1. Given the following plant conditions:

- A Reactor trip from 100% power occurred.
- All systems respond normally to actuation signals.
- E-0, "Reactor Trip or Safety Injection", Step 4 is being implemented.
- Containment pressure is 0.6 psig and stable.
- Pressurizer (PRZR) Level is 24% and slowly lowering.
- RCS pressure is 2000 psig and lowering.
- Control Rods J-13 **AND** H-8 are indicating 36 steps.

Which of the following actions is procedurally **REQUIRED** to be taken in order to ensure sufficient shutdown margin is maintained?

A. Transition to ES-0.1, "Reactor Trip Response", AND initiate normal boration for two stuck rods.

B. Transition to ES-0.1, "Reactor Trip Response", AND initiate emergency boration for two stuck rods.

C. Continue with E-0, "Reactor Trip or Safety Injection" AND initiate Safety Injection due to low PRZR Pressure **AND** two stuck rods.

D. Continue with E-0, "Reactor Trip or Safety Injection" AND initiate Safety Injection due to low PRZR Level **AND** two stuck rods.

Answer: B

Explanation/Justification: K/A is met with the knowledge that to maintain shutdown margin after the reactor has tripped with two or more control rods not fully inserted, the Reactor Operator must emergency borate to ensure adequate shutdown margin.

A. Incorrect. It is true that a transition to ES-0.1 is made after completion of IOA's, however, emergency as opposed to normal boration is procedurally required to ensure adequate shutdown margin is maintained.
B. Correct. Plant conditions do not support safety injection initiation criteria so therefore a transition to ES-0.1 is warranted. ES-0.1 requires emergency boration for more than one stuck control rod to ensure adequate shutdown margin is maintained iaw ES-0.1 major action step 1.
C. Incorrect. The additional boron from safety Injection is not required for these plant conditions to maintain adequate S/D Margin. Safety Injection not required unless pressurizer pressure is < 1860 psig nor is it required for two stuck rods. 2000 psig and dropping is a normal plant response following reactor trip.
D. Incorrect. The additional boron from safety Injection is not required for these plant conditions to maintain adequate S/D Margin. Safety Injection is not required based on PRZR Level nor is it required for the two stuck rods. PRZR level trending to 20% is a normal plant response on a reactor trip.

Sys #	System	Category	KA Statement
000007	Reactor Trip, Stabilization, Recovery / 1	EK1 Knowledge of the operational implications of the following concepts as they apply to the reactor trip:	Shutdown margin

K/A# EK1.02 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.E-0, Iss. 3 Rev. 3 pg. 5
1OM-53A.1.ES-0.1, Iss. 3 Rev. 0 pg. 8
1OM-53B.4.ES-0.1, Iss. 3 Rev. 0 pg. 2

Question Source: Bank – 2LOT7 Q1

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

Objective: 3SQS-53.3, Rev. 5 Obj. 6. Given a set of conditions, locate and apply the proper Emergency Operating Procedures, IAW BVPS-EOP Executive Volume.

2. Initial Conditions:
   - The plant is at 100% and 2235 psig.
   - The PRT is at 3 psig.

   Current Conditions:
   - Annunciator A4-5, PRESSURIZER POWER OPER RELIEF VALVE OPEN is LIT
   - The PRT is 10 psig and rising
   - Reactor Pressure is 2190 psig and lowering

   1) The leaking valve is confirmed by the associated downstream temperature corresponding to approximately _____(1)_____.

   2) Based on the above conditions, per ARP A4-5, the crew is required to _____(2)_____.

A. 1) the saturation pressure of the Pressurizer
   2) trip the reactor and enter E-0, "Reactor Trip or Safety Injection

B. 1) the saturation pressure of the Pressurizer
   2) close the associated PORV block valve

C. 1) the saturation pressure of the PRT
   2) trip the reactor and enter E-0, "Reactor Trip or Safety Injection

D. 1) the saturation pressure of the PRT
   2) close the associated PORV block valve

Answer: D

Explanation/Justification: K/A is met with the candidate's knowledge that when discharging the PRZR vapor space to the PRT, the temperature at the outlet of a throttled valve (PORV) will be dependent on the saturation pressure of the PRT.

A. Incorrect. Plausible if the candidate believes that the temperature of the steam in the Pressurizer is the same temperature as the steam entering the PRT. This was the error made at the TMI accident. Tripping the reactor is plausible because the ARP requires the reactor trip if the PORV cannot be isolated and RCS pressure is < 2000 psig and lowering.

B. Incorrect. Plausible if the candidate believes that the temperature of the steam in the Pressurizer is the same temperature as the steam entering the PRT. This was the error made at the TMI accident. The ARP directs the closing of the associated block valve to isolate the leaking PORV.

C. Incorrect. The enthalpy of the saturated fluid in the vapor space does not change as it passes through a safety valve, resulting in a temperature indication corresponding to the pressure in the PRT. Tripping the reactor is plausible because the ARP requires the reactor trip if the PORV cannot be isolated and RCS pressure is < 2000 psig and lowering.

D. Correct. The enthalpy of the saturated fluid in the vapor space does not change as it passes through a safety valve, resulting in a temperature indication corresponding to the pressure in the PRT. The ARP directs the closing of the associated block valve to isolate the leaking PORV.

Sys #	System	Category	KA Statement
000008	Pressurizer Vapor Space Accident / 3	AK1. Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident:	Thermodynamics and flow characteristics of open or leaking valve

K/A# AK1.01 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.4.ACE Iss. 3 Rev. 0 pg. 1

Question Source: Bank – Surry 2012 Q2

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(14)

Objective: 1OM-6.4, Rev. 14 Obj. 21. Given a change in plant conditions due to system or component failure, analyze the Pressurizer and Pressurizer Relief System to determine what failure has occurred.

1OM-6.4, Rev. 14 Obj. 23. Given a Pressurizer and Pressurizer Relief System alarm condition and using the Alarm Response Procedure(s), determine the appropriate alarm response, including automatic and operator actions in the control room.

3. The plant was operating at 100% power when the following annunciators alarm.

- A1-35, CONTAINMENT AIR TOTAL PRESS HIGH-HIGH CH I
- A1-49, CONTAINMENT SUMP LEVEL HIGH
- A3-58, CHARGING PUMP DISCH FLOW HIGH-LOW
- A4-4, PRESSURIZER CONTROL LOW LEVEL DEVIATION
- A4-12, PRESSURIZER CONTROL LOW PRESS DEVIATION
- A4-43, REACTOR CORE MARGIN TO SATURATION LOW ICC TRAIN A
- A4-71, RADIATION MONITORING HIGH

Current plant parameters:

- Reactor power is 100%
- RCS pressure is 2025 psig and lowering
- PRZR level is 32% and lowering
- FI-1CH-122A, Charging Pump Flow is 160 gpm and stable
- Cnmt Pressure is -0.7 psig and rising

What type of event is in progress?

A. RCS Leakage

B. Small break LOCA

C. Steam Generator Tube Rupture

D. Steam Line Break

Answer: B

Explanation/Justification: K/A is met with the candidate's ability to interpret control room annunciators to determine that a small break LOCA is in progress.

A. Incorrect. Plausible because RCS inventory loss is occurring, but it is larger than the capacity of one charging pump which must be recognized by the maximum.charging flow given, and przr level and pressure are still lowering.

B. Correct. Small break LOCA is indicated by all of the annunciators and plant conditions given. The student must recognize that safety injection is required to be actuated to accommodate the loss of RCS inventory. This is not a large break LOCA because RCS pressure is 2035 psig and lowering at this time.

C. Incorrect. Plausible because many of the alarms are in during a SGTR, but a SGTR will not cause the cnmt parameters to change as indicated.

D. Incorrect. Plausible because several of the alarms would be present during a steam line break, but reactor power remained at 100%, whereas, if it had been a steam line break power would have risen due to a lower Tavg. A steam line break will not cause a radiation monitor alarm either.

Sys #	System	Category	KA Statement
000009	Small Break LOCA / 3	Generic	Ability to prioritize and interpret the significance of each annunciator or alarm.

K/A# 2.4.45 **K/A Importance** 4.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53B.4.E-1 Iss. 3 Rev. 1 pg. 2

Question Source: New

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

Objective: GO-ATA 4.2 Rev. 8 Obj. 2. Explain the purpose and general methodology of the UFSAR safety analyses. Include the assumptions and conclusions of the analyses.

4. The plant was operating at 25% power when the ‘A’ RCP tripped, which of the following correctly describes the effect on the ‘A’ SG water level after the RCP trip?

   Assuming no operator action is taken, ‘A’ S/G level will initially ________________.

A. lower, then return to program value.

B. lower, then stabilize below program value.

C. rise, then return to program value.

D. rise, then stabilize above program value.

Answer: A

Explanation/Justification: K/A is met with the candidate’s ability to understand and monitor how the SG water level will respond to an RCP trip without a reactor trip, and how the SG water level control system will restore level back to the program level.

A. Correct. Steaming rate will lower due to less thermal energy in the idle loop causing shrink to occur (SG level lowers), then the SG water level control system will restore the level to program level.
B. Incorrect. Plausible because SG level will lower, but SG water level control will restore level to program level of 65%. Candidate may think that SGWLC is variable like the przr level control.
C. Incorrect. Plausible if the candidate thinks the steaming rate will rise causing swell (SG level rises). SG water level control will restore level to program level of 65%.
D. Incorrect. Plausible if the candidate thinks the steaming rate will rise causing swell (SG level rises). SG water level control will restore level to program level of 65%. Candidate may think that SGWLC is variable like the przr level control.

Sys #	System	Category	KA Statement
000015	Reactor Coolant Pump Malfunctions / 4	AA1. Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow):	S/G LCS

K/A# AA1.08 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** GO-3ATA3.2 Rev. 4 pg. 26-31

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(14)

Objective: GO-3ATA 3.2, Rev. 4 Obj. 1. Predict and analyze the plant response (T_{AVG}, Reactor Power, Net Reactivity, Pressurizer Pressure, Pressurizer Level, Steam Generator Pressure, Steam Generator Level, and Steam Flow) to the following transients: e. RCP Trip at Power (Affected and Unaffected Loops)

5. The plant was operating at 100% power when the following conditions occurred:

- 1B Charging Pump is RUNNING.
- 1A Charging Pump is SECURED.
- FCV-1CH-122, CHV FLOW TO REGEN HX INLET CONTROL VALVE is in AUTO.
- CHARGING PUMP 1B AMPS, begins to oscillate.
- FI-1CH-122A, CHARGING PUMP FLOW, begins to oscillate.

Subsequently, the following annunciators come into alarm:

- A3-58, CHARGING PUMP DISCH FLOW HIGH-LOW
- A3-78, REACT COOL PP SEAL INJECTION FLOW LOW
- A3-115, REGEN HX LETDOWN OUTLET TEMP HIGH

Which one of the following completes the statements below?

1) The alarms and indications above are indicative of _______________.

2) The setpoint for A3-78, REACT COOL PP SEAL INJECTION FLOW LOW is _______.

A. 1) FCV-1CH-122 closing
2) 3gpm

B. 1) FCV-1CH-122 closing
2) 6.5 gpm

C. 1) 1B Charging Pump Cavitation
2) 3 gpm

D. 1) 1B Charging Pump Cavitation
2) 6.5 gpm

Question 5

Answer: D

Explanation/Justification: K/A is met by the candidate's ability to analyze given plant alarms, and determine that the charging pump is cavitating.

A. Incorrect. Plausible because when FCV-1CH-122 closes, A3-58 and A3-115 will alarm, but seal injection flow will remain within range, and charging pump amps and flow will slowly lower, but will not oscillate. Second part is plausible if the candidate thinks about 10 gpm (3.3 gpm to each pump) total seal injection flow which is obtained in the Excessive Primary Plant Leakage AOP when trying to establish RCS leakrate and thinking that they can't reduce seal injection flowrate below the alarm setpoint, which is not the case.

B. Incorrect. Plausible because when FCV-1CH-122 closes, A3-58 and A3-115 will alarm, but seal injection flow will remain within range, and charging pump amps and flow will slowly lower, but will not oscillate. Setpoint for A3-78, REACT COOL PP SEAL INJECTION FLOW LOW is 6.5 gpm iaw the ARP.

C. Incorrect. Charging pump amps and flowrate oscillating is an indication of pump cavitation. Second part is plausible if the candidate thinks about 10 gpm (3.3 gpm to each pump) total seal injection flow which is obtained in the Excessive Primary Plant Leakage AOP when trying to establish RCS leakrate and thinking that they can't reduce seal injection flowrate below the alarm setpoint, which is not the case.

D. Correct. Charging pump amps and flowrate oscillating is an indication of pump cavitation. Setpoint for A3-78, REACT COOL PP SEAL INJECTION FLOW LOW is 6.5 gpm iaw the ARP.

Sys #	System	Category	KA Statement
000022	Loss of Reactor Coolant Makeup / 2	AA2. Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup:	Charging pump problems

K/A# AA2.02 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.7.1 Rev. 12 pg. 6
1OM-7.4.ABB Rev. 7 pg. 2

Question Source: Bank – Farley 2015 NRC Exam Q22

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(5)

Objective: 1SQS-7.1, Rev. 20 Obj. 22. Given a change in plant conditions due to system or component failure, analyze the Chemical and Volume Control System to determine what failure has occurred.

1SQS-6.3 Rev. 15 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.

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

- The reactor was shutdown from an extended Full Power run at 1200, FOUR days ago.
- It is currently 1600.
- RCS Temperature is 180 °F.
- RCS Pressure is 235 psig.
- PRZR Water Level is 22%.
- A Complete Loss of RHR occurred.
- All THREE Loops are available for heat removal.

Based on AOP 1.10.1, and the stated plant conditions, which of the following is the estimated time to saturation?

(Reference Provided)

A. < 1/2 hour

B. > 1/2 hour and < 1 hours.

C. > 1 hour but < 2 hours.

D. > 2 hours but < 3 hours.

Answer: D

Explanation/Justification: K/A is met with the knowledge of the operational implications that a loss of RHR will have on the RCS time to saturation.

A. Incorrect. Plausible if the candidate uses Attachment 5 as referenced by step 4 of AOP 1.10.1 but doesn't recognize that this attachment is for loop isolation valves closed and the RCS at atmospheric conditions. Times would be 15-25 minutes.
B. Incorrect. Plausible if the candidate uses Attachment 4 as referenced by step 4 of AOP 1.10.1 but doesn't recognize that this attachment is for RCS at atmospheric conditions. Using the 140F line, time to saturation would be 45 minutes.
C. Incorrect. Plausible if the candidate incorrectly calculates H/U rate, improperly applies the curves, or makes a mathematical error.
D. Correct. Using Attachment 3 of AOP 1.10.1 and steam tables, the candidate will determine that Tsat for 235 psig (250 psia) is 401 F. Present RCS temperature is 180 F. Time since shutdown is 100 hours. Using Attachment 7, the current H/U rate is 1.5 F/min. 221 F/1.5 F/min = 147.3 minutes.

Sys #	System	Category	KA Statement
000025	Loss of Residual Heat Removal System / 4	AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System:	Loss of RHRS during all modes of operation

K/A# AK1.01 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate 1OM-53C.4.1.10.1 Rev. 19 pg. 7, 30-36 Steam Tables **Technical References:** 1OM-53C.4.1.10.1, Rev. 19 pg. 7, 30, 34 Steam Tables

Question Source: Bank – 1LOT8Q6

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

Objective: 1SQS-53C.1 Rev. 12 Obj. 6. Given a set of conditions, apply the correct AOP

7. Which of the following describes the basis for the Component Cooling Water (CCR) System valve realignment upon receipt of a Containment Isolation Phase B (CIB) actuation?

A. Ensures CCR System is not an additional potential radioactive release path from Containment.

B. Reduces heat load on CCR System by eliminating unnecessary cooling requirements.

C. Reduces Diesel Generator loading requirements with Containment Spray in operation.

D. Ensures that CCR System meets design cooling function for loads within Containment during Design Basis Loss of Coolant Accident.

Answer: A

Explanation/Justification: K/A is met with the knowledge that when an automatic CIB occurs at 11.1 psig in the containment, the containment will be isolated (including CCR isolation) to provide another fission product barrier in the event of a LOCA or Steam Line Break inside Containment.

A. Correct. Isolation of Component Cooling Water is required for Containment Integrity.
B. Incorrect. Plausible because there would be a reduction in heat load on the CCR System, however, a heat load reduction is not required to meet design load limits.
C. Incorrect. Plausible if it is thought that Emergency Diesel Generator load changes due to the CIB actuation tripping the stub bus breaker, however, it is not the basis for the valve realignment.
D. Incorrect. Plausible because the CCR System has a heat removal function during a DBA LOCA, however, there are no loads associated with CCR operating in Containment.

Sys #	System	Category	KA Statement
000026	Loss of Component Cooling Water / 8	AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water:	The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFA

K/A# AK3.02 **K/A Importance** 3.6 **Exam Level** RO
References provided to Candidate None **Technical References:** 1OM-53B.4.FR-Z.1 Iss 3 Rev 0 Step 3
Question Source: Bank – 1LOT18 Q7 (LAST 2 EXAMS)
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(7)
Objective: 3SQS-1.1, Rev. 8 Obj. 1. Describe the function of the Reactor Protection System Trip Logics & Engineered Safeguards Features Actuation Signals and the associated major components as documented in Operating Manual, 1(2)OM-1.

8. The plant is at 100% power when the pressurizer master pressure controller fails to zero output while in AUTO.

   With no operator action, which of the following will occur?

A. Pressurizer pressure will rise and cycle at the PORV open setpoint.

B. Pressurizer pressure will lower and cycle at the PORV blocked setpoint.

C. The Reactor will trip on the high Pressurizer pressure setpoint.

D. Safety Injection will actuate on the low Pressurizer pressure setpoint.

Answer: A

Explanation/Justification: K/A is met by the candidate analyzing the master pressure controller malfunction, and predict the operation of the przr heaters, spray valves, and PORVs to control pressurizer pressure.

A. CORRECT: The master controller will close both spray valves and turn on all heaters, causing pressure to increase. This will cause PORV PCV-1RV-456 and 455D to open at ~2335 psig and reclose at P-11 (2000 psig). PORV PCV-1RV-455C will not cycle because it is controlled from the MPC.

B. INCORRECT: Plausible the candidate would think the pressure would lower with the output lowering and that it would lower until the PORV block setpoint P-11 (2000 psig) is reached. But the master controller will close both spray valves and turn on all heaters, causing pressure to continually increase until the other PORVs open.

C. INCORRECT: Plausible if the candidate recognizes that the master controller will close both spray valves and turn on all heaters, causing pressure to increase, and also recognizes PORV PCV-1RV-455C will not cycle because it is controlled from the MPC, but pressure will be maintained by the other PORVs, and the reactor trip setpoint of 2385 psig will not be reached.

D. INCORRECT: Plausible the candidate would think the pressure would lower with the output lowering, and that with the given malfunction the pressure would continue to lower until a low-pressure reactor trip and SI would occur.

Sys #	System	Category	KA Statement
000027	Pressurizer Pressure Control System Malfunction / 3	AK2. Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following:	Controllers and positioners

K/A# AK2.03 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.4.IF Rev. 11 pg. 23, 24

Question Source: Bank – 2017 South Texas Q12

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

Objective: 1SQS-6.4 Rev. 14 Obj. 6. Given a change in plant conditions, describe the response of the Pressurizer and Pressurizer Relief System field indication and control loops, including all automatic functions and changes in equipment status.

1SQS-6.4 Rev. 14 Obj. 17. Describe the control, protection and interlock functions for the control room components associated with the Pressurizer and Pressurizer Relief System, including automatic functions, setpoints and changes in equipment status as applicable.

9. Which of the following describes the reason that emergency boration is initiated in FR-S.1, Response To Nuclear Power Generation/ATWS?

A. It is performed because the UFSAR accident analysis does not take credit for local operator actions in the event of an ATWS.

B. It is performed to add negative reactivity to bring the reactor core subcritical.

C. It is the fastest mechanism for adding negative reactivity to the reactor core.

D. It bypasses the Boric Acid Filter to minimize loss of flow due to filter clogging.

Answer: B

Explanation/Justification: K/A is met with the candidate's knowledge of the reason that emergency boration is initiated during an ATWS is to bring the reactor core subcritical.

A. Incorrect. Plausible if the candidate assumes that credit is not taken for local operator actions to insert control rods, but as is stated in FR-S.1 major action step 2, emergency boration action is taken prior to initiating more time-consuming local actions to trip the reactor and/or turbine.
B. Correct. Per FR-S.1 step 3 background, the purpose of emergency boration is to add negative reactivity to bring the reactor core subcritical.
C. Incorrect. It is not the fastest method to add negative reactivity; the fastest method is to insert rods. Plausible distractor if candidate thinks boration injection would surpass the negative reactivity of the rods.
D. Incorrect. Filter clogging would be a concern as it would reduce emergency borate flow, however, any flowpath for emergency boration must go through the BA filter. Plausible distractor if the candidate does not have sound boration system knowledge.

Sys #	System	Category	KA Statement
000029	Anticipated Transient Without Scram / 1	EK3 Knowledge of the reasons for the following responses as the apply to the ATWS:	Initiating emergency boration

K/A# EK3.11 **K/A Importance** 4.2 **Exam Level** RO
References provided to Candidate None **Technical References:** 1OM-53B.4.FR-S.1 Iss 3 Rev 3 pg. 63
Question Source: Bank – Callaway 2013 Q8
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(5)
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.

10. A SGTR has occurred, and current conditions are as follows:

- The crew has entered E-3, Steam Generator Tube Rupture.
- The crew is depressurizing the RCS to less than ruptured SG pressure.
- RCS pressure has just dropped below 1950 psig.

Which signal is required to be blocked/reset by the crew at this point in E-3?

A. Low Steam Line pressure SI is required to be blocked.

B. Low Pressurizer pressure SI is required to be blocked.

C. Containment Isolation Phase A is required to be reset.

D. Feedwater Isolation is required to be reset.

Answer: A

Explanation/Justification: K/A is met with the candidates ability to operate the "Block Steam Line SI" switches when < 2000 psig (P-11) to remove the Main Steam Line Isolation on SG low pressure at 500 psig which would prevent cooldown using the condenser steam dumps.

A. Correct. Blocking low steam pressure SI performs two functions. The first is to block low steam pressure SI, and the second is to remove the low steam pressure MSI and replace it with a high rate MSI. Based on the need to prevent MSI when SGs are depressurized to cooldown the RCS to RHR conditions later in the E-3 series procedures. If the crew fails to do this, MSI will actuate during the cooldown, complicating the cooldown. There is no need to block SI signals, since SIS has already actuated, and a subsequent auto SIS actuation is already prevented by P-4 permissive.

B. Incorrect. Plausible because during a plant cooldown where SI was not required, the low pressurizer pressure SIS would be required to be blocked when <P-11 permissive.

C. Incorrect. Plausible since CIA was actuated when SI actuated, but it is reset later in E-3 (step 11), not when cooldown begins. Resetting CIA allows the opening of valves as directed in subsequent steps of the procedure.

D. Incorrect. Plausible because several places in the EOP network reset the FWI signal, but AFW is being used to feed the SGs.

Sys #	System	Category	KA Statement
000038	Steam Generator Tube Rupture / 3	EA1 Ability to operate and monitor the following as they apply to a SGTR:	Safety injection and containment isolation systems

K/A# EA1.30 **K/A Importance** 4.0 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.E-3 Iss 3 Rev 4 pg. 9, 10
1OM-53B.4.E-3 Iss 3 Rev 4 pg. 71,72

Question Source: Bank – Millstone 3 2013 Q10

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

Objective: 3SQS-53.3, Rev. 5 Obj. 4. Explain from memory the basis for ALL Cautions and Notes, IAW BVPS-EOP Executive Volume.

11. Given the following plant conditions:
    - The plant was operating at 100% power, a Reactor Trip and Safety Injection have occurred due to a steam line break in Containment on the 'B' SG

    Current plant conditions are as follows:
    - Containment pressure is 9.3 psig
    - The crew has transitioned from E-0, Reactor Trip or Safety Injection, and are at step 1 of E-2, Faulted Steam Generator Isolation

    Which of the following identifies the set of valves listed below that are expected to be in the SHUT position for the current plant conditions?

    1. All MSIV's
    2. TV-1IA-400, CNMT Inst Air Header Isolation valve
    3. MOV-1CH-289, Regen HX/Chg Header Inlet CNMT Isolation valve
    4. ONLY 'B' MSIV
    5. All SG Blowdown CNMT isolation valves

A. 1, 2 and 3 only

B. 4 and 5 only

C. 1, 3 and 5 only

D. 2 and 4 only

Answer: C

Explanation/Justification: K/A is met by the candidate demonstrating knowledge of the ESF actuations which occur during a steam line rupture in containment and the interrelations with various system valve actuations.

A. Incorrect. Plausible because TV-1IA-400 would close on a CIB signal (11.1 psig) but pressure has only reached 9.3 psig. MSIVs and MOV-1CH-289 would be closed.

B. Incorrect. Plausible that only B MSIV closed if they are thinking of a tube rupture which isolates only the affected SG, but containment pressure is greater than the MSLI setpoint of 7 psig, therefore all MSLI valves are closed. All SG Blowdown CNMT isolation valves are closed.

C. Correct. At 5 psig SI will occur. The SI signal will close MOV-1CH-289 and actuate CIA which will close all SG Blowdown CNMT isolation valves. At 7 psig, a MSLI isolation will occur closing the MSLI valves.

D. Incorrect. Plausible because TV-1IA-400 would close on a CIB signal (11.1 psig) but pressure has only reached 9.3 psig. Plausible that only B MSIV closed if they are thinking of a tube rupture which isolates only the affected SG, but containment pressure is greater than the MSLI setpoint of 7 psig, therefore all MSLI valves are closed.

Sys #	System	Category	KA Statement
000040	Steam Line Rupture—Excessive Heat Transfer / 4	AK2. Knowledge of the interrelations between the Steam Line Rupture and the following:	Valves

K/A# AK2.01 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.2.B Rev. 26 pg. 4, 1OM-53A.1.1-A Rev. 2 pg. 3
1OM-53A.1.1-B Rev. 3 pg. 4,
1OM-53A.1.1-D Rev. 1 pg. 3

Question Source: Modified – 2014 Harris Q10

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

Objective: 3SQS-1.1 Rev. 8 Obj. 4 Given a Reactor Protection System Trip Logics & Engineered Safeguards Features Actuation configuration and without referenced material, describe the Reactor Protection System Trip Logics & Engineered Safeguards Features Actuation Signals field response to the following actuation signals, including automatic functions and changes in equipment status: b. Main Steam Line Break Accident

12. Given the following plant conditions:
    - The plant was operating at 60% power
    - 'B' MDAFW pump is on clearance

    Subsequently the following occurs:
    - A manual Reactor Trip was initiated due to a loss of both MFPs
    - The TDAFW pump tripped after starting
    - MDAFW flow control valves are full open
    - SG NR levels are 52% and lowering
    - Containment pressure is -1.3 psig

    Based on the given conditions, which of the following parameters would require entry into FR-H.1, Response to Loss of Secondary Heat Sink?

    All SG NR levels are less than _____(1)_____ AND total AFW flow is less than _____(2)_____.

A. 1) 31%
   2) 370 gpm

B. 1) 31%
   2) 630 gpm

C. 1) 50%
   2) 370 gpm

D. 1) 50%
   2) 630 gpm

Answer: A

Explanation/Justification: K/A is met with the knowledge that to prevent a Loss of Secondary Heat Sink and provide decay heat removal capabilities to the primary coolant system after a loss of main feedwater, the secondary heat sink must be maintained by at least one SG NR level >31% (non-adverse), or 370 gpm AFW flow to the SGs.

A. Correct. To maintain a secondary heat sink for decay heat removal and prevent entering FR-H.1 for a loss of heat sink, AFW flow must be >370 gpm, or at least one steam generator NR level must be >31% (50% adverse cnmt). A normal cnmt pressure was given in the stem as -1.3 psig.

B. Incorrect. First part is correct. Second part is plausible because the required AFW flow during an ATWS event for decay heat removal if SG NR levels are <31% is 630 gpm.

C. Incorrect. First part is plausible because 50% is the SG NR level required for adverse cnmt conditions. Second part is correct.

D. Incorrect. First part is plausible because 50% is the SG NR level required for adverse cnmt conditions. Second part is plausible because the required AFW flow during an ATWS event for decay heat removal if SG NR levels are <31% is 630 gpm.

Sys #	System	Category	KA Statement
000054	Loss of Main Feedwater /4	Generic	Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

K/A# 2.4.21 **K/A Importance** 4.0 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.E-0 Iss. 3 Rev. 3 pg. 8; 1OM-53A.1.F-0.3 Iss. 3 Rev. 0

Question Source: Bank – 2LOT19 Q17

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

Objective: 3SQS-53.3, Rev. 5 Obj. 5. Explain from memory the basis for the decision blocks of each Status Tree, IAW BVPS-EOP Executive Volume.

13. Given the following plant conditions:

- A Loss of ALL AC Power occurred.
- The Control Room is performing actions of ECA-0.0, "Loss of All Emergency AC Power".
- ELAP was declared by the Unit Supervisor

Which of the following describes the design capacity of the 125VDC station batteries during a station blackout, and the purpose of declaring ELAP (extended loss of AC power) during ECA-0.0?

The design capacity of the 125 VDC station batteries is ____(1)____.

The purpose of declaring ELAP is to ____(2)____.

A. 1) 2 hours
2) increase monitoring and reduce loads on **both trains** of batteries.

B. 1) 2 hours
2) isolate **one train** of batteries, then monitor and reduce loads on the other train.

C. 1) 8 hours
2) increase monitoring and reduce loads on **both trains** of batteries.

D. 1) 8 hours
2) isolate **one train** of batteries, then monitor and reduce loads on the other train.

Answer: B

Explanation/Justification: K/A is met with the candidate's ability to perform ELAP (extended loss of AC power) DC bus load shed and management during a station blackout to support battery life preservation.

A. Incorrect. Two hours is the design capacity of the batteries. Second part is plausible because it sounds like a reasonable expectation when a loss of AC power starts depleting the batteries.

B. Correct. Two hours is the design capacity of the batteries. ELAP is declared if an AC bus is unable to be restored within one hour to ensure one train of batteries is isolated within two hours, and load shedding of the remaining train of batteries is completed within three hours to support battery life preservation.

C. Incorrect. Eight hours is plausible if the candidate thinks that each of the 4 batteries with a 2-hour capacity is combined for a total of eight hours. Second part is plausible because it sounds like a reasonable expectation when a loss of AC power starts depleting the batteries.

D. Incorrect. Eight hours is plausible if the candidate thinks that each of the 4 batteries with a 2-hour capacity is combined for a total of eight hours. Second part is correct.

Sys #	System	Category	KA Statement
000055	Station Blackout / 6	EA1 Ability to operate and monitor the following as they apply to a Station Blackout:	Reduction of loads on the battery

K/A# EA1.04 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None

Technical References:
1OM-39.1.B, Rev.1, pg. 4
3SQS-39.1, Rev. 9, pg. 3
1OM-53A.1.ECA-0.0 Iss. 3 Rev 3 pg. 11
1OM-53B.4.ECA-0.0 Iss. 3 Rev.3 pg 132

Question Source: New

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

Objective:
3SQS-39.1 Rev. 9 Obj. 17. Describe the battery capacity as it is affected by discharge rate.
3SQS-53.3, Rev. 5 Obj. 4. Explain from memory the basis for ALL Cautions and Notes, IAW BVPS-EOP Executive Volume.

14. Plant was operating at 100% power when a gasoline barge impacted the River Water Intake structure and exploded. The Intake Structure has been rendered inoperable.

    Which system is designed to provide the necessary cooling requirements to support a plant shutdown from 100% power and subsequent cooldown to <200°F during this accident?

A. Turbine Plant River Water System

B. Unit 2 Service Water System

C. Engine-driven Fire Pump

D. Auxiliary River Water System

Answer: D

Explanation/Justification: K/A is met with the candidate's knowledge of the function and purpose of the Auxiliary River Water system and its ability to allow a plant shutdown in the event of a loss of the River Water system.

A. Incorrect. Plausible because Turbine Plant – Reactor Plant Cross Connect valve 1RW-61 exists to supply selected TPRW system loads from the RPRW system when TPRW pumps are no longer needed and the system is to be shutdown.
B. Incorrect. Plausible because Unit 1 River Water System - Unit 2 Service Water System Cross Connection valve 1RW-60 exists to provide a path for either unit to supply the opposite unit with RW/SWS cooling during ECA-0.0, "Loss of all AC Power".
C. Incorrect. Plausible because the engine driven fire pump is capable of supplying the River Water header to supply limited loads such as AFW pumps, EDGs, charging pumps, and CR AC units.
D. Correct. The Auxiliary River Water System supplies water to the River Water system whenever the main intake structure is disabled and is designed to provide the necessary cooling requirements to support a unit shutdown from 100% power and subsequent cooldown to <200°F, in the event that the intake is disabled.

Sys #	**System**	**Category**	**KA Statement**
000062	Loss of Nuclear Service Water / 4	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:** 1SQS-30.2 Rev. 18 pg. 3, 4
Question Source: New
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(7)
Objective: 1SQS-30.2, Rev. 18 Obj. 1. Describe the function of the Reactor Plant River Water system and the associated major components as documented in Chapter 30 of the associated Operating Manual.

15. The plant is in Mode 3 with all systems in normal alignment for this mode.
- The reactor trip breakers are **OPEN**.
- A Loss of station instrument air occurs.
- Station instrument air header pressure is 0 psig.

What impact will this loss of station instrument air have on the following CVCS functions:

Charging will _______(1)_______, RCP seal injection will _______(2)_______, and RCP seal return will _______(3)_______.

A.	(1) isolate	(2) isolate	(3) isolate
B.	(1) isolate	(2) remain in service	(3) isolate
C.	(1) remain in service	(2) remain in service	(3) remain in service
D.	(1) remain in service	(2) isolate	(3) remain in service

Answer: C

Explanation/Justification: K/A is met by the candidate's ability to determine how each of the flowpaths are affected by a loss of instrument air with their knowledge of the individual failure modes of the air operated valves in the system.

A. Incorrect. See correct answer.
B. Incorrect. See correct answer.
C. Correct. Charging flow will remain in service because the only AOV in the flowpath is FCV-1CH-122 and it fails open. Seal injection will remain in service because HCV-1CH-186 fails open on loss of air. RCP seal return will remain in service because there are no AOVs in the seal return line. All other answers are plausible if the candidate is not familiar with the system flowpaths or the failure positions of the valves on a loss of air.
D. Incorrect. See correct answer.

Sys #	System	Category	KA Statement
000065	Loss of Instrument Air / 8	AA2. Ability to determine and interpret the following as they apply to the Loss of Instrument Air:	Failure modes of air-operated equipment

K/A# AA2.08 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.34.1 Rev. 29 pg. 11
U1 RM-0407-001 Rev. 40
U1 RM-0407-004 Rev. 30

Question Source: Bank – 2LOT6 Q15

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

Objective: 1SQS-7.1, Rev. 20 Obj. 8. Given a Chemical and Volume Control System configuration and without referenced material, describe the Chemical and Volume Control System field response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable. a. Loss of Instrument Air

16. The plant is at 100% power with all systems in NSA **EXCEPT** Emergency Diesel Generator (EDG #2) EE-EG-2 is OOS for maintenance on the engine driven fuel oil pump.

- The DLC System Operations Control Center notifies the control room of a significant threat to grid stability
- Grid voltage swings are verified in the control room
- The crew enters AOP 1/2.35.1, Degraded Grid

What actions, if any, regarding #1 EDG are required IAW AOP 1/2.35.1, Degraded Grid to place the plant in a safe operating condition?

A. **NO** actions required. Allow EDG #1 to Auto start if necessary.

B. Manually start EDG #1, flash the field **AND** manually CLOSE ACB 1E9, Emerg Gen1 Circuit Breaker.

C. Manually start EDG #1, flash the field **AND** manually OPEN ACB 1A10, 4KV Bus 1A to 1AE Circuit Breaker.

D. Manually start EDG #1, flash the field, synchronize the EDG to the 1AE bus, assume load, **AND** then manually OPEN ACB 1E7, 4KV Bus 1AE to 1A Circuit Breaker.

Answer: C

Explanation/Justification: K/A is met with the candidate's knowledge of the operation of the emergency diesel generator during a degraded grid condition as directed by the Degraded Grid AOP to ensure the plant is placed in a safe operating condition.

A. Incorrect. These are the correct actions for other AOPs that could possibly threaten offsite electrical power such as a tornado or high winds.

B. Incorrect. Plausible since it makes sense to get the EDG ready for service with manual action. However, this procedure will specifically direct getting the EDG off of any parallel operations and then performing choice C

C. Correct. There is a note prior to step 15 which ensures the 4KV emergency busses are powered from the respective diesel. It states that safety systems will preemptively be placed on emergency power supplies to place the plant in a safe operating or shutdown condition.

D. Incorrect. This will accomplish the task of divorcing safety related equipment from offsite power and establishing the EDG as the power supply to safety related equipment. However, this is not how the procedure directs this to be accomplished and violates the OM 36 P&L to not parallel EDGs with offsite power when anticipating a loss of offsite power.

Sys #	System	Category	KA Statement
000077	Generator Voltage and Electric Grid Disturbances / 6	AK3. Knowledge of the reasons for the following responses as they apply to Generator Voltage and Electric Grid Disturbances:	Actions contained in abnormal operating procedure for voltage and grid disturbances

K/A# AK3.02 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1/2OM-53C.4A.35.1 Rev. 10 pg. 9, 10

Question Source: Bank – 1LOT14 Q16

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

Objective: 1SQS-53C.1, Rev. 12 Obj. 4. Explain the basis for cautions and notes of each AOP.

17. The plant was operating at 100% power.

- A Loss of ALL Feedwater occurs
- The crew is performing actions of FR-H.1, “Response to Loss of Secondary Heat Sink”
- PRZR Pressure rises to 2230 psig

Which of the following indicates that a Loss of Secondary Heat Sink has occurred?

A. A smaller RCS loop Delta T because Tcold is rising.

B. A larger RCS loop Delta T because Tcold is lowering.

C. A smaller RCS loop Delta T because Thot is lowering.

D. A larger RCS loop Delta T because Thot is rising.

Answer: A

Explanation/Justification: K/A is met with the candidate’s knowledge of how the primary coolant hot and cold loops respond to a loss of Main Feedwater/Secondary Heat Sink.

A. Correct. Low loop Delta T indicates heat is not being removed. Lack of Heat removal due to high Tc means SG no longer acting as a heat sink.
B. Incorrect. High Delta T indicates natural circulation exists or is setting up. Tcold lowering would indicate heat removal does exist.
C. Incorrect. Low loop Delta T due to Th lowering could mean that heat sink is adequate and decay heat load is low.
D. Incorrect. High Delta T is a strong indication of natural circulation initiation.

Sys #	System	Category	KA Statement
WE05	Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4	EK2. Knowledge of the interrelations between the (Loss of Secondary Heat Sink) 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:** 1OM-53B.4.FR-H.1 Iss. 2 Rev. 4 pg. 31, 32
Question Source: Bank – 2018 Audit exam Q18

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(7)
Objective: GO-3ATA 4.3, Rev. 6 Obj. 2. Identify the “worst” case of initial conditions or, given a parameter, identify which direction of its magnitude would be “worse” for initial conditions for each listed accident.

18. Given the following plant conditions:

- A LOCA has occurred.
- Due to multiple equipment failures, the control room is performing actions of ECA-1.1, Loss of Emergency Coolant Recirculation.
- Two (2) Charging/HHSI pumps and two (2) LHSI pumps are running.
- One (1) Quench Spray pump is running.
- RCS pressure is 900 psig and SLOWLY LOWERING.
- Containment pressure is 13 psig and SLOWLY LOWERING.
- RWST Level is 1.7 feet and SLOWLY LOWERING.

Which of the following describes the **REQUIRED** action in accordance with ECA-1.1?

A. STOP ALL pumps taking suction from the RWST and verify no backflow from the RWST to CNMT sump.

B. STOP ALL pumps taking suction from the RWST and initiate secondary depressurization to facilitate SI accumulator injection.

C. STOP ONLY ONE (1) HHSI and ONLY ONE (1) LHSI pump and initiate secondary depressurization to facilitate SI accumulator injection. Secure the Quench Spray pump.

D. STOP BOTH LHSI pumps and ONE (1) HHSI pump. Maintain Quench Spray pump running until containment pressure is < 11 psig and then add makeup to RCS from alternate sources.

Answer: B

Explanation/Justification: K/A is met with the candidate's ability to interpret plant conditions during ECA-1.1, Loss of Emergency Coolant Recirculation, and determine that the RWST is considered empty at <2 feet, requiring all pumps to be stopped, and SGs are depressurized to inject the SI accumulators.

A. Incorrect. Correct that all pumps are stopped. Incorrect plausible action.

B. Correct. The RO candidate must know the overall mitigative strategy of ECA-1.1 and sequence of events. ECA 1.1 directs the operator to secure all pumps taking suction from the RWST when level is < 2 feet. Once stopped the procedure directs the operator to check if all intact S/Gs should be depressurized.

C. Incorrect. Incorrect but plausible that one HHSI and one LSHI pump are secured because one of the procedural mitigating strategies is to conserve RWST water and in fact the procedure does directs action to secure pumps. Correct that the crew will initiate secondary depressurization to facilitate SI accumulator injection. Also correct that the Quench Spray pump is secured.

D. Incorrect. Incorrect but plausible action to maintain one pump running as noted above. Also plausible but incorrect that the Quench Spray pump is maintained running until containment pressure is < 11 psig. Normally by procedure this would be a correct action.

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)	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A# EA2.2 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-1.1 Iss. 3 Rev. 1 pg. 3, 23
1OM-53B.4.ECA-1.1 Iss. 3 Rev. 1 pg. 3

Question Source: Bank – 2LOT8 Q18

Question Cognitive Level: Higher – 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 fuel assembly has been dropped in Containment during core off-load.

1) Which of the following Radiation Monitors will automatically actuate the Localized CNMT Evacuation Alarm?

2) From which of the following locations can the CNMT Evacuation Alarm be manually activated?

A. 1) RM-1RM-203, Manipulator Crane Area Monitor
   2) Control Room

B. 1) RM-1RM-203, Manipulator Crane Area Monitor
   2) CNMT Personnel Air Lock

C. 1) RM-1VS-104A, Containment Purge Exhaust Monitor
   2) Control Room

D. 1) RM-1VS-104A, Containment Purge Exhaust Monitor
   2) CNMT Personnel Air Lock

Answer: C

Explanation/Justification: K/A is met by testing the ability to determine which rad monitor will cause the containment evacuation alarm to automatically sound if a Fuel Handling Accident would occur, and the ability to locate and operate the containment evacuation alarm switch should an accident occur.

A. Incorrect. Plausible distractor because high radiation in CNMT would be just cause for evacuating per the ARP, but 1RM-203 does not provide input to the automatic CNMT evacuation alarm. Control Room is the correct location for manual actuation.

B. Incorrect. Plausible distractor because high radiation in CNMT would be just cause for evacuating per the ARP, but 1RM-203 does not provide input to the automatic CNMT evacuation alarm. CNMT Personnel Air Lock is incorrect. It is a plausible distractor because the PAL is a central location for communications and contains a red revolving warning light for incore moveable detector movement.

C. Correct. RM-1VS-104A will actuate the CNMT Evacuation Alarm when the setpoint is reached, and the alarm can be manually actuated from the CR and shutdown panel communication consoles.

D. Incorrect. RM-1VS-104A will actuate the alarm, but the CNMT Personnel Air Lock is incorrect. It is a plausible distractor because the PAL is a central location for communications and contains a red revolving warning light for incore moveable detector movement.

Sys #	System	Category	KA Statement
000036	Fuel-Handling Incidents / 8	AA1. Ability to operate and / or monitor the following as they apply to the Fuel Handling Incidents:	Reactor building containment evacuation alarm enable switch

K/A# AA1.03 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.49.1 Rev. 10, pg. 2
1OM-43.1.D Rev. 10, pg. 10

Question Source: Bank – 1LOT16 Q19 (Last 2 exams)

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

Objective: 1SQS-43.1-01-2: Describe the automatic actions that occur in the field when a given radiation monitor alarms.

1SQS-43.1-01-10: Given a Radiation Monitoring System alarm condition, determine the appropriate alarm response, including automatic and operator actions in the control room.

3SQS-40.1-01-02 2.Explain the different alarm notification systems that are used at BVPS including how and when they are activated.

20. The plant was operating at 100% power when a SG tube leak developed causing the crew to enter AOP-1.6.4, Steam Generator Tube Leak.

    Current plant conditions are:

    - PZR Level is 57% and STABLE
    - RCS Pressure is 2230 psig and STABLE
    - RCS Tavg is 578°F and STABLE
    - VCT level has lowered by 3.2% over a 2-minute trend

    1) Based on the above conditions, what is the current magnitude of the SG tube leak?
    2) During a controlled plant shutdown, how will the leak rate respond as power is lowered?

A. 1) 22 gpm
   2) The leakrate will lower.

B. 1) 22 gpm
   2) The leakrate will remain constant.

C. 1) 45 gpm
   2) The leakrate will lower.

D. 1) 45 gpm
   2) The leakrate will remain constant.

Answer: A

Explanation/Justification: K/A is met by the candidate's ability to determine a SG tube leakrate and understand how the leakrate will lower as a plant shutdown reduces power.

A. Correct. With RCS temperature and PRZR level stable, the note prior to step 7 states that the VCT is ~14 gal per percent. 1.6%/min X 14 gal = 22.4 gpm leakrate. The leakrate will lower as power is reduced. As power is reduced, SG pressures rise, this lowers the D/P between the SG and the RCS causing the leak rate to lower.

B. Incorrect. 22 gpm is correct. The second part is plausible as it is logical to think that the leak rate is independent of power level as a primary leak would be not associated with a SG.

C. Incorrect. Plausible if the candidate knows the 14gal/% but multiplies it by 3.2%. The leakrate will lower as power is reduced.

D. Incorrect. Plausible if the candidate knows the 14gal/% but multiplies it by 3.2%. The second part is plausible as it is logical to think that the leak rate is independent of power level as a primary leak would be not associated with a SG.

Sys #	System	Category	KA Statement
000037	Steam Generator Tube Leak / 3	AA2. Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak:	Flow rate of leak

K/A# AA2.12 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.6.4 Rev. 35 pg. 6

Question Source: New

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

Objective: 1SQS-53C.1, Rev. 12 Obj. 4. Explain the basis for cautions and notes of each AOP.

GO-3ATA 3.1, Rev. 3 Obj. 2. Given one of the above listed transients (Ten percent power changes), calculate, where applicable, final values of: d. Steam Pressure

21. RM-1LW-104, Liquid Waste Effluent radiation monitor reads out in ______(1)______ and isolates a Coolant Recovery Tank discharge to the Cooling Tower Blowdown when a ______(2)______ radiation alarm is received.

A. 1) µCi /cc
   2) High

B. 1) CPS
   2) High

C. 1) µCi /cc
   2) High-High

D. 1) CPS
   2) High-High

Answer: D

Explanation/Justification: K/A is met with the candidates knowledge of the meter indication units identifying the intensity of the radiation indicated on the Liquid Waste Effluent radiation monitor and recognizes that it will isolate a discharge on a High-High alarm to prevent an accidental liquid radwaste release.

A. Incorrect. Plausible because RM-1LW-104 is sensitive to Cs137 at a level to 1E-6 µCi /cc, but it reads out in CPM on the RMS Rack in the control room. High radiation alarm is plausible because it is an alarm available in the control room, but once it is verified, the discharge may continue provided the Operator logs the reading from RM-1LW-104 every 15 minutes on a temporary log.

B. Incorrect. CPS is the correct meter units for RM-1LW-104. High radiation alarm is plausible because it is an alarm available in the control room, but once it is verified, the discharge may continue provided the Operator logs the reading from RM-1LW-104 every 15 minutes on a temporary log.

C. Incorrect. Plausible because RM-1LW-104 is sensitive to Cs137 at a level to 1E-6 µCi /cc, but it reads out in CPM on the RMS Rack in the control room. High-High alarm will isolate the discharge.

D. Correct. CPS is the correct meter units for RM-1LW-104. High-High alarm will close FCV-1LW-104-1 and 2, and TV-1LW-105 to isolate the discharge to prevent the accidental discharge.

Sys #	System	Category	KA Statement
000059	Accidental Liquid Radwaste Release / 9	AK1. Knowledge of the operational implications of the following concepts as they apply to Accidental Liquid Radwaste Release:	Types of radiation, their units of intensity and the location of the sources of radiation in a nuclear power plant

K/A# AK1.01 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-43.1.E Rev. 21 pg. 4
1OM-43.1.C Rev. 12 pg. 4
1OM-43.4.ACQ Rev. 5 pg. 2

Question Source: New

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

Objective: 1SQS-43.1-01-1: Describe the function of the Radiation Monitoring systems and the associated major components.
1SQS-43.1-01-8: Given a specific plant condition, predict the response of the Radiation Monitoring 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

22. Given the following plant conditions:

- RCS Tavg is 310°F and Pressure is 400 psig during a scheduled plant shutdown.
- Personnel air lock inner door has been declared INOPERABLE due to a bad seal.
- Personnel air lock outer door is locked closed and OPERABLE.
- All other containment penetrations are OPERABLE.
- Overall containment leakage rate has remained within acceptable limits.
- Tech Spec 3.6.2, CONTAINMENT AIR LOCKS has been entered.

IAW Tech Spec 3.6.2, CONTAINMENT AIR LOCKS, ______(1)______ is the lowest Mode in which the Personnel air lock doors are required to be operable.

Tech Spec 3.6.1, CONTAINMENT, _____(2)_____ met for the current conditions.

A. 1) Mode 3
2) is

B. 1) Mode 3
2) is NOT

C. 1) Mode 4
2) is

D. 1) Mode 4
2) is NOT

Answer: C

Explanation/Justification: K/A is matched because it requires the candidate to determine compliance with Containment Integrity specification T.S. 3.6.1 when one air lock door is inoperable with the other air lock door locked closed. Both Tech Spec 3.6.1 and 3.6.2 are 1 hour tech specs and it is expected that an RO candidate knows the requirements for operability of each.

A. Incorrect. The first part is plausible because some specifications are applicable in Modes 1, 2 and 3. The second part is correct per T.S. 3.6.1 is met even if an airlock is inoperable as long as compliance is maintained via the action statements.

B. Incorrect. The first part is plausible because some specifications are applicable in Modes 1, 2 and 3. The second part is plausible because if both airlock doors were inoperable, then 3.6.1 would not be met.

C. Correct. The candidate must determine that the plant is in mode 3 due to RCS Temperature is 350F > Tavg > 200F. Mode 4 is the lowest mode of applicability for T.S. 3.6.2. T.S. 3.6.1, Containment Integrity is applicable in modes 1-4, and is met with one containment air lock door inoperable as long as the actions of T.S. 3.6.2 are maintained.

D. Incorrect. First part is correct. The second part is plausible because if both airlock doors were inoperable, then T.S. 3.6.1 would not be met.

Sys #	System	Category	KA Statement
000069	Loss of Containment Integrity / 5	AK2. Knowledge of the interrelations between the Loss of Containment Integrity and the following:	Personnel access hatch and emergency access hatch

K/A# AK2.03 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** T.S 3.6.2 pgs. 3.6.2-1, 2
T.S. 3.6.1 pg. 3.6.1-1
T.S. 3.6.1 pg. B3.6.1-1

Question Source: Bank – 2013 VC Summer Q47

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

Objective: 3SQS-CONT ITS, Rev. 1 Obj. 3. Given plant conditions, determine the criteria necessary to ensure compliance with each Section Containment Systems LCO and Licensing Requirement in accordance with the Bases, Surveillance Requirements, and the Applicability.

23. The plant was operating at 100% power.

- A Reactor Trip and Safety Injection occur due to a Loss Of Coolant Accident
- Plant conditions require entry into procedure FR-C.2, Response To Degraded Core Cooling
- Step 13, Depressurize All Intact SGs To 110 psig, is being performed, the cooldown rate is limited to 100 °F/Hr

The following table shows the trend of Cold Leg Temperature and RCS Pressure:

TIME	Cold Leg Temp	RCS Pressure
10:00	330 °F	1750 psig
10:30	310 °F	1200 psig
11:00	250 °F	940 psig
11:30	230 °F	800 psig

Refer to attached EOP Attachment 5-A, RCS Cooldown Limits – Technical Specifications

At 1200 hours, which of the following set of plant conditions is allowable per the procedure?

A. 140 °F, 400 psig

B. 160 °F, 550 psig

C. 170 °F, 450 psig

D. 180 °F, 600 psig

Answer: C

Explanation/Justification: K/A is met with the candidate's ability to interpret the given control room parameters for Tcold and RCS pressure during a degraded core cooling event, and use the supplied reference for RCS Cooldown Limits to ensure the operator complies with the proper procedural directives.

A. Incorrect. The cooldown will exceed the 100 °F/ hr limit but is in the Acceptable region of the curve.
B. Incorrect. The cooldown will not exceed the 100 °F/ hr limit, but is in the Unacceptable area of the curve, it is below the 50 °F/hr limit line.
C. Correct. Must recognize the 100 °F/hr limitations and extrapolate that information on the C/D curve. The cooldown is less than 100 °F/Hr and in the Acceptable region for this rate.
D. Incorrect. The cooldown will not exceed the 100 °F/ hr limit but is in the Unacceptable region of the curve at the common point for the lower C/D Rates.

Sys #	System	Category	KA Statement
000074	Inadequate Core Cooling / 4	Generic	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

K/A# 2.2.44 **K/A Importance** 4.2 **Exam Level** RO

References provided to Candidate 1OM-53A.1.5-A Rev 3 **Technical References:** 1OM-53A.1.FR-C.2 Iss. 3 Rev. 0 pg. 7
1OM-53A.1.5-A Rev 3

Question Source: Bank – 1LOT14 Q25

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

Objective: 3SQS-53.3, Rev. 5 Obj. 6. Given a set of conditions, locate and apply the proper Emergency Operating Procedures, IAW BVPS-EOP Executive Volume.

24. Given the following conditions:

- The crew has entered E-1, Loss of Reactor or Secondary Coolant due to a LOCA.
- Containment pressure is 7.8 psig and slowly rising.
- The crew is at the step to, "Check if SI flow should be reduced," with plant parameters as follows:
    - RCS pressure is 1725 psig and stable
    - CETCs indicate 550°F
    - No AFW flow is available
    - SG NR levels are 'A' - 44%, 'B' – 52%, 'C' – 48%
    - PRZR level is 21% and slowly rising

Based on the current conditions, which one of the following actions are the operators required to take at this time?

A. Continue in E-1, Loss of Reactor or Secondary Coolant

B. Transition to ES-1.1, SI Termination

C. Transition to ES-1.2, Post-LOCA Cooldown and Depressurization

D. Transition to FR-H.1, Response to Loss of Secondary Heat Sink

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Answer: A

Explanation/Justification: K/A is met by the candidate's ability to determine that the crew must remain in E-1 because SI termination criteria, and other EOP transitions are not met at this time.

A. Correct. Continuing in E-1 is correct because containment is adverse, therefore PRZR level must be >38% (17% non-adverse). Subcooling is 66F (54F adverse), 'B' SG is 52% (50% required in 1 SG adverse), RCS pressure is stable.
B. Incorrect. Plausible if it is not recognized that containment is adverse, and the candidate recognizes that normal SI termination conditions are thought to be satisfied.
C. Incorrect. Plausible distractor as this is where the crew may eventually transition, but not until much later (step 20) of the E-1 procedure.
D. Incorrect. Plausible because there is no AFW flow available, but NR level in at least one SG must be >50% (31% non-adverse), and the question stem states that all NR SG levels are 44, 52, and 48%, therefore at least one SG is >50% and transition to FR-H.1 is not warranted.

Sys #	System	Category	KA Statement
WE02	SI Termination / 3	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** 3.3 **Exam Level** RO
References provided to Candidate None **Technical References:** 1OM-53A.1.E-1 Iss. 3 Rev. 1 Step 9 pg. 8
Question Source: Bank – 2013 Vogtle Q67
Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(10)
Objective: 3SQS-53.2, Rev. 2 Obj. 3. State from memory the basis for SI termination criteria, IAW BVPS EOP Executive Volume.

25. Given the following conditions:

- A small break LOCA has occurred
- PCB-92 and PCB-83, 138KV Bus 1 and 2 Power Circuit Breakers opened following the Reactor Trip
- Containment pressure is 6.7 psig and slowly rising
- The crew is currently performing ES-1.2, Post LOCA Cooldown and Depressurization.
- They are currently at Step 16 for depressurizing the RCS

1) Based on the current conditions, ___________ will be used for depressurizing the RCS?

2) The purpose of this step is to ensure the Pressurizer is refilled to a MINIMUM required level of greater than ___________.

A. 1) normal PRZR spray
2) 76%

B. 1) normal PRZR spray
2) 50%

C. 1) one PRZR PORV
2) 76%

D. 1) one PRZR PORV
2) 50%

Answer: D

Explanation/Justification: K/A is met with the candidate's knowledge for the reason of stopping RCS depressurization when the przr level rapidly rises while refilling the przr iaw ES-1.2 to ensure the procedural step note is adhered to which prevents the przr from going solid.

A. Incorrect. Normal spray is not available due to the loss of offsite power de-energizing the RCPs. 76% przr level is plausible because it is one of the depressurization stop criteria identified in E-3 when depressurizing the RCS to refill the przr.

B. Incorrect. Normal spray is not available due to the loss of offsite power de-energizing the RCPs. PRZR level must be raised to >50% since the containment is adverse (cnmt pressure >5 psig) in preparation of an RCP start

C. Incorrect. One PORV is correct since no RCPs are operating. 76% przr level is plausible because it is one of the depressurization stop criteria identified in E-3 when depressurizing the RCS to refill the przr.

D. Correct. No RCPs are available due to the loss of off-site power (PCB-92 and PCB-83 supply offsite power to the plant after the auto bus transfer occurs after a reactor trip), therefore one PORV must be used to depressurize the RCS to restore przr level. PRZR level must be raised to >50% since the containment is adverse (cnmt pressure >5 psig) in preparation of an RCP start

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Sys #	System	Category	KA Statement
WE03	LOCA Cooldown—Depressurization / 4	EK3. Knowledge of the reasons for the following responses as they apply to the (LOCA Cooldown and Depressurization)	RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

K/A# EK3.4 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.ES-1.2 Iss. 3 Rev. 2 pg. 11
1OM-53B.4.ES-1.2 Iss. 3 Rev. 2 pg. 2, 30, 31

Question Source: Modified -2014 Catawba Q25

Question Cognitive Level: Higher – 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.

26. Given the following plant conditions:

- 'A' S/G Pressure is 1150 psig
- 'A' S/G Narrow range level is 82%
- RCS hot leg temperatures are 563 °F
- FR-H.2, "Response to Steam Generator Overpressure" has been entered
- All MSIVs are CLOSED
- PCV-1MS-101A, 'A' SG ATM STM DUMP has failed CLOSED
- HCV-1MS-104, Residual Heat Release Valve is CLOSED and will not open
- 1FW-P-2, Turbine Driven AFW pump is out of service for bearing replacement

Which of the following describes the preferred method to reduce 'A' S/G pressure in accordance with FR-H.2?

A. Feed 'A' SG with AFW and commence an RCS cooldown to less than 525 °F using 'B' & 'C' Steam Generators.

B. Feed 'A' SG with AFW and establish Blowdown from the 'A' Steam Generator.

C. Isolate AFW to the 'A' SG and commence RCS cooldown to less than 525 °F using 'B' & 'C' Steam Generators.

D. Isolate AFW to the 'A' SG and establish Blowdown from the 'A' Steam Generator.

Answer: C

Explanation/Justification: K/A met by candidate's ability to operate the plant iaw FR-H.2, "Response to Steam Generator Overpressure" major action steps of Controlling the affected SG pressure and initiate C/D using the unaffected SGs. In the stem of the question the ASD is unavailable for 'A' SG, therefore the candidate must recognize that cooldown is required on the unaffected SGs.

A. Incorrect. Feeding the A SG is not permitted (or procedurally driven) because feed may be the cause of the overpressure. Cooling the RCS using the unaffected SGs is the correct answer if steam cannot be dumped from the affected SG.
B. Incorrect. Feeding the A SG is not permitted (or procedurally driven) because feed may be the cause of the overpressure. Establishing blowdown from A SG is not procedurally driven.
C. Correct. Major action steps of FR-H.2 are Control the affected SG pressure and initiate C/D using the unaffected SGs. Given the initial conditions in the stem it is determined that there are no means to control pressure in the affected SG, therefore it will be necessary to isolate AFW to the 'A' SG and cooldown using the other SG by dumping steam using B and/or C ADV.
D. Incorrect. Isolating AFW is correct but establishing blowdown from A SG is not procedurally driven.

Sys #	System	Category	KA Statement
WE13	Steam Generator Overpressure / 4	EA1. Ability to operate and / or monitor the following as they apply to the (Steam Generator Overpressure)	Operating behavior characteristics of the facility.

K/A# EA1.2 **K/A Importance** 3.0 **Exam Level** RO

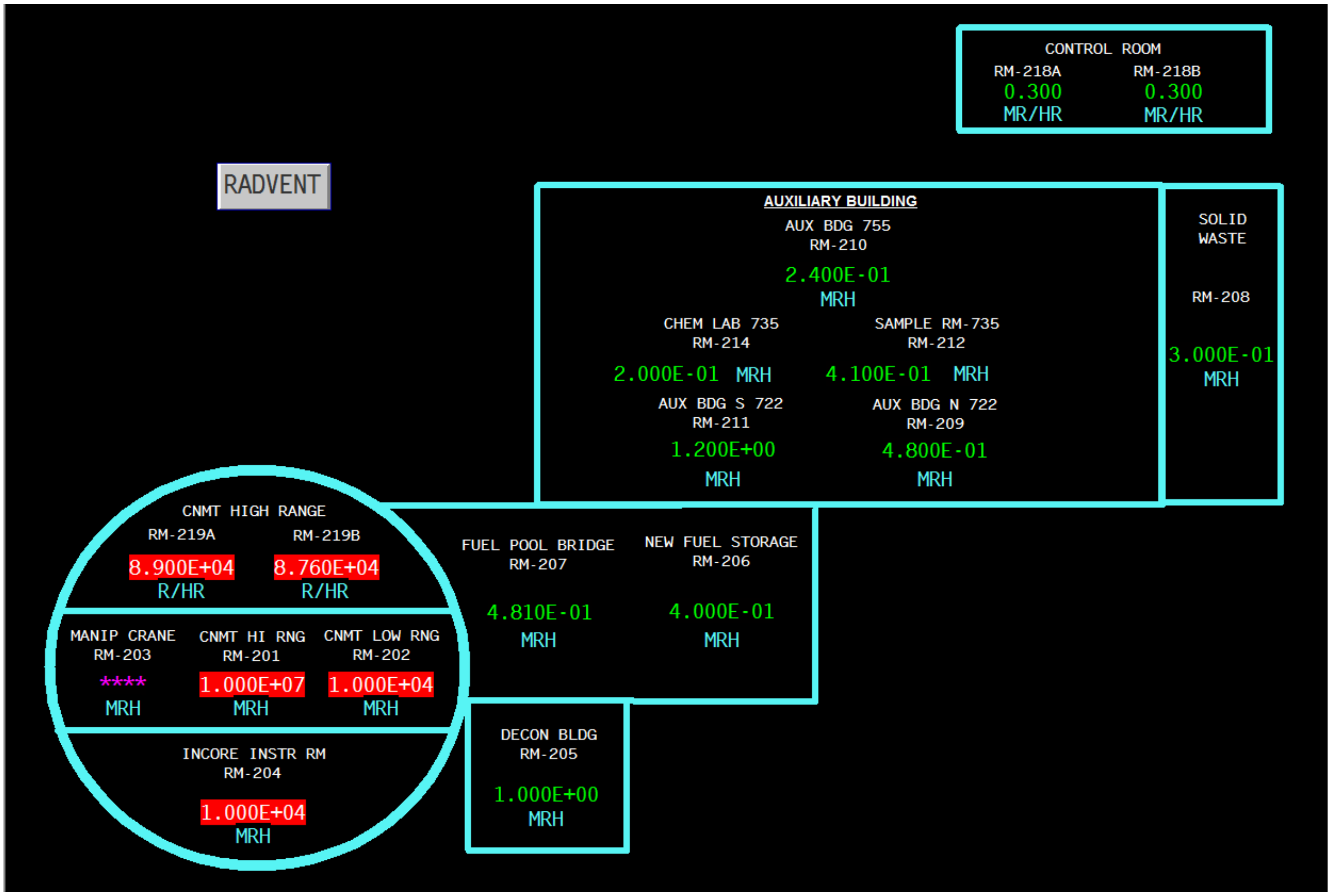
References provided to Candidate None **Technical References:** 1OM-53A.1.FR-H.2 Iss. 3 Rev. 0 pg. 3
1OM-53B.4.FR-H.2 Iss. 3 Rev. 0 pg. 2, 3

Question Source: Bank – 2LOT15 Q27

Question Cognitive Level: Higher – 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.

27.



1) Based **only** on the above IPC screen printout, and assuming the values observed are the peak radiation levels received during the event, is the Containment Adverse?

2) What is the basis for the Containment Radiation Adverse criteria?

A. 1) Adverse
2) To ensure conservative values for plant operation due to instrument inaccuracies.

B. 1) Adverse
2) To ensure plant operating conditions will not cause the 10CFR50.67 radiation limits in the Control Room to be exceeded.

C. 1) NOT Adverse
2) To ensure conservative values for plant operation due to instrument inaccuracies.

D. 1) NOT Adverse
2) To ensure plant operating conditions will not cause the 10CFR50.67 radiation limits in the Control Room to be exceeded.

Question 27

Answer: C

Explanation/Justification: K/A met by the candidate's ability to monitor containment radiation monitors and based on radiation levels determines if containment is adverse or not. Then demonstrate the knowledge of the bases for the containment high radiation operating characteristics on the plant instrumentation.

A. Incorrect. Plausible because radiation monitor RM-201 in cnmt is reading 1E+7 memr/hr, but this is incorrect because RM-1RM-219A(B) are not >1E+5 R/HR. Second part is correct.

B. Incorrect. Plausible because radiation monitor RM-201 in cnmt is reading 1E+7 memr/hr, but this is incorrect because RM-1RM-219A(B) are not >1E+5 R/HR. Second part is plausible because it is like the bases for the Steam generator tube leakage limits of tech specs.

C. Correct. Per the EOP network, containment is Adverse when radiation levels on RM-1RM-219A(B) are ≥1E+5 R/HR, therefore with RM-1RM-219A reading 8.9E+4 and RM-1RM-219B reading 8.76E+4, the containment is not adverse based on radiation levels. No assumptions can be made that the cnmt is adverse based on integrated dose rate because the stem states "peak radiation levels". Instrument inaccuracies based on cnmt environment is the correct basis for Containment Radiation Adverse criteria.

D. Incorrect. First part is correct. Second part is plausible because it is like the bases for the Steam generator tube leakage limits of tech specs.

Sys #	System	Category	KA Statement
WE16	High Containment Radiation /9	EK3. Knowledge of the reasons for the following responses as they apply to the (High Containment Radiation)	Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

K/A# EK3.1 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate IPC RADAREA screen **Technical References:** 1OM-53A.1.E-0 Iss 3 Rev 3 LHP; 1OM-53B.5.GI-2 Iss 2 Rev 0 pg. 12-13; 3SQS-53.2 HO Unit 1 53-2 Rev. 2 Issue 2 pg. 12

Question Source: Bank - 2LOT19 Q27

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

Objective: 3SQS-53.2, Rev. 2 Obj. 5. State from memory the basis for the foldout and left-hand page, IAW BVPS EOP Executive Volume.
3SQS-53.2, Rev. 2 Obj. 15. Define from memory adverse containment conditions, IAW BVPS EOP Executive Volume.

28. A plant startup is in progress with RCS Tavg at 285°F when an inadvertent Train ‘A’ CIA occurs.

    What affect will this have on the RCPs?

A. TV-1CC-103A, B, C (RCP CCR Inlet CNMT Isolation valves) will close isolating CCR to the RCP motor.

B. TV-1CC-107E1 (‘A’ RCP Thermal Barrier CCR Outlet Isolation valve) will close isolating CCR flow through the RCP thermal barrier.

C. MOV-1CH-308A, B, C (RCP Seal Injection Isolation valves) will close isolating Seal injection to #1 RCP seals.

D. MOV-1CH-378 (Seal Water Return CNMT Isolation valve) will close isolating RCP Seal Leakoff to the VCT.

Answer: D

Explanation/Justification: K/A is met with the candidate’s knowledge that a Train ‘A’ Containment Isolation Phase ‘A’ will cause the seal water return from the RCPs to isolate from the Volume control Tank.

A. Incorrect. Plausible distractor if the candidate thinks CCR will isolate to the RCP motors. TV-1CC-103A, B, C was chosen because they do auto close on a CIB isolation.
B. Incorrect. Plausible distractor if the candidate thinks CCR will isolate through the thermal barriers on a CIA. TV-1CC-107E1 was chosen to appear to be train ‘A’ specific, but this valve does not auto close on CIA, but it does auto close on CIB.
C. Incorrect. Plausible distractor if the candidate thinks seal injection isolates on CIA. MOV-1CH-308A, B, C was chosen because they are the isolation valves to each RCP but have no automatic closures associated with them.
D. Correct. MOV-1CH-378 closes on a train ‘A’ CIA signal causing seal leakoff to isolate to the VCT. This will cause RV-1CH-382A to open and relieve to the PRT.

Sys #	System	Category	KA Statement
003	Reactor Coolant Pump	K1 Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the following systems:	Containment isolation

K/A# K1.08 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** OM Fig. 7-4 (RM-0407-004 Rev. 30
1SQS-6.3 Rev.15 Iss. 1 pg. 51, 52

Question Source: New

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

Objective: 1SQS-6.3 Rev. 15 Obj. 7. Given a Reactor Coolant Pump and support system configuration and without referenced material, describe the Reactor Coolant Pump field response to the following actuation signals, including automatic functions and changes in equipment status. a. Safety Injection, b. Containment Isolation – Phase A, c. Containment Isolation – Phase B

29. ‘A’ RCP #2 Seal has shown signs of degraded performance and engineering suspects failure may be imminent.

Which of the following alarms would confirm failure of the #2 RCP seal?

1) ‘A’ RCP Seal Vent Pot Level ________ alarm.
2) ‘A’ RCP Seal Leakoff Flow ________ alarm.

A. 1) high
2) high

B. 1) high
2) low

C. 1) low
2) high

D. 1) low
2) low

Answer: B

Explanation/Justification: The KA is met with the candidate’s knowledge that a Reactor Coolant Pump #2 seal failure will cause seal leakoff flow to lower, and the seal vent pot level to rise due to the RCP seal configuration.

A. Incorrect. First part is correct. Second part is plausible because a #1 seal failure would result in increased seal leakoff flow.
B. Correct. A failure of the #2 seal will result in an increased flow through the seal causing Seal Vent Pot Level to rise to the high level setpoint. The #2 seal failure will redirect seal leakoff flow through the #2 seal causing seal leakoff flow to lower.
C. Incorrect. First part is plausible because a #3 seal failure would result in decreased flow to the Seal Vent Pot causing a low level to occur. Second part is plausible because a #1 RCP Seal failure would result in high seal leakoff flow.
D. Incorrect. First part is plausible because a #3 seal failure would result in decreased flow to the Seal Vent Pot causing a low level to occur. Second part is correct.

Sys #	System	Category	KA Statement
003	Reactor Coolant Pump	K6 Knowledge of the effect of a loss or malfunction on the following will have on the RCPS:	RCP seals and seal water supply

K/A# K6.02 **K/A Importance** 2.7 **Exam Level** RO
References provided to Candidate None **Technical References:** 1SQS-6.3 PPNT Rev. 15 Iss. 1 Slide 53
1OM-6.4.ABC Rev. 7 pg. 5

Question Source: Bank – 2015 Catawba Q28

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

Objective: 1SQS-6.3 Rev 15 Obj. 20. Given a specific plant condition, predict the response of the Reactor Coolant Pump and support 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.

30. Initial conditions:
    - The plant is at 75% power with all systems in normal alignment for this power level.

    Current conditions:
    - A 10% load rejection occurs.

    What is the **initial** effect on the Pressurizer and Charging flow?

A. Pressurizer pressure will lower due to resultant out-surge and charging flow will rise.

B. Pressurizer pressure will rise due to the resultant in-surge and charging flow will lower.

C. Pressurizer surge line temperature will rise due to the resultant in-surge and charging flow will lower.

D. Pressurizer surge line temperature will lower due to the resultant out-surge and charging flow will rise.

Answer: B

Explanation/Justification: K/A is met with the candidates understanding of the operational implications on the pressurizer during a load rejection which causes the pressurizer pressure and level to rise due to the in-surge caused by the increased RCS temperature and the effect it will have on the CVCS system.

A. Incorrect. Plausible answer if the candidate doesn't understand the integrated plant response that occurs during a load rejection. See correct answer.
B. Correct. Przr pressure and level will initially rise due to the przr in-surge caused by the RCS temperature rise which is caused by less heat being removed by the turbine after the load rejection. The question asks for the initial response because the control rods will automatically drive in to lower RCS temperature, and the przr pressure master controller will lower pressure, as the przr master level controller will restore przr level to the new reference level.
C. Incorrect. Plausible distractor if the candidate doesn't understand that an in-surge due to the load rejection will actually cause the surge line temperature to lower due to cooler water being pushed into the przr from the RCS loop. Charging flow will lower on a load rejection due to an in-surge occurring.
D. Incorrect. Plausible distractor if the candidate doesn't understand that an out-surge doesn't occur on a load rejection, and that an out-surge would actually cause the surge line temp to rise. Charging flow will lower on a load rejection due to an in-surge occurring.

Sys #	System	Category	KA Statement
004	Chemical and Volume Control	K5 Knowledge of the operational implications of the following concepts as they apply to the CVCS:	Pressure response in PZR during in-and-out surge

K/A# K5.44 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1SQS-6.4 Rev. 14 LP pg. 66, 67
1SQS-6.4 Rev. 14 PPNT slide 65

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(5)

Objective: 1SQS-6.4 Rev. 14 Obj. 6. Given a change in plant conditions, describe the response of the Pressurizer and Pressurizer Relief System field indication and control loops, including all automatic functions and changes in equipment status.
GO-3ATA 3.1, Rev. 3 Obj. 1. Predict and analyze the plant response (TAVG, Reactor Power, Net Reactivity, Pressurizer Pressure, Pressurizer level, Steam Generator Pressure, Steam Generator Level, and Steam Flow) to the following transients. b. Ten percent power changes. c. Large load rejections.

31. Given the following:

- A Large Break LOCA has occurred.
- The crew has entered the EOP network.
- RWST level is 13.2' and lowering slowly.

Assuming no operator action, what is the current lineup of the Safety Injection system?

HHSI pumps taking suction from ______, LHSI pumps taking suction from ______, HHSI pumps discharging to ______ leg injection line, LHSI pumps discharging to ______ leg injection line.

	HHSI pump suction	LHSI pump suction	HHSI discharging to:	LHSI discharging to:
A.	RWST	RWST	Cold leg injection	Cold leg injection
B.	RWST	RWST	Hot leg injection	Cold leg injection
C.	LHSI discharge	CNMT sump	Cold leg injection	Hot leg injection
D.	LHSI discharge	CNMT sump	Cold leg injection	Cold leg injection

Answer: D

Explanation/Justification: NRC exam Chief said to write a question keep with the intent of the K/A, therefore, instead of the RHR system piggy-backing with HHSI which is not possible at BV, the question is written as LHSI piggybacking with HHSI.

K/A is met with the candidate's knowledge that when the RWST level reaches the automatic transfer setpoint of the RWST, the SI system will automatically shift the suction of the LHSI pumps aligning to the cnmt sump, and the HHSI pumps taking suction from the LHSI pump discharges to recirc the cnmt sump to the reactor vessel.

A. Incorrect. Plausible distractor since this is the SI lineup prior to the auto transfer to recirc mode at RWST level of <14' 0.5".

B. Incorrect. Plausible distractor if it is thought that the SI lineup prior to the auto transfer to recirc mode at RWST level of <14' 0.5" is split discharge to both hot and cold leg, and the candidate doesn't know the recirc mode setpoint.

C. Incorrect. Plausible distractor since these are the correct suctions when RWST is <14' 0.5", but the discharge lineup is the hot and cold leg recirc lineup which occurs 6.5 hrs. after the LOCA occurred.

D. Correct. When the RWST level reached <14' 0.5", auto SI transfer to recirculation mode will align the LHSI pump suction to the cnmt sump, and the HHSI pumps to take suction from the LHSI pump discharge, and isolate the RWST. Both HHSI and LHSI discharge to the cold legs.

Sys #	System	Category	KA Statement
005	Residual Heat Removal	K4 Knowledge of RHRS design feature(s) and/or interlock(s) which provide or the following:	Lineup for "piggy-back" mode with high-pressure injection

K/A# K4.08 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.5.B.8 Rev. 0 pg. 6
1SQS-11.1 PPNT Rev. 15 Slide 93

Question Source: New

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

Objective: 1SQS-11.1-01-18 Rev. 15 Describe the Safety Injection System flow path(s) from the suction source (RWST or CNMT Sump) to the Reactor Core for the following ESF Core Cooling Phases: a. Cold Leg Injection, b. Cold Leg Recirculation, c. Simultaneous Hot and Cold Leg Recirculation

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

    - The plant is operating at 100% power with all systems in NSA.
    - A Main Turbine trip occurred causing a Rx Trip
    - Normal 4KV Bus 'D' failed to energize on the transfer to off-site power
    - A9-12, DIESEL GEN 2 OVERCURRENT PROTECTION is LIT

    Two minutes after the Rx Trip, a Large Break LOCA occurred.

    - 'A' Charging Pump has a bright white light lit on BB-A

    Based on the given conditions, what flowrate will be indicated on HI HEAD SI TO BIT FLOW indicator FI-1SI-943?

A. 0 gpm

B. 150 gpm

C. 550 gpm

D. 900 gpm

Answer: A

Explanation/Justification: K/A is met with the candidate's ability to recognize that with the given plant conditions, no High Head Safety Injection pumps are running causing a loss of High Head Injection flow to the RCS Cold Legs. Exam chief said to address the first part of the K/A because the second part cannot be met without making the question SRO level. The intent of the overall ECCS flow path is met for the loss of the HHSI flow path.

A. Correct. 0 gpm is correct because the candidate must recognize that no HHSI pumps are running because 'A' HHSI pump tripped (bright white light is lit indicating pump is tripped), and Emergency 4KV bus DF is de-energized because the normal 4KV bus 'D' failed to energize, and the #2 Diesel Generator cannot start due to #2 DG overcurrent.
B. Incorrect. Plausible because 150 gpm is the approximate design flowrate at 2500 psig.
C. Incorrect. Plausible because 550 gpm is the approximate runout flow of one HHSI pump.
D. Incorrect. Plausible because 900 gpm is the approximate runout flow of two HHSI pumps.

Sys #	System	Category	KA Statement
006	Emergency Core Cooling	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of flow path

K/A# A2.02 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-7.1.C Rev. 9 pg. 2
1SQS-11.1 PPNT Rev. 15 Slide 43
1OM-36.4.AEH Rev. 4 pg. 2

Question Source: New

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

Objective: 1SQS-7.1, Rev. 20 Obj. 4. Identify the power supplies for the components identified on the Normal-System-Arrangement System Flow path drawing which are powered from the class 1E electrical distribution system.

3SQS-36.1, Rev. 12 Obj. 16. Given a specific plant condition, predict the response of the 4KV Distribution 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.

33. What is the power supply to the SI accumulator discharge valves MOV-1SI-865B & C?

A. Bus 8N

B. Stub Bus 8N1

C. Bus 9P

D. Stub Bus 9P1

Answer: C

Explanation/Justification: K/A is met with candidate's knowledge of the power supply to the SI accumulator discharge valves.

A. Incorrect. Plausible because MOV-1SI-865A is powered from MCC1E-5 which is supplied from the 8N bus, and it could be mistaken because the question is asking what the power supply for MOV-1SI-865C is.
B. Incorrect. Plausible because Stub Bus 8N1 is 1 of 4 emergency busses available.
C. Correct. MOV-1SI-865B & C are powered from MCC1E-6 which is supplied from Bus 9P.
D. Incorrect. Plausible because Stub Bus 9P1 is 1 of 4 emergency busses available.

Sys #	**System**	**Category**	**KA Statement**
006	Emergency Core Cooling	K2 Knowledge of bus power supplies to the following:	Valve operators for accumulators

K/A# K2.02 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1SQS-11.1 PPNT Rev. 15 Slide 67
3SQS-37.1 U1 PPNT Rev.9 Iss.2 Slide 24

Question Source: New

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

Objective: 1SQS-11.1-01-20 Identify the power supplies for the components identified on the Normal-System-Arrangement System Flow-path drawing which are powered from the class 1E electrical distribution system. (For the 4160v system, include the power train and bus designation. For the 480v system, include only the power train.)

34. Given the following plant conditions:

- A Load Rejection has occurred.
- Pressurizer (PRZR) PORV operation has resulted in high pressure and temperature in the PRZR Relief Tank.
- Annunciator A4-38, PRESSURIZER RELIEF TANK TEMP HIGH" is received due to high tank temperature above 125°F.
- The PORV has reseated.

Which of the following describes the operational design features which provide PRZR Relief Tank (PRT) Cooling?

1) MOV-1RC-516, PRT Spray Valve ________________.
2) TV-1RC-519, PRT Primary Water Supply Isolation Valve ________________.

A. 1) automatically opens
   2) automatically opens

B. 1) automatically opens
   2) must be manually opened

C. 1) is NSA open
   2) automatically opens

D. 1) is NSA open
   2) must be manually opened

Answer: D

Explanation/Justification: K/A is met with the knowledge that the PRZR Relief Tank is designed to use primary water to spray down to condense and cool a discharge of the przr steam.

A. Incorrect. Neither valve is designed to automatically open.
B. Incorrect. MOV-1RC-516 is NSA open and there are no automatic features associated with the valve. TV-1RC-519 must be manually opened.
C. Incorrect. MOV-1RC-516 is NSA open, but TV-1RC-519 must be manually opened. There are no automatic features associated with the valve for PRT pressure or temperature.
D. Correct. MOV-1RC-516 is NSA open and there are no automatic features associated with the valve. TV-1RC-519 must be manually opened and there are no automatic features associated with this valve. In accordance with 1OM-6.4AAB (ARP A4-38) 516 is checked open, and 519 is manually opened from the control room. Both valves must be open to reduce tank temperature.

Sys #	System	Category	KA Statement
007	Pressurizer Relief/Quench Tank	K4 Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following:	Quench tank cooling

K/A# K4.01 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.4.AAB Rev 2 pg. 2
1OM-6.3.B.1 Rev 20 pg. 18

Question Source: Bank - 1LOT18 Q33 (LAST 2 EXAMS)

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

Objective: 1SQS-6.4, Rev. 14 Obj. 6. Given a change in plant conditions, describe the response of the Pressurizer and Pressurizer Relief System field indication and control loops, including all automatic functions and changes in equipment status

35. Complete the following statements regarding the Reactor Plant Component Cooling Water system.

Component Cooling Surge Tank Level Control valve, LCV-1CC-100A supplies make-up water from the ______(1)______ system to maintain the CCR Surge Tank (1CC-TK-1) normal operating range of ______(2)______.

A. 1) Domestic Water
   2) 24-36 inches

B. 1) Primary Water
   2) 24-36 inches

C. 1) Domestic Water
   2) 68-76 inches

D. 1) Primary Water
   2) 68-76 inches

Answer: B

Explanation/Justification: K/A is met with the candidate's ability to monitor the automatic operation of the Component Cooling Water surge tank level control valve to maintain the CCR surge tank within the normal operating range.

A. Incorrect. Domestic water is incorrect, but plausible because it is a clean water source which is available in the auxiliary building. CCR surge tank level normal operating range is 24-36 inches.
B. Correct. Primary water is supplied to the CCR surge tank via LCV-1CC-100A. CCR surge tank level normal operating range is 24-36 inches.
C. Incorrect. Domestic water is incorrect, but plausible because it is a clean water source which is available in the auxiliary building. Normal operating range of 68-76 inches is incorrect, but plausible because it is the Turbine Plant Component Cooling Water surge tank normal operating range.
D. Incorrect. Primary water is supplied to the CCR surge tank via LCV-1CC-100A. Normal operating range of 68-76 inches is incorrect, but plausible because it is the Turbine Plant Component Cooling Water surge tank normal operating range.

Sys #	System	Category	KA Statement
008	Component Cooling Water	A3 Ability to monitor automatic operation of the CCWS, including:	Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS

K/A# A3.01 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1SQS-15.1 Rev.14 pg. 19, 20
U1 RM-0415-001 Rev. 24

Question Source: New

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

Objective: 1SQS-15.1 Rev. 14 Obj. 4. State the purposes of the CCR surge tank and the makeup sources to the surge tank.
1SQS-15.1 Rev. 14 Obj 10. Given a change in CCR system surge tank level, summarize how the system will automatically respond to the change in level.

36. Given the following conditions:

- Plant is in MODE 3
- RCS Tave is 547°F and stable
- PRZR pressure is 2235 psig and stable
- PRT pressure is 20 psig and stable

Subsequently:

- PT-1RC-445 fails HIGH

1) Which PORV(s) opened due to this failure?

2) What is the state of the fluid downstream the PORV if a PORV fails to reseat and causes PRZR pressure to lower to 1500 psig?

A. 1) PCV-1RC-455C
2) Saturated steam

B. 1) PCV-1RC-455C
2) Superheated steam

C. 1) PCV-1RC-455D and PCV-1RC-456
2) Saturated steam

D. 1) PCV-1RC-455D and PCV-1RC-456
2) Superheated steam

Question 36

Answer: D

Explanation/Justification: K/A is met with the candidates knowledge that a leaking PORV will act to throttle the fluid from 1500 psig (or lower in this case) to the PRT which is at 20 psig, and knowing that throttling is constant enthalpy process, they will be able to use the Mollier diagram to determine that the fluid is superheated because it is above the saturation line.

A. Incorrect. PCV-1RC-455C would open if PT-1RC-444 failed high. Saturated steam is plausible if the candidate doesn't understand the mollier diagram or believes that since it is saturated in the PRZR, it will be saturated downstream of the PORV. The downstream fluid would be saturated is initial pressure was greater than ~1700 psig.

B. Incorrect. PCV-1RC-455C would open if PT-1RC-444 failed high. Second part is correct, see correct answer.

C. Incorrect. The PCS will open PCV-1RC-455D and PCV-1RC-456 when pressure on PT-1RC-445 goes above 2335 psig. Second part is incorrect, see incorrect answer justification above.

D. Correct. The PCS will open PCV-1RC-455D and PCV-1RC-456 when pressure on PT-1RC-445 goes above 2335 psig. The fluid at the outlet of the PORV will be superheated steam due to the constant enthalpy process caused by the throttling characteristics of the PORV failing to seat. The candidate will have to find 1500 psig (or lower pressure due to the PORV leaking) on the Saturation line, and follow it across to the right on the enthalpy line until it intersects with the 20 psig line, which is in the superheated region.

Sys #	System	Category	KA Statement
010	Pressurizer Pressure Control	K5 Knowledge of the operational implications of the following concepts as the apply to the PZR PCS:	Constant enthalpy expansion through a valve

K/A# K5.02 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate Steam Tables/ Mollier diagram **Technical References:** 1OM-6.4.IF Rev. 11 pg. 23
Steam Tables/Mollier diagram

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(14)

Objective: 1SQS-6.4 Rev. 14 Obj. 17. Describe the control, protection and interlock functions for the control room components associated with the Pressurizer and Pressurizer Relief System, including automatic functions, setpoints and changes in equipment status as applicable.

GOGPF-T4 Rev. 1 Iss. 3 Obj. 22. Explain the reduction of process pressure from throttling using an enthalpy-entropy (h-s) diagram or temperature-entropy (T-s) diagram.

37. The plant is at 100% power with all systems NSA **EXCEPT**, PCV-1RC-455A, PRZR Spray Valve is in its FAIL position due to a broken air line.

- RCS Pressure is 2235 psig and STABLE.
- RCS Tavg is 578°F and STABLE.
- PT-1RC-444, “Pressurizer (PRZR) Control Channel”, fails **HIGH** over a one minute period.
- PRZR Backup Heater Group 2A and 2B control switches are RED Targeted.

With no operator action, which of the following describes how the PRZR Pressure Control System will **INITIALLY** respond?

A. ONLY TWO (2) PRZR PORVs will be OPEN.

B. ONLY ONE (1) PRZR PORV and ONE (1) PRZR Spray Valve will be OPEN.

C. ALL PRZR B/U heaters will be OFF and TWO (2) PRZR PORVs will be OPEN

D. ALL PRZR B/U heaters will be OFF and ONE (1) PRZR Spray Valve will be OPEN.

Answer: B

Explanation/Justification: K/A is met with the knowledge that air operated spray valve PCV-1RC-455A will fail closed with a broken air line and will not respond when the pressurizer control system demands it to open due to PT-1RC-444 failing high.

A. Incorrect. These indications are indicative of PT-1RC-445 failing in the high direction. Plausible if the candidate confuses the PRZR pressure control system inputs or does not understand the impact of the items which are OOS for this system.
B. Correct. A failure of PT-1RC-444 in the high direction will typically result in PCV-1RC-455C opening and both PRZR spray Valves failing full open. PCV-1RC-455A fails closed on a loss of air, therefore only PCV-1RC-455B will open
C. Incorrect. Correct PRZR B/U heater response except that NSA two heaters will be ON, so therefore not all heaters will be OFF. The two PORVs are controlled by PT 445 not PT444, if the candidate misdiagnoses the transmitter failure, this distractor is plausible
D. Incorrect. Correct PRZR B/U heater response except that NSA two heaters will be ON, so therefore not all heaters will be OFF. At 100% power, PCV-1RC-455A fails closed on a loss of air, therefore PCV-1RC-455B will open.

Sys #	System	Category	KA Statement
010	Pressurizer Pressure Control	K6 Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS:	PZR sprays and heaters

K/A# K6.03 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.34.1 Rev. 28 pg. 11
1om-6.4.IF Rev. 11 pgs. 19, 23, 24

Question Source: Bank – 2LOT19 Q36

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(3)

Objective: ISQS-6.4 Rev. 14 Obj. 20. Given a specific plant condition, predict the response of the Pressurizer and Pressurizer Relief 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. c. Process Instrument Failure

38. Given the following:

- A plant startup is in progress and all Nuclear Instrumentation is observed to be operating normally.
- Power Range channel N-41 and N-42 are 9%.
- Power Range channel N-43 and N-44 are 11%.
- NO manual actions have been taken.

Which of the following is correct concerning RPS trips?

A. Power Range, high setpoint trip and Source Range high flux trip are enabled.

B. Power Range, low setpoint trip and Source Range high flux trip are disabled.

C. Power Range, high setpoint trip is enabled and Intermediate Range high flux trip is enabled.

D. Power Range, low setpoint trip is enabled and Intermediate Range high flux trip is disabled.

Answer: C

Explanation/Justification: K/A is met by the candidate's ability to interpret what reactor protection trips are enabled or disabled based on the reactor power range indications in the control room.

A. Incorrect. The Power Range high setpoint trip is always enabled, but the Source Range was should have been manually blocked >P-6, or it is automatically blocked when >P-10.
B. Incorrect. The Power Range low setpoint trip is still enabled because it has not been manually blocked which is permitted when >P-10. Source Range is disabled either manually at >P-6, or automatically when >P-10.
C. Correct. The Power Range high setpoint trip is always enabled and the Intermediate Range high flux trip is still enabled since it has not been manually blocked (no manual actions taken as stated) when >P-10 (2/4 Power Range >10%).
D. Incorrect. The Power Range low setpoint trip is still enabled because it has not been manually blocked which is permitted when >P-10. The IR high flux trip is not disabled because it must be manually blocked (no manual actions taken as stated) when >P-10 (2/4 Power Range >10%).

Sys #	System	Category	KA Statement
012	Reactor Protection	A4 Ability to manually operate and/or monitor in the control room:	Channel blocks and bypasses

K/A# A4.03 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.1.B Rev. 10 pg. 5, 6
1OM-1.5.B.8 Rev. 0 pg. 3, 5

Question Source: Bank - 1LOT7 Q17

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

Objective: 3SQS-1.1 Rev. * Obj.9. Describe the control, protection and interlock functions for the control room components associated with the Reactor Protection System Trip Logics & Engineered Safeguards Features Actuation Signals, including automatic functions, setpoints and changes in equipment status as applicable.

39. Given the following conditions:

- A Large Break LOCA occurred 7 hours ago
- The TSC has directed the crew to transition to ES-1.4, Transfer to Simultaneous Cold and Hot Leg Recirculation.
- ES-1.4 has been completed

At the completion of ES-1.4, what will be the charging/HHSI pump discharge flowpath(s)?

Charging/HHSI pumps will discharge to the ____________________.

A. hot leg injection only

B. cold leg injection only

C. cold leg and hot leg injection only

D. charging header, cold leg injection, and hot leg injection

Answer: B

Explanation/Justification: K/A is met with the candidate's knowledge of how the Safety Injection system and the CVCS (Charging/HHSI pumps) system is aligned to inject flow into the Cold Leg injection lines 6.5 hours after an event when ES-1.4, Transfer to Simultaneous Cold and Hot Leg Recirculation is performed.

A. Incorrect. Plausible distractor if the candidate doesn't know how Safety Injection flow is aligned to the reactor vessel in ES-1.4. Charging/HHSI pumps are aligned to the cold leg headers, and LHSI pumps are aligned to the hot leg injection headers.
B. Correct. ES-1.4 continues to have the Charging/HHSI pumps aligned to the cold leg injection headers, The Charging flowpath was isolated by the initial SI actuation, and LHSI pumps discharge to the HHSI pump suctions and the hot leg injection headers.
C. Incorrect. Plausible distractor if the candidate thinks that the Charging/HHSI pumps align to both cold and hot legs when in ES-1.4.as is inferred in the procedure title.
D. Incorrect. Plausible distractor if the candidate doesn't know that charging header is isolated on Safety Injection and thinks that Charging HHSI pumps are aligned to both the hot and cold leg injection headers.

Sys #	System	Category	KA Statement
013	Engineered Safety Features Actuation System	K1 Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems:	CVCS

K/A# K1.11 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-11.1.B Rev. 4 pg. 3
1SQS-11.1 PPNT Rev. 15 slide 109

Question Source: New

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

Objective: 1SQS-11.1-01-18 Rev. 15 Describe the Safety Injection System flow path(s) from the suction source (RWST or CNMT Sump) to the Reactor Core for the following ESF Core Cooling Phases: a. Cold Leg Injection, b. Cold Leg Recirculation, c. Simultaneous Hot and Cold Leg Recirculation

40. Given the following conditions:

- The plant is at 100% power
- PT-1LM-100D, Containment Pressure Channel IV pressure indication was oscillating and has been removed from service IAW 1OM-1.4.IF, “Instrument Failure Procedure”

Which of the following identifies the logic associated with the HIGH and HIGH-HIGH Containment Pressure actuations after the Channel IV is removed from service?

	HIGH Cnmt Press SI Actuation	HIGH-HIGH Cnmt Press CIB Actuation
A.	1/2	2/3
B.	1/2	1/3
C.	1/3	2/3
D.	1/3	1/3

Answer: A

Explanation/Justification: K/A is met by demonstrating knowledge of the containment pressure channel inputs to both Safety injection and Cnmt Isol phase B actuation logics, and how these inputs are removed from service for the reliability of the actuation coincidence.

A. Correct. Channel IV (PT-1LM-100D) was removed from service iaw 1OM-1.4.IF. This procedure and Tech Specs has the bistable tripped for High Cnmt Press (SI) which then makes the logic 1/2. The bistable for the High-High Cnmt Press (CIB) is required to be placed in bypass, which then requires a 2/3 coincidence to initiate a CIB. Both of these bistable configurations satisfies redundancy requirements.
B. Incorrect. Plausible if both bistables are tripped. High Cnmt Press (SI) is normally a 2/3 logic but changes to 1/2 when one of the logic channels are tripped. For High-High Cnmt Press (CIB), 3/4 logic, the channel is bypassed, so 2/3 is required.
C. Incorrect. Plausible if the logic for both are 2/4 and only High Cnmt Press is tripped.
D. Incorrect. Plausible if the logic for both are 2/4 and both are tripped.

Sys #	System	Category	KA Statement
013	Engineered Safety Features Actuation	K5 Knowledge of the operational implications of the following concepts as they apply to the ESFAS:	Safety system logic and reliability

K/A# K5.02 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.4.IF Att. 1 Rev. 9 pg. 5-7
BVPS TS Bases pg. B3.3.2-38 & 39

Question Source: Bank – 2LOT15 Q38

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

Objective: 3SQS-1.1 Rev. 8 Obj. 3. Given a change in plant conditions, describe the response of the Reactor Protection System Trip Logics & Engineered Safeguards Features Actuation Signals field indication and control loops, including all automatic functions and changes in equipment status.

41. Which signal will automatically CLOSE **ALL** of the following Chilled Water valves to the Containment Air Recirculation Cooling Coils?

    TV-1CC-110E2, CNMT Recirc Clg Coils AC Sys Inlet CNMT Isol Valve
    TV-1CC-110E3, CNMT Recirc Clg Coils AC/CCW Inlet CNMT Isol Valve
    TV-1CC-110D, CNMT Recirc Clg Coils AC/RW Outlet CNMT Isol Valve
    TV-1CC-110F2, CNMT Recirc Clg Coils AC Sys Outlet CNMT Isol Valve

A. SI only

B. CIA only

C. CIB only

D. MSLI only

Answer: C

Explanation/Justification: K/A is met by the candidate demonstrating knowledge of the Chilled Water system isolation to the Containment air recirculation (CAR) cooling coils when a CIB actuation signal occurs.

A. Incorrect. Cnmt cooling remains aligned during Safety Injection.
B. Incorrect. Cnmt cooling remains aligned during a CIA actuation.
C. Correct. The Chill Water cnmt inlet and outlet isolation valves all close upon a receipt of a CIB signal. This isolates chilled water cooling to the containment air recirculation heat exchangers which provide cooling to containment.
D. Incorrect. Cnmt cooling remains aligned during a MSLI actuation.

Sys #	System	Category	KA Statement
022	Containment Cooling	K1 Knowledge of the physical connections and/or cause effect relationships between the CCS and the following systems:	Chilled water

K/A# K1.04 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1SQS-44C.1 PPNT Rev. 10 Iss. 1 Slide 14
1OM-53A.1.1-E Rev. 6 pg. 2, 3

Question Source: Bank – 2LOT19 Q41

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

Objective: 1SQS-44C.1 Rev. 10 Obj. 5. Given a Containment Ventilation System configuration and without reference material, describe the Containment Ventilation System field response to the following actuation signals, including automatic functions and changes in equipment status. a. Containment Isolation Phase “B”

42. Given the following conditions:

- The plant was at 75% power when a large break LOCA occurred.
- LT-1QS-100A (RWST level ch. III) and LT-1QS-100C (RWST level ch. I) are not responding to RWST level change due to being frozen during extreme cold weather.

Due to these failures, Recirc Spray pumps ______(1)______ auto start, and the automatic SI transfer to recirculation mode ______(2)______ occur when the appropriate RWST level setpoints are reached.

A. 1) will
   2) will

B. 1) will
   2) will NOT

C. 1) will NOT
   2) will

D. 1) will NOT
   2) will NOT

Answer: C

Explanation/Justification: K/A is met with the candidate's knowledge of the effect of two of the four RWST level transmitters (Quench Spray system) failing as-is will have on the automatic start of the Recirculation spray system during a large break LOCA. BV1 has both Quench Spray and Recirc Spray systems.

A. Incorrect. Plausible distractor. See correct answer explanation.
B. Incorrect. Plausible distractor. See correct answer explanation.
C. Correct. Recirc Spray pumps will auto start when 2/3 RWST levels reach < 27' 7.5" coincident with a CIB. The candidate must know that RWST level transmitters 100A, B, & C are used for this logic, therefore the Recirc spray pumps will not start. Transfer to Recirc automatically occurs when 2/4 RWST levels reach < 14' 0.5" coincident with a Si signal. In this case the candidate must know that RWST level transmitters 100A, B, C, & D are used for this logic, therefore the automatic SI transfer to recirculation will occur. Candidate must know that a large break LOCA will actuate both an SI actuation and a CIB actuation.
D. Incorrect. Plausible distractor. See correct answer explanation.

Sys #	System	Category	KA Statement
026	Containment Spray	K3 Knowledge of the effect that a loss or malfunction of the CSS will have on the following:	Recirculation spray system

K/A# K3.02 **K/A Importance** 4.2 **Exam Level** RO
References provided to Candidate None **Technical References:** 1SQS-13.1 PPNT Rev. 15 Slides 26, 37
U1 LSK-026-002C Rev. 11
U1 LSK-027-001A Rev. 11
U1 LSK-027-001B Rev. 10

Question Source: Bank – 2014 North Anna Q42

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

Objective: 1SQS-13.1 Rev 15 Obj. 18. Describe the control, protection and interlock functions for the control room components associated with the Containment Depressurization System, including automatic functions, set points and changes in equipment status as applicable.

43. The plant has tripped from 100% power coincident with a loss of offsite power.

- Both Emergency Diesel Generators have failed to start
- Procedure ECA-0.0, Loss of All Emergency 4KV AC Power has been entered
- Instrument Air Pressure is 0 psig
- Chemistry reports that there is high RCS activity due to failed fuel

1) **IF** a Steam Generator Tube Rupture were to subsequently occur, which Radiation Monitors would be used to identify the ruptured Steam Generator?

2) **IF** the Steam Generator pressure rises due to decay heat, with no operator actions, what is the highest pressure that will exist in the ruptured Steam Generator, assuming all systems function as designed?

A. 1) N-16 Steam Generator Leak Monitor (RM-1MS-102A, B, C)
2) 1060 psig

B. 1) N-16 Steam Generator Leak Monitor (RM-1MS-102A, B, C)
2) 1075 psig

C. 1) Steam Relief Monitors (RM-1MS-100A, B, C)
2) 1060 psig

D. 1) Steam Relief Monitors (RM-1MS-100A, B, C)
2) 1075 psig

Answer: D

Explanation/Justification: K/A is met with the candidate's ability to determine that with the loss of instrument air the Main steam line stops will be closed, and a Steam Generator Tube Rupture will have to be detected by the Steam Relief Rad monitors which monitor the discharge of the lowest safety relief and the SG atmospheric dump valves.

A. Incorrect. Plausible because the N-16 monitors are used at power (> 20% rx power) and are located downstream of the MSIVs which will be closed. 1060 is plausible because it is the setpoint for the Atmospherics, but they are not available due to the loss of air.

B. Incorrect. Plausible because the N-16 monitors are used at power (> 20% rx power) and are located downstream of the MSIVs which will be closed. 1075 psig is the setpoint is for the lowest Safety valve which is correct.

C. Incorrect. Steam Relief Rad monitors monitor the discharge of the lowest safety relief and the SG atmospheric dump valves, and the SGTR would be monitored on these rad monitors because the MSIVs are closed on a loss of air. 1060 is plausible because it is the setpoint for the Atmospherics, but they are not available due to the loss of air.

D. Correct. Steam Relief Rad monitors monitor the discharge of the lowest safety relief and the SG atmospheric dump valves, and the SGTR would be monitored on these rad monitors because the MSIVs are closed on a loss of air. 1075 psig is the setpoint is for the lowest Safety Relief valve.

Sys #	System	Category	KA Statement
039	Main and Reheat Steam	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including:	Main steam line radiation monitors

K/A# A1.09 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-43.1.C Rev. 12 pg 5 & 6
1OM-21.2.B Rev. 6 pg. 3

Question Source: Bank – 1LOT14 Q45

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(5)

Objective: 1SQS-21.1, Rev. 16 Obj. 12. Given a Main Steam Supply System configuration and without referenced material, describe the Main Steam Supply System control room response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable. a. Loss of instrument air b. Loss of electrical power
1SQS-43.1-01-8: Given a specific plant condition, predict the response of the Radiation Monitoring 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

44. Given the following plant conditions:

- The plant was operating at 40% power when the ‘B’ MFRV failed closed.
- The crew manually tripped the reactor.
- Steam Generator parameters currently are as follows:

SG	NR Level	Pressure
A	44%	980 psig
B	17%	990 psig
C	48%	980 psig

- Assume NO other operator actions have been taken.

Which of the following identifies the expected position of the following valves?

1) MOV-1MS-105, AFW TURB STEAM ISOLATION VALVE will be ______(1)______.

2) TV-1MS-105A and B, AFW TURB STEAM SUPPLY A(B) TRAIN TRIP VALVES will be ______(2)______.

A. 1) Open
2) Open

B. 1) Open
2) Closed

C. 1) Closed
2) Closed

D. 1) Closed
2) Open

Question 44

Answer: A

Explanation/Justification: K/A is met by the candidate's ability to monitor the Main Steam supply valves to the Turbine Driven AFW pump and know the expected valve positions when a TDAFW pump start condition has been met.

A. Correct. MOV-1MS-105 is NSA open. TV-1MS-105A and B will be open due to 'B' SG level falling below the SG low-low level of 19.6% on one SG which would open TV-1MS-105A and B to start the turbine driven AFW pump.

B. Incorrect. Plausible because this is the NSA position for these Main Steam valves supplying the terry turbine and the candidate could get the start signal for the TDAFW pump confused with the Motor Driven AFW pump which requires 2/3 SGs low-low level to start.

C. Incorrect. Plausible if the candidate thinks this is the NSA position of these valves, and doesn't recognize that the 'B' SG level has fallen below the SG low-low level of 19.6% on one SG which would open TV-1MS-105A and B and start the turbine driven AFW pump.

D. Incorrect. Plausible if the candidate thinks this is the NSA position of these valves and doesn't recognize that the 'B' SG level has fallen below the SG low-low level of 19.6% on one SG which would start the turbine driven AFW pump, but this is actually opposite of the NSA positions of these valves.

Sys #	System	Category	KA Statement
039	Main and Reheat Steam	A4 Ability to manually operate and/or monitor in the control room:	Emergency feedwater pump turbines

K/A# A4.04 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.5.B.8 Rev. 0 pg. 4
1OM-24.1.D Rev. 6 pg. 3,4
1OM-24 Fig. 24-9 (LSK-005-013B Rev. 20)

Question Source: New

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

Objective: 1SQS-24.1 Rev. 20 Obj. 15. Given a Main Feedwater, Dedicated Auxiliary Feedwater, Auxiliary Feedwater System or Steam Generator Water Level Control System configuration and without referenced material, describe the associated system's control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable: a. Reactor Trip b. Safety Injection c. SG Low-Low Level d.SG High-High Level

45. The plant has experienced a Rx trip and safety injection with a failure of all Auxiliary Feedwater. The crew has entered FR-H. 1, Response to Loss of Secondary Heat Sink.

Plant conditions are as follows:

- RCPs are stopped
- Containment Conditions:
  - Atmospheric Air Temperature is 155°F and rising
  - Pressure is 0.5 psig and slowly rising
  - Radiation Levels have increased slightly post trip
- SG Wide Range (WR) Levels
  - ‘A’ SG is 16% and slowly lowering
  - ‘B’ SG is 13% and slowly lowering
  - ‘C’ SG is 17% and slowly lowering

In accordance with FR-H.1, which of the following describes the MINIMUM required actions to initiate Main Feedwater flow to the SGs after resetting the Safety Injection Signal?

A. Open Feedwater Isolation Valves, start a Main Feedwater Pump, and throttle open the appropriate Feedwater Bypass Valve(s).

B. Reset Feedwater Isolation, open Feedwater Isolation Valves, throttle open the appropriate Main Feedwater Regulating Valve(s), and start a Main Feedwater Pump.

C. Reset Feedwater Isolation, open Feedwater Isolation Valves, start a Main Feedwater Pump, and throttle open the appropriate Feedwater Bypass Valve(s).

D. Initiate RCS Bleed and Feed at this time, then restore Main Feedwater flow to the SGs.

Answer: C

Explanation/Justification: K/A is met with the candidate’s ability to determine the manual actions required in the control room to align feedwater to the steam generator after an automatic feedwater isolation was generated by the Safety Injection actuation.

A. Incorrect. Plausible distractor because the SI reset does allow the MFP and FW isolation valves to be operated, but the FWI must be reset to allow the BFRVs to be opened.

B. Incorrect. Plausible distractor because it identifies most of the correct steps/components, except the MFRVs are not used in FR-H.1, and the sequence is incorrect.

C. Correct. To establish MFW after the SI has been reset requires the FWI to be reset to allow the BFRVs to be opened to have feedwater aligned to the SGs. FR-H.1 step 7b RNO and step 7c identifies the sequence. Interlock logics for feedwater component operation is required system knowledge. The question setup places the candidate within the FR-H.1 procedure to meet the K/A, but detailed procedural knowledge is not required to answer the question, therefore this is RO knowledge.

D. Incorrect. Plausible distractor because if 2 SGs were <14% WR, then this would be correct iaw FR-H.1 continuous action step 3, but cnmt is not adverse and only 1 SG is <14% WR.

Sys #	System	Category	KA Statement
059	Main Feedwater	A4 Ability to manually operate and monitor in the control room:	Recovery from automatic feedwater isolation

K/A# A4.11 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-H.1 Iss 2 Rev 4 pgs. 2-5

Question Source: Bank – 1LOT18 Q17 (LAST 2 EXAMS)

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

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.

46. The plant is operating at 25% power with all systems in NSA for this power level.

- EDG #1 is on clearance for a lube oil change-out **AND** maintenance has just removed all lube oil from the crankcase.
- An inadvertent reactor trip occurs **COINCIDENT** with a loss of offsite power.
- All SG levels "Shrink" to 25% NR as a result of the trip.
- All systems function as designed.
- No Operator actions have occurred.

Based on these conditions:

Which auxiliary feed pumps, if any, will be running?

A. **NO** AFW pumps

B. **ALL** AFW pumps

C. **ONLY** the "B" AFW pump

D. **BOTH** the Steam driven AFW pump **AND** "B" AFW pump

Answer: D

Explanation/Justification: K/A is met with the candidate's knowledge of both the power supplies of the motor driven AFW pumps, and the automatic start logics associated with all AFW pumps.

A. Incorrect. Steam driven AFW pump will start on 2/3 RCP bus undervoltage BUT not on low-low S/G water level. B AFW pump will start on the trip of all running main feed pump signal BUT not on low-low S/G water level. (Low-low SG water level setpoint is 19.6%)

B. Incorrect. A AFW pump will NOT have power because AE bus is de-energized, and EDG #1 is on clearance.

C. Incorrect. Steam driven AFW pump will start on 2/3 RCP bus undervoltage.

D. Correct. Steam driven AFW pump will start on 2/3 RCP bus undervoltage and B AFW pump will start on the trip of all running main feed pumps because it is supplied by the DF emergency bus which is powered from the #2 EDG. To get this correct, the student will need to know the start signals and power supplies to the AFW pumps. They will also need to know that a loss of offsite power at 25% power will cause the last running main feed pump to trip and cause 2/3 RCP bus undervoltage.

Sys #	System	Category	KA Statement
061	Auxiliary/Emergency Feedwater	K2 Knowledge of bus power supplies to the following:	AFW electric drive pumps

K/A# K2.02 **K/A Importance** 3.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-24.1.D Rev. 6 pg. 3, 4

Question Source: Bank – 2LOT6 Q46

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

Objective: 1SQS-24.1 Rev. 20 Obj. 3. Identify the power supplies for the components identified on the Normal-System-Arrangement System Flow-path drawing which are powered from the class 1E electrical distribution system.

1SQS-24.1 Rev. 20 Obj. 14. Describe the control, protection and interlock functions for the control room components associated with the Main Feedwater, Dedicated Auxiliary Feedwater, Auxiliary Feedwater System and Steam Generator Water Level Control Systems, including automatic functions, setpoints and changes in equipment status as applicable.

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

- The plant suffered a Loss of Off-Site Power.
- Both Emergency Diesel Generators (EDGs) are supplying emergency busses.
- Grid stability is confirmed, and the Operations Manager has granted permission to return to the grid.
- The Control Room is performing 1OM-36.4.Q, “Transferring Emergency Busses 1AE and 1DF From Emergency Feed to Normal Feed”, beginning with Bus 1AE.
- ACB-1A10, 4KV Bus 1A To 1AE is closed.
- EDG 1-1 is synchronized to the grid and ACB-1E7, 4KV Bus 1AE to 1A was closed.
- Upon breaker closure, the following annunciator sequence occurs:
    - A8-107, “4160V EMERG BUS 1AE ACB-1E7 OVERCURRENT TRIP” received.
    - A8-106, “4160V EMERGENCY BUS 1AE ACB-1E7 AUTO TRIP” received.
    - A8-107, “4160V EMERG BUS 1AE ACB-1E7 OVERCURRENT TRIP” clears.

1) What is the status of EDG 1-1?

2) Besides ACB-1E7, 4KV Bus 1AE To 1A, what other breaker needs to be verified open IAW ARP A8-106 and A8-107?

A. 1) Tripped
   2) ACB-1A10, 4KV Bus 1A To 1AE

B. 1) Tripped
   2) ACB-1E9, Emerg Gen 1 Circuit Breaker

C. 1) Running
   2) ACB-1A10, 4KV Bus 1A To 1AE

D. 1) Running
   2) ACB-1E9, Emerg Gen 1 Circuit Breaker

Question 47

Answer: C

Explanation/Justification: K/A is met by the candidates ability to predict that when an instantaneous overcurrent occurs when paralleling the emergency 4KV bus to the 'A' normal 4KV bus, the EDG will remain running, and based on knowledge of the Alarm Response procedure, will verify open both feeder breakers to ensure the fault is isolated and the diesel is running.

A. Incorrect. EDG 1-1 does not trip but remains running. Plausible that the EDG would trip on an overcurrent condition, however, protection in this scenario is provided by ACB 1E7. Verifying ACB-1A10 is correct.

B. Incorrect. EDG 1-1 does not trip but remains running. Plausible that the EDG would trip on an overcurrent condition, however, protection in this scenario is provided by ACB 1E7. Verifying ACB-1E9 is plausible per the AOP if EDG 1-1 was NOT connected to 4KV Bus 1AE.

C. Correct. For the given conditions, EDG 1-1 was running paralleled to the grid briefly. An overcurrent condition was caused by the closure of ACB1E7 and results in ACB 1A10 & 1E7 automatically opening. Upon ACB-1E7 opening, the overcurrent condition clears which is indicative of the problem being downstream of ACB 1E7. The EDG will continue to run. The ARP states that if the EDG was synchronized to the grid, then verify ACB-1E7 and ACB-1A10 are open. A note in the ARPs states if an instantaneous trip has fault, the AE bus should remain energized.

D. Incorrect. EDG 1-1 will be running, see explanation above. Verifying ACB-1E9 is plausible per the AOP if EDG 1-1 was NOT connected to 4KV Bus 1AE.

Sys #	System	Category	KA Statement
062	AC Electrical Distribution	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Consequences of exceeding current limitations

K/A# A2.09 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-36.4.ADB Rev. 4 pg. 2
1OM-36.4.ADA Iss. 3 Rev. 0 pg. 1
1SQS-36.2 Rev. 19 pg. 38, 39

Question Source: Modified – 1LOT8 Q48

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

Objective: 3SQS-36.1, Rev. 12 Obj. 3. Given a change in plant conditions, describe the response of the 4KV Distribution System field indication and control loops, including all automatic functions and changes in equipment status.

3SQS-36.1, Rev. 12 Obj. 14. Describe the control, protection and interlock functions for the control room components associated with the 4KV Distribution System, including automatic functions, setpoints and changes in equipment status as applicable.

48. Given the following plant conditions:

- The plant is operating at 100% power with all systems in NSA.
- Vertical Board ‘C’ indicator for No. 1 DC Bus Grd Volts meter indicates a +110 VDC ground.
- The crew has entered 1OM-39.4.E, Clearing Grounds (125 VDC Busses 1-1 and 1-2).

IAW 1OM-39.4.E, which of the following is the impact if more than one ground exists **AND** what action will be taken to preclude this impact according to this procedure?

The impact of multiple DC grounds is that ____ (1) ____

To preclude this impact ____ (2) ____

A. 1) inadvertent actuations may occur.
2) de-energize DC Bus 1-1 until grounds are located.

B. 1) inadvertent actuations may occur.
2) open knife switches or breakers prior to resetting relays.

C. 1) control functions may not occur when called upon.
2) de-energize DC Bus 1-1 until grounds are located.

D. 1) control functions may not occur when called upon.
2) the unit must be shutdown within 1 hour if grounds are not isolated.

Answer: B

Explanation/Justification: K/A is met with the candidate’s ability to predict the impact that multiple grounds could have when resetting relays, and identifying the correct action to be taken to prevent the inadvertent actuations from occurring iaw the plant procedure for clearing grounds.

A. Incorrect. Correct impact according to 1OM-39.4.E precautions and limitations. Incorrect but plausible action. Grounds are located by isolating individual components supplied by DC-Bus 1-1. If the entire bus were de-energized it would be difficult to locate and isolate the ground.

B. Correct. The candidate must be able to predict the impacts of multiple DC grounds on DC Bus 1-1. According to 1OM-39.4.E P&Ls, the impact of multiple grounds is that inadvertent actuations may occur. BVPS has had some actual OE regarding this issue. The candidate must also be able to use procedures to control the impact of multiple grounds. According to the precautions and limitations of the same reference, the method of control is to open knife switches or breakers prior to resetting relays.

C. Incorrect. Correct impact not referenced in our procedure, however, from research it is also possible with a DC ground that control functions may not operate when called upon depending on the resistance of the circuit. Incorrect but plausible action as explained in A above.

D. Incorrect. Correct impact not referenced in our procedure, however, from research it is also possible with a DC ground that control functions may not operate when called upon depending on the resistance of the circuit. Plausible incorrect action. TS 3.8.4 actions if a battery charger is inoperable is a 2 hour action. The RO is required to know ≤ 1 hour TS actions from memory.

Sys #	System	Category	KA Statement
063	DC Electrical Distribution	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Grounds

K/A# A2.01 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-39.4.E Rev. 7 pg. 2

Question Source: Bank – 2LOT8 Q48

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(5)

Objective: 3SQS-39.1 Rev.9 Obj. 18. Given a set of plant conditions and the appropriate procedure(s), apply the operational sequence, parameter limits, precautions and limitations, and cautions & notes applicable to the completion of the task activities in the control room.

49. The plant is at 100% power.

    Which of the following combinations of Control Room indications would **REQUIRE** a reactor trip?

A. Battery Charger # 1 GREEN light NOT lit and RED light lit.
   Battery Breaker # 1 GREEN light lit and RED light NOT lit.

B. Battery Charger # 2 GREEN light lit and RED light NOT lit.
   Battery Breaker # 2 GREEN light lit and RED light NOT lit.

C. Battery Charger # 3 GREEN light lit and RED light NOT lit.
   Battery Breaker # 3 GREEN light lit and RED light NOT lit.

D. Battery Charger # 5 GREEN light NOT lit and RED light lit.
   Battery Breaker # 5 GREEN light NOT lit and RED light lit.

Answer: B

Explanation/Justification: K/A is met with the candidate's ability to interpret the DC battery charger and breaker position indications located in the Control Room and determine which of the indications would require a reactor trip because of the loss of plant equipment.

A. Incorrect. The Battery Charger will supply power to 125 VDC Bus #1, there is no need to trip the reactor.
B. Correct. The student must evaluate the breaker indicating lights and determine that the 125 VDC Bus #2 is deenergized since both the Battery and Charger supply breakers are open. The reactor is tripped because the loss of 125 VDC Bus 2 results in a loss of CCR to all RCPs, requiring them to be secured. Also the MFRVs and Bypass valves will fail closed.
C. Incorrect. These indications would result in a loss of power to 125 VDC Bus #3, however no reactor trip signal is generated or required for a loss of 125 VDC Bus #3.
D. Incorrect. The Battery will supply power to 125 VDC Bus #5, and there is no need to trip the reactor for a loss of 125 VDC Bus #5.

Sys #	System	Category	KA Statement
063	DC Electrical Distribution	Generic	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

K/A# 2.2.44 **K/A Importance** 4.2 **Exam Level** RO
References provided to Candidate None **Technical References:** 1OM-53C.4.1.39.1B Rev. 6 pg 1 & 9
Question Source: Bank – 1LOT14 Q49
Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(7)
Objective: 3SQS-39.1 Rev. 9 Obj. 15. Describe the instrumentation associated with the 125 VDC Distribution System located in the Main Control Room.
3SQS-39.1 Rev. 9 Obj. 21. Given a change in plant conditions due to system/component failure, analyze the 125 VDC Distribution System to determine what failure has occurred.

50. A rupture occurs at the discharge of the three DG Starting Air receivers which causes pressure to lower in one of EDG1-1 air banks.

1) What is the minimum Tech Spec required Air Receiver pressure?

2) If the rupture continues, causing all three air receivers on that bank to fully depressurize, EDG 1-1 ______(1)______ start upon receipt of an auto start signal.

A. 1) ≥165 psig
2) will

B. 1) ≥380 psig
2) will

C. 1) ≥165 psig
2) will NOT

D. 1) ≥380 psig
2) will NOT

Answer: A

Explanation/Justification: K/A is met by requiring knowledge of the EDG air system configuration and lineup, and the effects that a loss of one bank of air receivers will have on the starting capabilities of the EDG.

A. Correct. Tech Spec minimum air pressure is 165 psig. The knowledge of the correct value is gained through performing OSTs, tech specs, and log taking. DG will start even with one bank fully depressurized due to there being two banks per DG, and only two out of three air tanks in one of the two air banks at the specified air pressure is required to start the DG.

B. Incorrect. 380 psig is plausible because BV1 & 2 use combined Tech Specs which identify both unit air pressures on the same page. The DG will start with only one air bank available.

C. Incorrect. 165 psig is correct. Second part is plausible if candidate doesn't understand the DG air start system flowpaths or design, consisting of two air banks and six air receivers.

D. Incorrect. 380 psig is plausible because BV1 & 2 use combined Tech Specs which identify both unit air pressures on the same page. Second part is plausible if candidate doesn't understand the DG air start system flowpaths or design, consisting of two air banks and six air receivers.

Sys #	System	Category	KA Statement
064	Emergency Diesel Generator	K6 Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system:	Air receivers

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

References provided to Candidate None **Technical References:** Tech Spec pg. 3.8.3-1 & B 3.8.3-7
1SQS-36.2 Rev. 19 PPNT Slide 16
1OM-36.1.C Rev. 7 pg. 8

Question Source: Bank – 2LOT15 Q50

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

Objective: 1SQS-36.2 Rev. 19 Obj. 9. Identify the EDG field instruments, subsystems and components that are required to be operable by the Technical Specifications.

51. The plant is at 100% power with all systems in NSA **EXCEPT** for the following:

- 1OST-24.4, Steam Turbine Driven Auxiliary Feed Pump Test [1FW-P-2] is being performed
- Unit 1 Waste Gas Decay Tank [GW-TK-1B] is being discharged

Based on these conditions, which of the below Radiation Monitor **High-High** alarms will cause the Main Filter Banks to automatically realign to filter the discharging effluents?

A. Gaseous Waste Gas [RM-GW-108B]

B. Condenser Air Ejector Vent [RM-SV-100]

C. 1FW-P-2 Exhaust [RM-MS-101]

D. Aux Building Vent Sys B Gas [RM-VS-102B]

Answer: D

Explanation/Justification: K/A is met with the candidate's ability to determine that on an Aux Building Ventilation radiation monitor High-High alarm, the ventilation will realign to filter exhaust effluent to prevent exceeding the design limits for plant releases.

A. Incorrect. Hi-Hi- alarm will automatically isolate the discharge but will not re-align the discharge thru the MFB.
B. Incorrect. Hi-Hi- alarm will automatically isolate the normal discharge flowpath and align the discharge to CNMT but will not re-align the discharge thru the MFB.
C. Incorrect. This monitor does not have any automatic functions. If the Hi-Hi- alarm is received, the effluent will continue to discharge. The monitor is purely for monitoring purposes and offsite dose projections.
D. Correct. The Main Filter Banks will filter the effluent and reduce the release to values that are within acceptable design limits for 10CFR20.

Sys #	System	Category	KA Statement
073	Process Radiation Monitoring	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including:	Radiation levels

K/A# A1.01 **K/A Importance** 3.2 **Exam Level** RO
References provided to Candidate None **Technical References:** 1OM-43.1.C Rev. 12 pg. 9
Question Source: Bank – 1LOT14 Q51
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(5)
Objective: 1SQS-43.1-01-2: Describe the automatic actions that occur in the field when a given radiation monitor alarms.

52. The Tech Spec bases for Reactor Plant River Water requires that River Water needs to be supplied to which of the following components to place the plant in safe shutdown following a Design Basis Accident (DBA)?

    1. Diesel Generator Cooling System Heat Exchangers
    2. Recirculation Spray System Heat Exchangers
    3. Containment Air Recirculation Coolers
    4. Charging Pump Lube Oil Coolers
    5. Component Cooling Water Heat Exchangers
    6. Control Room Emergency Cooling Coils

A. 1, 2, 4, and 5 **only**

B. 1, 2, 3, 4, and 5 **only**

C. 1, 2, 4, 5, and 6 **only**

D. 1, 2, 3, 4, 5, and 6

Answer: C

Explanation/Justification: K/A is met with the candidate's knowledge of the Reactor Plant River Water components which must be supplied with water to ensure the Tech Spec 3.7.8 bases is met in the event a Design Basis Accident. These components will ensure the plant is capable of being placed in a safe shutdown condition.

A. Incorrect. See correct answer. Plausible because of the answer variations.
B. Incorrect. Incorrect. See correct answer. Plausible because of the answer variations.
C. Correct. DG, RSS, Charging pumps, CCW, and CR cooling coils are all identified in TS 3.7.8 as required to place the plant in a safe shutdown condition following a DBA accident. Containment Air Recirculation Coolers are attached to the RPRW system, but Chilled Water normally supplies them and they are not required for a DBA accident.
D. Incorrect. Incorrect. See correct answer. Plausible because of the answer variations.

Sys #	System	Category	KA Statement
076	Service Water	Generic	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

K/A# 2.2.25 **K/A Importance** 3.2 **Exam Level** RO
References provided to Candidate None **Technical References:** Tech Spec 3.78 Bases pg. B3.7.8-1
Question Source: New
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(10)
Objective: 3SQS-PLTSYS ITS, Rev. 2 Obj. 2. State the purpose of each Plant Systems System specification as described in the Applicable Safety Analyses section of the Bases.

53. Given the following plant conditions:

- The Unit has been operating at 75% steady state power.
- Control Rods are in MANUAL all other systems are in NSA.
- The air line to TCV-1CH-144, Non-Regenerative Heat Exchanger Discharge Temperature Control Valve shears causing this valve to reposition to its FAIL position.
- No operator action occurs.
- All systems function as designed.

Which of the following describes the **INITIAL** plant response to this malfunction?

A. Program PRZR level will increase.

B. Program PRZR level will decrease.

C. PRZR spray valve will throttle open and equilibrium xenon will rise.

D. PRZR spray valves will throttle closed and equilibrium xenon will drop.

Answer: A

Explanation/Justification: K/A is met with the candidate's knowledge that a loss of instrument air to the Non-Regenerative Heat Exchanger Discharge Temperature Control Valve will cause the valve to fail open causing CVCS letdown temperature to lower. The lower temperatures will cause the resin to remove more boron from coolant which will effect RCS Tavg.

A. Correct. TCV-1CH-144 fails to the full OPEN position. With more CCW flow through the NRHX, the CVCS temp will go down prior to entering demineralizer. The boron affinity of a resin bed is affected by the temperature of the coolant passed through the bed. The result of this characteristic is that at lower temperatures the resins are more efficient at removing boron from the coolant than at higher temperatures. (A saturated resin bed will actually release boron as temperature is increased.) The removal of boron is a positive reactivity event, RCS Temp must increase to counteract this reactivity. The increase in Tavg will cause program PZR level to initially increase.

B. Incorrect. Opposite of correct response. Plausible if the candidate believes that the TCV fails in the closed direction or they have misconceptions of boron affinity.

C. Incorrect. Correct that PRZR spray valves will initially open in response to increase Tavg which will increase PRZR level which results in a small initial pressure increase. Incorrect that Xenon will initially increase.

D. Incorrect. Incorrect PRZR spray valve response, however, plausible if the candidate has a misconception about the TCV-1CH-144 fail position or impact on Tavg. Plausible that Xenon will initially drop if the candidate believes power increases due to this event or has xenon concept confused.

Sys #	System	Category	KA Statement
078	Instrument Air	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:** 1OM-53C.4.1.34.1, Rev 29, pg 11
1SQS-7.1 Rev. 20 LP pg. 16, 20, 21

Question Source: Bank – Vision 256818

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

Objective: 1SQS-7.1 Rev. 20 Obj. 8. Given a Chemical and Volume Control System configuration and without referenced material, describe the Chemical and Volume Control System field response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable. a. Loss of Instrument Air

GO-GPF.C4 Rev. 2 Obj. 22. Describe the demineralizer characteristics that can cause a change in boron concentration.

54. Given the following plant conditions:

- The plant is operating at 75% power when a steam leak occurs in containment.
- Initial Containment temperature is 97°F.
- Containment temperature is rising at 1°F every 5 minutes at a constant rate.
- No auto or manual plant shutdown actions have occurred.

1) How long will it take for the containment temperature to exceed Tech Spec LCO 3.6.5, Containment Air Temperature?

2) LCO 3.6.5 upper limit is based on the Containment Liner not exceeding what design temperature limit?

A. 1) 45 minutes
2) 200°F

B. 1) 45 minutes
2) 280°F

C. 1) 60 minutes
2) 200°F

D. 1) 60 minutes
2) 280°F

Answer: D

Explanation/Justification: K/A is met by the candidate's ability to predict how long it will take for containment air temperature to exceed the Tech Spec LCO limit of 108F and understand that the upper limit is based on preventing the containment liner from exceeding its design temperature.

A. Incorrect. Plausible distractor because 45 minutes is equal to 105F which is when RPRW is supplied to the Containment Air Recirculation Cooling Coils when Chilled Water system cannot handle the demand on the system. 200F is plausible because it is a lower value than the actual liner temperature, and it is a temperature that is familiar to modes of operation.

B. Incorrect. Plausible distractor because 45 minutes is equal to 105F which is when RPRW is supplied to the CAR Cooling Coils when Chilled Water system cannot handle the demand on the system. IAW TS 3.6.5, the design temperature of the containment liner is 280F.

C. Incorrect. 60 minutes is correct based on TS 3.6.5 upper limit of ≤108F, therefore 108-97F=11F, 5min/1F=55 minutes. 60 minutes (109F) will exceed the TS limit. 200F is plausible because it is a lower value than the actual liner temperature, and it is a temperature that is familiar to modes of operation.

D. Correct. 60 minutes is correct based on TS 3.6.5 upper limit of ≤108F, therefore 108-97F=11F, 5min/1F=55 minutes. 60 minutes (109F) will exceed the TS limit. IAW TS 3.6.5, the design temperature of the containment liner is 280F.

Sys #	System	Category	KA Statement
103	Containment	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:** T.S 3.6.5 pg. 3.6.5-1 & B3.6.5-2

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(5)

Objective: 3SQS-CONT.ITS Rev. 1 Obj. 1. Apply the following definitions to ensure compliance with applicable requirements: a. OPERABLE - OPERABILITY

3SQS-CONT.ITS Rev. 1 Obj 2. State the purpose of each Containment Systems specification as described in the Applicable Safety Analyses section of the Bases.

55. What is the **minimum** containment pressure allowed in accordance with Tech Spec 3.6.4, Containment Pressure, and what is the bases for this **minimum** limit?

A. 12.8 psia, to ensure a maximum peak pressure from a LOCA does not exceed the containment design pressure.

B. 14.2 psia, to ensure a maximum peak pressure from a LOCA does not exceed the containment design pressure.

C. 12.8 psia, to ensure the containment will not exceed the design pressure limit following an inadvertent actuation of the Containment Spray System.

D. 14.2 psia, to ensure the containment will not exceed the design pressure limit following an inadvertent actuation of the Containment Spray System.

Answer: C

Explanation/Justification: K/A is met with the candidate's knowledge of the containment minimum pressure per Tech Spec 3.6.4, Containment Pressure, and the bases for the minimum pressure.

A. Incorrect. 12.8 psia is the minimum pressure per TS 3.6.4. Second part is plausible because this is the bases for maximum cnmt pressure per TS 3.6.4, but not the minimum and the candidate may be confused with the pressures and bases.
B. Incorrect. Plausible because 14.2 is TS 3.6.4 maximum pressure and the student could be confused since the spec is in psia. Plausible because this is the bases for maximum cnmt pressure per TS 3.6.4, but not the minimum.
C. Correct. Tech Spec 3.6.4 minimum cnmt pressure is ≥12.8 psig. This is to ensure that an inadvertent Quench Spray actuation will not lower cnmt pressure below the design internal pressure of 8.0 psia.
D. Incorrect. Plausible because 14.2 is TS 3.6.4 maximum pressure and the student could be confused since the spec is in psia. The bases is correct for the minimum cnmt pressure per TS 3.6.4.

Sys #	**System**	**Category**	**KA Statement**
103	Containment	Generic	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

K/A# 2.2.25 **K/A Importance** 3.2 **Exam Level** RO
References provided to Candidate None **Technical References:** T.S. 3.6.4 and bases pg. B3.6.4-1
Question Source: New
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(10)

Objective: 3SQS-CONT ITS, Rev. 1 Obj. 1. Apply the following definitions to ensure compliance with applicable requirements: a. OPERABLE - OPERABILITY
3SQS-CONT ITS, Rev. 1 Obj. 2. State the purpose of each Containment Systems specification as described in the Applicable Safety Analyses section of the Bases.

56. Given the following conditions:

- The plant is in Mode 3 performing a reactor startup in accordance with 1OM-50.4.D1, Reactor Startup from Mode 3 to Mode 2 - Initial Criticality.
- Both RDMG sets are operating supplying the CRDMs
- Shutdown control rods are fully withdrawn
- Control bank ‘A’ rods are being withdrawn
- A4-99, ROD CONTROL MG SET 1A TRIP alarms

1) A loss of which of the following power supplies will cause the ‘A’ RDMG set to trip?

2) What procedure would be used to mitigate the consequences of the ‘A’ RDMG set trip?

A. 1) 480 VAC Substation 8N
2) E-0, Reactor Trip and Safety Injection because the reactor has tripped.

B. 1) 480 VAC Substation 8N
2) 1OM-1.4B, Full-Length Rod Control System Startup to restart the ‘A’ RDMG set.

C. 1) 480 VAC Substation 1-1
2) E-0, Reactor Trip and Safety Injection because the reactor has tripped.

D. 1) 480 VAC Substation 1-1
2) 1OM-1.4B, Full-Length Rod Control System Startup to restart the ‘A’ RDMG set.

Answer: D

Explanation/Justification: K/A is met with the candidate’s ability to determine the impact on the CRDMs with a loss of one RDMG set during plant startup, and recognize that a loss of one RDMG set will not cause the reactor to trip because power to the CRDMs is provided from dual, parallel Rod Drive Motor Generator (RDMG) sets.

A. Incorrect. Plausible if the candidate thinks the ‘A’ RDMG set is supplied from the emergency 480-volt bus. E-0 is plausible if the candidate doesn’t understand the electrical flowpath from the RDMG sets to the CRDMs and thinks that a loss of the ‘A’ RDMG set will de-energize the ‘A’ Rx trip breaker.

B. Incorrect. Plausible if the candidate thinks the ‘A’ RDMG set is supplied from the emergency 480-volt bus. Power to the CRDMs is provided from dual, parallel Rod Drive Motor Generator (RDMG) sets, therefore the loss of the ‘A’ RDMG set does not cause the reactor to trip, and the ARP directs the crew to 1OM-1.4B to restart the ‘A’ RDMG set.

C. Incorrect. 480 VAC Substation 1-1 supplies power to the ‘A’ RDMG set. E-0 is plausible if the candidate doesn’t understand the electrical flowpath from the RDMG sets to the CRDMs and thinks that a loss of the ‘A’ RDMG set will de-energize the ‘A’ Rx trip breaker.

D. Correct. 480 VAC Substation 1-1 supplies power to the ‘A’ RDMG set. Power to the CRDMs is provided from dual, parallel Rod Drive Motor Generator (RDMG) sets, therefore the loss of the ‘A’ RDMG set does not cause the reactor to trip, and the ARP directs the crew to 1OM-1.4B to restart the ‘A’ RDMG set.

Sys #	System	Category	KA Statement
001	Control Rod Drive	A2 Ability to (a) predict the impacts of the following malfunction or operations on the CRDS- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of power source to reactor trip breakers

K/A# A2.02 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 3SQS-1.3 PPNT Rev. 8 Slide 26
1OM-1.4.AAP Iss. 3 Rev. 1

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(6)

Objective: 3SQS-1.3 Rev. 8, Iss. 1 Obj. 4. Describe the operation, purpose, and location of the RDMG Sets.
3SQS-1.3 Rev. 8, Iss. 1 Obj. 14. Given a System configuration, and without referenced material, describe the Rod Control System response to loss of electrical power.

57. While recording Critical Data at 10^{-8} amps, a fault causes a loss of Vital Bus 3.

Which of the following is the expected plant response due to the Nuclear Instrumentation system?

A. Reactor will trip due to Intermediate Range NI High Flux trip.

B. Bistables for Power Range NI N-43 will trip, Reactor will NOT trip.

C. Bistables for Power Range NI N-42 will trip, Reactor will NOT trip.

D. Reactor will trip due to removal of P-6 block reenergizing the Source Range NIs.

Answer: B

Explanation/Justification: K/A is met with the candidate's knowledge of vital bus power supplies to the various nuclear instruments, and determine how the reactor protection system will respond to a loss of vital bus 3.

A. Incorrect. Plausible because Critical Data is taken at 1X10-8 amps where the IR High Flux Trip is still active and a loss of power to an IR Channel would cause a trip due to a 1/2 coincidence when power is <P-10, however, IR channels N-35 and N-36 are powered from vital bus 1 and 2 respectively and no IR channels will be lost.

B. Correct. Power Range N-43 is powered from vital bus 3 and on a loss of power, the detector bistables will trip to their fail-safe position. Since the coincidence for a Reactor Trip is 2/4 channels, the Reactor will NOT trip.

C. Incorrect. Plausible because bistables for Power Range N-42 would trip upon a loss of power, however, N-42 is powered from vital bus 2 and will not lose power.

D. Incorrect. Plausible because removal of P-6 will reenergize Source Ranges and since Critical Data is taken at a power above the Source Range High Flux setpoint, a Reactor Trip would occur, however, the P-6 permissive signal is generated from the IR channels (1 of 2 >10^{-10} amps). It is automatically removed when 2 of 2 IR channels are < 10^{-10} amps. Since both IR channels remain powered, P-6 remains active. N-31 and N-32 are powered from vital bus 1 and 2 respectively and no SR channels will be lost.

Sys #	System	Category	KA Statement
015	Nuclear Instrumentation	K1 Knowledge of the physical connections and/or cause effect relationships between the NIS and the following systems:	Vital ac systems

K/A# K1.02 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 3SQS-2.1 Rev. 9 pg. 10
1OM-1.5.B.8 Rev. 0 pg. 3

Question Source: Bank – 2012 Comanche Peak Q30

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(6)

Objective: 3SQS-2.1, Rev. 9 Obj. 6. Describe the control, protection, alarm and interlock signals generated by the NIS including trip setpoints, coincidences, permissives, blocks, automatic rod motion inhibit signals, and the basis for each.

3SQS-2.1, Rev. 9 Obj 14. Given a specific plant condition, predict the response of the NIS, including all automatic functions and changes in equipment status, for a change in plant conditions.

58. The plant is at 75% Reactor power with the following conditions:

- PRZR level control is in automatic controlling PRZR level on program.
- TI-1RC-412D, Loop ‘A’ Tavg is 571ºF.
- TI-1RC-422D, Loop ‘B’ Tavg is 568ºF.
- TI-1RC-432D, Loop ‘C’ Tavg is 567ºF.

Subsequently:

- Loop ‘B’ Tavg Channel fails to 600ºF.

Automatic PRZR level control input will be from _______________ after this failure.

A. Loop ‘A’ Tavg.

B. Loop ‘B’ Tavg.

C. Loop ‘C’ Tavg.

D. Reference Tavg.

Answer: A

Explanation/Justification: K/A is met by the candidate analyzing a Loop ‘B’ Tavg failure and the affect it will have on the input to the automatic pressurizer level control system due to the Tavg Median select process.

A. Correct. With the przr level control program using the median select Tavg, the highest Tavg (Loop B - 600F) and the lowest Tavg (Loop C - 567F) will be automatically rejected. This will make Loop ‘A’ Tavg at 571F the median Tavg input to the przr level control program.
B. Incorrect. Plausible if the candidate thinks the highest Tavg channel is used for the przr level control program.
C. Incorrect. Plausible if the candidate thinks the lowest Tavg channel is used for the przr level control program.
D. Incorrect. Plausible because Reference Tavg is an input to the analog summer for the przr level control program, but the controlling Tavg will be Loop A due to the Tavg median selector. T ref (547F) could be considered a conservative decision.

Sys #	System	Category	KA Statement
016	Nonnuclear Instrumentation	A3 Ability to monitor automatic operation of the NNIS, including:	Relationship between meter readings and actual parameter value

K/A# A3.02 **K/A Importance** 2.9 **Exam Level** RO
References provided to Candidate None **Technical References:** 1SQS-6.5 PPNT Rev. 14 slide 26
Question Source: Bank - 2LOT19 Q59
Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(7)
Objective: 1SQS-6.5 Obj. 21. Given a specific plant condition, predict the response of the Reactor Coolant 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. b. Process Instrument Failure

59. Given the following conditions:

- The plant is in Mode 5.
- Containment entry has NOT occurred.
- A Containment purge was established IAW 1OM-44C.4.A, Containment Purge Supply and Exhaust System Startup and is aligned to the Ventilation Vent.

Subsequently, 1VS-D-5-3B, CNMT Isol Purge Exhaust Damper failed CLOSED due to an electrical malfunction.

Based on the given conditions, containment pressure will ______(1)______, and Containment Purge flowrate through the Ventilation Vent Header (FR-1VS-101) will ______(2)______.

A. 1) lower
2) remain the same

B. 1) lower
2) lower

C. 1) rise
2) remain the same

D. 1) rise
2) lower

Answer: D

Explanation/Justification: K/A is met with the candidate's ability to determine what occurs when a containment purge exhaust damper closes during a containment purge and monitor how containment pressure and purge flowrate is affected.

A. Incorrect. Plausible for cnmt pressure to lower if the candidate does not recognize that there is a purge supply fan to supply air into the containment during cnmt purge and thinks all airflow in cnmt will stop. Plausible for the candidate to think Ventilation Vent Header flowrate will remain the same due to the Auxiliary Building Exhausts fans also discharging to the ventilation vent.

B. Incorrect. Plausible for cnmt pressure to lower if the candidate does not recognize that there is a purge supply fan to supply air into the containment during cnmt purge and thinks all airflow in cnmt will stop. The purge flowrate as indicated on the ventilation vent header indication will lower because the purge exhaust fan is isolated from the cnmt by the CNMT Isol Purge Exhaust Damper.

C. Incorrect. Cnmt pressure will rise due to the Cnmt Purge Supply fan continuing to run supplying 27000 scfm into the CNMT Isol Purge Exhaust Damper closed. The flowrate through the Ventilation Vent Header will have to lower due to the exhaust fan suction is isolated from cnmt. Plausible if the candidate doesn't understand the cnmt purge components and flowpath.

D. Correct. Since the CNMT Isol Purge Exhaust Damper closes, the exhaust fan (30scfm) suction is isolated, but the Cnmt Purge supply fan will still be running supplying 27,000 scfm airflow into cnmt, therefore cnmt pressure will rise. The purge flowrate as indicated on the ventilation vent header indication will lower because the purge exhaust fan is isolated from the cnmt by the CNMT Isol Purge Exhaust Damper.

Sys #	System	Category	KA Statement
029	Containment Purge	A4 Ability to manually operate and/or monitor in the control room:	Containment purge flow rate

K/A# A4.01 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** U1 RM-0416-001 Rev. 18

Question Source: New

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

Objective: 1SQS-44C.1 Rev. 10 Obj. 6. Given a Containment Ventilation System configuration and without reference material, describe the Containment Ventilation System field response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable.

60. The plant is in Mode 6. A fuel assembly is being lowered into the core.

    IF the fuel assembly "**BINDS**" against another fuel assembly, downward motion of the hoist will be automatically stopped to prevent fuel assembly damage.

    What manipulator crane interlock, if any, provides this protection?

A. Underload

B. Tube Down

C. Overload

D. None

Answer: A

Explanation/Justification: K/A is met with the candidate's knowledge of the fuel handling manipulator crane interlock which prevents damage to the fuel bundle during fuel movement evolutions.

A. Correct. Underload stops downward motion when the gripper is engaged and the load cell indicates approximately 150 lbs below the weight of the latched fuel assembly.
B. Incorrect. Tube down interlock will stop hoist downward motion when the hoist is all the way down.
C. Incorrect. Overload will stop UPWARD motion if an assembly is binding while moving upward.
D. Incorrect. None is plausible if the candidate thinks that there are no automatic interlocks that would stop assembly movement.

Sys #	System	Category	KA Statement
034	Fuel-Handling Equipment	K4 Knowledge of design feature(s) and/or interlock(s) which provide for the following:	Fuel movement

K/A# K4.02 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** LP 3SQS-6.13 Rev. 6 pg 12 & Slide 104
1RP-3.3, Rev. 10 pg 36

Question Source: Bank – 1LOT14 Q61

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

Objective: 3SQS-6.13, Rev. 6 Obj. 2. Describe the control, protection and interlock functions for the fuel handling equipment, including automatic functions, setpoints and changes in equipment status as applicable.

61. Given the following:

- The plant is operating at Rated Thermal Power
- An electrical malfunction causes a Main Generator lockout
- The Steam Dump Mode Selector switch is in Tavg

Which of the following complete these statements?

In response to this event, steam dump valves trip open, then modulate to control TAVG ______(1)______.

If the Condenser Steam Dumps fail to ARM, RCS pressure will rise and ______(2)______.

A. (1) within 3°F of TREF
(2) exceed 110% of design pressure

B. (1) at 547°F
(2) remain below 110% of design pressure

C. (1) within 3° of TREF
(2) remain below 110% of design pressure

D. (1) at 547°F
(2) exceed 110% of design pressure

Answer: B

Explanation/Justification: K/A is met with the candidates knowledge that when the turbine trips from full power even if there were not a direct reactor trip, that the RCS pressure will not exceed the 110% design pressure of the RCS if the Condenser Steam Dumps fail to Arm due to the Atmospheric Steam Dumps and the Main Steam Safety valves.

A. Incorrect. Within 3°F of TREF is plausible because this is where the Steam Dump load rejection controller would control Tavg, but since the reactor tripped due to being >P9, the Rx trip controller maintains RCS temperature at 547F. Exceeding 110% of design pressure is plausible if the candidate does not understand that without the condenser steam dumps, the atmospheric steam dumps, and Main steam safety valves ensure that the RCS pressure will not exceed the 110% of RCS design pressure when the turbine trips from full power.

B. Correct. Since the plant was at Rated Thermal Power (>P9) when the generator lockout occurred, a reactor trip occurred, therefore the Steam Dump Rx trip controller maintains the RCS temperature at no load temperature of 547F. The RCS will remain below 110% of design pressure even if the condenser steam dumps fail to arm because the atmospheric steam dumps, and the main steam safety valves will provide the heat sink required.

C. Incorrect. Within 3°F of TREF is plausible because this is where the Steam Dump load rejection controller would control Tavg, but since the reactor tripped due to being >P9, the Rx trip controller maintains RCS temperature at 547F. The RCS will remain below 110% of design pressure even if the condenser steam dumps fail to arm because the atmospheric steam dumps, and the main steam safety valves will provide the heat sink required.

D. Incorrect: First part is correct because the Steam Dump Rx trip controller will maintain RCS temperature at 547F. Exceeding 110% of design pressure is plausible if the candidate does not understand that without the condenser steam dumps, the atmospheric steam dumps, and Main steam safety valves ensure that the RCS pressure will not exceed the 110% of RCS design pressure when the turbine trips from full power.

Sys #	System	Category	KA Statement
041	Steam Dump/Turbine Bypass Control	K5 Knowledge of the operational implications of the following concepts as the apply to the SDS:	Basis for RCS design pressure limits

K/A# K5.05 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1SQS-21.1 PPNT Rev. 16 Iss. 1 Slides 50, 51
Tech Spec Bases pg. B 3.7.1-1 and 2

Question Source: Bank – 2017 Point Beach Q61

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(5)

Objective: 1SQS-21.1 Rev. 16 Obj. 11. Given a Main Steam Supply System configuration and without referenced material, describe the Main Steam Supply System control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable. Reactor Trip

62. The plant is discharging the contents of a gaseous waste decay tank 1GW-TK-1A to the atmosphere IAW 1OM-19.4.E, Decay Tank Discharge when a HIGH-HIGH radiation monitor alarm actuates on RM-1GW-108B, Gaseous Waste Gas Monitor.

Which of the following components will automatically operate to isolate the discharge?

1. FCV-1GW-105, Decay Tank Bleed Control Valve
2. 1GW-F-1A (1B), GW Disposal Blowers
3. TV-1GW-103A2, 1A GW Decay Tk Bleed Valve
4. TV-1GW-103, GW Decay Tank Disch to CTWR Valve

A. 1 and 2 only

B. 3 and 4 only

C. 1, 2 and 3 only

D. 1, 2, 3, and 4

Answer: B

Explanation/Justification: K/A is met with the candidate's knowledge that a high-high process radiation monitor alarm will cause the gaseous waste discharge flowpath to isolate to minimize an offsite dose release.

A. Incorrect. Plausible because FCV-1GW-105 is throttled during discharge and other plant FCVs have automatic isolation features, and GW Disposal Blowers are plausible because they provide GW flow, but they only trip on motor electrical protection trip.
B. Correct. TV-1GW-103A2 and TV-1GW-103 will both automatically close to isolate the GW discharge on process rad monitor RM-1GW-108B High-High radiation alarm.
C. Incorrect. Plausible because FCV-1GW-105 is throttled during discharge and other plant FCVs have automatic isolation features, and GW Disposal Blowers are plausible because they provide GW flow, but they only trip on motor electrical protection trip. TV-1GW-103A2 does isolate on high-high radiation alarm.
D. Incorrect. Plausible because TV-1GW-103A2 and TV-1GW-103 will both automatically close on high-high radiation alarm, but FCV-1GW-105 does not automatically close, and the GW Disposal Blowers do not stop on a high-high radiation alarm.

Sys #	System	Category	KA Statement
071	Waste Gas Disposal	K3 Knowledge of the effect that a loss or malfunction of the Waste Gas Disposal System will have on the following:	ARM and PRM systems

K/A# K3.05 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-19.1.D Rev. 1 pg. 5, 6
1OM-43.5.B.2 Rev. 4 pg. 2

Question Source: New

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

Objective: 1SQS-19.1 Rev. 17 Obj. 6. Given a Gaseous Waste Disposal System configuration and without reference material, describe the Gaseous Waste Disposal System field response to the following actuation signals, including automatic functions and changes in equipment status. a. High Radiation

63. Complete the following statements.

An event has caused the radiation levels in the Control room to rise.

1) When Control Room Area Radiation Monitor, RM-1RM-218A or RM-1RM-218B reach the _______________ setpoint, the Control Room Emergency Ventilation System (CREVS) will automatically isolate to protect Control Room personnel.

2) If CREVS does not automatically isolate and pressurize the Control Room due the elevated radiation levels, the control room staff must manually start CREVS by pushing the _______________ pushbuttons.

A. 1) WARN
2) Manual Control Room Emergency Air Supply Isolation

B. 1) HIGH
2) Manual Control Room Emergency Air Supply Isolation

C. 1) WARN
2) Control Room Emergency Air Supply Actuation

D. 1) HIGH
2) Control Room Emergency Air Supply Actuation

Answer: D

Explanation/Justification: K/A is met by the candidate's ability to predict when a Control Room area radiation monitor alarm will cause the Control Room Emergency Ventilation System to isolate and identifying what control switches need to be operated is the automatic isolation fails to occur.

A. Incorrect. WARN setpoint is plausible, but it is lower than the HIGH setpoint which actuates CREVS. Manual Control Room Emergency Air Supply Isolation pushbutton is plausible because it is also on the BSP and is used for closing the control room dampers during a Toxic Gas event, but the pressurization fans will not start (FN241A/B).

B. Incorrect. RM-1RM-218A or RM-1RM-218B in HIGH alarm will each actuate a train of CREVS to isolate the Control Room in the event of high area radiation. Manual Control Room Emergency Air Supply Isolation pushbutton is plausible because it is also on the BSP, and is used for closing the control room dampers during a Toxic Gas event, but the pressurization fans will not start (FN241A/B).

C. Incorrect. WARN setpoint is plausible, but it is lower than the HIGH setpoint which actuates CREVS. Control Room Emergency Air Supply Actuation is correct.

D. Correct. RM-1RM-218A or RM-1RM-218B in HIGH alarm will each actuate a train of CREVS to isolate the Control Room and start 2HVC-FN241A in 90 seconds, or FN241B in 120 seconds if FN241A fails to start in the event of high area radiation. Control Room Emergency Air Supply Actuation is correct.

Sys #	System	Category	KA Statement
072	Area Radiation Monitoring	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including:	Radiation levels

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

References provided to Candidate None **Technical References:** 1OM-43.4.ADP Rev. 3 pg. 3
1OM-44A.1.D Rev. 7 pg. 4, 10

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(5)

Objective: 1SQS-44A.1 Rev 7 Obj. 12. Describe the control, protection and interlock functions for the Control Room components associated with the Unit-1 Control Area Ventilation System, including automatic functions, setpoints and changes in equipment status as applicable.

64. Which of the below set of conditions are the **MINIMUM REQUIRED** conditions to actuate (light up) the C-9 "Condenser Available" Status light on Panel 622?

A. 1/2 Condenser Pressure >20" Hg vacuum
and
1/4 Circulating Water Pumps running

B. 1/2 Condenser Pressure >20" Hg vacuum
and
2/4 Circulating Water Pumps running

C. 2/2 Condenser Pressure >20" Hg vacuum
and
1/4 Circulating Water Pumps running

D. 2/2 Condenser Pressure >20" Hg vacuum
and
2/4 Circulating Water Pumps running

Answer: C

Explanation/Justification: K/A is met by the candidate's knowledge of the minimum inputs from the operating Circulating Water Pumps and Condenser Pressure transmitters required to actuate (light) the C-9, Condenser Available control permissive Status Light on Control Room Panel 622.

A. Incorrect. Plausible distractor but 2/2 Condenser Pressures must be >20" Hg vacuum. 1/4 Circulating Water Pumps running is the correct minimum to meet C-9 permissive.
B. Incorrect. Plausible distractor but 2/2 Condenser Pressures must be >20" Hg vacuum. 2/4 Circulating Water Pumps running is more CW pumps than are needed to meet C-9 permissive, therefore not the minimum.
C. Correct. 2/2 Condenser Pressures must be >20" Hg vacuum and 1/4 Circulating Water Pumps running are the correct minimums to meet C-9 permissive.
D. Incorrect. 2/2 Condenser Pressures must be >20" Hg vacuum is correct. 2/4 Circulating Water Pumps running is more CW pumps than are needed to meet C-9 permissive, therefore not the minimum.

Sys #	System	Category	KA Statement
075	Circulating Water	Generic	Knowledge of annunciator alarms, indications, or response procedures.

K/A# 2.4.31 **K/A Importance** 4.2 **Exam Level** RO
References provided to Candidate None **Technical References:** 1OM-1.5.B.8 Rev. 0 pg. 6

Question Source: New
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(7)

Objective: 1SQS-31.1, Rev. 14 Obj. 3. Describe the instrumentation, control, protection, and interlock functions for the field components associated with the Circulating Water System, including automatic functions, setpoints and changes in equipment status as applicable.

65. The plant is at 100% power.

TWO Heat Actuating devices (HD-1FP-6-1 & HD-1FP-6-2) on Unit Station Transformer 1C fail high and generate the following indication:

- Annunciator A8-60, Unit Sta Serv Trans 1C FIRE alarms

Assuming all systems function as designed, what impact, if any, will these conditions have on the operation of USST 1C?

The USST 1C Deluge valve will ______(1)______ and 4KV Busses 1A and 1B _____(2)_____ transfer to Offsite Power.

A. 1) OPEN
2) will

B. 1) not OPEN
2) will

C. 1) OPEN
2) will not

D. 1) not OPEN
2) will not

Answer: C

Explanation/Justification: K/A is met with the candidate's knowledge that failed heat actuating devices on the USST transformer will cause the fire protection deluge valve to open and spray down the transformer. Talked with Exam Chief about the K/A, he said to write the question as though a fire detector is bad and the effects it will have on the FP system.

A. Incorrect. The two failed HADs will actuate the Deluge valve. Second part is plausible since a transformer high differential overcurrent would cause an automatic transfer to offsite power and automatically open the deluge valve.

B. Incorrect. Plausible if the candidate thinks that it takes more than 2 HADs (There are 6 HADS on the transformer), or they think that a transformer high differential overcurrent is required. The 4KV busses will not transfer to offsite because a transformer high differential overcurrent does not exist.

C. Correct. The two failed HADs will actuate the Deluge valve. The 4KV busses will not transfer to offsite power because a transformer high differential overcurrent would be required to automatically transfer to offsite power and automatically open the deluge valve.

D. Incorrect. Plausible if the candidate thinks that it takes more than 2 HADs (There are 6 HADS on the transformer), or they think that a transformer high differential overcurrent is required. It is correct that 4KV busses will not transfer to offsite.

Sys #	System	Category	KA Statement
086	Fire Protection	K6 Knowledge of the effect of a loss or malfunction on the Fire Protection System following will have on the:	Fire, smoke, and heat detectors

K/A# K6.04 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-33.4.AAE Rev. 2 pg 2
1OM-33.1.D Rev. 6 Pg 4
3SQS-33.1 Rev. 10 pg. 29, 30, 79

Question Source: Bank – 1LOT14 Q65

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(5)

Objective: 3SQS-33.1-1-03: Describe the control, protection and interlock functions for the field components associated with the Fire Protection System, including automatic functions, setpoints and changes in equipment status as applicable.

3SQS-33.1-1-08: Given a specific plant condition, predict the response of the Fire Protection 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.

66. Based on NOP-OP-1002, Conduct of Operations, complete the following statement.

______(1)______ are for short-term communication only, and are only valid for __(2)__ days.

A. 1) Night Orders
2) 14

B. 1) Night Orders
2) 30

C. 1) Standing Orders
2) 14

D. 1) Standing Orders
2) 30

Answer: A

Explanation/Justification: K/A is met with the candidate's knowledge of the difference between Night Orders and Standing Orders as described in the Conduct of Operations procedure.

A. Correct. Night Orders are for short-term communication only and are only valid for 14 days per Conduct of Operations section 4.15.2.2.
B. Incorrect. Night Orders are for short-term communication. 30 days is a plausible distractor for short term communications.
C. Incorrect. Plausible because Standing Orders and Night Orders are used to disseminate operations required information, but Standing Orders are utilized to communicate longer term issues and should not be in place longer than a fuel cycle. 14 days is correct for night orders.
D. Incorrect. Plausible because Standing Orders and Night Orders are used to disseminate operations required information, but Standing Orders are utilized to communicate longer term issues and should not be in place longer than a fuel cycle. 30 days is a plausible distractor for short term communications.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

K/A# 2.1.15 **K/A Importance** 2.7 **Exam Level** RO
References provided to Candidate None **Technical References:** NOP-OP-1002 Rev. 15 pg. 75
Question Source: New
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(10)
Objective: 3SQS-48.1 Rev. 2 Obj. 7. From memory, describe the Operations shift rules of practice including: a. communications

67. While performing actions in E-3, Steam Generator Tube Rupture, the Unit Supervisor directs the BOP Operator to "Check ruptured Steam Generator pressure greater than 380 psig".

    Which ONE of the following BOP Operator responses would satisfy BVBP-OPS-0024, Transient Response Guidelines?

A. No, Ruptured Steam Generator pressure is 360 psig and rising.

B. No, Ruptured Steam Generator pressure is 350 psig.

C. Yes, Ruptured Steam Generator pressure is greater than 380 psig.

D. Yes, Ruptured Steam Generator pressure is rising rapidly.

Answer: A

Explanation/Justification: K/A is met with the candidate's ability to make clear and concise verbal reports to the Unit Supervisor when performing steps in the EOP network.

A. Correct. Transient response Guidelines requires a response containing yes/no, value, and trend.
B. Incorrect. Plausible distractor but the report is missing the expected trend.
C. Incorrect. Plausible distractor but the report is missing a value, and a trend. It does answer the US question, but does not meet the expectations.
D. Incorrect. Plausible distractor but the report is missing a value.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to make accurate, clear, and concise verbal reports.

K/A# 2.1.17 **K/A Importance** 3.9 **Exam Level** RO
References provided to Candidate None **Technical References:** BVBP-OPS-0024 Rev. 13 pg. 11
Question Source: Modified – 2017 Point Beach Q67
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(10)
Objective: 3SQS-48.1 Rev. 2 Obj. 7. From memory, describe the Operations shift rules of practice including: a. communications

68. Given the following conditions:

- You are a Licensed Reactor Operator at Beaver Valley.
- You have been Licensed for Five and one-half years.
- Your License renewal medical examination (NRC Form 396, Certification Of Medical Examination By Facility Licensee) is due to the NRC Regional Administrator in 6 months.
- Your License is "Active" and you are currently assigned as the Unit 1 Reactor Operator.
- Your License contains **NO** medical restrictions.
- You have been experiencing some difficulties with your "distant" vision.

On your first relief day, your personal physician (a licensed optometrist) determines that your "distant" vision has permanently degraded, and you will **NOW** be required to wear corrective lenses at all times.

IAW 10CFR 50.74, Notification of Change In Operator or Senior Operator Status, what is the maximum **REQUIRED** time that the NRC Regional Administrator must be notified of this change in your medical status?

A. Immediately.

B. Within 30 days of the diagnosis.

C. Within 60 days of the diagnosis.

D. At the time of License renewal.

Answer: B

Explanation/Justification: K/A is met with the required knowledge that if a medical condition changes (corrective lenses) which requires NRC notification iaw 10CFR50.74 and 10CFR55, the licensed induvial must notify the NRC within 30 days of learning of the diagnosis.

A. Incorrect. You should immediately begin wearing the corrective lenses, but not required to report for 30 days.
B. Correct. IAW 10CFR 50.74, 55.25, 55.23, NRC form 396. (Beaver Valley specific OE CR 07-22311)
C. Incorrect. Plausible but 60 days exceeds the 30 day notification requirement.
D. Incorrect. Plausible because corrective lenses are a condition of the license as identified on NRC Form 396, but notification of the change of condition must be reported within 30 days of the diagnosis.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc..

K/A# 2.1.4 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** 3SQS-48.1 PPNT Rev. 24 Slide 223, 224 which reference 10CFR 50.74, 55.25, 55.23, NRC form 396

Question Source: Bank – 2LOT6 Q67

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

Objective: 3SQS-48.1, Rev. 24 Obj. 25. Given a set of conditions and appropriate reference material, explain the licensee responsibilities IAW 10CFR50.74, Reg Guide 1.134 and ANSI/ANS Standard 3.4.

69. Reactor startup in progress in accordance with 1OM-50.4.D1, Reactor Startup from Mode 3 to Mode 2 - Initial Criticality.
The ATC is preparing to withdraw Control Bank 'A' Rods.

Which of the choices below completes the following statement?

While withdrawing Control Bank 'A' rods IAW 1OM-50.4.D1, the CONTROL ROD BANK SEL SWITCH is required to be in the ______(1)______ position and the maximum Intermediate Startup Rate (SUR) allowed is ______(2)______.

A. 1) CBA
2) 0.3 DPM

B. 1) CBA
2) 0.5 DPM

C. 1) MAN
2) 0.3 DPM

D. 1) MAN
2) 0.5 DPM

Answer: D

Explanation/Justification: K/A is met by demonstrating the ability to manipulate the Control Rod Bank Selector Switch, and control reactivity during a reactor startup.

A. Incorrect. Plausible because the shutdown banks are withdrawn individually, but the control banks A-D are withdrawn sequentially which is controlled by cod control circuitry. Second part is plausible because -0.3 dpm is the SUR after a reactor trip until the delayed neutrons decay off.
B. Incorrect. Plausible because the shutdown banks are withdrawn individually, but the control banks A-D are withdrawn sequentially which is controlled by rod control circuitry. 0.5 dpm is the maximum SUR allowed per P&L #12 of 1OM-50.4.D1.
C. Incorrect. Manual is correct. Second part is plausible because -0.3 dpm is the SUR after a reactor trip until the delayed neutrons decay off.
D. Correct. Control Rod Bank Selector Switch is in Manual when withdrawing control banks A-D because the rod control circuitry controls the bank withdrawal sequentially. 0.5 dpm is the maximum SUR allowed per P&L #12 of 1OM-50.4.D1.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Generic	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

K/A# 2.2.2 **K/A Importance** 4.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-50.4.D1 Rev. 1 pg. 15, 25

Question Source: New

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

Objective: 3LOT-M4D1, Rev. 5 Obj. 1. Explain Precautions and Limitations, Reactor Theory and Kinetics applicable to the startup, in accordance with 1OM-50.4.D (2OM-50.4.D2), Reactor Startup from Mode 3 to Mode 2 and BVPS Reactor Theory Manual.
3LOT-M4D1, Rev. 5 Obj. 3. Perform a Reactor Startup, in accordance with 1OM-50.4.D (2OM-50.4.D2), Reactor Startup from Mode 3 to Mode 2.

70. Given the following conditions:

- The plant is in Mode 3.
- The following leaks are known to the crew:
  - The ‘A’ S/G has a 0.04 gpm tube leak
  - The ‘B’ S/G has a 0.00 gpm tube leak
  - The ‘C’ S/G has a 0.03 gpm tube leak
  - PCV-1RC-456, PRZR PORV has 1.2 gpm of seat leakage
  - MOV-1RC-557A, ‘A’ RCL DRAIN valve inlet weld has a 0.2 gpm leak
  - MOV-1CH-289, Regen Hx/Chg Hdr Inlet valve has 0.2 gpm valve stem leakage
  - MOV-1RC-591, ‘A’ RCL Cold Leg Isolation valve has 0.4 gpm valve stem leakage

The RCS Water Inventory Balance surveillance was just performed and indicates **4 gpm** of total RCS leakage.

Which Tech Spec RCS Operational leakage limits have been exceeded?

A. Pressure Boundary Leakage and Unidentified Leakage

B. Unidentified Leakage and Identified Leakage

C. Identified Leakage and Primary to Secondary Leakage

D. Primary to Secondary Leakage and Pressure Boundary Leakage

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Answer: A

Explanation/Justification: K/A is met with the candidate’s ability to recognize the entry conditions into Tech Spec 3.4.13 for RCS Operational LEAKAGE.
TS 3.4.13 states operational leakage shall be limited to 1) No pressure boundary leakage 2) 1 gpm unidentified 3) 10 gpm identified 4) 150 gpd though any one SG.

A. Correct. Pressure boundary leakage is correct due to MOV-1RC-557A valve inlet weld leakage of 0.2 gpm and TS allows 0.0 gpm leakage from a pressure boundary. Unidentified Leakage is correct as it is 2 gpm and TS only allows 1 gpm leakrate.
B. Incorrect. Unidentified Leakage is correct as it is 2 gpm and TS only allows 1 gpm leakrate. Identified Leakage is plausible because the candidate may get confused with the 1 gpm Unidentified leakage allowed by TS, but Identified is allowed to be 10 gpm, and the above total is 2 gpm.
C. Incorrect. Identified Leakage is plausible because the candidate may get confused with the 1 gpm Unidentified leakage allowed by TS, but Identified is allowed to be 10 gpm, and the above total is 2 gpm. Primary to Secondary Leakage is plausible because there was tube leakage identified which equates to ‘A’ SG – 57.6 gpd and ‘C’ SG -43.2 gpd, but TS allowable leakage is 150 gpd though any one SG.
D. Incorrect. Primary to Secondary Leakage is plausible because there was tube leakage identified which equates to ‘A’ SG – 57.6 gpd and ‘C’ SG - 43.2 gpd, but TS allowable leakage is 150 gpd though any one SG. Pressure boundary leakage is correct due to MOV-1RC-557A valve inlet weld leakage of 0.2 gpm and TS allows 0.0 gpm leakage from a pressure boundary.

Sys #	System	Category	KA Statement
N/A	N/A	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:** Tech. Spec 3.4.13 pg. 3.4.13 - 1

Question Source: New

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

Objective: 3SQS-RCS ITS, Rev. 1 Obj. 1. Apply the following definitions to ensure compliance with applicable requirements: a. LEAKAGE
3SQS-RCS ITS, Rev. 1 Obj. 3. Given plant conditions, determine the criteria necessary to ensure compliance with each RCS LCO and Licensing Requirement in accordance with the Bases, Surveillance Requirements, and the Applicability.

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

- A Reactor Startup is in progress following a refuel outage.
- The RO has just completed taking critical data and is directed to raise power to 4%.
- The RO establishes a positive startup rate and releases the IN-HOLD-OUT switch.
- Rods continue to step outward as indicated on group step counters and CERPI.

Which of the following describes the first required immediate action **AND** the consequence if **NO** operator action is taken?

Per the appropriate AOP, the first required immediate action is to ___(1)___.

The consequence of inaction is that reactor power will increase to ___(2)___ before being automatically terminated by the Reactor Protection System _____(3)_____.

A. 1) manually trip the reactor
2) 20%
3) Intermediate Range Rod Stop Signal

B. 1) place Control Rod Group Selector to Auto
2) 25%
3) Power Range Reactor Trip Signal

C. 1) manually trip the reactor
2) 25%
3) Intermediate Range Reactor Trip Signal

D. 1) place Control Rod Group Selector to Auto
2) 20%
3) Power Range Reactor Trip Signal

Question 71

Answer: B

Explanation/Justification: K/A is met by the candidates ability to interpret indications provided and recognize an uncontrolled rod withdrawal is in progress, then apply the appropriate immediate action and systems knowledge to demonstrate understanding of operator actions and impact on system conditions if these actions are not taken.

A. Incorrect. Manually tripping the reactor is an action required if the control rods are in auto and rod motion continues, but since the rods are manually being withdrawn, the immediate action is to check/place the group selector switch to auto. Since there is an intermediate rod stop at 20% (C-1), this is a plausible distractor.

B. Correct. The immediate action for unexpected control rod movement is to check the control rods in auto, if not, then place control rod bank select switch to auto. A reactor trip will automatically occur in this scenario when 2/4 power range channels are above 25% because the low power trip was not manually blocked above P-10 (10%) because no operator action is taken.

C. Incorrect. Manually tripping the reactor is an action required if the control rods are in auto and rod motion continues, but since the rods are manually being withdrawn, the immediate action is to check/place the group selector switch to auto. It is correct that a reactor trip will occur at 25% based on intermediate current equivalent neutron flux since reactor power is <P-10.

D. Incorrect. Correct action. Incorrect low power range setpoint. 20% is plausible because of the intermediate rod stop at 20% (C-1).

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

K/A# 2.2.44 **K/A Importance** 4.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.1.3 Rev. 15, pg. 2; 1OM-1.5.B.8 Rev. 0 pg. 3

Question Source: Modified – 1LOT8 Q69

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.41.b(5)

Objective: 1SQS-53C.1 Rev. 12 Obj. 1. State all Immediate Operator Actions associated with the AOPs.

1SQS-53C.1 Rev. 12 Obj. 2. State the conditions or symptoms that would require entry into the AOPs.

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.

72. A ______(1)______ Radiation Monitor alarm on RM-1SV-100, Condenser Air Ejector Vent Radiation Monitor is expected to align the Air Ejector discharge to the ______(2)______ to prevent a radioactive release during a Steam Generator Tube Leak.

A. 1) High
   2) SLCRS Main Filter Bank

B. 1) High
   2) Containment

C. 1) High-High
   2) SLCRS Main Filter Bank

D. 1) High-High
   2) Containment

Answer: D

Explanation/Justification: K/A is met with the candidate's ability to understand that a SG Tube Leak could generate a High-High radiation alarm on the Air Ejector discharge which causes the AE discharge to be automatically aligned to the Containment to prevent a radiological release to the environment.

A. Incorrect. Plausible because a high rad monitor alarm is an indication of a SG tube leak, but the automatic alignment of the AE discharge does not occur until the High-High alarm is received. SLCRS Main Filter Bank is plausible because several radiation monitors realign flow to the main filter banks on a high-high radiation alarm signal.

B. Incorrect. Plausible because a high rad monitor alarm is an indication of a SG tube leak, but the automatic alignment of the AE discharge does not occur until the High-High alarm is received.

C. Incorrect. A High-High Radiation Monitor alarm on RM-1SV-100 will Close TV-1SV-100B, Air Ejector Air Disch. to Gas Waste, and Open TV-1SV-100A, Air Ejector Air Disch to Containment to prevent the radioactive release. SLCRS Main Filter Bank is plausible because several radiation monitors realign flow to the main filter banks on a high-high radiation alarm signal.

D. Correct. A High-High Radiation Monitor alarm on RM-1SV-100 will Close TV-1SV-100B, Air Ejector Air Disch. to Gas Waste, and Open TV-1SV-100A, Air Ejector Air Disch to Containment to prevent the radioactive release.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Generic	Ability to control radiation releases.

K/A# 2.3.11 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-43.1.C Rev. 12 pg. 8

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(12)

Objective: 1SQS-43.1-01-10 Rev 16 Given a Radiation Monitoring System alarm condition, determine the appropriate alarm response, including automatic and operator actions in the control room.
1SQS-43.1-01-7: Rev 16 Describe the control, protection and interlock functions for the control room components associated with the Radiation Monitoring System, including automatic functions, and changes in equipment status as applicable.

73. 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. Continue venting the system until complete, or the EAD dose limit alarms, then exit the area.

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, exit the area, and notify Radiation Protection (RP).

Answer: D

Explanation/Justification: K/A is met with the candidate's knowledge of the expected actions is they receive an unexpected dose rate alarm while performing a job inside a Locked High Radiation Area (LHRA).

A. Incorrect. This could be possible if the candidate thought that the dose rate alarm was a warning, or getting the job done was more important.
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-4101-06 Rev. 3 form
FEN-RWT Rev 4 Chapter 6 pg. 4

Question Source: Bank – 2LOT17 Q72

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41 b(12)

Objective: FEN-RWT Rev 4 Chapter 6 obj. 6. State the action(s) to be taken if dosimetry is lost, off-scale, or alarming.

74. The Unit 1 Shift Manager was outside the Control Room and unable to return due to health issues, when an event occurred at Unit 1 which required implementation of the Emergency Plan.

Which of the following positions will assume the role of Emergency Director until the Technical Support Center (TSC) is operational?

A. Unit 1 Unit Supervisor

B. Shift Technical Advisor

C. Unit 2 Unit Supervisor

D. Unit 2 Shift Manger

Answer: D

Explanation/Justification: K/A is met with the candidate's knowledge that during implementation of the emergency plan, if the Shift Manager is unavailable, the opposite unit Shift Manager will assume the authority for the site.

A. Incorrect. This would be a plausible choice because the Unit 1 Unit Supervisor is a qualified SRO which could fill the role, but must remain focused on directing the crew response to the emergency situation.

B. Incorrect. Plausible because the STA could fill the SM position if SRO licensed but it is not permitted iaw EEP Plan section 5 page 5-40 note (e).

C. Incorrect. Plausible because the Unit 2 US is a licensed SRO but must remain in the Command SRO position for Unit 2.

D. Correct. Per NOP-OP-1002, At BV, if the SM cannot immediately return to the Control Room, the US SHALL request the other Unit's SM to assume EPP responsibility as Emergency Director.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the lines of authority during implementation of the emergency plan.

K/A# 2.4.37 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-OP-1002 Rev. 15 pg 13, 14

Question Source: New

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

Objective: 3SQS-48.1 Rev. 24 Obj. 1. From memory, explain the duties and responsibilities of Operations personnel.
3SQS-48.1 Rev. 24 Obj. 3. From memory, describe the required actions if less than the minimum shift staffing complement exists.

75. Given the following conditions:
- The plant is cooling down for a refueling outage.
- RCS Tavg is 325°F
- RCS pressure is 300 psig.

1) What is the current mode of operation?

2) For the current plant conditions, Tech Specs require how many trains of Safety Injection to be OPERABLE to mitigate a Loss of Coolant Accident?

A. 1) Mode 3
2) One ECCS train shall be OPERABLE.

B. 1) Mode 3
2) Two ECCS trains shall be OPERABLE.

C. 1) Mode 4
2) One ECCS train shall be OPERABLE.

D. 1) Mode 4
2) Two ECCS trains shall be OPERABLE.

Answer: C

Explanation/Justification: K/A is met with the candidate's knowledge that only one train of ECCS is required to be operable when the plant is in hot shutdown to mitigate a loss of coolant accident due to the reduced probability of occurrence of the accident.

A. Incorrect. Plausible if the candidate is not familiar with the Modes of Operation, and the criteria for each. Mode 3 is hot Standby with Tavg ≥350F. Per TS 3.5.3, only one train of ECCS is required to be OPERABLE in Mode 4 due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident.
B. Incorrect. Plausible if the candidate is not familiar with the Modes of Operation, and the criteria for each. Mode 3 is hot Standby with Tavg ≥350F. Two ECCS trains required to be OPERABLE is plausible because TS 3.5.2 ECCS – Operating requires two trains to be OPERABLE when the plant is in modes 1-3.
C. Correct. The plant is in mode 4, hot shutdown with 350F>Tavg>200F. Per TS 3.5.3, only one train of ECCS is required to be OPERABLE in Mode 4 due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident.
D. Incorrect. The plant is in mode 4, hot shutdown with 350F>Tavg>200F. Two ECCS trains required to be OPERABLE is plausible because TS 3.5.2 ECCS – Operating requires two trains to be OPERABLE when the plant is in modes 1-3.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

K/A# 2.4.9 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** Tech Spec Table 1.1-1
Tech Spec 3.5.3 Page 3.5.3 - 1

Question Source: New

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

Objective: 3SQS-ECCS ITS, Rev. 1 Obj. 3. Given plant conditions, determine the criteria necessary to ensure compliance with each Section Emergency Core Cooling System LCO and Licensing Requirement in accordance with the Bases, Surveillance Requirements, and the Applicability.

76. The plant was operating at 100% power when a large break LOCA occurred coincident with a Loss of 4KV Bus DF.

The follow conditions exist:

- ‘A’ Quench Spray Pump [1QS-P-1A] tripped on startup
- 3 Max CETs indicate 810^{o}F
- CNMT Pressure is 31 psig
- CNMT Temperature is 240^{o}F
- All RCPs have been tripped
- RVLIS Full Range indicates 35%
- 4KV Bus DF remains de-energized during this event

Based on the above conditions, which answer below completes the following statement?

The required EAL classification is based upon the _______________.

A. LOSS of one fission product barrier and POTENTIAL Loss of another barrier

B. LOSS of two fission product barriers

C. LOSS of two fission product barriers and a POTENTIAL loss of a third barrier

D. LOSS of all three fission product barriers

Question 76

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II G page 9 seventh bullet. SRO is required to have knowledge of the Emergency Classifications. This is an SRO position function only.

K/A is met by demonstrating the knowledge to determine the event classification based on the conditions given using the provided EPP classification chart.

Candidate will have to recognize a General Emergency would be declared based on the following conditions which they will have to interpret from the conditions given in the stem.
FC – Loss due to FR-C.1 Red Path Entry
RCS - Loss due to an automatic or manual ECCS (SI) actuation required by EITHER an Unisolable RCS Leakage or SG tube rupture.
CT – Potential Loss due to Cnmt pressure >11 psig AND less than one full train of depressurization equipment operating. One train of QS System and one train of RS System comprise one full train of depressurization equipment as designed and with 'A' QS tripped, and DF bus de-energized, both QS and RSS pumps are not available on 'B' train. The candidate may select a potential loss for Core cooling and procedures not effective based on note 1.

A. Incorrect. Plausible if it is not recognized that FR-C.1 entry conditions have been met for Fuel Clad failure, or a Loss based on RCS Leak Rate.
B. Incorrect. Plausible if it is not recognized that Potential Loss due to Cnmt pressure >11 psig AND less than one full train of depressurization equipment operating.
C. Correct. GE based on answer explanation above.
D. Incorrect. Plausible if it is not recognized that containment barrier is a potential loss based on Cnmt pressure >11 psig AND less than one full train of depressurization equipment operating, or possibly thinking Core cooling conditions met and procedures are not effective, but there is no loss of the barrier.

Sys #	System	Category	KA Statement
000011	Large Break LOCA / 3	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 EPP Wallboard **Technical References:** 1OM-53A.1.F-0.2 Iss. 3 Rev. 0
EPP-PLAN-SECTION-4 Rev. 33 pg. 4-192

Question Source: Bank – 2LOT15 Q77

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: EPP-9281, Rev. 13 Obj. 11. Given specific plant conditions, classify the condition in accordance with EPP I-1a & b.

77. The plant is shutting down for a Refueling Outage.

- RCS pressure is 250 psig and lowering
- Tcold is 190°F and lowering
- Annunciator A4-3, Pressurizer Control Level Low is in alarm
- PRZR Cold Calibration level instrument LI-1RC-462 is indicating 0%
- Vessel Level is NOT in service
- Annunciator A4-71, Radiation Monitoring High is in alarm
- RM-1RM-204, Incore Instrument Room radiation monitor is in High alarm
- Cnmt sump level is rising
- Both RHR pump amps and flowrates are fluctuating

Based on the above conditions:

1) What procedure will the Unit Supervisor enter to address these plant conditions?
2) What actions will be taken for the current conditions per the AOP?

A. 1) AOP-1.6.5, Shutdown LOCA
2) Trip the RHR pumps and isolate the RHR system from the RCS.

B. 1) AOP-1.6.5, Shutdown LOCA
2) Adjust RHR flow to maintain >1000 gpm and maintain stable pump amps and flowrate.

C. 1) AOP-1.10.1, Loss of Residual Heat Removal Capability
2) Trip the RHR pumps and isolate the RHR system from the RCS.

D. 1) AOP-1.10.1, Loss of Residual Heat Removal Capability
2) Adjust RHR flow to maintain >1000 gpm and maintain stable pump amps and flowrate.

Question 77

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E first paragraph on page 6. Specifically, the SRO must have specific knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose. In the case of this question, the candidate must have detailed knowledge of the Loss of Shutdown Cooling AOP and the requirements to trip the RHR pumps when the indications of cavitation are present, verses minimizing RHR pump flow to stabilize flow and amps based on their knowledge of przr level indication systems available in Mode 5 during a loss of coolant event.

K/A is met with the candidate's ability to interpret the przr level alarm in combination with the przr cold calibrated level indication, and pump cavitation indications, and determine that the RHR pumps must be tripped iaw the Loss of Shutdown Cooling AOP when a LOCA occurs in Mode 5. NRC Chief said that verifying alarm setpoints would be met by the candidate choosing a plant procedure and determining what actions must be taken.

A. Incorrect. AOP1.6.5 is plausible because it would be entered if the plant was in mode 3 with the SI accumulators isolated, or mode 4, but the stem states Tcold is 190F which indicates mode 5. Tripping the RHR pumps and isolating RHR is correct.

B. Incorrect. AOP1.6.5 is plausible because it would be entered if the plant was in mode 3 with the SI accumulators isolated, or mode 4, but the stem states Tcold is 190F which indicates mode 5. Adjusting RHR flow to maintain >1000 gpm and maintain stable pump current is plausible because it is the required action if the przr cold calibrated level were >5%, or <5% with Vessel level in service indicating >61 inches with the pumps showing signs of cavitation.

C. Correct. AOP-1.10.1 is correct with Tcold at 190F, the plant is in Mode 5 which requires entry into AOP-1.10.1 for przr level lowering. Tripping the RHR pumps is correct since przr cold cal level is <5% and Vessel level is not in service iaw step 1c of the AOP.

D. Incorrect. AOP-1.10.1 is correct with Tcold at 190F, the plant is in Mode 5 which requires entry into AOP-1.10.1 for przr level lowering. Adjusting RHR flow to maintain >1000 gpm and maintain stable pump current is plausible because it is the required action if the przr cold calibrated level were >5%, or <5% with Vessel level in service indicating >61 inches with the pumps showing signs of cavitation.

Sys #	System	Category	KA Statement
000025	Loss of Residual Heat Removal System / 4	Generic	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

K/A# 2.4.50 **K/A Importance** 4.0 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.10.1 Rev. 19 pg. 1-5

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(10)

Objective: 1SQS-53C.1, Rev. 12 Obj. 5. Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.

1SQS-53C.1, Rev. 12 Obj. 6. Given a set of conditions, apply the correct AOP.

78. The plant was operating at 100% power and experienced a LOCA causing a Reactor Trip and Safety Injection to occur.

- A loss of All AC power occurred on the transfer to Off-Site power.
- Both Diesel Generators failed to start
- The crew is performing ECA-0.0, Loss of All Emergency 4KV Power.
- 4160VAC Bus 1DF has been selected for cross-tie with Unit 2 and is expected to be completed in 45 minutes.
- The crew is at step 30, Depressurize Intact SGs to 620 psig.
- The Field Operator just reported Attachment 2-E, Local Action to Restore AC Power is complete, and EDG #1 is supplying emergency bus 1AE.

Based on the above conditions, what is the status of the High Head Safety Injection (HHSI) pumps when the crew is exiting ECA-0.0?

A. Only ‘A’ HHSI pump is running

B. Only ‘B’ HHSI pump is running

C. Both HHSI pumps are running

D. No HHSI pumps are running

Question 78

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E first paragraph on page 6. Specifically, the SRO must have specific knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose. In the case of this question, the candidate must have detailed knowledge of ECA-0.0 and determine that the safety injection pumps will not be started in ECA-0.0 after power has been restored.

K/A is met with the candidate's ability to determine that even though an emergency power source has been restored in ECA-0.0, Loss of All Emergency 4KV Power, the Safety Injection/HHSI pumps are not energized in the procedure due to procedural design of ECA-0.0, to ensure RCP damage does not occur.

A. Incorrect. Plausible to think only the 'A' HHSI pump is running because the caution statement prior to step 47 states all subsequent steps and procedures must be performed using equipment that is powered from a BV-1 source. This is where the crew would be when a BV1 emergency bus is restored and 'A' HHSI would be the BV-1 electrical supplied pump, but no HHSI pumps are started in ECA-0.0.

B. Incorrect. Plausible to think only the 'B' HHSI pump is running if the candidate thinks that the transition from ECA-0.0 does not occur until the SBO cross-tie is complete, and thinks it is a more reliable source, or is concerned with EDG loading based on several notes/cautions in ECA-0.0.

C. Incorrect. Plausible because two HHSI pumps would appear to be expected since there was a LOCA, but ECA-0.0 has the recovery procedures (ECA-0.1 or ECA-0.2) ensure the RCP seals have been isolated prior to starting the charging/HHSI pumps to ensure RCP seal failure is minimized.

D. Correct. No HHSI pumps are started in ECA-0.0 to ensure RCP seal damage will not occur. Both procedures that ECA-0.0 transitions to, ECA-0.1 and ECA-0.2, both ensure that the RCP seals are isolated at the beginning of the procedure to prevent seal damage when the Charging/HHSI pump is started.

Sys #	System	Category	KA Statement
000056	Loss of Offsite Power / 6	AA2. Ability to determine and interpret the following as they apply to the Loss of Offsite Power:	Operational status of safety injection pump

K/A# AA2.03 **K/A Importance** 3.9 **Exam Level** SRO

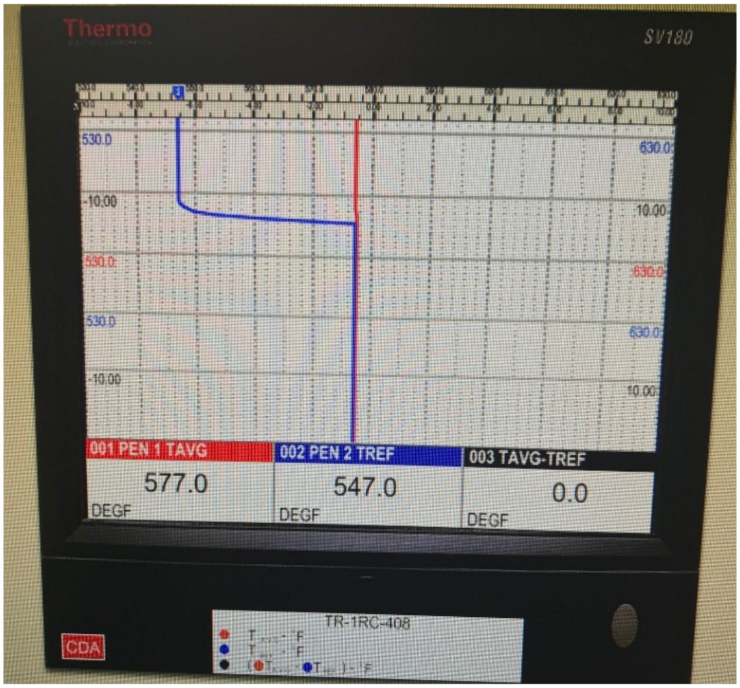
References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-0.0 Iss.3 Rev. 3 pg. 32
1OM-53B.4.ECA-0.0 Iss.3 Rev. 3 pg. 191,193,194

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.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.

79.



Given the following conditions:

- The plant is at 100% power
- Control rods are in Manual for testing
- Turbine 1st Stage Press selector switch is selected to the '446' position
- No operator actions have been taken

1) Based on the above plant conditions and the Tavg/Tref picture, what failure has occurred?

2) Which of the following Tech Spec LCOs must be entered?
   3.8.7, Inverters – Operating
   3.8.9, Distribution Systems – Operating

A. 1) Loss of Vital Bus 3
   2) Both 3.8.7 and 3.8.9

B. 1) Loss of Vital Bus 3
   2) 3.8.9 only

C. 1) Loss of Vital Bus 4
   2) Both 3.8.7 and 3.8.9

D. 1) Loss of Vital Bus 4
   2) 3.8.9 only

Question 79

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B third bullet on page 3. Specifically, the SRO must have knowledge of the bases for TS TS 3.8.7, Inverters – Operating and TS 3.8.9, Distribution Systems – Operating, so that they can recognize that vital bus 3 being de-energized will enter both LCOs.

K/A is met by the candidate recognizing that PT446 first stage pressure indicator is selected, and is indicated on the Tref chart recorded, then determines that vital bus 3 has been lost by the downscale temperature indication of 547F.

A. Correct. Candidate must recognize that PT-1MS-446, Ch. III First Stage Pressure has lost power, and recognize that it is powered from vital bus 3. Tech Spec LCO 3.8.7 and 3.8.9 both must be entered due to the loss of the vital bus. For TS 3.8.7, Inverters – Operating, the LCO state the required train A and B inverters shall be operable. The candidate must be knowledgeable of the TS bases and know that an OPERABLE inverter requires the associated vital bus to be powered by the inverter with output voltage within tolerances, and power input to the inverter from a 125 VDC station battery. Alternatively, power supply may be a battery charger or from an internal AC source via rectifier as long as the station battery is available as the uninterruptible power supply. Tech Spec 3.8.9, Distribution Systems – Operating bases requires the AC, DC, and AC vital bus electrical power distribution subsystems to be OPERABLE, and that an OPERABLE vital bus electrical power distribution subsystems require the associated buses to be energized to their correct voltage from the associated inverter via inverted DC voltage, inverter using internal AC source, or Class 1E constant voltage transformer.

B. Incorrect. Vital bus 3 is correct. TS 3.8.9 only is plausible if the candidate does not know the TS 3.8.7 bases which states an OPERABLE inverter requires the associated vital bus to be powered by the inverter with output voltage within tolerances.

C. Incorrect. Vital bus 4 is plausible because PT-1MS-447, Ch. IV First Stage Pressure is powered from Vital bus 4, but the stem of the question states that PT446 is selected which causes the chart recorder temperature to drop. Both TS 3.8.7 and 3.8.9 are correct as stated above.

D. Incorrect. Vital bus 4 is plausible because PT-1MS-447, Ch. IV First Stage Pressure is powered from Vital bus 4, but the stem of the question states that PT446 is selected which causes the chart recorder temperature to drop. TS 3.8.9 only is plausible if the candidate does not know the TS 3.8.7 bases which states an OPERABLE inverter requires the associated vital bus to be powered by the inverter with output voltage within tolerances.

Sys #	System	Category	KA Statement
000057	Loss of Vital AC Instrument Bus / 6	AA2. Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus:	T-ave. and T-ref. chart recorder.

K/A# AA2.09 **K/A Importance** 3.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.38.1C Rev. 5 pg. 13
Tech Spec 3.8.7 pgs. 3.8.7-1, B3.8.7-2
Tech Spec 3.8.9 pgs. 3.8.9-1, B3.8.9-2

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 3SQS-ELEC ITS, Rev. 1 Obj. 3. Given plant conditions, determine the criteria necessary to ensure compliance with each Section Electrical Power Systems LCO and Licensing Requirement in accordance with the Bases, Surveillance Requirements, and the Applicability.

80. The plant is operating at 15% power with all systems in normal alignment for this power level.

- SGWLC is being maintained automatically by the SG Feedwater Bypass Control Valves FCV-1FW-479(489)(499).
- Annunciator A6-109, Station Instrument Air Receiver Tank Discharge Pressure Low is received.
- Station Instrument Air Header Pressure is 80 psig and slowly dropping.

1) IAW AOP 1.34.1, Loss of Station Instrument Air, what directions are you **REQUIRED** to give the local operator to address the degrading Station Instrument Air Header Pressure?

2) IF Station Instrument Air Header Pressure continues to drop below 30 psig, how will the SG Feedwater Bypass Control Valves **FAIL**?

A. 1) Place the Standby Instrument Air dryer in service, **THEN** isolate the malfunctioned Instrument Air dryer IAW 1OM-34.4.N, Operation of Instrument Air Dryers.
2) **OPEN**

B. 1) Place the Standby Instrument Air dryer in service, **THEN** isolate the malfunctioned Instrument Air dryer IAW 1OM-34.4.N, Operation of Instrument Air Dryers.
2) **CLOSED**

C. 1) Place the Instrument Air Bypass filters in service, **THEN** isolate the Instrument Air dryers IAW AOP 1.34.1, Att. C, Locally Bypassing Instrument Air Dryers.
2) **OPEN**

D. 1) Place the Instrument Air Bypass filters in service, **THEN** isolate the Instrument Air dryers IAW AOP 1.34.1, Att. C, Locally Bypassing Instrument Air Dryers.
2) **CLOSED**

Question 80

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E first bullet on page 7. Specifically, the SRO must have specific knowledge of the Attachments included in the Loss of Instrument Air AOP and understand what work direction is given to the field operator to mitigate the loss of Instrument Air.

K/A is met by the candidate's ability to navigate the Loss of Instrument Air AOP and determine that they must direct the field operator to locally bypass the instrument air dryers in accordance with the AOP attachment to assist in mitigating the event. Discussed the K/A being RO level with the NRC and was instructed to word the question as to the SRO directing an activity outside the control room as directed by a plant procedure.

A. Incorrect. Plausible distractor because there are two instrument air dryers, and with the malfunction of the in-service air dryer it would appear reasonable to place the standby air dryer in service, but attachment C directs placing the bypass filters in service and isolating both air dryers. Second part is plausible because some air operated valves fail open on a loss of, or low air pressure, but the BFRVs fail closed

B. Incorrect. Plausible distractor because there are two instrument air dryers, and with the malfunction of the in-service air dryer it would appear reasonable to place the standby air dryer in service, but attachment C directs placing the bypass filters in service and isolating both air dryers. Second part is correct.

C. Incorrect. First part is correct. Second part is plausible because some air operated valves fail open on a loss of, or low air pressure, but the BFRVs fail closed.

D. Correct. IAW AOP-1.34.1 step 4 refers to attachment C which places the bypass filters in service and isolates the air dryers. A note prior to step 13 states that when station instrument air pressure drops to <30 psig, the valves listed in attachment A will begin to travel to their failed position. Attachment A identifies the SG Bypass FRVs as failing closed.

Sys #	System	Category	KA Statement
000065	Loss of Instrument Air / 8	Generic	Ability to locate and operate components, including local controls.

K/A# 2.1.30 **K/A Importance** 4.0 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.34.1 Rev. 28 pg. 3, 7, 19, 25

Question Source: Bank – 2LOT19 Q90

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 1SQS-53C.1, Rev. 12 Obj. 5. Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.

1SQS-34.1, Rev. 15 Obj. 5. Given a Compressed Air System configuration and without referenced material, describe the Compressed Air System field response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable: a. Loss of instrument air

81. The plant is operating at 100% power with all systems in NSA.

- A LOCA **OUTSIDE** containment occurs.
- At step 20 of E-0, Reactor Trip or Safety Injection, the crew enters ECA 1.2, LOCA Outside Containment.
- At the completion of ECA 1.2, the crew has been **UNABLE** to locate and isolate the break.

The following plant conditions **NOW** exist:

- All SG pressures are 800 psig and stable.
- All SG NR levels are 35% and slowly rising.
- All Secondary radiation monitors are consistent with pre-event values.
- CNMT Pressure is -1.0 psig and stable.
- CNMT sump level is consistent with pre-event values.
- CNMT radiation is consistent with pre-event values.
- RCS Subcooling is 40°F and slowly dropping.
- AFW flow is 700 gpm and stable.
- RCS Pressure is 1125 psig and slowly dropping.
- PRZR level is 12% and slowly dropping.
- Auxiliary Building Radiation levels are rising.
- Auxiliary Building sump levels are rising.

Based on these conditions:

What procedural transition is **REQUIRED**?

A. E-0, Reactor Trip or Safety Injection

B. ECA-1.1, Loss of Emergency Coolant Recirculation.

C. E-1, Loss of Reactor or Secondary Coolant.

D. ES-1.2, Post-LOCA Cooldown and Depressurization.

Question 81

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E 2nd bullet on page 7. Specifically, the SRO must have knowledge of diagnostic steps and decision points in ECA-1.2, LOCA Outside Containment that require a transition to ECA-1.1 at the end of the procedure due the RCS break not being isolated.

K/A is met by requiring the candidate to determine that a transition to ECA-1.1, Loss Of Emergency Coolant Recirculation is required at the end of ECA-1.2, LOCA Outside Containment due to the break not being isolated. The question demonstrates integrated plant procedure use in the EOP network.

A. Incorrect. Plausible since many of the procedures in the EOP network have the crew returning to procedure and step in effect. There are also procedures that have the crew do this even if the procedure was ineffective in correcting the problem.
B. Correct. IAW ECA-1.2 step 4 RNO. SRO level since this requires a candidate to have a detailed understanding of what the required transition would be when ECA-1.2 is essentially ineffective. ROs would NOT be required to have this detailed knowledge.
C. Incorrect. Plausible since E-1 would be the appropriate entry if RCS pressure were rising. Since RCS pressure is NOT rising, ECA-1.1 is the appropriate entry procedure to enter.
D. Incorrect. Plausible since plant conditions support entry into ES-1.2 from E-1 but NOT from ECA-1.2.

Sys #	System	Category	KA Statement
WE04	LOCA Outside Containment / 3	EA2. Ability to determine and interpret the following as they apply to the (LOCA Outside Containment)	Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A# EA2.1 **K/A Importance** 4.3 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-1.2 Iss. 3 Rev. 0 Step 4 RNO

Question Source: Bank – 2LOT6 Q81

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 3SQS-53.3, Rev. 5 Obj. 6. Given a set of conditions, locate and apply the proper Emergency Operating Procedures, IAW BVPS-EOP Executive Volume.

82. Given the following conditions:
    - The Plant is at 100% power
    - I&C has removed PRZR Level Instrument, LT-1RC-461 from service IAW 1MSP-6.25.I, L-461 Pressurizer Level Channel III Test.
    - PRZR Level Channel Selector Switch selected to position Pos 1 459/460.
    - An instrument malfunction causes the following plant response:
        - A4-4, PRESSURIZER CONTROL LOW LEVEL DEVIATION
        - Letdown automatically isolated.
        - Normal charging flow control valve FCV-1CH-122 fully opened.

    Which of the following identifies the failed channel, and includes the required Technical Specification action for the plant conditions?

A. LT-1RC-459 has failed; place LT-459 in a tripped condition within 72 hours

B. LT-1RC-459 has failed; place the plant in Mode 3 within 7 hours

C. LT-1RC-460 has failed; place LT-460 in a tripped condition within 72 hours

D. LT-1RC-460 has failed; place the plant in Mode 3 within 7 hours

Question 82

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B on page 3. Specifically, the SRO must know that with przr level LT-1RC-461 being worked on by I&C under an MSP, that it is currently in a tripped condition per LCO 3.3.1 condition K. When przr level LT-1RC-459 fails low, a condition does not exist for two of the three przr level detectors being out of service, therefore LCO 3.0.3 must be enter, which requires plant shutdown to mode 3 within 7 hrs.

K/A is met with the candidate's ability to determine that the pressurizer level transmitter LT-1RC-459 failed low causing the przr level low deviation alarm, letdown to isolate, and FCV-1CH-122 to fully open due to the pressurizer control logic seeing a low level on the flow side of the pressurizer level control logic.

A. Incorrect. First part is correct. Placing LT-459 in a tripped condition within 72 hours is plausible since it is the correct action per TS 3.3.1 function 9 and condition K if only one transmitter were failed. In the case of this question, placing LT-1RC-459 in a tripped condition would result in a reactor trip which is the incorrect action.

B. Correct. Based on the conditions given, the controlling channel (LT-1RC-459 in this case) has failed low which would cause letdown to isolate and FCV-1CH-122 to fully open. LCO 3.0.3 must be entered requiring placing the plant in Mode 3 within 7 hours because TS 3.3.1 does not have a condition for two przr level instruments failed. The candidate must know that LT-1RC-461 is in a tripped condition due to I&C testing iaw the MSP which is expected SRO knowledge.

C. Incorrect. Plausible that LT-1RC-460 was the failed controlling instrument if the candidate wasn't familiar with the przr level control logic, but LT-1RC-459 controls przr level. If LT-460 were the failed instrument, placing it in a tripped condition within 72 hours is plausible because it is the correct action per TS 3.3.1 function 9 and condition K if only one transmitter were failed. In the case of this question, placing LT-1RC-459 or LT-460 in a tripped condition would result in a reactor trip which is the incorrect action.

D. Incorrect. Plausible that LT-1RC-460 was the failed controlling instrument if the candidate wasn't familiar with the przr level control logic, but LT-1RC-459 controls przr level. Second part is correct.

Sys #	System	Category	KA Statement
000028	Pressurizer (PZR) Level Control Malfunction / 2	AA2. Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions:	Cause for PZR level deviation alarm: controller malfunction or other instrumentation malfunction

K/A# AA2.12 **K/A Importance** 3.5 **Exam Level** SRO

References provided to Candidate None

Technical References:
1OM-6.4.IF Rev. 11 pg. 12
1MSP-6.25-I Iss. 4 Rev. 13 pg. 5
Tech Spec 3.3.1 pg. 3.3.1 - 4 and 14
Tech Spec 3.0.3 pg. 3.0-1

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 1SQS-6.4, Rev. 14 Obj. 21. Given a change in plant conditions due to system or component failure, analyze the Pressurizer and Pressurizer Relief System to determine what failure has occurred.
3SQS-RULES ITS, Rev. 3 Obj. 6. State the actions that are required if we do not meet an LCO (LCO 3.0.3)

83. Given the following conditions:

- The plant is operating at 100% power.
- VCT level is 43.5% and lowering
- Radiation Monitoring High-High is lit for Incore Inst Transfer Device Area Rad Monitor.
- The crew has entered AOP-1.6.7, Excessive Primary Plant Leakage
- RCS Tavg is 578F and stable
- PRZR level is 65% and stable
- VCT level dropped 1.0%/minute

At 1000, the crew performed step 9, Quantify Leakage AND Check if Leakage is in CVCS.

- Charging and Letdown were isolated
- The crew could not obtain a 10 gpm net RCS input
- Net RCS input of 20 gpm had been established
- The ATC determined that the PRZR level had risen 0.1% over 2 minutes

At 1200, Containment entry is complete, and the leak is determined to be from MOV-1CH-310, REGEN HX CHG HDR OUT ISOL valve outlet weld.

(Reference Provided)

1) Based on the above leakrate at 1000, is the Shift Manager required to evaluate EPP IAW AOP-1.6.7?

2) Based on the given conditions, what is the required **Shutdown Initiation Time (SIT)** IAW 1/2OM-48.1.I, Technical Specification Compliance?

A. 1) Yes, EPP evaluation is required.
2) Shutdown Initiation Time is 1400 hours.

B. 1) No, EPP evaluation is not required.
2) Shutdown Initiation Time is 1400 hours.

C. 1) Yes, EPP evaluation is required.
2) Shutdown Initiation Time is 1700 hours.

D. 1) No, EPP evaluation is not required.
2) Shutdown Initiation Time is 1700 hours.

Question 83

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E first paragraph on page 6. Specifically, the SRO must have specific knowledge of the content of the Excessive Primary Plant Leakage AOP versus knowledge of the procedure's overall mitigative strategy or purpose. In the case of this question, the candidate must have detailed knowledge of evaluating a RCS leakrate and make a determination that an EPP evaluation is required due to the leakage is in excess of 10 gpm. The question also requires SRO knowledge of plant administrative procedures to determine the appropriate Shutdown Initiation Time for a Tech Spec required shutdown.

K/A is met with the candidate's ability to recognize that with a rising przr level of only 0.05%/min, that a >10gpm RCS leak exists and the Excessive Primary Plant Leakage AOP requires the Shift Manager to evaluate EPP.

A. Correct. AOP-1.6.7 step 10 requires an EPP evaluation of PRZR level if RCS net input was established at 20 gpm and PRZR level is rising <0.2%/min. With PRZR level rising at <0.2%/min (question infers 0.05%/min rise), a leakrate of >10gpm unidentified leakage has been identified and EPP evaluation is required. At 1200, the leak was determined to be Pressure Boundary Leakage which requires entry into TS 3.4.13 cond. B. Condition B requires the plant to be in Mode 3 within 6 hours (1800 hrs). SIT is defined as the time beyond discovery requiring plant shutdown to commence. When establishing SIT, the candidate must ensure the time allotted to ensure a safe shutdown can be performed within the remaining Completion Time limit. IF plant at 100% power the SIT shall NOT be less than 4 hours remaining on Completion Time. Therefore, with discovery and entry into Condition B at 1200, the SIT will be 1400 hours.

B. Incorrect. Plausible that no EPP evaluation is required if the candidate does not understand how to evaluate the PRZR level, and apply the AOP steps, but the current leakrate is >10 gpm per step 10.a RNO. Shutdown Initiation Time of 1400 hours is correct as explained in correct answer.

C. Incorrect. EPP evaluation is required because the RCS leakrate is >10 gpm as explained in correct answer. Shutdown Initiation Time of 1700 hours is plausible if the candidate thinks that a maximum time to attempt to stop the leakage is required, and decides to use AOP 1.51.1, Unplanned Power Reduction, which will shut the plant down in less than 1 hour. (Plant shutdown at 2% or 5% per minute)

D. Incorrect. Plausible that no EPP evaluation is required if the candidate does not understand how to evaluate the PRZR level, and apply the AOP steps, but the current leakrate is >10 gpm per step 10.a RNO. Shutdown Initiation Time of 1700 hours is plausible if the candidate thinks that a maximum time to attempt to stop the leakage is required, and decides to use AOP 1.51.1, Unplanned Power Reduction, which will shut the plant down in less than 1 hour. (Plant shutdown at 2% or 5% per minute)

Sys #	System	Category	KA Statement
000078	RCS Leak / 3	Generic	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

K/A# 2.4.47 **K/A Importance** 4.2 **Exam Level** SRO

References provided to Candidate Tech Spec 3.4.13 pg. 3.4.13-1 with Leakage values redacted.

Technical References: 1OM-53C.4.1.6.7 Rev. 3 pg. 6, 7
1/2OM-48.1.I Rev. 36 pg. 2, 13
Tech Spec 3.4.13 pg. 3.4.13-1

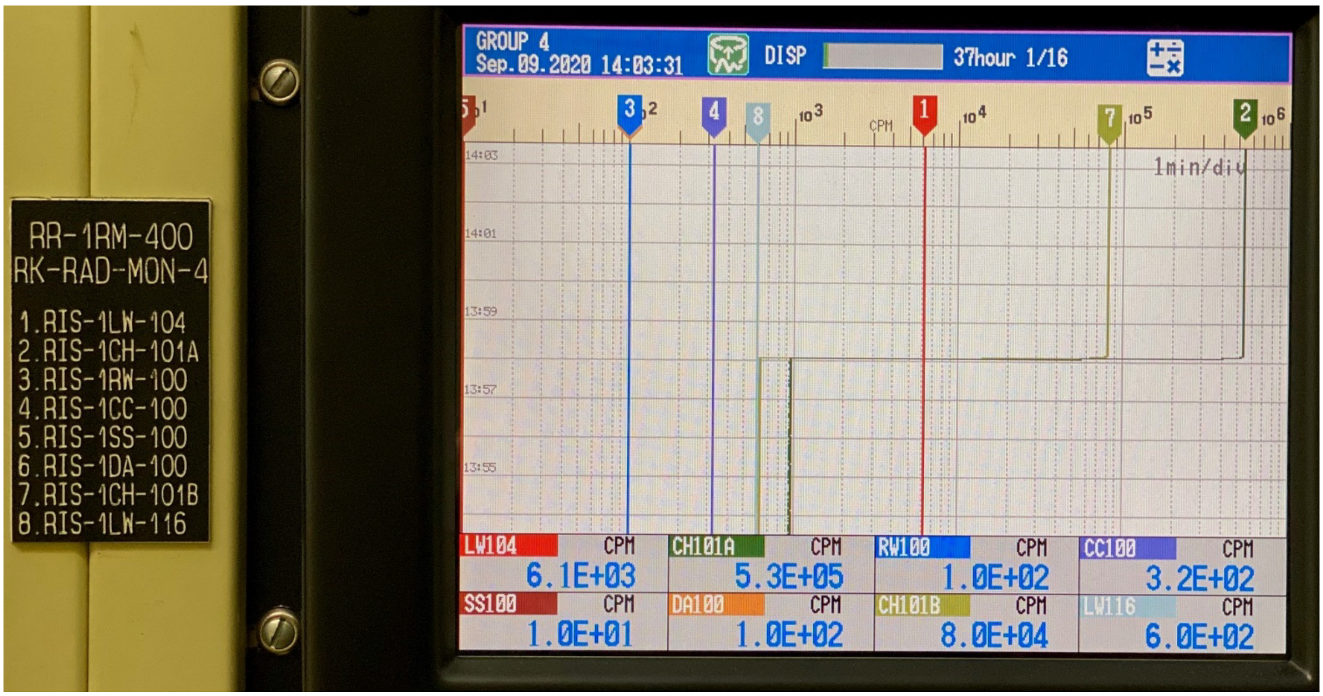
Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 1SQS-53C.1, Rev. 12 Obj. 6. Given a set of conditions, apply the correct AOP.
3SQS-48.1 Rev. 24 Obj. 24. From memory, explain the procedure for Technical Specification Compliance.

84. Given the following conditions:

- The plant was operating at 100% power when an inadvertant Rx trip occurred.
- RCS pressure is currently 2230 psig and stable.
- RCS temperature is 547°F and stable.
- Radiation Monitor Annuciators A4-71 (High) and A4-72 (High-High) are in alarm.
- The crew has entered AOP-1.6.6, High Reactor Coolant System Activity.
- Chemistry has been directed to sample the RCS for I-131 and Gross Activity.
- Below is the chart recorder for Reactor Coolant Letdown radiation monitors.



1) Based on the above indications, would an EPP declaration be required?

2) What are the accident assumptions for Tech. Spec. 3.4.16, RCS Specific Activity?

A. 1) Yes, an EPP declaration would be declared.
   2) Main Steam Line Break and a Steam Generator Tube Rupture

B. 1) No, an EPP declaration would **NOT** be required.
   2) Main Steam Line Break and a Steam Generator Tube Rupture

C. 1) Yes, an EPP declaration would be declared.
   2) Steam Generator Tube Rupture **ONLY**

D. 1) No, an EPP declaration would **NOT** be required.
   2) Steam Generator Tube Rupture **ONLY**

Question 84

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II G page 9 seventh bullet. Specifically, the SRO must recognize that the Letdown Radiation monitor levels would require an EPP declaration be made. It also meets the SRO only guidance of ES-401 Attachment 2 section II.B third bullet. Specifically, knowledge of the TS bases for the Specific Activity Safety Analysis which is an SRO responsibility.

K/A is met with the candidate's ability to interpret the chart recorder for Reactor Coolant Letdown radiation monitors and determine that radiation levels would require an EPP declaration for High Reactor Coolant Activity.

A. Correct. EPP SU4.1 states letdown rad monitor RM-1CH-101A **OR** 101B reading >5.1E+5 cpm is classified as an Unusual Event. In the picture above, RM-1CH-101A is reading 5.3E+5 cpm, therefore an Unusual Event must be declared. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

B. Incorrect. Plausible if the candidate answers no declaration is made if they look at the EPP wallboard under Abnormal Rad Levels/ Rad Effluent because the letdown radiation monitors are not identified in this area. The candidate must recognize that declaration must be made under System Malfunctions – RCS Activity. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

C. Incorrect. It is correct that a declaration will be made. Incorrect accident analysis because both the SLB and SGTR are analyzed per TS bases.

D. Incorrect. Plausible if the candidate answers no declaration is made if they look at the EPP wallboard under Abnormal Rad Levels/ Rad Effluent because the letdown radiation monitors are not identified in this area. The candidate must recognize that declaration must be made under System Malfunctions – RCS Activity. Incorrect accident analysis because both the SLB and SGTR are analyzed per TS bases.

Sys #	System	Category	KA Statement
000076	High Reactor Coolant Activity / 9	AA2. Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity:	Process effluent radiation chart recorder.

K/A# AA2.04 **K/A Importance** 3.0 **Exam Level** SRO

References provided to Candidate EPP Wallboard **Technical References:** EPP Wallboard (EPP-I-1a.F01)
Tech Spec 3.4.16 Bases pg. 3.4.16-1

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 1SQS-53C.1, Rev. 12 Obj. 2. State the conditions or symptoms that would require entry into the AOPs.
EPP-9281, Rev. 13 Obj. 11. Given specific plant conditions, classify the condition in accordance with EPP I-1a & b.

85.

Given the following conditions:

- The crew is currently in ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel (With RVLIS).
- The crew has just established the allowable cooldown rate per step 3.
- The US has directed the ‘B’ RCP to be started iaw EOP attachment 2-V.

1) What is the allowable cooldown rate per IAW ES-0.3?

2) Using Att. 2-V, and the vertical board indications above, determine if the ‘B’ RCP start criteria of step 4 have been met. **(Reference Provided)**

A. 1) Maximum cooldown rate.
   2) Criteria is met for ‘B’ RCP start.

B. 1) Maximum cooldown rate.
   2) Criteria is **not** met for ‘B’ RCP start.

C. 1) Less than 100°F/Hr.
   2) Criteria is met for ‘B’ RCP start.

D. 1) Less than 100°F/Hr.
   2) Criteria is **not** met for ‘B’ RCP start.

Question 85

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 sect. II.E pg. 6 first paragraph. Specifically, the SRO must know the required cooldown rates for the various EOP procedures. Detailed procedural knowledge is required for the 100F/hr cooldown rate of ES-0.3 Natural Circulation Cooldown with Steam Void in Vessel (With RVLIS).

K/A is met with the candidate's ability to interpret the given control room indications and determine if the RCP meets the startup requirements of the EOP attachment as directed in ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel (With RVLIS).

A. Incorrect. Plausible because maximum cooldown rate is required in E-3, Steam Generator Tube Rupture. It is one of four cooldown rates allowed within the EOP network. All criteria of Att. 2-V are met for the starting of 'B' RCP.

B. Incorrect. Plausible because maximum cooldown rate is required in E-3, Steam Generator Tube Rupture. It is one of four cooldown rates allowed within the EOP network. Second part is plausible if the candidate observes the wrong RCP parameters, such that the CCR flow is isolated to the 'A' RCP or misreads one of the other indications.

C. Correct. Cooldown rate per step 3 of ES-0.3 is limited to less than 100°F/Hr. All criteria of Att. 2-V are met for the starting of 'B' RCP.

D. Incorrect. Cooldown rate per step 3 of ES-0.3 is limited to less than 100°F/Hr. Second part is plausible if the candidate observes the wrong RCP parameters, such that the CCR flow is isolated to the 'A' RCP or misreads one of the other indications.

Sys #	System	Category	KA Statement
WE10	Natural Circulation with Steam Void/4	Generic	Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

K/A# 2.1.31 **K/A Importance** 4.3 **Exam Level** SRO

References provided to Candidate 1OM-53A.1.2-V Rev. 2 pg. 5 **Technical References:** 1OM-53A.1.ES-0.3 Iss. 3 Rev. 1 pg. 2, 4
1OM-53A.1.2-V Rev. 2 pg. 5

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 1SQS-6.3 Rev. 15 Obj. 22. 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.
3SQS-53.3, Rev. 5 Obj. 2. Describe from memory the overall purpose of each procedure, IAW BVPS-EOP Executive Volume.

86. Given the following plant conditions:

- RCS pressure is 320 psig and stable.
- RCS Cold Leg Temperature is 270°F and slowly lowering.
- BOTH "A" and "B" Residual Heat Removal (RHR) Pumps and Heat Exchangers (HX) in service.
- No Reactor Coolant Pumps (RCPs) are operating but they are OPERABLE.
- All NR Steam Generator Levels are 25% and STABLE.
- All systems are in NSA for the current mode of operation.
- The Reactor Operator reports MOV-1RH-700, "RHR Inlet Isolation Valve", has drifted to the FULL CLOSED position and will **NOT** respond.

Which of the following describes the specific impact shortly after MOV-1RH-700 fails CLOSED?

RHR flow will ________(1)________.

In accordance with Technical Specifications, ______(2)______ RCS Loops are OPERABLE.

A. 1) drop
2) zero

B. 1) remain the same
2) zero

C. 1) drop
2) three

D. 1) remain the same
2) three

Question 86

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B third bullet on page 3. Specifically, the SRO must have knowledge of the bases for TS 3.4.6, RCS Loops- mode 4, and recognize that the TS bases requires >28% SG NR level for an RCS loop to be considered operable for decay heat removal.

K/A is met with the RHR inlet valve failing closed and the candidate predicting the impact on RHR flow through the system, and the SRO making a Tech Spec operability determination of the RCS loops based on T.S. 3.4.6 bases.

A. Correct. If MOV-1RH-700 fails closed, system flow as indicated by FT-1RH-605 will drop significantly. Since no suction from the RCS loop to the RHR pump is available, the pump will continue to run and recirculate whatever water is left in the RHR loops. The candidate must determine that the plant is in Mode 4, and that in accordance with the TS 3.4.6 (RCS Loops - MODE 4) bases, an OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG, which has the minimum water level specified in SR 3.4.6.2 (>28 %). Therefore, no RCS Loops are operable due to SG NR being 25%. This question requires the SRO to make an operability determination based on system knowledge and TS bases.

B. Incorrect. Incorrect system response because with MOV-1RH-700 failing closed the RHR system is isolated from the RCS. This reflects system response if MOV-1RH-758 failed open. Also reflects an accurate configuration if there were separate RHR loops, similar to Unit 2 design. Correct that zero RCS loops are operable (refer to explanation above)

C. Incorrect. Correct that FT-1RH-605 will drop significantly. Since no suction from the RCS loop to the RHR pump is available, the pump will continue to run and recirculate whatever water is left in the RHR loops. Incorrect that three RCS loops are operable because the SG NR levels are <28% which is required for an operable RCS loop per SR 3.4.6.2. To ensure the candidate does not challenge the fact that RCPs are operable, it is added to the stem of the question. This is still highly plausible because they must possess the knowledge that RCS loops are operable based on S/G water level > 28% and RCPs need not be running

D. Incorrect. Incorrect system response. Incorrect that RCS Loops are not operable IAW TS 3.4.6. (refer to correct answer explanation.

Sys #	System	Category	KA Statement
005	Residual Heat Removal	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	RHR valve malfunction

K/A# A2.04 **K/A Importance** 2.9 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-10.1.C, Issue 4, Rev. 0, Pg. 2 & 3
Op Manual Fig. 10-1 (RM-0410-001, Rev. 16)
TS 3.4.6 Bases, Rev. 0, Pg. B 3.4.6-2 & 4

Question Source: Modified – 1LOT8 Q86

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 1SQS-10.1 Rev. 18 Obj. 19. Given a specific plant condition, predict the response of the RHR control room indications 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.

3SQS-RCS ITS, Rev. 1 Obj. 3. Given plant conditions, determine the criteria necessary to ensure compliance with each RCS LCO and Licensing Requirement in accordance with the Bases, Surveillance Requirements, and the Applicability.

87. The plant is at 100% power. The crew is required to swap CCR pumps IAW 1OM-15.4.H, Securing a CCR Pump Or Placing The Spare CCR Pump In Service, in preparation for clearing the 'A' CCR pump for preventive maintenance.

The following conditions exist:

- 'B' CCR pump is RUNNING
- 'A' CCR pump is in PTL
- 'C' CCR pump breaker is racked onto the AE bus
- 'A' CCR pump breaker is racked off AE bus

Which of the following is applicable to LCO 3.7.7, CCW System entry and why?

LCO 3.7.7 is ____________________

A. entered when 'A' CCR pump is placed to PTL because the 'A' CCR is unable to auto start if required.

B. entered when both 'A' and 'C' CCR pump breakers are racked onto AE bus because neither pump will auto start if required.

C. entered when 'A' CCR pump breaker is racked off the AE bus because 'A' CCR pump is no longer aligned to the emergency bus if the EDG would start.

D. not entered because operability of the system was not affected since either 'A' or 'C' pump could be started manually.

Question 87

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B 4 Fig. 2-1 3rd bullet on page 5. Specifically, the SRO must have knowledge of of Tech Spec bases that is required to analyze Tech Spec required actions and terminology. In this case the SRO must determine Component Cooling Water system operability based on TS bases and operating manual precautions and limitations.

K/A is met by the candidates ability to recognize that when swapping Component Cooling Water pumps, the system does not become inoperable and have the knowledge to explain why the system is not inoperable as stated in the operating manual precautions and limitations and Tech Spec Bases.

A. Incorrect. Plausible because when River Water is taken to PTL it is no longer considered operable. Candidate may get the two systems with swing pumps confused. CCW is considered operable if it can be placed in service manually.

B. Incorrect. Plausible because there are P&L's which state if two CCR pumps are racked onto the same emergency bus at any time, the 'C' pump will not auto load on diesel sequencing. Also, River Water which has a swing pump may not have two pumps aligned to same emergency bus. This would cause both pumps to be inoperable.

C. Incorrect. Plausible if the candidate thinks the CCR swing bus does not replace the 'A' CCR pump for operability requirements due to the P&L's which state if two CCR pumps are racked onto the same emergency bus at any time, the 'C' pump will not auto load on diesel sequencing.

D. Correct. Even with the 'A' CCR pump in PTL and both pumps being racked on to the AE bus, both trains of CCR are operable as stated in the OM procedure P&Ls, and the bases for LCO 3.7.7 which states a CCW train is considered OPERABLE when the pump and associated surge tank are OPERABLE and, the associated piping, valves, heat exchanger, and instrumentation and controls required to perform the required function are OPERABLE. Each CCW train is considered OPERABLE if it is operating or if it can be placed in service manually.

Sys #	System	Category	KA Statement
008	Component Cooling Water	Generic	Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

K/A# 2.2.42 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-15.4.H Rev. 16 pg. 2 P&L A & E
TS 3.7.7. pg B3.7.7.-2

Question Source: Bank - 1LOT18 Q86 (Last 2 Exams)

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 1SQS-15.1 Rev 14 Obj. 14. Summarize the Precaution and Limitations as outlined in the Operating Manual 10M-15.1 procedures.

3SQS-PLTSYS ITS, Rev. 2 Obj. 1. Apply the following definitions to ensure compliance with applicable requirements: a. OPERABLE – OPERABILITY

88. Given:

- The plant is at 100% power
- PRZR level is at program level
- PRZR Backup Heater Group D is OOS

Subsequently:

- Annunciator A4-34, Pressurizer Control Heater Power Controller Trouble alarms
- PRZR Proportional Heater Group C Breaker is tripped

Given the conditions above, LCO 3.4.9 “Pressurizer” requirements for pressurizer heaters _____(1)_____ met. This requirement ensures the pressurizer heaters maintain subcooling during a _____(2)_____ event.

A. 1) is
2) loss of offsite power (LOOP)

B. 1) is
2) small break loss of coolant accident (SBLOCA)

C. 1) is NOT
2) loss of offsite power (LOOP)

D. 1) is NOT
2) small break loss of coolant accident (SBLOCA)

Question 88

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B third bullet on page 3. Specifically, the SRO must have knowledge of the bases for TS 3.4.9, Pressurizer, and recognize that even though the UFSAR does not take credit for the pressurizer heater operation in accident analysis, LCO 3.4.9 bases requires the heaters to maintain subcooling in the long term during a loss of offsite power event.

K/A is met with the candidate's ability to determine whether given przr heater failures would require entry into the Pressurizer TS LCO and understand the bases of the przr heater requirements to mitigate a loss of power event.

A. Correct. The LCO requires two sets of OPERABLE pressurizer heaters with each set consisting of ≥ 150 kW capacity and powered from an emergency bus. Three groups of przr heaters are still OPERABLE with the capacity of each group >150 KW and capable of being powered from redundant ESF power supplies, the Group A, B, and E przr heaters are still available (215, 215, and 270 KW respectively). The przr proportional heater group C is powered from the 1B 480 bus which is a normal 480V bus. The bases states that the przr heaters provide the capability to maintain subcooling in the long-term during loss of offsite power.

B. Incorrect. First part is correct. Second part is plausible if the examinee believes that the basis for PZR heaters is to help with a SBLOCA to help maintain PZR pressure control to maintain RCS pressure while cooling down during a SBLOCA.

C. Incorrect. LCO not being met is plausible if the candidate doesn't understand the LCO requirement of two banks powered from emergency busses. Second part is correct.

D. Incorrect. LCO not being met is plausible if the candidate doesn't understand the LCO requirement of two banks powered from emergency busses. Second part is plausible if the examinee believes that the basis for PZR heaters is to help with a SBLOCA to help maintain PZR pressure control to maintain RCS pressure while cooling down during a SBLOCA.

Sys #	System	Category	KA Statement
010	Pressurizer Pressure Control	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Heater failures

K/A# A2.01 **K/A Importance** 3.6 **Exam Level** SRO

References provided to Candidate None **Technical References:** Tech Spec LCO 3.4.9 pgs. 3.4.9-1, B 3.4.9-2
1SQS-6.4 Rev. 14 pg. 45, 48

Question Source: Bank – 2014 Braidwood Q79

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 3SQS-RCS ITS Rev. 1 Obj. 2. State the purpose of each RCS specification as described in the Applicable Safety Analyses section of the Bases.

3SQS-RCS ITS Rev. 1 Obj. 3. Given plant conditions, determine the criteria necessary to ensure compliance with each RCS LCO and Licensing Requirement in accordance with the Bases, Surveillance Requirements, and the Applicability.

89. A large break LOCA has occurred from 100% power.

- The crew is currently performing step 5, “Verify SI system aligned for Recirculation” in ES-1.3, Transfer to Cold Leg Recirculation
- RWST level is 18 feet and lowering
- Cnmt pressure peaked at 34 psig, and is currently 10 psig and slowly lowering.

1) Based on the above conditions, what is the status of the Quench and Recirc Spray pumps?

2) What is the bases for securing 2 Recirc Spray pumps in ES-1.3?

A. 1) Two Quench Spray pumps and two Recirc Spray pumps are running.
2) To ensure containment pressure remains above the minimum design pressure.

B. 1) Only two Recirc Spray pumps are running.
2) To ensure containment pressure remains above the minimum design pressure.

C. 1) Two Quench Spray pumps and two Recirc Spray pumps are running.
2) To maintain NPSH for the pumps taking suction on the containment sump.

D. 1) Only two Recirc Spray pumps are running.
2) To maintain NPSH for the pumps taking suction on the containment sump.

Question 89

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E first paragraph on page 6. Specifically, the SRO must have specific knowledge of the content of ES-1.3, Transfer to Cold Leg Recirculation and the bases for securing two cnmt spray pumps prior to the transfer to recirculation.

K/A is met with the candidate's ability to interpret control indications of cnmt pressure and RWST level to determine how many cnmt spray pumps should be running prior to transfer to recirculation and why this combination is required,

A. Incorrect. First part is correct. Second part is plausible because it is the bases for securing QS pumps when <8psig in cnmt and throttling river water to the RS Hxs to maintain cnmt pressure above the design limit for minimum cnmt pressure.

B. Incorrect. Plausible if the candidate thinks that the QS pumps are secured to conserve RWST volume, but at this point of the event, cnmt pressure is >8 psig, and QS is assisting in adding water to the cnmt sump. Second part is plausible because it is the bases for securing QS pumps when <8psig in cnmt and throttling river water to the RS Hxs to maintain cnmt pressure above the design limit for minimum cnmt pressure.

C. Correct. Two QS pumps will be running lowering cnmt pressure and supplying water for the LHSI and RS pumps from the RWST. All four RS pumps started when RWST level <27' 7.5" coincident with a CIB, but ES-1.3 requires two RS pumps to be shut down prior to transfer to recirc to ensure sufficient NPSH is available in the cnmt sump for the LHSI pumps and two RS pump. Securing 2 RS pumps prior to transfer to recirc is a time critical operator action.

D. Incorrect. Plausible if the candidate thinks that the QS pumps are secured to conserve RWST volume, but at this point of the event, cnmt pressure is >8 psig, and QS is assisting in adding water to the cnmt sump. Second part is correct.

Sys #	System	Category	KA Statement
026	Containment Spray	Generic	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

K/A# 2.2.44 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-1.5.B.8 Rev 0 pg. 6
1OM-53A.1.ES-1.3 Iss. 3 Rev. 0 pg. 3
1OM-53B.4.ES-1.3 Iss. 3 Rev. 0 pg. 10

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 1SQS-13.1 Rev 15 Obj. 18. Describe the control, protection and interlock functions for the control room components associated with the Containment Depressurization System, including automatic functions, set points and changes in equipment status as applicable.

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.

90. Given the following conditions:

- The plant is at 20% during a plant startup
- The Tap Changer for ‘D’ 4KV Bus has malfunctioned causing the ‘D’ bus voltage to lower to 110 VAC
- A8-24, SYS STA SERV TRANS 1B LOAD TAP CHANGER AT END POSITION is LIT
- A8-117, 4160V EMERGENCY BUS 1DF UNDERVOLTAGE is LIT

1) What is the status of EDG1-2, **45 seconds** after the alarm comes in?

2) What is the required System Service Transformer 1B (SSST-1B) secondary voltage range required to be to meet the acceptance requirements of 1OST-36.7, Offsite to Onsite Power Distribution System Breaker Alignment Verification with the TAP changer in **MANUAL**?

A. 1) Running
2) 119-126 VAC

B. 1) Running
2) 122-126 VAC

C. 1) Standby
2) 119-126 VAC

D. 1) Standby
2) 122-126 VAC

Question 90

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E first paragraph on page 6. Specifically, the SRO must have specific knowledge of the content of the required Tech Spec OST to determine that when the System Station Service Transformer (SSST) Tap Changer is in manual, the voltage must be maintained between 122-126 to allow for grid variations.

K/A is met with the candidates knowledge of how a Tap Changer failing can cause a degraded voltage condition on the 4160VAC emergency bus undervoltage, and by understanding the undervoltage protection logic scheme determine that with the given conditions that the EDG will not have started yet. Then, knowing that the required Tech Spec OST will require the emergency bus voltage to be maintained between 122-126 VAC while the Tap Changer is in manual.

A. Incorrect. Plausible that the candidate thinks the EDG is running if they are not familiar with the emergency bus undervoltage protection and thinks that bus voltage has lowered to 75% which would cause the EDG to be running. 119-126 VAC is plausible because this is allowable bus voltage for the emergency bus, but not when the Tap changer is in manual.

B. Incorrect. Plausible that the candidate thinks the EDG is running if they are not familiar with the emergency bus undervoltage protection and thinks that bus voltage has lowered to 75% which would cause the EDG to be running. Second part is correct.

C. Incorrect. EDG1-2 will be in standby because the SSST load tap changer has lowered 'D' bus and 'DF' bus to 110 VAC (3813 VAC or 91.7%) which caused the 4160V Emergency Bus 1DF Undervoltage annunciator to alarm at 111.4 VAC. When emergency power lowers to 93.7% (112.4 VAC/3898VAC), a 90 second timer will start, after 90 seconds, emergency bus supply breakers 1A10 & 1E7 will open which after a 0.5 second time delay will start the EDG. Since the question asks the status of the EDG 45 seconds after the alarm comes in, the EDG will not be running. 119-126 VAC is plausible because this is allowable bus voltage for the emergency bus, but not when the Tap changer is in manual.

D. Correct. EDG1-2 will be in standby because the SSST load tap changer has lowered 'D' bus and 'DF' bus to 110 VAC (3813 VAC or 91.7%) which caused the 4160V Emergency Bus 1DF Undervoltage annunciator to alarm at 111.4 VAC. When emergency power lowers to 93.7% (112.4 VAC/3898VAC), a 90 second timer will start, after 90 seconds, emergency bus supply breakers 1A10 & 1E7 will open which after a 0.5 second time delay will start the EDG. Since the question asks the status of the EDG 45 seconds after the alarm comes in, the EDG will not be running. 1OST-36.7 precaution states that if bus 1D is being supplied by the SSST and the Tap changer is in Manual, the bus voltage must be within 122-126 VAC.

Sys #	System	Category	KA Statement
062	AC Electrical Distribution	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Degraded system voltages

K/A# A2.16 **K/A Importance** 2.9 **Exam Level** SRO

References provided to Candidate None

Technical References:
3SQS-36.1 U1 PPNT Rev. 12 Iss. 1 Slide 52
1OST-36.7 Rev. 25 pg. 7
1OM-36.4.ADK Rev. 5 pg. 2
1OM-36.4.AAO Rev. 1 pg. 3

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 3SQS-36.1, Rev. 12 Obj. 16. Given a specific plant condition, predict the response of the 4KV Distribution 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.

3SQS-ELEC ITS, Rev. 1 Obj. 3. Given plant conditions, determine the criteria necessary to ensure compliance with each Section Electrical Power Systems LCO and Licensing Requirement in accordance with the Bases, Surveillance Requirements, and the Applicability.

91. The following sequence of events have occurred:

- The plant was operating at 100% power
- An earthquake caused a break on the 'A' Main Steam Line
- Main Steam Isolation Valves (MSIV) would not close
- Field Operators are currently trying to close any MSIV
- The Control Room has entered ECA-2.1, "Uncontrolled Depressurization of all Steam Generators" and just completed step 3, Control Feed Flow to Minimize RCS Cooldown

Subsequently, a Field Operator reports that the 'B' Steam Generator MSIV has been closed.

Which of the following describes the expected plant response, and the actions to be taken for the above conditions?

A. 'B' SG pressure will rise, and the crew should remain in ECA-2.1 until Safety Injection is terminated.

B. 'B' SG pressure will rise, and the crew should transition to E-2, "Faulted Steam Generator Isolation".

C. 'B' SG pressure will continue to lower due to cooldown, and the crew should remain in ECA-2.1 until Safety Injection is terminated.

D. 'B' SG pressure will continue to lower due to cooldown, and the crew should transition to E-2, "Faulted Steam Generator Isolation".

Question 91

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E second bullet on page 7. Specifically, the SRO must have specific knowledge of the content of ECA-2.1, and based on that knowledge determine that a transition to E-2 is required.

K/A is met with the candidates ability to predict that steam pressure will rise on a non-faulted SG when the main steam isolation valve is closed while in ECA-2.1, "Uncontrolled Depressurization of all Steam Generators", and then determine the correct procedural transition is to E-2, "Faulted Steam Generator Isolation".

A. Incorrect. Plausible because pressure will rise, and the candidate may know that the left hand page states go to E-3 if any SG pressure is rising except while performing SI termination, in which case the candidate wouldn't understand the sequence of ECA-2.1 major action steps.
B. Correct. SG pressure will rise, and the candidate should know that step 3 is too early to begin SI termination. The foldout page instructs a transition to E-2 if pressure begins to increase in any SG unless SI termination actions are in progress.
C. Incorrect. Plausible because a large cooldown of RCS could cause an isolated SG pressure to lower; however, since AFW is throttled, this should not occur. In addition, the foldout page states that if SI termination is in progress, it should continue.
D. Incorrect. Plausible because a large cooldown of RCS could cause an isolated SG pressure to lower; however, since AFW is throttled, this should not occur. In addition, the transition to E-2 is correct.

Sys #	System	Category	KA Statement
035	Steam Generator	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Faulted or ruptured S/Gs

K/A# A2.01 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-2.1 Iss. 3 Rev. 2 pg. 1, 3, LHP

Question Source: Bank – 2014 Indian Point Q93

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

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.

92. Given the following initial conditions:

- The plant is at 55% power
- PR-1CN-103, Condenser Backpressure Recorder is reading 3 In. Hg. Abs and stable

Current conditions

- Annunciator A7-4, Condenser Vacuum Low is LIT
- PR-1CN-103, Condenser Backpressure Recorder is reading 5.5 In. Hg. Abs and rising
- Turbine Load reduction was commenced to restore condenser backpressure to normal.
- Reactor power is currently 40% and stable

Based on the above conditions, which of the following people and/or groups are required to be notified in accordance with NOP-OP-1015, Event Notifications?

(Reference Provided)

1. NRC Ops Center
2. SLT/ELT
3. NRC Resident Inspector
4. Tenaska Power Services

A. 1, 2 only

B. 2, 3 only

C. 1, 2, 4 only

D. 2, 3, 4 only

Question 92

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E third bullet on page 8. Specifically, the SRO must have specific knowledge of the administrative procedures requiring internal organization and external agencies which must be notified due to a lowering condenser vacuum with a turbine/Reactor power reduction.

K/A is met with the candidate's ability to recognize that actions have been taken as a result of lowering condenser vacuum with a turbine/Reactor power reduction, and based on the plant events determine who must be notified of the events iaw NOP-OP-1015, Plant Notifications.

A. Incorrect. Plausible because NRC Ops Center is notified for many issues which happen to the plant or the site. SLT/ELT is required to be notified per NOBP-OP-1015 due to unplanned power change, and equipment malfunctions which impact plant generation.

B. Incorrect. Both notification of SLT/ELT and the Resident Inspector are required, but due to the 15% power reduction, Tenaska Power Services must be notified also for any power change > 100MWe or MVAR.

C. Incorrect. Plausible because NRC Ops Center is notified for many issues which happen to the plant or the site. The SLT/ELT and Tenaska Power Services must be notified, as is the Resident Inspector.

D. Correct. SLT/ELT must be notified due to unplanned power change, and equipment malfunctions which impact plant generation (NOP-OP-1015 pg. 18 & NOBP-OP-1015 pg. 41). NRC Resident Inspector must be notified due to any event or condition that involves an unplanned change (> 5% RTP) in reactor power level (NOP-OP-1015 pg. 18). Tenaska Power Services must be notified for any power change >100MWe or MVAR (NOP-OP-1015 pg. 23)

Sys #	System	Category	KA Statement
055	Condenser Air Removal	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 Rev. 8
NOBP-OP-1015 Rev. 17 Att. 9 pg. 37-41

Technical References: 1OM-26.4.AAS Rev. 11 pg. 2, 3
NOP-OP-1015 Rev. 8 pg. 18, 23
NOBP-OP-1015 Rev. 17 pg. 41

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43 b(4)

Objective: 3SSG-Admin Rev. 9 Obj. 22. DESCRIBE the requirements for notifying outside agencies and plant management of significant operating events in accordance with: a. NOP-OP-1015, Event Notifications

93. The plant was operating at 20% power when the following events occur:

- A rupture has occurred on the Instrument Air header piping causing a Reactor Trip.
- 1SA-C-1A, Station Air Compressor TRIPPED.
- 1SA-C-1B, Station Air Compressor failed to start.
- 1IA-C-4, DIESEL DRIVEN AIR COMPRESSOR failed to start.
- PI-1IA-106, STA INSTR AIR HEADER PRESSURE is 20 psig and lowering.
- ES-0.1, Reactor Trip Response and AOP-1.34.1, Loss of Station Instrument Air are in progress.

Which of the following completes the statements below?

TV-1SA-105, Service Air Header Isolation valve is designed to auto close at ____(1)____ to isolate the Instrument Air header.

To control Pressurizer level, the crew is required to cycle _____(2)_____.

A. 1) 95 psig
2) the operating charging pump IAW 1OM-7.2.A, CVCS Precautions and Limitations

B. 1) 95 psig
2) MOV-1CH-289, Regen Hx/Chg Header Inlet Cnmt Isolation Valve IAW AOP-1.34.1

C. 1) 100 psig
2) the operating charging pump IAW 1OM-7.2.A, CVCS Precautions and Limitations

D. 1) 100 psig
2) MOV-1CH-289, Regen Hx/Chg Header Inlet Cnmt Isolation Valve IAW AOP-1.34.1

Question 93

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E first paragraph on page 6. Specifically, the SRO must have specific knowledge of the content of AOP-1.34.1 which requires the pressurizer level be maintained my manually cycling the charging header isolation valve due to the charging FCV failing open during a loss of air event.

K/A is met with the candidates knowledge that during a loss of air event, the station air header will automatically isolate from the instrument air header at 95 psig to attempt to minimize the severity of the event, and with a complete loss of instrument air still occurring, use the knowledge of AOP-1.34.1 to control pressurizer level.

A. Incorrect. First part is correct. Cycling the charging pump is plausible if the candidate recognizes that FCV122 has failed open and it would control przr level, but it is not procedurally permitted by AOP-1.34.1, or by the starting duty limits of OM-7 P&L's. Starting and stopping the charging pump is a contingency action directed in the Unit 1 Control room Fire AOP.

B. Correct. TV-1SA-105 automatically closes at 95 psig to isolate the station air loads on low instrument air pressure so that all of the compressed air will be supplied to the instrument air loads. Per step 15 of AOP-1.34.1, MOV-1CH-289 is cycled to maintain przr level at 22% until local manual isolation and bypass control of FCV-1CH-122 is maintaining the przr level.

C. Incorrect. 100 psig is plausible because it is the auto start pressure of the standby station air compressor. Cycling the charging pump is plausible if the candidate recognizes that FCV122 has failed open and it would control przr level, but it is not procedurally permitted by AOP-1.34.1, or by the starting duty limits of OM-7 P&L's. Starting and stopping the charging pump is a contingency action directed in the Unit 1 Control room Fire AOP.

D. Incorrect. 100 psig is plausible because it is the auto start pressure of the standby station air compressor. Second part is correct.

Sys #	System	Category	KA Statement
079	Station Air	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the SAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Cross-connection with IAS

K/A# A2.01 **K/A Importance** 3.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-34.2.B Rev. 8 pg. 2
1OM-53C.4.1.34.1 Rev. 28 pg. 8

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 1SQS-34.1, Rev. 15 Obj. 3. Describe the control, protection and interlock functions for the field components associated with the Compressed Air System, including automatic functions, setpoints and changes in equipment status as applicable.
1SQS-53C.1 Rev. 12 Obj. 5. Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.

94. Which of the following contains items that are required to be reviewed and signed for on the **Unit Supervisor Turnover Checklist**?

A. Discuss xenon and delta Flux trends resulting from load changes accomplished during the shift.
Conduct a walkdown of the control board discussing evolutions in progress and planned, unusual equipment or system status and any unusual alarms or indications

B. Discuss frequency and amount of boration or dilution during the shift.
Review all open ODMI's to ensure assumptions are still valid and trigger point actions are met.

C. Review list of Tech Spec action items in effect.
Review all Night Orders since last time signed on.

D. Discuss any temporary logs, parameter trending or temporary procedures in effect.
Review SAP Notifications generated during the previous shift for safety significance and operability.

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section III in that it is unique to the SRO position per plant procedures and the task is NOT listed in the RO task list. Unit Supervisor turnover practices are specifically an SRO responsibility.

K/A is met with the candidate's knowledge of the Unit Supervisor turnover checklist expectations which is required to be reviewed by the oncoming and off going Unit Supervisors every shift.

A. Incorrect. Plausible because discussion of xenon and delta Flux trends and conducting a walkdown of the control board discussing evolutions are RO turnover checklist items.
B. Incorrect. Plausible because discussing frequency and amount of boration or dilution is a RO turnover checklist item, but reviewing all open ODMI's is a SM turnover checklist item.
C. Correct. Reviewing the list of Tech Spec action items in effect and reviewing all Night Orders are both on the Unit Supervisor checklist.
D. Incorrect. Plausible because discussion of any temporary logs, parameter trending or temporary procedures in effect is a RO turnover checklist item, but discussion of general plant status, operations and maintenance activities in progress and planned is a SM turnover item.

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** 3.9 **Exam Level** SRO
References provided to Candidate None **Technical References:** 1/2OM-48.1.C.F02, Rev. 5 (US Checklist)
Question Source: New
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(10)
Objective: 3SQS-48.1 Rev. 24 Obj. 6. From memory, explain how the logs and the shift turnover should be completed.

95. Given the following conditions:

- A reactor startup is in progress following an inadvertent plant trip.
- 1OM-50.4.D2, Reactor Startup From Mode 3 To Mode 2 is in progress.
- Control Bank 'C' is being withdrawn.
- The allowable critical rod position on CBD is 20-210 steps (+/-1000 pcm of ECP)
- The calculated Estimated Critical Position (ECP) is Control Bank 'D' at 100 steps.
- The last two 1/M plots indicate that criticality will be achieved on Control Bank 'D' at approximately 5 steps.

The Command SRO is required to direct the RO to ________________________________.

A. manually Trip the reactor AND go to E-0, Reactor Trip Response

B. manually insert all Control Bank rods to zero steps

C. manually insert Control Bank 'C' rods to zero steps

D. suspend pulling rods and have Reactor Engineering evaluate the ECP before proceeding

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Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E first bullet on page 8. Specifically, the SRO must have specific knowledge of the content of the Reactor startup and ECP procedures to determine that all control rods must be manually inserted into the core when it is determined that the reactor will go critical outside the +/-1000 pcm band established in the ECP.

K/A is met with the candidate's ability to direct the operation the reactor control rods to shutdown the reactor during a reactor startup because it is determined that the reactor will go critical outside the allowable band established in the ECP.

A. Incorrect. Plausible because this is the action required if the reactor goes critical below the rod insertion limit.
B. Correct. All control banks must be manually inserted to zero steps because predicted criticality is < 1000 pcm below the expected rod height ban established in the ECP.
C. Incorrect. Plausible because the candidate may think that since control bank 'C' is being withdrawn at the time it is discovered that the reactor is predicted to go critical early, that only CBC must be inserted.
D. Incorrect. Plausible if the candidate thinks that since the reactor is not critical yet, that stopping rod withdrawal and having reactor engineering evaluate the ECP is possible, but the procedure requires the control rods be inserted to zero steps because it is predicted for the reactor to go critical <1000 pcm of the expected critical position.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

K/A# 2.2.1 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-50.4.D2 Rev. 3 pg. 29
1OM-50.4.F Rev. 11 pg. 16

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(6)

Objective: 3LOT-M4D1, Rev. 5 Obj. 4. Explain the lessons to be learned from SOER 88-2, Premature Criticality and the response/policy to prevent a similar occurrence from happening at BVPS.

96. Given the following plant conditions:

- The Unit is operating at 100%.
- You have just returned from a day off and are reviewing the narrative logs.
- 36 hours ago, a valve was repositioned out of NSA and selected as an OPEN item using the Short Term Configuration Change Process.

Based on the requirements of NOP-OP-1014, "Plant Status Control", does this comply with the Short Term Configuration Change Process?

A. No; a clearance should have been posted 12 hours ago.

B. No; a system status print sheet should have been issued 12 hours ago.

C. Yes; a clearance will only be necessary if restoration does not occur within the next 12 hours.

D. Yes; a system status print sheet will be necessary if restoration does not occur within the next 12 hours.

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E first paragraph on page 6. Specifically, the SRO must have specific knowledge of the content and their responsibly for plant configuration control as stated in the administrative procedure for Plant Status control.

K/A is met by the ability to determine a valve has exceeded the short term configuration change process, and identify the correct actions that should have been taken in accordance with NOP-OP-1014, Plant Status Control procedure.

A. Correct. According to NOP-OP-1014, if a component is not restored to its normal configuration within 24 hours, then a clearance is hung to provide a plant status control tracking method and documentation of the deviation from the components normal alignment. A clearance should have been posted 12 hours ago. The SRO is responsible for operating changes and configuration control in the facility.
B. Incorrect. Correct that it does not comply with the short term configuration control process. A System Status Print is required to be filled out at all times reflecting system status conditions, if the system is deemed necessary by the Ops Manager. Either way if not deemed necessary the system status print would not be required. If deemed necessary than it should have been filled out 36 hours ago.
C. Incorrect. Refer to correct answer explanation. The candidate may believe the requirement is 48 hours as opposed to 24 hours.
D. Incorrect. Refer to incorrect choice B explanation. Plausible and balanced distractor.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.

K/A# 2.2.15 **K/A Importance** 4.3 **Exam Level** SRO

References provided to Candidate None **Technical References:** NOP-OP-1014, Rev. 7 pg. 14
1/2OM-48.3.D, Rev. 8 pg. 8

Question Source: Bank – 1LOT8 Q96

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(3)

Objective: 3SQS-48.1 Rev. 24 Obj. 20. From memory, explain all of the Operations Expectations.

97. In accordance with 1/2-EPP-IP-5.3, "Emergency Exposure Criteria and Control":

1) Whose authorization is required to exceed the emergency exposure limits of 10 CFR 20 "Standards for Protection Against Radiation" to prevent the failure of equipment necessary to protect the public health and safety during an emergency?
2) What TEDE limit is this authorization limited to?

A. 1) Emergency Recovery Manager
   2) 10 Rem TEDE

B. 1) Emergency Recovery Manager
   2) 75 Rem TEDE

C. 1) Emergency Director
   2) 10 Rem TEDE

D. 1) Emergency Director
   2) 75 Rem TEDE

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II E page 21 third bullet. SRO is required to have knowledge of the Emergency Plan and position responsibilities for the Emergency Director. This is a SRO position function only

K/A met with knowledge of 10CFR20 emergency limits, and who may authorize emergency exposure limits, and what the limit is to prevent the failure of equipment necessary to protect the public health and safety.

A. Incorrect. Plausible because the Emergency Recovery Manager has many responsibilities, and will discuss information with the ED, but ONLY the ED is authorized to grant exceeding 10CFR20 emergency limits. 10 Rem is correct because it is the 10CFR20 emergency exposure limit for preventing the failure of equipment necessary to protect the public health and safety.

B. Incorrect. Plausible because the Emergency Recovery Manager has many responsibilities, and will discuss information with the ED, but ONLY the ED is authorized to grant exceeding 75 Rem is plausible because the ED may authorize up to 75 Rem to save human life, mitigate radioactive releases comparable to EPA PAGS, or restore plant equipment necessary for maintaining Critical Safety Functions.

C. Correct. The Emergency Director, with the advice of the Radiological Controls Coordinator, shall approve all emergency exposures in excess of 10 CFR 20 limits. 10 Rem is correct because it is the 10CFR20 emergency exposure limit for preventing the failure of equipment necessary to protect the public health and safety.

D. Incorrect. The Emergency Director, with the advice of the Radiological Controls Coordinator, shall approve all emergency exposures in excess of 10 CFR 20 limits. 75 Rem is plausible because the ED may authorize up to 75 Rem to save human life, mitigate radioactive releases comparable to EPA PAGS, or restore plant equipment necessary for maintaining Critical Safety Functions.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of radiation exposure limits under normal or emergency conditions.

K/A# 2.3.4 **K/A Importance** 3.7 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1/2-EPP-IP-5.3, Rev. 11 pg. 3, 4, 9

Question Source: Modified – 2LOT15 Q97

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(4)

Objective: EPP-9281, Rev. 13 Obj. 4. State the duties of the Emergency Director that cannot be delegated.

98. A 10 Rem/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: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section D 2nd bullet on page 6. Specifically, the SRO must determine if the Fuel Pool diving activity is an orange radiological risk at BV and based on the radiological conditions who must approve the radiation work permit.

K/A is met with the candidate's ability to determine that the Radiation Protection Manager must approve the radiation work permit for a Fuel Pool diving activity in a 10R/hr radiation field.

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 18; NOP-OP-4010 rev. 8

Technical References: NOP-OP-4010 rev. 8 page 10; NOP-OP-4107 rev. 18 page 26 and 29

Question Source: Bank – 2LOT17 Q98

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(7)

Objective: SSG-Admin Rev, 9 Obj. 16. DESCRIBE the controls for maintaining personnel exposures ALARA in accordance with NOP-OP-4107, Radiation Work Permit (RWP).

99. Which of the following **local field operator actions** are required to be taken due to a SGTR on 'C' SG in accordance with E-3, Steam Generator Tube Rupture, and why?
Assume all equipment operates as expected.

1. Manually isolate the 'A' OR 'B' SG Atmospheric Steam Dump valve to maintain the pressure in ONE intact SG higher than the ruptured SG.
2. Manually isolate the 'C' SG Atmospheric Steam Dump valve to maintain the pressure in the 'C' SG higher than an intact SG's following RCS cooldown.
3. Manually isolate the RHR valve from the 'C' SG to minimize radiological release and maintain the pressure in the 'C' SG higher than the intact SG's following RCS cooldown.
4. Manually isolate the TDAFW steam supply from 'C' SG to minimize a radiological release from occurring.

A. 1 and 4 only.

B. 3 and 4 only

C. 1, 3, and 4 only

D. 2, 3, and 4 only

Question 99

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section E 1st paragraph on pg. 6. Specifically, the SRO must know that E-3 locally isolates the RHR valve and the TDAFW pump steam supply from the affected SG to minimize radiological releases, and ensure the affected SG pressure is higher than the intact SGs, which ensures the primary to secondary leak stops after the RCS cooldown.

K/A is met with the SRO's knowledge of specific local operator actions to isolate steam flow from a SG with a tube rupture in E-3, and have an understanding of the operational effects that the isolation will have on stopping the primary to secondary leak, and radioactive releases.

A. Incorrect. Plausible if the candidate thinks that one intact SG should be isolated to ensure the pressure remains higher in an intact SG than in the ruptured SG, but this is incorrect. It is also wrong to think any SG Atm dump valves should be isolated if they are functional during E-3 because even the ruptured SG Atm dump valve must remain available to limit SG pressure. Isolating TDAFW steam supply from 'C' SG is correct.

B. Correct. Locally isolating the RHR valve and the TDAFW steam supply from 'C' SG is correct to minimize radiological releases and ensure the ruptured SG pressure remains higher than the intact SGs following RCS cooldown to stop primary to secondary leakage.

C. Incorrect. Plausible if the candidate thinks that one intact SG should be isolated to ensure the pressure remains higher in an intact SG than in the ruptured SG, but this is incorrect. It is also wrong to think any SG Atm dump valves should be isolated if they are functional during E-3 because even the ruptured SG Atm dump valve must remain available to limit SG pressure. Locally isolating the RHR valve and the TDAFW steam supply from 'C' SG is correct.

D. Incorrect. Plausible to isolate the ruptured 'C' SG Atmospheric Steam Dump valve, but it would not be locally isolated if it is functioning properly, which the stem states assume all equipment operates as expected. The SRO must know from the background document that the ASD valve from the ruptured SG should remain available to limit SG pressure unless it fails open. This minimizes any challenges to the code safety valves. Locally isolating the RHR valve and the TDAFW steam supply from 'C' SG is correct.

Sys #	System	Category	KA Statement
N/A	N/A	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:** 1OM-53A.1.E-3 Iss. 3 Rev. 4 pg. 5, 6
1OM-53B.4.E-3 Iss. 3 Rev. 4 pg. 59, 60

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(5)

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.

3SQS-53.3, Rev. 5 Obj. 6. Given a set of conditions, locate and apply the proper Emergency Operating Procedures, IAW BVPS-EOP Executive Volume.

100 Given the following plant conditions:

- The Emergency Director declared a Site Area Emergency at 1215.
- The initial report to state and local government was completed at 1227.
- An upgrade to General Emergency was declared at 1245.
- The Initial Protective Action Recommendation (PAR) was made without a dose projection.
- A Valid dose projection is now available that requires an upgraded PAR at 1255.

The Initial **AND** Upgraded (PAR) to the State/County Agencies **must be** given by which of the following times?

	INITIAL	UPGRADED
A.	1245	1255
B.	1300	1310
C.	1300	1345
D.	1327	1355

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II E page 7 third bullet. SRO is required to have knowledge of the reporting requirements required by the BV Emergency Protection Plan procedure. This is an SRO position function only.

K/A is met with the candidate's ability to take action to inform outside agencies of changing plant conditions within the allotted times set forth by the guidelines of the plant emergency procedures.

A. Incorrect. All distractors are plausible but incorrect as they are intervals of the given time in the question stem.
B. Correct. The Initial PAR must be declared within 15 minutes of declaration of a GE. The upgraded PAR does not change emergency classification status. Upgraded PAR determination must be completed within 15 minutes of assessment being available (ie: dose projection)
C. Incorrect. All distractors are plausible but incorrect as they are intervals of the given time in the question stem.
D. Incorrect. All distractors are plausible but incorrect as they are intervals of the given time in the question stem.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.

K/A# 2.4.38 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** ½-EPP-IP-4.1, Rev. 34, pg. 11, 13

Question Source: Bank – 1LOT8 Q100

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(5)

Objective: EPP-9281, Rev. 13 Obj. 10. Describe the time constraints related to assessment and declaration of emergency conditions.
EPP-9281, Rev. 13 Obj. 15. Describe the Emergency Director's responsibilities relative to notification of the offsite agencies.