying the initiation signals to the Squib Valves and RWCU system therefore the RWCU isolation would occur automatically.

Distractor 3: Plausible because a scram from rated power will result in a low RPV water level and the MO 1-1201-5 valve is powered from 250 VDC. Incorrect because the power supply to the MO 1-1201-5 valve is 250 VDC Bus 1A.

Reference: 4E-1503A Rev.AX, 4E-1503B Rev.BE, 4E 1507 Rev.AF, QOA 7000-01 Rev.40
Reference provided during examination: None

Cognitive level: High

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295003AK3.06 Knowledge of the reasons for the following responses or actions as they apply to PARTIAL OR COMPLETE LOSS OF AC POWER: Containment isolation

Importance: 3.5
10 CFR Part 55 Content: 41.5/45.6

Question Source: New
Question History: N/A
Comments: None

3 **ID: 2474307** **Points: 1.00**

Unit 2 is at 100% Reactor power when 125 VDC Reactor Building Distribution Panel 2 is lost.

What action, if any, is required to restore control power to 4KV Bus 23-1?

A. Crosstie Bus 13-1 and Bus 23-1 control power per QCOS 6900-07, OPERATING CYCLE 125 VDC CROSSTIE OF BUSES 13-1 AND 23-1 FOR APPENDIX R.

B. Transfer Bus 23-1 control power feed to Turbine Building 125 VDC Bus 2B-1 per QCOP 6900-47, TRANSFERRING UNIT 2 4 KV BUS 125 VDC CONTROL POWER FEEDS.

C. NO action is necessary; Bus 23-1 control power auto-transfers to 125 VDC SBO Distribution Panel 6A-1 via the Auto Transfer Switch (ATS).

D. NO action is necessary; Bus 23-1 control power auto-transfers to 125 VDC SBO Distribution Panel 7A-1 via the Auto Transfer Switch (ATS).

Answer: D

Answer Explanation

Answer Explanation: Per QOA 6900-13, LOSS OF POWER TO REACTOR BUILDING DISTRIBUTION PANEL 2, control power will automatically transfer to the SBO Bus 7A-1. The Auto Transfer Switch is shown on Electrical Drawing 4E-2318B.

Distractor 1: Plausible because the procedure provides direction to supply 125 VDC control power from Bus 13-1. Incorrect because this is only done as a surveillance for testing or in the event of an Appendix R fire.

Distractor 2**:** Plausible because QCOP 6900-47 provides procedural guidance to supply Bus 23-1 control power from its reserve feed at TURB BLDG DIST PNL 2B-1. Incorrect because control power is automatically restored via the ATS from Distribution Panel 7A-1.

Distractor 3: Plausible because control power will automatically transfer to the SBO Battery. Incorrect because SBO Distribution Panel 6A-1 is supplied by the Unit 1 SBO Battery which will supply Bus 13-1 with control power, not Bus 23-1.

Reference: QOA 6900-13 Rev.22, 4E-2318B Rev.J
Reference provided during examination: None

Cognitive level: Memory

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295004AA2.04 Ability to determine or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: System lineups

Importance: RO 3.2
10 CFR Part 55 Content: 41.10/43.5/45.13

Question Source: Modified
Question History: 2012 ILT NRC Exam Q.3
Comments: Question significantly modified by changing the stem so that one of the distractors is now the correct answer. Question stem previously read: “What action (if any) is required to restore control power to 4KV Bus 23-1 from a safety related power supply?”

4 **ID: 2474308** **Points: 1.00**

Unit 2 is at 35% power when a transient occurs. Moments later the NSO reports:

- Feedwater temperature lowering
- Reactor power slowly rising

Which of the events would cause this trend?

A. Main Turbine trip

B. B MSIV fails closed

C. 1A Recirc Pump run up

D. Control rod H-14 drifts out

Answer: A

Answer Explanation

Answer Explanation: Feedwater temperature rises with increasing reactor power if extraction steam is available. From 25% to 100% power, feedwater temperature will rise approximately 90°F. On a Main Turbine trip, Extraction Steam valves close and Extraction Bypass valves open. The loss of feedwater heating results in a cold water transient which causes reactor power to rise.

Distractor 1: Plausible because MSIV closure will cause a momentary void collapse resulting in a small power spike. Incorrect because feedwater heating is unaffected at this power level and feedwater temperature remains stable.

Distractor 2: Plausible because a speed increase in one or both Recirc pumps will raise power level. Incorrect because raising power, increases extraction steam flow and raises feedwater temperature.

Distractor 3: Plausible because a control rod drifting out of the core raises reactor power. Incorrect because raising power, increases extraction steam flow and raises feedwater temperature.

Reference: QOA 5600-04 Rev.31, QCOA 0400-01 Rev.26
Reference provided during examination: None

Cognitive level: Memory

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295005AA2.06 Ability to determine or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP: Feedwater temperature

Importance: RO 3.5
10 CFR Part 55 Content: 41.10/43.5/45.13

Question Source: New
Question History: N/A
Comments: None

5 **ID: 2474320** **Points: 1.00**

Unit 1 was at 100% power when APRM 3 failed upscale.
Solenoid Groups 2 and 3 on RPS Channel A did NOT de-energize.

What is the FIRST required procedure to enter?

A. QCOA 0500-01, PARTIAL SCRAM ACTUATION

B. QCOP 0500-04, INSERTING MANUAL SCRAMS

C. QCOS 0500-02, MANUAL SCRAM INSTRUMENTATION FUNCTIONAL TEST

D. QCOP 0700-04, AVERAGE POWER RANGE MONITORING SYSTEM OPERATION

Answer: A

Answer Explanation

Answer Explanation: The appropriate procedure block to enter are the operating abnormals, ie. QOAs or QCOAs. The required system actions did not occur starting with the failure of APRM #3 then the incomplete RPS Channel A 1/2 scram. QCOA 0500-01, PARTIAL SCRAM ACTUATION step D.1. states: IF partial actuation of a "Half Scram" exists, THEN insert a manual “Half Scram” by depressing the associated RPS Trip System RX SCRAM CH A OR RX SCRAM CH B pushbutton.

Distractor 1: Plausible because the procedure provides guidance on inserting scrams. Incorrect because the QCOP provides guidance for inserting full scrams during shutdown or 1/2 scrams in any mode during normal operation.

Distractor 2: Plausible because the procedure provides guidance for inserting 1/2 scrams. Incorrect because it is a procedure for a logic system functional test required by Technical Specification 3.3.1.1.

Distractor 3: Plausible because APRM #3 has failed will require bypassing it per QCOP 0700-04. Incorrect because the higher priority is to complete the required automatic action of an RPS Channel A 1/2 scram.

Reference: QCOA 0500-01 Rev.08
Reference provided during examination: None

Cognitive level: High

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295006 G2.4.14 SCRAM: EMERGENCY PROCEDURES/PLAN: Knowledge of general guidelines for emergency and abnormal operating procedures usage.

Importance: RO 3.8
10 CFR Part 55 Content: 41.10/43.1/45.13

Question Source: New
Question History: N/A
Comments: None

6 **ID: 2487698** **Points: 1.00**

A control room evacuation is required due to hazardous atmosphere in the Control Room.

The Unit NSO will relocate to the __(1)__.

FWRV operational duties will be transferred to the _(2)_.

A. (1) Aux Electric Room
(2) Unit Assist NSO

B. (1) Aux Electric Room
(2) Unit Equipment Operator

C. (1) Reactor Building 2nd Floor
(2) Unit Equipment Operator

D. (1) Reactor Building 2nd Floor
(2) Unit Assist NSO

Answer: C

Answer Explanation

Answer Explanation: Per QOA 0010-05, PLANT OPERATION WITH THE CONTROL ROOM INACCESSIBLE, step D.5. designates the following reporting locations and assignments: "Shift Manager (SM), Shift Technical Advisor (STA), and the on-shift Communicator to the TSC. The Unit Supervisor and Unit NSO report to the 2201(2)-5 AND 2201(2)-6 Instrument Racks on the Reactor Building 2nd floor. The Assist NSO is stationed in the Aux Electric Room. Step D.5.b allows the STA to be relocated after the TSC has assumed command and control. Step D.5.c directs the Unit EO to buses 11(21) and 12(22) and FWRVs. The Unit assist NSOs report to the Aux Electrical Room and DISPATCH EOs as necessary per note in Step D.5.d.

Distractor 1: Plausible because the Unit Assist NSO reports to the Aux Electric Room. Incorrect because the FWRVs are operated locally by the UNIT EO not the Assist NSO and additionally, the UNIT NSO reports to the Reactor building 2nd floor.

Distractor 2: Plausible because the Unit EO does operate the FWRVs. Incorrect because the Unit NSO reports to Reactor Building 2nd Floor, not the Aux Electric Room.

Distractor 3: Plausible because the Unit NSO relocates to the Reactor Building 2nd Floor. Incorrect because the Unit Assist NSO does not operate the FWRVs.

Reference: QOA 0010-05 Rev.26
Reference provided during examination: None

Cognitive level: Memory

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295016 AA2.08 Ability to determine or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Successful transfer

Importance: RO 4.3
10 CFR Part 55 Content: 41.10/43.5/45.13

Question Source: New
Question History: N/A
Comments: None

7 **ID: 2481352** **Points: 1.00**

A Loss of Service Water has occurred on Unit 1 and Unit 2.
The Safe Shutdown Makeup Pump (SSMP) is injecting at 400 gpm to Unit 1.

Which of the following systems can be used for alternate cooling to the SSMP Room Cooler?

A. RBCCW

B. TBCCW

C. Fire Header

D. Clean Demin

Answer: C

Answer Explanation

Answer Explanation: The SSMP Room Cooler maintains room temperature during operation. The normal cooling water supply to the room cooler is Service Water. Procedural guidance to align the fire main water to cooler is provided in QCOP 2900-02, SAFE SHUTDOWN MAKEUP PUMP SYSTEM STARTUP. QGA 100, RPV CONTROL references QCOP 2900-02, SAFE SHUTDOWN MAKEUP PUMP SYSTEM STARTUP for injection which provides steps for room cooler operation and swapping to alternate cooling for use in an emergency situation.

Distractor 1: Plausible because this a cooling water supply for compressors and pumps. Incorrect because RBCCW is the component cooling water for Reactor Building systems and would be unavailable with a loss of Service Water.
Distractor 2: Plausible because TBCCW is a cooling water supply for equipment in the Turbine Building. Incorrect because TBCCW is not the backup cooling water supply for the SSMP Room Cooler.
Distractor 3: Plausible because it is a high quality water used in various plant systems. Incorrect because it is used for makeup water, not component cooling.
Reference: QCOP 2900-01, Rev.42
Reference provided during examination: None

Cognitive level: Memory

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295018 AK3.07 Knowledge of the reasons for the following responses or actions as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Cross-connecting with backup systems

Importance: 3.1
10 CFR Part 55 Content: 41.5/45.6

Question Source: New
Question History: N/A
Comments: None

8 **ID: 2474338** **Points: 1.00**

Unit 2 is operating at rated power when a leak results in a loss of Instrument Air to the Low Pressure Heater Bay.

(1) Which of the following plant components will be affected?
(2) What, if any, is the operational impact?

A. (1) RFP Min Flow valves FAIL CLOSED, (2) with no effect on steady state plant operation.

B. (1) SJAE Suction valves FAIL CLOSED, (2) resulting in a loss of Main Condenser vacuum.

C. (1) Feedwater Heater Emergency LCVs FAIL CLOSED, (2) resulting in a positive reactivity addition.

D. (1) Extraction Steam Bypass valves FAIL CLOSED, (2) with no effect on steady state plant operation.

Answer: B

Answer Explanation

Answer Explanation: The Steam Jet Air Ejector (SJAE) Suction valves are air operated valves that fail closed on a loss of Instrument Air. QOA 4700-06, LOSS OF INSTRUMENT AIR directs the NSO to monitor Main Condenser backpressure AND Off-Gas flow for an Off-Gas System isolation.

Distractor 1: Plausible because the RFP Min Flow valves air operated valves supplied by the Instrument Air system. Incorrect because the valves fail open and would require a response by the Feed Reg valves to maintain RPV water level.

Distractor 2: Plausible because the Emergency LCVs are normally closed during steady state operation. Incorrect because they fail open on a loss of Instrument Air. During normal operation this would lower Heater Drain levels and Feedwater Heater outlet temperature.

Distractor 3: Plausible because the Extraction Steam Bypass valves are closed during steady state operation. Incorrect because they fail open on a loss of Instrument Air causing a reduction of Feedwater heating during normal operation.

Reference: QOA 4700-06 Rev.29
Reference provided during examination: None

Cognitive level: High

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295019 AK2.06 Knowledge of the relationship between the PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following systems or components: Offgas system

Importance: 3.2
10 CFR Part 55 Content: 41.7/45.8

Question Source: New
Question History: N/A
Comments: None

9 **ID: 2481636** **Points: 1.00**

Unit 2 is shutdown.
Shutdown Cooling has just been placed in operation on the 2A RHR Loop.

Moments later the NSO reports:
- The 2A RHR pump has tripped and closes MO 2-1001-29A, INBD LPCI INJ VLV.
- RPV pressure is 120 psig.

What actions are required to maintain RPV water temperature < 212°F?

A. Start Shutdown Cooling on the B RHR Loop ONLY.

B. Restart Shutdown Cooling using the 2B RHR pump ONLY.

C. Restart the 2A RHR pump and re-open MO 2-1001-29A, INBD LPCI INJ VLV.

D. Lower RPV pressure to <100 psig and restart shutdown cooling using the 2B RHR pump.

Answer: D

Answer Explanation

Answer Explanation: RPV pressure must be lowered to < 100 psig or until alarm 901-3 E-15 is received because MO 2-1001-29A, INBD LPCI INJ VLV is interlocked closed at RPV pressure ≥ 100 psig. Shutdown Cooling can then be restarted on either RHR Loop. The most efficient restoration is to start the 2B RHR pump instead of realigning the B RHR Loop.
The RHR Discharge valve (MO 1-1001-29A/B) is closed on a pump trip and/or pump shutdown to maintain the system fill, thereby preventing water hammer on a restart. This is in QCOP 1000-05 step F.6.a, for removing a SDC loop from operation.

Distractor 1: Plausible because starting shutdown cooling on the B RHR Loop will bring the reactor to cold shutdown. Incorrect because RPV pressure must also be lowered to clear RHR valve interlocks.

Distractor 2: Plausible because the 2B RHR pump is available and is the most time efficient method to restart shutdown cooling. Incorrect because RPV pressure must also be lowered to clear RHR valve interlocks.

Distractor 3: Plausible because restarting an RHR pump and opening the MO 2-1001-29A valve will restore shutdown cooling. Incorrect because RPV pressure must also be lowered to clear RHR valve interlocks.

Reference: QCOP 1000-05 Rev.64, QCAN 901(2)-3 E-15 Rev.08
Reference provided during examination: None

Cognitive level: High

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295021 G2.2.44 LOSS OF SHUTDOWN COOLING: EQUIPMENT CONTROL: 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.

Importance: RO 4.2
10 CFR Part 55 Content: 41.5/43.5/45.12

Question Source: Bank
Question History: 2009 ILT NRC Exam
Comments: None

10 **ID: 2474339** **Points: 1.00**

Three days into the Unit 1 refueling outage a fuel assembly is dropped and damaged in the fuel pool during a core offload.

Why is it possible that no radiation alarms may be received from this event?

A. The major part of the release will be alpha radiation which will not affect gamma detectors.

B. The major part of the release will be beta radiation which will not affect gamma detectors.

C. The fuel pool is designed to absorb the release of radioactive material resulting from a fuel drop accident.

D. Enough time has elapsed since shutdown to eliminate the possibility of releasing any airborne activity.

Answer: B

Answer Explanation

Answer Explanation: QCFHP 0110-04, NEW/IRRADIATED FUEL DAMAGE, contains the following CAUTION statement regarding the radioactive release from a damaged fuel bundle: "Refuel Floor Rad Monitors and CAMS may not alarm if the fission product released is the beta emitting isotope Krypton-85." This is because the Refuel Floor and Fuel Pool Radiation monitors are gamma detectors and therefore not sensitive to beta radiation.

Distractor 1: Plausible because alpha radiation will not effect gamma detectors. Incorrect because the radiation released from a damaged fuel bundle is comprised of beta and gamma components, not alpha.

Distractor 2: Plausible because the required water level above irradiated fuel is designed to contain/absorb water soluble fission product gases. Incorrect because there will be non-soluble gasses released.

Distractor 3: Plausible because time after shutdown will reduce the amount of radioactivity reduction. Incorrect because a significant amount of radioactivity is still present.

Reference: QCFHP 0110-04 Rev.05
Reference provided during examination: None

Cognitive level: Memory

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295023 AA2.01 Ability to determine or interpret the following as they apply to REFUELING ACCIDENTS: Radiation levels

Importance: RO 4.1
10 CFR Part 55 Content: 41.10/43.5/45.13

Question Source: New
Question History: N/A
Comments: None

11 **ID: 2474378** **Points: 1.00**

Following a LOCA on Unit 2, the following conditions exist:

- Drywell pressure is 7.5 psig and RISING
- Drywell temperature is 265ºF and RISING
- Torus pressure is 6.0 psig and RISING

Which of the following procedures does the NSO need to perform mitigating actions for, given the present conditions?

A. QCOP 5750-19, DRYWELL COOLER OPERATION

B. QCOP 1000-30, POST-ACCIDENT RHR OPERATION

C. QCOP 1600-02, TORUS PRESSURE RELIEF THROUGH SBGT

D. QCOP 1600-13, POST ACCIDENT VENTING OF THE PRIMARY CONTAINMENT

Answer: B

Answer Explanation

Answer Explanation: Starting Drywell Sprays is used to reduce and Drywell temperature per QGA 200 and is the priority under the present plant conditions. The guidance is provided in QCOP 1000-30.

Distractor 1: Plausible because restarting Drywell cooling is a mitigation strategy in the Drywell Temperature leg of QGA 200. Incorrect because restoring RBCCW flow to the Drywell Coolers is prohibited with containment temperature > 260ºF.

Distractor 2: Plausible because Torus venting through the SBGT is a mitigation strategy that is used in QGA 200 to hold Torus and Drywell pressure below 2.5 psig. Incorrect because QCOP 1600-02 does not contain steps to bypass the Group II Isolation (DW pressure >2.5 psig).

Distractor 3: Plausible because venting containment is a mitigation strategy in the Primary Containment Pressure leg of QGA 200. Incorrect because this procedure is used to vent containment when needed to stay below PCPL.

Reference: QGA 200 Rev.13, QCOP 1000-30 Rev.33
Reference provided during examination: None

Cognitive level: High

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295024 G2.4.39 HIGH DRYWELL PRESSURE: EMERGENCY PROCEDURES/PLAN: Knowledge of RO responsibilities in emergency plan implementing procedures.

Importance: RO 3.9
10 CFR Part 55 Content: 41.10/45.11

Question Source: New
Question History: N/A
Comments: None

12 **ID: 2487330** **Points: 1.00**

Unit 1 was operating at 100% power when a spurious Group I isolation occurred.

The Unit Supervisor directed an RPV pressure band of 800 to 1000 psig using relief valves.

The ANSO manually operated ERV 1-0203-3B lowering RPV pressure from 1000 to 825 psig.

- Reactor pressure is now 997 psig and RISING.

Which ERV will be used to LOWER RPV pressure (1), and why (2)?

A. (1) ANY ERV EXCEPT 1-0203-3A
(2) to ensure ability to perform an RPV Blowdown.

B. (1) 1-0203-3E
(2) to use a valve equipped with automatic reopening protection.

C. (1) 1-0203-3C
(2) to use a valve equipped with automatic reopening protection.

D. (1) 1-0203-3D
(2) to evenly disburse the heat load in the Torus.

Answer: C

Answer Explanation

Answer Explanation: QCOP 0203-01, step F.2 directs alternating use of B and C if repeated cycling of relief valves is required. ERV B and C are the only valves with auto reopening protection. If the ERV is reopened quickly the water drawn into the relief valve discharge line from the condensed steam creating a vacuum would be shot out creating possible torus structural damage (see QCOP 0203-01 step B.1).

Distractor 1: Plausible because it is one of the five relief valves that can be used for pressure relief. Incorrect because it is the last in order of manual operation due its air operated actuation system, and if used, should remain continuously open while cycling the other relief valves.

Distractor 2: Plausible because it is one of the five relief valves that can be used for pressure relief. Incorrect because the 3E relief valve does not have automatic reopening protection (14 second delay).

Distractor 3: Plausible because it is one of the five relief valves that can be used for pressure relief and additionally would disperse the heat load in the Torus if operated after the B relief valve. Incorrect because the D relief valve does not have automatic reopening protection.

Reference: QCOP 0203-01 Rev.17
Reference provided during examination: None

Cognitive level: Memory

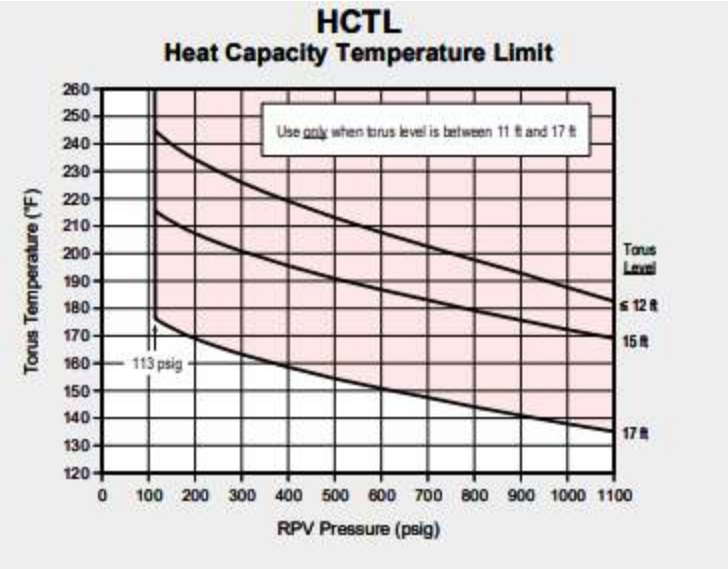
L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295025 EA1.03 Ability to operate or monitor the following as they apply to HIGH REACTOR PRESSURE: Safety/relief valves.

Importance: 4.4
10 CFR Part 55 Content: 41.7 / 45.6

Question Source: Bank
Question History: ILT 12-1 COMP, ILT 09-01 COMP
Comments: Bank question modified to a (1) / (2) style format for clarity. NO OTHER CHANGES MADE.

13 **ID: 2479638** **Points: 1.00**



Complete the following statement describing the Heat Capacity Temperature Limit (HCTL) curve.

The HCTL curve is ...

A. used to maintain the suppression pool boundary design load if SRVs are opened.

B. used to avoid containment failure or deinertion following the initiation of Drywell sprays.

C. used to preserve primary containment integrity, containment vent valve and ADS valve operability.

D. used to determine the highest Torus temperature from which a blowdown will not raise Torus pressure above the Primary Containment Pressure Limit.

Answer: D

Answer Explanation

Answer Explanation: From EPG/SAG Rev. 4 and its associated bases, the definition of the Heat Capacity Temperature Limit curve is:

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

- Suppression chamber temperature above the 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 within the capacity of the containment vent.

It is vital to maintain Torus temperature below this threshold during emergencies to preserve primary containment and continue safe operation of the plant.

Distractor 1: Plausible because it describes a QGA 200 curve based on Torus level and pressure. Incorrect because it describes the Pressure Suppression Pressure (PSP) curve.

Distractor 2: Plausible because it describes a QGA 200 curve used to protect containment. Incorrect because it describes the Drywell Spray Initiation Limit (DSIL) curve.

Distractor 3: Plausible because it describes a QGA 200 curve to protect containment related to the HCTL. Incorrect because it describes the design of the Primary Containment Pressure Limit (PCPL) curve.

Reference: EPGs/SAGs Appendix B Vol. I Rev. 4.9
Reference provided during examination: QGA 200 (HCTL Curve)

Cognitive level: Memory

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295026 EK1.02 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the SUPPRESSION POOL HIGH WATER TEMPERATURE: Heat Capacity

Importance: 4.1
10 CFR Part 55 Content: 41.8 to 41.10

Question Source: Bank
Question History: 2016 ILT NRC Exam
Comments: None

14 **ID: 2487937** **Points: 1.00**

The following conditions exist on Unit 1 during a LOCA:

- Reactor pressure is 380 psig and STEADY
- Indicated Fuel Zone reactor water level is -142 inches and STEADY
- Drywell temperature is 220°F and STEADY
- Both Reactor Recirc Pumps are TRIPPED

Using the table from the 901-5 panel provided below, what is the corrected RPV water level? (Reference Provided)

FUEL ZONE CORRECTION

INDICATED LEVEL (inches)			ACTUAL LEVEL (inches)
>800 PSIG	>400 PSIG	>100 PSIG	
-62	-40	-13	10
-70	-48	-22	0
-77	-57	-32	-10
-85	-65	-41	-20
-101	-82	-60	-40
-115	-96	-78	-59
-132	-116	-97	-80
-147	-133	-116	-100
-163	-151	-135	-120
-180	-169	-155	-142
-196	-186	-173	-162
-205	-196	-184	-174
-218	-211	-201	-191
-225	-219	-209	-200
-272	-270	-265	-260
-303	-304	-303	-300

A. -110 inches

B. -128 inches

C. -142 inches

D. Level cannot be corrected due to reference leg flashing.

Answer: B

Answer Explanation

Answer Explanation: Use the PRESSURE column and cross reference to ACTUAL LEVEL.

Since 380 psig is less than 400 psig but greater than 100 psig, use the >100 psig column. -142 inches is 7" lower than -135" (between -135 and -155). Interpolating between -120 and -142 the actual water level is -128".

Fuel Zone instruments can only be used to determine Reactor water level when the Reactor Recirculation pumps are OFF and their indicated level has been corrected.

Distractor 1: Plausible if the Actual and Indicated Level columns are reversed to derive the answer. Incorrect because the indicated level is required to interpolated to determine the actual level. This plaque and it's associated procedure were revised to attempt to reduce the risk of this occurring.

Distractor 2: Plausible because water level indication on this instrument is read when the recirc pumps are off. Incorrect because the indicated RPV water level must still be pressure corrected.

Distractor 3: Plausible because reference leg flashing could potentially occur with Drywell temperature > 212°F. Incorrect since the correct temperature for flashing at 380 psig would be ~440 °F.

Reference: QCAP 0200-10 Rev. 60
Reference provided during examination: 901-5 Panel RPV Water Level Correction Table, Steam Tables

Cognitive level: High

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295028 EK2.02 Knowledge of the relationship between the HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY) and the following systems or components: Components internal to the drywell.

Importance: 3.2
10 CFR Part 55 Content: 41.7/45.8

Question Source: New
Question History: None
Comments: None.

15 **ID: 2474382** **Points: 1.00**

Approximately 1 hour ago, a seismic event resulted in a LOCA and Torus leak on Unit 1.
The crew has been taking actions in QGAs 100, 200, and 300.

- RPV pressure is 40 psig.
- Drywell pressure is 8 psig.
- Torus pressure is 6 psig as read from PI 1-1640-20, TORUS BTM PRESS.
- Torus level is 10 ft.
- Torus temperature is 160°F.
- Bus 14-1 is DE-ENERGIZED.
- All available ECCS pumps are running.

What is the approximate MAXIMUM allowable RHR flow per pump?

(Reference provided**)**

A. 3750 gpm.

B. 4500 gpm.

C. 4750 gpm.

D. 5200 gpm.

Answer: C

Answer Explanation

Answer Explanation: With the loss of Bus 14-1, the maximum number of ECCS pumps available are 2 RHR pumps and 1 Core Spray pump. HPCI and RCIC are isolated on low RPV pressure. QCAP 0200-10 Attachment R directs the use of Attachment V page 1 to determine RHR pump flow. Using the 6 psig Torus pressure line, it intersects the 160°F Torus temperature at 4750 GPM.

Distractor 1: Plausible because it is a NPSH determined from Attachment S page1. Incorrect because it is use when 3 or 4 RHR pumps and HPCI and/or RCIC are running.

Distractor 2: Plausible because it is an NPSH determined from Attachment U, page 1. Incorrect because the curve is used when HPCI and/or RCIC are also operating. HPCI and RCIC are both isolated on low RPV pressure.

Distractor 3: Plausible because it is an NPSH determined from Attachment V, page 2. Incorrect because the Core Spray curve is used instead of the RHR curve.

Reference: QCAP 0200-10, Attach V, Rev. 60
Reference provided during examination: QCAP 0200-10 Attachments R through W.

Cognitive level: High

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295030 EK1.02 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the LOW SUPPRESSION POOL WATER LEVEL: Pump NPSH

Importance: 3.9
10 CFR Part 55 Content: 41.8 to 41.10

Question Source: New
Question History: None
Comments: None

16 **ID: 2481379** **Points: 1.00**

A Loss of Offsite Power (LOOP) and a Loss of Coolant Accident (LOCA) have occurred on Unit 1.

- RPV pressure is 700 psig and is LOWERING at 2 psig per minute.
- RCIC is injecting at 400 gpm.
- HPCI tripped and cannot be restarted.
- RPV water level is -49 in and LOWERING at 10 in per minute.
- Drywell pressure is 2.0 psig and RISING at 0.5 psig per minute.
- Both Emergency Diesel Generators have started and energized their respective busses.

How will the Automatic Depressurization System (ADS) Relief valves respond with the current trend?

The ADS relief valves will…

A. NOT automatically open

B. automatically open in 60 seconds (1 minute)

C. automatically open in 170 seconds (2 minutes and 50 seconds)

D. automatically open in 570 seconds (9 minutes and 30 seconds)

Answer: C

Answer Explanation

Answer Explanation: In 60 seconds, RPV water level will be at -59 inches and Drywell pressure will be > 2.5 psig. At this time the ADS 110 second Timer will start. Low Pressure ECCS pumps are running as the EDGs started and energized their respective busses. Therefore, after 170 seconds total time, all ADS logic conditions are met for an automatic blowdown to initiate.

Distractor 1: Plausible because under current conditions ADS valves will not automatically open. Incorrect because in one minute, conditions will be met to start the ADS 110 second Timer, after which, an automatic depressurization will be initiated.

Distractor 2: Plausible because in 60 seconds both RPV level and Drywell pressure will be at their ADS setpoints. Incorrect because the 110 second Timer must elapse to complete the logic and start the blowdown.

Distractor 3: Plausible because in one minute the 8.5 minute (510 seconds) Time Delay Relay will start. The initiation setpoint of -59 inches for 8.5 minutes will be met in a total time of 570 seconds. Incorrect because with Drywell pressure at 2.5 psig, the ADS setpoint for a blowdown will be met in 170 seconds.

Reference: QCAP 0200-10 Rev.60, 4E-1462 Sh.1 Rev.AO, 4E-1461 Sh.2 Rev.AP, 4E-1462 Sh.3 Rev.AK
Reference provided during examination: None

Cognitive level: High

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295031 EA1.07 Ability to operate or monitor the following as they apply to REACTOR LOW WATER LEVEL: Safety/relief valves

Importance: 4.1
10 CFR Part 55 Content: 41.7/45.6

Question Source: New
Question History: N/A
Comments: None

17 **ID: 2474398** **Points: 1.00**

The initial direction in the pressure leg of QGA 101, RPV CONTROL (ATWS), if ADS valves are cycling, states:

"Open ADS valves to lower RPV pressure to 940 psig."

What is the operational effect of this direction?

A. Prevent a rapid RPV depressurization and a resultant Low Pressure ECCS injection.

B. Prevent actuation of the Safety valves and subsequent pressurization of the Drywell.

C. Prevent pressure and level oscillations which can result in significant power oscillations.

D. Prevent actuating SBLC pump discharge relief valves which can result in delaying reactor shutdown.

Answer: C

Answer Explanation

Answer Explanation: Per the EPG/SAG B.I-8-36: SRVs should generally be opened manually when used to augment RPV pressure control, particularly under ATWS conditions. Manual operation affords direct, positive control over valve operation and thus minimizes the possibility of inadvertent pressure and power transients under ATWS conditions.

Distractor 1: Plausible because the ADS setpoint (-59 inches) may be reached when RPV water level is lowered. Incorrect because this is the reason for the initial ATWS action of inhibiting ADS.

Distractor 2: Plausible because actuation of a safety valve would result in Drywell pressurization. Incorrect because this would only occur when reactor power initially exceeded pressure relief capability through relief valves and and/or main turbine bypass valves. In this case, the operator would not immediately be capable of controlling RPV pressure below the relief valve or scram setpoint.

Distractor 3: Plausible because high RPV pressure could result in the SBLC relief valves lifting and delaying the reactor shutdown. Incorrect because a QCOP 1100-02, CAUTION statement warns the operator of the possibility of lifting SBLC relief valves when RPV pressure is > 1200 psig. The procedure also prohibits 2 pump SBLC operation if RPV pressure is > 1200 psig.

Reference: EPGs/SAGs Appendix B Vol. I, Rev.4.9
Reference provided during examination: None

Cognitive level: Memory.

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295037 EK1.01 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor pressure effects on reactor power

Importance: 4.3
10 CFR Part 55 Content: 41.8 to 41.10

Question Source: Bank
Question History: 2016 ILT NRC Exam
Comments: None

18 **ID: 2479738** **Points: 1.00**

Unit 1 was operating at 100% power when a LOCA occurred.

- The B MSL failed to isolate.
- There is a large steam leak on the Main Turbine Floor.
- Drywell radiation is 10000 R/hr and rising.
- Off-site release rate has reached the Alert level.
- Group II and Group III isolations are verified.
- Several Turbine Building Area Radiation Monitors are alarming

Which of the following configurations satisfies the Control Room ventilation requirements within the next 40 minutes?

A. The A Train Air Handling Unit in Recirc Mode ONLY.

B. The A Train Air Handling Unit in Recirc mode WITH the Air Filtration Unit.

C. The B Train Air Handling Unit in Recirc mode ONLY.

D. The B Train Air Handling Unit in Normal mode WITH the Air Filtration Unit.

Answer: B

Answer Explanation

Answer Explanation: In accordance with QCOP 5750-09, step E.9, the Control Room HVAC AFU 1/2A or 1/2B Booster Fan must be started within 40 minutes of a LOCA. With a reactor scram from 100% power, RPV water level will lower to < 0 inches before recovery. Per QCAP 0200-10 Attachment M, Control Room ventilation will automatically go into Recirc Mode on an RPV low water level condition.

Distractor 1: Plausible because A Train AHU is normally on and should have transferred to Recirc Mode on the Group II. Incorrect because the AFU is required to be in operation within 40 minutes of a LOCA.

Distractor 2: Plausible because B Train AHU will ventilate and maintain habitability requirements of the Control Room Emergency Zones. Incorrect because the AFU is required to be in operation within 40 minutes of a LOCA.

Distractor 3: Plausible because B Train AHU can supply Control Room ventilation requirements and the AFU operation is required. Incorrect because Control Room Ventilation is required to be in Recirc Mode.

Reference: QCOP 5750-09 Rev.67., QCAP 0200-10 Rev.60
Reference provided during examination: None

Cognitive level: High

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 295038 EA1.06 Ability to operate or monitor the following as they apply to HIGH OFFSITE RELEASE RATE: Plant ventilation systems

Importance: 3.6
10 CFR Part 55 Content: 41.7/45.6

Question Source: New
Question History: N/A
Comments: None

19 **ID: 2474407** **Points: 1.00**

- A severe fire is occurring in the Unit 1 LP Heater Bay.
- The Shift Manager has determined entry into QCARP 0030-01, TB-III INJECTION WITH SSMP AND BRINGING THE UNIT TO COLD SHUTDOWN is required.
- A manual Reactor Scram is inserted.

All remaining control room actions must be completed in no later than _(1)_ in order to _(2)_.

A. (1) 10 minutes
(2) establish reactor vessel injection

B. (1) 10 minutes
(2) preserve reactor vessel inventory

C. (1) 32 minutes
(2) establish reactor vessel injection

D. (1) 32 minutes
(2) preserve reactor vessel inventory

Answer: B

Answer Explanation

Answer Explanation: The QCARPs are designed such that specific actions must be accomplished within specified times. The time clock starts when the reactor is scrammed. These actions are:

a. Within 10 minutes, automatic actuations must be disabled. Most of these actions are to preserve Reactor vessel inventory.
b. Within 32 minutes, Reactor vessel injection must be established.
c. Within 3 hours, Torus Cooling must be established.
d. Within 4 hours, Torus level monitoring must be established.

Distractor 1: Plausible because 10 minutes is the correct time period. Incorrect because the purpose of the actions is to preserve RPV inventory.

Distractor 2: Plausible because the QCARPs have 32 minute actions to establish RPV injection. Incorrect because the control room actions have a 10 minute requirement and are designed to preserve RPV inventory.

Distractor 3: Plausible because the purpose is to preserve RPV inventory. Incorrect because the actions are required to be completed within 10 minutes.

Reference: QCARP 0030-01 Rev.34, step E.2.
Reference provided during examination: None

Cognitive level: High

L**evel (RO/SRO):** RO
Tier: 1 **Group:** 1

K/A: 600000 AK3.04 Knowledge of the reasons for the following responses or actions as they apply to PLANT FIRE ON SITE: Actions contained in the fire response procedures for a plant fire on site.

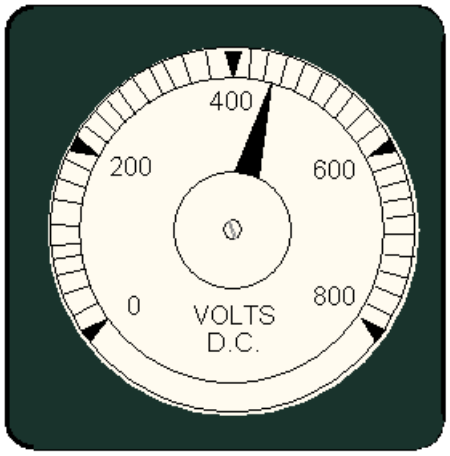
Importance: 3.6
10 CFR Part 55 Content: 41.5/41.10/45.6/45.13

Question Source: Bank
Question History: 2016 ILT NRC Exam
Comments: None

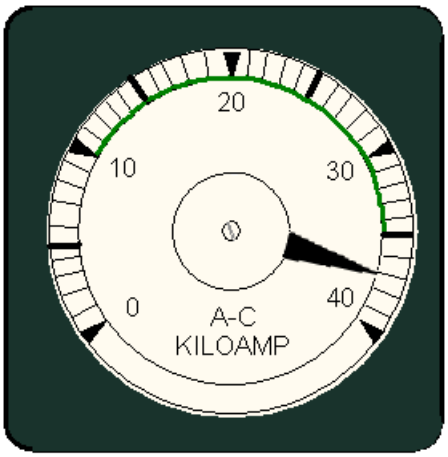
20 **ID: 2479760** **Points: 1.00**

- Unit 1 is operating at 100% power.
- A Grid disturbance results in steadily LOWERING 345 KV Switchyard voltage.
- The Auto Voltage Regulator (AVR) responds by RAISING Main Generator Terminal Voltage.
- Annunciator 901-8 H-10, GEN 1 EXCITER FIELD OVERCURRENT, alarms.

U1 Generator indications are as shown below:



FIELD VOLT



STATOR OUTPUT CUR

Main Generator Megawatts and Frequency remain stable at their normal values throughout the transient.

Based on the above indications, operator actions are required to prevent...

A. a Main Turbine Overspeed Trip.

B. the Main Generator from slipping a pole.

C. a Main Generator Trip on Reverse Power.

D. the Main Generator rotor and stator from overheating.

Answer: D

Answer Explanation

Answer Explanation: The given conditions (lowering grid voltage) will cause the generator automatic voltage regulator to raise the rotor excitation current in order to raise generator terminal voltage, in an attempt to raise grid voltage. The AVR will continue to raise excitation current to the field until the over-excitation limit is reached, at which point the Exciter Field Overcurrent annunciator alarms.
The graphics show the Stator Output current beyond the generator nameplate rating (>34,256 amps).
Although the generator is designed to exceed nameplate ratings for short periods of time (~2 mins), operator actions are required to control parameters to prevent stator and rotor overheating.

Distractor 1: Plausible because the generator could trip on field over-voltage or generator over-excitation if conditions continue to deteriorate. A sudden removal of all electrical loads from the generator may cause the turbine to overspeed and trip. Incorrect because a generator trip will directly trip the turbine, preventing an overspeed condition.

Distractor 2: Plausible because lowering grid voltage would cause the generator field to weaken (a condition that would be true if the field were under-excited). Incorrect because the field is not in an under-excited condition.

Distractor 3: Plausible because the lowering grid voltage will initially cause MVARS to lower, which could lead to negative MVARS. Incorrect because a reverse power condition occurs when generator megawatts lower until megawatts are negative, or the generator becomes motorized.

Reference: QCAN 901-8 H-10 Rev.03, QCOA 6000-02 Rev. 22, QCOP 6000-02 Rev. 23
Reference provided during examination: Field volts and Stator Output Current meter faces.

Cognitive level: High.

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

K/A: 700000 AK2.08 Knowledge of the relationship between GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES and the following systems or components: Main Turbine generator and auxiliary systems

Importance: 3.1
10 CFR Part 55 Content: 41.4/41.5/41.7/41.10/45.8

Question Source: Bank
Question History: Quad Cities 2012 ILT NRC Exam
Comments: None

21 **ID: 2474419** **Points: 1.00**

When all RPV water level indication is lost, QGA 500-4, RPV FLOODING actions are prioritized over QGA 100, RPV CONTROL Level Leg actions to ...

A. restore RPV water level indication by backfilling the RPV water level instrument reference legs.

B. assure adequate core cooling by a combination of submergence and steam cooling.

C. assure adequate core cooling by filling the RPV to the main steam lines.

D. solely implement the steam cooling method of heat transfer for core cooling.

Answer: C

Answer Explanation

Answer Explanation: The QGA 100 Level leg override states, that if RPV water level is unknown, (which is the case when all RPV water level instrument reference legs have flashed), then exit this procedure and enter QGA 500-4. The Technical Support Guidelines (TSG) Reference Manual states that if RPV water level cannot be determined, positive assurance of adequate core cooling is provided only if the RPV is flooded to the main steam lines.

Distractor 1: Plausible because QGA 500-4 actions will result in backfilling the reference legs if the RPV is flooded to the MSLs. Incorrect because the strategy of QGA 500-4 is to cool the core by submergence, and the only positive indication is the RPV flooded to the MSL when level indication is lost.

Distractor 2: Plausible because this is a QGA 500-4 action. Incorrect because it applies to when the reactor is not shutdown. The transition would have been from QGA 101 and not QGA 100.

Distractor 3: Plausible because steam cooling is a viable core cooling strategy. Incorrect because QGA 500-2 is entered from QGA 100 when available L3 Injection subsystems cannot provide adequate core cooling. It is not a method used because RPV water level is unknown.

Reference: QGA 100 Rev.13, QGA 500-4 Rev.16, TSG Manual Rev.07
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 295008 G2.4.23 HIGH REACTOR WATER LEVEL: EMERGENCY PROCEDURES/PLAN: Knowledge of the bases for prioritizing emergency operating procedures implementation

Importance: 3.4
10 CFR Part 55 Content: 41.10/43.5/45.13

Question Source: New
Question History: N/A
Comments: None

22 **ID: 2479858** **Points: 1.00**

Unit 1 is operating at full power.

Which of the following conditions REDUCES the available net positive suction head (NPSH) for the Recirculation Pumps?

A. Raising reactor pressure

B. Lowering RWCU Demin flow

C. Lowering reactor water level

D. Lowering feedwater inlet temperature

Answer: C

Answer Explanation

Answer Explanation: Lowering reactor water level lowers the pressure at the suction of the recirc pumps, ie. reduces the static pressure head. This decreases the availability of NPSH to the Recirculation Pumps. Without the recirculation pumps operating properly, sufficient cooling cannot be maintained during emergency or abnormal situations. It is vital to maintain appropriate NPSH for these pumps.

Distractor 1: Plausible because RPV pressure affects Recirc Pump NSPH. Incorrect because raising reactor pressure will improve the available NPSH (further from saturation pressure).

Distractor 2: Plausible because the RWCU system takes its suction flow from the Recirc pump suction line. Incorrect because lowering RWCU Demin flow will raise NPSH in the recirc pump suction line.

Distractor 3: Plausible because feedwater inlet temperature affects Recirc Pump NPSH. Incorrect because lowering feedwater temperature will improve the available NPSH (lowers the corresponding saturation pressure).

Reference: UFSAR Rev. 16, 5.4.1.3.2.3 Recirculation Pump Performance
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 295009 AK1.02 Knowledge of the operational implications and/or cause and effect relationships of the following concepts as they apply to the LOW REACTOR WATER LEVEL: Recirculation pump net positive suction head

Importance: 3.4
10 CFR Part 55 Content: 41.8 to 41.10

Question Source: New
Question History: N/A
Comments: None

Test Answer Key

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23 **ID: 2474493** **Points: 1.00**

Unit 1 was at rated power when a transient occurred.
QGA 100 and 200 actions are in progress.

The following annunciators are in alarm:

- 901-3 H-4, DRYWELL HIGH AIR TEMP
- 901-3 G-4, DRYWELL HIGH PRESSURE
- 901-5 F-8, RX VESSEL LOW LEVEL

The highest Drywell temperature reached was 250ºF.

Restoring RBCCW flow to the Drywell Coolers _(1)_ prohibited.

The RBCCW Pumps _(2)_ be restarted without BYPASSING the trip signal.

A. (1) IS
(2) CAN

B. (1) IS
(2) CAN NOT

C. (1) IS NOT
(2) CAN

D. (1) IS NOT
(2) CAN NOT

Answer: D

Answer Explanation

Answer Explanation: With Drywell pressure at 2.5 psig, Core Spray logic automatically load sheds the Drywell Cooler fans and RBCCW pumps. The Drywell Cooler fans and RBCCW pumps can be manually started if the LOCA trip signal is bypassed on the 912-5 panel using keylock switches when an ECCS initiation signal (2.5 psig DW press) is present. QCOP 5750-19, DRYWELL COOLER OPERATION, prohibits restoring RBCCW flow to the Drywell coolers if DW temperature has been above 260ºF due to water hammer and two phase flow concerns.

Distractor 1: Plausible because restarting RBCCW flow is prohibited at high Drywell temperatures. Incorrect because the temperature limit is 260°F and the RBCCW pumps cannot be restarted without bypassing the LOCA trip signal.

Distractor 2: Plausible because the RBCCW pumps cannot be restarted without bypassing the LOCA trip signal. Incorrect because restarting RBCCW flow to the DW coolers is not prohibited.

Distractor 3: Plausible because restarting RBCCW flow is not prohibited. Incorrect because the LOCA trip signal must be bypassed to restart the RBCCW pumps.

Reference: QCOA 0201-01 Rev.30, QCOP 5750-19 Rev.12
Reference provided during examination: None

Cognitive level: High.

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

K/A: 295012 AA2.04 Ability to determine or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: System/component operating limitations

Importance: 3.8
10 CFR Part 55 Content: 41.10/43.5/45.13

Question Source: New
Question History: N/A
Comments: None

24 **ID: 2474499** **Points: 1.00**

Unit 2 was at 90% power when the following occurs:

- Reactor power is 91% and rising.
- Reactor pressure is 1000 psig and rising.
- MO 2-3403, LP HTR STRING BYP VLV, is currently full OPEN.
- Annunciator 902-6 G-1, HEATER 1A1 HIGH LEVEL, is in alarm.

What is the next required operator action?

A. Insert a Manual Scram.

B. Reduce Recirc pump speeds to limit the power increase.

C. Reduce Feedwater flow to $\leq$ 7.6 Mlbm/hr using CRAM rods ONLY.

D. Insert control rods by sequence step in reverse order to limit the power increase.

Answer: B

Answer Explanation

Answer Explanation: Per QCOA 3500-01, FEEDWATER TEMPERATURE REDUCTION WITH MAIN TURBINE ONLINE, the Immediate Operator Action step C.1 is: Initiate Emergency Power Reduction to limit the power increase. This is followed with a caution statement that power reduction should start first with Recirc Flow to prevent LHGR violations and possible fuel failures. QCGP 3-1, REACTOR POWER OPERATIONS step F.1 has a note stating that the use of Recirc flow prior to Rod insertion is preferred.

Distractor 1: Plausible because this is a subsequent action in QCOA 3500-01. Incorrect because inserting a manual scram is done if reactor power cannot be maintained below 105%.

Distractor 2: Plausible because this is also a subsequent action in QCOA 3500-01 to reduce Feedwater flow to < 7.6 Mlb/hr, allowing the operator to close MO 2-3403, LP HTR STRING BYP VLV and terminate the reactivity addition. Incorrect because you would not have to use CRAM rods only in an emergency power reduction and it is specifically stated in the procedure to use recircs first.

CAUTION

Power reduction should start first with Recirc Flow to prevent LHGR violations and possible fuel failures.

Distractor 3: Plausible because control rod insertion is also part of the emergency power reduction step. Incorrect because at this power level, insertion of the CRAM rods is done first. Insertion of control rods by sequence step in reverse order is done only after the CRAM rods are inserted.

Reference: QCOA 3500-01 Rev.44, QCGP 3-1 Rev.93
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 295014 AK2.11 Knowledge of the relationship between the INADVERTENT REACTIVITY ADDITION and the following systems or components: Recirculation flow control.

Importance: 3.9
10 CFR Part 55 Content: 41.7/45.8

Question Source: New
Question History: N/A
Comments: None

25 **ID: 2474503** **Points: 1.00**

Unit 1 was operating at 100% power when a small line break on the 1A Recirc pump discharge line occurred.

QGA 100, 200 and 300 actions are in progress.

- RPV water level is -50 in and rising
- Drywell pressure is 3.0 psig and slowly rising
- Drywell temperature is 170°F and slowly rising
- Torus pressure is 2.5 psig and slowly rising
- HPCI autostarted and is injecting
- Annunciator 901-3 F-12, HPCI PUMP AREA HI TEMP, is in alarm
- HPCI Room temperature is 175°F and rising

(1) What actions are required?
(2) What is the basis for the actions taken?

A. (1) Trip HPCI and close both Steam Isolation Valves.
(2) Prevents a release of radioactive material to the environment that will result in exceeding offsite dose rate limits to the general public.

B. (1) Continue HPCI operation and monitor room temperature.
(2) HPCI operation is required to maintain adequate core cooling.

C. (1) Trip HPCI and close both Steam Isolation Valves.
(2) To minimize moisture buildup and overheating in the Standby Gas Treatment System charcoal beds.

D. (1) Continue HPCI operation and monitor room temperature.
(2) HPCI startup and injection is designed for a small break LOCA and operation should not be terminated until RPV pressure is < 150 psig.

Answer: A

Answer Explanation

Answer Explanation: The conditions suggest that there is a steam leak in the room. HPCI operation is not required to maintain RPV water level and should be tripped and isolated. The objective of the primary containment isolation system (PCIS) is to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the primary containment. The PCIS system provides automatic isolation of appropriate pipelines which penetrate the primary containment whenever certain monitored variables exceed their preselected operational limits.

Specifically, one criteria is that PCIS prevents the release of radioactive materials in excess of the limits in 10 CFR 100 (or 10 CFR 50.67 as applicable) as a result of the design basis accidents.

Distractor 1: Plausible because HPCI room temperature is expected to rise during extended operation. Incorrect because it is not needed to maintain RPV water level and should be isolated to stop the steam leak. The Condensate/Feed system is available and HPCI injection is raising water level.

Distractor 2: Plausible because HPCI must be tripped and isolated to stop the steam leak. Incorrect because the SBGTS Demister is designed to remove water particles entrained in the steam-air mixture routed through the SBGTS.

Distractor 3: Plausible because HPCI room temperature is expected to rise during extended operation. Incorrect because HPCI operation is not needed to maintain adequate core cooling and should be isolated to stop the steam leak.

Reference: UFSAR Section 7.3.2 Rev 10
Reference provided during examination: None

Cognitive level: High

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

K/A: 295032 EK3.03 Knowledge of the reasons for the following responses or actions as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Isolating affected systems

Importance: 4.0
10 CFR Part 55 Content: 41.5/45.6

Question Source: New
Question History: N/A
Comments: None

Justification for how references supplied during 2023 ILT exam do not give away answers: First part of question requires known value for max safe temperature for HPCI room. That information is redacted in the references provided. Also, the portion of the QGA 300 that states above **max safe/alarm level** is redacted in the reference. The second part of the question (basis) is not in the included references at all.

26 **ID: 2487278** **Points: 1.00**

An accident has occurred on Unit 1 requiring operation of a system to monitor for Hydrogen in the primary containment.

At time 0800, the 'CAM/ACAD PWR CONT' Control Switch on the 901-56 panel has been moved from 'AUTO' to 'ON'.

If applicable, what is the EARLIEST time that the CAM analyzer instrumentation will display reliable information?

A. 1 minute later

B. 45 minutes later

C. 2 hours later

D. Not applicable, the High Radiation Sampling System (HRSS) must be used to sample containment hydrogen

Answer: B

Answer Explanation

Answer Explanation: A note in QCOP 2400-01, CAM Subsystem Operation, states: "Allow approximately 45 minutes for analyzer indication to stabilize in order to obtain reliable information." It is expected that the Hydrogen and Oxygen readings will oscillate given the volume of the containment space. Forty-five (45) minutes allows time for readings to stabilize and therefore obtain a more accurate and representative sample.

Distractor 1: Plausible because it is a precaution for the Oxygen Analyzer system. Incorrect since the Hydrogen system requires approximately 45 minutes to be reliable.
Distractor 2: Plausible because two to six hours may be required if the heater boxes are not turned on and the detectors have cooled. Incorrect because the heater boxes are normally on.
Distractor 3: Plausible because chemistry is able to sample containment hydrogen per CY-QC-110-204 using HRSS system. Incorrect because the CAM system **can** be used.
Reference: QCOP 2400-01 Rev. 23
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 500000 EA1.01 Ability to operate or monitor the following as they apply to HIGH CONTAIMENT HYDROGEN CONCENTRATION: Primary containment hydrogen instrumentation.

Importance: 3.7
10 CFR Part 55 Content: 41.7 / 45.6

Question Source: New
Question History: N/A
Comments: None

27 **ID: 2474699** **Points: 1.00**

Unit 1 is operating at 100% power.

- A leak has developed in the 1A Reactor Recirculation Loop.
- Drywell pressure is 2.5 psig.
- Reactor level is 30 inches.
- Reactor pressure is 900 psig.
- LPCI Loop Select Logic has selected the B Reactor Recirculation Loop for injection.

Which of the following valves will be CLOSED by LPCI Loop Select?

A. MO 1-202-5B, PMP DISCH VLV
MO 1-1001-28A, OTBD LPCI INJECTION VLV

B. MO 1-202-5A, PMP DISCH VLV
MO 1-1001-28A, OTBD LPCI INJECTION VLV

C. MO 1-202-5B, PMP DISCH VLV
MO 1-1001-28B, OTBD LPCI INJECTION VLV

D. MO 1-202-5A, PMP DISCH VLV
MO 1-1001-28B, OTBD LPCI INJECTION VLV

Answer: A

Answer Explanation

Answer Explanation: Per Technical Specification Basis 3.5.1-2, "The LPCI System is equipped with a loop select logic that determines which, if any, of the recirculation loops has been broken and selects the non-broken loop for injection." A leak in the A Recirculation Loop will cause closure of the Recirc Pump Discharge valve on the intact loop and closure of the broken loop RHR Injection valve. Therefore the "B" Reactor Recirculation pump discharge valve, MO 1-202-5B will close and the RHR Inboard Injection valve, MO 1-1001-28A will also close.

Distractor 1: Plausible because RHR MO 1-1001-28A closes. Incorrect because Recirc Pump Discharge Valve MO 1-202-5A does not close.

Distractor 2: Plausible because Recirc Pump Discharge Valve, MO 1-202-5B closes. Incorrect because RHR MO 1-1001-28B does not close.

Distractor 3: Plausible because a Recirc Pump Discharge valve and an RHR Injection Valve will close. Incorrect because the valve closures are consistent with a leak on the 1B Reactor Recirculation Loop.

Reference: TS Basis 3.5.1-2 Rev. 0, 4E-1438C Rev.AU, 4E-1438D Rev.U
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 203000 A3.08 Ability to monitor automatic operation of the RHR/LPCI: INJECTION MODE, including: System initiation sequence

Importance: 4.0
10 CFR Part 55 Content: 41.7/45.7

Question Source: New
Question History: N/A
Comments: None

28 **ID: 2482353** **Points: 1.00**

Unit 2 was operating at 100% power when a loss of all feedwater occurred.

The RCIC system was manually started and is currently injecting into the RPV for water level control.

If RPV water level reaches +48 inches, the RCIC Turbine will trip by closing __(1)__.

The RCIC high RPV water level trip __(2)__.

A. (1) 2-1303B, Trip Throttle Valve
(2) automatically resets

B. (1) 2-1303B, Trip Throttle Valve
(2) must be Manually reset

C. (1) MO 2-1301-61, STM TO TURB VLV
(2) automatically resets

D. (1) MO 2-1301-61, STM TO TURB VLV
(2) must be Manually reset

Answer: C

Answer Explanation

Answer Explanation:
Upon receiving the respective trip signal, the Trip Throttle Valve or the steam supply valve will close. The system will remain in the trip condition until the conditions for reset are met. The exception is for a high water level trip. After the high water level signal has cleared, and RPV water level lowers, the K11A relay de-energizes. In this condition, if the low RPV water level initiate signal is reached, the steam supply valve will open and the system will inject water into the feed system/reactor. All but overspeed and local trip lever close 1301-61.
Therefore, the RCIC Steam Supply valve (1301-61) closes on a high RPV level trip. It auto-resets when the high RPV level condition clears.

Also, QCOA 1300-01, step B. states if RCIC Turbine trip caused by Turbine overspeed, then RCIC TRIP THROTTLE VLV will close. If RCIC Turbine trip was not caused by Turbine overspeed, then MO 1(2)-1301-60, MIN FLOW VLV, and MO 1(2)-1301-61, STM TO TURB VLV, will both close. Step D.3. states if RCIC Turbine trip was caused by Reactor high water level trip signal, then no action is required to reset the trip signal.

Distractor 1 is plausible because RCIC high RPV water level trip automatically resets is correct, however if RPV water reaches +48 inches, the RCIC Turbine will trip by closing MO 2-1301-61, STM TO TURB VLV, not the Trip Throttle Valve.

Distractor 2 is plausible because upon receiving the respective trip signal, the Trip Throttle Valve or the steam supply valve will close. However, the RCIC Steam Supply valve (1301-61) closes on a high RPV level trip and auto-resets when the high RPV level condition clears.

Distractor 3 is plausible because when RPV water reaches +48 inches, the RCIC Turbine will trip by closing MO 2-1301-61, STM TO TURB VLV is correct. However, the RCIC high RPV water level trip will automatically reset, does not require a manual reset.

Reference: QCOA 1300-01, rev 18. QCOP 1300-02, rev 33. UFSAR Ch. 5.4.6.2 Rev 16
Reference provided during examination: None

Cognitive level: High

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

K/A: 217000 A4.03 - RCIC Reactor Core Isolation Cooling System - Ability to manually operate and/or monitor in the control room: System valves.
Importance: 3.8
10 CFR Part 55 Content: 41.7/45.5 to 45.8

Question Source: NRC Bank
Question History: 18-1 ILT NRC Exam
Comments: None

29 **ID: 2474592** **Points: 1.00**

Unit 1 was in Mode 3 with the RHR Loop A of Shutdown Cooling (SDC) in service and ONLY the 1B Reactor Recirc pump running when the following occurred:

- 1B Reactor Recirc pump tripped.
- RPV water level is 30 in.

The DIFFERENCE between the RPV bottom head coolant temperature and the RPV coolant temperature is 185°F.

Which of the following actions will reduce thermal stratification without the possibility of exceeding RPV metal heat-up rate limits?

A. Start the 1A Reactor Recirc pump to establish forced circulation through the core.

B. Raise Reactor water level to 90" to 100", then perform a feed and bleed to maintain level constant.

C. Increase RWCU rejection and replace water inventory with the CRD system to maintain RPV water level at 30".

D. Increase RBCCW flow to maximum on RWCU system non-regenerative heat exchangers to increase heat removal from the Reactor vessel.

Answer: B

Answer Explanation

Answer Explanation: Raising RPV water level to 95" will bring level to above the steam separators and establish a flowpath for natural circulation, reducing stratification.

Distractor 1: Plausible because establishing forced circulation will eliminate stratification. Incorrect because a prerequisite in QCOP 0202-43 limits the start of a Recirc pump to < 145°F between bottom head and RPV coolant temperature to avoid violating the P/T limits of the vessel.

Distractor 2: Plausible because it is an action that will increase heat removal from the core which is required when SDC is lost. Incorrect because with level at 30", there is no path for natural circulation and increasing CRD flow will add cold water to the bottom of the vessel.

Distractor 3: Plausible because it is an action that will increase heat removal from the core which is required when SDC is lost. Incorrect because with level at 30", there is no path for natural circulation and increasing RBCCW flow will result in colder water collecting in the bottom head of the vessel.

Reference: QCOA 1000-02 Rev. 23
Reference provided during examination: None

Cognitive level: High

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

K/A: 205000 K3.02 Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on the following systems or system parameters: Reactor water level

Importance: 3.7
10 CFR Part 55 Content: 41.7/45.4

Question Source: New
Question History: N/A
Comments: None

30 **ID: 2474594** **Points: 1.00**

Unit 1 was operating at rated power when a LOCA with a Loss of Offsite power occurred approximately 1 hour ago.

- HPCI is running and maintaining RPV water level at 30 inches.
- The 1/2 EDG is supplying Division I.
- Bus 14-1 tripped on overcurrent.

Moments later, the following annunciators alarm:

- 901-3, C-10 HPCI GRP 4 PCI VLVS NOT OPEN
- 901-3, C-12 HPCI STEAM LINE HIGH DP

What is the effect on the following HPCI valves:

(1) MO 1-2301-4, STM ISOL VLV.
(2) MO 1-2301-5, STM ISOL VLV.

A. (1) CLOSED
(2) CLOSED

B. (1) OPEN
(2) OPEN

C. (1) OPEN
(2) CLOSED

D. (1) CLOSED
(2) OPEN

Answer: C

Answer Explanation

Answer Explanation: With a Bus 14-1 overcurrent condition, the Unit 1 EDG will not load and Bus 19 will be de-energized. The alarms indicate a steam leak causing a Group IV isolation on area high temperature. The MO 1-2301-4 valve is powered by MCC 19-1 which is presently deenergized preventing its closure on the isolation signal. MO 1-2301-5 valve is powered by 250 VDC Bus 1A and closes.

Distractor 1: Plausible because MO 1-2301-5 valve is closed. Incorrect because MO 1-2301-4 valve is deenergized and cannot close on the isolation signal.

Distractor 2: Plausible because MO 1-2301-4 valve is open as it is deenergized. Incorrect because the MO 1-2301-5 valve is powered and closes on the isolation signal.

Distractor 3: Plausible because MO 1-2301-4 closes on a Group IV isolation. Incorrect because MO 1-2301-4 valve is deenergized and cannot close on the isolation signal.

Reference: QCAP 0200-10 Rev.60, QOM 1-2301-2 Rev.07, QCAN 901(2)-3 B-12 Rev.06
Reference provided during examination: None

Cognitive level: High

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

K/A: 206000 K6.10 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the HIGH PRESSURE COOLANT INJECTION SYSTEM: HPCI initiation/isolation logic

Importance: 4.3
10 CFR Part 55 Content: 41.7/45.7

Question Source: New
Question History: N/A
Comments: None

31 **ID: 2480158** **Points: 1.00**

A LOCA has occurred on Unit 2.

- Reactor pressure is 800 psig and LOWERING 10 psig/min.
- Drywell pressure is 12 psig and RISING 2 psig/min.
- Reactor Water Level is -60 in and LOWERING 10 in/min.
- The 2A Core Spray Pump was the ONLY low pressure ECCS pump that started.
- The 2A Core Spray Pump discharge pressure is indicating 0 psig.
- 902-3, B-13 AUTO BLOWDN TIMER START has just alarmed.

Given these conditions and NO other operator actions, the Automatic Depressurization System will...

A. NOT automatically actuate.

B. IMMEDIATELY actuate.

C. actuate in approximately 110 seconds.

D. actuate in approximately 8.5 minutes.

Answer: A

Answer Explanation

Answer Explanation: ADS auto initiation occurs on the following conditions: High Drywell pressure (2.5 psig) concurrent with low reactor water level (-59 in.) for 110 seconds OR a low reactor water level (-59 in.) for 8.5 minutes. In both cases, the logic must sense at least one low pressure ECCS pump running. All RHR and Core Spray pumps have discharge pressure switches that actuate when the pumps are running and input into the ADS logic as a permissive for the auto blowdown. These switches also actuate annunciator 902-3 C-15. If this alarm is not present, then the discharge pressure switch has malfunctioned and is not satisfying the pump running permissive for ADS logic.

Distractor 1: Plausible because high drywell pressure concurrent with low reactor water level are conditions for an ADS blowdown. Incorrect because the 110 second timer has not expired and the ECCS pump running permissive is not met.

Distractor 2: Plausible because a low reactor water level for 8.5 minutes is an ADS actuation signal. Incorrect because ECCS pump permissive is not met.

Distractor 3: Plausible because high drywell pressure concurrent with low reactor water level for 110 seconds are conditions for an ADS blowdown. Incorrect because ECCS pump running permissive is not met.

Reference: QCAN 901(2)-3 B-13 Rev.07, QCAN 901(2)-3 C-15 Rev.07
Reference provided during examination: None

Cognitive level: High

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

K/A: 209001 K3.02 Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on the following systems or system parameters: ADS logic

Importance: 4.1
10 CFR Part 55 Content: 41.7/45.4

Question Source: New
Question History: N/A
Comments: None

32 **ID: 2487805** **Points: 1.00**

Unit 2 was at 100% power with the RWCU system in operation with both pumps running and both RWCU Demins on line.

On Panel 902-5, the SBLC Initiation switch is placed in the SYS 1 position.

With the above conditions,

MO 2-1201-2, RWCU Suction DW Inboard Isolation valve, is expected to be ___(1)___.
The 2A and 2B RWCU Demin Holding pumps are expected to be ___(2)___.

A. (1) OPEN
(2) RUNNING

B. (1) OPEN
(2) OFF

C. (1) CLOSED
(2) OFF

D. (1) CLOSED
(2) RUNNING

Answer: D

Answer Explanation

Answer Explanation: Initiating the SBLC system isolates all RWCU isolation valves which in turn trip the RWCU pumps. The flow reduction through the demin causes the hold pumps to autostart.

Distractor 1: Plausible because RWCU Demin Hold pumps will be running. Incorrect because the MO 2-1201-2 valve will close as part of the Group 3 isolation caused by SBLC initiation.

Distractor 2: Plausible because the RWCU isolation on the Group III initiation signal is 0 inches RPV water level and RPV water level is at 30 inches. Incorrect because a SBLC system initiation is also a RWCU isolation which closes the MO 2-1201-2 valve and starts the Demin Hold pumps on low system flow.

Distractor 3: Plausible because the MO 2-1201-2 valve will be close as part of the Group 3 isolation caused by SBLC initiation. Incorrect because the RWCU Demin Hold pumps will be running.

Reference: QCOP 1100-02 Rev.14, QCOP 1200-11 Rev.42, QCAP 0200-10 Rev.60
Reference provided during examination: None

Cognitive level: High

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

K/A: 211000 A4.06 Ability to manually operate and/or monitor the STANDY LIQUID CONTROL SYSTEM in the control room: RWCU system isolation

Importance: 4.0
10 CFR Part 55 Content: 41.7/45.5 to 45.8

Question Source: Bank
Question History: ILT 19-1 Cert Exam
Comments: None

33 **ID: 2474600** **Points: 1.00**

Unit 2 is operating at 40% reactor power when the Main Turbine Stop Valves 1 and 3 failed closed.

Which of the following alarms are expected?

A. 902-5 D-10, CHANNEL A REACTOR SCRAM
902-5 A-12, CHANNEL A/B STOP VLVS CLOSE TRIP

B. 902-5 D-15, CHANNEL B REACTOR SCRAM
902-5 A-12, CHANNEL A/B STOP VLVS CLOSE TRIP

C. 902-5 D-15, CHANNEL B REACTOR SCRAM
902-5 C-13, CHANNEL A/B REACTOR HIGH PRESSURE

D. 902-5 D-10, CHANNEL A REACTOR SCRAM
902-5 C-13, CHANNEL A/B REACTOR HIGH PRESSURE

Answer: B

Answer Explanation

Answer Explanation: An automatic reactor scram on a Turbine trip is bypassed when Reactor power is < 38.5% rated thermal power. With the reactor at 40% power, the combination of Stop Valves 1 and 3 closing will result in a 1/2 scram on Channel B (902-5 D-15). At this low power level, Stop Valves 2 and 4 will open and accommodate the steam flow without a significant RPV pressure spike or Turbine bypass valve actuation. Annunciator 902-5 A-12, is actuated by closure of at least 2 Stop valves with reactor power > 40%.

Distractor 1: Plausible because a 1/2 scram occurs and annunciator 902-5 A-12 will alarm. Incorrect because the 1/2 scram occurs on RPS Channel B.

Distractor 2: Plausible because the 1/2 scram occurs on RPS Channel B. Incorrect because RPV pressure will rise but not to the alarm setpoint.

Distractor 3: Plausible because RPV pressure will rise but not to the alarm setpoint. Incorrect because the 1/2 scram occurs on RPS Channel B.

Reference: QOA 900-5 A-12 Rev.4, QCAN 901(2)-5 D-15 Rev.6, 4E-1466 Sh.1 Rev.AG and Sh.3 Rev. AP
Reference provided during examination: None

Cognitive level: High

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

K/A: 212000 A1.11 Ability to predict and/or monitor changes in parameters associated with operation of the REACTOR PROTECTION SYSTEM including: Lights and alarms

Importance: 3.7
10 CFR Part 55 Content: 41.5/45.5

Question Source: New
Question History: N/A
Comments: None

34 **ID: 2480851** **Points: 1.00**

Which of the following conditions will cause an IRM INOP rod block?

1. IRM downscale on Range 1
2. Any module unplugged
3. High voltage to the detector becomes low
4. IRM upscale with Rx. Mode Switch in RUN

A. 1 and 2 ONLY

B. 2 and 3 ONLY

C. 3 and 4 ONLY

D. 1 and 4 ONLY

Answer: B

Answer Explanation

Answer Explanation: Per QCOP 0700-02, INTERMEDIATE RANGE MONITOR (IRM) OPERATION step E.1: IRM Rod Block setpoints are as follows:

a. IRM downscale allowable value:
   (1) > 5/125 of full scale.
   (2) Bypassed when on Range 1.
b. IRM high allowable value: < 112/125 of full scale.
c. IRM detector "NOT FULL IN" while in MODE 2 (Startup).
d. IRM Inoperable.

The IRM INOP condition is generated by placing the Function switch out of the OPERATE position, unplugging any of the modules in the IRM Drawer, or the IRM detector voltage is low. This is bypassed only when the Mode Switch is in RUN.

Distractor 1: Plausible because unplugging any module in the IRM Drawer results in a 1/2 scram and rod block, (Mode Switch NOT in RUN). Incorrect because an IRM downscale on range 1 does not generate a rod block.

Distractor 2: Plausible because an IRM detector low voltage results in a 1/2 scram and rod block (Mode Switch NOT in RUN). Incorrect because an IRM upscale does not generate a rod block when the Mode Switch is in RUN.

Distractor 3: Plausible because an IRM upscale 105/125 generates an IRM rod block. Incorrect because the IRM upscale rod block is bypassed when the Mode Switch is in RUN. Also an IRM downscale on range 1 does not generate a rod block.

References: QCOP 0700-02 Rev.21, 4E-1411 Rev.W
Reference provided during examination: None

Cognitive level: Memory.

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

K/A: 215003 K4.01 Knowledge of INTERMEDIATE RANGE MONITOR SYSTEM design features and/or interlocks that provide for the following: Rod withdrawal blocks

Importance: 4.0
10 CFR Part 55 Content: 41.7

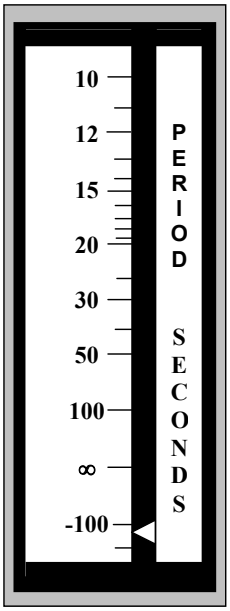
Question Source: New
Question History: N/A
Comments: None

35 **ID: 2474617** **Points: 1.00**

The following conditions exist on Unit 1:

- Startup is in progress.
- Reactor power is STEADY on IRM Range 4.
- SRMs are currently being withdrawn.
- Core flux is top peaked

While withdrawing the SRMs, the NSO notices the SRM Period Meters indicate as shown below.



Is the SRM response EXPECTED or NOT EXPECTED and why?

A. This reading is EXPECTED. The period meter is reflecting cable noise as a result of SRM movement.

B. This reading is EXPECTED. The period meter is responding to detector movement from a high flux to a low flux region.

C. This reading is NOT EXPECTED. The period meter should be indicating infinity because reactor power is steady.

D. This reading is NOT EXPECTED. The period meter should indicate a positive period because the detectors are being withdrawn from the core.

Answer: B

Answer Explanation

Answer Explanation: While withdrawing a SRM from the core, power will appear to drop (negative period) with a top-peaked core. The detector at full insertion is just above the core midplane. As it is withdrawing, it is moving to a lower flux region of the core. However, actual power remains steady (actual period is infinity).

Distractor 1: Plausible because neutron instrumentation is "small" signal transmission that could be affected by cable noise resulting from movement. Incorrect because pulse preamplifiers outside of the drywell improve the signal to noise ratio prior to sending it to the control room, nullifying the noise created by detector movement.

Distractor 2: Plausible because reactor power is steady. Incorrect because the SRMs are moving from a high to low flux region of the core and should register a negative period.

Distractor 3: Plausible because this is the indication expected when inserting the SRMs and also expected if the core was bottom-peaked (flux). Incorrect because the SRMs are being withdrawn and moving from a high to low flux region.

Reference: QCOP 0700-01 Rev.17
Reference provided during examination: Imbedded reference of Period Meter picture

Cognitive level: High

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

K/A: 215004 A1.02 Ability to predict and/or monitor changes in parameters associated with operation of the SOURCE RANGE MONITOR SYSTEM including: Reactor power

Importance: 4.0
10 CFR Part 55 Content: 41.5/45.5

Question Source: New
Question History: N/A
Comments: None

36 **ID: 2474614** **Points: 1.00**

A Unit 1 startup is in progress with SRM's fully inserted and reading approximately 10,000 cps.

- Annunciator 901-5 A-4, SRM HIGH OR INOP, alarms and the associated rod block occurs.
- SRM 21 is now reading approximately 5,000 cps.
- SRM's 22, 23, and 24 are still indicating 10,000 cps.

Which of the following would result in all of the above indications?

A. Low Detector voltage

B. Low 24/48 Bus A voltage

C. Loss of 120 VAC Essential Service

D. SRM Mode Switch moved from OPERATE to STANDBY

Answer: A

Answer Explanation

Answer Explanation: SRM fission detectors operate in the Proportional Region of the Gas Amplification Curve. The higher detector bias voltage (600 VDC) is required for greater sensitivity. The gas amplification factor increases with increasing voltage. Therefore if the applied voltage is low, sensitivity falls and the detector will yield a lower countrate. The low detector voltage will also cause annunciator 901-5 A-4 to alarm.

Distractor 1: Plausible because the SRM Drawer containing the power supply circuitry to the detector is fed from the 24/48 VDC System. Incorrect because 24/48 Bus A supplies power to both SRM 21 and 22 Drawers thus effecting both meters. The only meter affected was SRM 21.

Distractor 2: Plausible because Essential Service powers the SRM recorders. Incorrect because the SRM Drawer which contains the detector power supply circuitry is fed from the 24/48 VDC System.

Distractor 3: Plausible because placing the FUNCTION switch to STANDBY will cause annunciator 901-5 A-4 to alarm. Incorrect because the STANDBY position does not affect meter indication.

Reference: QCAN 901(2)-5 A-4 Rev.6, QCOP 0700-01 Rev.17
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 215004 K6.04 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the SOURCE RANGE MONITOR SYSTEM: Detectors

Importance: 3.2
10 CFR Part 55 Content: 41.7/45.7

Question Source: New
Question History: N/A
Comments: None

37 **ID: 2474639** **Points: 1.00**

- Unit 1 was operating at 75% power when an APRM spiked high due to a failed LPRM.
- The Unit Supervisor has directed the NSO to bypass the LPRM.

Prior to bypassing the LPRM, how is the number of LPRM inputs to APRM determined?

A. Place the APRM meter Function switch to AVERAGE.
Divide the upper scale meter reading by 5 to obtain the number of OPERABLE inputs.

B. Place the APRM meter Function switch to COUNT.
Divide the upper scale meter reading by 5 to obtain the number of OPERABLE inputs.

C. Place the APRM meter Function switch to AVERAGE.
The number of OPERABLE LPRMs is directly displayed on the meter upper scale.

D. Place the APRM meter Function switch to COUNT.
The number of OPERABLE LPRMs is directly displayed on the meter upper scale.

Answer: B

Answer Explanation

Answer Explanation: Per QCOP 0700-04, AVERAGE POWER RANGE MONITORING SYSTEM OPERATION, step F.1.c. states:

"To determine the number of LPRM Detectors feeding that APRM Channel, place METER FUNCTION switch to COUNT and divide the meter indication by five."

Distractor 1: Plausible because the APRM meter Function switch must be placed to COUNT. Incorrect because the number of OPERABLE LPRMs is not directly displayed.

Distractor 2: Plausible because the APRM meter upper scale is used in the determination of OPERABLE LPRMs. Incorrect because the Function Switch must be in COUNT and it is not a direct reading.

Distractor 3: Plausible because the APRM meter upper scale reading must be divided by 5 to determine the OPERABLE LPRMs. Incorrect because the Function Switch must be in COUNT.

Reference: QCOP 0700-04 Rev.18
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 215005 G2.1.30 AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR: CONDUCT OF OPERATIONS: Ability to locate and operate components, including local controls.

Importance: RO 4.4
10 CFR Part 55 Content: 41.7/45.7

Question Source: New
Question History: N/A
Comments: None

38 **ID: 2474644** **Points: 1.00**

Which of the following actions will stop an Automatic Depressurization System (ADS) Blowdown once it has automatically initiated?

A. Raising RPV water level to -45 inches.

B. Lowering Drywell pressure to +2.0 psig.

C. Placing the ADS INHIBIT switch to INHIBIT.

D. Placing ALL ADS valve control switches to OFF.

Answer: C

Answer Explanation

Answer Explanation: Per Electrical Drawing 4E-1462 Sh.1 & 2, when the Blowdown is initiated, the Low-Low RPV water level and High Drywell pressure signals are sealed in via the 287-106A/B and 287-107A/B relays. The same is true for Low-Low RPV water level for 8.5 minutes as it is sealed in via the same relays. The relays can be de-energized by the ADS INHIBIT switches on the 901-3 panel or 901-32 panel in the Aux Electric Room. The relays can also be deenergized by placing all Low Pressure ECCS pumps in PTL.

Distractor 1: Plausible because Low-Low RPV water level is an initiating condition and raising RPV water level above -59 inches resets the 8.5 minute timer. Incorrect because the Low-Low RPV water level condition is sealed in after 8.5 minutes or after 110 seconds if concurrent with high Drywell pressure.

Distractor 2: Plausible because high Drywell pressure is an initiating condition. Incorrect because the High Drywell pressure (2.5 psig) signal seals in and can only be reset using the DRYWELL PRESS RESET key on the 901-3 panel.

Distractor 3: Plausible because placing the control switch to OFF inhibits one of the ADS valve auto functions. Incorrect because it prevents ADS valve response to RPV pressure, ie. the relief valve function becomes inoperable. The ADS blowdown function is not affected.

Reference: 4E 1462 Sh.1 Rev. AO, 4E-1462 Sh.2 Rev.AP
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 218000 K4.05 Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design features and/or interlocks that provide for the following: Inhibiting automatic initiation of ADS logic

Importance: 4.2
10 CFR Part 55 Content: 41.7

Question Source: New
Question History: N/A
Comments: None

Test Answer Key

2023 ILT NRC RO Exam Final Version
Test ID: 369039

39 **ID: 2480258** **Points: 1.00**

Unit 1 was at rated power when the following conditions occur:

- 125 VDC Bus 1A-1 supply breaker trips
- RPS A is de-energized
- A LOCA results in RPV water level LOWERING to -75 inches.

What is the status of the Group 1 Isolation Valves?

A. ALL Group 1 Isolation Valves are CLOSED.

B. ALL Group 1 Isolation Valves have remained AS IS.

C. The Inboard MSIVs are OPEN. ALL other Group 1 Isolation valves are CLOSED.

D. The Inboard MSIVs are CLOSED. ALL other Group 1 Isolation valves have remained AS IS.

Answer: A

Answer Explanation

Answer Explanation: 125 VDC Bus 1A-1 provides power to the DC solenoids for the Inboard MSIVs. RPS A provides power to the AC solenoids for the Inboard MSIVs. The inboard MSIVs will close with no power to the solenoids. When -59 inches is reached during the LOCA, Division II PCIS Group I will be received and a full Group I isolation occurs and all valves not previously closed (Outboard MSIVs) will close.

Distractor 1: Plausible because PCI Division I did not process the low-low RPV water level signal due to loss of logic power, (Group I signal processed by Division II only). Incorrect because the Inboard MSIVs close on loss of power to the AC and DC solenoids. The Outboard MSIVs close because Division II PCI Group I logic processes the low-low RPV water level signal and Division I PCI logic fails safe on loss of power supply (RPS A).

Distractor 2: Plausible because the low-low RPV water level signal is not processed by Division I PCI Logic due to loss of power, (RPS A). Incorrect because both AC and DC solenoids for the Inboard MSIVs have lost power, RPS A and 125 VDC Bus 1A-1 respectively.

Distractor 3: Plausible because the Inboard MSIVs close on loss of power to the AC and DC solenoids and the low-low RPV water level signal was only processed by Division II PCI logic. Incorrect because Division I PCI logic fails safe on loss of power and Division II PCI logic processes the low-low RPV water level signal resulting in a full Group I actuation and isolation.

Reference: QOA 6900-08 Rev.21, QCAP 0200-10 Rev.60, 4E-1503A Rev.AX, 4E-1503B Rev.BE
Reference provided during examination: None

Cognitive level: High

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

K/A: 223002 K5.01 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUTOFF: Primary containment integrity

Importance: 4.1
10 CFR Part 55 Content: 41.5/45.3

Question Source: Bank
Question History: Quad Cities 2012 ILT NRC Exam
Comments: None

40 **ID: 2474650** **Points: 1.00**

An ATWS with a Group I isolation has occurred on Unit 1.
RPV water level is in band at -50 inches and STEADY.
RPV pressure is being controlled with relief valves between 600 and 1000 psig.

What is the status of the 1-203-3A Target Rock Relief Valve?

A. The valve can ONLY be operated once due to its accumulator design.

B. The valve must be left closed because the Drywell Pneumatic system has isolated.

C. The valve can be operated as needed because Nitrogen backup automatically supplies the Drywell Pneumatic System.

D. The valve can be operated as needed because Instrument Air backup automatically supplies the Drywell Pneumatic System.

Answer: C

Answer Explanation

Answer Explanation: The Drywell Pneumatic Air System provides pneumatic control to devices inside primary containment. In the normal lineup, the compressor takes a suction from the Drywell through two isolation valves, (AO 1-4720 and AO 1-4721). These suction valves close on a Group II isolation signal (0 inches RPV water level) tripping the compressor on low suction pressure. The Nitrogen system is lined up to supply the compressor loads through AO 1-4723 which automatically opens when the supply header pressure drops to 82 psig. Instrument Air is also a backup supply but is normally isolated by the 1-4799-207 valve.

Distractor 1: Plausible because the accumulator is designed for one operation when isolated from the Drywell Pneumatic and Instrument Air systems. Incorrect because the supply header pressure is maintained by the Nitrogen backup allowing valve operation if needed.

Distractor 2: Plausible because the Drywell Pneumatic System has isolated on a Group II. Incorrect because the supply header pressure is maintained by the Nitrogen backup allowing valve operation if needed.

Distractor 3: Plausible because Instrument Air is a backup to the Drywell Pneumatic System. Incorrect because it is isolated by the 1-4799-207 valve when the containment is inerted.

References: M-24 Sh.12 Rev. J, M-24 Sh.13 Rev. K, QOA 900-4 G-13 Rev.3, QCOP 0203-01 Rev.17
Reference provided during examination: None

Cognitive level: High

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

K/A: 239002 K1.06 Knowledge of the physical connections and/or cause and effect relationships between the SAFETY RELIEF VALVES and the following systems: Drywell instrument air/drywell pneumatics.

Importance: 3.6
10 CFR Part 55 Content: 41.2 to 41.9/45.7 to 45.8

Question Source: New
Question History: N/A
Comments: None

41 **ID: 2474651** **Points: 1.00**

The 1-2201-190 and 901-18 Unit 1 DFWLC cabinets are powered from:

A. Unit 1 Instrument Bus and 125V DC via UPS.

B. Unit 1 Instrument Bus and 125V DC (auctioneered).

C. Unit 1 Essential Service Bus and Unit 1 Instrument Bus (auctioneered).

D. Unit 1 Essential Service Bus and Unit 2 Essential Service Bus via UPS.

Answer: C

Answer Explanation

Answer Explanation: Per Electrical print, 4E-1417, the1-2201-190 and the 901-18 cabinet are supplied with two independent highly reliable power sources 120 VAC Instrument Bus and 120 VAC ESS Bus.

Distractor 1: Plausible because the Unit 1 Instrument Bus is one of the DFWLC power supplies. Incorrect because the other power supply is the Unit 1 Essential Service (ESS) Bus.

Distractor 2: Plausible because the Unit 1 Instrument Bus is one of the DFWLC power supplies. Incorrect because the other power supply is the Unit 1 Essential Service (ESS) Bus.

Distractor 3: Plausible because the Unit 1 ESS Bus is one of the DFWLC power supplies. Incorrect because the other power supply is the Unit 1 Instrument Bus.

References: QOM 1-6800-01 Rev.14, QOM 1-6800-02 Rev.12, 4E-1417 Rev.AG, QOA 6800-03 Rev.52
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 259002 K2.02 REACTOR WATER LEVEL CONTROL SYSTEM: Knowledge of electrical power supplies to the following: Feedwater coolant injection (FWCI) initiation logic

Importance: 3.4
10 CFR Part 55 Content: 41.7

Question Source: New
Question History: None
Comments: None

42 **ID: 2474654** **Points: 1.00**

A Unit 1 shutdown is in progress due to a 1B Recirc pump seal leak which resulted in a SLOWLY RISING Drywell pressure.

- RPV pressure is currently 700 psig with a cooldown in progress.
- Primary containment venting is in progress per QCOP 1600-01, DRYWELL PRESSURE RELIEF THROUGH SBGT.
- B SBGT is running with the 1/2B SBGTS TRAIN MODE SELECTOR SWITCH in START.
- Drywell pressure is currently 2.2 psig and LOWERING SLOWLY as a result of venting.
- RPS B is on reserve power.

With no operator action, what is the containment response if Bus 19 tripped on overcurrent?

A. Slowly rises due to the trip of 1/2B SBGT.

B. Continues to lower with the 1/2A SBGT running.

C. Continues to lower with the 1/2B SBGT running.

D. Slowly rises due to closure of containment vent valves on loss of Bus 19.

Answer: A

Answer Explanation

Answer Explanation: With Bus 19 tripped, the B SBGTS Fan and AIR Heater stop. The B SBGTS Dampers fail as is. The A train will NOT auto start after approx 25 seconds because the 1/2B SBGTS TRAIN MODE SELECTOR SWITCH IS in START. With RPS B on reserve power, containment 2-inch vent valve AO 1-1601-63 to the SBGTS will remain open. With no SBGTS Train operating, the continued leakage from the Recirc pump seal will cause Drywell pressure to slowly rise.

Distractor 1: Plausible because the A SBGT has an autostart feature on B SBGTS low flow. Incorrect because the B SBGTS did not autostart, it was manually started, therefore the A SBGT will not autostart on B train low flow.

Distractor 2: Plausible because the loss of Bus 19 does not change the valve and damper lineup. Incorrect because B SBGTS is tripped on loss of Bus 19.

Distractor 3: Plausible because a trip of Bus 19 also deenergizes RPS B, closing AO 1601-63 VENT TO SBGTS. Incorrect because RPS B is on reserve power MCC 15-2.

References: QCAN 912-2 B-6 Rev.06, QOA 7000-01 Rev.40, QCOP 1600-01 Rev.17, 4E-1400C Rev.AG
Reference provided during examination: None

Cognitive level: High

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

K/A: 261000 K3.03 Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on the following system or system parameters: Primary containment pressure (Mark I and II)

Importance: 3.7
10 CFR Part 55 Content: 41.7/45.6

Question Source: New
Question History: None
Comments: None

43 **ID: 2474681** **Points: 1.00**

Unit 1 is at 100% power.

- BOTH Standby Gas Treatment (SBGT) trains are in their normal Standby lineup.
- A loss of the Unit 1 Essential Service Bus (ESS) occurs.

Determine the SBGT damper lineup and status of the RX BLDG VENT CH B radiation monitor under these conditions.

(1) 1/2-7505B INLET DMPR
(2) RX BLDG VENT CH B radiation monitor

A. (1) Open
(2) Downscale

B. (1) Open
(2) Reading On Scale

C. (1) Closed
(2) Downscale

D. (1) Closed
(2) Reading On Scale

Answer: C

Answer Explanation

Answer Explanation: The B Reactor Building Vent Rad Monitor trips on a loss of ESS. The meter indication is downscale as it has no power. The trip of the B RB Vent Rad Monitor is however a SBGTS initiation signal to both trains. Logic power for B SBGTS is also Unit 1 ESS and is deenergized preventing an autostart. However, since "B" Train does NOT start, the “A” Train starts on "B" Train low flow for 25 seconds. The "A" Train logic is powered from U-2 ESS and gives an open signal to the 1-7503 and a close signal to 2-7503. When "A" Train starts, the 1/2-7505A opens. The "B" Train essentially stays in a Standby lineup with the 1/2-7505B closed.

Distractor 1: Plausible because the radiation monitor is downscale on a loss of power. Incorrect because the B Train does not start and the 1/2-7505B damper remains closed.

Distractor 2: Plausible because the 1/2-7505 damper opens on an initiation signal from Unit 1. Incorrect because the 1/2B SBGTS logic does not process the signal, (loss of logic power). The 1/2A SBGTS initiates closing the 1/2-7505 damper. Also the radiation monitor indication is downscale due to the power loss.

Distractor 3: Plausible because the 1/2-7505B damper will remain closed. Incorrect because the radiation monitor indication is downscale due to the power loss.

Reference: QCOP 7500-01 Rev. 22, 4E-1400A Sh.1 Rev.Z, 4E-1400C Rev.AG
Reference provided during examination: None

Cognitive level: High

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

K/A: 261000 K6.04 Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the STANDBY GAS TREATMENT SYSTEM: Radiation monitoring system

Importance: 3.4
10 CFR Part 55 Content: 41.7/45.7

Question Source: New
Question History: None
Comments: None

44 **ID: 2480344** **Points: 1.00**

When establishing a Clearance Order boundary, why are 4KV circuit breakers racked out instead of ONLY tagging the control switch?

A. To allow remote operation of the breaker.

B. To allow remote indication of breaker position.

C. To allow control power to remain energized in the breaker.

D. To ensure the breaker's control and indication circuits are not functional.

Answer: D

Answer Explanation

Answer Explanation: Per QCOP 6500-04, RACKING OUT A 4160 VOLT HORIZONTAL TYPE AMHG, G26, OR GEHR CIRCUIT BREAKER, removing a breaker from service is done by tagging the control switch, removing the CLOSE and TRIP fuse blocks, racking the breaker out to the DISCONNECT position, and then fully removing the breaker from its cubicle. A protective cover is placed over the cubicle opening and a clearance order tag is placed on the circuit breaker. This ensures the breaker cannot be operated and provides worker protection.

Distractor 1: Plausible because a control switch with an INFO card may be manipulated. Incorrect because the breaker could potentially be operated and pose a worker risk.

Distractor 2: Plausible because a control switch with an INFO card may be manipulated. If taken out of PTL, it will provide breaker position indication. Incorrect because worker protection is not ensured by tagging the control switch only. When the breaker is removed and tagged out properly there will be no indication of breaker position as the CLOSE and TRIP fuse blocks are also removed.

Distractor 3: Plausible because tagging the control switch ONLY leaves the control power fuse blocks at the breaker cubicle installed. Incorrect because this would not provide adequate protection as the breaker could potentially be operated.

Reference: QCOP 6500-04 Rev.39
Reference provided during examination: None

Cognitive level: Memory.

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

K/A: 291008 K1.01 AC ELECTRICAL DISTRIBUTION COMPONENTS: Breakers, Relays, and Disconnects: Purpose for racking out breakers (deenergize components and associated control and indication circuits)

262001 A4.01 SF6 AC ELECTRICAL DISTRIBUTION: Ability to manually operate and/or monitor in the control room: Breakers and disconnects.

Importance: 3.6 / 3.7
10 CFR Part 55 Content: 41.7 / 41.7

Question Source: New
Question History: None
Comments: None

45 **ID: 2474700** **Points: 1.00**

Unit 1 is in cold shutdown. Electrical Maintenance has just finished performing an inspection of the ESS UPS Inverter.

- Annunciator 901-8 E-9, ESS SERV BUS ON EMERG SPLY, is in alarm and acknowledged (solid).
- The ESS UPS is in a normal lineup and ready to accept load.
- An EO has closed the ASCO Switch Normal Supply Breaker, supplying the output of the ESS UPS to the ESS ABT.

AFTER the EO places the ESS ABT 'TRANSFER CONTROL' switch in the 'RETRANSFER TO NORMAL' position, what is the status of the alarms?

Annunciator 901-8 E-9, ESS SERV BUS ON EMERG SPLY, ___(1)___ AND 901-8 E-9, ESS SERV UPS ON DC OR ALT AC, ___(2)___.

A. (1) REMAINS in alarm (solid)
(2) is NOT in alarm

B. (1) is NO LONGER in alarm (slow flash)
(2) is NOT in alarm

C. (1) REMAINS in alarm (solid)
(2) IS now IN alarm (fast flash)

D. (1) is NO LONGER in alarm (slow flash)
(2) IS now IN alarm (fast flash)

Answer: B

Answer Explanation

Answer Explanation: The ESS ABT supplies power to the ESS Bus from either the Normal Supply (ESS UPS output), or the Emergency Supply (MCC 18-2). Annunciator 901-8 E-9, ESS SERV BUS ON EMERG SPLY, indicates that the ESS bus is currently being supplied by the Emergency supply (MCC 18-2) through the ESS ABT. When the EO moves the transfer switch on the ABT to 'RETRANSFER', the ESS Bus will be supplied power from the output of the ESS UPS, and the 901-8 E-9 alarm in the control room will clear.

Distractor 1: Plausible because this would be the indication if the ESS Bus feed did not transfer, (incorrectly thinks additional manipulations are required). Incorrect because under the conditions described, the ESS Bus feed will transfer to the UPS.

Distractor 2: Plausible because the UPS has three feeds, Bus 18, 250 VDC, and Bus 17 (ALT). If the UPS Static Switch was lined up to the alternate source, annunciator 901-8 E-8 would alarm on the transfer. Incorrect because the ESS UPS is in the normal lineup with Bus 18 supplying the Static Switch.

Distractor 3: Plausible because annunciator 901-8 E-9 would clear. Incorrect because the ESS power source transfers to the UPS Normal feed (Bus 18) and not to the Alternate AC feed (Bus 17).

Reference: QOP 6800-03 Rev.39, QOA 900-8 E-8 Rev.04, QOA 900-8 E-9 Rev.03
Reference provided during examination: None

Cognitive level: High

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

K/A: 262002 A4.01 Ability to manually operate and/or monitor the UNINTERRUPTABLE POWER SUPPLY (AC/DC) in the control room: Transfer of power sources

Importance: 3.1
10 CFR Part 55 Content: 41.7/45.5

Question Source: Bank
Question History: 2012 ILT NRC Exam
Comments: Change made to make the distractor choices shorter. Added common portions of the old long distractors to the new stem. Editorial changes only.

46 **ID: 2480362** **Points: 1.00**

Unit 1 is in a Refuel outage.

The Electric plant is in a normal lineup with the following exceptions:

- Bus 14-1 and Bus 24-1 are crosstied.
- Bus 15 and Bus 16 are crosstied.

Bus 14 is then de-energized.

What is the status of the Computer UPS system?

A. The Computer UPS Static Switch will transfer to Bus 26 with no interruption of power to loads.

B. The Computer UPS will have to be manually transferred to Bus 26 to restore power to the loads.

C. The Computer UPS 250VDC battery will immediately assume Computer UPS loads with no interruption of power to loads.

D. The Computer UPS will have to be manually transferred to the Computer UPS 250VDC battery to restore power to the loads.

Answer: C

Answer Explanation

Answer Explanation: Per QOA 9900-01, step B.1 (Automatic Actions), an Auto transfer of computer UPS from normal AC supply to UPS 250 VDC battery occurs on loss of feed from Bus 17. Bus 17 is fed from Bus 14 for a Normal lineup and thus is lost when Bus 14 is deenergized. Therefore the Computer UPS loads will auto transfer to the UPS 250 VDC battery.

Distractor 1: Plausible because the Computer UPS loads will auto transfer to Bus 26 if the Inverter has failed. Incorrect because the Inverter has not failed, therefore on a loss of Bus 17 (Normal) the Computer UPS loads will by auctioneered to the Computer UPS 250 VDC battery.

Distractor 2: Plausible because Computer UPS loads can be manually transferred to Bus 26 through the Manual Bypass Switch. Incorrect because on a loss of Bus 17 (Normal) the Computer UPS loads will by auctioneered to the Computer UPS 250 VDC battery.

Distractor 3: Plausible because the loads will be carried by the Computer UPS 250 VDC battery. Incorrect because the transfer is automatic.

References: QOA 9900-01 Rev.14, QCOP 6800-04 Rev.09
Reference provided during examination: None

Cognitive level: High

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

K/A: 263000 K1.01 Knowledge of the physical connections and/or cause effect relationships between the DC ELECTRICAL DISTRIBUTION and the following systems: AC electrical distribution

Importance: 4.0
10 CFR Part 55 Content: 41.2 to 41.9/45.7 to 45.8

Question Source: New
Question History: N/A
Comments: None

47 **ID: 2474701** **Points: 1.00**

The 1/2 250 VDC Battery Charger can be fed from which two MCC's?

A. 15-2 or 25-2

B. 18-2 or 19-2

C. 18-2 or 28-2

D. 28-2 or 29-2

Answer: C

Answer Explanation

Answer Explanation: Per QOM 1-6700-T23 and QOM 2-6700-T21, MCC 18-2 or 28-2 cam power the ½ 250 VDC Battery Charger (key interlock at MCC 28-2 prevents closing both feeds into the ½ charger).

Distractor 1: Plausible because MCC 15-2 supplies power to the Non-Essential 250 VDC Battery Charger and MCC 25-2 is a 480 VAC MCC. Incorrect because the 1/2 250 VDC Battery Charger feeds are MCC 18-2 and MCC 28-2.

Distractor 2: Plausible because MCC 18-2 is a feed to the 1/2 250 VDC Battery Charger and MCC 19-2 feeds 125 VDC Battery Charger 1. Incorrect because MCC 28-2 is the other feed to the 1/2 250 VDC Battery Charger.

Distractor 3: Plausible because MCC 28-2 is a feed to the 1/2 250 VDC Battery Charger and MCC 29-2 feeds the 125 VDC Battery Charger 2. Incorrect because MCC 18-2 is the other feed to the 1/2 250 VDC Battery Charger.

References: QOM 1-6700-T23 Rev.09, QOM 2-6700-T21 Rev.09
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 263000 K2.03 DC ELECTRICAL DISTRIBUTION: Knowledge of electrical power supplies to the following: Battery chargers

Importance: 3.5
10 CFR Part 55 Content: 41.7

Question Source: New
Question History: N/A
Comments: None

48 **ID: 2480388** **Points: 1.00**

Unit 1 was at 100% power when a Loss of Offsite Power and a LOCA occurred.

- Annunciator 901-8 G-5, DIESEL GEN 1/2 RELAY TRIP, is in alarm.
- There are NO overcurrent alarms.

(1) What is the status of Bus 13-1?
(2) What action should be taken?

A. (1) Bus 13-1 is deenergized.
(2) Reset Lockout Relay 186-DG 1/2 at Bus 13-1 per QCOA 6600-02, DIESEL GENERATOR 1/2 FAILS TO START

B. (1) Bus 13-1 is deenergized.
(2) Place the 1/2 EDG Control Switch to START per QCOA 6600-02, DIESEL GENERATOR 1/2 FAILS TO START

C. (1) Bus 13-1 is energized.
(2) Verify and adjust frequency / voltage as necessary per QCOP 6600-05, SHARED UNIT DIESEL GENERATOR START UP

D. (1) Bus 13-1 is energized.
(2) Dispatch an EO to monitor 1/2 EDG operation per QCOP 6600-05, SHARED UNIT DIESEL GENERATOR START UP

Answer: A

Answer Explanation

Answer Explanation: Per QCOA 6600-02, step D.1: IF alarm 901-8 G-5 DIESEL GEN 1/2 RELAY TRIP is alarming (indicative of a possible spurious high differential current trip), THEN:
a. Direct Operator to reset Lockout Relay 186-DG 1/2 at Bus 13-1, cubicle 1.
b. IF DG failed to auto-start AND is required for emergency operations, THEN:
(1) Place DIESEL GEN 1/2 CONTROL SWITCH to START at Panel 901-8.
(2) Verify DG starts by indication of DG frequency and voltage at Panel 901-8 or 902-8.

Distractor 1: Plausible because Bus 13-1 is deenergized. Incorrect because the DG cannot be restarted until the lockout relay, 186-DG 1/2, is reset.

Distractor 2: Plausible because adjusting voltage and frequency as necessary are correct actions on a running DG. Incorrect because the DG has tripped and Bus 13-1 is deenergized.

Distractor 3: Plausible because dispatching an EO to the DG is a correct action. Incorrect because Bus 13-1 is deenergized.

References: QCOA 6600-02 Rev. 23, QCAN 901-8 G-5 Rev. 06
Reference provided during examination: None

Cognitive level: High

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

K/A: 264000 A2.11 Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Failure of emergency generator to start/load

Importance: 4.6
10 CFR Part 55 Content: 41.5/43.5/45.6

Question Source: New
Question History: N/A
Comments: None

49 **ID: 2480853** **Points: 1.00**

Which of the following would be a concern if there was a breakdown of Instrument Air Dryer desiccant and subsequent carryover into the Instrument Air System?

1. Endangering personnel operating on station supplied breathing air.
2. Loss of Low Flow Feedwater Regulating Valve control.
3. Control rods drifting into the core.
4. Inboard MSIVs drift closed.

A. 1 and 2 ONLY

B. 2 and 3 ONLY

C. 3 and 4 ONLY

D. 1 and 4 ONLY

Answer: B

Answer Explanation

Answer Explanation: Excessive moisture can result in desiccant carryover causing high differential pressures across the after filter (1/2, 1A and U-2 IAC only). A high d/p can be caused by dryer desiccant breakdown or oil carryover. After the filter becomes clogged or torn, it allows the desiccant to be blown down plugging orifices and Instrument Air lines thus preventing normal operation of Instrument Air operated valves. Instrument Air provides air to all listed except breathing air which is provided by the Service Air system.

Distractor 1: Plausible because clogging of the air supply line to the Low Flow Feedwater Regulating Valve (LFFRV) could occur. Incorrect because the Instrument Air system is not used for breathing air.

Distractor 2: Plausible because Instrument Air supplies the scram air header. Loss of scram air header pressure will result in control rod drift. Incorrect because the Inboard MSIVs are supplied by the Drywell Pneumatic Air system. The Outboard MSIVs are supplied by Instrument Air and may drift closed.

Distractor 3: Plausible because the MSIVs are air operated valves. Incorrect because the Instrument Air system is not used for breathing air and the Inboard MSIVs are supplied by the Drywell Pneumatic Air system.

References: QOA 4700-06 Rev. 29
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 300000 K5.15 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: High moisture content in instrument air

Importance: 3.1
10 CFR Part 55 Content: 41.5/45.3

Question Source: New
Question History: N/A
Comments: None

50 **ID: 2474739** **Points: 1.00**

Unit 2 is in a Refuel Outage.

- Cavity flooded / Fuel Pool gates are REMOVED
- 2A and 2B Fuel Pool Cooling pumps are ON
- CRD pump suction is from Fuel Pool Reject
- RWCU system is OOS
- Condensate system is OOS
- Fuel Pool temperature is 100°F

A partial core offload is in progress when a loss of RBCCW occurs.

What action, if any, must be taken to continue fuel moves in the Spent Fuel Pool?

A. Fuel moves cannot be resumed until the Condensate system is restored.

B. No action is necessary, both Fuel Pool Cooling Water pumps are still operating.

C. Start the RHR system in Fuel Pool Cooling Assist mode per QCOP 1000-11, RHR FUEL POOL COOLING ASSIST.

D. Start a feed and bleed using the Fire System for water addition and reject to the CCSTs per QCOA 1900-11, LOSS OF FUEL POOL COOLING WITH UNIT SHUTDOWN FOR REFUELING.

Answer: C

Answer Explanation

Answer Explanation: The RBCCW system provides cooling water to the Fuel Pool Cooling Heat Exchangers. QCOA 1900-02, FUEL STORAGE POOL HIGH TEMPERATURE and QCOA 1900-03, LOSS OF FUEL POOL COOLING WITH UNIT SHUTDOWN FOR REFUELING direct starting RHR in Fuel Pool Cooling Assist mode if necessary when normal cooling is lost.

Distractor 1: Plausible because starting a feed and bleed would control Fuel Pool and Reactor Cavity temperatures. Incorrect because this would require a reject path with a large capacity such as RWCU system. The question stem has the Condensate and RWCU systems OOS.

Distractor 2: Plausible because fuel pool temperature is presently < 125°F and may continue. Incorrect because temperature will rise even with circulation provided by the Fuel Pool Cooling Water pumps. Without any cooling and no further actions, a stoppage of fuel moves and possible evacuation of the Refuel Floor due to airborne radioactive iodine (140°F) will occur.

Distractor 3: Plausible because starting a feed and bleed would control Fuel Pool and Reactor Cavity temperatures. Incorrect because the Fire System is a low quality water source. Higher quality water systems are available and would be preferred.

Distractor 3: Plausible because starting a feed and bleed would control Fuel Pool and Reactor Cavity temperatures. Incorrect because this would require a reject path with a large capacity such as RWCU system. The question stem has the Condensate and RWCU systems OOS.

References: QCOA 1900-02 Rev.14, QCOA 1900-03 Rev.15
Reference provided during examination: None

Cognitive level: High

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

K/A: 400000 A2.09 Ability to (a) predict the impacts of the following on the COMPONENT COOLING WATER SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of cooling to spent fuel pool cooling system heat exchanger

Importance: 4.2
10 CFR Part 55 Content: 41.5/43.5/45.6

Question Source: New
Question History: N/A
Comments: None

51 **ID: 2474778** **Points: 1.00**

Which of the following sources provides cooling to the Emergency Diesel Generator Heat Exchangers?

A. RBCCW

B. TBCCW

C. Well Water

D. River Water

Answer: D

Answer Explanation
Answer Explanation: Per Drawing M-37 and M-22 Sh.3, Service Water is shown to provide cooling water to the Emergency Diesel Generators. **Distractor 1:** Plausible because it is a component cooling water system. Incorrect because it does not provide cooling water to the Emergency Diesel Generator Heat Exchanger. **Distractor 2:** Plausible because it is a component cooling water system. Incorrect because it does not provide cooling water to the Emergency Diesel Generator. **Distractor 3:** Plausible because it provides cooling to the EDGCWP seals. Incorrect because it does not provide cooling water to the Emergency Diesel Generator Heat Exchanger. **References:** M-22 Sh.3 Rev.AB, M-37 Rev.BM, **Reference provided during examination:** None **Cognitive level:** Memory **Level (RO/SRO):** RO **Tier:** 2 **Group:** 1 **K/A:** 510000 K1.04 Knowledge of the physical connections and/or cause and effect relationships between the SERVICE WATER SYSTEM and the following systems: Emergency generators (diesel/jet) **Importance:** 3.4 **10 CFR Part 55 Content:** 41.2 to 41.9/45.7 to 45.8 **Question Source:** New **Question History:** N/A **Comments:** None

52 **ID: 2480854** **Points: 1.00**

Unit 1 is operating at full load in a normal Service Water lineup with the 2A Service Water pump in operation.

Service Water pressure begins to lower at 2 psig per minute.

As Service Water system pressure lowers to 75 psig, the __(1)__ will autostart.

If Service Water system pressure continues to lower to 68 psig, the MO-1/2-3906, FIRE PROT SW SPLY VLV, will __(2)__.

A. (1) 1/2A Fire Diesel pump
(2) OPEN

B. (1) 1B Service Water pump
(2) CLOSE

C. (1) 1/2B Fire Diesel pump
(2) OPEN

D. (1) 1A Service Water pump
(2) CLOSE

Answer: B

Answer Explanation

Answer Explanation: Per QCOA 3900-01, SERVICE WATER SYSTEM FAILURE, the automatic actions listed in step B are:

B.1. IF Service Water header pressure decreases to 75 psig, THEN 1B Service Water Pump will Auto-start if available and in standby lineup.

B.2. Diesel Fire Pumps automatically start when fire header pressure fails below:

a. On A Diesel Fire Pump, 1/2-4101A, 70 psig.

b. On B Diesel Fire Pump, 1/2-4101B, 65 psig.

B.3. MO 1/2-3906 will close if service water pressure decreases to 68 psig.

Distractor 1: Plausible because the 1/2A Fire Diesel pump autostarts on 70 psig Service Water Header pressure. Incorrect because MO 1/2-3906 valve closes at 68 psig system pressure.

Distractor 2: Plausible because 1/2-3906 valve opens at 68 psig. Incorrect because the 1/2B Fire Diesel autostarts at 65 psig system pressure.

Distractor 3: Plausible because there is a Service Water pump autostart on low Service Water Header pressure. Additionally the 1/2-3906 valve closes on low system pressure. Incorrect because the autostart is on the 1B Service water pump at 75 psig system pressure.

References: QCOA 3900-01 Rev.26
Reference provided during examination: None

Cognitive level: High

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

K/A: 510000 A3.02 Ability to monitor automatic operation of the SERVICE WATER SYSTEM, including: Valve alignment

Importance: 3.3
10 CFR Part 55 Content: 41.7/45.7

Question Source: New
Question History: N/A
Comments: None

53 **ID: 2474819** **Points: 1.00**

Unit 1 is operating at 100% power with the 1A CRD pump operating.
Unit 2 is in MODE 4 during an outage with Buses 23 and 24 de-energized.

- Bus 17 trips on overcurrent.
- Annunciator 901-5 C-2, CRD PUMP A/B LOW SUCTION PRESSURE, alarms.

Which of the following actions will restore CRD system operation?

A. Swap suction filters and restart the 1A CRD pump per QCOA 0300-01, CONTROL ROD DRIVE PUMP FAILURE.

B. Swap drive water filters and restart the 1A CRD pump per QCOP 0300-13, CRD DISCHARGE FILTER OPERATION.

C. Manually OPEN MO 1-301-2B, 1B PMP DISCH VLV, and start the 1B CRD pump per QCOP 0300-23, CRD PUMP CHANGE OVER.

D. Lineup the CRD crosstie from the Unit 2 CRD system per QCOP 0300-33, CRD PUMP CROSS-TIE OPERATION USING UNIT 2 CRD PUMPS.

Answer: A

Answer Explanation

Answer Explanation: Annunciator 901-5 C-2 alarming results in a trip of the running CRD pump. QCAN 901(2)-5 C-2 refers the operator to QCOA 0300-01, CONTROL ROD DRIVE PUMP FAILURE. In the case of a CRD pump trip accompanied by annunciator 901-5 C-2, the QCOA directs the operator to swap CRD suction filters or place both suction filters online. The discharge valve for the 1B CRD pump is powered from MCC 17-2 and cannot be operated from the 901-5 panel, so the 1B CRD pump should not be started.

Distractor 1: Plausible because the trip of the CRD pump was caused by CRD system filter clogging. Incorrect because the CRD pump suction filters must be swapped; swapping the drive water filters will have no effect on the CRD pump low suction pressure condition.

Distractor 2: Plausible because the 1B CRD pump can be used if the 1-301-2B is manually opened. Incorrect because the CRD pump suction filters must still be swapped in order to avoid an auto-trip of the 1B CRD pump on low suction pressure.

Distractor 3: Plausible because it is procedurally allowed by QCOP 0300-33. Incorrect because the Unit 2 CRD pumps are unavailable due to the Unit 2 outage activities on Bus 23 and Bus 24.

References: QCAN 901(2)-5 C-2 Rev.08, QCOA 0300-01 Rev.18
Reference provided during examination: None

Cognitive level: High

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

K/A: 201001 A2.03 Ability to (a) predict the impacts of the following on the CRD HYDRAULIC SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Power supply failures

Importance: 4.0
10 CFR Part 55 Content: 41.2/45.5-7/41.10/45.1-6/41.12-13

Question Source: New
Question History: N/A
Comments: None

54 **ID: 2474820** **Points: 1.00**

Unit 1 is shutdown.
The indications below are observed when the RECIRC ASD 1A control switch is momentarily placed to START, then returned to NORMAL:

- RECIRC ASD ‘OFF’ and ‘TRIP’ indicating lights are illuminated.
- PRE-CHARGE STATUS ‘READY’ light extinguishes and ‘IN PROGRESS’ light illuminates.
- RECIRC ASD ‘OFF’ and ‘TRIP’ lights extinguishes and ‘ON’ light illuminates.
- PRE-CHARGE STATUS ‘IN PROGRESS’ light extinguishes.
- PRE-CHARGE STATUS ‘COMP’ light illuminates and then extinguishes.

Which event do the indications above describe?

A. Recirc Pump trip

B. Recirc Pump start

C. Aborted Pre-Charge Sequence

D. Completed Pre-Charge Sequence

Answer: D

Answer Explanation

Answer Explanation: QCOP 0202-43, REACTOR RECIRCULATION SYSTEM STARTUP, step F.6 Note describes the Pre-Charge Sequence as follows:

Step F.6.k initiates the following sequence of actions:
1. RECIRC ASD 1(2)A/1(2)B BKR 'OFF' and 'TRIP' indicating lights are illuminated.
2. PRE-CHARGE STATUS 'READY' light is extinguished and 'IN PROGRES' light is illuminated.
3. RECIRC ASD 1(2)A/1(2)B 'OFF' and 'TRIP' lights are extinguished, and 'ON' light is illuminated.
4. PRE-CHARGE STATUS 'IN PROGRES' light is extinguished and 'COMP' light is illuminated.

IF breaker does NOT close within 3 seconds of completing the pre-charge sequence, THEN the sequence must be repeated.

Step F.6.k. is placing the ASD control switch to START, then returning it to NORMAL

Distractor 1: Plausible because the RECIRC ASD 1A BKR 'OFF' and 'TRIP' indicating lights illuminate. Incorrect because the 1A ASD has not been started. The 1A ASD is started when the ASD-A START pushbutton is depressed.

Distractor 2: Plausible because pumps are started when a control switch is taken to START. Incorrect because the ASDs are started when the ASD-A START pushbutton is depressed.

Distractor 3: Plausible because several of the indications may occur before the sequence is aborted. Incorrect because the above indications describe a completed Pre-Charge sequence.

References: QCOP 0202-43 Rev.25
Reference provided during examination: None

Cognitive level: High

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

K/A: 202001 A3.10 Ability to monitor automatic operation of the RECIRCULATION SYSTEM including: VFD start sequence

Importance: 3.4
10 CFR Part 55 Content: 41.7/45.7

Question Source: New
Question History: N/A
Comments: None

55 **ID: 2474839** **Points: 1.00**

Unit 1 is operating at 50% power with a startup in progress when the following occurs:

- Annunciators 901-5 G-5, RPIS INOP, and 901-5 B-3, ROD WORTH MIN BLOCK, alarm.
- Position indication for six control rods have been lost on the Full Core Display and on the RWM.
- The same six control rods also indicate BAD on an OD-7 printout.

The NSO selects a control rod and observes the white backlight is LIT on the Select Matrix and Four Rod Display, but **NOT** LIT on the Full Core Display.

How is control rod movement from the 901-5 panel affected?

A. Rods may be inserted using Emergency Rod In.

B. Rods can be inserted by depressing the Manual Scram pushbuttons.

C. In-sequence control rods may be inserted or withdrawn normally using the RMCS.

D. Rods indicating "??" on the RWM may be INSERTED to a known position, all other control rod motion is blocked.

Answer: B

Answer Explanation

Answer Explanation: Per QCAN 901-G-5, RPIS INOP, the automatic action is a Reactor Manual Control Select Block. Relay –28/-138 drops out on an RPIS inop signal. This prevents the Rod Select Relays from energizing when a control rod is selected on the Pushbutton Matrix. The pushbutton will backlight and the Four Rod Display "Selected Rod" light will light, but the Directional Control Valve Solenoids cannot be energized preventing control rod movement with the RMCS. The control rods can only be scrammed until the Select Block is reset.

Distractor 1: Plausible because the Emergency In function bypasses the Rod Control Relays and directly energizes 305-121 and 305-123 Drive In Directional Solenoids, (4E-1415). Incorrect because the Rod Select Relays cannot be energized, therefore the Directional Control Valve solenoids for any selected rod cannot be energized until the Select Block is reset.

Distractor 2: Plausible because some of the indications for a selected control rod are present, ie. Select Matrix and Four Rod Display. Incorrect because the Rod Select Relays cannot be energized. The Select Block must be reset to move control rods.

Distractor 3: Plausible because the RWM will allow insertion of control rods to a known position with an RPIS failure. Incorrect because control rods cannot be moved until the Select Block is reset.

References: QCOA 0280-01 Rev.17, QCAN 901(2)-5 G-5 Rev.06, 4E-1410 Rev.I, 4E-1413 Rev.Q, 4E-1414 Rev.AL, 4E-1415 Rev.T
Reference provided during examination: None

Cognitive level: High

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

K/A: 214000 K5.01 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the ROD POSITION INFORMATION SYSTEM: Rod position indication failures

Importance: 3.5
10 CFR Part 55 Content: 41.5/45.3

Question Source: New
Question History: N/A
Comments: None

56 **ID: 2474840** **Points: 1.00**

Which of the following conditions will generate a Rod Withdrawal Block by the Rod Block Monitor?

A. Out of sequence control rod selected.

B. Reference APRM fails downscale.

C. Flow Converter 1 output at 110%.

D. Failure to null.

Answer: D

Answer Explanation

Answer Explanation: QCAN 901(2)-5 A-7, RBM HIGH OR INOP, lists the signals which generate a Rod Block Monitor control rod withdrawal block. These are:

1. RBM HIGH: 0.65W + 52.5% increasing.
2. RBM INOP:
   a. No Rod selected.
   b. More than one rod selected.
   c. Failure to null.
   d. Mode switch not in OPERATE.
   e. Module unplugged.
   f. Less than 50% assigned inputs.
   g. Nulling sequence.

Distractor 1: Plausible because the RWM generates a rod block when an out of sequence rod is selected. Incorrect because it is NOT an RBM generated rod out block.

Distractor 2: Plausible because an APRM downscale with the Reactor Mode Switch in RUN will generate a control rod block. Incorrect because it is NOT an RBM generated rod out block.

Distractor 3: Plausible because a Flow Converter 110% (increasing) channel output will cause a rod out block. Incorrect because it is NOT an RBM generated rod out block.

References: QCAN 901(2)-5 A-7 Rev.07
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 215002 K4.01 Knowledge of ROD BLOCK MONITOR SYSTEM design features and/or interlocks that provide for the following: Rod withdrawal blocks

Importance: 3.9
10 CFR Part 55 Content: 41.7

Question Source: New
Question History: N/A
Comments: None

57 **ID: 2474941** **Points: 1.00**

Unit 2 has just completed leak inspections for the hydrostatic pressure test.

Reactor pressure has been lowered to 800 psig for performance of QCIS 0200-22, REACTOR VESSEL EXCESS FLOW CHECK VALVE FUNCTIONAL TEST on the 2-2202-6 rack.

A valving error resulted in an improper isolation of Reactor Water Level Instrument LT 2-0263-58B.

As a result, the reference leg of LT 2-0263-58B was pressurized.

Which of the following describes the effect on LT 2-0263-58B water level indication and actuations?

A. Rising level and a HPCI Pump trip.

B. Rising level and a Main Turbine trip.

C. Lowering level and a full reactor scram.

D. Lowering level and a 1/2 reactor scram only.

Answer: D

Answer Explanation

Answer Explanation: The 2-2202-6 rack contains the LT 2-0263-58A and 58B reactor water level instruments. The 58A level instrument feeds RPS A (A2 trip string) and the 58B feeds RPS B (B2 trip string) therefore due to the lowering indicated level (result of a reference leg pressurization) the reactor will scram on a 0" signal. Since only LT 2-0263-58B was improperly isolated, the logic for a 1/2 scram on RPS Channel B is met.

Distractor 1: Plausible because a higher indicated water level can result from pressurizing an instrument leg (variable). Incorrect because LT 2-0263-58B does not input into the HPCI Pump trip logic.

Distractor 2: Plausible because a higher indicated water level can result from pressurizing an instrument leg (variable). Incorrect because the reference leg was pressurized and the Main Turbine trip logic is initiated from LT 2-0263-23B and LT 2-0263-23D.

Distractor 3: Plausible because pressurizing the reference leg results in a lower indicated water level. Incorrect because the logic for a full reactor scram is not met.

References: QOA 6800-04 Rev.20, QCOA 6800-05 Rev.08
Reference provided during examination: None

Cognitive level: High

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

K/A: 216000 A1.02 Ability to predict and/or monitor changes in parameters associated with operation of the NUCLEAR BOILER INSTRUMENTATION including: Removing or returning a sensor (transmitter) to service

Importance: 3.1
10 CFR Part 55 Content: 41.5/45.5

Question Source: New
Question History: N/A
Comments: None

58 **ID: 2474960** **Points: 1.00**

The following conditions exist on Unit 1 during a LOCA:

- Reactor water level is -30 inches and LOWERING.
- Drywell pressure is 3.4 psig and LOWERING.
- Torus pressure is 3.0 psig and LOWERING.
- RHR Loop A/B CONTAINMENT CLG PERMISSIVE SWITCH 17 is in the ON position.
- RHR Loop A/B RHR SW START PERMISSIVE SWITCH 19 is in the MANUAL OVERRIDE position.
- Torus and Drywell sprays are ON.
- All other switches are in their NORMAL positions.

What will be the impact to containment sprays if reactor water level drops to < -191 inches?

A. ONLY Torus sprays will isolate.

B. ONLY Drywell sprays will isolate.

C. Drywell and Torus sprays will isolate.

D. Drywell and Torus sprays will continue to operate.

Answer: C

Answer Explanation

Answer Explanation: Since the CONTAINMENT COOLING 2/3 LEVEL & ECCS INIT BYPASS keylock switch is NOT in the MANUAL OVERRD position, Torus and Drywell sprays (and Torus Cooling valves) will be isolated, providing full RHR flow to the reactor.

Distractor 1: Plausible because Torus sprays will isolate. Incorrect because Drywell sprays will also isolate.

Distractor 2: Plausible because Drywell sprays will isolate. Incorrect because Torus sprays will also isolate.

Distractor 3: Plausible because the CONTAINMENT CLG PERMISSIVE SWITCH 17 bypasses a LOCA signal (2.5 psig DW pressure) which isolates Drywell and Torus sprays. Incorrect because it does not bypass the 2/3 core height (-191 in.) containment cooling isolation signal.

Reference: QCAN 901(2)-3 A-7 Rev.02, 4E-1438E Rev.AO
Reference provided during examination: None

Cognitive level: High

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

K/A: 226001 A4.07 Ability to manually operate and/or monitor the RHR/LPCI: CONTAINMENT SPRAY MODE SYSTEM MODE in the control room: Valve logic reset/bypass/override

Importance: 3.8
10 CFR Part 55 Content: 41.7/45.5 to 45.8

Question Source: New
Question History: N/A
Comments: None

59 **ID: 2474978** **Points: 1.00**

Unit 1 is at 100% power after a refueling outage.

Unit 1 Fuel Pool cleanup work is in progress.

When the REFUEL BRIDGE INTERLOCK BYPASS SWITCH in Panel 901-28 was placed in BYPASS prior to startup, a malfunction in the keylock switch occurred resulting in the switch contacts remaining in the NORM position.

The crew is positioning the Refuel Bridge over the Reactor Cavity Shield Plugs.

What effect, if any, does this have on control rod movement?

(Reference provided**)**

A. No effect on control rod movement

B. Rod Out Block #1 ONLY

C. Rod Out Block #2 ONLY

D. Rod Out Block #1 and Rod Out Block #2

Answer: C

Answer Explanation

Answer Explanation: With the REFUEL BRIDGE INTERLOCK BYPASS SWITCH in NORM, the control rod withdrawal blocks generated from the refuel bridge position are active. Electrical drawing 4E-1411 shows with the Mode Switch in RUN and the refuel bridge moved over the core, Rod Out Block #2 is received.

Distractor 1: Plausible because the refuel bridge can be moved over the core without a resultant control rod block when the REFUEL BRIDGE INTERLOCK BYPASS SWITCH is in BYPASS. Incorrect because a control rod out block will occur if it is moved near/over the core with the switch in NORM.

Distractor 2: Plausible because a rod out block is actuated under these conditions. Incorrect because Rod Out Block #2 Only is received.

Distractor 3: Plausible because a rod out block is actuated under these conditions. Incorrect because Rod Out Block #2 Only is received.

References: 4E-1411 Rev.W, 4E-1414 Rev.AL, QCFHP 0100-01 Rev.40
Reference provided during examination: 4E-1411, 4E-1414

Cognitive level: High

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

K/A: 234000 K3.01 Knowledge of the effect that a loss or malfunction of the FUEL HANDLING will have on the following systems or system parameters: Reactor manual control system

Importance: 2.8
10 CFR Part 55 Content: 41.7/45.4

Question Source: New
Question History: N/A
Comments: None

60 **ID: 2475039** **Points: 1.00**

A Unit 1 startup is in progress.

- The Main Turbine is RESET
- RPV pressure is 50 psig and RISING SLOWLY
- CLOSE VALVES is selected on the on <CONTROL> <SPEED-LOAD> screen

Note: See DEHC screen for CLOSE VALVES lineup on provided reference.

Which of the following describes the response of the Main Turbine valves when SHELL WARMING is selected on the <CONTROL><PRE-WARMING> screen?

A. MSVs, CVs, and IVs open, ISVs close.

B. MSVs close but MSV#2 can now be throttled, CVs open, IVs open, ISVs close.

C. MSVs remain closed but MSV#2 can be throttled, CVs and IVs remain closed, ISVs remain open.

D. MSVs remain closed but MSV#2 can now be throttled, CVs open, IVs remain closed, ISVs close.

Answer: D

Answer Explanation

Answer Explanation: In accordance with QCOP 5600-04, step F.2 describes the Main Turbine valve position for Shell Warming as:
All Control valves open to 100%, MSV #2 can now be throttled, CIV stop valves close, and all other valves remain closed.

Distractor 1: Plausible because all CVs open. Incorrect because MSVs and IVs are NOT open and MSV#2 can be throttled.

Distractor 2: Plausible because all CVs open, and MSV #2 can now be throttled are correct. Incorrect because IVs and ISVs are NOT closed.

Distractor 3: Plausible because this is a Main Turbine warming valve lineup. Incorrect because it is the valve lineup for CHEST WARMING.

Reference: QCOP 5600-04, Rev 27.
Reference provided during examination: DEHC screen with CLOSE VALVES lineup.

Cognitive level: High

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

K/A: 245000 K5.03 Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: Hydraulically operated valve operation

Importance: 2.9
10 CFR Part 55 Content: 41.5/45.3

Question Source: New
Question History: N/A
Comments: None

61 **ID: 2476319** **Points: 1.00**

- Unit 1 is operating at 50% power with a normal electric plant line-up.
- Transformer 12 trips but Auxiliary power DOES NOT transfer.

Which Reactor Feed Pumps are available?

A. 1A RFP ONLY

B. 1B RFP ONLY

C. 1A RFP and 1C RFP

D. 1B RFP and 1C RFP

Answer: C

Answer Explanation

Answer Explanation: In the normal at power electric plant lineup, Transformer 12 feeds Bus 12. If Aux Power does not transfer on a trip of Transformer 12, then Bus 12 is deenergized leaving the 1B RFP unavailable. 1A RFP remains running and the 1C RFP is available because of its dual feed from Bus 11.

Distractor 1: Plausible because 1A RFP is fed from Bus 11. Incorrect because 1C RFP can be fed from Bus 11 and is available.

Distractor 2: Plausible because Bus 12 can also be energized from Transformer 11. Incorrect because the auto transfer failed leaving Bus 12 deenergized.

Distractor 3: Plausible because the 1C RFP is available. Incorrect because the 1B RFP is unavailable due to the failure of Aux power to transfer.

Reference: QOA 6100-01 Rev.37, QOM 1-6500-T01 REV.06, QOM 1-6500-T02 Rev.06
Reference provided during examination: None

Cognitive level: High

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

K/A: 259001 K2.01 FEEDWATER SYSTEM Knowledge of electrical power supplies to the following: Reactor feedwater pump(s): motor-driven-only

Importance: 3.8
10 CFR Part 55 Content: 41.7

Question Source: New
Question History: N/A
Comments: None

62 **ID: 2487806** **Points: 1.00**

Unit 1 has experienced a large LOCA.

- Low Pressure ECCS system have NOT responded as expected, requiring the use of Alternate Injection Systems.
- QCOP 4100-11, USING DIESEL FIRE PUMPS VIA SAFE SHUTDOWN HOSE LINE FOR EMERGENCY USE has been directed.

How is the Safe Shutdown Hose Line installed?

Attach one end from the fire header strainer valve and the other end ...

A. at the flange connection upstream of MO 1-3205B, 1B RX FW INLT VLV.

B. to the quick disconnect fitting between the A Loop RHR Drywell Spray valves.

C. to the quick disconnect fitting upstream of MO 1-1402-24B, CS PMP OUTBD DISCH VLV.

D. to the quick disconnect fitting downstream of the 1-0301-25, U-1 CRD CHARGING WTR SV.

Answer: B

Answer Explanation

Answer Explanation: QCOP 4100-11, USING DIESEL FIRE PUMPS VIA SAFE SHUTDOWN HOSE LINE FOR EMERGENCY USE, step F.1.i. provides the direction for installation between the Drywell Spray valves on A RHR Loop as follows:

1. Attach one end of hose to quick connect fitting on fire header strainer valve 1(2)-4199-291, located on el 595' east wall.
2. Attach other end of hose to quick connect fitting at 1(2)-1099-166, located on el 595' north of X1 hatch.

Distractor 1: Plausible because the Condensate system injects into the RPV and can be supplied by river water via Standby Coolant. Incorrect because there is no direct connection to the Fire Header.

Distractor 2: Plausible because the Core Spray system injects into the RPV and can be cross connected with other water sources. Incorrect because there is no connection between Core Spray and the Fire Header.

Distractor 3: Plausible because the CRD system injects water into the RPV. Incorrect because the CRD and Fire systems can not be cross connected.

References: QCOP 4100-11 Rev, 21, M-39 Sh.1 Rev. BW
Reference provided during examination: None.

Cognitive level: Memory

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

K/A: 286000 K1.12 Knowledge of the physical connections and/or cause and effect relationships between the FIRE PROTECTION SYSTEM and the following systems: Emergency core cooling system

Importance: 3.1
10 CFR Part 55 Content: 41.2 to 41.9/45.7 to 45.8

Question Source: New
Question History: N/A
Comments: None

63 **ID: 2476428** **Points: 1.00**

Unit 1 was operating at full power when the Control Room Ventilation System isolated due to High Main Steam Line flow.

After the alarm condition clears, the reset of the isolation should be performed by the...

A. RO ONLY

B. EO ONLY

C. RO AND EO

D. RO AND SS

Answer: C

Answer Explanation
Answer Explanation: An isolation of the CR HVAC system due to High Main Steam Line flow will need to be reset at both the B Control Room HVAC local control panel 1/2-9400-105 panel and at the control room 901-5 panel. **Distractor 1:** Plausible because the signal must be reset at the 901-5 panel. Incorrect because it must also be reset at the 1/2-9400-105 panel. **Distractor 2:** Plausible because the signal must be reset at the 1/2-9400-105 panel. Incorrect because it must also be reset in the control room. **Distractor 3:** Plausible because the signal must be reset at the 901-5 panel. Incorrect because a Shift Supervisor is NOT required to reset the isolation at the local control panel. **Reference:** QOA 900-5 D-8 Rev.02 **Reference provided during examination:** None **Cognitive level:** Memory **Level (RO/SRO):** RO **Tier:** 2 **Group:** 2 **K/A:** 290003 G2.1.47 CONTROL ROOM VENTILATION: CONDUCT OF OPERATIONS: Ability to direct non-licensed personnel activities inside the control room **Importance:** 3.2 **10 CFR Part 55 Content:** 41.10/43.5/45.5/45.12/45.13 **Question Source:** Bank **Question History:** 2012 ILT NRC Exam **Comments:** None

64 **ID: 2476435** **Points: 1.00**

A startup is in progress on Unit 1.
Initial SRM count rate was 300 cps.

The NSO observes the following indications after each control rod movement:

- All four SRM's are indicating 6000-6500 cps.
- Initial period response is shorter and the time to return to infinity is longer.

Control rod withdrawal of ...

A. any rod from position 00 to 36 may be CONTINUOUS.

B. any rod from position 06 to 36 must be SINGLE NOTCH ONLY.

C. any rod from position 00 to 48 must be SINGLE NOTCH ONLY.

D. peripheral rods ONLY may be CONTINUOUS from position 00 to 48.

Answer: B

Answer Explanation

Answer Explanation: QCGP 4-1, CONTROL ROD MOVEMENTS AND CONTROL ROD SEQUENCE, step D.3. states: To minimize the risk of inadvertent short periods, NOTCH OVERRIDE shall NOT be used between positions 06 and 36 from the time the count rate on any SRM has doubled 3 times (8 times initial reading). Indications have SRMs between 4 and 5 doublings and approaching criticality, therefore the restriction on use of notch override applies.

Distractor 1: Plausible because continuous withdrawal is allowed before the SRM count rate has doubled 3 times. Incorrect because indications are that the reactor is approaching criticality and restrictions between positions 06 and 36 applies.

Distractor 2: Plausible because there is a single notch restriction. Incorrect because it is between positions 06 and 36.

Distractor 3: Plausible because peripheral rods do not insert much positive reactivity, ie. low worth. Incorrect because the limitation for use of notch override at this time does not have an exception for peripheral control rods.

References: QCGP 4-1 Rev. 51
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): RO
Tier: 3

K/A: G2.1.7 CONDUCT OF OPERATIONS: Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation

Importance: 4.4
10 CFR Part 55 Content: 41.5/43.5/45.12/45.13

Question Source: New
Question History: N/A
Comments: None

65 **ID: 2476434** **Points: 1.00**

Shift turnover is in progress on Unit 1.
Reactor power is 2957 MWth and slowly RISING, due to Xenon decay.
The off-going Unit Supervisor directs the off-going NSO to lower power using Reactor Recirculation pumps.

Which of the following describes the station's policy for completing the power change during shift turnover?

A. The shift turnover should proceed, the thermal power limit is NOT being exceeded.

B. The off-going NSO should put the turnover on hold, make the power change, then resume the turnover.

C. The power change should be delayed until the shift turnover is complete and on-coming personnel have assumed the watch.

D. The on-coming NSO should assume the At-The-Controls NSO duties and make the power change, then complete the shift turnover.

Answer: B

Answer Explanation

Answer Explanation: The station policy for reactivity changes is in OP-AA-300, REACTIVITY MANAGEMENT, as follows:
Step 4.8.9. "... Operators who are in the process of reactivity manipulations are focused on the task and not involved with concurrent activities that would cause a distraction."

Distractor 1: Plausible because the thermal power limit is not exceeded at the present time. Incorrect because it may result in a hastily performed power reduction as the reactor is approaching the thermal power limit. OP-AA-300 requires reactivity manipulations to be made in a
deliberate, carefully controlled manner.

Distractor 2: Plausible because it assures the power change will occur with no distractions. Incorrect because delaying the turnover may result in exceeding the Operating License Limit on thermal power output.

Distractor 3: Plausible as this outcome is not strictly prohibited. Incorrect because it does not represent conservative decision making.

References: OP-AA-300 Rev. 14
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): RO
Tier: 3

K/A: G2.1.39 CONDUCT OF OPERATIONS: Knowledge of conservative decision-making practices

Importance: 3.6
10 CFR Part 55 Content: 41.10/43.5/45.12

Question Source: Bank
Question History: 2019 NRC Initial License Exam
Comments: None

66 **ID: 2476436** **Points: 1.00**

During the course of a Complex Troubleshooting activity, who maintains the overall responsibility for Operability, Reportability, and Quarantines?

A. Technical Team Leader (TTL)

B. Operations Management (SRO)

C. Event Response Team Leader (ERT)

D. Troubleshooting Team Manager (TTM)

Answer: B

Answer Explanation

Answer Explanation: OP-AA-101-111, ROLES AND RESPONSIBILITIES OF ON-SHIFT PERSONNEL and OP-AA-106-101-1005, Quarantine of Areas Equipment and Records delineate the responsibility to the Operations Department for determining Operability, Reportability, or Quarantines. The Troubleshooting team members and responsibilities are outlined in MA-AA-716-004, CONDUCT OF TROUBLESHOOTING.

Distractor 1: Plausible because during Complex Troubleshooting a Technical Team Lead (TTL) is assigned in addition to the Troubleshooting Team Manager. The TTL assists in coordinating the Complex Troubleshooting, but does not maintain overall responsibility. In this case, the SRO makes all Operability determinations.

Distractor 2: Plausible because the ERT reviews all system maintenance and testing for the systems. The ERT assists in coordinating the Complex Troubleshooting but does not maintain overall responsibility. In this case, the SRO makes all Operability determinations.

Distractor 3: Plausible because the TTM is the highest level of Plant Management on the Complex Troubleshooting team; He/she will usually be a branch Manager or higher level from Engineering, The TTM manages the Complex Troubleshooting, but does not maintain overall responsibility for Operability determinations.

References: MA-AA-716-004 Rev. 19, OP-AA-101-111 Rev. 13, OP-AA-106-101-1005 Rev. 02
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 3

K/A: G2.2.20 EQUIPMENT CONTROL: Knowledge of the process for managing troubleshooting activities.

Importance: 2.6
10 CFR Part 55 Content: 41.10/43.5/45.13

Question Source: Bank
Question History: 2016 NRC Initial License Exam
Comments: None

Test Answer Key

2023 ILT NRC RO Exam Final Version
Test ID: 369039

67 **ID: 2476439** **Points: 1.00**

- Reactor Mode Switch in SHUTDOWN
- All control rods are inserted to 00 inches
- Reactor water temperature 190 °F
- 3 Reactor vessel head closure bolts are DETENSIONED

What operational mode is the plant in?

A. Mode 2

B. Mode 3

C. Mode 4

D. Mode 5

Answer: D

Answer Explanation

Answer Explanation: Per Technical Specifications Definition Table 1.1-1:
Mode 5 is defined as having the Mode Switch in Shutdown OR Refuel and one or more reactor vessel head closure bolts less than fully tensioned. The average reactor coolant temperature is not applicable.

Distractor 1: Plausible because it is a mode of operation in which the Mode Switch position may be in Refuel and the coolant temperature is not applicable. Incorrect because Mode 2 requires all reactor head closure bolts fully tensioned.

Distractor 2: Plausible because the Mode Switch position is Shutdown. Incorrect because coolant temperature for Mode 3 is > 212°F and all reactor head closure bolts must be fully tensioned.

Distractor 3: Plausible because the Mode Switch is in Shutdown and coolant temperature is < 212°F. Incorrect because Mode 4 requires all reactor head closure bolts fully tensioned.

References: Technical Specification Table 1.1-1 Amendment No. 279/274
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 3

K/A: G2.2.35 EQUIPMENT CONTROL: Ability to determine technical specification mode of operation

Importance: 3.6
10 CFR Part 55 Content: 41.7/41.10/43.2/45.13

Question Source: New
Question History: N/A
Comments: None

68 **ID: 2481782** **Points: 1.00**

A Loss of Cooling Accident has occurred on Unit 2:

- All control rods inserted to position 00.
- Several Fuel assemblies were damaged during the Reactor scram.
- Radioactive release is in progress.
- Drywell pressure is 4.5 psig and slowly rising.
- Torus pressure is 5.0 psig and slowly rising.
- The Unit Supervisor has ordered venting the Torus using the SBGT system.

Which of the following are valid reasons for this decision?

1. To prevent unnecessary cycling of the Reactor Building to Torus vacuum breakers.
2. To prevent unnecessary cycling of the Torus to Drywell vacuum breakers.
3. To provide a scrubbed release path.
4. To minimize the amount of radioactive discharge and allow for a monitored ground level release.

A. 1 and 2

B. 2 and 3

C. 1 and 4

D. 3 and 4

Answer: B

Answer Explanation

Answer Explanation: QCOP 1600-13, step B.4. POST-ACCIDENT VENTING OF THE PRIMARY CONTAINMENT, Rev. 33 states that the preferred method for venting the Primary Containment is through the Torus. Studies performed by General Electric indicate a decontamination factor of up to four is provided when vented gases are required to bubble through water first. This will also help in preventing the Torus to Drywell Vacuum Breakers from cycling.
UFSAR, Section 6.2.1.2.4.5.2.1 Rev. 2, Design Basis states that the normally selected vent path would be from the pressure suppression chamber only, to take advantage of the scrubbing effect of the suppression pool.
The cycling of the Reactor Building to Torus vacuum breakers is incorrect since as stated in the UFSAR Section 6.2.1.2.4.1, Rev 13. These vacuum breakers relieve pressure from the Reactor Building to the Torus and therefore would not be cycling at high Torus pressure or excessively no matter which vent path was chosen.
SBGT System discharges to the Main Chimney. This serves to minimize the amount of radioactive discharge and *prevents* a ground level release.

Distractor 1: Plausible because venting the Torus using SBGT system prevents unnecessary cycling of the Torus to Drywell vacuum breakers is correct. However, in this case Reactor Building to Torus vacuum breakers are not affected.

Distractor 2: Plausible because venting the Torus using SBGT system provides a scrubbed release path is correct. However, in this case Reactor Building to Torus vacuum breakers are not affected.

Distractor 3: Plausible because venting the Torus using SBGT system prevents unnecessary cycling of the Torus to Drywell vacuum breakers is correct. Also, SBGT system does minimize the amount of radioactive discharge, however SBGT system is designed to maintain the Reactor Building at a negative pressure of .25 inches of water while treating and exhausting 4000 scfm to the Main Chimney thus <u>preventing</u> a ground level release.

References: UFSAR Section 6.2.1.2.4.1 Rev. 13, UFSAR section 6.2.3.2 Rev. 6, UFSAR Section 6.2.1.2.4.5.2.1 Rev. 2, QCOP 1600-13 Rev. 33.
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): RO
Tier: 3

K/A: G2.3.11 RADIATION CONTROL: Ability to control radiation releases.

Importance: 3.8
10 CFR Part 55 Content: 41.11/43.4/45.10

Question Source: New
Question History: N/A
Comments: None

69 **ID: 2476441** **Points: 1.00**

Unit 2 was operating at rated power when a lightning strike in the 345KV Yard resulted in GCB 1-11 and GCB 10-11 opening.

Which of the annunciators received should have the HIGHEST priority in the report to the Unit Supervisor?

- 902-5 A-12, CHANNEL A/B STOP VLVS CLOSED
- 902-5 B-13, CHANNEL A/B REACTOR LOW LEVEL
- 902-5 D-8, CONTROL ROOM VENT ISOLATED
- 902-5 A-14, CHANNEL A/B DISCH VOLUME HIGH LEVEL

A. 902-5 D-8, CONTROL ROOM VENT ISOLATED

B. 902-5 A-12, CHANNEL A/B STOP VLVS CLOSED

C. 902-5 B-13, CHANNEL A/B REACTOR LOW LEVEL

D. 902-5 A-14, CHANNEL A/B DISCH VOLUME HIGH LEVEL

Answer: C

Answer Explanation

Answer Explanation: All alarms are associated with the reactor scram and expected for the transient described. However, the reactor low level alarm is also a QGA 100 entry, which will prescribe the highest priority actions to stabilize the plant.

Distractor 1: Plausible because it is associated with the reactor scram and expected for the transient. Incorrect because annunciator 902-5 D-8 is NOT a direct QGA entry.

Distractor 2: Plausible because it is associated with the reactor scram and expected for the transient. Incorrect because annunciator 902-5 A-12 is NOT a direct QGA entry.

Distractor 3: Plausible because it is associated with the reactor scram and expected for the transient. Incorrect because annunciator 902-5 A-14 is NOT a direct QGA entry.

References: QGA 100 Rev.13
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): RO
Tier: 3

K/A: G2.4.45 EMERGENCY PROCEDURES/PPLAN: Ability to prioritize and interpret the significance of each annunciator or alarm

Importance: 4.1
10 CFR Part 55 Content: 41.10/43.5/45.3/45.12

Question Source: New
Question History: N/A
Comments: None

Test Answer Key

2023 ILT NRC RO Exam Final Version
Test ID: 369039

70 **ID: 2478611** **Points: 1.00**

Unit 1 is in Mode 4 conducting a heat up. As reactor coolant temperature rises, its moderator characteristics are as follows:

The macroscopic absorption cross section is ___(1)___ and the average neutron energy loss per collision is ___(2)___.

A. (1) large
(2) small

B. (1) large
(2) large

C. (1) small
(2) small

D. (1) small
(2) large

Answer: D

Answer Explanation

Answer Explanation: The ideal moderator requires a small thermal neutron absorption cross section to minimize neutron losses. A large moderator absorption cross section would result in significant neutron losses. The moderator would act as a poison by removing neutrons necessary to carry out the chain reaction. A large neutron energy loss per collision with the moderator atoms is also preferred as neutron are less likely to leak from the core before reaching thermal equilibrium or be absorbed in non-fuel materials. This property is expressed as the Logarithmic Energy Decrement, (average energy loss per collision in a moderator).

Distractor 1: Plausible because a large absorption cross sections are desired in some non-fuel materials, ie, control rods and burnable poisons. Incorrect because this is not a desired property for the moderator.

Distractor 2: Plausible because a large energy loss per collision is desired. Incorrect because a large absorption cross section results in non-fuel thermal neutron absorption.

Distractor 3: Plausible because a small absorption cross section is desired. Incorrect because a small logarithmic energy decrement results in larger neutron losses to leakage outside the core and absorption in non-fuel materials.

References: BWR Generic Fundamentals, General Physics Corporation 2007
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 4

K/A: 292001 K1.05 NEUTRONS: Identify characteristics of good moderators

Importance: 2.6
10 CFR Part 55 Content: 41.1

Question Source: New
Question History: N/A
Comments: None

71 **ID: 2478614** **Points: 1.00**

Unit 2 was steady at 100% power when Control Rod 30-31 scrams into the core.

Reactor power undergoes a prompt drop that...

A. is a result of the rapid response of the fuel temperature coefficient.

B. occurs in response to rapid decrease in the prompt neutron population.

C. occurs at a period of -80 seconds regardless of the size of the reactivity insertion.

D. is caused by the magnitude of reactivity inserted exceeding the value of the average effective delayed neutron fraction.

Answer: B

Answer Explanation

Answer Explanation: When negative reactivity is inserted, the power drops quickly prior to reaching a stable reactor period. It is an initial rapid decrease in the prompt neutron population. A trace of the power level shows an almost instantaneous drop followed by an asymptotic approach to a value as the period stabilizes due to the decay of the shorter lived delayed neutron precursors. The prompt neutron population is affected before the delayed neutron population.

Distractor 1: Plausible because the fuel temperature coefficient inserts negative reactivity. Incorrect because the fuel temperature coefficient becomes prominent when a large positive reactivity insertion occurs.

Distractor 2: Plausible because the -80 second period occurs after a large negative reactivity insertion, ie. scram. Incorrect because a prompt drop is the result of a step insertion of negative reactivity such as notch insertion of individual control rods.

Distractor 3: Plausible because a negative reactivity insertion exceeding the value of the effective delayed neutron would drop reactor power. Incorrect because a reactivity insertion of this magnitude would result in a subcritical reactor and not a prompt drop followed a by a stable reactor period.

References: BWR Generic Fundamentals, General Physics Corporation 2007
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 4

K/A: 292003 K1.07 REACTOR KINETICS AND NEUTRON SOURCES: Explain prompt critical, prompt jump, and prompt drop

Importance: 3.3
10 CFR Part 55 Content: 41.1

Question Source: New
Question History: N/A
Comments: None

72 **ID: 2478617** **Points: 1.00**

A startup to 100% power is in progress on Unit 1. The plant is currently at 45% power and withdrawing rods.

The POSITIVE reactivity from rod withdrawal is dampened by the NEGATIVE reactivity added due to an increased void fraction, which ____.

A. increases the number of neutrons absorbed in the control rods.

B. increases the number of neutrons that achieve thermal energy.

C. decreases neutron moderation and increases neutron absorption in Xenon.

D. increases the number of neutrons that leak from the core or are resonantly absorbed.

Answer: D

Answer Explanation

Answer Explanation: Increasing the core void fraction decreases the moderator density. A decreasing moderator density results in greater neutron leakage from the core, less absorption by moderator molecules, and an increased resonance absorption in U-238.

Distractor 1: Plausible because the neutron mean free path increases with a decrease in moderator density. This increases the probability of neutrons reaching the control rod. Incorrect because as the moderator density lowers, the slowing down or thermalization time also increases. Control rods are thermal neutron absorbers, therefore less absorption occurs as the neutron energy remains in the epithermal range longer.

Distractor 2: Plausible because the neutron population will increase in an energy range. Incorrect because the neutron population lowers in the thermal range. A lower moderator density will result in neutrons remaining in the epithermal region longer.

Distractor 3: Plausible because neutron moderation decreases. Incorrect because Xenon is a thermal neutron absorber. Xenon absorption lowers with decreasing neutron moderation.

References: BWR Generic Fundamentals, General Physics Corporation 2007
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 4

K/A: 292004 K1.10 REACTIVITY COEFFICIENTS: Define the void coefficient of reactivity

Importance: 3.2
10 CFR Part 55 Content: 41.1

Question Source: New
Question History: N/A
Comments: None

73 **ID: 2478619** **Points: 1.00**

Unit 2 licensed limit for reactor power is measured by percent _____ power.

A. thermal neutron flux

B. rated thermal output

C. rated electrical output

D. rated Recirc Drive flow

Answer: B

Answer Explanation

Answer Explanation: Per QCOP 0700-06, APRM FLOW BIASED HIGH FLUX (HEAT BALANCE) CALIBRATION TEST, the percent rated reactor power is the ratio of core thermal power (CTP_{MWt}) to rated thermal power (2957 MWt) multiplied by 100.

% Rated Power = (CTP_{MWt}) / 2957 MWt X 100

Distractor 1: Plausible because thermal flux is correlated to thermal power. Incorrect because flux is a measure of the neutron population which is subject to reactions other than nuclear fission. It is therefore not a direct measure of the thermal energy output of the core.

Distractor 2: Plausible because turbine electrical output is related to core thermal power. Incorrect because the electrical output varies with main turbine design, main condenser and feedwater heater efficiencies.

Distractor 3: Plausible because Recirc Drive flow is correlated to thermal power. Incorrect because the percent rated Drive flow (WD/WT X100) where WD is the Recirc pump flow and WD is the Recirc pump flow at rated power. It is only reliable at high flow conditions, ie. > 70 Mlb/hr. The WD/WT relationship changes at low Core Flow conditions. The Recirc pump output to obtain rated flow changes during the operating cycle due to changes in the core resistance to flow.

References: QCOP 0700-06 Rev.32
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 4

K/A: 293007 K1.12 HEAT TRANSFER: CORE THERMAL POWER: Define percent reactor power

Importance: 2.7
10 CFR Part 55 Content: 41.14

Question Source: New
Question History: N/A
Comments: None

74 **ID: 2478620** **Points: 1.00**

The fuel installed in Unit 1 and Unit 2 is designed with a radial gap between the UO_2 fuel pellet and Zircalloy cladding to _____.

A. enhance heat transfer from the fuel pellet to the cladding.

B. accommodate fission gasses built up by the decay of fission products.

C. accommodate differential axial expansion of the fuel pellet and the cladding.

D. accommodate differential radial expansion of the fuel pellet and the cladding.

Answer: D

Answer Explanation

Answer Explanation: A radial gap is provided between the fuel pellet and cladding to allow for pellet growth due to thermal expansion and irradiation swelling. This will prevent excessive cladding strain that can be caused by a pellet-clad interaction.

Distractor 1: Plausible because the heat transfer across the pellet/clad gap changes over the operating cycle. Incorrect because the gap is not designed to enhance the heat transfer between fuel and cladding. It is designed to handle thermal expansion and irradiated swelling of the pellet and prevent PCI failure.

Distractor 2: Plausible because fission gasses are formed during plan operation. Incorrect because the space at the top of the fuel rod called the plenum provides for the accumulation of fission gases.

Distractor 3: Plausible because the fuel pellet undergoes axial and radial changes over the operating cycle. Incorrect because the plenum spring is designed to maintain a constant force on the pellet stack ensuring no gaps occur between the fuel pellets.

References: BWR Generic Fundamentals, General Physics Corporation 2007
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 4

K/A: 293009 K1.33 CORE THERMAL LIMITS: PELLET-CLAD INTERACTION: Describe the purpose of the pellet to clad gap

Importance: 2.8
10 CFR Part 55 Content: 41.14

Question Source: New
Question History: N/A
Comments: None

75 **ID: 2478624** **Points: 1.00**

QGA 100, RPV CONTROL requires a cooldown at less than 100°F/hr.
The operational concern for exceeding the limit is a large .....

A. tensile stress at the inner wall of the reactor vessel.

B. tensile stress at the outer wall of the reactor vessel.

C. compressive stress at the inner wall of the reactor vessel.

D. compressive stress at the outer wall of the reactor vessel.

Answer: A

Answer Explanation

Answer Explanation: During cooldown, the thermal stresses are additive to the pressure, embrittlement, and residual stresses on the inner wall of the vessel. The inner wall is closer to the total allowable stress as compared to the outer wall where the stresses tend to cancel each other. All mentioned above are tensile at 1/4 thickness of the wall during a cooldown.

Distractor 1: Plausible because a large tensile stress exists. Incorrect because it is larger at the inner wall than the outer wall.

Distractor 2: Plausible because a compressive stress is created by the temperature gradient. Incorrect because the total stress is tensile and largest at the inner wall.

Distractor 3: Plausible because a compressive stress is created at the outer wall by the temperature gradient. Incorrect because the total stress is tensile and largest at the inner wall.

References: BWR Generic Fundamentals, General Physics Corporation 2007, (Chapter 10, Figure 10-10 Cooldown Stress Profile).
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): RO
Tier: 4

K/A: 293010 K1.04 BRITTLE FRACTURE AND VESSEL THERMAL STRESS: State how the possibility of brittle fracture is minimized by operational limitations

Importance: 3.2
10 CFR Part 55 Content: 41.14

Question Source: New
Question History: N/A
Comments: None

Test Answer Key

2023 ILT NRC SRO Exam Final Version
Test ID: 369040

76 **ID: 2478692** **Points: 1.00**

Unit 1 is in at 100% power.
Unit 2 is in Mode 3 preparing for a startup.

- QCGP 1-2, NORMAL UNIT 2 STARTUP, prerequisites are in progress.
- An EO reports the "EMERGENCY SOURCE ACCEPTED" light on the Unit 2 ESS ABT is NOT lit.
- All other portions of the electrical distribution are in a normal lineup.

What actions, if any, are required for Unit 1 and Unit 2?

(Reference provided**)**

A. Unit 1: None
Unit 2: Continue with the startup

B. Unit 1: None
Unit 2: Remain in Mode 3 until ESS is operable

C. Unit 1: 7 days to restore ESS
Unit 2: Immediately enter TS 3.0.3

D. Unit 1: 7 days to restore ESS
Unit 2: Remain in Mode 3 until ESS is operable

Answer: D

Answer Explanation

Answer Explanation: Per the Technical Specification 3.8.7 Bases, Distribution Systems -- Operating, " the 120 VAC ESS Bus must be capable of being energized from Bus 18-2 (28-2)". The Modes of Applicability are 1,2, and 3 placing the Unit in TS 3.8.7 Condition A with 8 hours to restore ESS to operable status. If not met, entry into Condition D is required and the Unit must be in Mode 3 in 12 hours which is already met. Therefore, startup cannot continue, ie. Mode Switch to STARTUP, until the ABT is repaired. There are no Division II requirements per Table B 3.8.7-1. Unit 1 is in a 7 day LCO per TS 3.8.7 Condition C as the opposite unit AC subsystem required for SBGTS operability is not met.

Distractor 1: Plausible because all Busses and MCCs are energized and fed from their normal sources on Unit 1. Incorrect because the emergency supply to Unit 2 ESS (MCC-28-2) is required for operability in Mode 3, therefore startup cannot continue.

Distractor 2: Plausible because Unit 2 must remain in Mode 3 until ESS is operable. Incorrect because Unit 1 requires the opposite unit AC electrical sources that support SBGTS, CREV, and CREV (AC).

Distractor 3: Plausible because TS 3.8.7 b. requires the opposite unit AC electrical subsystems for SBGTS. Incorrect because Unit 2 is already met the requirement to be in Mode 3 and is not required to be in Mode 4.

Reference: TS 3.8.7 and TS 3.8.7 Bases
Reference provided during examination: TS 3.8.7

Cognitive level: High

L**evel (RO/SRO):** SRO
Tier: 1 **Group:** 1

KA: 295003 G2.2.1 PARTIAL OR COMPLETE LOSS OF AC POWER: EQUIPMENT CONTROL- Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Importance: SRO= 4.4
10 CFR Part 55 Content: 41.5/41.10/43.5/43.6/45.1

SRO Justification:
Can the question be answered solely by knowing ≤ 1-hour TS/TRM action? **No**
Can the question be answered solely by knowing the LCO/TRM information listed “above the line”? **No**
Can the question be answered solely by knowing the TS safety limits? **No**
Can the question be answered solely by knowing the TS bases information associated with the “above-the-line” LCO information or general systems knowledge? **No**
Does the question involve one or more of the following for TS, TRM, or ODCM: **Yes**
• application of required actions (TS Section 3) and SRs (TS Section 4) in accordance with rules of application requirements (TS Section 1)
• application of generic LCO requirements (LCOs 3.0.1 through 3.0.7 and LCOs 4.0.1 through 4.0.4)
• knowledge of TS bases that is required to analyze TS required actions and terminology

Content: 10 CFR 55.43(b)(2)

Question Source: New
Question History: None
Comments: None

Test Answer Key

2023 ILT NRC SRO Exam Final Version
Test ID: 369040

77 **ID: 2478714** **Points: 1.00**

Unit 1 is operating at 100% power, when a total loss of 120 VAC ESS Bus occurs.
The Unit Supervisor orders a manual scram.
Nuclear Instruments fully inserted as expected and all APRM Downscale lights are lit.

At this time, the Unit Supervisor should be directing actions of…

A. QGA 100, RPV CONTROL, ONLY.

B. QCGP 2-3, REACTOR SCRAM, ONLY.

C. QCGP 2-3, REACTOR SCRAM and QGA 101, RPV CONTROL (ATWS).

D. QCGP 2-3, REACTOR SCRAM and QGA 100, RPV CONTROL, ONLY.

Answer: C

Answer Explanation

Answer Explanation: At rated power, insertion of a manual scram will result in RPV water level lowering to < 0 inches. This requires entry into QGA 100. However, with the total loss of Essential Service bus, control rod position indication is lost on the Full-Core Display as well as the RWM. ALL Rods IN cannot be directly determined which results in a transition to QGA 101, (Technical Support Guidelines section 3.2.1 basis). Entry into QCGP 2-3 is concurrent with the direction to insert a manual scram.

Distractor 1: Plausible because entry into QGA 100 was initially required. Incorrect because transition to QGA 101 should have been made and QCGP 2-3 actions should also be in progress.

Distractor 2: Plausible because entry of QCGP 2-3 is directed with the insertion of a manual scram. Incorrect because QGA 101 entry is also required.

Distractor 3: Plausible because QGA 100 is initially entered on low RPV water level and QCGP 2-3 is entered. Incorrect because at this time transition to QGA 101 should have been made.

Reference: TSG-3.2.1, QGA 100
References provided during examination: None

Cognitive level: High

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

K/A: 295006 AA2.02 Ability to determine or interpret the following as they apply to SCRAM: Control rod position.

Importance: SRO 4.5

10 CFR 55 Content: 41.10/43.5/45.13

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: New
Question History: None
Comments: None

Justification for how references supplied during 2023 ILT exam do not give away answers: Requires knowledge of power supplies to RPI/RWM, in order to determine 101 actions are necessary. This power supply is not listed in the references anywhere. No information in QGA 101 helps at all with this question.

78 **ID: 2478723** **Points: 1.00**

QOA 0010-05, PLANT OPERATION WITH THE CONTROL ROOM INACCESSIBLE, has been entered. The Unit 1 Reactor was scrammed and RPV water level stabilized before personnel evacuated the control room. Personnel are at their assigned locations and communications established.

As the Unit Supervisor:

(1) What actions are assigned for RPV pressure control?
(2) What actions are assigned for RPV level control?

A. (1) Establish HPCI in pressure control mode per QCOP 2300-06, HPCI SYSTEM MANUAL STARTUP
(INJECTION / PRESSURE CONTROL).
(2) Establish RCIC in level control mode per QCOP 1300-09, RCIC LOCAL MANUAL OPERATION.

B. (1) Establish HPCI in pressure control mode per QCOP 2300-06, HPCI SYSTEM MANUAL STARTUP
(INJECTION / PRESSURE CONTROL).
(2) Establish RHR in level control mode per QCOP 1000-30, POST-ACCIDENT RHR OPERATION.

C. (1) Manually operate Relief Valves and start Torus Cooling per QCOP 1000-07, TORUS COOLING WITH THE CONTROL ROOM INACCESSIBLE.
(2) Establish RHR in level control mode per QCOP 1000-30, POST-ACCIDENT RHR OPERATION.

D. (1) Manually operate Relief Valves and start Torus Cooling per QCOP 1000-07, TORUS COOLING WITH THE CONTROL ROOM INACCESSIBLE.
(2) Control RPV water level by locally operating FWRVs per QCOP 0600-18, MAIN FEEDWATER REGULATOR OPERATION.

Answer: D

Answer Explanation

Answer Explanation: QOA 0010-05, PLANT OPERATION WITH CONTROL ROOM INACCESSIBLE, directs an RPV cooldown established using Relief Valves and local operation of the FWRVs to control RPV water level.

Distractor 1: Plausible because HPCI can be used for pressure and RCIC for level control. Incorrect because QOA 0010-05 uses relief valves for an RPV cooldown and the Feedwater system for RPV water level control.

Distractor 1: Plausible because HPCI can be used for pressure control and RHR can be operated in injection mode. Incorrect because QOA 0010-05 specifically directs the use of relief valves for RPV pressure control and the Feedwater system for level.

Distractor 3: Plausible because Relief valves are directed by QOA 0010-05 for RPV pressure control. Incorrect because the Feedwater system is used for RPV water level control.

Reference: QOA 0010-05 Rev 26, QCOP 1000-07 Rev 19, QCAP 0200-10 Rev. 60
Reference provided during examination: None

Cognitive level: High

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

K/A: 295016 AA2.07 Ability to determine or interpret the following as they apply to CONTROL ROOM ABANDONMENT: Suppression chamber pressure

Importance: SRO 3.4
10 CFR Part 55 Content: 41.10/43.5/45.13

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: Bank
Question History: 2009 ILT NRC Exam
Comments: None

79 **ID: 2478769** **Points: 1.00**

Unit One is in Mode 4.

- Shutdown Cooling is lined up on the "A" loop of RHR.
- "B" RHR Loop is OOS due to planned maintenance.
- When the NSO attempts to start up Shutdown Cooling, the MO 1-1001-29A valve breaker trips open and will NOT reset.
- RPV water temperature is 190°F and RISING at 10°F/hr.

What direction should be given to the crew?

A. Secure both Recirc Pumps per QCOP 0202-33, UNIT 1 REACTOR RECIRCULATION SYSTEM SHUTDOWN and maintain RPV water level at 30 inches.

B. Raise the RBCCW TCV setpoint per QCOP 1200-13, RWCU SYSTEM HIGH TEMPERATURE ALARM AND ISOLATION SETPOINT ADJUSTMENT.

C. Increase RWCU reject flow per QCOP 1200-07, RWCU SYSTEM COOLANT REJECTION and replace inventory with the CRD and/or Condensate Systems.

D. Open Reactor Head Vent, AO 1-220-46 and AO 1-220-47, OUTBD/INBD VENT VLVs per QCGP 2-1, NORMAL UNIT SHUTDOWN, to prevent pressurization of the vessel.

Answer: C

Answer Explanation

Answer Explanation: Per QCOA 1000-02, LOSS OF SHUTDOWN COOLING, step D.8.b., states if a vacuum is not present in the main condenser, increase reject flow and replace water with the CRD or Condensate systems

Distractor 1: Plausible because tripping recirc pumps removes the pump heat addition into the core. Incorrect because if both recirc pumps are secured, then RPV water level must be raised to a band of 90 to 100 inches to establish natural circulation.

Distractor 2: Plausible because RBCCW TCV setpoint can be adjusted to provide more heat removal. Incorrect because the setpoint should be lowered.

Distractor 3: Plausible because opening the Head Vent valves would prevent the RPV from repressurizing. Incorrect because the Head Vent Valves are not to be opened until RPV water temperature is < 190 deg. F.

Reference: QCOA 1000-02 Rev 23
Reference provided during examination: None

Cognitive level: High

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

K/A: 295021 G2.1.46 LOSS OF SHUTDOWN COOLING: CONDUCT OF OPERATIONS- Ability to use integrated control systems to operate plant systems or components.

Importance: SRO 3.3
10 CFR Part 55 Content: 41.10/45.12/45.13

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: New
Question History: N/A
Comments: None

80 **ID: 2478785** **Points: 1.00**

A leak in the B Main Steam Line resulted in a reactor scram from 100% power.
QGA 100, RPV CONTROL and QGA 200, PRIMARY CONTAINMENT CONTROL, have been entered.

- RPV pressure 500 psig
- Drywell temperature 250°F
- Drywell pressure 6 psig
- Torus pressure 4 psig
- Torus water temperature 110°F

Initiating Drywell Sprays under these conditions ...

(Reference Provided)

A. is required to prevent reference leg flashing.

B. could result in a higher Drywell O_2 concentration.

C. is required because Drywell temperature is at the design limit.

D. could result in damage from drawing a negative pressure in excess of primary containment design limits.

Answer: B

Answer Explanation

Answer Explanation: Initiating Drywell Sprays to the left of the DSIL curve could result in opening the Drywell-Torus Vacuum Breakers and the Reactor Building-Torus Vacuum Breakers. Since the containment pressure is below atmospheric, air would be drawn into the containment and increasing the O2 concentration.

Distractor 1: Plausible because 250 deg.F exceeds the boiling point of water at atmospheric pressure. Incorrect because with 500 psig reactor pressure and 250 deg.F drywell temperature, saturation conditions are not present in the Drywell and reference leg flashing would not be expected.

Distractor 2: Plausible because QGA 200 direction is to spray the Drywell before the temperature design limit is reached. Incorrect because the Drywell temperature design limit is 281 deg. F.

Distractor 3: Plausible because spraying the drywell will cause drywell pressure to drop. Incorrect because lowering Drywell pressure would open the torus to drywell Vacuum Breakers in order to equalize pressure and once DW pressure is 0.5 psig higher than torus pressure the vacuum breakers would shut preventing a negative pressure from being drawn.

Reference: Quad Cities Nuclear Power Station Technical Support Guidelines (TSG) Reference Manual, Rev.07, Sections 3.3.8 and 5.5
Reference provided during examination: QGA 200

Cognitive level: High

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

K/A: 295028 EA2.01 Ability to determine or interpret the following as they apply to HIGH DRYWELL TEMPERATURE (MARK I AND MARK II ONLY): Drywell temperature.

Importance: SRO 3.3
10 CFR Part 55 Content: 41.10/43.5/45.13

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10CFR 55.43(b)(5)

Question Source: New
Question History: N/A
Comments: None

81 **ID: 2478816** **Points: 1.00**

An ATWS has occurred on Unit 2.
A small ECCS Suction Header break on the Torus has developed.
The Unit Supervisor is directing actions of QGA 101, RPV CONTROL (ATWS) and QGA 200, PRIMARY CONTAINMENT CONTROL.
Initial ATWS actions have been taken, however SBLC failed to inject.
Mechanical Maintenance has reported that the leak can be patched.

- Reactor power is 20% and OSCILLATING
- Reactor pressure is 900 psig and SLOWLY RISING
- Reactor water level is -130 inches and LOWERING
- Drywell pressure is 4 psig and STABLE
- Torus water temperature is 180°F and RISING
- Torus water level is 13.5 feet and SLOWLY LOWERING

What action is required NEXT?

(Reference Provided)

A. Direct an RPV BLOWDOWN by opening all five ADS valves.

B. Anticipate RPV Blowdown and OPEN all Main Turbine Bypass Valves.

C. Continue to let RPV water level DROP to -142 inches, and LOWER reactor pressure.

D. Maintain RPV water level between -35 and -162 inches and MAXIMIZE Torus Cooling by aligning B RHR Loop in Torus Cooling mode

Answer: C

Answer Explanation

Answer Explanation: Reactor pressure and torus temperature are both rising and approaching the Heat Capacity Temperature Limit. The required actions are to continue to let reactor water level lower to suppress power (let drop to -142) and to lower reactor pressure to stay within the Heat Capacity Temperature Limit.

Distractor 1: Plausible because this is required if Torus water level lowers to 11 ft. Incorrect because actions to add water to the Torus should be attempted and attempts to isolate the leak are still in progress.

Distractor 2: Plausible because anticipating blowdown removes energy from the containment and is a correct action in QGA 100. Incorrect because QGA 100 is exited and the override to anticipate blowdown is not applicable in an ATWS due to power concerns.

Distractor 3: Plausible because maximizing Torus cooling is a required step in QGA 200 Torus Temperature leg. Incorrect because RPV water level should continue to lower until reactor power is < 5% OR RPV water level drops to -142 inches OR all ADS valves stay closed and Drywell pressure stays below 2.5 psig.

Reference: QGA 200 Rev.13, Quad Cities Nuclear Power Station Technical Support Guidelines (TSG) Reference Manual, Rev.7
Reference provided during examination: QGA 101, QGA 200

Cognitive level: High

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

K/A: 295030 EA2.02 Ability to determine or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: Suppression pool temperature

Importance: SRO 3.8
10 CFR Part 55 Content: 41.10/43.5/45.13

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: New
Question History: N/A
Comments: None

82 **ID: 2478850** **Points: 1.00**

A Loss of Offsite Power and a LOCA occurred on Unit 1.
The Unit 1 Emergency Diesel failed to start.
Bus 14-1 to Bus 24-1 crosstie tripped.
The Unit 1 SBO is OOS.
The HPCI turbine auto started and is injecting at 5000 gpm.
All other ECCS systems responded as designed.

- RPV water level is 0 inches and rising slowly
- Drywell pressure is 2.6 psig and rising slowly
- Torus pressure is 1.6 psig and rising slowly
- Drywell temperature is 170°F and stable

Which of the following actions has the HIGHEST PRIORITY?

A. Initiate Drywell sprays per QCOP 1000-30, POST-ACCIDENT RHR OPERATION.

B. Start the SSMP per QCOP 2900-02, SAFE SHUTDOWN MAKEUP PUMP SYSTEM START UP.

C. Dispatch an EO to start the U-1 EDG per QCOP 6600-11, DIESEL GENERATOR LOCAL OPERATION.

D. Dispatch an EO to restore RPS B per QCOP 7000-03, UNIT 1 REACTOR PROTECTION SYSTEM MG SETS.

Answer: C

Answer Explanation

Answer Explanation: QGA 100 requires verification of auto actions, if they do not occur the manual actuation should be performed as these are needed systems for the DBA LOCA.

Distractor 1: Plausible because Drywell temperature and pressure are elevated and rising slowly. Incorrect because no entry condition for QGA 200 has been reached and Torus pressure is below 5 psig.

Distractor 2: Plausible because another injection source would allow RPV water level to be restored more rapidly. Incorrect because HPCI is sufficient at the moment and establishing another injection source is not the highest priority.

Distractor 3: Plausible because RPS B was de-energized on the LOOP and is needed to validate QGA isolations. Incorrect because the higher priority is to restore Division II power after which both RPS busses can be energized. (required for full validation of containment isolations)

Reference: QGA 100 Rev.13, QGA 200 Rev.13
Reference provided during examination: None

Cognitive level: High

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

K/A: 295031 G2.1.8 REACTOR LOW WATER LEVEL: CONDUCT OF OPERATIONS: Ability to coordinate personnel activities outside the control room.

Importance: SRO 4.1
10 CFR Part 55 Content: 41.10/43.1/45.5/45.12/45.13

SRO Justification: Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: New
Question History: N/A
Comments: None

Justification for how references supplied during 2023 ILT exam do not give away answers: This question is asking the examinee to prioritize actions based on plant conditions. The examinee must determine that the EDG has failed to start from the plant conditions given. The examinee must determine that torus pressure has not reached **5 psig** and that Torus temperature is not yet **280 °F** (both redacted in references) so sprays are not the correct answer. The examinee must know that SSMP is not required due to the plant conditions stated (RWL going up) that are not in reference. Examinee must know that restoration of RPS B is not a priority over the other applicable action to take (EDG start).

83 **ID: 2478860** **Points: 1.00**

Unit 2 has been operating at 100% power with a failing Outboard Seal on the 2A Reactor Recirc pump. Inboard and Outboard Seals have been indicating 1000 psig and 250 psig respectively.

- Inboard Seal pressures is 900 psig
- Outboard Seal pressure is 300 psig
- Drywell Pressure is 1.40 psig
- Drywell Average Temperature is 153°F

The NSO will FIRST be directed to _(1)_.

The Unit Supervisor is required to enter _(2)_.

A. (1) trip the 2A Recirc pump, and close the MO 2-0202-4A, PMP SUCT VLV and MO 2-0202-5A, PMP DISCH VLV.

(2) TS LCO 3.6.1.5 Drywell Air Temperature ONLY.

B. (1) start ALL available Drywell Coolers.

(2) TS LCO 3.6.1.5 Drywell Air Temperature ONLY.

C. (1) trip the 2A Recirc pump, and close the MO 2-0202-4A, PMP SUCT VLV and MO 2-0202-5A, PMP DISCH VLV.

(2) TS LCO 3.6.1.4 Drywell Pressure and TS LCO 3.6.1.5 Drywell Air Temperature.

D. (1) start ALL available Drywell Coolers.

(2) TS LCO 3.6.1.4 Drywell Pressure and TS LCO 3.6.1.5 Drywell Air Temperature.

Answer: A

Answer Explanation

Answer Explanation: Per QCOA 0202-06, RECIRCULATION PUMP SEAL FAILURE, if an increase in Drywell pressure OR temperature is observed, trip the Recirc Pump and close MO 1(2)-202-4A/B, PMP SUCT VLV and MO 1(2)-202-5A/B, PMP DISCH VLV. TS 3.6.1.4 has an LCO entry condition of Drywell pressure > 1.5 psig while the entry condition for TS 3.6.1.5 is Drywell temperature > 150 deg.F. Therefore only entry into TS 3.6.1.5 is required. Additionally, QCOA 0201-01 states all following steps may be performed in any order EXCEPT for investigating the cause and attempting to isolate the leak. That must be performed first.

Distractor 1: Plausible because the crew would have also entered QCOA 0201-01, INCREASING DRYWELL PRESSURE which directs starting another Drywell Cooler and entry into TS 3.6.1.5 Drywell Air Temperature AFTER attempting to isolate the leak. Incorrect because tripping and isolating the Recirc pump takes priority procedurally over starting another Drywell Cooler.

Distractor 2: Plausible because the crew would have also entered QCOA 0201-01, INCREASING DRYWELL PRESSURE which directs starting another Drywell Cooler AFTER attempting to isolate the leak. Incorrect because entry into TS 3.6.1.4 Drywell Pressure is not required until >1.5 psig in the drywell.

Distractor 3: Plausible because Drywell pressure and temperature started increasing and QCOA 0202-06, RECIRCULATION PUMP SEAL FAILURE requires the Recirc pump be taken off line and isolated to stop the leak. Incorrect because entry into TS LCO 3.6.1.4 Drywell Pressure is not required until >1.5 psig in the drywell.

Reference: QCOA 0202-06 Rev.27, TS 3.6.1.4 Amend 248/243 and TS 3.6.1.5 Amend 248/243, QCOA 0201-01 Rev. 30
Reference provided during examination: None

Cognitive level: High

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

K/A: 295010 AA2.06 Ability to determine or interpret the following as they apply to HIGH DRYWELL PRESSURE: Drywell temperature

Importance: SRO 3.7
10 CFR Part 55 Content: 41.10/43.5/45.13

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: New
Question History: N/A
Comments: None

84 **ID: 2478903** **Points: 1.00**

QCOS 1300-01, RCIC PUMP OPERABILITY TEST was in progress on Unit 1 when the following annunciators alarm:

- 901-4 C-14 RCIC TURBINE EXHAUST DISCH HIGH PRESSURE
- 901-3 G-3, RX BLDG VENT CHANNEL A HI HI RADIATION
- 901-3 H-3, RX BLDG VENT CHANNEL B HI HI RADIATION

- 1-1705-8A, RX BLD VENT RAD MON is reading 20 mr/hr.
- 1-1805-17, RCIC CUBICLE radiation monitor is reading 2500 mr/hr and RISING.
- MO 1-1301-16, STM SPLY ISOL VLV and MO 1-1301-17, STM SPLY ISOL VLV will NOT close.

The EO has reported steam and water filling the RCIC room.
All other automatic actions have occurred.

As the Unit Supervisor which of the following actions has the HIGHEST priority?

(Reference Provided)

A. Insert a manual reactor scram and enter QGA 100, RPV CONTROL.

B. Verify open the 1-1301-64, U-1 RCIC TURB EXH STOP CK VLV.

C. Shutdown the reactor per QCGP 2-1, NORMAL UNIT SHUTDOWN.

D. Perform QOA 5750-07, REACTOR BUILDING VENTILATION ISOLATION.

Answer: A

Answer Explanation

Answer Explanation: In accordance with QGA 300, Temperature/Radiation Leg, if a primary system is discharging into the Reactor Building and cannot be isolated, then before any temperature, radiation, or water level reaches the max safe value per Table R, T, W, scram the reactor and enter QGA 100. This takes priority over other QOA, QCOP or QCAN required actions.

Distractor 1: Plausible because it si a require action in QCAN 901(2)-4 C-14, HIGH RCIC TURBINE EXHAUST DISCHARGE PRESSURE. Incorrect because the valve is located high up on the wall between the RCIC room and the Torus requiring a scaffold to reach and the QGA required action has a higher priority.

Distractor 2: Plausible because it is a QGA 300 action. Incorrect because it is performed if the discharge can be isolated and 2 or more areas are approaching a max safe value.

Distractor 3: Plausible because Reactor Building Ventilation has isolated. Incorrect because the QGA 300 actions have a higher priority,

Reference: QGA 300 Rev.14
Reference provided during examination: QGA 300

Cognitive level: High

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

K/A: 295034 G2.4.16 SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: EMERGENCY PROCEDURE/PLAN: Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, or severe accident management guidelines.

Importance: SRO 4.4
10 CFR Part 55 Content: 41.10/43.5/45.13

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: New
Question History: N/A
Comments: None

85 **ID: 2478958** **Points: 1.00**

Unit 1 and Unit 2 are operating at rated power when Security reports a tornado sighted in the Protected Area.

- Annunciators 912-5 C-1, RX BLD 1 LOW DP and 912-5 C-4, RX BLD 2 LOW DP are in alarm.
- The 1-5740, RX BLDG TO ATMOS DP gauge is indicating +0.5 psid.
- Several damaged blowout panels on the Refuel Floor has just been reported by an EO.

The Unit Supervisors have entered and are coordinating actions per QCOA 0010-10, TORNADO WATCH / WARNING, SEVERE THUNDERSTORM WARNING, OR SEVERE WINDS.

Who has the responsibility for making a 10CFR50.72 notification to the NRC, if required?

(Reference provided**)**

A. Unit Supervisor

B. Shift Manager

C. Station Emergency Director

D. NRC notification is NOT required

Answer: B

Answer Explanation

Answer Explanation: The report of a Tornado sighting, damaged blowout panels, and loss of Reactor Building DP is indicative of a strike on the Reactor Building. A Tornado strike within the Protected Area is an Unusual Event per EAL HU6. Per EP-AA-112-100-F-01, Shift Emergency Director Checklist, NRC notification is required after notifying State and Local agencies but not later than (1) hour after the time of classification.
The Shift Manager assumes the role of Shift Emergency Director when the E-Plan is activated.

Distractor 1: Plausible because QCOA 0010-10 requires notification of organizations on site which falls under the Unit Supervisors responsibility. Incorrect because the procedure requires the Shift Manager to review the E-Plan. An NRC notification is not a direct step.

Distractor 2: Plausible because the TSC assumes responsibility for notification of outside agencies. Incorrect because activation of the TSC is not required for an Unusual Event classification. However, the Shift Manager would have the option of activating the ERO. If the ERO is activated, the required time to have TSC minimal staffing is within 1 hour. Until command and control is transferred, the responsibility resides with the Shift Manager/Shift Emergency Director.

Distractor 3: Plausible because a tornado touchdown or siting can occur without requiring activation of the E-Plan if it is not within the Protected Area. Incorrect because the question stem gives indication that the tornado stuck the reactor building and therefore within the Protected Area.

Reference: EP-AA-1006 Addendum 3, Rev.10, EP-AA-112-100-F-01, Rev.AD
Reference provided during examination: EP-AA-1006 ADDENDUM 3, Rev.10, Hot Matrix, pg. QC 2-11

Cognitive level: High

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

K/A: 295035 G2.4.37 SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: EMERGENCY PROCEDURES/PLAN: Knowledge of the lines of authority during implementation of the emergency plan implementing procedures.

Importance: SRO 4.1
10 CFR Part 55 Content: 41.10/45.13

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: New
Question History: N/A
Comments: None

86 **ID: 2478921** **Points: 1.00**

A transient occurred from 100% power on Unit 1 with the following alarms and automatic actions:
(NOT ALL ALARMS LISTED)

- 901-5 B-7, GROUP 1 ISOL CH TRIP
- 901-5 A-8, GROUP II ISOL CH TRIP
- RWCU system isolates
- Reactor Building Ventilation isolates
- B SBGTS autostarts and is running at 3500 SCFM
- APRM 1, 2, and 3 DOWNSCALE, HIGH and HIGH-HIGH lights lit

(1) What direction should be given to the NSO?

(2) Concerning the SBGTS ONLY, what, if any, Technical Specifications actions are required?

A. (1) Enter QOA 6800-03, 120/240 VAC ESSENTIAL SERVICE BUS FAILURE for a loss of the ESS Bus.
(2) No Technical Specification actions required.

B. (1) Enter QOA 7000-01, 120 VAC REACTOR PROTECTION BUS FAILURE (ONE OR BOTH BUSES) for a loss of RPS A.
(2) Enter TS 3.6.4.3 Condition A, One SGT subsystem inoperable.

C. (1) Enter QCOA 6800-01, 120/240 VAC INSTRUMENT BUS FAILURE, for a loss of the Instrument Bus.
(2) Enter TS 3.6.4.3 Condition A, One SGT subsystem inoperable.

D. (1) Enter QOA 7000-01, 120 VAC REACTOR PROTECTION BUS FAILURE (ONE OR BOTH BUSES) for a loss of RPS B.
(2) No Technical Specification actions required.

Answer: B

Answer Explanation

Answer Explanation: QOA 7000-01 lists the alarms and actuations of a loss of RPS A and RPS B. The alarms and actuations are nearly identical except for APRM 1, 2, and 3 which are powered from RPS A. The B SBGTS flow of 3500 scfm is below the minimum required 3600 scfm flowrate and is inoperable. TS LCO 3.6.4.3, Two SGT subsystems shall be OPERABLE, is NOT met. Entry into Condition A, One SGT subsystem inoperable is required with a 7 day completion time to restore B SBGTS to operable status.

Distractor 1: Plausible because alarms and actuations are consistent with a loss of an RPS Bus. Incorrect because RPS A has lost power and Tech Spec entry is required for one inoperable SGT subsystem.

Distractor 2: Plausible because some of the actuations are consistent with a loss of the Instrument Bus and entry into TS LCO 3.6.4.3 is required. Incorrect because the alarms, RWCU, and APRM indications are due to a loss of the Instrument Bus.

Distractor 3: Plausible because some of the alarms and actuations are consistent with a loss of the Essential Service Bus. Incorrect because entry into TS LCO 3.6.4.3 is required.

Reference: QOA 7000-01 Rev.40, TS 3.6.4.3 Amendment No. 273/268
Reference provided during examination: None

Cognitive level: High

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

K/A: 212000 G2.4.46 REACTOR PROTECTION SYSTEM: EMERGENCY PROCEDURES/PLAN: Ability to verify that the alarms are consistent with plant conditions.

Importance: SRO 4.2
10 CFR Part 55 Content: 41.10/43.5/45.3/45.12

SRO Justification:
Can the question be answered solely by knowing ≤ 1-hour TS/TRM action? **No**
Can the question be answered solely by knowing the LCO/TRM information listed “above the line”? **No**
Can the question be answered solely by knowing the TS safety limits? **No**
Can the question be answered solely by knowing the TS bases information associated with the “above-the-line” LCO information or general systems knowledge? **No**
Does the question involve one or more of the following for TS, TRM, or ODCM: **Yes**
• application of required actions (TS Section 3) and SRs (TS Section 4) in accordance with rules of application requirements (TS Section 1)
• application of generic LCO requirements (LCOs 3.0.1 through 3.0.7 and LCOs 4.0.1 through 4.0.4)
• knowledge of TS bases that is required to analyze TS required actions and terminology

Content: 10 CFR 55.43(b)(2)

Question Source: New
Question History: N/A
Comments: None

87 **ID: 2479013** **Points: 1.00**

- Unit 2 is in Mode 2.
- RPV pressure is 918 psig.
- One (1) Main Turbine Bypass valve 80% open.
- The crew is preparing to place the Mode Switch in RUN.
- APRM and IRM readings are:

APRM 1	APRM 2	APRM 3	APRM 4	APRM 5	APRM 6
6%	5.5%	2%	5.8%	6%	6%

IRM 11	IRM 12	IRM 13	IRM 14	IRM 15	IRM 16	IRM 17	IRM 18	
90	90	95	70	80	95	99	90	IRM Units
10	10	10	10	10	10	10	9	IRM Range

All IRM counts are slowly rising.
All other surveillances and plant conditions are met for the Mode Switch to be placed in RUN.

The Unit Supervisor should direct the NSO to...

A. bypass APRM 3, then place the Mode Switch in RUN.

B. insert a 1/2 scram on RPS Channel A, then place the Mode Switch in RUN.

C. insert control rods to maintain IRM countrate on scale, then recalibrate APRM 3.

D. partially withdraw IRM 13 to maintain countrate on scale, then continue control rod withdrawal until APRM 3 downscale clears.

Answer: A

Answer Explanation

Answer Explanation: Per QCGP 1-2, NORMAL UNIT 2 STARTUP, all APRM downscale lights must be clear. APRM 1 may be bypassed without entering any Technical Specification LCOs. Since all other surveillance and plant conditions are met the Mode Switch then can be placed in RUN.

Distractor 1: Plausible because with the Mode Switch in RUN, a 1/2 scram will occur with an APRM downscale and the companion IRM HI-HI (125 units). Incorrect because QCGP 1-2 step F.6.nn has the crew verify scram conditions are not present, (Condenser vacuum /APRM downscale lights clear), prior to moving the Mode Switch to RUN.

Distractor 2: Plausible because inserting control rods would result in lowering the countrate on all IRMs and a recalibration of APRM 3 may result in a power indication closer to 5%. Incorrect because inserting control rods may result in a subcritical reactor and down-ranging IRMs places an additional challenge on the crew. Control rod insertion should be used during a startup primarily to control excessive heatup rate or reactor period ONLY. Per QCGP 1-2 APRM calibration at low power (<~10%) should only be performed when operationally necessary (e.g. rod block or downscale). The given conditions do not meet this requirement.

Distractor 3: Plausible because SRMs are partially withdrawn to avoid an SRM HI control rod blocks until IRMs are on scale (overlap). Incorrect because withdrawing IRMs from the full in position results in a control rod block.

Reference: QCGP 1-2 Rev.51
Reference provided during examination: None

Cognitive level: High

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

K/A: 215003 A2.08 Ability to (a) predict the impacts of the following on the (SF7 IRM) INTERMEDIATE RANGE MONITOR SYSTEM and (b) based on those predictions to correct, control, or mitigate the consequences of those abnormal operations: Improper overlap

Importance: SRO 3.5
10 CFR Part 55 Content: 41.5/43.5/45.6

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: New
Question History: N/A
Comments: None

88 **ID: 2474641** **Points: 1.00**

Power ascension is in progress with Unit 1 at 10% RTP in Mode 1, when the following occurs:

- Annunciator 901-5 E-7, LPRM DOWNSCALE alarms.
- Annunciator 901-5 C-6, APRM DOWNSCALE alarms.

- LPRM 08-41A assigned to APRM 1 is reading downscale.
- APRM Channel 1 currently has 13 input LPRMs in OPERATE.

(1) What impact, if any, does the LPRM failure have on control rod withdrawal?
(2) What direction should the Unit Supervisor give to the NSO?

A. (1) A rod block exists.
(2) Bypass the failed LPRM ONLY per QCOP 0700-03, LOCAL POWER RANGE (LPRM) MONITORING.

B. (1) A rod block exists.
(2) Bypass APRM Channel 1 per QCOP 0700-04, AVERAGE POWER RANGE MONITORING SYSTEM OPERATION and then the failed LPRM per QCOP 0700-03, LOCAL POWER RANGE (LPRM) MONITORING.

C. (1) NO rod block exists.
(2) Bypass the failed LPRM ONLY per QCOP 0700-03, LOCAL POWER RANGE (LPRM) MONITORING.

D. (1) NO rod block exists.
(2) Bypass APRM Channel 1 per QCOP 0700-04, AVERAGE POWER RANGE MONITORING SYSTEM OPERATION and then the failed LPRM per QCOP 0700-03, LOCAL POWER RANGE (LPRM) MONITORING.

Answer: B

Answer Explanation

Answer Explanation: A rod block is issued when any APRM is downscale with the Reactor Mode switch in RUN. Additionally, if bypassing the LPRM results in only 12 LPRM inputs to APRM 1 it will become inoperable (setpoint < 13 LPRM inputs) generating a rod block and 1/2 scram. QCOP 0700-03, directs bypassing the APRM first if bypassing an LPRM will render it inoperable. Therefore the APRM and LPRM must be bypassed to continue rod withdrawal.

Distractor 1: Plausible because a rod block is generated by the APRM downscale and the LPRM must be bypassed. Incorrect because the APRM will become inoperable when the LPRM is bypassed generating a rod withdrawal block. The APRM must also be bypassed.

Distractor 2: Plausible because not all Nuclear Instrumentation downscale conditions generate a rod block, (ie. IRMs on range 1). Additionally, bypassing the LPRM is a correct action. Incorrect because a rod block is generated and the LPRM and associated APRM must be bypassed.

Distractor 3: Plausible because not all Nuclear Instrumentation downscale conditions generate a rod block, (ie. IRMs on range 1). Additionally, bypassing the LPRM and APRM are the correct actions. Incorrect because a rod block is generated by the APRM downscale and the APRM inoperable (too few inputs) when the LPRM is bypassed.

Reference: QCOP 0700-03 Rev.22, TS Bases 3.3.1.1 Section 2.a. Rev.0, QCAN 901(2)-5 C-12 Rev.10, QCAN 901(2)-5 C-6 Rev.05
Reference provided during examination: None

Cognitive level: High

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

K/A: 215005.A2.02: Ability to (a) predict the impacts of the following on the (SF7 PRMS) AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Upscale or downscale trips.

Importance: SRO 3.9
10 CFR Part 55 Content: 41.5/43.5/45.6

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: Bank
Question History: 2011 ILT NRC Exam
Comments: None

89 **ID: 2479161** **Points: 1.00**

Technical Specification surveillance requirement SR 3.6.1.3.6 is met by performing QCOS 0250-04, MSIV CLOSURE TIMING.
The surveillance frequency of every refueling outage is determined by the Inservice Testing (IST) Program.

Who's permission is required to change SR 3.6.1.3.6, QCOS 0250-04, MSIV CLOSURE TIMING, frequency from required every 2 years to required every 4 years? (Not all permissions required are listed)

A. Pre-Define Coordinator and Maintenance Manager.

B. Operations Management and Engineering Manager.

C. Radiation Protection Manager and Work Control Manager.

D. Work Control Manager and a Subject Matter Expert (SME).

Answer: B

Answer Explanation

Answer Explanation: Per ER-AA-425-1003, SURVEILLANCE FREQUENCY CONTROL PROGRAM – INTEGRATED DECISIONMAKING PANEL (IDP) ROLES AND RESPONSIBILITY, states the Operations and Engineering are voting members on the Integrated Decision Making Panel (IDP). This panel is responsible for:

- Reviewing a proposed Surveillance Test Interval (STI) change for both qualitative considerations and quantitative results
- Making recommendations on the way revised surveillance intervals are implemented
- Reviewing the cumulative impact of all changes carried out over a period of time
- Monitoring the impact of changes on failure rates

Distractor 1: Plausible because the Pre-Define Coordinator schedules IST program activities. Incorrect because the Pre-Define Coordinator is not involved in approval of technical changes or test frequencies associated with regulatory requirements. The Maintenance Manager is responsible for providing resources for tests/inspections required by the IST program, but not for approval of surveillance frequencies.

Distractor 2: Plausible because the Work Control Manager is responsible scheduling and coordinating test activities. Incorrect because neither the Work Control Manager or the Radiation Protection Manager would be required to approve surveillance frequency changes.

Distractor 3: Plausible because the Work Control Manager is a voting member of the approval panel. Incorrect because the SME is a non-voting member.

Reference: ER-AA-425-1003, Rev.03
Reference provided during examination: None

Cognitive level: Memory

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

K/A: 223002 G2.2.6 PRIMARY CONTAINMEMT / NUCLEAR STEAM SUPPLY SHUTOFF: EQUIPMENT CONTROL: Knowledge of the process for making changes to procedures.

Importance: SRO 3.6
10 CFR Part 55 Content: 41.10/43.3/45.13

SRO Justification:
Some examples of SRO-only examination items for this topic include the following:
• screening and evaluation processes under 10 CFR 50.59, “Changes, tests, and experiments”
• administrative processes for temporary modifications
• administrative processes for disabling annunciators
• administrative processes for the installation of temporary instrumentation
• processes for changing the plant or plant procedures **

Content: 10 CFR 55.43(b)(3)

Question Source: New
Question History: N/A
Comments: None

90 **ID: 2487799** **Points: 1.00**

Unit 1 was operating at full power when the following events occur:

- The 1-203-3D Relief Valve spuriously opens.
- The NSO places the 1-203-3D key switch to OFF and verifies the valve solenoid de-energizes but the Relief Valve does NOT close.
- Torus temperature is 90°F and RISING at ~5°/minute.

The next required actions directed by the Unit Supervisor are to ___(1)___,

and the 1-203-3D SRV is ___(2)___.

A. (1) insert a manual scram and enter QCGP 2-3, REACTOR SCRAM
(2) operable

B. (1) cycle the 1-203-3D key switch between MANUAL and AUTO and verify the valve closes
(2) inoperable

C. (1) remove all four control power fuses for the 1-203-3D Relief Valve and verify the valve closes
(2) operable

D. (1) enter QCGP 2-1, NORMAL UNIT SHUTDOWN and start both loops of Torus Cooling per QCOP 1000-09, TORUS COOLING START UP AND OPERATION
(2) inoperable

Answer: B

Answer Explanation

Answer Explanation: QCOA 0203-01, FAILURE OF A RELIEF VALVE TO CLOSE OR RESEAT PROPERLY, step D.1 requires cycling the key switch between MANUAL and AUTO to try and close the relief valve. If not successful, step D.2. requires a reactor scram before 95 deg F Torus temperature is reached. QCOS 0203-03 states the performance acceptance criteria for each relief valve is that it can be manually opened and closed therefore the SRV is inoperable.

Distractor 1: Plausible because inserting a manual reactor scram is required if the relief valve cannot be closed. Incorrect because the SRV is inoperable and the Torus temperature has not reached 95 °F. There is time to make another procedurally directed attempt to close the relief valve (1 minute).

Distractor 2: Plausible because QCOA 0203-01 step D.4. states, "IF Relief Valve solenoid can NOT be de-energized with keylock switch, THEN remove all four control power fuses from Panel 2201-32." and because the SRV is inoperable. Incorrect because the solenoid was deenergized when the key switch was taken to OFF and relief valve did NOT close.

Distractor 3: Plausible because QCOA 0203-01 step D.5 directs the start of Torus Cooling as does annunciator 901-4 G-17, TORUS WTR HIGH TEMP which is in alarm with Torus temperature at 90 °F and the SRV is inoperable. Incorrect because an attempt to close the relief valve takes priority over starting Torus Cooling.

Reference: QCOA 0203-01, Rev 13, QCAN 901(2)-4 G-17, Rev.14 QCOS 0203-03, Rev. 28.
Reference provided during examination: None

Cognitive level: High

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

K/A: 239002 A2.03 Ability to (a) predict the impacts of the following on the (SF3 SRV) SAFETY RELIEF VALVES and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Stuck-open SRV.

Importance: SRO 4.4
10 CFR Part 55 Content: 41.5/43.5/45.6

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: New
Question History: N/A
Comments: None

91 **ID: 2479167** **Points: 1.00**

An automatic scram occurred on Unit 2.
The crew has entered QCGP 2-3, REACTOR SCRAM.
Several minutes after the RWCU system was restarted in Reject Mode, the NSO reports:

- Annunciator 902-4 F-12, CU SYSTEM AFTER NONREG HX HIGH TEMP is in alarm.
- The RWCU system has NOT isolated.
- TI 2-1290-21. RWCU LOOP TEMP, Pt 3, NON-REGEN HX OUTLET is reading 100°F.

Select the appropriate action.

(Reference provided**)**

A. Raise RWCU reject flow until alarm 902-4 F-12, CU SYSTEM AFTER NONREG HX HIGH TEMP clears.

B. Dispatch an EO to raise the RWCU isolation setpoint per QCOP 1200-13, RWCU SYSTEM HIGH TEMPERATURE ALARM AND ISOLATION SETPOINT ADJUSTMENT.

C. Add the alarm to the Equipment Status Tag Log (EST Log) and place an EST per OP-AA-108-101, Control of Equipment and System Status.

D. Add the alarm to the MCR Equipment Deficiency List and place an Equipment Deficiency Tag per OP-AA-108-105-1001, MCR AND RWCR EQUIPMENT DEFICIENCY MANAGEMENT AND PERFORMANCE INDICATOR SCREENING.

Answer: D

Answer Explanation

Answer Explanation: OP-AA-108-105-1001, step 2.1 includes annunciators and alarms that "impact operator ability to assess plant operation and to take action, but do not rise to the level of an operator workaround or burden". An Equipment Deficiency Tag or sticker is generated and placed either on the alarm tile or on the 901-4 panel near the RWCU system controls.

Distractor 1: Plausible because adjusting Reject flow is directed by QCAN 901(2)-4 F-12 if the system did not isolate. Incorrect because the Nonregen Heat Exchanger outlet temperature was 100 deg.F. This is below the alarm setpoint (130 deg.F) and the high temp isolation setpoint (140 deg.F).

Distractor 2: Plausible because QCOP 1200-07, RWCU SYSTEM COOLANT REJECTION allows adjustment of the high temperature and alarm setpoint under certain conditions. Incorrect because the Nonregen Heat Exchanger outlet temperature is 100 deg.F, which is below the alarm setpoint and high temp isolation setpoint.

Distractor 3: Plausible because ESTs are used to identify temporary equipment status. Incorrect because the EST program is used to control equipment positioning (switches, valves, etc.) to ensure configuration control.

Reference: OP-AA-108-105-1001 Rev.09, QCAN 901(2)-4 F-12 Rev.08
Reference provided during examination: QCAN 901(2)-4 F-12

Cognitive level: High

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

K/A: 204000 G2.2.43 REACTOR WATER CLEANUP SYSTEM: EQUIPMENT CONTROL: Knowledge of the process used to track inoperable alarms.

Importance: SRO 3.3
10 CFR Part 55 Content: 41.10/43.5/45.13

SRO Justification:
Some examples of SRO-only examination items for this topic include the following:
• screening and evaluation processes under 10 CFR 50.59, “Changes, tests, and experiments”
• administrative processes for temporary modifications
• administrative processes for disabling annunciators **
• administrative processes for the installation of temporary instrumentation
• processes for changing the plant or plant procedures

Content: 10 CFR 55.43(b)(3)

Question Source: New
Question History: N/A
Comments: None

92 **ID: 2479240** **Points: 1.00**

A Loss of Offsite power and a LOCA has occurred on Unit 1.
Emergency Diesel Generators are supplying the ECCS Busses and are backfeeding Bus 13 and Bus 14.
QGA 100, RPV CONTROL and QGA 200, PRIMARY CONTAINMENT CONTROL actions are in progress.
Drywell and Torus sprays are operating on Division I.
Torus Cooling is operating on Division II.

- RPV water level is 0 inches and STEADY
- RPV pressure is 700 psig and LOWERING SLOWLY
- Drywell pressure is 7 psig and SLOWLY LOWERING
- Torus pressure is 5 psig and SLOWLY LOWERING
- Drywell temperature is 170°F and LOWERING
- Torus temperature is 100°F and SLOWLY RISING

An EO calls in and reports:
- Smoke is coming from the 1/2 EDG room.

The NSO reports voltage and frequency oscillations on 1/2 EDG.

Which of the following actions should be assigned?

A. Continue operation and locally monitor the 1/2 EDG.

B. Secure the 1/2 EDG and start another RHRSW pump on B Loop.

C. Secure the 1/2 EDG, cross tie Bus 13-1 and Bus 23-1, and restart Torus sprays.

D. Secure the 1/2 EDG, energize Bus 13-1 with the U1 SBO Diesel, and restart Torus sprays.

Answer: C

Answer Explanation

Answer Explanation: The reports of smoke in the 1/2 EDG room and voltage/frequency oscillations suggest an imminent failure of the 1/2 EDG. The next available safety related power supply for Bus 13-1 is through the crosstie breakers from Bus 23-1. Securing the 1/2 EDG and re-energizing Bus 13-1 will allow the crew to re-establish Torus sprays and then Drywell sprays after verifying containment parameters are within the DSIL curve.

Distractor 1: Plausible because the plant is in an emergency and 1/2 EDG operation is needed. Incorrect because 1/2 EDG failure is imminent, and the fluctuating voltages and frequency will damage required equipment. QCAN 901-8 F-4, SHARED DIESEL LOW FREQUENCY states to shut down the Diesel when another source of power is available.

Distractor 2: Plausible because starting an RHRSW pump on Division II replaces the loss of system cooling provided by the RHRSW pump on Division I. RHR flow to the Torus spray valves is still provided through the crosstie valves (MO 1-1001-19A and MO 1-1001-19B) or they can be established on B RHR Loop. Incorrect because the U-1 EDG will exceed maximum loading with both RHR and RHRSW pumps running, especially if the CS pump is still operating (QCOA 6100-03 Loss of Offsite Power).

Distractor 3: Plausible because the U-1 SBO is capable of supplying Bus 13-1. Incorrect because under accident conditions, a safety related power source is preferred over a non-safety related power source and for station blackout it specifies attempting to use the crosstie first per QCOA 6100-04.

Reference: QGA 200 Rev.13, QCOA 6100-03 Rev.43, QCOA 6100-04, Rev. 27, QCAN 901-8 F-4 Rev. 8.
Reference provided during examination: None

Cognitive level: High
Level (RO/SRO): SRO
Tier: 2 **Group:** 2

K/A: 230000 A2.07 Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Emergency generator failure.
Importance: SRO 3.7
10 CFR Part 55 Content: 41.5/43.5/45.6

SRO Justification: Can the question be answered solely by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)
Question Source: New
Question History: N/A
Comments: None
Justification for how references supplied during 2023 ILT exam do not give away answers: This is a power supply / procedural priority question for Station Black Out. The QGAs do not provide any useful information to help answer this question. Examinee must know power supplies to RHRSW Pump, limitations on a diesel for loading, and priority procedurally between SBO and cross-tie when an ECCS bus is lost. None of that information is provided in the QGA references.

Test Answer Key

2023 ILT NRC SRO Exam Final Version
Test ID: 369040

93 **ID: 2479280** **Points: 1.00**

Unit 1 is currently in Mode 4 in preparation for a scheduled refuel outage.
Unit 2 is operating at 100% power when the following occurs:

- Annunciator 912-5 C-1, RX BLD 1 LOW DP, alarms.
- Reactor Building to Atmosphere DP gauge is indicating +0.1 in.H_2O ~~vac~~.

Shortly after, an EO calls in to the control room and reports:

- Both Interlock Doors between the Unit 1 Reactor Feed Pump room and the Reactor Building are blocked open by contractors moving heavy equipment into the Reactor Building.
- It will take about 20 minutes to clear the doorways.
- This was NOT a pre-planned evolution.

What actions are REQUIRED?

(Reference provided**)**

A. Close one Interlock Door within 1 hour and lock it within 24 hours ONLY.

B. Close one Interlock Door within 4 hours and notify the NRC via the ENS within 1 hour per SAF 1.3.

C. Close one Interlock Door within 4 hours and notify the NRC via the ENS within 8 hours per SAF 1.4.

D. Close one Interlock Door within 4 hours and notify the NRC via the ENS within 8 hours per SAF 1.8.

Answer: D

Answer Explanation

*** Updated with NRC concurrence based on an examinee question early during exam administration.**

Answer Explanation: Per QCOP 1600-24, SECONDARY CONTAINMENT INTEGRITY step F.1.b. (1) through (7) requires secondary containment interlock doors to be operable with at least one door in each interlock closed. TS LCO 3.6.4.1 Applicability requires an operable secondary containment in MODES 1, 2, and 3, and during movement of recently irradiated fuel assemblies in the secondary containment. Since Unit 2 is in Mode 1, Condition A must be entered requiring a restoration of secondary containment within 4 hours. Per LS-AA-1110, a loss of secondary containment is a reportable event under SAF 1.8 if the following criteria is met: (1) secondary containment is required in the current mode, (2) declared inoperable, and (3) the breach was not a planned evolution.

Distractor 1: Plausible because this is a TS required action for an inoperable Air Lock Door. Incorrect because this action pertains to primary containment.

Distractor 2: Plausible because TS 3.6.4.1 and QCOP 1600-24 require one interlock door closed within 4 hours. Incorrect because reportability under SAF 1.3 does not apply as this is not a deviation from TS due to an emergency action needed to protect the health and safety of the public (50.54(x).

Distractor 3: Plausible because TS 3.6.4.1 and QCOP 1600-24 require one interlock door closed within 4 hours. Incorrect because reportability under SAF 1.4 does not apply because it does not qualify as a serious or unanalyzed degradation of secondary containment. Starting a SBGTS train would restore a negative DP until at least one interlock door is closed.

Reference: QCOP 1600-24 Rev.09, TS 3.6.4.1 Amendment No. 273/268, LS-AA-1110 Rev.33
Reference provided during examination: TS 3.6.4.1, TS 3.6.1.2, LS-AA-1110 SAF 1.3, 1.4, and 1.8.

Cognitive level: High

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

K/A: 290001 A2.08 Ability to (a) predict the impacts of the following on the SECONDARY CONTAINMENT and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of secondary containment integrity.

Importance: SRO 3.9
10 CFR Part 55 Content: 41.5/43.5/45.6

SRO Justification:
Can the question be answered solely by knowing ≤ 1-hour TS/TRM action? **No**
Can the question be answered solely by knowing the LCO/TRM information listed “above the line”? **No**
Can the question be answered solely by knowing the TS safety limits? **No**
Can the question be answered solely by knowing the TS bases information associated with the “above-the-line” LCO information or general systems knowledge? **No**
Does the question involve one or more of the following for TS, TRM, or ODCM: **Yes**
• application of required actions (TS Section 3) and SRs (TS Section 4) in accordance with rules of application requirements (TS Section 1)
• application of generic LCO requirements (LCOs 3.0.1 through 3.0.7 and LCOs 4.0.1 through 4.0.4)
• knowledge of TS bases that is required to analyze TS required actions and terminology

Content: 10 CFR 55.43(b)(2)

Question Source: New
Question History: N/A
Comments: None

Test Answer Key

2023 ILT NRC SRO Exam Final Version
Test ID: 369040

94 **ID: 2479311** **Points: 1.00**

- You are the Shift Manager.
- An ATWS has occurred on Unit 2.
- The Unit 2 Supervisor and Shift Manager are the only qualified STAs on shift.

The Unit 2 Supervisor and STA duties are to be fulfilled by the _(1)_ and _(2)_, respectively.

A. (1) Unit 2 Supervisor
(2) Shift Manager

B. (1) WEC SRO
(2) Unit 2 Supervisor

C. (1) Unit 2 Supervisor
(2) Unit 1 Supervisor

D. (1) Shift Manager
(2) Unit 2 Supervisor

Answer: B

Answer Explanation

Answer Explanation: OP-AA-101-111, ROLES AND RESPONSIBILITIES OF ON-SHIFT PERSONNEL, step 4.5. states that if a station uses the STA to fill the role of a Unit SRO position, then in cases where the EOP's are entered, or a scram or transient occurs, the STA turns over the Unit Supervisor position to another qualified SRO, and then performs STA functions. In this case, the Unit 2 Supervisor will turnover to the WEC SRO and then perform the STA function.

Distractor 1: Plausible because the U2 Supervisor and Shift Manager are qualified to fill either position. Incorrect because the Shift Manager i is required to assume the duties of the Shift Emergency Director.

Distractor 2: Plausible because the U2 Supervisor is required to perform STA duties. Incorrect because the Shift Manager is required to perform the Shift Emergency Director role.

Distractor 3: Plausible because the U2 Supervisor is qualified to direct the EOPs. Incorrect be the U1 Supervisor is NOT STA qualified.

Reference: OP-AA-101-111 Rev.13
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 3

K/A: G2.1.6 CONDUCT OF OPERATIONS: Ability to manage the control room crew during plant transients (SRO Only)

Importance: SRO 4.8
10 CFR Part 55 Content: 43.5/45.12/45.13

SRO Justification:
Examples of SRO-only examination items for this topic include the following:
• reporting requirements when the maximum licensed thermal power output is exceeded administration of fire protection program requirements, such as compensatory actions associated with inoperable sprinkler systems and fire doors
• required actions necessary when a facility does not meet the administrative controls listed in Technical Specifications (TS) Section 5 or 6, depending on the facility (e.g., shift staffing requirements) ***
• National Pollutant Discharge Elimination System requirements, if applicable
• processes for TS and final safety analysis report changes

Content: 10 CFR 55.43(b)(1)

Question Source: New
Question History: N/A
Comments: None

95 **ID: 2487338** **Points: 1.00**

Which of the following set of parameters would be indicative of a failed containment (1) and why (2)?

Set Number #	Drywell Pressure	Torus Pressure
#1	7 psig	12 psig
#2	9.6 psig	10 psig
#3	10 psig	10 psig
#4	40 psig	38 psig

A. (1) Set #1 ONLY
(2) At least 1 Torus-to-Drywell vacuum breaker has become inoperable.

B. (1) Set #1 and Set #2 ONLY
(2) At least 1 Torus-to-Drywell vacuum breaker has become inoperable.

C. (1) Set #3 ONLY
(2) Drywell pressure and Torus pressure should NEVER equalize.

D. (1) Set #4 ONLY
(2) The design pressure for the Drywell and Torus have been exceeded.

Answer: A

Answer Explanation

Answer Explanation: Per T.S. Bases 3.6.1.8 Suppression chamber-to-drywell vacuum breakers prevent an excessive negative differential pressure across the suppression chamber-drywell boundary. This prevents the Torus pressure from exceeding the Drywell pressure. The surveillance requirements and T.S. Bases state that the operability requirement for a torus vacuum breaker is to open $\leq$ 0.5 psid. UFSAR Section 6.2. also states that the Torus-to-Drywell vacuum breakers should have opened at/before .5 psid.

Distractor 1: Plausible because Torus pressure has exceeded Drywell pressure for both sets. Incorrect because the Drywell-Torus differential pressure has not yet reached the Drywell to Torus Vacuum Breaker full lift setpoint.

Distractor 2: Plausible because Drywell-Torus differential pressure is not maintained. Incorrect because Drywell and Torus pressures will eventually equalize according to the long term LOCA analysis.

Distractor 3: Plausible because Torus pressure has exceeded the PSP Limit. Incorrect because Containment design pressure is 56 psig per UFSAR section 6.2.1.1.

Reference: T.S Bases 3.6.1.8 , Rev. 0, UFSAR Section 6.2.1.1 Rev. 6,
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 3

K/A: G2.1.19 CONDUCT OF OPERATIONS: Ability to use available indications to evaluate system or component status.

Importance: SRO 3.8
10 CFR Part 55 Content: 41.10 / 45.12

SRO Justification:
Can the question be answered solely by knowing ≤ 1-hour TS/TRM action? **No**
Can the question be answered solely by knowing the LCO/TRM information listed “above the line”? **No**
Can the question be answered solely by knowing the TS safety limits? **No**
Can the question be answered solely by knowing the TS bases information associated with the “above-the-line” LCO information or general systems knowledge? **No**
Does the question involve one or more of the following for TS, TRM, or ODCM: **Yes**
• application of required actions (TS Section 3) and SRs (TS Section 4) in accordance with rules of application requirements (TS Section 1) ***Yes**
• application of generic LCO requirements (LCOs 3.0.1 through 3.0.7 and LCOs 4.0.1 through 4.0.4)
• knowledge of TS bases that is required to analyze TS required actions and terminology ***Yes**

Content: 10 CFR 55.43(b)(2)

Question Source: New
Question History: N/A
Comments: None

96 **ID: 2479369** **Points: 1.00**

Which of the following is a VIOLATION of the Conditions and Limitations in the Facility License?

A. Operation in Mode 1 with a Reactor Coolant System UNIDENTIFIED leakage rate of 3 gpm.

B. An Appendix R procedure change that results in 80 hours to achieve cold shutdown, WITHOUT NRC approval.

C. Control Room staff is momentarily ONE Senior Reactor Operator and THREE Reactor Operators with Unit 1 in Mode 1 and Unit 2 in Mode 5.

D. Steady state thermal power below the license thermal power limit, with momentary indications ABOVE the limit as a result of normal process fluctuations.

Answer: B

Answer Explanation

Answer Explanation: The 10 CFR 50 Appendix R requirement is to achieve cold shutdown in 72 hours. The Facility license allows changes to the fire protection program without NRC approval if those "changes WOULD NOT adversely affect the ability to achieve and maintain safe shutdown in the event of a fire."

Distractor 1: Plausible because operation with unidentified leakage is allowed. Incorrect because the Technical Specification limit for unidentified RCS leakage is < 5 gpm. (Reference TS LCO 3.4.4 (b)). Operation is within the TS LCO and NOT a violation of the Facility Operating License.

Distractor 2: Plausible because staffing for Modes 1, 2, and 3 require an SRO on each Unit. Incorrect because an SRO is not required for a unit in Mode 4 or Mode 5 per Tech Spec 5.2.2 and 10 CFR 50.54(m)(2)(i).

Distractor 3: Plausible because the Facility License has a maximum thermal power limit. Incorrect because power fluctuations as a result of normal oscillations due to stabilizing after control rod movement or flow adjustments are NOT considered "intentional operation above the Facility License limit of 2957 MWth". Per QCGP 3-1, During full power operation the Reactor Operator will closely monitor thermal power trends and take action as required to maintain a goal of keeping the 1-hour thermal power average at or below the license limit.

Reference: 10 CFR50 Appendix R, DPR 29
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): SRO
Tier: 3

K/A: G2.2.38 EQUIPMENT CONTROL: Knowledge of conditions and limitations in the facility license.

Importance: SRO 4.5
10 CFR Part 55 Content: 41.7/41.10/43.1/45.13

SRO Justification:
Examples of SRO-only examination items for this topic include the following:
• reporting requirements when the maximum licensed thermal power output is exceeded administration of fire protection program requirements, such as compensatory actions associated with inoperable sprinkler systems and fire doors
• required actions necessary when a facility does not meet the administrative controls listed in Technical Specifications (TS) Section 5 or 6, depending on the facility (e.g., shift staffing requirements)
• National Pollutant Discharge Elimination System requirements, if applicable
• processes for TS and final safety analysis report changes ***

Content: 10 CFR 55.43(b)(1)

Question Source: Bank
Question History: 2009 ILT NRC Exam
Comments: None

97 **ID: 2487804** **Points: 1.00**

Unit 2 is at 100% power when Chemistry determines that reactor coolant specific activity is 0.6 μCi/gm Dose Equivalent Iodine (I-131).

Complete the following statements:

- This is __(1)__ the Technical Specification limit for reactor coolant specific activity.
- Per the bases of TS 3.4.6, RCS Specific Activity, the reactor coolant specific activity limit __(2)__.

A. (1) below
(2) ensures the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) is not exceeded

B. (1) above
(2) prevents stress corrosion cracking of the stainless steel reactor materials that are in contact with the coolant

C. (1) below
(2) prevents stress corrosion cracking of the stainless steel reactor materials that are in contact with the coolant

D. (1) above
(2) ensures the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) is not exceeded

Answer: D

Answer Explanation

Answer Explanation: The specific iodine activity is limited to < 0.2 mCi/gm Dose Equivalent Iodine (I-131). This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so the dose consequence of any release of radioactivity to the environment during a MSLB is less than a small fraction of the 10 CFR 50.67 limits at the site boundary and less than 10 CFR 50.67 limits for the control room.

Distractor 1: Plausible since the reactivity coolant activity limit does ensure the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) is not exceeded and because the TLCO 3.4.b limit for RCS conductivity is 1.0 mmhos/cm at 25°C. Incorrect because the I-131 limit is ≤ 0.2 uCi/gm.

Distractor 2: Plausible because it is above the limit of 0.2 uCi/gm. Incorrect because chloride limits listed in TLCO 3.4.b, RCS Chemistry, are to prevent stress corrosion cracking of the stainless steel materials that are in contact with the coolant not for activity.

Distractor 3: Plausible because the TLCO 3.4.b limit for RCS conductivity is 1.0 mmhos/cm at 25°C and RCS conductivity limit is based on preventing stress corrosion cracking. Incorrect because the question is asking for limits and concerns for the activity associated with I-131.

Reference: TS 3.4.6 and bases Amendment 223/218
Reference provided during examination: None.

Cognitive level: Memory

Level (RO/SRO): SRO
Tier: 3

K/A: G2.2.42 EQUIPMENT CONTROL: Ability to recognize system parameters that are entry-level conditions for technical specifications.

Importance: SRO 4.6
10 CFR Part 55 Content: 41.7/41.10/43.2/43.3/45.3

SRO Justification:
Can the question be answered solely by knowing ≤ 1-hour TS/TRM action? **No**
Can the question be answered solely by knowing the LCO/TRM information listed “above the line”? **No**
Can the question be answered solely by knowing the TS safety limits? **No**
Can the question be answered solely by knowing the TS bases information associated with the “above-the-line” LCO information or general systems knowledge? **No**
Does the question involve one or more of the following for TS, TRM, or ODCM: **Yes**
• application of required actions (TS Section 3) and SRs (TS Section 4) in accordance with rules of application requirements (TS Section 1)
• application of generic LCO requirements (LCOs 3.0.1 through 3.0.7 and LCOs 4.0.1 through 4.0.4)
• knowledge of TS bases that is required to analyze TS required actions and terminology
Content: 10 CFR 55.43(b)(2)

Question Source: Bank
Question History: 2016 NRC Exam
Comments: No significant changes made. Only revisions and template updates.

98 **ID: 2479400** **Points: 1.00**

Acting as the Station Emergency Director, you have been requested to authorize Emergency Exposure Limits to allow entry into an area to save the lives of an EO and RP Technician.

A volunteer, who is FULLY AWARE of the risks involved, has agreed to attempt the rescue. The volunteer is an adult male, Constellation employee (Occupational Worker) who does NOT have high lifetime exposure AND has NOT had any Planned Special Exposures OR administrative increases in his exposure limits.

Complete the following two statements regarding the volunteer's Total Effective Dose (TEDE) limits.

PRIOR TO this event, company policy administratively limits the ANNUAL exposure of a worker to _(1)_ REM.
DURING this event, the MAXIMUM exposure you can authorize is _(2)_.

A. (1) 2
(2) 25 REM

B. (1) 5
(2) 25 REM

C. (1) 2
(2) GREATER than 25 REM

D. (1) 5
(2) GREATER than 25 REM

Answer: C

Answer Explanation

Answer Explanation: Per RP-AA-203, EXPOSURE CONTROL AND AUTHORIZATION, the Exelon administrative limit is 2000 mrem routine cumulative TEDE/yr for adults. For a lifesaving activity, the worker fully aware of the risks involved and voluntary, the limit for dose exposure is > 25 REM.

Distractor 1: Plausible because the Company Administrative Dose Limit to an Occupational Worker is 2 REM. Incorrect because the
25 REM is the lifesaving limit for individuals NOT fully aware, ie. briefed on the possible health effects of the exposure.

Distractor 2: Plausible because 5 REM is the NRC annual limit. Incorrect because 25 REM is the lifesaving limit for individuals NOT fully aware.

Distractor 3: Plausible because the lifesaving exposure limit for a fully aware worker is > 25 REM. Incorrect because 5 REM is the NRC annual limit for an occupational worker.

Reference: RP-AA-203 Rev.06
Reference provided during examination: None

Cognitive level: Memory

Level (RO/SRO): SRO
Tier: 3

K/A: G2.3.14 RADIATION CONTROL: Knowledge of radiation or contamination hazards that may arise during normal, or emergency conditions or activities, such as analysis and interpretation of radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures, or analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits.

Importance: SRO 3.1
10 CFR Part 55 Content: 43.4/45.10

SRO Justification:
Some examples of SRO-only examination items for this topic include the following:
• process for gaseous/liquid release approvals (i.e., release permits)
• analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures
• analysis and interpretation of coolant activity, including comparison to emergency plan criteria or regulatory limits (or both)
• process for authorizing emergency exposure limits ***

Content: 10 CFR 55.43(b)(4)

Question Source: Bank
Question History: 2009 ILT NRC Exam
Comments: None

99 **ID: 2479411** **Points: 1.00**

Unit 2 was at 100% power when a manual scram was inserted due to a LOCA.
After RPV water level was recovered, Drywell Pressure exceeded the entry condition value for QGA 100, RPV Control and QGA 200, Primary Containment Control.

(1) What automatic PCIS Group Isolation have occurred?
(2) What Emergency Plan Declaration is required?

(Reference provided**)**

A. (1) Group II ONLY
(2) ALERT

B. (1) Group II and Group III
(2) ALERT

C. (1) Group II ONLY
(2) UNUSUAL EVENT

D. (1) Group II and Group III
(2) UNUSUAL EVENT

Answer: B

Answer Explanation

Answer Explanation: The entry condition into QGA 100 and QGA 200 for Drywell Pressure is 2.5 psig. A reactor scram from full power will result in an RPV water level drop to < 0 inches. The setpoints for the Groups II isolation are 0 inches RPV water level, 2.5 psig Drywell Pressure, and 100 R/hr Drywell radiation. The setpoint for the Group III isolation is 0 inches RPV water level. Therefore, the Group II and Group III isolations have occurred. With the LOCA resulting in 2.5 psig Drywell pressure, a loss of the RCS barrier per RC3 is met. Loss of the RCS or Fuel Clad barrier meets EAL FA1. Therefore, an E-Plan ALERT classification is required.

Distractor 1: Plausible because a Group II isolation occurs on RPV water level and Drywell pressure. The E-Plan ALERT declaration is also met. Incorrect because a Group III isolation has also occurred due to RPV low water level.

Distractor 2: Plausible because a Group II isolation occurs on RPV water level and Drywell pressure. Incorrect because a Group 3 isolation has also occurred and the E-Plan declaration of an UNUSUAL EVENT is incorrect.

Distractor 3: Plausible because Group II and Group III isolations have occurred. Incorrect because the E-Plan declaration is an ALERT.

Reference: QGA 100 Rev.13, QGA 200 Rev.13, QCAP 0200-10 Rev.60, EP-AA-1006 Addendum 3 Rev.10
Reference provided during examination: EP-AA-1006 Addendum 3, Fission Product Barrier Matrix, pg. QC 2-3

Cognitive level: High
Level (RO/SRO): SRO
Tier: 3

K/A: G2.4.2 Knowledge of system setpoints, interlocks and automatic actions associated with emergency and abnormal operating procedure entry conditions.

Importance: SRO 4.6
10 CFR Part 55 Content: 41.7/45.7/45.8

SRO Justification: Can the question be answered solely by knowing “systems knowledge”, i.e., how the system works, flowpath, logic, component location? **No**
Can the question be answered solely by knowing immediate operator actions? **No**
Can the question be answered solely by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs? **No**
Can the question be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? **No**
Content: 10 CFR 55.43(b)(5)

Question Source: New
Question History: N/A
Comments: None

Justification for how references supplied during 2023 ILT exam do not give away answers: The question requires entry condition knowledge for QGA 100 and QGA 200. That information is not included in the references provided (it is redacted in QGA 200 and QGA 100 is not provided).

100 **ID: 2479430** **Points: 1.00**

QGA 400, RADIOACTIVITY RELEASE CONTROL, directs operators to perform an Emergency Depressurization before off-site release rates reach a specified level.

What is the basis for this action?

(1) To maintain Turbine Building habitability.

(2) To place the primary system in its lowest energy state.

(3) Prevent the spread of radioactive material to the Main Condenser via the bypass valves.

(4) Reduce the discharge rate to the environment.

A. (1) and (3) ONLY

B. (1) and (4) ONLY

C. (2) and (4) ONLY

D. (2) and (3) ONLY

Answer: C

Answer Explanation

Answer Explanation: TSG guidelines state that an RPV blowdown may be required to do the following: (1) Maximize injection flow into the RPV from motor-driven systems, (2) Minimize discharge of reactor coolant from an unisolable primary system break, (3) Reduce the energy contained in the RPV in anticipation of loss of pressure suppression capability, (4) Minimize radioactivity release from the RPV into the primary containment, reactor building, or areas outside the primary and secondary containments, (5) Minimize continued energy addition from the RPV to the primary or secondary containment. Specifically, the QGA 400 Blowdown is performed to accomplish (3) and (4). QGA 300 blowdown actions are based on minimizing releases into secondary containment, and QGA 200 blowdown actions protect primary containment.

Distractor 1: Plausible because highly radioactive steam flow to other areas would be reduced and result in lower general area dose rates. Incorrect because the blowdown is performed to limit off site release rates and MSIV closure is designed to isolate the reactor from the Main Turbine.

Distractor 2: Plausible because it does reduce release rate. Incorrect because maintaining Turbine Building habitability is accomplished by operating Turbine Building ventilation fans.

Distractor 3: Plausible because reducing the energy state of the primary system is an action to terminate the primary system leakage. Incorrect because MSIV closure is designed to isolate the reactor from the Main Turbine.

Reference: Quad Cities Nuclear Power Station Technical Support Guidelines (TSG) Reference Manual Rev.07
Reference provided during examination: None

Cognitive level: High

Level (RO/SRO): SRO
Tier: 3

K/A: G2.4.6 EMERGENCY PROCEDURES/PLAN: Knowledge of emergency and abnormal operating procedures major action categories.

Importance: SRO 4.7
10 CFR Part 55 Content: 41.10/43.5/45.13

SRO Justification:
Some examples of SRO-only examination items for this topic include the following:
• process for gaseous/liquid release approvals (i.e., release permits)
• analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures**
• analysis and interpretation of coolant activity, including comparison to emergency plan criteria or regulatory limits (or both) **
• process for authorizing emergency exposure limits

Content: 10 CFR 55.43(b)(4)

Question Source: Bank
Question History: 2016 ILT NRC Exam
Comments: None