# Beaver Valley Unit 2 NRC Written Exam (2LOT17)

1. Which of the following are **NOT** DIRECT or RNO operator actions called for by FR-S.1 in the event of an ATWS?

    1.) Locally Open Reactor Trip Breakers
    2.) Check PRZR pressure LESS THAN 2375 PSIG
    3.) Locally Open Rod Drive MG Sets Output Breakers
    4.) Verify TDAFW Pump Steam Supply Valves – OPEN
    5.) Initiate Safety Injection
    6.) Initiate Main Steam Line Isolation

A. 1 and 3

B. 2 and 6

C. 2 and 5

D. 4 and 6

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**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Both are valid FRS1 actions.
**B.** Incorrect. Initiate Main Steam Line Isolation (6) is an RNO action.
**C.** Correct. Actual actions are to verify < 2330 psig for emergency boration and check SI initiated.
**D.** Incorrect. Both are valid FRS1 actions.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 007 | Reactor Trip | EA2. Ability to determine or interpret the following as they apply to a reactor trip: | Proper actions to be taken if the automatic safety functions have not taken place |

**K/A#** EA2.02 **K/A Importance** 4.3 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.FR-S.1 Iss. 2 Rev. 0

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41 (b)10

**Objective:**

2. The plant is operating at 100% power.

- A PRZR vapor space break accident occurs.
- PRZR pressure drops to 1200 psig.
- The Highest Steam Generator pressure is 1000 psig.
- HHSI flow is 800 gpm and stable.
- NO Orange or Red path conditions exist.
- The crew is performing the actions of E-1, Loss of Reactor or Secondary Coolant.
- At Step 2, Check if RCPs should be stopped, the crew is directed to Stop ALL RCPs.

Why MUST the RCPs be stopped at this time?

The RCPs are tripped to ______________

A. prevent possible pump damage by running the RCPs under highly voided conditions in order to save the pumps for potential future use.

B. prevent excessive depletion of RCS water inventory which might lead to severe core uncovery if the RCPs were tripped later in the event.

C. remove their added heat input, thereby ensuring the steam generators will be capable of performing the subsequent RCS cooldown.

D. ensure the RCP seal package is not damaged by the excessive temperature or steam voiding associated with this event.

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**Answer: B**

**Explanation/Justification:**

**A.** Incorrect Plausible because this is the reason they are stopped in FR-C.2, but not for a SBLOCA.

**B.** Correct. IAW with E-1 step 2 basis and RCP trip generic issue. D/P < 205 psid between highest SG pressure and RCS pressure, and HHSI is indicated.

**C.** Incorrect. Plausible since RCPs are tripped in FR-H.1 to remove their heat input, but it is done to extend the effectiveness of the remaining inventory. This is also plausible since tripping the pumps will remove the heat input and removing the heat input will ensure the only heat that will be required to be removed is from decay heat alone BUT this is not the reason why they are tripped in E-I.

**D.** Incorrect. Plausible since this is the consequence of losing both seal injection and RCP thermal barrier cooling.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000008 | Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open) | AK3. Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: | RCP tripping requirements |

**K/A#** AK3.04 **K/A Importance** 4.2 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53B.4.E-1 Iss 3 Rev 1 step 2 bases
2OM-53B.5GI-6 Iss 2 Rev 0 page 6 2nd para.

**Question Source:** Bank - 2LOT6 Q2

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** (CFR 41.5,41.10 / 45.6 / 45.13)

**Objective:**

3. The plant has experienced a LOCA and loss of all feed from 100% power. Plant conditions are:

- All SG pressures 900 psig, trending down slowly
- All SG narrow range levels are off scale low
- RCS pressure 600 psig, trending down slowly
- All other ECCS equipment has operated properly
- The crew has just entered FR-H.1 from E-0, step 9, Verify AFW Status

What procedural action is required?

A. Remain in FR-H.1 to restore feed.

B. Exit FR-H.1, return to E-0 due to plant conditions.

C. Remain in FR-H.1, perform in parallel with E-0.

D. Exit FR-H.1, transition to E-1.

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**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Plausible if RCS pressure was greater than any non-faulted SG pressure indicating that a heat sink was required IAW step 1.

**B.** Correct. Transition back to E-0 is warranted due to a heat sink not being required due to SG pressure being greater than RCS pressure IAW step 1 RNO.

**C.** Incorrect. Plausible if the candidate thinks that Functional Restoration Guidelines are performed in parallel with Emergency Operating Procedures.

**D.** Incorrect. Plausible because a transition to E-1 is available in FR-H.1 but only after feedwater is reestablished and if an RCS bleed path is open IAW step 35.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000009 | Small Break LOCA / 3 | Generic | Knowledge of EOP mitigation strategies. |

**K/A#** 2.4.6 **K/A Importance** 3.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.FR-H.1 Iss 2 Rev 1

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41 (b)10

**Objective:**

4. The plant is at 100% power.

- A Large Break LOCA occurs
- EOP Procedure E-1, Loss Of Reactor Or Secondary Coolant has been entered
- Step 15 of E-1, Check If Diesel Generators Should Be Stopped is being performed

What action will be taken for the Emergency Diesel Generators (EDGs) and what is the bases for this action?

The crew will stop the EDGs and enable ___(1)___.

The EDGs are secured because ___(2)___.

A. 1) a Manual start ONLY
2) the EDGs should not be run extensively unless they are carrying load

B. 1) a Manual start ONLY
2) the load sequencer must be allowed to reset prior to reclosing the output breaker

C. 1) both a Manual or Automatic start
2) the EDGs should not be run extensively unless they are carrying load

D. 1) both a Manual or Automatic start
2) the load sequencer must be allowed to reset prior to reclosing the output breaker

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**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. EDGs are secured and then set up for a Manual or Automatic start. The bases is correct, unloaded diesels are to be secured per manufacturer direction.

**B.** Incorrect. EDGs are secured and then set up for a Manual or Automatic start. The bases is not correct for unloaded diesels, it is a valid concern for the EDG sequencers. The load sequencer uses electromechanical timers that do not reset instantly but must time-out their cycle.

**C.** Correct. EDGs are secured and then set up for a Manual or Automatic start. unloaded EDGs are not to be run for an extended period of time per manufacturer recommendation and the EOP step bases.

**D.** Incorrect. EDGs are secured and then set up for a Manual or Automatic start, the bases is not related to the sequencer operation.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 011 | Large Break LOCA | EA1. Ability to operate and monitor the following as they apply to a Large Break LOCA: | D/Gs |

**K/A#** EA1.06 **K/A Importance** 4.2 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53B.4.E-1 iss. 3 rev.1 page 66.

**Question Source:** 1LOT14 #3

**Question Cognitive Level:** Memory or Fundamental Knowledge **10 CFR Part 55 Content:** 55.41 b(8)

**Objective:**

5. Which of the following conditions will require an immediate trip of an RCP?

A. #1 seal leakoff flow is >6 gpm.

B. The seal return line has isolated.

C. CCP to the thermal barrier heat exchanger has been isolated >1 min.

D. The seal leakoff line relief has lifted.

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. AOP 2.6.8 LHP identifies seal leakoff flow >6 gpm as an immediate RCP shutdown criteria.

**B.** Incorrect. Plausible if they feel that this will stop flow through the seal and cause parameters to meet immediate RCP shutdown criteria. This is not the case due to 2CHS-RV382A opening and relieving to the PRT when 2CHS-MOV381 goes closed.

**C.** Incorrect. Plausible to think a loss of thermal barrier flow could be an immediate RCP shutdown criterion, but the actual criteria is Loss of both the thermal barrier and Seal Injection flows for ≥4 minutes.

**D.** Incorrect. Plausible to think that if 2CHS-RV382A opens that seal leakoff flow would increase, but the leakoff flow does not change due to leakoff returning to the PRT or the VCT.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000015 | Reactor Coolant Pump Malfunctions / 4 | AK2. Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: | RCP seals |

**K/A#** AK2.07 **K/A Importance** 2.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.6.8 Rev. 13

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41 (b)10

**Objective:**

6. Given the following conditions:

- The reactor was shutdown from an extended full power run at 1400 THREE (3) days ago
- RCS temperature is 140°F
- RCS pressure is 200 psig
- PRZR level is 22%
- It is currently 1700 hrs, and a COMPLETE Loss of RHR occurred
- All RCS loop isolation valves are closed

IAW AOP 2.10.1, Loss of Residual Heat Removal Capability, to the nearest time, when will the RCS be at saturation?

**(References provided)**

A. 1715 hrs

B. 1740 hrs

C. 1800 hrs

D. 1935 hrs

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**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Plausible if the applicant uses Figure 1B and uses the 140 °F temperature line. This would be incorrect because figures 1A-C assume the RCS is at atmospheric pressure, when the stem states that the RCS is at 200 psig.

**B.** Incorrect. Plausible if the applicant uses Figure 2C, determining heatup rate to be 6.2 °F/min. The Tsat for 215 psia is 387.8 °F (387.8-140)/6.2 = 39.97 minutes. Also plausible if the applicant uses Figure 1A, and 45 minutes is the intersection of time to boiling from 140 °F.

**C.** Correct. The applicant correctly uses Att. 3 and uses Figure 2B, determining the heatup rate to be 4.3 °F/min. (387.8 -140)/4.3 = 57.63 minutes.

**D.** Incorrect. Plausible if the applicant uses the incorrect Figure 2A, determining heatup rate to be 1.6 °F/min. (387.8 -140)/1.6 = 154.88 minutes.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 025 | Loss of Residual Heat Removal System (RHRS) | AA2. Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: | Limitations on LPI flow and temperature rates of change |

**K/A#** AA2.05 **K/A Importance** 3.1 **Exam Level** RO

**References provided to Candidate** Steam tables AOP 2.10.1 Figures 1A, B, C; 2A, B, C, and Att. 3

**Technical References:** AOP 2.10.1 rev 12

**Question Source:** New

**Question Cognitive Level:** Comprehension or Analysis

**10 CFR Part 55 Content:** 55.41 b(14)

**Objective:**

7. Which of the following results in a Technical Specification action with an **Immediate** Completion Time?

A. Loss of both trains of CCP in Mode 3.

B. RCS pressure boundary leakage exceeds the RCS operational leakage limit in Mode 1.

C. Both Motor Driven Auxiliary Feedwater Pumps inoperable in Mode 3.

D. Annunciator A4-9D, Rod Control Bank 'D' Low-low annunciates in Mode 1.

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. TS 3.7.7 Condition C states two CCP trains inoperable, initiate action to restore one train of CCW to OPERABLE status immediately.

**B.** Incorrect. TS 3.4.13 states in MODES 1, 2, 3, & 4, NO pressure boundary leakage is permitted, therefore cond. B is met which requires being in MODE 3 within 6 hours and Mode 5 within 36 hours. Plausible because it sounds undesirable for any continued plant operations

**C.** Incorrect. TS 3.7.5 states In MODES 1, 2, and 3 that three trains of AFW must be operable. Plausible distractor because this is a 6 hr completion time under condition D. If three trains were inoperable in mode 3 (Cond. E), the completion time would be immediately.

**D.** Incorrect. A4-9D alarms when Control Bank 'D' is below Low-low RIL which is entry condition for TS 3.1.6. Verify SDM is within the limits specified in the COLR within 1 hour.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000026 | Loss of Component Cooling Water / 8 | Generic | Knowledge of conditions and limitations in the facility license |

**K/A#** 2.2.38 **K/A Importance** 2.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** Tech Spec 3.7.7

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41 (b)8

**Objective:**

8. The plant is operating at 100% power when one main feed pump trips. Manual reactor trip FAILS.

   After verifying the reactor will not trip, in order, what actions should be taken next?

A. Insert control rods, manually trip the turbine, and ensure steamlines isolated.

B. Verify AFW status, insert control rods, and manually trip the turbine.

C. Manually trip the turbine, ensure steamlines isolated, and insert control rods.

D. Manually trip the turbine, insert control rods, and verify AFW status.

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**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Ensuring steamlines isolated is not an immediate operator action of this procedure.
**B.** Incorrect. Verifying AFW status is the third option.
**C.** Incorrect. Ensuring steamlines isolated is not an immediate operator action.
**D.** Correct. This is the correct order of steps IAW FR-S.1

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 029 | ATWS | Generic | Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. |

**K/A#** 2.4.49 **K/A Importance** 4.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.FR-S.1 Iss 2 rev 0

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental Knowledge **10 CFR Part 55 Content:** 55.41 b(10)

**Objective:**

9. The plant is operating at 100% power.

1) Which of the following Radiation monitors will give the earliest indication of a SG tube leak?

2) At what leakrate will this Radiation monitor initially alarm?

A. 1) Air Ejector monitor, 2ARC-RQ100
2) ALERT at 5 gpd

B. 1) Steam Generator N16 monitor, 2MSS-RQ102A, B, C
2) ALERT at 5 gpd

C. 1) Air Ejector monitor, 2ARC-RQ100
2) ALERT at 30 gpd.

D. 1) Steam Generator N16 monitor, 2MSS-RQ102A, B, C
2) ALERT at 30 gpd

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Plausible because both AE and N16 detectors are monitored and the candidate may think the AE detector Alert setpoint is set at 5GPD instead of the actual setpoint of 30GPD.

**B.** Correct. SG N16 monitor will alarm the Digital Radiation Monitoring System (DRMS) at Alert level of 5GPD.

**C.** Incorrect. Plausible because 30 GPD is the Alert level for the AE discharge, but it is not the earliest indication of a SG tube leak.

**D.** Incorrect. Plausible because 30GPD is the High alarm for the N16, but it is not the earliest indication of a SG tube leak.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 038 | Steam Generator Tube Rupture (SGTR) | EA2 Ability to determine or interpret the following as they apply to a SGTR: | Radiation levels (MREM/hr) |

**K/A#** EA2.04 **K/A Importance** 3.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-4.2.6.4 Rev 28 step 1
2OM-43.4.AEK rev 2

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41 (b)11

**Objective:**

10. A plant shutdown is in progress for a refueling outage with the following conditions:

- Pressurizer pressure is 1700 psig
- Steam Line Pressure is 700 psig

**Subsequently:**

- A large steam line break occurs upstream of the "B" MSIV and outside containment.
- "B" SG completely depressurizes.

No operator action has occurred.

Based on these conditions:

Safety Injection ______(1)______ automatically occur.

Main Steam Isolation automatically occurred due to ______(2)______ **ONLY**.

A. 1) did NOT
   2) HIGH negative rate of "B" steam line pressure

B. 1) did NOT
   2) LOW "B" steam line pressure

C. 1) did
   2) HIGH negative rate of "B" steam line pressure

D. 1) did
   2) LOW "B" steam line pressure.

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. Since below P-11 (2000 psig), SI is blocked to prevent automatic actuation of the low pressure setpoint of 500 psig. Steam Lines will isolate on high negative rate on "B" steam line pressure only. The low steam line pressure is blocked at P-11 in order to prevent a SLI at the low pressure setpoint of 500 psig. However, when the low pressure SLI is blocked, a SLI can still occur due to a high rate of decrease (100 psig/sec with 50 second time constant).

**B.** Incorrect. First Part is correct. Second part is plausible because the applicant may not realize that this input would have been blocked.

**C.** Incorrect. First part is plausible if the applicant realizes that S/G pressure has previously decreased below the SLI setpoint but does not realize that pressurizer low pressure SI is also below the setpoint. Second part is correct.

**D.** Incorrect. See first part of distractor "C". Second part see second part of distractor "B."

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 040 | Steam Line Rupture | AK2. Knowledge of the interrelations between the Steam Line Rupture and the following: | Sensors and detectors |

**K/A#** AK2.02 **K/A Importance** 2.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2SQS21.1 rev. 23 PPNT slide 43<br>UFSAR logic Fig 7.3-12 Rev 12<br>2OM-1.5.B.2 Iss 4 rev 0

**Question Source:** Bank - 2013 Catawba Q10

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(4)

**Objective:** 2SQS21.1 rev 23 OBJ 11

11. Which of the following describes the capacity of the station batteries during a station blackout?

A. 2 hours during worst case loading (no ECA-0.0 load shed).

B. 2 hours assuming ECA-0.0 load shedding.

C. 8 hours during worst case loading (no ECA-0.0 load shed).

D. 8 hours assuming ECA-0.0 load shedding.

**Answer: A**

**Explanation/Justification:**

**A.** Correct. A loss of all AC will cause a loss of battery chargers to occur, therefore the batteries will commence supplying DC loads and commence discharging. The batteries are designed to supply normal and emergency DC loads for 2 Hours.

**B.** Incorrect. The batteries are designed to supply normal and emergency DC loads for 2 hours without load shedding the DC loads. ECA-0.0 ELAP will actually extend the battery DC power to select loads if needed.

**C.** Incorrect. 8 hours is plausible if the candidate thinks that each of the 4 batteries with a 2 hour capacity is combined for a total of 8 hours.

**D.** Incorrect. 8 hours is plausible if the candidate thinks that each of the 4 batteries with a 2 hour capacity is combined for a total of 8 hours.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000055 | Station Blackout / 6 | EK3: Knowledge of the reasons for the following responses as they apply to the Station Blackout | Length of time for which battery capacity is designed |

**K/A#** EK3.01 **K/A Importance** 2.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-39.1.B, Iss. 4, Rev. 0, pg 2

**Question Source:** BVPS Bank

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** (CFR 41.5 / 41.10 / 45.6 / 45.13)

**Objective:**

12. The plant is operating at 100% power with all systems in NSA, and an auto makeup to the VCT is in progress.

A plant trip and loss of offsite power occurs. EDGs start and load normally.

How does this affect the auto makeup?

A. Auto makeup will automatically resume once the Emergency busses load.

B. Auto makeup does not resume due to Boric Acid and Primary Grade Water flow deviation.

C. The makeup system trips to manual.

D. Auto makeup is available after the Boric Acid and Primary Grade Water volume setpoints are reentered.

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**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. The Blender Control Switch must be stopped, then taken back to start.

**B.** Correct. On the loss of power, the Boric Acid pumps powered from E13 & E14 will be de-energized until the EDG sequences the loads and repowers the pumps. When the pump is stopped, it will cause a Boric Acid flow setpoint deviation which will require the Boric Acid Makeup Blender Control Switch to be placed in STOP, then back to START to commence the auto makeup.

**C.** Incorrect. The Mode selector would have been in AUTO since an auto makeup was in progress, and the blender control does not automatically realign to manual mode of operation.

**D.** Incorrect. Auto make up is available after the Blender Control Switch is manipulated, but the volume totalizers have no input into the auto makeup. The volume setpoints are used during manual makeups.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 056 | Loss of Offsite Power | AA2. Ability to determine and interpret the following as they apply to the Loss of Offsite Power: | CVCS makeup |

**K/A#** AA2.75 **K/A Importance** 3.0 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-7.4.AAA Rev 9 pg 3; 2OM-7.1.D Iss 4 Rev 3 pg 27; 2OM-7.1.C Rev 9 pg 6

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(7/8)

**Objective:**

13. The plant is operating at 60% power when the following alarm is received:

    - A8-9A, 125V DC BUS 2-1 TROUBLE

    Several minutes after the alarm is received:

    - The plant continues to operate at 60% power.
    - DC BUS 2-1 Voltage indicates approximately 124 VDC and is slowly DROPPING.
    - Battery Charger 2-1 AC Input breaker, BAT-CHG2-1-B301, has been verified closed and 480V MCC*2-E05 is energized.
    - No operator actions have occurred.

    For the given indications, which of the following describes DC BUS 2-1 status?

A. Battery Charger 2-1 has failed. Station Battery 2-1 is supplying DC BUS 2-1.

B. Station Battery 2-1 has failed. Battery Charger 2-1 is supplying DC BUS 2-1.

C. Station Battery 2-1 and Battery Charger 2-1 have failed. DC BUS 2-1 is degraded.

D. Station Battery 2-1 and Battery Charger 2-1 are operating normally. A large ground has occurred on DC BUS 2-1.

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. When a battery charger is lost, the station battery will automatically supply power to the loads on the effected bus, and DC BUS 2-1 voltage will lower to the battery voltage.

**B.** Incorrect. Plausible since DC Bus 2-1 is indicating DC voltage, but if Battery Charger 2-1 is supplying DC BUS 2-1, voltage would indicate between 130-140 VDC. This is the DC output voltage set by 2OM-39.4.A when placing Battery Charger 2-1 in service.

**C.** Incorrect. Plausible if the candidate thought the spare battery charger (BAT-CHG2-7) picked up the DC loads, but the spare chargers have to be placed in service manually.

**D.** Incorrect. Plausible if the candidate thinks this is normal voltage indication for DC Bus 2-1, and thinks A8-9A alarms for DC grounds. A DC Ground would alarm A8-10, 125V DC Bus 2-1 Ground.

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 058 | Loss of DC Power | AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: | Battery charger equipment and instrumentation |

**K/A#** AK1.01 **K/A Importance** 2.8 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-39.4.AAD Rev 7 pg 4-5<br>2OM-39.4.A Rev 14 pg 6

**Question Source:** Bank, mod distractor "d"

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41 (b)7

**Objective:**

14. The plant is operating at 100% power.

- AOP-2.30.1, Service Water/Main Intake Structure Loss AOP has been entered due to a loss of the 'B' SWS header.
- A Steam Generator Blowdown Test Tank discharge is in progress.

Step 4 of AOP-2.30.1 instructs the crew to secure any liquid waste discharge IF service water header pressure cannot be restored above 34 psig.

Under these conditions, WHY is the liquid waste discharge secured?

A. The required liquid waste discharge dilution water flow cannot be assured.

B. The liquid waste discharge radiation monitor will be inoperable.

C. Liquid waste discharge flow control will be unavailable.

D. Steam generator cleanup ion exchanger temperature control cannot be assured.

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**Answer: A**

**Explanation/Justification:**

A. Correct. IAW OM Fig. 31-1 and 25-4 Dilution water for liquid waste discharges is provided by the service water system.

B. Incorrect. The liquid waste discharge radiation monitor is not cooled by river water and will remain operable during loss of service water.

C. Incorrect. Air will still be available to the flow control valve since domestic water is manually aligned to cool the air compressors. Therefore, there will be no loss of air.

D. Incorrect. Steam Generator Blowdown Test Tank ion exchangers are used to clean-up the water before they are prepared for discharge NOT during the discharge. Additionally, the need to cool evaporator distillate at Unit 2 has been removed since the evaporators have been retired in place.

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| **Sys.#** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 062 | Loss of Nuclear Service Water | AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: | Guidance actions contained in EOP for Loss of nuclear service water |

**K/A#** AK3.03 **K/A Importance** 4.0 **Exam Level** RO

**References provided to Candidate** None

**Technical References:** 2OM-53C.4.2.30.1 Rev 9<br>2OM-25.4.L Rev 34<br>U2 RM-0425-004 Rev 7, U2 RM-0431-001 Rev 7

**Question Source:** Bank - 2LOT6 Q14

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(12)

**Objective:**

15. What effect does a loss of instrument air to containment have on the RCPs?

A. Loss of stator and bearing cooling only.

B. Loss of seal injection flow only.

C. Loss of thermal barrier cooling flow only.

D. Loss of all stator, bearing, and thermal barrier cooling only.

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**Answer: C**

**Explanation/Justification:**

- **A.** Incorrect. 2CCP-MOV103A, B, C supplies the upper and lower bearing oil coolers, and the stator air cooler. The supply to these RCP components is not affected by a loss of air.
- **B.** Incorrect. AOP 2.34.1 pg 20 identifies 2CHS-HCV186 as failing open on a loss of instrument air.
- **C.** Correct. AOP 2.34.1 pg 31 identifies 2CCP-AOV107A, B, C as failing closed on a loss of instrument air.
- **D.** Incorrect. Plausible if the candidate is not familiar the failure positions of the various supplies. As stated above stator and bearing oil cooler supplies do not change position, therefore there is no loss due to the instrument air failure. Thermal barrier cooling flow does fail closed.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000065 | Loss of Instrument Air / 8 | AK3: Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: | Knowing effects on plant operation of isolating certain equipment from instrument air |

**K/A#** AK3.03 **K/A Importance** 2.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C4.2.34.1 Rev 20 pg 31
U2 RM-04-003 Rev 9
2SQS-15.1, Rev. 9, Iss. 2 pg 20-21

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41 (b)4

**Objective:**

16. Initial Main Generator/345KV Switchyard Conditions:

- Real Load: 960 MWe
- Reactive Load: 240 MVARS out
- Switchyard Voltage: 345KV
- Frequency: 60.0 Hz

The grid becomes unstable, and the BOP reports the following parameters:

Switchyard Voltage has dropped to 330 KV.
Frequency has remained at 60.0 Hz.

Assuming the reactor does NOT trip, how will Main Generator Amps respond to this event; and which limit (MWe or MVAR) will most likely be exceeded?

| | Generator Amps | Limit Most Likely to be Exceeded |
|---|---|---|
| A. | Increase | MVARs |
| B. | Decrease | MVARs |
| C. | Increase | MWe |
| D. | Decrease | MWe |

**Answer: A**

**Explanation/Justification:** Since Frequency has not changed, turbine speed remains constant, since it is tied to the grid. The turbine control valves will remain steady, maintaining real load constant ("C" and "D" wrong). A generator's MVAR load increases when generator terminal voltage increases above grid voltage. This can be caused either by raising excitation voltage, or by decreasing grid voltage, which has happened in the transient described in the stem of the question. "A" is correct, and "B" wrong, since raising MVARs increases generator amps. "B", "C", and "D" are plausible, since a transient is in progress.

**A.** Correct. See answer explanation/justification.
**B.** Incorrect. See answer explanation/justification.
**C.** Incorrect. See answer explanation/justification.
**D.** Incorrect. See answer explanation/justification.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 077 | Generator Voltage and Electric Grid Disturbances | AK1. Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances: | Definition of terms: volts, watts, amps, VARs, power factor |

**K/A#** AK1.01 **K/A Importance** 3.3 **Exam Level** RO
**References provided to Candidate** None **Technical References:** GO-GPF-C5 rev. 2 page 126, 132-135
**Question Source:** Bank - 2011 Millstone 3 Q18
**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(4)
**Objective:**

17. A safety injection has occurred on low pressurizer pressure. Plant conditions are:

- RCS pressure is 1850 psig, trending down
- Reactor and turbine are tripped
- Pressurizer level decreasing rapidly
- All SG pressures are 900 psig, trending down slowly
- Containment pressure - stable
- Containment rad levels - no alarms
- Containment sump level - stable
- A11-10D, AUXILIARY BUILDING SUMP LEVEL HIGH is in alarm
- Auxiliary Building rad level - Hi alarm

Which of the following events are in progress?

A. LOCA inside containment

B. LOCA outside containment

C. Multiple Faulted SGs inside containment

D. Multiple Faulted SGs outside containment

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Cnmt pressure, sump levels, or radiation levels are not rising. No indication of LOCA inside cnmt.

**B.** Correct. RCS pressure and przr level are lowering, aux building rad levels and sump level are rising are indications of a LOCA outside of cnmt.

**C.** Incorrect. SG pressures are slowly lowering as expected due to SI, Cnmt pressure or sump levels are not rising. No indications of faulted SG inside cnmt.

**D.** Incorrect. SG pressures are slowly lowering as expected due to SI.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| WE04 | LOCA Outside Containment / 3 | EA1. Ability to operate and / or monitor the following as they apply to the (LOCA Outside Containment) | Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. |

**K/A#** EA1.1 **K/A Importance** 4.0 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.E-0 Iss. 3 Rev. 0 step 20 & Bkgd.

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41 (b)5

**Objective:**

18. The crew is performing ECA-1.1, Loss of Emergency Coolant Recirculation.

    1) Which of the following describes the reason for depressurizing the RCS IAW ECA-1.1?

    2) After commencing the RCS cooldown in ECA-1.1, which of the following sequences will be used to depressurize the RCS?

A. 1) To minimize RCS leakage.
   2) Reduce/terminate HHSI flow, then depressurize the RCS while maintaining minimum RCS subcooling.

B. 1) To determine if recirculation spray pumps are cavitating due to CNMT sump blockage.
   2) Reduce/terminate HHSI flow, then depressurize the RCS while maintaining minimum RCS subcooling.

C. 1) To minimize RCS leakage.
   2) Depressurize the RCS while maintaining maximum RCS subcooling, then stabilize RCS temperature while attempting to restore makeup sources.

D. 1) To determine if recirculation spray pumps are cavitating due to CNMT sump blockage.
   2) Depressurize the RCS while maintaining maximum RCS subcooling, then stabilize RCS temperature while attempting to restore makeup sources.

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. The RCS is depressurized to minimize RCS leakage. This is accomplished by commencing plant cooldown, reducing or terminating the HHSI flow to help minimize the RWST outflow, then the RCS is depressurized to minimize subcooling and reduce flow from the LOCA. (Major action step 4)

**B.** Incorrect. The first step of ECA-1.1 checks for RSS pump cavitation due to containment sump blockage. Plausible distractor because reducing RCS pressure will raise RSS pump flow, which could cause cavitation indications to appear, but this is not the reason for depressurization which comes much later in ECA-1.1. Second part is correct.

**C.** Incorrect. The RCS is depressurized to minimize RCS leakage. Second part is plausible because the candidate must understand that depressurization of the RCS is to minimize, not maximize subcooling so RCS pressure can be decreased to the lowest pressure possible without losing subcooling. After depressurization, plant cooldown will still occur by procedure and attempting to restore makeup sources will continue.

**D.** Incorrect. The first step of ECA-1.1 checks for RSS pump cavitation due to containment sump blockage. Plausible distractor because reducing RCS pressure will raise RSS pump flow, which could cause cavitation indications to appear, but this is not the reason for depressurization which comes much later in ECA-1.1. Second part is plausible because the candidate must understand that depressurization of the RCS is to minimize, not maximize subcooling so RCS pressure can be decreased to the lowest pressure possible without losing subcooling. After depressurization, plant cooldown will still occur by procedure and attempting to restore makeup sources will continue.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| W/E11 | Loss of Emergency Coolant Recirc. / 4 | EK2. Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: | Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility |

**K/A#** EK2.2 **K/A Importance** 3.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.ECA-1.1 Iss 3 Rev 0 pg. 1
2OM-53B.4.ECA-1.1 Iss. 3 Rev. 0 pgs. 3 & 50

**Question Source:** Bank - 1LOT16 Q17

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(10)

**Objective:** 3SQS-53.3 Rev 5 Obj. 3 State from memory the basis and sequence for the Major Action Steps of each EOP procedure, IAW BVPS-EOP Executive Volume.

19. A rapid load reduction at 5% per minute has been initiated from 100% power. Emergency boration is in progress as directed by AOP 2.51.1, Unplanned Power Reduction.

    Which of the following conditions requires a reactor trip?

A. Rods fail to insert in AUTO at +1.5°F temperature deviation.

B. Temperature deviation exceeds +/- 5°F and cannot be corrected.

C. Temperature deviation exceeds +/- 10°F and cannot be corrected.

D. Tave decreases to 543°F when load reduction is complete.

---

**Answer: C**

**Explanation/Justification:**

A. Incorrect. Plausible because candidate may think an auto rod malfunction warrants a Rx trip, but the action required is to take manual control of the rods.

B. Incorrect. Plausible because this is the desired control band during the unplanned power reduction, but if +/-5F is not being maintained, then the operator will take manual control of rods and maintain +/-5F and AFD.

C. Correct. Step 4 of AOP 2.51.1 states if Tavg cannot be maintained +/-10F, then trip the Rx, this is also reinforced with the Transient Response Guidelines of BVPS.

D. Incorrect. Plausible if the candidate thinks 543F is the minimum temp for criticality, but the actual temp is 541F which does require a rx trip per BV operating procedures.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 024 | Emergency Boration | AA1. Ability to operate and / or monitor the following as they apply to Emergency Boration: | T-ave. meters |

**K/A#** AA1.16 **K/A Importance** 3.3 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.51.1 Rev 18 pg 3; BVBP-OPS-0024 Rev 10 pg 16

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** CFR 41.7 / 45.5 / 45.6

**Objective:**

20. The plant has been at 100% power for the last 50 days. Pressurizer level is in the normal programmed band.

- 2RCS-LT460 and LT461 are in service (PRZR LEVEL CONTROL CHANNEL SELECTOR in position II & III)
- 2RCS-LT461 fails HIGH

1) If **NO** operator action is taken, what effect, if any, is on PRZR level?

2) Two seconds after 2RCS-LT461 failing high, A4-1B, "PRESSURIZER CONTROL LEVEL HIGH/LOW", annunciator _________ lit.

A. 1) No effect on PRZR level.
2) will NOT be

B. 1) No effect on PRZR level.
2) will be

C. 1) PRZR level decreases.
2) will NOT be

D. 1) PRZR level decreases.
2) will be

---

**Answer: C**

**Explanation/Justification:** In position II & III, LT-461 will be the controlling channel and send a false high value to the level error comparator. PZR Level Control 2RCS-LK459F compares charging flow and the level error and will send a negative value to the Charging Pumps Discharge Flow Control VLV 2CHS*FCV122. 2CHS*FCV122 will start to close causing PZR level to decrease. LT-461 does not feed the high level bistable in position II & III, but does in position I & III. The A4-1B, "Pressurizer Control Level High/Low" annunciator does come in on low level due to LT-460, but WILL NOT come in right away as it takes time for level to drop to <14%.
All other distractors are plausible because it can be correct depending on the position of the switch. To answer this question, the applicant requires knowledge of the circuit.

**A.** Incorrect. See answer explanation/justification.
**B.** Incorrect. See answer explanation/justification.
**C.** Correct. See answer explanation/justification.
**D.** Incorrect. See answer explanation/justification.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 028 | Pressurizer (PZR) Level Control Malfunction | Generic | Ability to verify that the alarms are consistent with the plant conditions. |

**K/A#** 2.4.46 **K/A Importance** 4.2 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-6.4.IF rev. 13 page 12

**Question Source:** New

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(7)

**Objective:**

21. With the plant at 100% power, NSA, If the N31 Source Range Detector High Voltage (HV) Manual Switch on the N31 drawer is placed in the HV ON position, the N31 detector high voltage will:

A. HV Turns ON; no reactor trip will occur.

B. HV Remains OFF due to the P-10 interlock; no reactor trip will occur.

C. HV Remains OFF due to the P-6 interlock; no reactor trip will occur.

D. HV Turns ON; resulting in a reactor trip.

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. When the HV Manual On/Off switch is placed in the HV ON position, N31 will be energized, and no rx trip will occur because the source range high rx trip is blocked manually when above P-6, or automatically above P-10 and the stem states that the plant is at 100% power.

**B.** Incorrect. Plausible distractor of HV remain off because the P-10 interlock is input to the N31 being de-energized but only if the HV Manual On/Off switch is in the NORMAL position. Second part, no rx trip is correct.

**C.** Incorrect. Plausible distractor of HV remain off because the P-6 interlock is input to the N31 being de-energized but only if the SR trip is manually blocked when above P-6. Second part, no rx trip is correct.

**D.** Incorrect. First part is correct, N31 will be energized. Second part in incorrect but plausible if the candidate doesn't know that at 100% power the source range high rx trip is blocked manually when above P-6, or automatically above P-10.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000032 | Loss of Source Range Nuclear Instrumentation | AK2. Knowledge of the interrelations between the Loss of Source Range Nuclear Instrumentation and the following: | Power supplies, including proper switch positions |

**K/A#** AK2.01 **K/A Importance** 2.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-2.1.C Rev 2 pg. 9
UFSAR Fig. 7.3-8 Rev 14
3SQS-2.1 PPNT Rev 9 slide 51

**Question Source:** Bank Modified

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41(b)7

**Objective:**

22. The plant was operating at 100% power when a Tube Leak occurred on 'B' SG. The plant is currently shutdown in Mode 3, and Tavg is 535°F.

(1) What is the maximum cooldown rate, and preferred method and, (2) why?

A. 1) not to exceed 25°F/hr to atmosphere
2) The procedure assumes no condenser available.

B. 1) not to exceed 90°F/hr to atmosphere
2) The procedure assumes no condenser available.

C. 1) not to exceed 25°F/hr to condenser
2) To minimize offsite release.

D. 1) not to exceed 90°F/hr to condenser
2) To minimize offsite release.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Incorrect answer with incorrect explanation.
**B.** Incorrect. Correct rate (to wrong location) with incorrect reason.
**C.** Incorrect. Incorrect rate (to correct location) with correct reason.
**D.** Correct. AOP 2.6.4 step 15 & 26 both require cooldown at ≤ 90F/Hr (Plant administrative cooldown limit) using the Condenser Steam Dumps is preferred to minimize radiological releases and conserve feedwater supply.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 037 | Steam Generator (S/G) Tube Leak | AK3. Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak: | Normal operating precautions to preclude or minimize SGTR |

**K/A#** AK3.06 **K/A Importance** 3.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.6.4 Rev 28
2OM-53B.4.E-3 Iss 3 Rev 0 pg 69

**Question Source:** New

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** (41.5,41.10 / 45.6 / 45.13)

**Objective:**

23. Given the following conditions:

- A plant power ascension is in progress after an outage.
- The plant is currently at 375 MW.
- The crew notices a slow trend down in MW and diagnoses a lowering condenser vacuum.
- A6-5G, CONDENSER VACUUM LOW/LOW-LOW is in alarm.
- 2CNM-PR103, Condenser Vacuum Side A & Side B recorder indicate a slowly lowering trend which has stabilized at 20.8" Hg-Vac.

What is the proper response in accordance with plant procedures?

A. Trip the reactor, then enter E-0, Reactor Trip or Safety Injection.

B. Trip the turbine, then enter AOP 2.26.1, Turbine and Generator Trip.

C. Restore vacuum to greater than 24.1" Hg-vac within 5 minutes. If vacuum cannot be restored within 5 minutes, trip the turbine and enter AOP 2.26.1, Turbine and Generator Trip.

D. Reduce turbine load to restore vacuum. If vacuum cannot be restored to >24.1" Hg-Vac, then trip the reactor and enter E-0, Reactor Trip or Safety Injection.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. The plant is 375MW (<P9), therefore a reactor trip is not required. Plausible because a turbine trip is required, but conditions are not warranted for Rx trip.

**B.** Correct. With the plant <P9 and condenser vacuum below the turbine trip setpoint of 21.1"Hg-Vac, the turbine must be tripped because the automatic trip has failed to occur. A reactor trip is not required and AOP 2.26.1 will give the guidance on reducing plant power further.

**C.** Incorrect. Plausible distractor since the plant is <P9, and vacuum is at a low value. The candidate could think that vacuum could be high enough to regain control using AOP 2.26.2 Loss of condenser Vacuum. This is the identified action of the AOP step 3.

**D.** Incorrect. Plausible distractor because these are the actions required by AOP 2.26.2 Loss of condenser Vacuum step 2 if power is >P9.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000051 | Loss of Condenser Vacuum / 4 | Generic | Knowledge of abnormal condition procedures. |

**K/A#** 2.4.11 **K/A Importance** 4.0 **Exam Level** RO

**References provided to Candidate** None

**Technical References:**
2OM-53C.4.2.26.2 rev. 1
2OM-26.2.B rev 21 pg 9
2OM-26.4.AAK rev 15 pg 6
2OM-26.4.AAB rev 2 pg 3
NOP-OP-1002 Rev 12 pg 62

**Question Source:** New

**Question Cognitive Level:** Memory

**10 CFR Part 55 Content:** 55.41 (b)10

**Objective:**

24. Given the following:

- The plant has experienced a Steam Generator tube leak with high RCS activity from a fuel failure
- The Auxiliary Building has rising radiation indications
- The Control Room Radiation Monitors, 2RMC-RQ201 and RQ202 both are in ALERT
- The appropriate Alarm Response Procedure was entered
- The Control Room is determined to be habitable

Which of the choices below completes the following statement?

In accordance with the Control Room Area Radiation Monitor ARP, the crew will depress the Control Room Emergency Air Supply ____(1)____ pushbutton because ____(2)____.

A. 1) Actuation
2) this trips 2HVC-FN241A,B to isolate outside air from the control room

B. 1) Actuation
2) this aligns filtered air to the Control Room envelope to maintain an acceptable environment for operating personnel

C. 1) Isolation
2) this trips 2HVC-FN241A,B to isolate outside air from the control room

D. 1) Isolation
2) this aligns filtered air to the Control Room envelope to maintain an acceptable environment for operating personnel

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. First part correct. Second part plausible because the CR Emergency Supply fans (2HVC-FN241A,B) are NOT used in toxic gas.

**B.** Correct. The crew will depress the Control Room Emergency Ventilation System Actuation pushbutton for high radiation to start 2HVC-FN241A,B to maintain an acceptable environment.

**C.** Incorrect. Isolation is plausible because this is what the crew would do for a toxic gas environment. (AOP-1/2.44A.1). The second part is plausible because that does happen when the isolation pushbutton is depressed.

**D.** Incorrect. See distractor "C" for reason for first part. Second part is plausible if the applicant gets the functions of isolation and actuation confused.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 061 | Area Radiation Monitoring (ARM) System Alarms | AK3. Knowledge of the reasons for the following responses as they apply to the Area Radiation Monitoring (ARM) System Alarms: | Guidance contained in alarm response for ARM system |

**K/A#** AK3.02 **K/A Importance** 3.4 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-43.4.AAU/AAV rev. 3 step 1.d
2SQS 44A.1 rev. 6 PPNT slide 44, 45

**Question Source:** Modified - 2014 North Anna Q24

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(11)

**Objective:**

25. Given the following conditions:

- A LOCA in containment has occurred.
- RCS pressure is 1000 psig and slowly lowering.
- All HHSI and LHSI pumps have failed; all other components have actuated properly.
- CNMT pressure is 8 psig and slowly rising.
- The crew is has entered FR-C.1, Response to Inadequate Core Cooling.
- The Unit Supervisor directs intact SG depressurization IAW step 13.

How will the SG depressurization be accomplished?

A. Dump steam using condenser steam dumps at a rate not to exceed 100°F/hour.

B. Dump steam using condenser steam dumps at maximum rate.

C. Dump steam using SG atmospheric steam dumps at a rate not to exceed 100°F/hour.

D. Dump steam using SG atmospheric steam dumps at maximum rate.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. LOCA will cause cnmt pressure to rise above 7 psig and cause a MSLI which makes the condenser steam dumps unavailable. 100F/hr is plausible as it is used throughout the EOP network.

**B.** Incorrect. LOCA will cause cnmt pressure to rise above 7 psig and cause a MSLI which makes the condenser steam dumps unavailable. Max rate is correct per FR-C.1 step 13.

**C.** Incorrect. SG atmospheric steam dumps is correct. 100F/hr is plausible as it is used throughout the EOP network.

**D.** Correct. The candidate will have to recognize that a LOCA will cause cnmt pressure to rise above 7 psig and cause a MSLI which makes the condenser steam dumps unavailable. The crew will have to use the SG atmospheric steam dumps. Maximum rate is correct.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000074 | Inadequate Core Cooling / 4 | EA2 Ability to determine or interpret the following as they apply to a Inadequate Core Cooling: | Availability of turbine bypass valves for cooldown |

**K/A#** EA2.03 **K/A Importance** 3.8 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.FR-C.1 Iss 3 Rev 0 step 13 pg1 & 7

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41 (b)4

**Objective:**

26. The plant is in Mode 3, starting up after a refuel outage. A loose part left in the reactor vessel has caused damage to a fuel assembly.

What would indicate that the fuel is damaged, and which AOP do you enter?

A. High radiation as indicated by 2CHS-RQ101A, Letdown Radiation Monitor, and enter AOP 2.6.6, High Reactor Coolant Activity.

B. High radiation as indicated by 2RMR-RQI303, Containment Airborne Radiation Monitor, and enter AOP 2.6.6, High Reactor Coolant Activity.

C. High radiation as indicated by 2CHS-RQ101A, Letdown Radiation Monitor, and enter AOP 2.49.1, Irradiated Fuel Damage.

D. High radiation as indicated by 2RMR-RQI303, Containment Airborne Radiation Monitor, and enter AOP 2.49.1, Irradiated Fuel Damage.

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. The letdown radiation monitor is the radiation monitor that monitors for failed fuel. AOP 2.6.6 is the procedure to be entered.

**B.** Incorrect. Plausible radiation monitor if the applicant thinks that the containment airborne radiation monitor will pick up on activity inside the RCS. Correct procedure to be entered

**C.** Incorrect. Correct radiation monitor. Plausible AOP given the name Irradiated Fuel Damage.

**D.** Incorrect. Incorrect radiation monitor and incorrect AOP.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 076 | High Reactor Coolant Activity | AA1. Ability to operate and / or monitor the following as they apply to the High Reactor Coolant Activity: | Failed fuel-monitoring equipment |

**K/A#** AA1.04 **K/A Importance** 3.2 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.6.6 Rev 3<br>2OM-53C.4.2.49.1 Rev 10

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(11)

**Objective:**

27. Given the following conditions:

- A safety injection has occurred on low pressurizer pressure
- RCS pressure is 1700 psig, trending down
- The turbine has failed to trip. Main Steamlines have been isolated
- Containment radiation levels are rising
- Containment humidity is rising
- Containment sump level is rising
- All SG pressures are 800 psig, trending down slowly
- SG levels are in the wide range, trending down slowly
- SG blowdown has isolated

Which of the following events are in progress?

A. LOCA inside Containment

B. Steamline Break in Containment

C. Feedline Break in Containment

D. SGTR coincident with Steam Break in Containment

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. Cnmt pressure increases and cnmt radiation levels will rise.
**B.** Incorrect. Plausible because cnmt pressure will increase, but the radiation level will not change.
**C.** Incorrect. Plausible because cnmt pressure will increase, SG WR levels will lower, but the radiation level will not change.
**D.** Incorrect. Plausible because cnmt pressure will increase, and radiation levels would increase, but all SG levels are lowering slowly, and all SG pressures are 800 psig and lowering. Based on these conditions there is not a tube rupture in progress.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| WE16 | High Containment Radiation /9 | EK1. Knowledge of the operational implications of the following concepts as they apply to the (High Containment Radiation) | Annunciators and conditions indicating signals, and remedial actions associated with the (High Containment Radiation). |

**K/A#** EK1.3 **K/A Importance** 3.0 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.E-0 Iss 3 Rev 0 step 16

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41 (b)11

**Objective:**

28. The Reactor Operator is in the process of a normal Reactor Coolant Pump startup IAW 2OM-6.4.A, Reactor Coolant Pump Startup. The RO is ready to place the 'C' Reactor Coolant Pump (RCP) control switch to START.

Which of the following is the expected sequence of events following placement of the RCP control switch to START?

(All components and indications below are referring to the 'C' RCP)

1) RCP starts
2) Bearing Lift Oil pump starts
3) Starting amps drop to normal running amps
4) Bearing Lift Oil Pump STOPS after a time delay
5) Bearing Lift Oil Pump remains RUNNING during RCP operation

A. 1, 2, 3, 4

B. 1, 3, 2, 5

C. 2, 4, 1, 3

D. 2, 1, 3, 4

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**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Plausible if the candidate thinks the RCP starting will cause the bearing lift oil pump to run until the RCP amperage drops to normal running amps, and then the bearing lift oil pump would stop after RCP amperage stabilizes. In actuality, the bearing lift oil pump operates for approx. 2 minutes prior to the RCP starting.

**B.** Incorrect. Plausible if the candidate thinks the RCP starts, and the bearing lift oil pump doesn't begin to run until the RCP amperage drops to normal running amps, and then the bearing lift oil pump remains running while the RCP is in operation. See the correct answer for proper sequence.

**C.** Incorrect. Plausible because the candidate may think that the bearing lift oil pump will start, and shutdown, to prelube the bearings and lift the shaft prior to RCP start. The bearing lift oil pump actually runs for approx. 50 seconds after the RCP starts.

**D.** Correct. The bearing lift oil pump will start and run for approx. 2 minutes before the 'C' RCP will start. After the RCP starts, amps will drop to normal running amps after approx. 30 seconds. The bearing lift oil pump will stop approx. 50 seconds after the RCP starts.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 003 | Reactor Coolant Pump System (RCPS) | A3 Ability to monitor automatic operation of the RCPS, including: | RCP lube oil and bearing lift pumps |

**K/A#** A3.05 **K/A Importance** 2.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-6.4.A Rev 19 sect. IV.C.11.e

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41

**Objective:** 2SQS-6.3, Rev. 12 Obj. 17 - Describe the control, protection and interlock functions for the control room components associated with the Reactor Coolant Pump and support system, including automatic functions, setpoints and changes in equipment status as applicable.

29. The Non-Regenerative Heat Exchanger Cooling Water Discharge temperature control valve, 2CCP-TCV144 has failed CLOSED.

    What effect does this have on plant operation?

A. Letdown diverts to degasifiers.

B. Bypass of letdown demineralizers.

C. Inability to use auxiliary spray due to high charging to spray line delta T.

D. Potential flashing in the letdown line.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Plausible if the candidate thinks that 2CHS-LCV115A will divert to the degasifiers on high inlet temperature to the VCT.

**B.** Correct. If TCV144 fails closed, the NRHX outlet temperature will rise. When the NRHX outlet temperature reaches 134F, the L/D Demin Bypass Temperature Divert Valve (2CHS-TCV143) will realign to the VCT position to protect the demins from high temperature.

**C.** Incorrect. Plausible if the candidate thinks the letdown temperatures will get colder, which would cause a higher delta T for aux spray and PRZR vapor temperature. This would be incorrect because delta T will actually become slightly smaller due to higher charging temperatures.

**D.** Incorrect. Plausible if the candidate thinks that letdown would isolate due to the higher temperatures, but flashing should not occur since letdown flow remains in service and backpressure is controlled by the NRHX Discharge Pressure Control valve.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 004 | Chemical and Volume Control System (CVCS) | K6 Knowledge of the effect of a loss or malfunction on the following CVCS components: | Heat exchangers and condensers |

**K/A#** K6.07 **K/A Importance** 2.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** U2 RM-0407-001A Rev 25<br>U2 RM-0415-005 Rev 10<br>2OM-7.4.AAF Rev 17 Sect. E

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41(b)8

**Objective:**

30. Given the following conditions:

- The RCS is solid
- The RCS pressure is at 300 psig and stable
- The RCS temperature is 180°F and stable
- RHR is in service and supplying letdown
- CVCS Letdown Orifices are open

If the air line separates from the actuator on the following valves, how will RCS pressure respond?

(Answer each of the following statements as **separate failures**)

1) 2CHS*PCV145, NRHX Discharge Pressure Control Valve, RCS pressure will __________.

2) 2CHS*HCV142, RHS Letdown Flow Control Valve, RCS pressure will ______________.

A. 1) increase
2) increase

B. 1) increase
2) decrease

C. 1) decrease
2) decrease

D. 1) decrease
2) increase

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. See correct answer.
**B.** Incorrect. See correct answer.
**C.** Incorrect. See correct answer.
**D.** Correct. 145 fails open which will cause pressure to lower. 142 fails shut and controls the letdown flow out of the RHR heat exchanger. Once this valve goes shut, RCS pressure will increase due to the mass imbalance in a solid system.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 005 | Residual Heat Removal System (RHRS) | K5 Knowledge of the operational implications of the following concepts as they apply the RHRS: | Plant response during "solid plant": pressure change due to the relative incompressibility of water |

**K/A#** K5.05 **K/A Importance** 2.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.34.1 Rev 20 pgs 20 & 30
RM-0407-001A Rev 25

**Question Source:** Bank Modified - 2012 North Anna Q8

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(7)

**Objective:**

31. Which of the following conditions is **NOT** in compliance with Technical Specifications?

A. Plant is in Mode 5, loops unisolated and filled.
Two SGs are at 30% Narrow Range level.
One RHR pump is in operation.
One RHR pump is tagged for maintenance.

B. Plant is in Mode 5, midloop.
Two SGs are at 30% Narrow Range level.
Both RHR pumps are operable.
Both RHR pumps were stopped for 5 minutes during equipment rotation.

C. Plant is in Mode 5, loops unisolated and filled.
Two SGs are at 30% Narrow Range level.
One RHR pump is in operation; one RHR pump is tagged for maintenance.
PRZR level is being raised.

D. Plant is in Mode 5, midloop.
Two SGs are at 30% Narrow Range level.
One RHR pump is in operation.
One RHR pump is tagged for maintenance.

---

**Answer: D**

**Explanation/Justification:**

- **A.** Incorrect. LCO 3.4.7 is met. 1 RHR loop is in operation, and at least 1 SG water level is ≥15.5% NR.
- **B.** Incorrect. LCO 3.4.8 is met. 2 RHR loops operable and 1 loop in operation. TS note allows for both RHR pumps to be removed from operation for ≤15 minutes for equipment rotation.
- **C.** Incorrect. LCO 3.4.7 is met. 1 RHR loop is in operation, and at least 1 SG water level is ≥15.5% NR.
- **D.** Correct. With the plant in Midloop condition, TS 3.4.8 is not met. TS requires 2 RHR loops operable and 1 RHR loop in operation. Only 1 RHR is in operation with the other tagged for maintenance. The steam generators (SGs) are not available as a heat sink when the loops are not filled or isolated in Midloop condition.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 005 | Residual Heat Removal | Generic | Ability to recognize system parameters that are entry-level conditions for Technical Specifications. |

**K/A#** 2.2.42 **K/A Importance** 3.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** TS 3.4.8

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41(b)10

**Objective:**

32. The plant is operating at 100% power. A spurious SI signal occurs. The reactor trip breakers (RTBs) fail to open. All functions not dependent on open reactor trip breakers operate normally.

What is the effect on Main Feedwater?

A. MFW will continue to control SG level; no FW isolation will occur because the RTBs have not opened.

B. SI will provide a signal to actuate the P-4 interlock MFRV closure on low Tave.

C. SI will initiate a full feedwater isolation and trip of main feed pumps.

D. SI will close 2FWS-HYV157A,B,& C, SG Feedwater Isolation Valves; MFW pumps will not trip because the RTBs have not opened.

---

**Answer: C**

**Explanation/Justification:**

A. Incorrect. The SI will provide a full FW isolation, therefore MFW will not control SG level, and AFW will be supplying the SGs due to the auto start on the SI signal.
**B.** Incorrect. SI provides no signal to P4. A partial FW isolation which closes the MFRVs occurs when P-4B and 2/3 Tavg (<554F) are met.
**C.** Correct. The SI will initiate a full FW isolation which will trip the MFW pumps.
**D.** Incorrect. The SI will close the SG Feedwater Isolation Valves, and it will also trip the MFW pumps. RTBs provide no input to the MFPs.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 006 | Emergency Core Cooling | K1: Knowledge of the physical connections and/or cause effect relationships between the ECCS and the following systems: | MFW System |

**K/A#** K1.07 **K/A Importance** 2.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-1.5.B.4A Iss 4 Rev 1 pg1
2OM-1.5.B.4E Iss 4 Rev 0

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41(b)7

**Objective:**

33. Which one of the following statements describes how a steam bubble is formed in the PRZR during an RCS heatup in preparation to returning the plant to service?

A. The PRZR is filled with one PORV OPEN until a level rise is observed in the PRT. The PORV is then closed, and the PRZR is heated to 250-275°F, THEN RCS Pressure is lowered to 25-30 PSIG as level is lowered. At these conditions the combination of heaters and Letdown flow being greater than Charging will cause a bubble to form in the PRZR

B. The PRZR is filled until water vents from 2RCS*600 by local observation. Then the PRZR is heated to 250-275°F, THEN RCS Pressure is lowered to 25-30 PSIG as level is lowered. At these conditions the combination of heaters and Letdown flow being greater than Charging will cause a bubble to form in the PRZR

C. The PRZR is filled to ≥ 80% with 2RCS*600 open. Then 2RCS*600 is closed and the PRZR is heated until RCS pressure is approximately 150 psig. THEN the PORV is cycled and RCS Pressure is kept at 150 PSIG as level is lowered. At these conditions, the PRZR is purged of non-condensable gas as a steam bubble is formed.

D. The PRZR is filled to ≥ 80% with one PORV open. The PORV is then CLOSED, and the PRZR is heated to 250-275°F. THEN RCS Pressure is lowered to 25-30 PSIG as level is lowered. At these conditions, dissolved non-condensable gases are minimized while avoiding solid plant conditions.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Plausible due to correct bubble forming process
**B.** Correct. IAW 2OM-50.4.L step IV.F and H.
**C.** Incorrect. Plausible because 80% is solid plant, incorporates 150# to open PORV
**D.** Incorrect. Plausible due to 80%, correct bubble parameters, requires knowledge of PORV ops

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 007 | Pressurizer Relief Tank/Quench Tank System (PRTS) | K5 Knowledge of the operational implications of the following concepts as the apply to PRTS: | Method of forming a steam bubble in the PZR |

**K/A#** K5.02 **K/A Importance** 3.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-50.4.L rev 19 pg 18 & 27

**Question Source:** Modified to address PRT

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41(b)10

**Objective:**

34. The plant is cooling down to mode 4 for maintenance.

    - RCS Temperature is 520°F
    - RCS pressure is 1970 psig and stable
    - PRZR Low Press Safety Injection Train A and B have been BLOCKED
    - Steam Line Safety Injection Train A and B have been BLOCKED

    A steam line break inside containment has occurred.

    Which of the following will result in an automatic **Safety Injection**?

A. Pressurizer pressure at 1840 psig

B. Containment Pressure at 6 psig

C. Steam Generator Steam Pressure at 500 psig

D. Steam Generator Steam Pressure at 900 psig and lowering at 200 psig/min

---

**Answer: B**

**Explanation/Justification:**

A. Incorrect. Plausible because PRZR pressure <1856 psig would actuate Safety injection, but not when the PRZR Low Press Safety Injection Train A and B have been BLOCKED.

B. Correct. Cnmt pressure >5 psig will actuate Safety Injection.

C. Incorrect. Plausible because Steam line pressure low (500 psig) would actuate Safety Injection, but not when Steam Line Safety Injection Train A and B have been BLOCKED.

D. Incorrect. Plausible because Safety Injection would actuate at 500 psig plus rate, but this is blocked by the Steam Line Safety Injection Train A and B being BLOCKED. The rate of 200 psig/min would still initiate a MSLI due to the block installing a 1oo psig with 50 sec TC.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 103 | Containment System | Generic | Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. |

**K/A#** 2.4.4 **K/A Importance** 4.5 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-11.2.B Rev 5 pg 4; USFAR Fig. 7.3-13 Rev 13; BVPS Brainbook U2 ESF Setpoints

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(7)

**Objective:**

35. Given the following conditions:

- The crew is responding to a Primary Component Cooling Water System (CCP) Leak
- The "A" and "B" headers have been split
- The crew is isolating the "A" containment header in an attempt to identify the location of the leak.

After isolating the "A" containment header, which of the following components will still be supplied with CCP flow?

A. 'B' RCP Thermal barrier HX

B. Neutron Shield Tank Cooler

C. Excess Letdown HX

D. Primary Drains Cooler

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. The B RCP thermal barrier is supplied from the B CCP header.
**B.** Incorrect. NST is supplied from the A CCP header.
**C.** Incorrect. Excess Letdown HX is supplied from the A CCP header
**D.** Incorrect. Primary Drains Cooler is supplied from the A CCP header

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 008 | Component Cooling Water | K1: Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: | Loads cooled by CCWS |

**K/A#** K1.02 **K/A Importance** 3.3 **Exam Level** RO

**References provided to Candidate** None **Technical References:** RM-0415-002 Rev 19
2SQS-15.1 PPNT Rev 9 Iss 2 slide 9
2OM-53C.4.2.15.1 Rev 4

**Question Source:** Bank

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41(b)4

**Objective:**

36. Given the following indication:

- 2CCP-PI145A, 'A' CCP pump discharge pressure is 120 psig and stable.
- Annunciator A6-1H, PRI COMP COOLING WATER SYSTEM TROUBLE is LIT.
- 2CCP-RQ100, Component Cooling Water Radiation Monitor is in High Alarm.

1) What has caused these indications?

2) If no operator action is taken, what is the **main** concern associated with this event?

A. 1) Surge Tank level RISING due to CCP system in leakage.
2) Spread of contamination and radioactivity to the public.

B. 1) Surge Tank level RISING due to CCP system in leakage.
2) Loss of CCP pumps from cavitation.

C. 1) Surge Tank level DROPPING due to CCP system out leakage.
2) Spread of contamination and radioactivity to the public.

D. 1) Surge Tank level DROPPING due to CCP system out leakage.
2) Loss of CCP pumps from cavitation.

---

**Answer: A**

**Explanation/Justification:** (1) CCW system in leakage occurred due to the rising surge tank level causing 6H-1H annunciator to come in and high radiation on 2CCP-RQI100. The annunciator would come in for out-leakage on low level, but the radiation would not.
(2) Spreading contamination and radiation is a concern with system in leakage. Loss of CCP pumps from cavitation is plausible but is not the primary concern.

**A.** Correct. See answer explanation/justification above.
**B.** Incorrect. See answer explanation/justification above.
**C.** Incorrect. See answer explanation/justification above.
**D.** Incorrect. See answer explanation/justification above.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 008 | Component Cooling Water System (CCWS) | A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: | Surge tank level |

**K/A#** A1.04 **K/A Importance** 3.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2SQS15.1 PPNT rev. 9 iss 2, slide 42
2OM-53C.4.2.15.1 Rev 4 pg 1, 4, 13

**Question Source:** New

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.5 / 45.5)

**Objective:** 2SQS15.1 Rev 9 OBJ 18 - Given in-leakage or out-leakage to/from the CCP system, describe all the means by which the in-leakage can be detected, the consequences of the leakage and the actions taken to correct the leakage.

37. Auxiliary spray is being placed in service due to a loss of all RCPs. The RCS is at normal operating temperature. The VCT is at 120°F.

How is thermal shock to the PRZR spray nozzle prevented when charging is aligned to provide auxiliary spray?

A. The PRZR spray bypass valves maintain the spray line warm.

B. Aux spray is not placed in service unless letdown and the regenerative HX are in service.

C. Spray line cooldown rate is monitored and maintained < 100°F/hour.

D. Spray is not initiated if PRZR to spray temperature difference is > 100°F.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Plausible is the RCS where operating maintaining some flow through the spray line bypass valves.

**B.** Correct. As stated in P&L 62 of 2OM-6.2.A. Auxiliary spray flow from the charging line should NOT be initiated unless the regenerative heat exchanger is in service and letdown flow is passing through the shell side.

**C.** Incorrect. Plausible if the candidate is thinking the spray line cooldown limits are the same as the RCS cooldown limits which is not the case for this component. LRM states max aux spray temperature differential is 380°F.

**D.** Incorrect. Plausible because if przr to spray temperature difference is > 100°F, aux spray flow must be initiated slowly.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 010 | Pressurizer Pressure Control System (PZR PCS) | K4 Knowledge of PZR PCS design feature(s) and/or interlock(s) which provide for the following: | Spray valve warm-up |

**K/A#** K4.01 **K/A Importance** 2.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-6.2.A Rev 23 P&L 62

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41(b)4

**Objective:**

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

- The reactor automatically trips.
- **ONLY** the "B" Reactor Trip Breaker opens.

Which of the following describes the response of the Steam Dump Valve System **DIRECTLY** following the reactor trip? (Assume no operator action)

A. The Reactor Trip Controller modulates all 4 banks of steam dump valves.

B. The Load Rejection Controller modulates all 4 banks of steam dump valves.

C. The Reactor Trip Controller modulates ONLY the 1st and 2nd banks of steam dump valves.

D. The Load Rejection Controller modulates ONLY the 1st and 2nd banks of steam dump valves.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Correct that the reactor trip controller is in service; however, only two versus all four banks modulate to control temperature. It is true that all four banks are armed since the "A" did not open. In order to close the C-7B contact, the reset switch would need to be taken to reset.

**B.** Incorrect. The Reactor Trip Controller is in service due to the "B" RTB opening. Incorrect that all four banks modulate.

**C.** Correct. Opening "B" RTB places the Reactor Trip Controller into service. The first and second banks of steam dump valves arm from a C-7A signal, the third and fourth bank banks arm from the C-7B signal. Since the Reactor Trip Controller maximum output of 12 ma can only open the first and second banks of steam dumps, they are the only banks that modulate to bring Tavg to 547F.

**D.** Incorrect. The Reactor Trip Controller is in service. Correct that only two banks modulate.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 012 | Reactor Protection System | K3 Knowledge of the effect that a loss or malfunction of the RPS will have on the following: | SDS |

**K/A#** K3.03 **K/A Importance** 3.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-21.5.A.12 Rev 3<br>2OM-21.5.A.13 Rev 3<br>2OM-21.1.D Rev 1 pg 6-9

**Question Source:** Bank - 17 from 2LP-SQS-21.1

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41 b(7)

**Objective:**

39. The plant is operating at 100% power.
    - Pressurizer Pressure channel 1 fails LOW.

    What effect does this have on the setpoints for OP$\Delta$T and OT$\Delta$T?

A. Loop 1 OP$\Delta$T only is affected, and indicates LOWER than normal.

B. Loop 1 OT$\Delta$T only is affected, and indicates LOWER than normal.

C. Loop 1 OP$\Delta$T only is affected, and indicates HIGHER than normal.

D. Loop 1 OT$\Delta$T only is affected, and indicates HIGHER than normal.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. No pressure input to overpower delta T
**B.** Correct. Pressure low is a negative penalty
**C.** Incorrect. No pressure input
**D.** Incorrect. Negative penalty

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 012 | Reactor Protection System | A3 Ability to monitor automatic operation of the RPS, including: | Single and multiple channel trip indicators |

**K/A#** A3.05 **K/A Importance** 3.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 3SQS-1.1 Rev 8 pg 25
3SQS-1.1 Rev 8 PPNT slide 60
T.S. pgs 3.3.1-19,20,21

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41(b)10

**Objective:** 3SQS-1.1 Rev 8 Obj. 11 - Given a specific plant condition, predict or describe the response of the Reactor Protection System Trip Logics & Engineered Safeguards Features Actuation Signals control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off-normal condition.

40. Given the following plant conditions:

- The plant is operating at 100% with all systems in NSA.
- A Feed Flow instrument failure on the 'B' Steam Generator (S/G) causes 2FWS*FCV488, "Main Feed Regulating Valve" to go full open.
- A turbine/reactor trip occurs from the resultant 'B' Hi-Hi S/G water level.
- Reactor Coolant System (RCS) temperature is 547°F and stable.
- All equipment operates as designed and no operator action has occurred.

Based on these conditions, what will be the status of the following feedwater components **30 seconds after the trip**?

| | 2FWS*FCV488 (Main FRV) | 2FWS*FCV489 (Main FRV Bypass) | Main Feedwater Pumps |
|---|---|---|---|
| A. | OPEN | CLOSED | RUNNING |
| B. | CLOSED | CLOSED | TRIPPED |
| C. | CLOSED | CLOSED | RUNNING |
| D. | CLOSED | OPEN | TRIPPED |

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. 2FWS*FCV488 closes on a full FWI signal. It is plausible that this valve would reopen once the Hi-HI S/G water level signal clears as it taps off above the set/reset device. It is possible that on a reactor trip, the S/G water level will shrink down far enough to clear the Hi-HI signal and thus any feedwater/steamflow mismatch would reopen 2FWS*FCV488. The MFP will trip due to Full FWI and will not restart without operator action.

**B.** Correct. In accordance with the feedwater logic diagrams, Feedwater Isolation occurs on a Hi-HI S/G level (92.2%). This results in MFRV's and MFRBV's closing and the MFP's tripping.

**C.** Incorrect. This would be the correct response for a partial FWI signal. On a Low Tavg (< 554 F) and a reactor trip, a partial feedwater isolation will occur. As a result, 2FWS*FCV488 will close and not reopen. MFRBV's NSA position is closed in manual, so therefore 2FWS*FCV489 will be closed. The MFP will remain running on a partial FWI.

**D.** Incorrect. Incorrect 2FWS*FCV489 position. Correct MFP status. Correct 2FWS*FCV488 status. Plausible if the candidate does not understand the feedwater isolation logic.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 013 | Engineered Safety Features Actuation System (ESFAS) | K4 Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: | MFW isolation/reset |

**K/A#** K4.13 **K/A Importance** 3.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** UFSAR Figure 7.3-18, Rev. 9

**Question Source:** Bank - 2LOT7 Q44

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(7)

**Objective:**

41. Given the following conditions:

- The plant is at 20% power.
- 2FWS-LT494, 'C' SG NR level channel has been placed in the tripped condition IAW 2OM-24.4.IF, Instrument Failure procedure.
- Subsequently, 2FWS-LT495, 'C' SG NR level channel fails low.

What is the plant response?

The Reactor will trip, and ______________

A. all AFW pumps start.

B. no AFW pumps start.

C. only the MDAFW pumps start.

D. only the TDAFW pump starts.

---

**Answer: D**

**Explanation/Justification:** Power at 20% to allow the question to test the AFW pump start coincidence verses SG shrink below 20.5% on trip.

**A.** Incorrect. Plausible if 2/3 low-low levels were on 2/3 SGs. In this case only 1 SG is affected by these conditions.

**B.** Incorrect. Plausible if the candidate feels that LT495 trip condition is high, and know that 1 SG level detector failing low does not start an AFW.

**C.** Incorrect. Plausible if the candidate does not know that 2/3 low-low levels on 2/3 SGs coincidence is required to start the MDAFW pumps.

**D.** Correct. LT494 will be in the trip condition of low-low, therefore when LT495 fails low, the 2/3 low-low level on 1/3 SGs coincidence required to start the TDAFW pump will be met.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 013 | Engineered Safety Features Actuation | K6 Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: | Sensors and detectors |

**K/A#** K6.01 **K/A Importance** 2.7 **Exam Level** RO

**References provided to Candidate** None

**Technical References:** UFSAR Logic Fig. 7.3-12 Rev 12
UFSAR Logic Fig. 7.3-19 Rev 7
2OM-24.1.D Rev 6 pg 18

**Question Source:** New

**Question Cognitive Level:** Memory

**10 CFR Part 55 Content:** 55.41(b)7

**Objective:**

42. Given the following initial conditions:

- Plant is at 100% power
- 'A' and 'B' CNMT Air Recirc (CAR) Fans are RUNNING
- 'C' CNMT Air Recirc Fan is aligned to Bus 2N, with the control switch GREEN targeted

Subsequent conditions:
- A11-6G, Containment Air Recirc Fan Auto Stop is in alarm
- 'A' CNMT Air Recirc Fan indicating light is Bright WHITE

Based on the above conditions, complete the following statements.

'C' CAR fan will ______(1)______.

Cooling water ______(2)______ automatically aligned to the 'C' Containment Air Recirculation Cooling Coils.

A. 1) auto start
2) is

B. 1) auto start
2) is NOT

C. 1) remain shutdown
2) is

D. 1) remain shutdown
2) is NOT

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Cnmt Air Recirc Fans do not auto start. Plausible distractor because the CS in the control room is STOP-AUTO-START, and the stem stated the CS was green targeted and aligned to the same bus that 'A' CAR fan is energized from. Second part is plausible because some valves do auto align on a system startup.

**B.** Incorrect. Cnmt Air Recirc Fans do not auto start. Plausible distractor because the CS in the control room is STOP-AUTO-START, and the stem stated the CS was green targeted and aligned to the same bus that 'A' CAR fan is energized from. Second part is correct.

**C.** Incorrect. 'C' CAR fan will remain shutdown. Second part is plausible because some valves do auto align on a system startup.

**D.** Correct. 'C' CAR fan will remain shutdown. Second part is also correct. OM-44C.4.D states to align cooling water if it is necessary for containment temperature control.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 022 | Containment Cooling System (CCS) | A2 Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: | Fan motor thermal overload/high-speed operation |

**K/A#** A2.03 **K/A Importance** 2.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-44C.4.AAC rev. 6
2OM-44C.4.D rev. 19

**Question Source:** Bank - 1LOT16 Q40

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(9)

**Objective:**

43. The plant is operating at full power with all equipment in NSA when the following events occur:

| Time | Event |
|---|---|
| T = 0 | The reactor trips due to a LOCA inside CNMT. |
| T = 30 seconds | Bus 2AE de-energizes due to an overcurrent fault. |
| T = 10 minutes | Containment Isolation-Phase B (CIB) automatically actuates. |
| T = 40 minutes | The RWST level is 367 inches and lowering on all channels. |

Which of the following conditions exist 40 minutes after the LOCA occurred?

1. Recirc Spray Pump 'B' discharge aligned to the High Head Safety Injection suction header.
2. Service Water flowing through the 'B' Train Recirc spray heat exchangers.
3. Recirculation Spray Pump 'A' running with discharge aligned to the spray rings.
4. Quench Spray Pump 'A' discharge isolation valve 2QSS*MOV101A open.
5. Recirc Spray Pump 'D' discharge aligned to the High Head Safety Injection suction header.

A. 1 and 2 only.

B. 2, 4, and 5 only.

C. 1, 3, and 4 only.

D. 3 and 5 only.

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. 'B' RSS pump does not align to the HHSI suction, only 'C' and 'D' RSS pumps align to HHSI suction on transfer to recirc. Service water will be flowing through the 'B' train RSS Hxs after the CIB.

**B.** Correct. Service water will be flowing through the 'B' train RSS Hxs after the CIB. 'A' QSS pump discharge isolation valve 2QSS*MOV101A is NSA open, and the loss of power to the 2AE bus does not change the position. 'D' RSS pump will be aligned to HHSI suction on transfer to recirc at 369 RWST level coincident with SI signal.

**C.** Incorrect. 'B' RSS pump does not align to the HHSI suction, only 'C' and 'D' RSS pumps align to HHSI suction on transfer to recirc. 'A' RSS pump will not be running due to the overcurrent fault on bus 2AE causing the bus to be de-energized. 'A' QSS pump discharge isolation valve 2QSS*MOV101A is NSA open.

**D.** Incorrect. 'A' RSS pump will not be running due to the overcurrent fault on bus 2AE causing the bus to be de-energized. 'D' RSS pump will be aligned to HHSI suction on transfer to recirc at 369 RWST level coincident with SI signal.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 026 | Containment Spray System (CSS) | K2 Knowledge of bus power supplies to the following: | Containment spray pumps |

**K/A#** K2.01 **K/A Importance** 3.4 **Exam Level** RO

**References provided to Candidate** None

**Technical References:** 2OM-11.1.D Rev 1 pg 14, 2OM-11.2.B Rev 5 pg 4
2OM-13.3.B.1 Rev 11 pg 3
2OM-13.1.D Rev 4 pg 6
2OM-30.1.D Rev 8 pg 7

**Question Source:** Bank- modified

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41(b)7

**Objective:**

44. A DBA LOCA has occurred. Quench Spray pumps cannot be started.

What are the consequences of this malfunction?

A. Containment failure will occur by the end of the LOCA blowdown phase.

B. Recirc spray pumps must be crosstied to backup Quench Spray to maintain containment pressure to within design limits.

C. Sodium Tetra-Borate (NaTB) will not be dissolved, and containment sump pH will not be adequate to maintain off site releases within accident analysis containment leakage assumptions.

D. Peak containment pressure will be turned by passive heat sinks, but containment pressure reduction may not occur within accident analysis assumptions.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect, peak pressure is turned by passive heat sinks as seen on 2SQS13.1 PPT slide 6.

**B.** Incorrect, no such crosstie. Plausible if the applicant does not know the system flowpath and plausible since they both are spray systems.

**C.** Incorrect, NaTB baskets are a passive system and will be mixed with containment sump water and spread using the recirculation spray system.

**D.** Correct.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 026 | Containment Spray System (CSS) | A2 Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: | Failure of spray pump |

**K/A#** A2.04 **K/A Importance** 3.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** Tech Spec Bases pgs B3.6.6-2-4, 2SQS13.1 PPT rev. 18

**Question Source:** New

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(8)

**Objective:**

45. Given the following conditions:

- A load rejection from 100% power has occurred
- Reactor power is currently 65%
- Tave is 11°F higher than Tref

How many banks of condenser steam dumps will be fully or partially open?

A. 1

B. 2

C. 3

D. 4

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. See correct answer.
**B.** Correct. The Load Rejection controller modulates between 4-20 ma demand for a temperature error of 3-24F. This controller is what causes the Bank 1 dumps to be full open at 7.5ma (7.6°F error) and the Bank 2 dumps to be full open at 11.3ma (12.7°F error), therefore Bank 1 will be full open and Bank 2 will be partially open. Bank 3 will not begin to open until 11.3ma (12.7°F error) only if C-7B is picked up.
**C.** Incorrect. See correct answer.
**D.** Incorrect. See correct answer.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 039 | Main and Reheat Steam System (MRSS) | A4 Ability to manually operate and/or monitor in the control room: | Steam dump valves. |

**K/A#** A4.07 **K/A Importance** 2.8 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-21.5.A.12 Rev 3
2OM-21.5.A.16 Rev 2
2OM-21.1.C Rev 7 pg 8
2SQS-21.1 Rev 23 pg. 46

**Question Source:** Bank

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41(b)5

**Objective:**

46. Given the following plant conditions and sequence of events:

- The plant is operating at 100% power with all systems in NSA EXCEPT:
  - 2FWE*P23B, "B Motor Driven Auxiliary Feedwater Pump" is OOS for 4 hours, and applicable actions of Tech Spec 3.7.5, AFW System have been completed.
- A Loss of Offsite power coincident with a turbine trip occurs.
- Bus 2AE has an overcurrent lockout.
- All systems function as designed.

With no operator action, which of the following describes the response of the Auxiliary Feedwater (AFW) System?

A total AFW flow of approximately ____(1)____ GPM will be provided to ALL Steam Generators through the ____(2)____.

A. (1) 375
(2) "A" Header ONLY.

B. (1) 750
(2) "B" Header ONLY.

C. (1) 750
(2) "A" Header ONLY.

D. (1) 900
(2) "A" AND "B" Headers.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Incorrect capacity and incorrect header. Plausible if the candidate believes NSA is to the "A" header or believes the impact of 2FWE*P23A is realignment of 2FWE-P22 to the "A" header.

**B.** Correct. Correct capacity. Correct header. A loss of offsite power coincident with a turbine trip results in a reactor trip and subsequent loss of both MFW pumps. The EDGs are designed to start on a loss of power to AE and DF bus which will power both electric AFW pumps. In the stated conditions, with an overcurrent condition on the AE bus, 2FWE*P23A will not have power. Since 2FWE*P23B is already OOS, only 2FWE-P22 (Turbine Driven AFW pump) will start to provide approximately 750 gpm AFW flow. The AFW system is designed to feed all three S/G based on NSA requirements. When 2FWE*P23B is OOS, corrective action has 2FWE-P22 aligned to the "B" Header.

**C.** Incorrect. Plausible if the candidate believes NSA is to the "A" header or believes the impact of 2FWE*P23B is realignment of 2FWE-P22 to the "A" header.

**D.** Incorrect. Correct capacity. If the candidate does not know the capacities or understand the impact based on initial plant conditions, then it is plausible that AFW flow would be provided through the "A" header by 2FWE-P22 and the "B" header by 2FWE*P23A. In this case the total flow will be 900 gpm based on limiting orifices which limit flow to 300 gpm per S/G. Incorrect because 2FWE*P23B has no power.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 059 | Main Feedwater (MFW) System | K3 Knowledge of the effect that a loss or malfunction of the MFW will have on the following: | AFW system |

**K/A#** K3.02 **K/A Importance** 3.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-24.1.C, Rev.3, pg. 5 & 6
2SQS-24.1, Rev. 26 PPNT slide 71
2OM-24.2.A Rev 15 pg 6

**Question Source:** Bank Modified – 2LOT8 Q42

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(7)

**Objective:**

47. A loss of offsite power has occurred at 100% power.

1) What is the design heat removal capacity with the Primary Plant Demineralized Water Storage Tank (PPDWST) 2FWE-TK210 as the water source?

2) Per 2OM53A.1.A-1.8, Makeup to PPDWST, what are the water sources for the Auxiliary Feed Pumps as the PPDWST is depleted?

A. 1) Maintain Hot Standby for 16 hours.
2) Demineralized Water Storage Tank **and** Service Water.

B. 1) Maintain Hot Standby for 9 hours.
2) Demineralized Water Storage Tank **and** Service Water.

C. 1) Maintain Hot Standby for 16 hours.
2) Service Water **and** Fire Protection header.

D. 1) Maintain Hot Standby for 9 hours.
2) Service Water **and** Fire Protection header.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Plausible distractor due to the volume of the PPDWST. Second part is correct.

**B.** Correct. The usable volume of the PPDWST is sufficient to maintain the plant in hot stby for at least 9 hours following a LOOP. Second part is correct IAW Att. A-1.8 step 8, and when level reaches 25", Service Water is aligned to the AFW pumps.

**C.** Incorrect. Plausible distractor due to the volume of the PPDWST. Second part is plausible because Fire Protection header sounds like a viable source of makeup water, but it is not used as a makeup source for the AFW pumps.

**D.** Incorrect. First part is correct. Second part is plausible because Fire Protection header sounds like a viable source of makeup water, but it is not used as a makeup source for the AFW pumps.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 061 | Auxiliary / Emergency Feedwater (AFW) System | K1. Knowledge of the physical connections and/or cause effect relationships between the AFW and the following systems: | Emergency water source |

**K/A#** K1.07 **K/A Importance** 3.6 **Exam Level** RO

**References provided to Candidate** None

**Technical References:** 2OM-53A.1.A-1.8 Rev. 7 pg. 2 & 3
TS Bases pg. B-3.7.6-1
2SQS-24.1 Rev. 26

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** CFR: 41.2 to 41.9 / 45.7 to 45.8)

**Objective:**

48. Given the following plant conditions:

- The plant is operating at 100% power.
- 2OST-36.2, "Emergency Diesel Generator (2EGS*EG2-2) Monthly Test" is in progress.
- 2-2 EDG is paralleled to the grid, carrying about 50% load.
- A grid disturbance causes frequency to drop very slightly.
- Grid Voltage remains constant.

Which of the following describes the response of 2-2 EDG **AND** what is the significance of operating the EDG above 4535 KW for extended periods of time?

The response of 2-2 EDG is that ______(1)______, AND the significance of operating this EDG > 4535 KW is excessive ______(2)______.

A. (1) KW output RISES and KVAR output is STABLE
(2) mechanical stress on the EDG engine

B. (1) KW output LOWERS and KVAR output is STABLE
(2) accumulation of combustion and lubricating products in the exhaust system

C. (1) KW output and KVAR output RISES
(2) mechanical stress on the EDG engine

D. (1) KW output and KVAR output LOWERS
(2) accumulation of combustion and lubricating products in the exhaust system

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. If frequency drops, the EDG will attempt to increase speed, which will pick up real load. TS Surveillance 3.8.1.3 bases states that the load band (3814 to 4238 kW) which is more restrictive than the rated load in 2OST-36.2 (4535 kW) is to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations for DG OPERABILITY.

**B.** Incorrect. KW output will rise when the EDG attempts to raise grid frequency. The reason for significance of EDG loading is for ensuring loading is maintained >50% for an hour when operating the EDG at low loads for extended periods of time. This limit is plausible in that it is more associated with operating the EDG at low loads and could be confused by the candidate.

**C.** Incorrect. KVAR output will remain essentially constant if grid voltage is constant. If it did change it would change in the opposite direction of KW. Significance of operating above rated limit is correct as explained above.

**D.** Incorrect. KW will rise. Reason for load limit is incorrect as explained above.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 062 | A.C. Electrical Distribution | A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: | Significance of D/G load limits |

**K/A#** A1.01 **K/A Importance** 3.4 **Exam Level** RO

**References provided to Candidate** None **Technical References:** GO-GPF.C5 Rev. 2 pgs. 133-135
2OST-36.2, Rev. 73
TS 3.8.1 Amend. 292/179 & Bases. Rev. 29

**Question Source:** Bank - 2LOT7 Q47

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** (41.5 / 43.5 / 45.3 / 45.13)

**Objective:** 12. Given a specific plant condition, predict the response of the 4KV Distribution System control room indication control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off-normal condition.

49. Which of the following states the correct required action for the stated plant mode and malfunctions?

    All equipment is **OPERABLE** other than the stated failures.

A. Mode 1: A loss of "A" 4kv bus occurs.
   Required Action: restore within one hour.

B. Mode 6: A loss of EDG 2-1 occurs.
   Required Action: immediately suspend core alterations.

C. Mode 1: A loss of "2AE" 4kv bus and 125V DC bus 2-2 occurs.
   Required Action: enter T.S. 3.0.3.

D. Mode 6: A loss of 125V DC bus 2-2 occurs.
   Required Action: immediately suspend core alterations.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Plausible because TS 3.8.1 requires 2OST-36.7, Breaker Alignment Verification to be performed within one hour, but the bus is required to be restored within 72 hours.

**B.** Incorrect. Plausible because TS 3.8.2 requires one EDG and one qualified circuit between offsite and onsite. The question stem states all equipment is operable, therefore EDG 2-2 is operable and the LCO is met. Suspending core alts would be correct if both EDGs were inoperable.

**C.** Correct. TS 3.8.9 cond. E, the '2AE' bus is one AC source, and the loss of DC bus 2-2 removes control power from the train 'B' components, and removes the starting capability of EDG 2-2. TS 3.8.1 cond. H have been met by the loss of 2AE bus means no offsite alignment is available, the 2-1 DG is not available, and DC bus 2-2 removes control power to 2F7 & 2F10 breakers, as well as the inability to start 2-2 EDG.

**D.** Incorrect. Plausible because TS 3.8.5 requires One DC electrical power subsystem shall be operable in Mode 5 and 6. The candidate may think that both DC electrical power subsystem as in TS 3.8.4.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 063 | D.C. Electrical Distribution | Generic | Knowledge of less than or equal to one hour Technical Specification action statements for systems. |

**K/A#** 2.2.39 **K/A Importance** 3.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** T.S. 3.8.1 and bases; T.S. 3.8.9 and bases

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41(b)10

**Objective:**

50. Which of the following provides control power to the 2-1 EDG?

A. DC*SWBD 2-1

B. DC*SWBD 2-3

C. DC*SWBD 2-5

D. DC*SWBD 2-6

**Answer: A**

**Explanation/Justification:**

**A.** Correct. Supplies EDG 2-1 control power and other DC loads on the EDG.
**B.** Incorrect. Plausible because it is 1 of 2 Train A safety related DC busses.
**C.** Incorrect. Plausible because it supplies DC Control power to normal 480 and 4160VAC busses.
**D.** Incorrect. Alternate supply to DC bus 2-5.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 064 | Emergency Diesel Generators (ED/G) | K2 Knowledge of bus power supplies to the following: | Control power |

**K/A#** K2.03 **K/A Importance** 3.2 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2SQS-36.2 LP rev 21 pg 31
2SQS 36.2 PPNT rev 21 slide 94
2OM-36.3.C.8 Rev 13 pg 13

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(8)

**Objective:**

51. An EDG has started and sequence loaded during a station blackout. Which of the following engine or breaker trips are still active in this event?

A. Crankcase pressure high

B. Jacket water temperature high

C. Lube oil pressure low

D. Engine overspeed

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Plausible distractor because Crankcase pressure high could damage the DG and does give a local annunciator.

**B.** Incorrect. Plausible distractor because Jacket water temperature high is an automatic shutdown of the DG during any operation except while in emergency run mode.

**C.** Incorrect. Plausible distractor because Lube oil pressure low is an automatic shutdown of the DG during any operation except while in emergency run mode.

**D.** Correct. While in emergency run mode, automatic shutdown of the DG will only be accomplished by engine overspeed and generator and excitation system electrical protection relay action.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 064 | Emergency Diesel Generator | K4 Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: | Trips while loading the ED/G (frequency, voltage, speed) |

**K/A#** K4.01 **K/A Importance** 3.8 **Exam Level** RO

**References provided to Candidate** None

**Technical References:** 2OM-36.1.D Iss 4 Rev 3 pg 24<br>U2 LSK-022-006D rev 7<br>U2 LSK-022-006F rev 8

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41(b)7

**Objective:**

52. Given the following plant conditions:

- The plant is operating at 100%.
- A liquid waste discharge is in progress to the Unit 1 Cooling Tower Blowdown.
- 2SWS-RQI102, Component Cooling HX SW fails upscale **HIGH**.
- 2CCP-RQI100, Component Cooling Water is reading normal and is unchanged.
- The following alarms are received:
  - A4-5A, RADIATION MONITORING SYSTEM TROUBLE
  - A4-5C, RADIATION MONITORING LEVEL HIGH
- On RM-11, it is confirmed that COMPONENT COOLING HX SW, 2SWS-RQI102 is blinking RED.
- No other alarms are present and no operator action has occurred.
- All systems function as designed.

What will be the impact of this process monitor failure on the effluent release in progress?

The release will ______________

A. automatically terminate immediately.

B. continue and **IS** required to be manually terminated.

C. automatically terminate after a short time delay.

D. continue and is **NOT** required to be manually terminated.

---

**Answer: D**

**Explanation/Justification:**

A. Incorrect. Plausible if the candidate confuses this monitor with 2SGC-RQ100 which would result in auto termination if an upscale failure occurred.

B. Incorrect. Correct that release will continue and plausible but incorrect that the release must be manually terminated. The candidate must understand system interrelationships. With CCP in normal there is no reason to believe there is any radiation coming from the CCP system into the SW system.

C. Incorrect. Plausible if the candidate confuses this monitor with 2SGC-RQ100 which would result in auto termination if an upscale failure occurred and they also confuse or do not know that there is no time delay with this failure.

D. Correct. The candidate must understand how the failure of 2SWS-RQI102 will impact the effluent release in progress. There is no automatic action associated with this radiation monitor. Therefore, an upscale failure will have no impact on the release. The candidate must also know the system interrelationships between CCP and SW. If there were a leak from the RCS into CCP, then there would be an alarm from 2CCP-RQI100. The ARP does not require any release to be terminated.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 073 | Process Radiation Monitoring (PRM) System | A4 Ability to manually operate and/or monitor in the control room: | Effluent release |

**K/A#** A4.01 **K/A Importance** 3.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-43.4.AAA rev. 9, 2OM-43.4.AAC rev 1, 2OM-43.4.ACG rev 5, 2OM-43.4.AEI rev 7, 2OM-43.4.ACO rev 7, 2OM-43.1.C rev. 5

**Question Source:** Bank - 2LOT8 #51

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(7)

**Objective:**

53. While operating at 100% power, a small break LOCA and LOOP occurs.

- SI, CIA, and CIB signals actuate.
- A fault occurs on the 2DF bus, and the supply breakers 2D10 and 2F7 trip.
- The 2-2 EDG starts but does not load due to the faults.

Which of the following states consequences of the fault?

A. 2SWS*MOV113A, 2SIS*MOV865A, and 2SIS*MOV865B have lost power.
MCCs E05 & E06 must be jumpered to isolate Accumulators during the post LOCA cooldown.

B. 2SWS*MOV113D, 2SIS*MOV865B, and 2SIS*MOV865C have lost power.
MCCs E05 & E06 must be jumpered to isolate Accumulators during the post LOCA cooldown.

C. 2SWS*MOV113A, 2SIS*MOV865A, and 2SIS*MOV865B have lost power.
Accumulators must be vented during post LOCA cooldown.

D. 2SWS*MOV113D, 2SIS*MOV865B, and 2SIS*MOV865C have lost power.
Accumulators must be vented during post LOCA cooldown.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. wrong ACCs affected, plausible because crosstie is possible.
**B.** Incorrect. wrong EDG SW valve, plausible because partially correct.
**C.** Incorrect. wrong ACCs affected, plausible because crosstie possible.
**D.** Correct.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 076 | Service Water System (SWS) | K2 Knowledge of bus power supplies to the following: | ESF-actuated MOVs |

**K/A#** K2.08 **K/A Importance** 3.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM37.5.B.7 Rev 43 table 37-7 pg 108 & 117
2OM-53A.1.ES-1.2 Iss 3 Rev 0 pg 21

**Question Source:** Modified/New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41(b)4

**Objective:**

54. Following a loss of instrument air, which of the following valves will fail OPEN?

A. 2CHS-FCV114B, Boric Acid Dilute Injection to VCT

B. 2CHS-FCV113B, Boric Acid Blender Discharge to Injection Pumps

C. 2CHS-HCV186, RCP Water Injection Filter Control Valve

D. 2CHS-PCV116A, VCT Pressure Control Valve

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Fails closed.
**B.** Incorrect. Fails closed.
**C.** Correct. AOP 2.34.1 pg 20.
**D.** Incorrect. Fails closed.

| Sys # | System | Category | KA Statement |
| --- | --- | --- | --- |
| 078 | Instrument Air System (IAS) | K3 Knowledge of the effect that a loss or malfunction of the IAS will have on the following: | Systems having pneumatic valves and controls |

**K/A#** K3.02 **K/A Importance** 3.4 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.34.1 rev 20 pg 20

**Question Source:** Modified - 2005 NRC #27 common

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(4)

**Objective:**

55. A large break LOCA occurred 24 hours ago. Containment sprays have reduced containment pressure and radiation levels. Plant conditions are as follows:

- Containment pressure peaked at 50 psig, but has been reduced to 4 psig.
- Containment area radiation was 1E+5 R/hr for the first 12 hours, but has been reduced to 1E+4 R/hr.

Is the use of ADVERSE containment values still required?

A. Yes, once ADVERSE values have been required they remain in effect even after the entry criteria has been cleared.

B. Yes, the integrated CNMT radiation criteria has been exceeded.

C. Yes, CNMT radiation still exceeds the ADVERSE value.

D. No, the use of ADVERSE values is no longer required.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect.
**B.** Correct. 1.2E+6 integrated exposure.
**C.** Incorrect.
**D.** Incorrect. The Integrated rad value of 1E+6 R/Hr was exceeded.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 103 | Containment System | A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: | Containment pressure, temperature, and humidity |

**K/A#** A1.01 **K/A Importance** 3.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.E-0 Iss. 3 Rev 0 LHP
1/2OM-53B.5.GI16 Iss. 1C Rev 0 pg.76

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41(b)10

**Objective:** 3SQS-53.2, Rev. 2 Obj. 15 - Define from memory adverse containment conditions, IAW BVPS EOP Executive Volume.

56. Which of the following describes the sequence of components from power supply to the Control Rod Drive Mechanism (CRDM's)?

(RTB's = Reactor trip Breakers)
(RDMG's = Rod Drive Motor Generators)

A. 480 VAC Substation 2-8 & 2-9, RDMG's, RTB's, Power Cabinets.

B. 480 VAC Substation 2-8 & 2-9, Power Cabinets, RDMG's, RTB's.

C. 480 VAC Substation 2-1 & 2-2, RDMG's, RTB's, Power Cabinets.

D. 480 VAC Substation 2-1 & 2-2, Power Cabinets, RDMG's, RTB's.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Plausible incorrect emergency power supply with correct flowpath.
**B.** Incorrect. Plausible incorrect emergency power supply with incorrect flowpath.
**C.** Correct. The candidate must know the power supply to the Motor Generator Sets and have understanding of the flowpath of this power to the Control Rod Drive Mechanisms. 480 VAC Substation 2-1 supplies power to 2RDS-MG21 and 480 VAC Substation 2-2 supplies power to 2RDS-MG22. The proper flowpath is via the RDMGs via the RTBs through the power cabinets to the CRDMs.
**D.** Incorrect. Correct power supply with plausible incorrect flowpath.

---

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 001 | Control Rod Drive System | K2 Knowledge of bus power supplies to the following: | M/G sets |

**K/A#** K2.05 **K/A Importance** 3.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-1.3.C, Rev. 27, pg. 13
3SQS1.3 PPNT Rev. 7 Iss 1 slide 62

**Question Source:** Bank – 2LOT8 Q56

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.41 b(2)

**Objective:**

57. The plant was at 100% power when a load rejection occurred.

Current conditions:

- Rx power is 71% and stable
- Tavg is 565F and stable
- RCS pressure is 2250 psig and stable
- All PRZR heaters are in AUTO
- 2RCS-PK444A, PRZR Press Control is in AUTO at 42% output
- 8% PRZR level insurge has occurred
- NO operator actions have occurred

What is the current status of the pressurizer heaters?

A. All heaters are OFF.

B. All heaters are ON.

C. Only Backup heaters are ON.

D. Only Proportional heaters are ON.

---

**Answer: C**

**Explanation/Justification:**

A. Incorrect. Plausible because the load rejection caused a przr insurge, which causes pressure to rise. The candidate may think that normal przr pressure control will cause all of the heaters to be de-energized, but przr level control will energize the BU heaters due to the level deviation.

B. Incorrect. Plausible if the candidate recognizes that the level deviation will energize heaters, but only the BU heaters are energized, the proportional/control heaters are not effected by przr level, and with the master pressure controller at 42% demand output, there is no demand for heaters.

C. Correct. When the przr high level deviation reaches 5% above program, the BU heaters are energized to return the przr to saturation conditions. The conditions given of an 8% insurge causes the backup heaters to energize from the level control logic. The proportional heaters will be de-energized since they are controlled from the przr pressure controller which is at 42% and has no heater control demand.

D. Incorrect. Plausible if the candidate recognizes that the level deviation will energize heaters, but only the BU heaters are energized, the proportional/control heaters are not effected by przr level, and are only effected by the przr pressure controller which will have driven the proportional heaters off when at 42% demand output.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 011 | Pressurizer Level Control System (PZR LCS) | A4 Ability to manually operate and/or monitor in the control room: | PZR heaters |

**K/A#** A4.03 **K/A Importance** 3.3 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-6.4.IF Rev 13 Att 1 pg. 12 & Att 2 pg. 24, 25

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41(b)7

**Objective:**

58. The plant is at 100% RTP with Rod Control in AUTO, when a 30% load rejection occurred. Assume no operator actions have been taken.

What happens to Axial Flux Difference ($\Delta$I), and what is the preferred method to restore it?

1) Axial Flux Difference ($\Delta$I) will be ______(1)______.

2) The preferred method to restore Axial Flux Difference ($\Delta$I) from this condition is to ______(2)______.

A. 1) more negative
2) dilute

B. 1) more negative
2) borate

C. 1) more positive
2) dilute

D. 1) more positive
2) borate

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. See correct answer justification.

**B.** Correct. As rods insert power is suppressed in the upper region of the core. AFD is defined as Power at top-Power at bottom, therefore, with rods suppressing power at the top, the power at the bottom is higher causing AFD to lower (or become more negative). The correct action IAW AOP 2.35.2, is to borate.

**C.** Incorrect. See correct answer justification.

**D.** Incorrect. See correct answer justification.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 014 | Rod Position Indication System (RPIS) | A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPIS controls, including: | Axial and radial power distribution |

**K/A#** A1.04 **K/A Importance** 3.5 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.35.2 Rev 20 step 1
3SQS-2.1 Rev 9 PPNT slide 146-147
GO-GPF.R5 Rev 1 pg 39

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental Knowledge **10 CFR Part 55 Content:** 55.41 b(1)

**Objective:**

59. Which one of the following describes the impact on the indication of a core exit thermocouple if it were to short or become an open circuit?

| | Short | Open |
|---|---|---|
| A. | Low | Low |
| B. | Low | High |
| C. | High | Low |
| D. | High | High |

---

**Answer: A**

**Explanation/Justification:** Comment – part b) of KA not met, not operationally relevant due to 51 TCs.

**A.** Correct. TCs are based on voltage diff between metals, open or short means no delta volts, therefore indicates low.

**B.** Incorrect. Plausible if the candidate does not know that both an open and a short indicate low.

**C.** Incorrect. Plausible if the candidate does not know that both an open and a short indicate low.

**D.** Incorrect. Plausible if the candidate does not know that both an open and a short indicate low.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 017 | In-Core Temperature Monitor System (ITM) | A2 Ability to (a) predict the impacts of the following malfunctions or operations on the ITM system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: | Thermocouple open and short circuits |

**K/A#** A2.01 **K/A Importance** 3.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 3SQS-3.1, Rev. 6 pg 16

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41(b)2

**Objective:** 3SQS-3.1 obj. 10 - Describe the response of a thermocouple readout to open and short circuits.

60. The plant is operating at 100% power with all systems in normal alignment **EXCEPT** Emergency 480V Bus 2N is de-energized for emergent maintenance.

- An inadvertent turbine trip occurs.
- The "B" reactor trip breaker **FAILS** to OPEN.
- The Condenser steam dumps fail to OPEN.

Without any operator action, where will RCS temperature automatically stabilize?

A. 541°F

B. 547°F

C. 550°F

D. 554°F

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. This is where RCS would stabilize if it were relying on the steam dump lo-lo Tavg interlock to stop a cooldown.

**B.** Incorrect. This is where RCS would stabilize if it were being controlled by the Rx trip controller. However, with "B" trip breaker still closed, the steam dumps would function on the load rejection controller.

**C.** Incorrect. This is where the RCS temperature would stabilize if they operated properly on the load rejection controller. which has a 3°F deadband before it will open the steam dumps, but the stem states the condenser steam dumps fail to open.

**D.** Correct. This is where RCS will stabilize when relying on the SG safeties to control temperature. SG Safety valves 2MSS*SV101A-C have a setpoint of 1075 psig. The Atmospheric steam dumps will be failed closed by the loss of 2N bus which supplies all 3 Atm dumps.

| Sys # | System | Category | KA Statement |
| --- | --- | --- | --- |
| 041 | Steam Dump System (SDS) and Turbine Bypass Control | K3 Knowledge of the effect that a loss or malfunction of the SDS will have on the following: | RCS |

| | | | | | |
| --- | --- | --- | --- | --- | --- |
| **K/A#** | K3.02 | **K/A Importance** | 3.8 | **Exam Level** | RO |
| **References provided to Candidate** | | None | | **Technical References:** | 2OM-21.2.B Rev 7 pg 5<br>2OM-21.1.C Rev 7 pg 10<br>2OM-21.3.C Rev 9 pg 3<br>2OM-21.5.A.12 Rev 3<br>3SQS-37.1 Rev 9 Iss 2 U2 PPNT slide 14 |

**Question Source:** Modified - 2LOT6 Q62

**Question Cognitive Level:** Comprehension or Analysis

**10 CFR Part 55 Content:** 55.41 b(4)

**Objective:**

61. A pipe break occurs in the spent fuel pool cooling system.

What is the lowest design level to which water can be siphoned by the break, and what is basis for that level?

A. 23' over the top of fuel assemblies in the fuel racks, to ensure building accessibility and cooling if the break occurs during fuel movement.

B. 10' over the top of an assembly being moved, to ensure compliance with technical specifications in the event of a pipe break.

C. 23' over the top of fuel assemblies in the fuel racks, to ensure compliance with technical specifications in the event of a pipe break.

D. 10' over the top of fuel assemblies in the fuel racks, to ensure building accessibility and fuel cooling in the event of a pipe break.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Plausible if applicant remembers 23', but not what it's for.
**B.** Incorrect. Plausible because 10' above seated fuel is the actual design value.
**C.** Incorrect. Plausible because 23' is the TS number, but not anti-siphon number.
**D.** Correct.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 033 | SFP Cooling | K4 Knowledge of design feature(s) and/or interlock(s) which provide for the following: | Anti-siphon devices |

**K/A#** K4.03 **K/A Importance** 2.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-20.1.B Rev 2, 2SQS-20.1 Rev 13

**Question Source:** Bank

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41(b)4

**Objective:**

62. Which component's function is to maneuver the control rod drive shaft unlatching tool?

A. Manipulator Crane Gripper

B. Manipulator Crane Auxiliary Hoist

C. Spent Fuel Bridge Crane

D. Manipulator Crane Mast

---

**Answer: B**

**Explanation/Justification:**

- **A.** Incorrect. The manipulator crane gripper engages the top of the fuel assembly.
- **B.** Correct. The manipulator crane auxiliary hoist is used to manipulate the control rod drive shaft unlatching tool.
- **C.** Incorrect. The spent fuel bridge crane is used to transfer fuel between the new fuel storage racks, new fuel elevator, the spent fuel storage racks and the fuel transfer system.
- **D.** Incorrect. The manipulator crane mast contains the gripper tube gives lateral support to the fuel assembly during movement.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 034 | Fuel Handling | Generic | Knowledge of system purpose and/or function. |

**K/A#** 2.1.27 **K/A Importance** 3.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 3SQS-6.13, Rev. 6 pg 2

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(13)

**Objective:** 3SQS-6.13 Rev 6 Obj 1 - 1. Describe the function of the following Fuel Handling equipment as documented in the Refueling Procedures: b. Manipulator crane auxiliary hoist

63. Given the following conditions:

- The plant is at 80% power when a steam leak occurs on the 'A' Main Steamline outside of CNMT.
- The reactor does not automatically trip.
- No operator actions have been taken.

What is the response of the 'A' Steam Generator water level? Assume normal response of the SG level control system.

A. Initial level rise, then return to program value.

B. Initial level decrease, then return to program value.

C. Initial level rise, then stabilize above program value.

D. Initial level decrease, then stabilize below program value.

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. Swell will cause 'A' SG level to rise due to an increase in steam flow causing SG pressure to lower. The SGWLC system will initially respond to the increased steam flow, but will restore SGWL to the program level of 44%.

**B.** Incorrect. Plausible if the candidate thinks that excess steam demand will cause 'A' SGWL to lower until the 'A' FRV opens to restore SGWL to program. The candidate may think the SGWLC will reduce flow until the level control responds.

**C.** Incorrect. Plausible because 'A' SG level will rise due to swell, but the second part is incorrect. The SGWLC system will restore 'A' SGWL to the program level of 44%. The candidate may think the flow error affects program level.

**D.** Incorrect. Plausible if the candidate thinks that excess steam demand will cause 'A' SGWL to lower, and based on the current SGWL, SGWLC system maintains the levels at the new SGWL which is below the program value. The candidate may think the flow error affects program level.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 035 | Steam Generator System (S/GS) | K5 Knowledge of operational implications of the following concepts as the apply to the S/GS: | Shrink and swell concept |

**K/A#** K5.03 **K/A Importance** 2.8 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2SQS-24.1 PPNT, Rev. 26 Slides 45, 55-57

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.41(b)5

**Objective:**

64. Given the following:

- The Unit is operating at 100% power with all systems in NSA.
- A liquid waste discharge is in progress to the Unit 2 cooling tower blowdown.
- A **HIGH** radiation alarm occurs on 2SGC-RQI100, Liquid Waste Process Monitor.

Which of the following describes the action that will occur?

(2SGC-HCV100 - Liquid Waste Effluent High Radiation Isolation Valve)

A. Manually close 2SGC-HCV100, terminating the release.

B. Verify 2SGC-HCV100 closes automatically, immediately terminating the release.

C. Manually open 2SGC-HCV100, diverting the release to the Unit 1 boron recovery test tank.

D. Verify 2SGC-HCV100 opens automatically, immediately diverting the release to the Unit 1 boron recovery test tank.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. The valve closes automatically. Plausible if the applicant believes that the HCV must be manually closed.

**B.** Correct. ARP 2OM-43.4.AEE, Liquid Waste Process Effluent 2SGC-RQI100 High Alarm states verify closed 2SGC-HCV100 and stop 2SGC-P26A/B SGBD test tank pumps. 2SGC-HCV100 receives an automatic signal to close upon high radiation signal.

**C.** Incorrect. Opening HCV100 will not divert to U1 boron recovery system. Plausible because the path to Unit 1 BRS is DOWNSTREAM of HCV100 and would prevent liquid waste from being discharged.

**D.** Incorrect. Opening HCV100 will not divert to U1 boron recovery system.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 068 | Liquid Radwaste System (LRS) | A3. Ability to monitor automatic operation of the Liquid Radwaste System including: | Automatic isolation |

**K/A#** A3.02 **K/A Importance** 3.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-43.4.AEE Iss 1 Rev 5; 2SQS-17.1 Rev 9 PPNT slide 23; 2OM-25.1.D Iss 4 Rev 1 pg 10

**Question Source:** Bank - BVPS U2 2005 Q36

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(13)

**Objective:** 2SQS-17.1 Rev 9 Obj. 10 - Given a specific plant condition, predict the response of the Liquid Waste Disposal System control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off-normal condition.

65. Which of the following describes the purpose and operation of the following Radiation monitors?

- 2MSS-RQ101A, B, & C; Main Steam Line Radiation Monitors
- 2MSS-RQ102A, B, & C; Main Steam Line N16 Radiation Monitors

A. RQ101 channels are high range monitors which are continuously in-service.
RQ102 channels are SG tube leak monitors.

B. RQ101 channels are SG tube leak monitors.
RQ102 channels are release monitors which are in-service whenever a steam release flowpath is open.

C. RQ101 channels have an auto isolation function.
RQ102 channels are indication only monitors.

D. RQ101 channels are release monitors that auto align on an SI signal.
RQ102 channels are SG tube leak monitors with NO auto isolation function.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. RQ101 are high range detectors but are only placed in service manually or automatically with a Safety Injection signal. RQ102s are a SG tube leak monitor.

**B.** Incorrect. RQ101s are high range detectors used during and after an accident, not for SG tube leak monitoring. RQ102s are SG tube leakage monitors which are continuously in service, and do not monitor steam releases.

**C.** Incorrect. RQ101s are normally isolated unless 2MSS-SOV120 is automatically opened by a Safety Injection signal, or manually during accident conditions. There are no auto isolation signals associated with RQ101s.RQ102s are SG tube leak indication monitors.

**D.** Correct. RQ101s auto align by automatically opening 2MSS-SOV120 on a Safety Injection signal, or manually, during accident conditions. RQ102s are SG tube leak monitors with no automatic features.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 072 | Area Radiation Monitoring (ARM) System | K1 Knowledge of the physical connections and/or cause-effect relationships between the ARM system and the following systems: | MRSS |

**K/A#** K1.05 **K/A Importance** 2.8 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-43.1.C Rev 5 pgs. 23 & 59
2OM-43.1.B Rev 2 pg 5

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.41(b)11

**Objective:**

66. You are instructed to escort a group of visitors around the plant to perform maintenance. They are required to go into the Protected and Vital areas.

1) What is the maximum number of visitors that you can escort by yourself into a **Vital area**?

2) As you pass through a fire door (not a Key Card door), you notice that the door will not automatically close to the latched position. What actions are you required to take?

A. 1) 5
2) Contact the Control Room to establish a continuous fire watch patrol on one side of the affected barrier.

B. 1) 5
2) Manually close and latch the fire door shut, and establish measures to assure the door remains latched except when opened for transit.

C. 1) 10
2) Contact the Control Room to establish a continuous fire watch patrol on one side of the affected barrier.

D. 1) 10
2) Manually close and latch the fire door shut, and establish measures to assure the door remains latched except when opened for transit.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. See answer explanation/justification.

**B.** Correct. IAW NOP-LP-1205, Visitor control, 5:1 ratio is the maximum for a vital area, 10:1 is for a protected area. Since they are going into a vital area, a maximum of 5 visitors is allowed. IAW 1/2-ADM-1900, if the door does not automatically close to the latched position, it must be manually closed and latched and normally a sign placed on the door. If the fire door cannot be manually closed and latched in the closed position, then a continuous fire watch must be established.

**C.** Incorrect. See answer explanation/justification.

**D.** Incorrect. See answer explanation/justification.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of facility requirements for controlling vital/controlled access. |

**K/A#** 2.1.13 **K/A Importance** 2.5 **Exam Level** RO

**References provided to Candidate** None **Technical References:** NOP-LP-1205 rev. 5 pg. 5
1/2-ADM-1900 rev 40 pgs. 88-89

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(9/10)

**Objective:** FEN-PAT, Rev. 10 Chap 5 Obj. 9 - Describe escorting responsibilities.

3SQS-33.1-1-06: Identify the Fire Protection System field instruments, subsystems and components that are required to be operable by the 1/2-ADM-1900 "Fire Protection".

67. A shift turnover **shall not** be conducted in which of the following circumstances?

A. A plant startup is in progress. The generator has just been synced to the grid, and SG level control is in manual.

B. A reactor startup is in progress. Source Range NIs have indicated 3 doublings since rod withdrawal commenced.

C. A diesel surveillance is in progress. The diesel is running loaded.

D. A one hour LCO is in effect. The LCO will expire during the turnover.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Plausible because infrequent evolution, component in manual

**B.** Correct. Per NOP-OP-1002 section 4.13.2.16, Shift turnover shall not be conducted during the approach to criticality.

**C.** Incorrect. Plausible because it is a surveillance test; Conduct of Operations says it is undesirable for evolution to take place and operator may need to remain on station, but turnover is not prohibited.

**D.** Incorrect. Plausible because the operator may need to remain on station, but turnover can proceed.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of shift or short-term relief turnover practices. |

**K/A#** 2.1.3 **K/A Importance** 3.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** NOP-OP-1002 Rev 12 pg. 71

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** (CFR: 41.10 / 45.13)

**Objective:**

68. In an emergency, which position can authorize the use of a Human Clearance?

A. WEC SRO

B. Operations Manager

C. Shift Manager

D. Unit Supervisor

---

**Answer: C**

**Explanation/Justification:**

- **A.** Incorrect. Plausible distractor because the Work Execution Center SRO authorizes daily and emergent work for the plant, but does not have the authority to authorize human clearance.
- **B.** Incorrect. Plausible because the Operations Manager has overall responsibility for the implementation of the procedure.
- **C.** Correct. IAW NOP-OP-1001, page 14, section 4.2.4, the SM "determines the appropriateness of the use of Human Clearances in times of emergency".
- **D.** Incorrect. Plausible because the Unit Supervisor is an SRO, but the Shift Manager has the authority.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of tagging and clearance procedures. |

**K/A#** 2.2.13 **K/A Importance** 4.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** NOP-OP-1001 Rev 24 pg 14

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(10)

**Objective:**

69. You are the first person performing the latest revision of a pump surveillance. While lining up for the surveillance, you realize based on your system knowledge that the procedure incorrectly requires you to verify "CLOSED" a valve that needs to be "OPEN".

    In accordance with NOP-LP-2601, Procedure/Work Instruction Use and Adherence, this would be defined as what type of error, and what action is required?

A. This is a <u>Typographical Error</u>; annotate a correction and continue with the surveillance.

B. The procedure requires an <u>Enhancement</u>; annotate a correction and continue the surveillance.

C. This is a <u>Deficiency</u>; the procedure cannot be continued until the procedure is revised.

D. This is a change of procedure <u>Intent</u>; the procedure cannot be continued until the procedure is revised.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Typographical Error - A simple keying, copying, pasting or transposition error made during procedure preparation that was not corrected during procedure review and approval processes. Whole word errors such as open and close, on and off, increase and decrease, or component identification errors of any sort, or Functional Location number errors (as present in SAP) are not Typographical Errors.

**B.** Incorrect. Procedure Enhancement – Procedure issues that are simply punctuation, style, insignificant word or title errors, and reference errors. This includes situations where functional location information in the procedure differs slightly from field labels or other information.

**C.** Correct. Deficiency - Essential information that is missing or incorrect. A deficiency exists when the procedure cannot or should not be implemented as written, and must be revised prior to continuing with the evolution.

**D.** Incorrect. "Intent" is not defined as a type of error. Plausible distractor because the intent of the procedure is evaluated when referring to procedural errors.

---

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of the process for making changes to procedures. |

**K/A#** 2.2.6 **K/A Importance** 3.0 **Exam Level** RO

**References provided to Candidate** None **Technical References:** NOP-LP-2601 Rev 6 pg 4, 5, 15, 16

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** (CFR: 41.10 / 43.3 / 45.13)

**Objective:**

70. When performing an OST procedure, which one of the following conditions PROHIBITS the use of "N/A" in the sign-off spaces provided?

A. Performance of partial tests.

B. Inability to perform the OST as written.

C. Performing an OST that pre-establishes conditions for non-performance of steps.

D. Performance of steps that cannot be performed due to plant conditions but do not change the intent of the procedure.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Partial tests allow N/A.
**B.** Correct. Situation requires issuing a revision after placing equipment in a safe condition.
**C.** Incorrect. N/A is specifically used for this condition.
**D.** Incorrect. May use N/A as long as procedure intent is not altered.

---

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of surveillance procedures. |

**K/A#** 2.2.12 **K/A Importance** 3.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** NOP-LP-2601 Rev 6 pg 7-9

**Question Source:** Bank – BVPS U2 2005 Q68

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(10)

**Objective:** 3SSG-ADMIN Rev 9 Obj. 5 - EXPLAIN the requirements for the use of plant procedures in accordance with: a. NOP-LP-2601, Procedure Use And Adherence

71. A large break LOCA is in progress. Core damage has occurred with high radiation levels in accessible plant areas.

To prevent the loss of an ECCS pump, an individual is dispatched to locally operate a failed MOV in an area surveyed at 100 R/hr area. He becomes injured and immobilized, and two individuals are sent to rescue him.

What are the BVPS voluntary exposure limits for the **original operation**, and the **subsequent rescue**?

A. valve operation: 5 REM as a Planned Special Exposure.
two rescuers: 10 REM each lifesaving activities.

B. valve operation: 5 REM as a Planned Special Exposure.
two rescuers: 10 MAN-REM combined total for all lifesaving activities.

C. valve operation: 10 REM as an emergency exposure.
two rescuers: 25 MAN-REM combined total for all lifesaving activities.

D. valve operation: 10 REM as an emergency exposure.
two rescuers: 25 REM each for all lifesaving activities.

---

**Answer: D**

**Explanation/Justification:**

- **A.** Incorrect. Incorrect dose values but the 5 rem is plausible because it is equivalent to the annual occupational dose limit, and the second part is the dose limit allowed for protecting valuable property.
- **B.** Incorrect. Incorrect dose values but the 5 rem is plausible because it is equivalent to the annual occupational dose limit, and the second part is is double this value.
- **C.** Incorrect. First part is correct. Second part is plausible if they think that the lifesaving activity total dose is only allowed to be 25 rem.
- **D.** Correct. 10 rem to protect valuable equipment and 25 rem each to save a life.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of Radiation exposure limit under normal or emergency conditions |

**K/A#** 2.3.4 **K/A Importance** 3.2 **Exam Level** RO

**References provided to Candidate** None **Technical References:** FEN-RWT Chap. 4, Rev 4 pg 5

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** (CFR: 41.12 / 43.4 / 45.10)

**Objective:**

72. You have been assigned the task of venting a radioactive system that is located in a Locked High Radiation Area (LHRA).

When you open the vent valve you receive an UNEXPECTED dose rate alarm on your electronic alarming dosimeter (EAD).

IAW NOP-OP-4101, Access Controls for Radiologically Controlled Areas, what are your REQUIRED actions for these conditions?

A. Immediately notify Radiation Protection (RP) and stay in the area to wait for personnel for decontamination.

B. Close the vent valve, and report the alarm to the Control Room and Radiation Protection (RP).

C. Immediately exit the area, perform whole body frisk, and notify Radiation Protection (RP).

D. Close the vent valve, immediately exit the area, and notify Radiation Protection (RP).

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. These are the correct actions personnel contamination.
**B.** Incorrect. These would be appropriate actions for an alarming air monitor.
**C.** Incorrect. Frisking is required before exiting the RCA but not necessarily required as part of LHRA exit.
**D.** Correct. If the dose rate alarm activates, place your work in a safe condition, notify your coworkers, exit the area and contact RP

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. |

**K/A#** 2.3.12 **K/A Importance** 3.2 **Exam Level** RO

**References provided to Candidate** None **Technical References:** NOP-OP-4101 Rev 12
FEN-RWT Rev 4 Chapter 6 pg. 4 & 8

**Question Source:** Bank – 2LOT6 Q71

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(12)

**Objective:**

73. A serious fire is in progress in the Primary Auxiliary Building, elevation 755. The Shift Manager has determined that the Safe Shutdown Following a Serious Fire in the Primary Auxiliary Building procedure, 2OM-56B.4.A, must be entered.

In which time critical order should the following actions be completed to maintain engineering commitments?

1. Isolate Hydrogen to the VCT
2. Isolate Charging Pump Suction from the VCT
3. Isolate PG Water

A. 3, 1, 2

B. 1, 3, 2

C. 3, 2, 1

D. 1, 2, 3

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. See correct answer.
**B.** Correct. 25 minutes to isolate Hydrogen, 29 to isolate PG water, and 2 hours to Isolate Charging Pump Suction from the VCT.
**C.** Incorrect. See correct answer.
**D.** Incorrect. See correct answer.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of fire protection procedures. |

**K/A#** 2.4.25 **K/A Importance** 3.3 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-56B.4.A Rev 20 pg 44

**Question Source:** Bank - Modified

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.13)

**Objective:**

74. Following a reactor trip, the crew entered E-0, Reactor Trip or Safety Injection.

1) When is the crew required to perform Continuous Action Steps?
2) When is the crew allowed to terminate monitoring for Continuous Action Steps?

A. 1) As soon as the crew enters the EOP.
2) After exiting that EOP.

B. 1) When that step is reached in the procedure.
2) After exiting that EOP.

C. 1) As soon as the crew enters the EOP.
2) After exiting **ALL** EOPs.

D. 1) When that step is reached in the procedure.
2) After exiting **ALL** EOPs.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. First part is plausible if the applicant confuses CAS with symptom based actions. Second part correct.
**B.** Correct. The action is performed when reaching that step in the procedure and continuous throughout the rest of that EOP.
**C.** Incorrect. First part is plausible if the applicant confuses CAS with symptom based actions. Second part is plausible if the applicant believes that a CAS is always valid in the EOPs.
**D.** Incorrect. First part correct. Second part is plausible if the applicant believes that a CAS is always valid in the EOPs.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of EOP terms and definitions. |

**K/A#** 2.4.17 **K/A Importance** 3.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 3SQS-53.1 rev 2 PPNT slide 36
1/2OM-53B.2 rev 9 pg 6-7

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(10)

**Objective:** 3SQS53.1 PPNT rev. 2: State from memory and apply ALL of the EOP user's guide rules of usage as defined in 1/2-OM53B.2

75. A station event has just been classified as a Site Area Emergency. You are the communicator.

    Which of the following describes the time requirement for Local and State notifications?

A. 30 minutes from the initiation of the event.

B. 15 minutes from the classification of the event.

C. 15 minutes from the completion of the Initial Notification Form.

D. 15 minutes from the time the communicator acknowledges the briefing from the Shift Manager/ Emergency Director.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Applicant may think 15 min to classify, then additional 15 min to notify.
**B.** Correct.
**C.** Incorrect. Applicant may think classification time includes form prep
**D.** Incorrect. Applicant may think he has 15 minutes from time he receives form.

---

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of the emergency plan. |

**K/A#** 2.4.29 **K/A Importance** 3.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 1/2-EPP-IP-1.1.F02 Rev. 24

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.11)

**Objective:**

76. A large Steam break accident inside containment has occurred.

- Containment pressure peaked at 20 psig.
- All Equipment functioned as designed **EXCEPT** all seal injection flow has been lost.
- SI, CIA, and CIB have all been reset.
- SWS has been restored to the CCP heat exchangers.
- CCP flow has been restored.
- While performing EOP Attachment A-1.2, Establishing RCP CCP Cooling and Seal Injection, the Reactor Operator is unable to OPEN 2CCP*AOV107A, 21A RCP Thermal Barrier Outlet Isol Vlv using the benchboard control switch.

To OPEN 2CCP-AOV107A, it will be necessary to defeat the 'CLOSE' signal to the valve.

IAW EOP Attachment A-1.2, what directions are you **REQUIRED** to give the local operator to defeat the 'CLOSE' signal to 2CCP*AOV107A?

A. Install jumpers across the opening contacts of the valve's control circuit.

B. Remove the valve's associated secondary process rack power supply card.

C. Remove the valve's associated control circuit power supply fuse.

D. Install jumpers across the contacts of the high discharge flow transmitter.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Although this may open the valve, it is NOT IAW EOP attachment A-1.2.
**B.** Correct. IAW EOP attachment A-1.2 step 4.b.3.
**C.** Incorrect. This action will fail the valve closed.
**D.** Incorrect. This action will only defeat the high flow signal BUT NOT the high pressure and it is NOT IAW EOP attachment A-1.2

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 015 | Reactor Coolant Pump (RCP) Malfunctions | Generic | Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. |

**K/A#** 2.4.35 **K/A Importance** 4.0 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.A-1.2 Rev 2

**Question Source:** Bank – 2LOT6 Q100

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.43 b(5)

**Objective:**

77. The crew has just entered FR-S.1 due to an ATWS from 100% power. Which of the following conditions must be met to allow the crew to exit FR-S.1?

1. Rods are fully inserted by any means
2. SI is required
3. Turbine is verified tripped
4. Power less than 5% with negative SUR

A. 1 only

B. 1 OR 2 only

C. 1 AND 3 only

D. 4 only

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Plausible because IOA step 1 of FR-S.1 inserts the control rods. Even though Rods are fully inserted, and negative reactivity has been added, it does not allow for a transition out of the procedure.

**B.** Incorrect. Plausible because IOA step 1 of FR-S.1 inserts the control rods and step 3 will initiate emergency boration via SI or a boration flowpath. Even though both steps add negative reactivity, neither of them alone will allow for a transition out of the procedure.

**C.** Incorrect. Plausible because IOA step 1 of FR-S.1 inserts the control rods, and trips the turbine. Even though these steps are adding negative reactivity and minimizing plant cooldown, they do not allow for a transition out of the procedure.

**D.** Correct. Step 7, a Continuous action step of FR-S.1 is the first transition out of the procedure. The step "Check if reactor is subcritical" checks PR <5% and negative SUR.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000029 | Anticipated Transient Without Scram / 1 | EA2 Ability to determine or interpret the following as they apply to a ATWS: | Occurrence of a main turbine/reactor trip |

**K/A#** EA2.09 **K/A Importance** 4.5 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.FR-S.1 Iss 2 Rev 0 step 7

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.43(b)5

**Objective:**

78. Given the following conditions:
    - The Unit was at 100% power when a Reactor Trip occurred due to a Loss of Main Feedwater.
    - E-0, Reactor Trip or Safety Injection, was exited at Step 4 and the crew transitioned to ES-0.1, Reactor Trip Response.
    - While in ES-0.1, a RED path developed on the Heat Sink Critical Safety Function Status Tree due to a loss of all Auxiliary Feedwater flow.
    - The crew transitioned to FR-H.1, Response to Loss of Secondary Heat Sink.
    - Both Charging Pumps are available.
    - PRZR pressure is 2000 psig and slowly rising.
    - The Reactor Coolant Pumps have been stopped.
    - Steam Generator (SG) narrow range levels are as follows:
        - SG A is 5%.
        - SG B is 7%.
        - SG C is 6%.

    Which of the following actions must be performed per FRH-1, Response to Loss of Secondary Heat Sink?

    FR-H.1 can be exited as soon as narrow range level in any Steam Generator exceeds _______

A. 12%. A transition back to ES-0.1, Reactor Trip Response, is performed and continues at the step in effect.

B. 31%. A transition back to ES-0.1, Reactor Trip Response, is performed and continues at the step in effect.

C. 12%. A transition to E-0, Reactor Trip or Safety Injection, is performed and proper Safety Injection actuation and alignment is verified.

D. 31%. A transition to E-0, Reactor Trip or Safety Injection, is performed and proper Safety Injection actuation and alignment is verified.

# QUESTION 78

**Answer: A**

**Explanation/Justification:**

**A.** Correct. Correct setpoint in FR-H.1. And step 8 says to return to procedure and step in effect.

**B.** Incorrect. Plausible setpoint as this is the setpoint for adverse containment. Correct transition.

**C.** Incorrect. Correct setpoint. Plausible transition because this would appear to be a transition if bleed and feed using Safety Injection was established, but it is not referenced within FR-H.1. A transition to ES-1.1 is required after Bleed and Feed.

**D.** Incorrect. Plausible setpoint, but nothing indicates Cnmt is adverse. Plausible transition because this would appear to be a transition if bleed and feed using Safety Injection was established, but it is not referenced within FR-H.1. A transition to ES-1.1 is required after Bleed and Feed.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 054 | Loss of Main Feedwater (MFW) | Generic | Knowledge of EOP mitigation strategies. |

**K/A#** 2.4.6 **K/A Importance** 4.7 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** FR-H.1 issue 2 rev. 1, steps 7, 8, and 15

**Question Source:** Bank - 2012 Comanche Peak Q76

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.43 b(5)

**Objective:**

79. The reactor is at 100% power when the following annunciators illuminate:

- A1-1A, DC DISTRIBUTION PANEL LOSS OF CONTROL DC
- A2-3H, SAFETY SYSTEM TRAIN A INOPERABLE
- A8-9A, 125V DC BUS 2-1 TROUBLE

Which of the following procedure sequences would be correct for this condition?

E-0, Reactor Trip or Safety Injection
AOP 2.39.1A, Loss of 125VDC Bus 2-1
AOP 2.51.1, Unplanned Power Reduction

A. Enter AOP 2.39.1A, perform E-0 IOAs from memory, and continue in AOP 2.39.1A.

B. Enter AOP 2.39.1A, and enter AOP 2.51.1 concurrently.

C. Enter E-0, perform E-0 or ES-0.1 and enter AOP 2.39.1A concurrently.

D. Enter E-0 if an SI occurs, otherwise enter and remain in AOP 2.39.1A.

---

**Answer: C**

**Explanation/Justification:**

A. Incorrect. AOP 2.39.1A is plausible because it is the correct AOP to enter, and the first step of the AOP has the crew verify the IOAs are completed, but in this case, the reactor will trip and E-0 must be entered.

B. Incorrect. Plausible distractor to enter AOP 2.39.1A and AOP 2.51.1 if the candidate does not recognize that a reactor trip occurs with the loss of DC bus 1, and feels that power must be lowered due to one of the DC bus loads being lost.

C. Correct. In accordance with the Transient Response Guideline section 4.14, E-0 entry on the Rx trip is correct based on this being the highest priority procedure during this event, then after E-0 IOAs, enter AOP 2.39.1A and perform both procedures concurrently.

D. Incorrect. Entering E-0 is correct based on this being the highest priority procedure, but it does not matter whether an SI occurred or not, both procedures will be performed concurrently.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000058 | Loss of DC Power / 6 | Generic | Knowledge of abnormal condition procedures |

**K/A#** 2.4.11 **K/A Importance** 4.2 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** BVPS-OPS-0024 Rev 9 pg 14-15
2OM-39.4.AAD Rev 7 pg 4
2OM-5A.4.AAC Rev 10 pg 73

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.43(b)5

**Objective:**

80. The Unit is operating at 100% power with all systems in NSA.

- A large leak occurs in the Service Water System.
- The control room receives A1-4H, "SERVICE WATER SYSTEM TROUBLE" followed shortly after by A1-4G, "SERVICE WATER HEADER PRESSURE LOW".
- "A" & "B" SW Header Pressures **BOTH** indicate 28 psig and slowly DROPPING.
- "A" & "B" CCS Water HX Service Water Supply Header Isolation Valves (2SWS-MOV107A/B/C/D) automatically isolate **AND** cannot be re-opened.
- **AFTER** 2SWS*MOV107A/B/C/D automatically isolate, "A" & "B" SW Header Pressures begin to RISE.

1) Based on these plant conditions, which Service Water System component is leaking?

2) IAW AOP 2.30.1, Service Water/Normal Intake Structure Loss which of the below listed components are **required** to be tripped?

- All Station Air Compressors
- All Main Feed Pumps
- All Heater Drain Pumps
- All Condensate Pumps

A. 1) The in-service Primary Component Cooling Heat Exchangers, 2CCP-E21A, B, C
2) ONLY the Main Feed Pumps and Condensate Pumps.

B. 1) The in-service Centrifugal Water Chillers, 2CDS-CHL23A, B, C
2) ONLY the Main Feed Pumps and Heater Drain Pumps.

C. 1) The in-service Primary Component Cooling Heat Exchangers, 2CCP-E21A, B, C
2) ONLY Station Air Compressors and Heater Drain Pumps.

D. 1) The in-service Centrifugal Water Chillers, 2CDS-CHL23A, B, C
2) ONLY the Station Air Compressors and Condensate Pumps.

# QUESTION 80

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Incorrect leaking component and condensate pumps are not to be tripped in AOP 2.30.1.

**B.** Correct. Since pressure recovered when 2SWS*MOV107A/B/C/D isolated, the leak must be in the secondary side header. The Centrifugal Water Chillers are on the secondary side header and the Primary Component Cooling Heat Exchangers are on the primary side header. The first part of the question can be answered with RO knowledge. The second part is SRO only since it requires specific knowledge of procedure content and cannot be answered with system knowledge alone. All four pumps listed in the stem of the question will lose cooling when the 107 valves close. The SRO must be familiar enough with the AOP content to know what actions are to be taken. The procedure directs the alignment of an alternate cooling water supply to the air compressors, and the starting of one condensate pump if none are running. The SRO must therefore assess plant conditions (normal, abnormal, or emergency) and then select the actions with which to proceed.

**C.** Incorrect. Incorrect leaking component and air compressors are not to be tripped in AOP 2.30.1.

**D.** Incorrect. Correct leaking component and air compressors and condensate pumps are not to be tripped in AOP 2.30.1.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 062 | Loss of Nuclear Service Water | AA2. Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: | Location of a leak in the SWS |

**K/A#** AA2.01 **K/A Importance** 3.5 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.30.1 Rev. 9 Steps 7,9, & 10
2SQS-30.1 rev 23 PPNT slides 7 & 10

**Question Source:** Bank - 2LOT8 Q80

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.43 b(5)

**Objective:**

81. The reactor has just tripped on low pressurizer pressure from 100% power.
The crew has just completed E-0, Reactor Trip or Safety Injection immediate operator actions.
Plant conditions are as follows:

- Reactor trip confirmed
- SI initiated
- RCS pressure is 1600 psig lowering rapidly
- PRZR level is empty
- SG pressures: approximately 900 psig and stable
- SG levels: below NR, rising on WR
- Containment pressure: -1.3 psig, stable
- Auxiliary Building radiation monitors: alarming
- Auxiliary Building sump: Hi level alarms

1) If the crew is unable to isolate the leak, what is the expected EOP flowpath?
2) IAW the EOP network, what additional action would be required if the RWST level decreases below 30 inches?

E-1, Loss of Reactor or Secondary Coolant
ES-1.3, Transfer to Cold Leg Recirculation
ECA-1.1, Loss of Emergency Coolant Recirculation
ECA-1.2, LOCA Outside Containment

A. 1) E-0 > ECA-1.2 > ECA-1.1
2) Makeup to RCS as directed by TSC.

B. 1) E-0 > ECA-1.2 > ECA-1.1
2) Fill Containment sump using Fire Protection system.

C. 1) E-0 > E-1 > ES-1.3
2) Makeup to RCS as directed by TSC.

D. 1) E-0 > E-1 > ES-1.3
2) Fill Containment sump using Fire Protection system.

# QUESTION 81

**Answer: A**

**Explanation/Justification:**

A. Correct. E-0 step 20 checks auxiliary building rad levels, and transitions to ECA-1.2. ECA-1.2 will attempt to isolate the LOCA, and have the crew transition to ECA-1.1 when the RCS pressure is lowering indicating the leak is not isolated as stated in the question stem. Second part is correct because ECA-1.1 step 38 checks if RWST <30 inches, and step 40 states adds to RCS as directed by TSC.

B. Incorrect. First part is correct. E-0 step 20 checks auxiliary building rad levels, and transitions to ECA-1.2. ECA-1.2 will attempt to isolate the LOCA, and have the crew transition to ECA-1.1 when the RCS pressure is lowering indicating the leak is not isolated as stated in the question stem. Second part is plausible if the candidate is thinking the leak is outside cnmt, and makeup must be added to the cnmt sump as is directed in SACRG-1 step12, fill cnmt sump to 112 inches using fire protection header, but this procedure is not applicable in this scenario.

C. Incorrect. Plausible if the candidate makes the assumption that the conditions warrant transition to E-1 from E-0. This is not correct because E-0 step 16 checks cnmt pressure, level, and radiation for the E-1 transition, none of which are elevated. ES-1.3 would be transferred to from E-1 LHP when RWST level reached 430 inches, but these transitions would not resolve the LOCA outside cnmt or the RWST depletion. Second part is correct because ECA step 38 checks if RWST <30 inches, and step 40 states adds to RCS as directed by TSC.

D. Incorrect. Plausible if the candidate makes the assumption that the conditions warrant transition to E-1 from E-0. This is not correct because E-0 step 16 checks cnmt pressure, level, and radiation for the E-1 transition, none of which are elevated. ES-1.3 would be transferred to from E-1 LHP when RWST level reached 430 inches, but these transitions would not resolve the LOCA outside cnmt or the RWST depletion. Second part is plausible if the candidate is thinking the leak is outside cnmt, and makeup must be added to the cnmt sump as is directed in SACRG-1 step12, fill cnmt sump to 112 inches using fire protection header, but this procedure is not applicable in this scenario.

| Sys # | System | Category | | KA Statement |
|---|---|---|---|---|
| WE11 | Loss of Emergency Coolant Recirculation / 4 | | EA2. Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation) | Facility conditions and selection of appropriate procedures during abnormal and emergency operations. |

**K/A#** EA2.1 **K/A Importance** 4.2 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.E-0 Iss 3 Rev 0 pg 16
2OM-53A.1.ECA-1.2 Iss 3 Rev 0 pg 3
2OM-53A.1.ECA-1.1 Iss 3 Rev 0 pg 26

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.43(b)5

**Objective:**

82. Given the following conditions:

- The plant is operating at 100% power with all systems in NSA.
- A4-8A, Rod Control System Urgent Alarm is LIT.

Which of the following is the cause for the Urgent Alarm **AND** what effect on the Control Rods does it have?

A. Failure of a Logic Cabinet Slave Cycler; Multiple rods drop, and the Rx must be tripped.

B. Failure of a Power Cabinet +24vdc power supply; One rod drops, and rod alignment must be restored within 1 hour.

C. Failure of a Logic Cabinet Slave Cycler; Rods will not drop, rods are trippable and operable.

D. Failure of a Power Cabinet +24vdc power supply; Rods will not drop, affected rods are inoperable and Rx power to must be reduced to <75% RTP within 2 hours.

---

**Answer: C**

**Explanation/Justification:**

A. Incorrect. First part is correct. Second part is plausible if the candidate does not know that an Urgent alarm will block all rod motion, and apply current to the stationary and movable coils to ensure the rods don't drop.

B. Incorrect. Plausible because a loss of a power cabinet +24vdc power supply will give a Rod Control System Non-Urgent Alarm due to there being auctioneered + and – 24vdc power supplies, therefore a loss of one of the supplies will not affect rod control. Second part is plausible if the candidate did not know that the power supplies are auctioneered, and thought that the loss of power would drop the control rod, and TS 3.1.4 cond B would require the rod alignment to be restored within 1 hour..

C. Correct. Logic Cabinet Slave Cycler failure will cause an Inhibit signal for the affected slave cycler since it is the cause of the Urgent Failure alarm, this will block all rod motion in manual and automatic. AOP 2.1.8, part B, step 4 RNO states that if the Rod Control System Urgent Alarm is lit, then the rods are considered trippable. TS 3.1.4 background states that if the rod is trippable, then it is operable.

D. Incorrect. Plausible because a loss of a power cabinet +24vdc power supply will give a Rod Control System Non-Urgent Alarm due to there being auctioneered + and – 24vdc power supplies, therefore a loss of one of the supplies will not affect rod control. Second part is plausible if the candidate did not know that the rods would still be operable due to the auctioneered power supplies, and thought that TS 3.1.4 Cond B was

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 003 | Dropped Control Rod | AK2. Knowledge of the interrelations between the Dropped Control Rod and the following: | Control rod drive power supplies and logic circuits |

**K/A#** AK2.05 **K/A Importance** 2.8 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-1.4.AAB Rev 7 pg 3
2OM-53C.4.2.1.8 Rev 5 pg 6
TS 3.1.4 Bases pg. B3.1.4-4
3SQS-1.3 PPNT Rev 7 Iss 1 slide 173

**Question Source:** Modified - 2005 BVPS Q58

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.43 b(6)

**Objective:**

83. Liquid waste process effluent monitor 2SGC-RQ100 is inoperable.

Under what conditions may a liquid release be initiated?

A. At least two independent samples are analyzed and release calculations and discharge valving independently verified per the ODCM.

B. Monitoring is initiated with the comparable alternate channel per the ODCM.

C. Sample the cooling tower blowdown line every four hours to ensure the release remains within ODCM limits.

D. The release may not be continued until the channel is repaired per Technical Specifications.

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. Per ODCM 3.03 attachment E, with 2SGC-RQ100 inoperable, and no alternate channels available, action 23 must be taken. Action 23 of attachment E states at least two independent samples are analyzed and release calculations and valving verified. The discharge procedure for 2SGC-TK23A/B, 2OM-25.4.L also states the requirements for discharging when 2SGC-RQ100 is inoperable.

**B.** Incorrect. Plausible due to ODCM wording, but no alternate channel exists for 2SGC-RQ100.

**C.** Incorrect. Plausible because this sounds like a reasonable action if the candidate knows that there is not an alternate channel to monitor, or doesn't realize the correct actions are available.

**D.** Incorrect. Plausible if the candidate knows that there is not an alternate channel to monitor, or doesn't realize the correct actions are available. The release may continue.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 000059 | Accidental Liquid Radwaste Release | AK2. Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following: | Radioactive-liquid monitors |

**K/A#** AK2.01 **K/A Importance** 2.8 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 1/2-ODC-3.03 Rev 14 Att. E pg 29-30<br>2OM-25.4.L Rev 34 pgs. 12 & 18<br>2SQS-17.1 PPNT Rev. 9 slide 29

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.43 (b)4

**Objective:**

84. In accordance with NOP-OP-1015, Event Notifications, which of the following events should be reported?

Event 1: A Seismic event was felt by plant personnel. No Seismic alarms were received at Unit 1 or Unit 2.

Event 2: An inadvertent radioactive gaseous release with an effective dose rate of 0.22 rem/hr for 24 hours. After the stated 24 hours, the release is terminated. Assume that the dose is from inhalation.

**(Reference provided)**

A. Event 1 **only**

B. Event 2 **only**

C. Both events

D. Neither event

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Plausible because an earthquake would require notification per NOP-OP-1015 pg 25 if the event was ≥ 0.02g, but the event 1 description states no alarms were received, and the candidate must know that A10-5H alarm setpoint is 0.008-0.06g). It is also plausible because an earthquake would require EPP notification for UE since it was felt by plant personnel, but alarm A10-5H did not come in as required in HU3.

**B.** Correct. Only need to report it if its >1 ALI if a person was standing there for 24 hours. 0.22*24 = 5.28 rem which would make this reportable under FE's event notification procedure to the NRC Operations Center within 24 hours. The other requirements are eye dose equivalent of exceeding 15 rems and a shallow dose to the skin or extremities exceeding 50 rems.

**C.** Incorrect. Plausible if they think that both events require outside notification.

**D.** Incorrect. Plausible if they think that neither event meets the outside notification requirements of NOP-OP-1015.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 060 | Accidental Gaseous Radwaste Release | Generic | Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. |

**K/A#** 2.4.30 **K/A Importance** 4.1 **Exam Level** SRO

**References provided to Candidate** NOP-OP-1015, Event Notifications, rev. 6 **Technical References:** NOP-OP-1015 Rev 6 Att. 2 pgs. 10, 26; 10 CFR 22.2202

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.43 b(4)

**Objective:**

85. Given the following:

- A Steam Generator safety valve has failed open on 'A' SG.
- The crew has terminated SI IAW ES-1.1, SI Termination, and has just established normal charging flow.

Then, a Steam Generator safety valve fails open on 'B' SG. The following plant conditions exist.

- PRZR level is 50%, trending down slowly.
- PRZR pressure is 2220 psig, trending down slowly
- Tc is 350°F, trending down slowly
- RCPs are stopped
- The initial event occurred 40 minutes ago.

What action is required IAW ES-1.1?

A. Reinitiate SI, transition to FR-P.1, Response To Imminent Pressurized Thermal Shock Condition.

B. Reinitiate SI, transition to ECA-2.1, Uncontrolled Depressurization Of All Steam Generators.

C. Transition to E-2, Faulted Steam Generator Isolation, and operate ECCS as directed.

D. Transition to ES-1.2, Post LOCA Cooldown And Depressurization.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Plausible distractor if the candidate thinks that SI reinitiation is required by ES-1.1 LHP. Second part is plausible if the candidate doesn't recognize that FR-P.1 is sat even though the initial cooldown was >100F/hr, based on T-cold at 350F, therefore entry to FR-P.1 is not warranted.

**B.** Incorrect. Plausible distractor if the candidate thinks that SI reinitiation is required by ES-1.1 LHP. Second part is plausible if the candidate thinks that a direct entry is warranted because 2 SG pressures are lowering, but they must enter E-2 to get to ECA-2.1.

**C.** Correct. Per ES-1.1 LHP. Enter E-2 if any SG pressure is dropping in an uncontrolled manner or has completely depressurized, and has not been isolated.

**D.** Incorrect. Plausible distractor if the candidate remembers that RCS pressure is checked early in the SI Termination procedure, and if it is lowering a transition to ES-1.2 is made per step 4 RNO.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| WE02 | SI Termination | EA2. Ability to determine and interpret the following as they apply to the (SI Termination) | Facility conditions and selection of appropriate procedures during abnormal and emergency operations. |

**K/A#** EA2.1 **K/A Importance** 4.2 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.ES-1.1 Iss 3 rev 0 LHP

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.43(b)5

**Objective:**

86. The plant is at 100% power.

- 2RCS-PCV551C, '**C**' PRZR Safety Relief Valve begins **LEAKING**.
- 2RCS-PI472, PRZR Relief Tank Pressure has **RISEN** to 50 psig.
- 2RCS-PCV551A, '**A**' PRZR Safety Relief Valve is found to lift at 2395 psia.

(1) What will be the **STABLE** PRZR Safety Relief line temperature, for these conditions?

(2) Which PRZR Safety Relief valve(s) is(are) **INOPERABLE** IAW LCO 3.4.10, "Pressurizer Safety Valves"?

A. 1) 281°F
2) 'A' PRZR Safety Relief Valve **only**.

B. 1) 281°F
2) **Both** 'A' and 'C' PRZR Safety Relief Valves.

C. 1) 298°F
2) 'A' PRZR Safety Relief Valve **only**.

D. 1) 298°F
2) **Both** 'A' and 'C' PRZR Safety Relief Valves.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Wrong Relief line temperature, this is the relief line temperature for 50 psia NOT 50 psig. Second part is correct because 'A' PRZR Safety Relief Valve lift setpoint is low out of the LCO 3.4.10 band of 2410.5-2524.7 psig.

**B.** Incorrect. Wrong Relief line temperature, this is the relief line temperature for 50 psia NOT 50 psig. Second part is incorrect because a leaking Safety valve is not inoperable.

**C.** Correct. Correct Relief line temperature. Second part is correct because 'A' PRZR Safety Relief Valve lift setpoint is low out of the LCO 3.4.10 band of 2410.5-2524.7 psig. Candidate must know that the leaking Safety valve is not inoperable under TS 3.4.10, and the leakage would be identified under TS 3.4.13 Operational Leakage.

**D.** Incorrect. Correct Relief line temperature. Second part is incorrect because a leaking Safety valve is not inoperable.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 007 | Pressurizer Relief Tank/Quench Tank System (PRTS) | A2 Ability to (a) predict the impacts of the following malfunctions or operations on the P S; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: | Stuck-open PORV or code safety |

**K/A#** A2.01 **K/A Importance** 4.2 **Exam Level** SRO

**References provided to Candidate** Steam Tables **Technical References:** Steam Tables; TS page 3.4.10-1

**Question Source:** Modified - 1LOT14 Q34

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.43 b(1)

**Objective:**

87. A fire in Benchboard 'A' has caused an inadvertent SI signal, and has forced the evacuation of the control room due to smoke.

- The reactor failed to trip manually from the control room.
- The crew was able to reduce power by driving rods.
- The reactor was tripped at 7% power by locally opening trip breakers.
- The control room was evacuated one minute after the trip breakers were opened and 5 minutes after the initial report of the fire.

19 minutes **after** the initial report of the fire, the fire is out and plant control has been established at the Alternate Shutdown Panel.

What is the event classification?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Plausible distractor because the fire was not extinguished within 15 minutes, but this is a lower classification that the correct answer.

**B.** Incorrect. Plausible distractor because CR evacuation with control of safety functions from an alternate location within 15 minutes is an Alert based on HA2. The stem states control from the ASP was completed in 14 minutes from the time of the CR evacuation..

**C.** Correct. Based on SS3. The inadvertent SI failed to automatically trip the Rx, AND manual trip actions taken from the CR bench boards failed to shutdown the Rx.

**D.** Incorrect. Plausible distractor because it is one of the four event classifications available.

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 012 | Reactor Protection System | Generic | Knowledge of the emergency action level thresholds and classifications. |

**K/A#** 2.4.41 **K/A Importance** 4.6 **Exam Level** SRO

**References provided to Candidate** EAL matrix **Technical References:** EPP-I-1b Rev 17 pg 18
EPP Plan Sect 4, Rev 30 pgs. 254-256

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.43 (b)5

**Objective:**

88. The plant is in Mode 5 preparing to enter Mode 4.

- Maintenance is performing 2MSP-13.01-I, 2QSS-L104A, Refueling Water Storage Tank 2QSS-TK21 Level Loop Channel I Test.
- All bistables associated with this RWST level channel have been placed in their Tech Spec required condition.
- Maintenance reports the as found setpoint for:

  2QSS-LSEL104A RWST Ext-Lo Level SI Switchover Comparator Trip is 29' 6" (Tech Spec Allowable Value is between 31' 8" and 31' 10")

  **AND**

  2QSS-LSL104A Recirc Spray Pump Start Interlock Comparator Trip is 32' 9" (Tech Spec Allowable Value is between 32' 8" and 32' 10")

1) What is the status of the following Tech Spec REQUIRED Engineered Safety Feature Actuation System (ESFAS) Instrumentation? (Based on the as found values)
   - RWST level Extreme low SI Switchover
   - RWST level low Recirc Spray Pump Start Interlock

2) Assuming all other requirements for Mode 4 entry have been met, what additional actions, if any, would be REQUIRED to enter Mode 4? (Assume the as found setpoints will remain as is)

1) This channel of RWST________________________________________.

2) Mode 4 entry is allowed________________________________________.

**(Reference provided)**

A. 1) level Extreme low AND level low are BOTH still OPERABLE
   2) with no additional actions required

B. 1) level Extreme low is INOPERABLE AND level low is still OPERABLE
   2) ONLY if an additional risk assessment is performed

C. 1) level Extreme low is INOPERABLE AND level low is still OPERABLE
   2) with no additional actions required

D. 1) level Extreme low AND level low are BOTH INOPERABLE
   2) ONLY if an additional risk assessment is performed

# QUESTION 88

**Answer: C**

**Explanation/Justification:**

- **A.** Incorrect. Extreme level low is inoperable since it is outside the allowable band on the low side. If it was outside on the high side, it could still be operable dependent on the outcome of an evaluation to determine if it could still perform its function. Since the bistables are already tripped, no additional actions would be required to enter Mode 4 IAW TS 3.0.4. since the action statements allow continued operation for an unlimited period of time.
- **B.** Incorrect. Correct operability determination. However, since the bistables are already tripped, no additional actions would be required to enter Mode 4 IAW TS 3.0.4 since the action statements allow continued operation for an unlimited period of time.
- **C.** Correct. IAW TS page 3.3.2-8 item 2.b.2.and page 3.3.2-12 item 7.b and TS 3.0.4
- **D.** Incorrect. Extreme level low is inoperable since it is outside the allowable band on the low side. If it was outside on the high side, it could still be operable dependent on the outcome of an evaluation to determine if it could still perform its function. Since the bistables are already tripped, no additional actions would be required to enter Mode 4 IAW TS 3.0.4 since the action statements allow continued operation for an unlimited period of time.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 013 | Engineered Safety Features Actuation System (ESFAS) | Generic | Ability to determine operability and/or availability of safety related equipment. |

**K/A#** 2.2.37 **K/A Importance** 4.6 **Exam Level** SRO

**References provided to Candidate** TS 3.3.2 pgs. 1-13 **Technical References:** TS page 3.3.2-8 item 2.b.2 and page 3.3.2-12 item 7.b and TS 3.0.4

**Question Source:** Bank - 2LOT8 Q88

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.43 (b)2

**Objective:**

89. Given the following conditions:

- The plant is at 100% power.
- 4KV bus 2AE has tripped due to a fault on the bus.

Which is the **most** limiting applicable Technical Specification required action?

A. Restore the inoperable bus within 2 hours.

B. Restore the inoperable bus within 8 hours.

C. Lineup the 2CHS-P22B Boric Acid Transfer Pump within 2 hours.

D. Place 2RCS-PCV455D PORV in manual within 1 hour.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Plausible because this is the completion time for a loss of vital bus IAW TS 3.8.9 cond. B, but it is not the most limiting

**B.** Incorrect. Plausible because this is the completion time for a loss of the emergency bus IAW TS 3.8.9 cond. A, but it is not the most limiting

**C.** Incorrect. Plausible because a Boric Acid Transfer pump is required to be functional, but 72 hours is permitted IAW LRM 3.1.6.

**D.** Correct. The loss of the 2AE bus causes a loss of power to 2RCS-MOV537 (E05) which makes the block valve inoperable. TS 3.4.11 cond. C requires that for an inoperable PORV block valve the PORV must be placed in manual control within 1 hour.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 062 | A.C. Electrical Distribution | Generic | Knowledge of limiting conditions for operations and safety limits. |

**K/A#** 2.2.22 **K/A Importance** 4.7 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** TS 3.4.11 Cond. C
T.S. 3.8.9

**Question Source:** New

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** 55.43 (b)2

**Objective:**

90. The plant is operating at 100% power with all systems in NSA.
    - SG Feed Pump 21B Recirculation Valve 2FWR-FCV150B inadvertently fails full OPEN.

    Assuming recirculation flow is within the control capacity of the MFRVs, answer the following:

    1) With NO operator action, what impact will this failure have on the Steam Generator Water Level Control system?

    2) IAW AOP-2.24.1, Loss of Main Feedwater, what actions will be required in response to this failure?

A. 1) SG levels will decrease continuously.
   2) Start the standby condensate pump and reduce power to 80%

B. 1) SG levels will initially decrease, then return to program level.
   2) Place the keylock switch for 2FWR-FCV150B to CLOSE.

C. 1) SG levels will decrease continuously.
   2) Start all AFW pumps and reduce power to 52%

D. 1) SG levels will initially decrease, then stabilize below program level.
   2) Reduce power to 80%, then place the keylock switch for 2FWR-FCV150B to CLOSE.

---

**Answer: B**

**Explanation/Justification:**

- **A.** Incorrect. Plausible if applicant does not know flow mismatch and level mismatch error signals. 80% is a power level in the procedure requiring trip
- **B.** Correct. Failure introduces a flow error, SGWLC will initially increase feed flow to match steam flow and stabilize level, level will then be corrected by integral level error raising level back to program. Correct action per procedure.
- **C.** Incorrect. Plausible is applicant does not understand separate flow and level error.
- **D.** Incorrect. Level will return to normal, no downpower required.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 059 | Main Feedwater (MFW) System | A2 Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: | Failure of feedwater control system |

**K/A#** A2.11 **K/A Importance** 3.3 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** AOP-2.24.1 rev 7 step 7 RNO
2SQS-24.1 Rev 26 LP page 19
2SQS-24.1 Rev 26 PPNT slide 55

**Question Source:** Modified - 2LOT6 Q45

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.43 b(5)

**Objective:**

91. Given the following:

- The Plant is operating at 100% power.
- Then, the lowest (bottom) Data 'A' coil fails (open circuit) for a control rod.

Which of the following describes the effect of this failed coil on the Digital Rod Position Indication (DRPI) System, and what actions would be required?

A. Rod bottom indication will be lost in the event of a reactor trip. Boration will be required due to the rod failing to indicate full insertion.

B. DRPI will switch to the half-accuracy mode. A General Warning Status light will flash for the affected rod. Enter AOP 2.1.7, RPI Malfunction.

C. DRPI will switch to the half-accuracy mode. Enter AOP 2.1.7, RPI Malfunction and T.S. 3.1.4, Rod Group Alignment Limits, due to indicated rod misalignment.

D. All indication for the effected rod will be lost. A General Warning Status light will flash for the affected rod. Enter T.S. 3.1.7.2 for a failed RPI.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Plausible if the candidate thinks that rod bottom indication would be lost, and the step in ES-0.1 states to emergency borate if only one rod bottom light is not lit. This would not be the case due to Data 'B' coils indicating the rod on bottom due to "half accuracy", and ES-0.1 requires two rod bottom lights not being lit to emergency borate.

**B.** Correct. When 'A' coil fails, the 'B' coils will indicate with an accuracy of +10/-4 in "half accuracy" mode, and a rod bottom light will be present. A General warning light will be lit for the affected rod due to data failure for the rod. AOP 2.1.7 would be entered to determine the extent of the DRPI system failure.

**C.** Incorrect. Plausible because DRPI will switch to "half accuracy" mode, and AOP 2.1.7 would be entered, but TS 3.1.4 Cond B for rod misalignment would not be entered due to the 'B' coils indicating with an accuracy of +10/-4 in "half accuracy" mode, therefore rod misalignment would be <12 steps.

**D.** Incorrect. Plausible if the candidate thinks that one coil in the Data 'A' cabinet would make all indication lost, but this is not the case with Data 'B' available. General warning light will be flashing. TS 3.1.7.2 would not be entered because there is no indication in the stem that Data 'B' is not working properly.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 014 | Rod Position Indication System (RPIS) | A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: | Loss of LVDT |

**K/A#** A2.06 **K/A Importance** 3.0 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** TS 3.1.7.2 bases pg B.3.1.7.2-2
2OM-53C.4.2.1.7 Rev 11
2SQS-1.4, Rev. 9 (RPI systems)

**Question Source:** Bank - Modified distractors

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.43 (b)1,5

**Objective:** 2SQS-1.4, Rev. 9 Obj. 11 - Explain the DRPI system accuracy in the normal and half accuracy modes of operation, and the indications that DRPI is operating at half accuracy.

92. Attached is a recent surveillance data sheet performed to evaluate CETs operability IAW 2OST-6.7, Accident Monitoring Instrumentation Channel Checks.

According to the data sheet, which condition(s), if any, are required to be entered IAW TS 3.3.3, "PAM Instrumentation"?

**(Reference provided)**

A. Sufficient operable CETs, no condition is entered

B. A only

C. C only

D. A & C only

---

**Answer: D**

**Explanation/Justification:** CETs identified on Data Sheet Train A I-4, II-1, III-6, IV-6 Train B I-6, II-1, III-4, IV-5

**A.** Incorrect. Plausible distractor if the candidate thinks that 2 CETs in quadrant II meets function 17b channel requirement, but a channel consists of 2 CETs per channel based on note c.

**B.** Incorrect. Plausible distractor if the candidate thinks that a separate entry for 1 channel on Train A & 1 channel on Train B for function 17b, but a channel consists of 2 CETs per channel based on note c. This is not a correct application of the TS.

**C.** Incorrect. Plausible because there are 2 channels inoperable in function 17b, but Condition A must also be entered due to it being met.

**D.** Correct. At least 2 core exit thermocouples per quadrant for Train A are required, AND, 2 core exit thermocouples per quadrant for Train B are required to be OPERABLE by SR 3.3.3.1(17 a, b, c, d). 2OST-6.7 Data Sheet 1 Identifies Train A - quadrant II, with only 1 CET, and Train B - quadrant II with only 1 CET. Based on this combination, both condition A & C would be entered.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 017 | In-Core Temperature Monitor System (ITM) | Generic | Knowledge of limiting conditions for operations and safety limits. |

**K/A#** 2.2.22 **K/A Importance** 4.7 **Exam Level** SRO

**References provided to Candidate** T.S 3.3.3 pgs. 1-4
Modified 2OST-6.7 pg 30

**Technical References:** TS 3.3.3 & TS 3.3.3 Bases pg. B3.3.3-12
2OST-6.7 Rev 21

**Question Source:** New

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.43 b(2)

**Objective:**

Q92



## DATA SHEET 1

| CORE EXIT THERMOCOUPLE DATA | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| TRAIN A | | | | | TRAIN B | | | | |
| Mark Number | Location | Quadrant | Data (°F) | Review | Mark Number | Location | Quadrant | Data (°F) | Review |
| TE01E | A8 | I | 573 | ✓ | TE03E | B10 | II | 585 | ✓ |
| TE02E | B5 | I | | 601105299 | TE06E | E12 | II | * | * |
| TE04E | E4 | I | * | * | TE07E | E14 | II | | 601105300 |
| TE05E | E7 | I | 630 | ✓ | TE08E | F3 | I | 621 | ✓ |
| TE10E | F9 | II | 627 | ✓ | TE09E | F5 | I | 623 | ✓ |
| TE11E | F11 | II | | 601005854 | TE12E | G1 | I | 564 | ✓ |
| TE13E | G6 | I | 626 | ✓ | TE17E | J10 | III | | 600896681 |
| TE14E | H8 | IV | 626 | ✓ | TE18E | J12 | III | 624 | ✓ |
| TE15E | H15 | III | 574 | ✓ | TE19E | K3 | IV | * | * |
| TE16E | J2 | IV | * | * | TE27E | C8 | I | 621 | ✓ |
| TE20E | K5 | IV | 624 | ✓ | TE29E | D3 | I | | 600816642 |
| TE21E | K8 | III | 631 | ✓ | TE30E | D5 | I | 628 | ✓ |
| TE22E | K11 | III | 627 | ✓ | TE31E | E8 | II | 32 | |
| TE23E | M9 | III | | 601105301 | TE32E | E10 | II | | 601927516 |
| TE24E | N6 | IV | 624 | ✓ | TE35E | G8 | I | 631 | ✓ |
| TE25E | P8 | III | 624 | ✓ | TE36E | G15 | II | * | * |
| TE26E | R7 | IV | 562 | ✓ | TE37E | H3 | IV | 619 | ✓ |
| TE28E | C12 | II | * | * | TE38E | H5 | IV | 622 | ✓ |
| TE33E | F13 | II | | 601005855 | TE42E | L6 | IV | 625 | ✓ |
| TE34E | G2 | I | 609 | ✓ | TE43E | L8 | III | 624 | ✓ |
| TE39E | H9 | III | 630 | ✓ | TE44E | L12 | III | 630 | ✓ |
| TE40E | H11 | II | * | * | TE45E | L14 | III | 574 | ✓ |
| TE41E | H13 | II | 32 | | TE46E | M3 | IV | 563 | ✓ |
| TE47E | M11 | III | 629 | ✓ | TE50E | N10 | III | * | * |
| TE48E | N4 | IV | 567 | ✓ | TE51E | P7 | IV | 612 | ✓ |
| TE49E | N8 | IV | 620 | ✓ | | | | | |
| A1 | REF JUNCTION | NA | 84 °F | | B1 | REF JUNCTION | NA | 84 °F | |
| A2 | REF JUNCTION | NA | 84 °F | | B2 | REF JUNCTION | NA | 85 °F | |
| A3 | REF JUNCTION | NA | 84 °F | | B3 | REF JUNCTION | NA | 84 °F | |

*Thermocouple has been capped by ECP 14-0339 and is **NOT** available.

TC / Today
**Initial / Date**

93. A fire in the cable spreading room (CB-2) has forced evacuation of the control room.

- All required control room actions were completed before the evacuation.
- As a result of the fire, a spurious SI has occurred and one PORV has failed open.

Which of the answer choices below lists the time critical actions in their correct priority in accordance with 2OM-56C.4.B?

1.) Close the PORV
2.) Reset SI, CIA, and CIB
3.) Isolate High Head Injection
4.) Start and Load DG 2-1 from the ASP
5.) Start and Load DG 2-2 from the ASP
6.) Restore AFW flow
7.) Restore Charging

A. 1, 5, 3, 7

B. 2, 1, 6, 3

C. 4, 3, 5, 1

D. 4, 3, 1, 7

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. . Plausible because the candidate may think the first priority is to close the PORV (20 min), but this is not correct. Starting and loading EDG 2-2 sounds like a time critical action, but 56C does not credit EDG 2-2 (train B), therefore it is not a critical action. #3 Isolating HHSI on an inadvertent SI (19 min) and #7 restoring charging (30 min ) are both time critical actions.

**B.** Incorrect. . Plausible because (#2) resetting SI, CIA, and CIB sounds logical for an inadvertent SI, but it is not a time critical action, and 56C de-energizes required valves and manually manipulates the system flowpaths. 1, 6, and 3 are time critical actions.

**C.** Incorrect. Plausible because (#4) starting EDG 2-1 is a high priority (10 min), and (#3) Isolating HHSI on an inadvertent SI (19 min) is important, as well as (#1) closing the PORV (20 min). But, 56C does not credit EDG 2-2 (train B), therefore it is not a critical action.

**D.** Correct. (#4) starting EDG 2-1 is a high priority (10 min) as this train A EDG is needed for alternate safe shutdown of the plant, and (#3) Isolating HHSI on an inadvertent SI (19 min) is also required, (#1) close the PORV (20 min), and (#7) restoring charging (30 min ) are critical actions in the correct priority. This answer does not include all required TCAs, but the ones listed are in order per 2OM-56C.4.B.

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 086 | Fire Protection System (FPS) | Generic | Knowledge of abnormal condition procedures. |

**K/A#** 2.4.11 **K/A Importance** 4.2 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-56.C.4.B Rev 35 pg 3

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.43 (b)5

**Objective:** 2SQS-56C.1, Rev. 7 Obj. 7 - Given a set of plant conditions and the appropriate procedure(s), apply the operational sequence, parameter limits, precaution and limitations, and cautions & notes applicable to the completion of the task activities in the Control Room.

Obj. 9 - Describe the design basis for Alternate Safe Shutdown and the associated major components as documented in the UFSAR.

94. In accordance with 1/2 RP-1.1, Refueling Administrative Section, which position(s) has(have) the authority to bypass interlocks on fuel handling equipment during fuel handling emergency conditions?

A. Fuel handling equipment interlocks may NEVER be defeated

B. Refueling SRO ONLY

C. Shift Manager ONLY

D. EITHER Refueling SRO OR Shift Manager

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**Answer: B**

**Explanation/Justification:**

A. Incorrect. Plausible because interlocks are designed to prevent damage to personnel or equipment. An applicant may think that these interlocks must never be defeated.
**B.** Correct. Also allowed if stated in the procedure.
**C.** Incorrect. Plausible because the SM is responsible for overall supervision of plant operations, but authority to bypass interlocks is a Refueling SRO responsibility per P&L 6 of 1/2RP-1.1.
**D.** Incorrect. Plausible because Refueling SRO is correct, but the authority to bypass interlocks is not a SM responsibility.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of the fuel-handling responsibilities of SROs. |

**K/A#** 2.1.35 **K/A Importance** 3.9 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 1/2RP-1.1 Rev 29 pg 16
3SQS-6.13 Rev 6 PPNT slide 145

**Question Source:** New

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.43

**Objective:**

95. Initial conditions:

- The Plant is operating at 100% steady state power.
- All Primary and Secondary plant chemistry parameters are within Technical Specification/LRM limits.

The shift chemist reports the following **STABLE** Primary and Secondary plant chemistry conditions, based on the **LATEST** sample:

- Secondary Specific Activity is 0.025 μCi/gm DOSE EQUIVALENT I-131
- RCS Specific Activity is 25.0 μCi/gm DOSE EQUIVALENT I-131
- RCS Dissolved Oxygen is 0.15 ppm
- RCS Chlorides are 0.10 ppm
- RCS Fluorides are 0.10 ppm

Based on these chemistry conditions, what Technical Specification/LRM actions are **REQUIRED** at this time?

Within ______________________

**(Reference provided)**

A. 6 hours be in Mode 3 and within 36 hours be in Mode 5.

B. 6 hours be in Mode 3 with $T_{avg}$ less than 500°F.

C. 24 hours restore DOSE EQUIVALENT I-131 to within its limit.

D. 48 hours restore RCS Dissolved Oxygen to within the steady state limit.

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**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Plausible distractor if the candidate thinks that the secondary specific activity has been exceeded IAW TS 3.7.13, as these are the actions required. This is not the case since given activity is 0.025 μCi/gm, and the limit is ≤0.10 μCi/gm.

**B.** Correct. DEI is 25 μCi/gm, therefore TS 3.4.16 condition A is entered, and figure 3.4.16-1 will be referenced. Acceptable range for 100% power is less than 20 μCi/gm, therefore DEI is in the unacceptable region, and condition C must be entered. IAW TS 3.4.16 condition C.1, the plant must be in Mode 3 with Tavg <500F within 6 hours.

**C.** Incorrect. Plausible distractor if the candidate does not recognize that DEI is in the unacceptable range IAW TS 3.4.16, but the required time is 48 hrs, not 24 hrs.

**D.** Incorrect. Plausible distractor if the candidate recognizes that Dissolved Oxygen is greater than the steady state limit IAW LR 3.4.2, but the required time is 24 hrs, not 48 hrs.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of primary and secondary plant chemistry limits |

**K/A#** 2.1.34 **K/A Importance** 3.5 **Exam Level** SRO

**References provided to Candidate** TS 3.4.16 pg 1-3<br>TS 3.7.13 pg 1<br>LRM 3.4.2 pg 1-3

**Technical References:** TS 3.4.16 condition C.1 and TS figure 3.4.16-1.

**Question Source:** Bank - BV 2LOT6 Q95

**Question Cognitive Level:** Comprehension **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.12)

**Objective:**

96. The plant is in Mode 2 during a reactor startup.

- 0800 on Monday, the CR receives a report 2 **battery chargers** on **Train B** are inoperable.
- 1000 on Monday, SR 3.8.4.1 and SR 3.8.6.1 were completed UNSAT.
- 1100 on Monday, a **battery** on **Train A** is found to be inoperable.

**IF** no repairs are made, what is the latest time the plant is REQUIRED to be in Mode 3?

**(Reference provided)**

A. 1600 on Monday

B. 1800 on Monday

C. 1700 on Tuesday

D. 0800 on Thursday

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. At 0800 TS 3.8.4 cond A is entered which requires terminal voltage to be restored ≥ float voltage within 2 hrs. 1000 surveillance results confirm Cond A is not met, therefore since no repairs are made, TS 3.8.4 Cond D must be entered at 1000, and Mode 3 by 1600 hrs.

**B.** Incorrect. Plausible because LCO 3.0.3 will be entered at 1100 due to 3 batteries being inoperable iaw TS 3.8.4. This requires mode 3 within 7 hours. Train B batteries were declared inoperable at 0900 when the SRs were completed unsat based on TS 3.8.6 cond F.

**C.** Incorrect. Plausible if the candidate thinks the Train B chargers being OOS will require TS 3.8.6 cond A not being met and the battery float voltage not being restored within 24 hrs

**D.** Incorrect. Plausible if the candidate thinks the chargers have 72 hours to restore iaw TS 3.8.4 cond A, and either doesn't recognize that the Train B batteries are inop, or that terminal voltage is not ≥ float voltage.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. |

**K/A#** 2.2.36 **K/A Importance** 4.2 **Exam Level** SRO

**References provided to Candidate** TS 3.8.4 pgs. 1-2, TS 3.8.6 pgs. 1-3 **Technical References:** TS 3.8.4

**Question Source:** New

**Question Cognitive Level:** Comprehension or Analysis **10 CFR Part 55 Content:** 55.43 b(2)

**Objective:**

97. The plant is operating at 100% power. The Unit Supervisor discovers that a monthly surveillance has not been performed for 35 days.

    What action is required?

A. Plant shutdown must be commenced IAW LCO 3.0.3.

B. The surveillance must be completed with 24 hours.

C. The surveillance must be completed within 1.25 times the surveillance frequency from the previous completion time.

D. The surveillance must be completed within 0.25 times the surveillance frequency from the time of discovery.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Plausible if the candidate thinks that the LCO is not met due to the missed surveillance and LCO 3.0.3 is required to be entered to commence plant shutdown.

**B.** Incorrect. Plausible because SR 3.0.3 states if it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. In this case the 1.25 times 31 days is greater.

**C.** Correct. Per SR 3.0.2 and 3.0.3, the surveillance must be completed within 1.25x the surveillance frequency of 31 days. This is the greater of 24 hours, or the 1.25 times the limit of the specified Frequency (SR 3.0.2).

**D.** Incorrect. Plausible if the candidate thinks the extension of SR 3.0.2 is .25 times the surveillance frequency from the time of discovery. This incorrect because SR3.0.2 requires the surveillance be completed in 1.25 times the surveillance frequency from time of previous completion.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of surveillance procedures. |

**K/A#** 2.2.12 **K/A Importance** 4.1 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** T.S. SR 3.02 and 3.03

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** 55.43 (b)1

**Objective:**

98. A 10 Rad/hr diving operation in the Spent Fuel Pool is planned to commence later in the shift.

Which of the following completes the statements below?

The dive ______(1)______ the requirements of NOP-OP-4010, "Determination of Radiological Risk," to be classified as an ORANGE risk activity.

Based on the radiation level in the area of the dive, in accordance with NOP-OP-4107, "Radiation Work Permit (RWP)," the RWP would be required to be approved by ____(2)____.

**(References provided)**

| | (1) | (2) |
|---|---|---|
| A. | meets | Radiation Protection Manager |
| B. | meets | Site Vice President |
| C. | does NOT meet | Radiation Protection Manager |
| D. | does NOT meet | Site Vice President |

**Answer: A**

**Explanation/Justification:**

- **A.** Correct. It meets orange activity for diving activities near irradiated components in excess of 1000 mrem/hr (NOP-OP-4010 att. 1, activity 6). For working in areas in excess of 2.5 rem/hr. Site Radiation Protection Manager approval is required per NOP-OP-4107 pg. 26.
- **B.** Incorrect. First part is correct. Second part is plausible because the Site VP will approve emergency exposures but this does not fit that criteria.
- **C.** Incorrect. First part is plausible because normal dive activities in the Fuel Pool are yellow risk per NOP-OP-4010 att. 1, activity 17, but the stem specifically states it is a 10 R/hr dive. Second part is correct.
- **D.** Incorrect. First part is plausible because normal dive activities in the Fuel Pool are yellow risk per NOP-OP-4010 att. 1, activity 17, but the stem specifically states it is a 10 R/hr dive. Second part is plausible because the Site VP will approve emergency exposures but this does not fit that criteria.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Ability to comply with radiation work permit requirements during normal or abnormal conditions. |

**K/A#** 2.3.7 **K/A Importance** 3.6 **Exam Level** SRO

**References provided to Candidate** NOP-OP-4107 rev 16
NOP-OP-4010 rev. 8

**Technical References:** NOP-OP-4107 rev. 16 page 26 and 29
NOP-OP-4010 rev. 8 page 10

**Question Source:** Modified - 2013 Sequoyah Q98

**Question Cognitive Level:** Memory or Fundamental **10 CFR Part 55 Content:** 55.43 b(7)

**Objective:**

99. Given the following conditions:
    - A general emergency is in progress.
    - A site evacuation has just been ordered due to high airborne radiation levels on site.

    Where do personnel in the Operations Support Center (OSC) go?

A. All personnel remain in the OSC.

B. Personnel relocate to the Alternate OSC.

C. Personnel relocate to the TSC.

D. Personnel relocate to a designated off-site muster area.

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**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Plausible because the Operations Support Center (OSC) is located in the Outage Central Complex above the BV 1 and 2 Control Rooms, but this area is not protected against radiological accidents.

**B.** Correct. The BV 1/2 Alternate Operations Support Center (OSC) is in the Process Instrumentation and Rod Position Instrumentation Area located below the BV-1 Control. This area is supplied by CR Emergency Ventilation System which contains HEPA and charcoal filters in the event of an emergency.

**C.** Incorrect. Plausible because the TSC is located inside the dedicated ERF with filtered ventilation, but this area would not support a rapid response to mitigate site damage.

**D.** Incorrect. Plausible because in the event of the site being inaccessible, OSC personnel would muster in the front of the EOF, but the question does not infer that the site is inaccessible, therefore personnel with EP assignments remain on site.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of emergency response facilities. |

**K/A#** 2.4.42 **K/A Importance** 3.8 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** EPP-PLAN-SECTION-7 Rev 30 sect. 7.1.2
1/2-EPP-IP-1.5 Rev 23 pg. 6

**Question Source:** New

**Question Cognitive Level:** Memory **10 CFR Part 55 Content:** (CFR: 41.10 / 45.11)

**Objective:**

100 Given the following plant conditions:

- The STA informs you of the following Critical Safety Function (CSF) Status Tree information:
  - All Narrow Range S/G Water levels are 5% and LOWERING.
  - Total available Feedwater Flow is Zero (0) GPM.
  - All Core Exit Thermocouples are 750°F and slowly RISING.
  - RVLIS Full Range is 55% and slowly DROPPING.
  - No RCPs are currently running.

Which procedure transition is immediately required **AND** why?

A. FR-C.1, "Response to Inadequate Core Cooling" due to Extreme Challenge to Clad/Matrix Barrier.

B. FR-C.1, "Response to Inadequate Core Cooling" due to Severe Challenge to Vessel/Containment Barrier.

C. FR-H.1, "Response to Loss of Secondary Heat Sink" due to Extreme Challenge to Clad/Matrix Barrier.

D. FR-H.1, "Response to Loss of Secondary Heat Sink" due to Severe Challenge to Vessel/Containment Barrier.

## QUESTION 100

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Although core cooling is a higher priority in terms of sequence, the red path will always trump an orange path condition according to EOP users guide. Incorrect reason why due to severe versus extreme and incorrect barriers challenged. Red path conditions are not met until RVLIS < 40%.

**B.** Incorrect. Incorrect in that an orange path is a lower priority than a red path. Based on stated plant conditions an orange core cooling path is met only. Correct reason why.

**C.** Correct. In accordance with 1/2OM-53B.2, even though Core Cooling is a higher priority than Heat Sink, the first red path encountered must be entered, FR-H.1 Bases states that a red path on heat sink is an extreme challenge to clad/matrix barrier and immediate operator attention is warranted, With NR S/G water levels < 12% and available total feedwater flow @ 0 GPM a Red Heat Sink Path exists, SRO is responsible for prioritizing and selecting appropriate procedure.

**D.** Incorrect. Correct procedure, the challenge to FR-H.1 is extreme versus severe since a red versus orange path exists. Also the challenge is to the clad versus vessel/containment.

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| N/A | N/A | Generic | Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. |

**K/A#** 2.4.22 **K/A Importance** 4.4 **Exam Level** SRO

**References provided to Candidate** None

**Technical References:**
1/2OM-53B.2 Rev 9 pg 9
2OM-53A.1.F-0.3 Iss 3 rev 0
2OM-53B.4.F-0.3 Iss 3 rev 0
2OM-53A.1.F-0.2 Iss 3 rev 0
2OM-53B.4.F-0.2 Iss 3 rev 0

**Question Source:** Bank - 2LOT7 Q99

**Question Cognitive Level:** Comprehension or Analysis

**10 CFR Part 55 Content:** 55.43 b(5)

**Objective:** 3SQS-53.1, Rev. 2 Obj. 2 - Concerning critical safety function restoration, IAW BVPS EOP Executive Volume, state from memory the following: a. The CFS in the order of priority. b. The priorities of the color-coded end points of the CSF status trees. c. The red path summary conditions from the EOPs.

3SQS-53.3, Rev. 5 obj. 8 - For a given event, apply the Critical Safety Function status form to advise the operating crew of CSF priorities