1. After a Reactor Trip from 100% power, indicated Reactor power will lower to _____(1)_____ power in 2 to 3 seconds and then lower to a subcritical equilibrium level at a rate of _____(2)_____ decades per minute.

A. 1) ~6%
   2) -0.1

B. 1) ~10%
   2) -0.1

C. 1) ~6%
   2) -0.3

D. 1) ~10%
   2) -0.3

Answer: C

Explanation/Justification: K/A is met with the knowledge of where reactor power promptly drops to after a reactor trip, and at what rate the power will decay after the prompt drop.

A. Incorrect. First part is correct. Second part is plausible because 0.1 dpm limit is used during the transition through the point of adding heat during reactor startup.
B. Incorrect: Plausible because 10% reactor power is a basis for other reactor limits such as when certain reactor trips need to be blocked/activated. Second part is plausible because 0.1 dpm limit is used during the transition through the point of adding heat during reactor startup.
C. Correct: After a reactor trip power makes a prompt drop to about 5-7% at which time delayed neutrons slow the decrease at a calculated rate of minus 0.3 decades per minute until power is stable in the source range.
D. Incorrect: Plausible because 10% reactor power is a basis for other reactor limits such as when certain reactor trips need to be blocked/activated. Second part is correct.

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:	Decrease in reactor power following reactor trip (prompt drop and subsequent decay)

K/A# EK1.04 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** GO-GPF.R8 Rev 1 APP..A pg. 76-77
GO-3ATA 3.2 Rev. 4 pg. 2-6

Question Source: Bank – South Texas Project 2016 NRC exam Q46

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

Objective: GO-GPF.R8, Rev. 1 Obj 22. Explain reactor response to a control rod insertion.

GO-GPF.R8, Rev. 1 Obj 23. Explain the shape of the curve of reactor power versus time after a reactor trip.

GO-3ATA 3.2, Rev. 4 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: Reactor Trip from 100% power

2. Given the following conditions:

- The plant has tripped from 100% power
- RCS Tave is 538°F and stable
- Pressurizer pressure is 1737 psig and lowering
- Pressurizer level is 45% and rising
- Containment pressure is -1.2 psig and stable
- PRT pressure is 20 psig and rising

Which of the following events has occurred?

A. Steam Generator feedwater line break outside Containment.

B. Charging flow control valve, 2CHS-FCV122, failed open.

C. PRZR Relief Tank Spray valve, 2RCS-MOV516, is open.

D. A Pressurizer PORV has failed open.

Answer: D

Explanation/Justification: K/A is met by diagnosing that the przr PORV has failed open causing the indications of a vapor space accident.

A. Incorrect. a SG feedline break outside of containment would be an overcooling type of event, however RCS temperature is stable, not going down (only SI flow affecting Tavg). Additionally, Przr. level should be lowering if an overcooling were occurring. Instead, it's going up.

B. Incorrect. If the charging flow control valve failed open it would normally cause a rise in Pressurizer level. However, based on the given conditions, a Safety Injection has occurred, and the charging line has been isolated by SI and Phase 'A' Isolation.

C. Incorrect. The PRT spray valve open would raise PRT level thus causing PRT pressure to rise, but the candidate must know that the spray valve is NSA closed and no water would be entering the PRT unless the PRZR Relief Tank Pri Grade M/U Water Inlet.2RCS-AOV519 is opened also.

D. Correct. Based on RCS pressure lowering with RCS temperature stable, the basic event going on is a loss of coolant and not an overcooling. With a Pressurizer PORV open, a low-pressure area exists in the top of the Przr causing RCS water to expand into the Przr. raising Przr. Level. There is no Containment pressure response because the PORV discharges to the PRT, and the rupture disks will not rupture at 20 psig.

Sys #	System	Category	KA Statement
000008	Pressurizer Vapor Space Accident / 3	AK2. Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following:	Valves

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

References provided to Candidate None **Technical References:** 2LOT-M5D11 rev 9 scenario 1

Question Source: Bank - 1LOT18 Q2

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

Objective: GO-ATA 4.2, Rev. 8 Obj. 14 Compare and contrast a Pressurizer steam space LOCA with a small break LOCA in the cold leg. Include predicted plant response.

2LOT-M5D11, Rev. 9 Obj. 2-1 Given a Pressurizer safety failed open, respond in accordance with E-1 Loss of Reactor or Secondary Coolant

3. Given the following plant conditions:

- A small break LOCA has occurred.
- All Reactor Coolant Pumps have been stopped.
- All SG narrow range levels are approximately 50% and stable.
- Operators are responding using ES-1.2, Post LOCA Cooldown and Depressurization.

Which of the choices below completes the following statements?

1) While natural circulation is occurring _____ will be at saturation temperature for the associated steam generator pressure.

2) After RCS level decreases to the point that steam voiding has just begun to occur in the RCS hot leg, the steam generators will _____.

A. 1) Thot
   2) stop removing heat due to interruption of natural circulation

B. 1) Tcold
   2) stop removing heat due to interruption of natural circulation

C. 1) Thot
   2) continue removing heat by reflux boiling

D. 1) Tcold
   2) continue removing heat by reflux boiling

Answer: D

Explanation/Justification: The K/A is met by the candidate demonstrating knowledge of reflux boiling effects on the RCS during Natural Circulation.

A. Incorrect. Plausible because Thot is also a form of verification for natural circulation, but Thot is verified to be stable or decreasing in order to verify natural circulation and not to be at saturation temperature for the steam generator pressure. The second part is plausible because steam voiding would appear to interrupt natural circulation, but reflux boiling occurs when some of the steam produced by core is condensed in steam generator tubes and flows back down hot legs to the core, thus transferring energy from core to steam generators.

B. Incorrect. First part is correct. Second part is plausible because steam voiding would appear to interrupt natural circulation, but reflux boiling occurs when some of the steam produced by core is condensed in steam generator tubes and flows back down hot legs to core, thus transferring energy from core to steam generators.

C. Incorrect. Plausible because Thot is also a form of verification for natural circulation, but Thot is verified to be stable or decreasing in order to verify natural circulation and not to be at saturation temperature for the steam generator pressure. Second part is correct.

D. Correct. In accordance with EOP attachment A-1.7, Tcold is verified to be at saturation temperature for the pressure of the steam generator to verify natural circulation. Heat removal from the RCS is not lost due to Reflux boiling which occurs when some of the steam produced by core is condensed in steam generator tubes and flows back down hot legs to the core, thus transferring energy from core to steam generators.

Sys #	System	Category	KA Statement
000009	Small Break LOCA / 3	EK1 Knowledge of the operational implications of the following concepts as they apply to the small break LOCA:	Natural circulation and cooling, including reflux boiling

K/A# EK1.01 **K/A Importance** 4.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-53A.1.A-1.7 Iss 1C Rev 1 pg. 3
2OM-53B.5.GI-12 Iss 2 Rev 0 pg 2
GO-GPF.T8 Rev 1 Iss 1 App. A pg. 51, 52

Question Source: Bank – Summer 2011 NRC exam Q56

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

Objective: GOGPF.T8 Iss 1 Rev 1 Obj. 26 - Describe the process of reflux boiling.

4. The plant was operating at 100% power when a Large Break LOCA occurred. No other events have occurred at this time.

Which of the following annunciators is **NOT** expected for this event within the first 10 minutes?

A. A4-2F, PRESSURE RELIEF BLOCK

B. A1-3F, RECIRC SPRAY PUMP AUTO-START/AUTO-STOP

C. A6-1H, PRI COMP COOLING PUMP AUTO-START/AUTO-STOP

D. A1-3A, STEAMLINE STOP VLV NOT FULLY OPEN BYPASS VLV NOT FULLY CLOSED

Answer: B

Explanation/Justification: K/A is met by the candidate demonstrating the ability to validate alarms which are associated with a Large Break LOCA. Specifically, determine that the Recirc Spray Pumps will not be operating 10 minutes after the LOCA occurs because the pumps will not start until the RWST low level occurs coincident with a CIB.

A. Incorrect. The Pressure Relief Block annunciator is expected but plausible because the candidate must recognize that a large break LOCA will cause RCS pressure <2185 psig.

B. Correct. Recirc Spray pumps will not start until the RWST level low (381") is reached coincident with the CIB. Based on pump flowrates and RWST capacity, this will occur at a time greater than 25 minutes based on the RWST Tech Spec level requirements, and the operating pump capacities until transfer to recirc occurs. This knowledge is also acquired during simulator scenario training.

C. Incorrect. Primary comp cooling pump auto-start/auto-stop is expected but plausible because the candidate must recognize that a CIB has occurred at 11.1 psig. in cnmt which will cause the CCP pumps to auto trip.

D. Incorrect. Steamline stop vlv not fully open bypass vlv not fully closed is expected but plausible because the candidate must recognize that a MSLI has occurred at 7 psig. in cnmt which will cause the MSLI valves to close.

Sys #	System	Category	KA Statement
000011	Large Break LOCA / 3	Generic	Ability to verify that the alarms are consistent with the plant conditions.

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

References provided to Candidate None

Technical References:
2OM-13.1.D Rev. 4 Pg. 5, 6
2OM-13.4.AAH Iss 1 Rev 7 pg. 1
2OM-6.4.AAV Rev. 1 pg. 3
2OM-15.4.AAB Rev. 6
2OM-21.1.D Rev. 1 pg. 3

Question Source: New

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

Objective: 2SQS-13.1, Rev. 18 Obj. 11. Describe the control, protection and interlock functions for the control room components associated with the Containment Depressurization System, including automatic functions, setpoints and changes in equipment status as applicable.

5. 1) What is the purpose of the flywheel during an RCP trip?
   2) What prevents the RCP from rotating backwards in an idle loop?

   1) The flywheel prolongs RCP coast down time to aid in maintaining _____ during certain loss of RCS flow events.
   2) _____ prevents the RCP from rotating backwards in an idle loop.

A. 1) hot channel factors at an acceptable level
   2) Closure of the Cold leg Loop Stop Valve

B. 1) DNBR within acceptable limits
   2) An Anti-Reverse Rotation Device mounted on the flywheel

C. 1) hot channel factors at an acceptable level
   2) An Anti-Reverse Rotation Device mounted on the flywheel

D. 1) DNBR within acceptable limits
   2) Closure of the Cold leg Loop Stop Valve

Answer: B

Explanation/Justification: K/A is met by demonstrating the knowledge that the RCP flywheel is designed to maintain DNBR within limits during RCP pump trips, and its associated Anti-Reverse Rotation Device function is to prevent the RCP from rotating backwards in an idle loop.

A. Incorrect. First part is plausible because the flywheel is designed to provide inertia to aid DNBR, not specifically for hot channel factors. Hot channel factors are affected by control rods. Second part is plausible since BV has Loop Stop Valves on each Loop Thot and Tcold leg.
B. Correct. The flywheel provides additional inertia to extend coast-down time of pump in event of loss of power which limits minimum DNBR during certain loss of flow transients. Reverse coolant flow is desired, so the flywheel Anti-Reverse Rotation Device consists of a ratchet plate and pawls that operate on centrifugal force and lock the pump from rotating backwards.
C. Incorrect. First part is plausible because the flywheel is designed to provide inertia to aid DNBR, not specifically for hot channel factors. Hot channel factors are affected by control rods. Second part is correct.
D. Incorrect. First part is correct. Second part is plausible since BV has Loop Stop Valves on each Loop Thot and Tcold leg.

Sys #	System	Category	KA Statement
000015	Reactor Coolant Pump Malfunctions / 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:** 2SQS-6.3 Rev. 14 pg. 44, 45
Question Source: Modified – 1LOT18 NRC Exam Q28
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(3)
Objective: 2SQS-6.3, Rev. 14 Obj. 1. Describe the function of the Reactor Coolant Pump, major components, and the associated support systems as documented in the Operating Manual Chapter 6.

6. Given the following conditions:

- The plant is operating at 55% power.
- 'A' Charging pump has tripped on overcurrent.
- The crew is taking the appropriate immediate Operator actions per the AOP.

In accordance with the appropriate AOP IOA's, which of the following valves are required to be CLOSED, and what is the reason for this action?

2CHS-AOV200A, Letdown Orifice 21 Isolation
2CHS-AOV200B, Letdown Orifice 22 Isolation
2CHS-AOV200C, Letdown Orifice 23 Isolation
2CHS-LCV460A, Regenerative Heat Exchanger Letdown Inlet Valve
2CHS-LCV460B, Regenerative Heat Exchanger Letdown Inlet Valve

A. 2CHS-AOV200A, B, and C are closed to minimize the potential for flashing downstream of the letdown orifices.

B. 2CHS-AOV200A, B, and C are closed to prevent lifting 2CHS-RV209, LTDN Demin Relief.

C. 2CHS-LCV460A and B are closed to minimize the potential for flashing downstream of the letdown orifices.

D. 2CHS-LCV460A and B are closed to prevent lifting 2CHS-RV209, LTDN Demin Relief.

Answer: A

Explanation/Justification: K/A is met by demonstrating the knowledge of what letdown system immediate operator actions must be taken when a loss of charging occurs, and what effects this transient will have on the CVCS system.

A. Correct. The first IOA of the Loss of Charging AOP is to close 2CHS-AOV200A, B, C. This is to preclude flashing downstream of the letdown orifices as stated in the CVCS precautions and limitations. The RO is required to know the IOAs of the AOP from memory, for a loss of the running charging pump AOP 2.7.1, is the applicable procedure and requires performance of the immediate operator actions.

B. Incorrect. First part is correct. Second part is plausible if the candidate doesn't know the letdown flowpath, and that letdown pressure is reduced via the letdown orifices and NRHX Discharge Pressure Control (2CHS*PCV145). 2CHS-RV209 is set at 190 psig and is located downstream of PCV145.

C. Incorrect. Plausible because 2CHS-LCV460A, B would isolate the letdown flowpath upstream of the regen Hx but are not the valves identified in the IOAs of the Loss of Charging AOP. Second part is the correct reason for isolating letdown during a Loss of Charging event.

D. Incorrect. Plausible because 2CHS-LCV460A, B would isolate the letdown flowpath upstream of the regen Hx but are not the valves identified in the IOAs of the Loss of Charging AOP. Second part is plausible if the candidate doesn't know the letdown flowpath, and that letdown pressure is reduced via the letdown orifices and NRHX Discharge Pressure Control (2CHS*PCV145). 2CHS-RV209 is set at 190 psig and is located downstream of PCV145

Sys #	System	Category	KA Statement
000022	Loss of Reactor Coolant Makeup / 2	AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup:	Need to avoid plant transients

K/A# AK3.05 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-53C.4.2.7.1 Rev. 8 pg. 2; 2OM-7.2.A Rev. 17 pg. 3

Question Source: Bank - 2016 Harris Q5

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

Objective: 2SQS-7.1 Rev. 23 Obj. 23. Given a set of plant conditions for the Chemical and Volume Control System, describe the appropriate operating procedure(s) operational sequence, and the applicable parameter limits, precautions and limitations, and cautions & notes used to complete the task activities from the control room.

7. The plant is in Mode 5 with RHR loop 'A' in service. The following conditions exist:
   - Pressurizer level is 25% and stable
   - RCS temperature is 190°F and stable

   A 500 gpm pipe break occurs at the discharge of the 'A' RHR Pump (2RHS-P21A) and the crew enters AOP 2.10.1, Loss of Residual Heat Removal Capability.

   Which of the below listed annunciators will ALARM because of this failure?

A. A1-5H, RESIDUAL HEAT REMOVAL SYSTEM TROUBLE

B. A1-2G, INCORE INST ROOM/CNMT SUMP LEVEL HIGH/VALVE NOT RESET

C. A2-5A, PRIMARY DRAINS TRANSFER TANK/PUMP TROUBLE

D. A11-10G, SAFEGUARDS AREA SUMP LEVEL HIGH

Answer: B

Explanation/Justification: K/A is met with the candidate's knowledge that an RHR pipe break will cause the containment (reactor building) sump level to rise causing annunciator A1-2G to alarm due to high sump level.

A. Incorrect. Plausible if the candidate thinks that 'A' train RHR flow RHS-FT605A Low will alarm, but this is not the case because flow will be maintained at FT605 >1500 gpm (alarm setpoint) by FCV605 opening to maintain to auto setpoint of 3200-4000 gpm.

B. Correct. The water from this leak will spill onto the CNMT floor and make its way to this sump via CNMT floor drains. In order to answer the question, the candidate will need to have knowledge of the inputs to each of choices and be familiar enough with the names and locations of each choice presented.

C. Incorrect. Plausible because only the RHR valve stem leak-off goes into Primary Drains Transfer Tank TK21. An RHR piping leak will enter the cnmt sump.

D. Incorrect. Plausible since the CNMT sump overflows to the 'A' & 'B' recirc spray pump pits which are in Safeguards. However, these pits are not the safeguards area sump.

Sys #	System	Category	KA Statement
000025	Loss of Residual Heat Removal System / 4	AK2. Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following:	Reactor building sump

K/A# AK2.05 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-9.4.AAA Rev. 4 pg. 2; 2OM-10.1.C Iss 4 Rev 0 pg. 1

Question Source: Bank – 1LOT14 NRC Exam Q6

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(3)

Objective: 2SQS-9.1, Rev. 14 Obj. 11. Recall all the inputs into the Containment Sump.

8. The plant is at 65% power with all systems in normal alignment for this power level when the crew enters AOP 2.28.1, Loss of Secondary Component Cooling Water.

Which condition, and required action will be directed by AOP 2.28.1 due to the high Secondary Component Cooling Water temperature condition?

A. 'B' Main Feedwater Pump Bearing temperature is 205°F, trip the Main Feedwater pump.

B. Main Generator cold gas temperature is 145°F, trip the Reactor.

C. #4 Turbine Journal Bearing Metal temperature is 221°F, trip the Turbine.

D. 'A' Main Condensate Pump Bearing temperature is 175°F, trip the Condensate pump.

Answer: B

Explanation/Justification: K/A is met by the candidate monitoring CCW temperatures during a Loss of Secondary Component Cooling Water event and determining that the Main Generator cold gas temperature has exceeded its high temperature setpoint requiring a reactor trip.

A. Incorrect. Plausible distractor since AOP-2.28.1 requires the MFP to be tripped if MFP Bearing temperature is ≥220°F. The AOP has the operator look at Rx power and if >52% with the another MFP operating (which it would be at 65% power), and trip the affected MFP. AOP & 2OM-24.2.A P&L B.3.d states if bearing temperature exceeds 220F, immediately shut down the affected MFP.

B. Correct. AOP-2.28.1 requires a Rx trip if Main Generator cold gas temperature is ≥133°F with Rx power >P9 (49%). This knowledge is gained through AOP 2.28.1 or through 2OM-35.2.A P&L #22.

C. Incorrect. Plausible distractor since AOP-2.28.1 requires a Turbine trip if Turbine Journal Bearing Metal temperature is ≥225°F with Rx power <P9 (49%), but temperature is less than the trip setpoint and power is >P9.

D. Incorrect. Plausible distractor since AOP-2.28.1 requires the pump tripped if Main Condensate Pump Bearing temperature is ≥180°F.

Sys #	System	Category	KA Statement
000026	Loss of Component Cooling Water / 8	AA1. Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water:	CCW temperature indications

K/A# AA1.01 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-53C.4.2.28.1 Rev. 6 pg. 3 and LHP 2OM-35.2.A Rev. 15 pg. 3

Question Source: Bank – 1LOT16 Q35

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

Objective: 2SQS-35.2, Rev. 9 Obj. 12. 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.

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

- The plant is at 50% power with all systems in normal alignment for this power level.
- The running Main Feedwater pump trips.
- Reactor and Turbine fail to trip both automatically and manually.

Which of the following will be the AMSAC system response?

AMSAC will initiate a turbine trip and _____ after a specified time delay.

A. start all three AFW pumps when feed flow in one feedwater loop is below 25%

B. start only the MDAFW pumps when feed flow in two feedwater loops is below 40%

C. start only the TDAFW pump when feed flow all three feedwater loops is below 40%

D. start all three AFW pumps when feed flow in two feedwater loops is below 25%

Answer: D

Explanation/Justification: K/A is met with the ability to determine that during an ATWS at >40% power, with a loss of main feedwater, the AMSAC (ATWS Mitigating System Actuation Circuitry) will trip the turbine and start all three AFW pumps when feed flow in two feedwater loops reduces to <25% feed flow after a time delay to allow the RPS to respond first.

A. Incorrect. Plausible because all three AFW pumps will start on AMSAC initiation, but feed flow in two feedwater loops must be < 25%,

B. Incorrect. Plausible if the candidate thinks only the MDAFW pumps receive a start signal from AMSAC but all three pumps receive a signal when two feedwater loops are <25%, not 40% flow.

C. Incorrect. Plausible if the candidate thinks only the TDAFW pump receives a start signal from AMSAC, but all three pumps receive a signal when two feedwater loops are <25% flow, not 40% flow.

D. Correct. All three AFW pumps will start if first stage pressure is >40% (C-20) and feed flow in two feedwater loops is below 25%. The stem states that Rx power is 50% with all systems in normal alignment for this power level, therefore it is expected to have only one MFP running. When the MFP trips feed flows will lower to zero flow.

Sys #	System	Category	KA Statement
000029	Anticipated Transient Without Scram / 1	EA1 Ability to operate and monitor the following as they apply to an ATWS:	AFW system

K/A# EA1.15 **K/A Importance** 4.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-1.1.B Rev. 6 pg. 19, 20; 3SQS-1.5 U2 AMSAC PPNT Rev. 6 slide 14

Question Source: New

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

Objective: 3SQS-1.5, Rev. 6 Obj. 8. Recall all the inputs into the AMSAC circuitry, the input parameter actuation setpoint and the coincidence required for AMSAC actuation.

10. The plant is operating at 25% power when a steam line fault occurs inside containment.

Based on the above event, which of the following current conditions would **require** a manual reactor trip per the precautions and limitations of 2OM-52, General Operating Instructions?

A. 'B' Loop Tavg is 540°F.

B. SG NR water level in all SGs is 29%.

C. SG pressure in all SGs is 855 psig.

D. Containment pressure 4.8 psig.

Answer: A

Explanation/Justification: K/A is met by the candidate's ability to determine when a manual reactor trip is required based on given conditions caused by a steam line fault.

A. Correct. A manual reactor trip is required when Tavg in one or more loops is <541°F due to Tech Spec 3.4.2 RCS Minimum Temperature for Criticality, and 2OM-52 precautions and limitations #31.

B. Incorrect. Plausible with SG level near the auto reactor trip setpoint of 20.5% for low SG water level, but a reactor trip is not required, and the stem does not infer that manual reactor trip criteria has been established per the transient response guidelines. Normal SG level is 44%, and manual trip criteria setpoint is 25% and lowering. The prec

C. Incorrect. Plausible because 855 psig is slightly reduced from normal steam pressure when operating at 25% power, but it is far enough away from the SI or MSLI actuation setpoint of 500 psig so that a manual reactor trip is not required.

D. Incorrect. Plausible with cnmt pressure approaching the Safety Injection (with would actuate a reactor trip) setpoint of 5 psig, but pre-emptively tripping the reactor is not required.

Sys #	System	Category	KA Statement
000040	Steam Line Rupture—Excessive Heat Transfer / 4	AA2. Ability to determine and interpret the following as they apply to the Steam Line Rupture:	Conditions requiring a reactor trip

K/A# AA2.02 **K/A Importance** 4.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-52.2.A Rev. 25 pg. 6; Tech Spec 3.4.2

Question Source: New

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

Objective: 3SQS-RCS ITS, Rev. 1 Obj. 5. From memory, identify a condition concerning the RCS Technical Specifications and Licensing Requirements that requires Tech Spec action to be taken in one hour or less.

11. Given the following plant conditions:

- The plant was operating at 100% power.
- A complete Loss of Main Feedwater occurred.
- The crew has transitioned to FR-H.1, "Response to Loss of Secondary Heat Sink" and initiated Bleed and Feed.
- Subsequently, an AFW pump has been started and it is desired to recover Steam Generator (SG) water level.
- SG Primary side temperature is 550°F and all SG WR levels are 5%.

Which of the following will be the method of recovering SG water level **AND** the associated reason why?

SG Water Level will be recovered by initially feeding ____(1)____ to ensure SG ____(2)____.

A. 1) all three SGs at ≤ 50 gpm
2) tubes remain wetted and are fully covered before raising flowrate

B. 1) all three SGs at ≤ 100 gpm
2) thermal stress is minimized

C. 1) only one SG ≤ 50 gpm
2) tubes remain wetted and are fully covered before raising flowrate

D. 1) only one SG ≤ 100 gpm
2) thermal stress is minimized

Answer: D

Explanation/Justification: K/A is met by the candidate's ability to determine that hot / dry steam generator conditions exist and the basis for establishing feedwater in a controlled manner.

A. Incorrect. Incorrect number of SGs and flowrate. The reason and flowrate are plausible if the candidate confuses the FR-H.1 background with ECA-2.1 background which states if cooldown rate is >100F/hr, then reduce feed to <50 gpm to each SG.

B. Incorrect. Correct flowrate and reason but incorrect number of SGs.

C. Incorrect. Correct number of SGs with incorrect flowrate. The reason and flowrate are plausible if the candidate confuses the FR-H.1 background with ECA-2.1 background which states if cooldown rate is >100F/hr, then reduce feed to <50 gpm to each SG.

D. Correct. The candidate must know the background document bases for FR-H.1 information on how to restore feedwater flow to a hot dry SG and understand the operational effect if this is not performed properly. Specifically, step 29 background states that a hot dry SG is defined as having primary side of the SG temperature > 525 F and <14% WR level. The background further states that the SG water level should be restored to one SG at a time and at a minimal flowrate not to exceed 100 gpm to minimize thermal stress.

Sys #	System	Category	KA Statement
000054	Loss of Main Feedwater /4	AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW):	Effects of feedwater introduction on dry S/G

K/A# AK1.02 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-53A.1.FR-H.1, Issue 2, Rev. 1, step 29
2OM-53B.4.FR-H.1, Issue 2, Rev. 1, pg 95 & 96

Question Source: Bank – 2LOT8 Q10

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

Objective: 3SQS-53.3, Obj. 3 State from memory the basis and sequence of major Action steps IAW BVPS EOP Executive Volume.

12. Given the following conditions:

- The plant is at 100% power.
- Offsite power has been lost to System Station Service Transformer 2B (SSST 2B) due to a lightning strike.

The maximum time to complete 2OST-36.7, Offsite to Onsite Power Distribution System Breaker Alignment Verification is _____(1)_____.

2OST-36.7 checks SSST power from the _____(2)_____ system.

A. 1) 30 minutes
   2) 138 KV

B. 1) 1 hour
   2) 138 KV

C. 1) 30 minutes
   2) 345 KV

D. 1) 1 hour
   2) 345 KV

Answer: B

Explanation/Justification: K/A is met with the knowledge that TS 3.8.1 requires the Offsite to Onsite Power Distribution System Breaker Alignment Verification to be completed within 1 hour to comply with Condition A of the LCO upon a loss of an offsite power source.

A. Incorrect. 30 minutes is plausible because it is less than the required RO knowledge of 1hour Tech Specs, but the surveillance has a 1-hour action statement. 138KV is correct.
B. Correct. Tech Spec 3.8.1 Cond. A state that SR 3.8.1.1 must be performed for the required operable offsite circuit within 1 hour. SR 3.8.1.1 is met by performing 2OST-36.7, Offsite to Onsite Power Distribution System Breaker Alignment Verification which is a common OST for the Reactor Operators. SSST 2B is supplied from 138 KV Bus 1 via PCB-94.
C. Incorrect. 30 minutes is plausible because it is less than the required RO knowledge of 1hour Tech Specs, but the surveillance has a 1-hour action statement. 345KV is plausible because the 345KV receives output from the Main Unit Transformer.
D. Incorrect. 1 hour is correct. 345KV is plausible because the 345KV receives output from the Main Unit Transformer.

Sys #	System	Category	KA Statement
000056	Loss of Offsite Power / 6	Generic	Knowledge of less than or equal to one hour Technical Specification action statements for systems.

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

References provided to Candidate None **Technical References:** Tech Spec. 3.8.1 pg. 3.8.1-1 & 5; 2SQS-35.7 PPNT Rev. 7 slide 42; 2OST-36.7 Rev. 20 pg. 3, 22

Question Source: New

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

Objective: 3SQS-ELEC ITS, Rev. 1, Obj. 5. From memory, identify a condition concerning the Electrical Power Systems Technical Specifications and Licensing Requirements that requires Tech Spec action to be taken in one hour or less.

3SQS-36.1, Rev. 12, Obj. 1. Describe the function of the 4KV Distribution System and the associated major components as documented in chapter 36 of the associated Operating Manual.

13. Given the following conditions:

The plant has experienced a small break LOCA 10 minutes ago and is performing E-1, Loss of Reactor or Secondary Coolant.

The crew has just entered AOP-2.30.1 due to receiving annunciator A1-4H, SERVICE WATER SYSTEM TROUBLE. No other SWS annunciators are in alarm.

The following conditions exist:

- RCS pressure is 1520 psig and slowly LOWERING
- CNMT pressure is 1.2 psig and slowly RISING
- BOP observes the following Service Water pressures:
  - 2SWS-PI113A, Service Water Header Pressure is 41 psig and STABLE
  - 2SWS-PI113B, Service Water Header Pressure is 106 psig and STABLE

Based on the above conditions, which of the following is the location of the Service Water leak?

A. 'A' Recirculation Spray Cooler

B. 'A' Primary Component Cooling Water Hx

C. 'A' Secondary Component Cooling Water Hx

D. Standby Service Water pump header

Answer: B

Explanation/Justification: K/A is met by demonstrating the ability to determine the location of a service water leak based on given plant conditions and SWS lineup based on those conditions.

A. Incorrect. Plausible because the candidate must recognize that CIB has not occurred, therefore MOV103A/B would still be NSA closed.
B. Correct. The leak can only be on the 'A' CCP Hx because the CIA split the SWS headers and isolated the secondary portion of the system, and MOV106A/B remain NSA open because no CIB has occurred.
C. Incorrect. Plausible because the candidate must recognize that a CIA has occurred due to the SI, which automatically closes MOV107A,B,C,D to isolate the Secondary portion of the system.
D. Incorrect. Plausible distractor because the stem states no other SWS annunciators in alarm, and header pressure has been maintained >34 psig, therefore the Standby Service Water pump header to SWS system isolation valves MOV116A/B remained NSA closed.

Sys #	System	Category	KA Statement
000062	Loss of Nuclear Service Water / 4	AA2. Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water:	Location of a leak in the SWS

K/A# AA2.01 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 2SQS-30.1 PPNT Rev 23 slide 7, 8

Question Source: New

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

Objective: 2SQS-30.1, Rev. 23 Obj. 14. Describe the control, protection and interlock functions for the control room components associated with the Service Water System, including automatic functions, setpoints and changes in equipment status as applicable.

2SQS-30.1, Rev. 23 Obj. 15. Given a Service Water System configuration and without referenced material, describe the Service Water System control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable. a. Safety Injection b. Containment Isolation Phase A c. Containment Isolation Phase B

14. Initial conditions:
    - Plant is operating at 100% power.
    - 2SAS-C21A, 'A' Station Air Compressor is in service.
    - 2SAS-C21B, 'B' Station Air Compressor is on clearance for maintenance.

    A 1000 scfm air leak occurs at the outlet flange of the Instrument Air Receiver.
    Two minutes later, 'A' Station Air Compressor trips due to electrical overload.
    2IAS-PI106, Station Instrument Air Header Pressure is 88 psig and lowering.
    No operator actions have occurred.

    1) If conditions continue to degrade, which air compressor(s) would be supplying the Station Instrument air header?
    2) IAW AOP-2.34.1, Loss of Station/Cnmt Instrument Air, a manual reactor trip is required if 2IAS-PI106, Station Instrument Air pressure lowers to what pressure?

A. 1) Condensate Polishing Air Compressor **and** Diesel Driven Air Compressor.
   2) 40 psig

B. 1) Condensate Polishing Air Compressor **and** Diesel Driven Air Compressor.
   2) 65 psig.

C. 1) Diesel Driven Air Compressor **only.**
   2) 40 psig

D. 1) Diesel Driven Air Compressor **only.**
   2) 65 psig.

Question 14

Answer: D

Explanation/Justification: K/A is met with the knowledge that all air systems are supplied by the station air compressors, but during a loss of air event condensate polishing air compressor and the diesel driven air compressors will start on lowering pressure, and the systems will be isolated by the auto closure of 2SAS-AOV105. This will allow the diesel driven air compressor to maintain air loads and prevent reactor shutdown due to valves failing closed.

A. Incorrect. Plausible if the candidate recognizes that the CPAC cannot maintain the header pressure and knows that the diesel driven air comp will start at 82 psig but forgets that AOV105 will close to separate the systems at 86 psig. Second part is plausible because it is the pressure at which the MSIVs will begin to fail closed, but the AOP requires the reactor to be tripped at 65 psig due to the FRVs closing.

B. Incorrect. Plausible if the candidate recognizes that the CPAC cannot maintain the header pressure and knows that the diesel driven air comp will start at 82 psig but forgets that AOV105 will close to separate the systems at 86 psig. Second part is correct.

C. Incorrect. First part is correct. Second part is plausible because it is the pressure at which the MSIVs will begin to fail closed, but the AOP requires the reactor to be tripped at 65 psig due to the FRVs closing.

D. Correct. Condensate Polishing Air Compressor will auto start at 90 psig and attempt to maintain the instrument air header. The capacity of the CPAC is ~400 scfm, therefore, header pressure will continue to lower. At 86 psig, 2SAS-AOV105 will auto close in order to separate the service air and instrument air systems. The instrument air header will be isolated from the CPAC which will cause the instrument air header pressure to continue to lower to 82 psig which is the auto start setpoint for the Diesel Driven Air compressor. AOP 2.34.1 states that the MFRV will fail closed at 65 psig, therefore the reactor is tripped at 65 psig because SGWL will not be maintained.

Sys #	System	Category	KA Statement
000065	Loss of Instrument Air / 8	AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air:	Cross-over to backup air supplies

K/A# AK3.04 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** 2SQS-34.1 PPNT Rev 18 lss 1 slide 6
2OM-34.2.B Rev. 8 pg. 2
2OM-53C.4.2.34.1 Rev. 21 pg. 3, 11, 22

Question Source: New

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

Objective: 2SQS-34.1, Rev. 18 Obj. 13. Describe the control, protection and interlock functions for the control room components associated with the various Unit 2 Compressed Air Systems, including automatic functions, setpoints and changes in equipment status as applicable.

2SQS-34.1, Rev. 18 Obj. 15. Given a Unit 2 Compressed Air System configuration and without referenced material, describe the Compressed Air 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

15. Given the following plant conditions:

- The plant is at 100% power with all systems in NSA.
- The DLC System Operations Control Center informs the control room of possible grid instability.
- The crew enters AOP 1/2.35.1, "Degraded Grid".

Which of the following describes how voltage regulator controls are affected by undervoltage/underfrequency **AND** Generator Overexcitation conditions associated with a Degraded Grid?

1. Voltage regulator transfer to MANUAL Exciter Base Adjust _____(1)_____ when voltage reaches +/- 15 volts from setpoint.

2. To compensate for Generator Overexcitation (107% Excitation Volts/Cycle) the Main Generator Voltage Adjuster and/or Exciter Base Adjustments shall be made in the ____(2)____ direction **ONLY**.

A. 1) will AUTO occur
2) RAISE

B. 1) will **NOT** occur
2) RAISE

C. 1) will AUTO occur
2) LOWER

D. 1) will **NOT** occur
2) LOWER

Answer: C

Explanation/Justification: K/A is met by demonstrating the candidate's knowledge and ability to operate the main generator voltage regulator controls during a degraded electrical grid condition by responding to a generator overexcitation condition.

A. Incorrect. First part is correct. Second part is plausible is the candidate is not familiar with the voltage regulator operation or the associated procedures. Over-excitation adjustment made shall be in the lower direction only.

B. Incorrect. First part is plausible is the candidate is not familiar with the voltage regulator operation and recognize that Auto transfer will occur. Second part is plausible is the candidate is not familiar with the voltage regulator operation or the associated procedures. Over-excitation adjustment made shall be in the lower direction only.

C. Correct. In accordance with AOP ½ 35.1, the voltage regulator will automatically transfer to the manual exciter base adjust mode when voltage reaches +/- 15 volts from setpoint. This is system level knowledge. Additionally, Attachment 2 Caution and A7-5B ARP, states that adjustments during this condition shall be made in the lower direction only. This is a time critical item in this AOP.

D. Incorrect. First part is plausible is the candidate is not familiar with the voltage regulator operation and recognize that Auto transfer will occur. Second part is correct.

Sys #	System	Category	KA Statement
000077	Generator Voltage and Electric Grid Disturbances / 6	AA1. Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances:	Voltage regulator controls

K/A# AA1.03 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1/2OM-53C.4A.35.1, Rev. 10 pg. 2 & 18
2OM-35.4.AAT Rev 5 pg. 3

Question Source: Bank – 2LOT7 Q15

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

Objective: 2SQS-35.3 Rev. 8 Obj. 5. Given a Main Generator and Exciter System configuration and without reference material, describe the Main Generator and Exciter System control room response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable. a. Voltage Regulator Failure. b. Loss of Exciter

16. Given the following plant conditions:

- A LOCA outside containment has occurred.
- The Crew is performing the actions in ECA-1.2, LOCA Outside Containment.

In accordance with ECA-1.2, which of the following indications will be used to determine if the leak has been isolated?

A. RCS Pressure RISING.

B. RVLIS Indication RISING.

C. Safety Injection Flow LOWERING.

D. Aux Building Sump Level LOWERING.

Answer: A

Explanation/Justification: K/A is met with the ability to recognize that a leak outside of containment has been isolated by during the performance of ECA-1.2 by monitoring RCS pressure rising. This indication is the determining factor on whether to transition to E-1 for the leak being isolated, or ECA-1.1 Loss of Emergency Coolant Recirculation, if it hasn't been isolated since there will be no water in the containment sump to restore recirculation flow, and preservation of RWST.

A. Correct. In accordance with ECA-1.2, RCS pressure is used as an indication of whether leak isolation has occurred and determines whether transition to E-1 or ECA 1.1 will occur.
B. Incorrect. Other E-1 series procedures use RVLIS as an indication, but other factors would also change level.
C. Incorrect. As RCS pressure rises, ECCS flow drops, but indication is not used in ECA-1.2.
D. Incorrect. Auxiliary building sump level will more than likely be dropping because of auto pump down feature depending on leak size, however, this is not an indication used in ECA-1.2 to determine if the leak has been isolated.

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

K/A# EA2.2 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-53A.1.ECA-1.2, Issue 3, Rev. 0, pg. 3
2OM-53B.4.ECA-1.2, Issue 3, Rev. 0, pg. 1, 2

Question Source: Bank – 2LOT8 Q16

Question Cognitive Level: Lower – Memory or Fundamental **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 Executive Volume.

17. 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 37% and lowering
    - Containment pressure is -1.3 psig

    Which of the following conditions 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) 12%
   2) 340 gpm

B. 1) 12%
   2) 700 gpm

C. 1) 31%
   2) 340 gpm

D. 1) 31%
   2) 700 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, the secondary heat sink must be maintained by at least one SG NR level >12% (non-adverse), or 340 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 >340 gpm, or at least one steam generator NR level must be >12% (31% 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 <12% is 700 gpm.

C. Incorrect. First part is plausible because 31% is the SG NR level required for adverse cnmt conditions. Second part is correct.

D. Incorrect. First part is plausible because 31% 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 <12% is 700 gpm.

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:** 2OM-53A.1.E-0 Iss. 3 Rev. 0 pg. 7
2OM-53A.1.F-0.3 Iss. 3 Rev. 0

Question Source: Bank – Harris 2014 Q16

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.

18. Which of the following is a **COMPLETE** list of the reasons for the Reactor Operator to perform a controlled depressurization of the Steam Generators during the performance of ECA-1.1, Loss of Emergency Coolant Recirculation?

List of Reasons

1. Minimize RCS dilution potential in case of a subsequent tube rupture.
2. To establish conditions for injection of the Safety Injection accumulators.
3. Minimize reactor coolant flow from the LOCA.
4. To establish conditions for RHS system operation.

A. 1 and 2 ONLY

B. 3 and 4 ONLY

C. 1, 2, and 3 ONLY

D. 2, 3, and 4 ONLY

Answer: D

Explanation/Justification: The K/A asks for the reasons for RO function (action) during a Loss of Emergency Coolant Recirculation, NOT necessarily what actions are performed. This question establishes a context of ECA-1.1 performance, including an important function performed by the RO during the Loss of Emergency Coolant Recirculation, and tests knowledge of the reasons (basis) for performing these actions, with an inherent knowledge component of why they are important (minimizing break flow, assuring injection, etc.). This also addresses the K/A component of (limitations are not violated).

A. Incorrect. Plausible, since reason 2 is correct. Reason 1 (RCS dilution) is plausible, because reducing SG pressure would reduce the potential for dilution, but it is misapplied here, and is not part of the basis, or reason, for the required RO actions.

B. Incorrect. Plausible, since Reasons 3 and 4 are correct. However, the applicant has failed to also recognize that Reason #2 is also correct. By excluding this selection, the applicant has failed to recognize an important aspect of mitigation of this event (and the reason for their actions).

C. Incorrect. Plausible, since Reasons 2 and 3 are correct. However, Reason 1 is NOT part of the basis for SG depressurization during the performance of ECA-1.1.

D. Correct. Per ECA-1.1 Major Action step basis, the reasons for a controlled depressurization of the SGs are to enable cold leg accumulators injection, minimize break flow, and to establish RHR conditions.

Sys #	System	Category	KA Statement
WE11	Loss of Emergency Coolant Recirculation / 4	EK3. Knowledge of the reasons for the following responses as they apply to the (Loss of Emergency Coolant Recirculation)	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** 3.6 **Exam Level** RO

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

Question Source: Bank – 2015 Catawba Q18

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.

19. Which of the following statements describe the Spent Fuel Pool Tech Spec limit which has been established to minimize the effects of a Fuel Handling accident, and what is the bases for the limit?

A. Water level must be maintained ≥10 feet over the top of irradiated fuel assemblies to ensure dose rates offsite, and in the Control Room are within limits.

B. Boron concentration must be ≥ 300 ppm to prevent an inadvertent criticality by ensuring K_{eff} remains ≤ 0.95.

C. Water level must be maintained ≥23 feet over the top of irradiated fuel assemblies to ensure dose rates offsite, and in the Control Room are within limits.

D. Boron concentration must be ≥ 100 ppm to prevent an inadvertent criticality by ensuring K_{eff} remains ≤ 0.95.

Answer: C

Explanation/Justification: K/A is met with the ability of the candidate to recognize Tech Spec limits and explain that the limit is established to maintain the dose rates in the Control Room and offsite within the limits of 10CFR50.57 after a fuel handling accident.

A. Incorrect. Plausible because 10 feet is the lowest level that will occur in the fuel pool if any pipe breaks, thus ensuring the fuel will remain covered with enough water to keep the building accessible and the fuel adequately cooled.
B. Incorrect. Plausible because the Tech Specs state that a Boron concentration of >495 ppm will maintain Keff <0.95 when used in conjunction with the specific design of the SFP racks. The 300 ppm listed is below the limit in TS.
C. Correct. TS 3.7.15 states ≥ 23 feet over the top of irradiated fuel assemblies is required to maintain the dose rates in the Control Room and offsite within the limits of 10CFR50.57 after a fuel handling accident.
D. Incorrect. Plausible because the Tech Specs state that a Boron concentration of >495 ppm will maintain Keff <0.95 when used in conjunction with the specific design of the SFP racks. The 100 ppm listed is below the limit in TS.

Sys #	System	Category	KA Statement
000036	Fuel-Handling Incidents / 8	Generic	Ability to explain and apply system limits and precautions.

K/A# 2.1.32 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None

Technical References: Tech Spec 3.7.15 and bases pg. B.7.15-1
GO3ATA 4.3 PPNT Rev. 6 Slide 58
Tech Spec 3.7.16
Tech Spec 4.3.1.1 pg 4.0-2

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.

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.

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

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

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

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

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

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

Answer: D

Explanation/Justification: K/A met by the candidate's knowledge of the required cooldown rate to preclude or minimize a SGTR during a cooldown for a SG tube leak, and the reason that the desired cooldown flowpath is to the condensers is to minimize an offsite release.

A. Incorrect. Plausible distractor because 25F/hr cooldown is used in ES-0.2, Natural Circ Cooldown, but SG Tube Leak is less restrictive on the cooldown. Dumping steam to the atmosphere is plausible with SG Atm steam dumps and the candidates thinking all MSIVs are closed, but only the affected MSIV is closed and it is not the preferred method due to radioactive releases during a SG tube leak.

B. Incorrect. Correct cooldown. Dumping steam to the atmosphere is plausible with SG Atm steam dumps and the candidates thinking all MSIVs are closed, but only the affected MSIV is closed and it is not the preferred method due to radioactive releases during a SG tube leak.

C. Incorrect. Plausible distractor because 25F/hr cooldown is used in ES-0.2, Natural Circ Cooldown, but SG Tube Leak is less restrictive on the cooldown. Second part is correct.

D. Correct. AOP 2.6.4 step 15 & 26 both require cooldown at ≤ 90F/Hr (Plant administrative cooldown limit) using the Condenser Steam Dumps is preferred to minimize radiological releases and conserve feedwater supply.

Sys #	System	Category	KA Statement
000037	Steam Generator Tube Leak / 3	AK3. Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak:	Normal operating precautions to preclude or minimize SGTR

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

References provided to Candidate None **Technical References:** 2OM-53C.4.2.6.4 Rev 28 step 15
2OM-53B.4.E-3 Iss 3 Rev 3 pg 9, 10

Question Source: Bank - 2LOT17 NRC Exam Q22 **(Last 2 exams)**

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

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.

21. The plant is operating at 60% power when air leakage into the condenser results in lowering condenser vacuum.
    - A load reduction is initiated at a rate of 5% per minute in accordance with AOP-2.51.1, Unplanned Power Reduction.
        - Three minutes after the load reduction was commenced, condenser vacuum has lowered to 23 In. Hg-Vac and is continuing to lower.
        - Eight minutes after the load reduction was commenced, condenser vacuum has lowered to 22 In. Hg-Vac. and is continuing to slowly lower.

What operator action is required by AOP-2.26.2, Loss of Condenser Vacuum?

A. Trip the reactor and go to E-0, Reactor Trip or Safety Injection.

B. Trip the turbine and go to AOP-2.26.1, Turbine and Generator Trip.

C. Continue the load reduction and place a priming ejector (Hogger) into service.

D. Continue the load reduction and place a second set of air ejectors in service.

Answer: B

Explanation/Justification: K/A is met by giving the candidate plant conditions requiring a power reduction due to lowering vacuum and making them interpret the conditions at different points of the power reduction to determine that a turbine trip is warranted.

A. Incorrect. After 3 minutes of load reduction, the power should be at ~45% power. Plausible distractor because the loss of vacuum AOP step 2 checks power >P9, if it is and vacuum is not >24.1 In. Hg-Vac within 5 minutes, then trip the reactor. Nothing in the stem requires a reactor trip at this time because the 5 minute clock starts at the 3 minute point of the event.

B. Correct. 3 minutes after the power reduction was started, the power will be <P-9 (49%), and the 5 minute clock started at the 3 minute point when vacuum was 23 In. Hg-Vac. Therefore, tripping the turbine and entering the Turbine and Generator Trip AOP is the correct course of action since vacuum was not restored to >24.1 In. Hg-Vac by 8 minutes. The question was setup assuming that the candidates will have specific procedural knowledge that the AOP requires a turbine trip if vacuum cannot be restored to >24.1 In. Hg-Vac within 5 minutes.

C. Incorrect. The Hogger is not placed in service in the loss of vacuum AOP, but it is a plausible distractor since it is used to draw the initial vacuum.

D. Incorrect. A second set of air ejectors are placed in service in the loss of vacuum AOP attachment 1, therefore this answer is plausible, but the vacuum has not been restored to >24.1 In. Hg-Vac, therefore the turbine should be tripped.

Sys #	System	Category	KA Statement
000051	Loss of Condenser Vacuum / 4	AA2. Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum:	Conditions requiring reactor and/or turbine trip

K/A# AA2.02 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-53C.4.2.26.2 Rev 1 pg. 2

Question Source: Bank – 1LOT18 Q22

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

Objective: 2SQS-53C.1, Rev. 11 Obj. 4 Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.
2SQS-53C.1, Rev. 11 Obj. 5. Given a set of conditions, apply the correct AOP.

22. The plant is discharging the contents of the Steam Generator Blowdown Test Tank 2SGC-TK23A. The following initial conditions exist:

- 2SGC-AOV126A, SG Blowdown Test Tk 23A Recirc Isolation Valve is CLOSED.
- 2SGC-P26A, SG Blowdown Test Tank Pump 26A is RUNNING.
- 2SGC-AOV128A, Evap Test Tank Pump 26A Discharge Valve is OPEN.
- 2SGC-HCV100, Liquid Waste Eff High Rad Isolation Valve is in AUTO set at 60 GPM.

A4-5C, RADIATION MONITORING LEVEL HIGH annunciator alarms.

- The Liquid Waste Process Effluent Radiation Monitor 2SGC-RQI100 is in high alarm on the RM-11 display.

Which of the following describes the plant's response to the above conditions?

A. The discharge is terminated when the Steam Generator Blowdown Test Tank Pump, 2SGC-P26A trips upon receipt of the high radiation signal.

B. The discharge is terminated when Liquid Waste Effluent High Radiation Isolation Valve, 2SGC-HCV100 closes due the high radiation signal.

C. The discharge is diverted back to the Steam Generator Blowdown Test Tank via the Steam Generator Blowdown Test Tank Recirculation Valve 2SGC-AOV126A opening.

D. The discharge continues until the Control Room operator isolates the release by closing Evaporator Test Tank Pump Discharge Valve 2SGC-AOV128A.

Answer: B

Explanation/Justification: KA is met by the candidate demonstrating knowledge of how an Accidental Liquid Radwaste Release is prevented by the automatic features associated with the Liquid Waste Process Effluent Radiation Monitor.

A. Incorrect. Plausible distractor because a pump trip is possible with an input into the trip circuitry, but the Steam Generator Blowdown Test Tank Pump does not trip on a rad monitor input. It is also plausible because the high rad monitor alarm procedure requires 2SGC-HCV100 to be verified closed and the Test Tank Pump must be manually stopped.

B. Correct. A high radiation signal will automatically close 2SGC-HCV100 if the valve is in auto as is stated in the question stem.

C. Incorrect. Plausible distractor because by opening the Steam Generator Blowdown Test Tank Recirculation Valve the discharge would appear to go back to the test tank and minimize the accidental discharge.

D. Incorrect. Plausible because this would stop a discharge, but it is not procedurally identified in plant procedures.

Sys #	System	Category	KA Statement
000059	Accidental Liquid Radwaste Release / 9	AK2. Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following:	Radioactive-liquid monitors

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

References provided to Candidate None **Technical References:** 2OM-43.1.E Rev. 6 pg.42

Question Source: New

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

Objective: 2SQS-17.1 Rev. 9 Obj. 7 Describe the control, protection and interlock functions for the control room components associated with the Liquid Waste Disposal System, including automatic functions, setpoints and changes in equipment status as applicable.

23. Which of the following constitutes a loss of an OPERABLE containment?

A. **MODE 3**, it is discovered that Containment Purge Vacuum Break Damper will **NOT OPEN**.

B. **MODE 4**, a review of Integrated Leak Rate test results show that leakage is **NOT WITHIN LIMITS**.

C. **MODE 5**, it is discovered that the Phase 'B' isolation valve for CCP to the RCPs will **NOT CLOSE**.

D. **MODE 6**, it is discovered that one of the Emergency Airlock (EAL) doors will **NOT CLOSE**.

Answer: B

Explanation/Justification: K/A is met with the ability to determine which condition given would constitute a loss of containment integrity in various modes of operation.

A. Incorrect. Plausible because the Containment Purge Vacuum Break Damper is a passive isolation which is normally locked shut in modes 1-4 and would not normally be opened in Mode 3. It is a cnmt isolation valve but would not be a loss of OPERABLE containment if it does not OPEN since it is in its accident position.

B. Correct. IAW Technical Specification 3.6.1 and its' bases an Operable containment which equates to CNMT integrity is required in Modes 1-4. In order to maintain the containment, operable leakage must be maintained less than or equal to 1.0 L_a. The situation posed in the stem of the question would require the containment to be restored to operable status within 1 hour, and at BVPS licensed ROs are required to know from memory all TS LCO conditions that would require less than 1-hour actions.

C. Incorrect. Plausible because this is a cnmt isolation valve, but an OPERABLE Containment TS 3.6.1 or Containment isolation Valves TS 3.6.3 are not required in Mode 5.

D. Incorrect. Plausible because one could think that the EAL door not closing would constitute an inoperable containment, but in Mode 6 TS 3.9.3 states one door in each air lock must be closed, and TS 3.6.2 for Containment Airlock doors is not applicable in mode 6.

Sys #	System	Category	KA Statement
000069	Loss of Containment Integrity / 5	AA2. Ability to determine and interpret the following as they apply to the Loss of Containment Integrity:	Loss of containment integrity

K/A# AA2.01 **K/A Importance** 3.7 **Exam Level** RO

References provided to Candidate None **Technical References:** TS 3.6.1

Question Source: Bank – 2LOT6 Q22

Question Cognitive Level: Higher – Comprehension or Analysis **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. 5. From memory, identify a condition concerning the Containment Systems Technical Specifications and Licensing Requirements that requires Tech Spec action to be taken in one hour or less.

24. Plant was at 100% power when the Rx Tripped due to a Faulted SG.
    - The Steam Generator has blown dry
    - The crew has transitioned to ES-1.1, SI TERMINATION
    - The crew has just secured "B" Charging Pump
    - "A" Charging Pump is running
    - RCS pressure is rising
    - PRZR level is rising
    - RWST level is lowering

    Which of the following describes the actions that will be taken next per ES-1.1, SI TERMINATION, and why should a reduction in SI flow be done expeditiously?

A. 1) Establish letdown flow.
   2) To preserve RWST inventory.

B. 1) Establish normal charging flow.
   2) To preserve RWST inventory.

C. 1) Establish letdown flow.
   2) To prevent the pressurizer from going solid.

D. 1) Establish normal charging flow.
   2) To prevent the pressurizer from going solid.

Question 24

Answer: D

Explanation/Justification: K/A is met by the candidate assessing the CR indications, and with knowledge of the SI termination procedure, determine that normal charging must be established prior to letdown. The candidate must also demonstrate knowledge why SI should be terminated quickly as to not take the PRZR solid.

A. Incorrect. Establishing letdown flow is plausible if candidate thinks that securing one charging pump (HHSI) pump in SI term has returned to a normal charging flowpath and restoring L/D would be the next logical step to perform. Securing SI expeditiously is not based on RWST depletion, but it is plausible with RWST level lowering.

B. Incorrect. Establish normal charging flow is correct. Securing SI expeditiously is not based on RWST depletion, but it is plausible with RWST level lowering.

C. Incorrect. Establishing letdown flow is plausible if candidate thinks that securing one charging pump (HHSI) pump in SI term has returned to a normal charging flowpath and restoring L/D would be the next logical step to perform. Preventing the pressurizer from going solid is the correct reason for securing SI expeditiously.

D. Correct. After isolating HHSI flow, establishing normal charging flow is correct. Restoring L/D is identified later in ES-1.1, but it would be incorrect to establish L/D without having charging flow restored to prevent flashing downstream of the L/D orifices. Preventing the pressurizer from going solid is the correct reason for securing SI expeditiously.

Sys #	System	Category	KA Statement
WE02	SI Termination / 3	EK3. Knowledge of the reasons for the following responses as they apply to the (SI Termination)	Normal, abnormal and emergency operating procedures associated with (SI Termination).

K/A# EK3.2 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-53A.1.ES-1.1 Iss. 3 rev. 0 pg. 13

Question Source: Bank – 2LOT15 Q24 (**Last 2 exams**)

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

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

25. Given the following plant conditions:

- The plant was operating at 100% power
- The plant has experienced a Small Break LOCA
- An automatic Reactor Trip and SI have occurred
- The crew has transitioned to ES-1.2, Post LOCA Cooldown and Depressurization

With the above conditions, which of the following describes how voiding can be identified in the RCS, in accordance with ES-1.2?

A. Rapidly increasing Pressurizer level

B. Decreasing safety injection flow

C. Increasing RCS pressure

D. Rapidly decreasing RCS subcooling

Answer: A

Explanation/Justification: The K/A is met since it requires the candidate to demonstrate the ability to monitor the operating behavior characteristics on the plant during Post LOCA Cooldown and Depressurization when a void occurs in the upper head region of the Rx vessel resulting in pressurizer level rapidly rising.

A. Correct. A rapid rise in Pressurize level is indication that voiding of the vessel head region has occurred as the rapid expansion of the bulk fluid from liquid to steam forces the fluid into the surge volume of the Pressurizer.
B. Incorrect. Plausible because the RCS pressure is lowering, and SI flow is expected to increase during depressurization of the RCS but lowering SI flow would indicate saturation conditions exist which would promote void growth.
C. Incorrect. Plausible because the RCS pressure is expected to be reduced during depressurization of the RCS, but increasing pressure would indicate saturation conditions exist to promote void growth.
D. Incorrect. Plausible because subcooling is expected to be reduced during depressurization of the RCS and would indicate saturation conditions exist to promote void growth.

Sys #	System	Category	KA Statement
WE03	LOCA Cooldown—Depressurization / 4	EA1. Ability to operate and / or monitor the following as they apply to the (LOCA Cooldown and Depressurization)	Operating behavior characteristics of the facility.

K/A# EA1.2 **K/A Importance** 3.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-53B.4.ES-1.2 Iss 3 Rev 0 pg.11

Question Source: Bank – 2012 Harris NRC Exam Q25

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

Objective: 3SQS-53.3, Rev. 5 Obj 4. Explain from memory the basis for ALL Cautions and Notes, IAW BVPS-EOP Executive Volume.
GO-ATA 4.2, Rev. 8 Obj. 4. Explain the significance of saturated RCS conditions during a small break LOCA.

26. Which of the following choices are the Major Action Categories, and sequential order which they are to be performed when implementing FR-C.2, Response to Degraded Core Cooling?

A. Establish Safety Injection flow to the RCS, then initiate a rapid depressurization of the SGs to depressurize the RCS.

B. Establish Safety Injection flow to the RCS, then initiate a controlled SG depressurization to cooldown and depressurize the RCS.

C. Initiate a controlled SG depressurization to cooldown and depressurize the RCS, then establish Safety Injection flow to the RCS.

D. Initiate a rapid depressurization of the SGs to depressurize the RCS, then establish Safety Injection flow to the RCS.

Answer: B

Explanation/Justification: K/A is met by the candidate's knowledge of the operational implications of performing the major action steps of FR-C.2, Degraded core cooling in the correct order.

A. Incorrect. First part is correct. Second part is plausible as a rapid depressurization of the SGs is a major action step of FR-C.1, but this method is not desired for FR-C.2 to prevent pressurized thermal shock.

B. Correct. Major action steps in FR-C.2 are sequentially established to improve core cooling and raise RCS inventory prior to continuing in the procedure. An attempt at establishing SI flow is first, if core cooling or inventory are reestablished, then controlled SG depressurization to cooldown and depressurize the RCS is not required. If some form of high pressure injection cannot be established or is ineffective in restoring core cooling, then the operator must take actions to reduce the RCS pressure in order for the SI accumulators and low head SI pumps to inject. A controlled secondary depressurization is an effective method for achieving this, while at the same time avoiding a rapid RCS cooldown that could cause problems with pressurized thermal shock.

C. Incorrect. Incorrect order of major action steps. See correct answer explanation.

D. Incorrect. Incorrect order and major action step. See correct answer explanation.

Sys #	System	Category	KA Statement
WE06	Degraded Core Cooling / 4	EK1. Knowledge of the operational implications of the following concepts as they apply to the (Degraded Core Cooling)	Normal, abnormal and emergency operating procedures associated with (Degraded Core Cooling).

K/A# EK1.2 **K/A Importance** 3.5 **Exam Level** RO

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

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **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 attached PCS 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.

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, and by knowing the bases for this limit they will be demonstrating operating behavior characteristics of the plant instrumentation.

A. Incorrect. Plausible because two radiation monitors in cnmt are reading 1E+5 mr/hr, but this is incorrect because 2RMR-RQ206 or 207 are not > 1E+5 R/HR. Second part is correct.

B. Incorrect. Plausible because two radiation monitors in cnmt are reading 1E+5 mr/hr, but this is incorrect because 2RMR-RQ206 or 207 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 2RMR-RQ206 or 207 are ≥1E+5 R/HR, therefore with RQ206 reading 8.9E+4 and RQ207 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	EA1. Ability to operate and / or monitor the following as they apply to the (High Containment Radiation)	Operating behavior characteristics of the facility.

K/A# EA1.2 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate Radiation Levels Picture

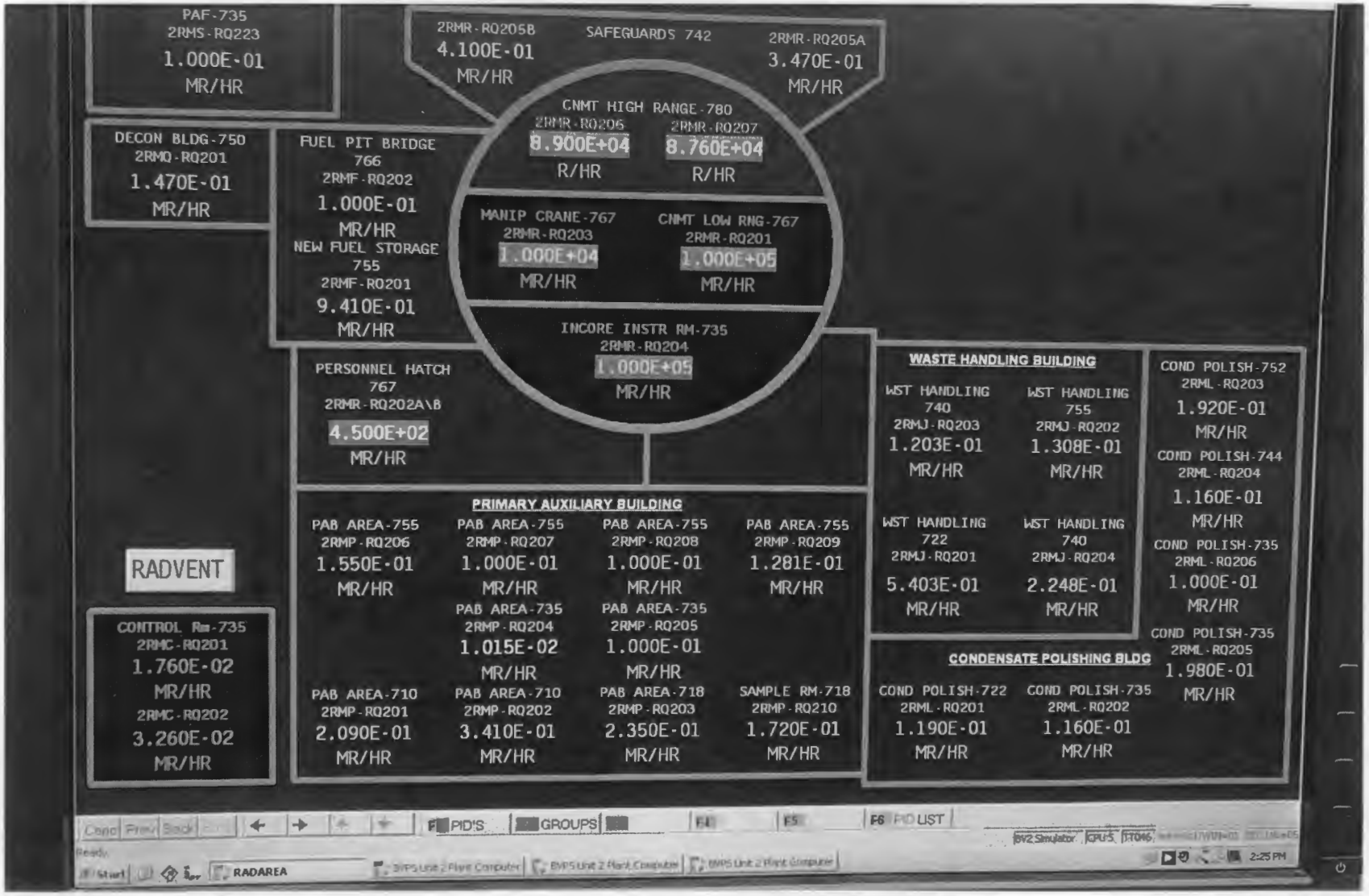
Technical References: 2OM-53A.1.E-0 Iss 3 Rev 0 LHP
2OM-53B.5.GI-2 Iss 2 Rev 0 pg. 12-13
3SQS-53.2 HO Unit 2 53-2 Rev. 2 Issue 2 pg. 12

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **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.

Question 27


PAF-735
2RMS-RQ223
1.000E-01
MR/HR
2RMR-RQ205B
4.100E-01
MR/HR
SAFEGUARDS 742
2RMR-RQ205A
3.470E-01
MR/HR
CNMT HIGH RANGE-780
2RMR-RQ206
8.900E+04
R/HR
2RMR-RQ207
8.760E+04
R/HR
DECON BLDG-750
2RMQ-RQ201
1.470E-01
MR/HR
FUEL PIT BRIDGE
766
2RMF-RQ202
1.000E-01
MR/HR
NEW FUEL STORAGE
755
2RMF-RQ201
9.410E-01
MR/HR
MANIP CRANE-767
2RMR-RQ203
1.000E+04
MR/HR
CNMT LOW RNG-767
2RMR-RQ201
1.000E+05
MR/HR
INCORE INSTR RM-735
2RMR-RQ204
1.000E+05
MR/HR
PERSONNEL HATCH
767
2RMR-RQ202A\B
4.500E+02
MR/HR
WASTE HANDLING BUILDING
WST HANDLING
740
2RMJ-RQ203
1.203E-01
MR/HR
WST HANDLING
755
2RMJ-RQ202
1.308E-01
MR/HR
WST HANDLING
722
2RMJ-RQ201
5.403E-01
MR/HR
WST HANDLING
740
2RMJ-RQ204
2.248E-01
MR/HR
COND POLISH-752
2RML-RQ203
1.920E-01
MR/HR
COND POLISH-744
2RML-RQ204
1.160E-01
MR/HR
COND POLISH-735
2RML-RQ206
1.000E-01
MR/HR
COND POLISH-735
2RML-RQ205
1.980E-01
MR/HR
PRIMARY AUXILIARY BUILDING
PAB AREA-755
2RMP-RQ206
1.550E-01
MR/HR
PAB AREA-755
2RMP-RQ207
1.000E-01
MR/HR
PAB AREA-755
2RMP-RQ208
1.000E-01
MR/HR
PAB AREA-755
2RMP-RQ209
1.281E-01
MR/HR
PAB AREA-735
2RMP-RQ204
1.015E-02
MR/HR
PAB AREA-735
2RMP-RQ205
1.000E-01
MR/HR
PAB AREA-710
2RMP-RQ201
2.090E-01
MR/HR
PAB AREA-710
2RMP-RQ202
3.410E-01
MR/HR
PAB AREA-718
2RMP-RQ203
2.350E-01
MR/HR
SAMPLE RM-718
2RMP-RQ210
1.720E-01
MR/HR
RADVENT
CONTROL Rm-735
2RMC-RQ201
1.760E-02
MR/HR
2RMC-RQ202
3.260E-02
MR/HR
CONDENSATE POLISHING BLDG
COND POLISH-722
2RML-RQ201
1.190E-01
MR/HR
COND POLISH-735
2RML-RQ202
1.160E-01
MR/HR
PID'S
GROUPS
PID LIST
RADAREA

28. The plant was operating at 100% power.

- 2CCP-P21A CS is in AFTER-START
- 2CCP-P21B is on clearance for maintenance
- 2CCP-P21C is racked on to DF bus with the CS in AFTER-STOP
- An inadvertent Reactor Trip has occurred
- During the transfer to Off-site power, annunciator A7-7F, SYSTEM STA SERV TRNSFORMER 2B DIFFERENTIAL/OVERCURRENT was received for a differential protection alarm

How many CCP pumps and RCPs will be operating **five minutes** after the Reactor Trip?

_____(1)_____ Primary Component Cooling Water pump(s) will be running.

_____(2)_____ Reactor Coolant pumps will be running.

A. 1) One
   2) No

B. 1) One
   2) Two

C. 1) Two
   2) No

D. 1) Two
   2) Two

Question 28

Answer: D

Explanation/Justification: K/A is met by placing the candidate in a changing plant lineup and demonstrating knowledge of which RCP and CCW pumps will be energized and running after a loss of one offsite power source.

A. Incorrect. Both parts are plausible if the candidate assumes that SSST 2B supplies 'A' & "B' Normal 4KV busses. Therefore the 'A' CCP pump losses power, and they recognize that the 'C' CCP pump will auto start. Second part is plausible because 2 RCPs would trip on undervoltage/under frequency if bus A & B lost power, causing the third RCP trip also due to 2/3 RCP under frequency trip.

B. Incorrect. First part plausible if the candidate recognizes power will be lost to one of the CCP pumps and doesn't recognize that the EDG sequencer will restart the lost CCP pump. Second part is correct, two RCPs will be running.

C. Incorrect. Two CCP pumps will be running. Second part is plausible because 2 RCPs would trip on undervoltage/under frequency if bus A & B lost power, causing the third RCP trip also due to 2/3 RCP under frequency trip.

D. Correct. SSST 2B supplies the C & D Normal 4KV busses. This will cause 'C' RCP to trip on undervoltage due to the loss of 'C' bus, leaving 'A' & 'B' RCPs running. It will also cause #2 EDG to auto start due to undervoltage on 'DF' bus. When the #2 EDG sequences, the 'C' CCP pump will auto start, and 2 CCP pumps will be running. The candidate must know that a SSST 2B differential protection alarm will cause the 'C' & 'D' busses to de-energize due to an internal transformer fault.

Sys #	System	Category	KA Statement
003	Reactor Coolant Pump	K2 Knowledge of bus power supplies to the following:	CCW pumps

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

References provided to Candidate None **Technical References:** 2OM-36.4.AAN Iss. 1 Rev. 2 pg 1
2OM-15.1.D Iss. 4 Rev. 1 pg. 5
2SQS-6.3 LP Rev 14 pg. 63
3SQS36.1 PPNT Unit 2 Rev 12 Iss. 1 slide 10

Question Source: Bank – 1LOT16 Q28

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

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.

2SQS-15.1, Rev. 9 Obj. 3 - Describe the electrical configuration of the CCP pumps and the special considerations that are required if that configuration is changed.

29. Which of the following completes the statements below?

During full power operation the Charging Pump Minimum Flow Isolation Valves, 2CHS*MOV275A, B, C will be _____(1)_____ to provide low flow protection to the pumps.

Flow from the Charging Pump Minimum Flow discharge header returns to the _____(2)_____.

A. 1) full open
   2) Charging pump suction

B. 1) full open
   2) top (inlet) of the VCT

C. 1) throttled
   2) Charging pump suction

D. 1) throttled
   2) top (inlet) of the VCT

Answer: A

Explanation/Justification:

A. Correct. The Charging Pump Minimum Flow valves are NSA de-energized open to eliminate the possibility of fire induced spurious operation which would close the valve. Second part is correct as the min flow discharges to the seal water heat exchanger and back to the charging pump suction.

B. Incorrect. First part is correct. Second part is plausible because other lines such as dilutions, letdown, and H2 discharge to the top of the VCT.

C. Incorrect. First part is plausible because several other system minimum flow valves throttle based on system flow. Second part is correct.

D. Incorrect. First part is plausible because several other system minimum flow valves throttle based on system flow. Second part is plausible because other lines such as dilutions, letdown, and H2 discharge to the top of the VCT.

Sys #	System	Category	KA Statement
004	Chemical and Volume Control	K4 Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following:	Interrelationships and design basis, including fluid flow splits in branching networks (e.g., charging and seal injection flow)

K/A# K4.05 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** 2SQS-7.1 PPNT Rev 23 slides 6 & 110
2OM-7.1.C Rev 9 pg. 30
2OM-7.3.B.1 Rev 34 pg. 18

Question Source: Bank – Harris 2014 Q30

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

Objective: 2SQS-7.1, Rev. 23 Obj.16. Draw and label the Chemical and Volume Control System Normal-System-Arrangement system flow path as it applies to a Licensed Operator and as illustrated on simplified one-line diagrams for, RCP Seal Injection, Excess Letdown and, CVCS Blender.

2SQS-7.1, Rev. 23 Obj. 17. List the nominal value of the control room operating parameters associated with the Chemical and Volume Control System.

30. The plant is performing a Startup from Cold Shutdown. The following conditions exist:

- The RCS is solid IAW 2OM-50.4.L, RCS Startup.
- 2CHS-HCV142, RHR Hx Outlet Flow controller, setting is at 80% demand.
- RCS pressure is 300 psig and STABLE.
- RCS temperature is 175°F and STABLE.
- Letdown is aligned to RHR.

The air supply to the valve operator for 2CHS-HCV142, RHS Letdown Flow Control valve has been severed.

1) With no operator action, how will this failure affect the RCS pressure?

2) IAW 2OM-50.4.L, what actions must the crew take to respond to this failure?

	___(1)___	___(2)___
A.	rise	stop the running charging pump
B.	rise	place all letdown orifices in service
C.	lower	raise charging flow
D.	lower	isolate normal letdown

Answer: A

Explanation/Justification: K/A is met by demonstrating the candidates ability to identify the affect that a failure of the RHR letdown valve will have on the RCS pressure during solid plant operations, and take the appropriate actions of 2OM-50.4.L to minimize the effect the high pressure transient will have on the plant.

A. Correct. 2CHS-HCV142 fails closed on the loss of air, therefore RCS pressure will rise with no operator action due to reduced letdown flow and increased net charging. Stopping the charging pump is the correct action for a sudden loss of letdown when the plant is solid iaw, 2OM-50.4.L P&L #4 and attachment 27.

B. Incorrect. First part is correct. Second part is plausible if they do not understand that all letdown orifices are in service during solid plant operations, and that due to low DP, letdown flow through normal letdown is greatly reduced (2OM-50.4.L P&L #2).

C. Incorrect. First part is plausible if they think the failure position of 2CHS-HCV142 is open, since this would lower RCS pressure when the plant is solid. Second part is the correct action if RCS pressure was low during solid plant ops because this would raise RCS pressure.

D. Incorrect. First part is plausible if they think the failure position of 2CHS-HCV142 is open, since this would lower RCS pressure when the plant is solid. Second part is plausible because isolating letdown would raise RCS pressure if it were low, and a charging pump was still running.

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:	Pressure transient protection during cold shutdown

K/A# A2.02 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-53C.4.2.34.1 rev. 21 pg. 30; 2OM-50.4.L Rev. 21 pgs. 150, 159-162

Question Source: New

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

Objective: 2SQS-10.1, Rev. 20 Obj. 7. Given a Residual Heat Removal System configuration and without reference material, describe the Residual Heat Removal System field response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable. a. Loss of instrument air

3LOT-M4D7&8&9, Rev. 10 Scenario #3 Obj. 3-8 Given Solid Plant Operation, adjust RCS Pressure in accordance with the current operations procedure.

31. Given the following:

- Reactor is at 100% and stable
- RO taking L5 logs notes the following for the 'B' accumulator:
  - Pressure is 485 psig
  - Level is 68%
  - Outlet valve is fully OPEN with lockout jack removed

If a LBLOCA were to occur now on the 'A' loop cold leg what would be the effect (if any) on the RCS?

A. None, the accumulator would inject as design.

B. Reactor could restart due to inadequate boron injection from the 'B' accumulator.

C. N2 injection into the RCS could impede natural circulation flow for the 'B' loop.

D. Fuel clad could overheat due to inadequate water injection from the 'B' accumulator.

Answer: D

Explanation/Justification: K/A is met with the knowledge that an SI accumulator (borated water source) is not within Tech Spec tolerance and will not perform the ECCS function of the accumulator as designed.

A. Incorrect – No effect is plausible if the candidate assumes accumulator pressure is normal, in which case 'B' and 'C' accumulator would inject as designed.

B. Incorrect - Reactor restart is plausible since with the lower N2 pressure not enough water from the 'B' accumulator would inject which has the boron in it and with less boron the reactor could restart. This is not the case because 'C' accumulator would inject and HHSI and LHSI would also inject.

C. Incorrect - N2 injection is plausible if the student confuses the pressure as too high and if that were true then the N2 would inject into the loops and end up in the SG u-tubes and stop Natural Circulation flow.

D. Correct – The candidate will recognize that 'B' accumulator pressure of 485 psig is out of Tech Spec tolerance of 611-685 psig (629-667 psig by L5 logs) and is low out of spec With the low N2 pressure on the accumulator the full volume of water would not inject, and the purpose of the accumulator is to flood the core on a LBLOCA until the ECCS pumps have started and begin to inject water to continue to remove heat. Knowing the affect that lower accumulator pressure will have on the RCS if the LOCA were to occur is system understanding for the purpose of the accumulator tanks, along with the assumption that one accumulator is expected to dump on the containment floor ('A' loop in this case) and the other two are expected to inject.

Sys #	System	Category	KA Statement
006	Emergency Core Cooling	K6 Knowledge of the effect of a loss or malfunction on the following will have on the ECCS:	BIT/borated water sources

K/A# K6.01 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-11.1.C Rev. 1 pg. 3, 4; 2OM-11.1.B Rev. 1 pg. 2

Question Source: Bank – Wolfe Creek 2015 NRC Exam Q32

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

Objective: 2SQS-11.1, Rev. 18 Obj. 1. Describe the function of the Safety Injection System and the associated major components as documented in Chapter 11 of the Operating Manual.

2SQS-11.1, Rev. 18 Obj 15. List the nominal value of the control room operating parameters associated with the Safety Injection System.

32. Given the following conditions:

- The plant is at 100% power.
- PRT temperature is 90°F and rising at 2°F/min due to tank in-leakage.
- PRT level is 74% and slowly rising.
- PRT pressure is 6.5 psig and slowly rising.

1) What is the minimum amount of time before A4-3H, Pressurizer Relief Tank Trouble alarms for PRT Temp High?
2) 2RCS-MOV516, PRZR Relief Tank Spray Valve _____ open(ed) when the PRT High temperature alarm is received.

A. 1) 18 minutes
   2) will automatically

B. 1) 56 minutes
   2) will automatically

C. 1) 18 minutes
   2) must be manually

D. 1) 56 minutes
   2) must be manually

Answer: C

Explanation/Justification: K/A is met by the candidate demonstrating the ability to determine when the PRT High temperature alarm will come in based on the current tank heatup rate due to PRT in leakage. Knowledge of the PRT high temp alarm setpoint is required to obtain the correct answer.

A. Incorrect. First part is correct. Second part is plausible because 2RCS-MOV516 closes automatically on PRT high level or pressure, and the candidate could confuse the auto features with auto opening on high temperature.

B. Incorrect. 56 minutes is plausible if the candidate thinks the high temperature alarm is above 200F, because the spray rate is designed to cool the PRT from 200F to 120F in about an hour. Second part is plausible because 2RCS-MOV516 closes automatically on PRT high level or pressure, and the candidate could confuse the auto features with auto opening on high temperature.

C. Correct. The PRT high temperature alarms at 125F, therefore 125-90=35/2 is 17.5 (18) minutes. 2RCS-MOV516 is manually opened per the ARP when the high temperature condition exists. The spray valve has no auto open features associated with it.

D. Incorrect. 56 minutes is plausible if the candidate thinks the high temperature alarm is above 200F, because the spray rate is designed to cool the PRT from 200F to 120F in about an hour. Second part is correct. 2RCS-MOV516 is manually operated.

Sys #	System	Category	KA Statement
007	Pressurizer Relief/Quench Tank	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including:	Monitoring quench tank temperature

K/A# A1.03 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-6.4.AAY Rev. 11 pg. 3; 2OM-6.1.D Rev. 3 pg. 7, 8

Question Source: New

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

Objective: 2SQS-6.4, Rev. 17 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.

2SQS-6.4, Rev. 17 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.

33. Given the following:

- The plant is at 100% power
- Containment pressure is 13.4 psia
- Containment temperature is 93°F

Subsequently:

- A load rejection results in a reactor trip
- Following the trip, a Pressurizer Safety valve opens, and will NOT reseat
- The PRT rupture disks function as designed
- Containment pressure is rising at 0.1 psia every 5 minutes
- Containment temperature is rising at 1°F every 5 minutes

Assuming these conditions remain constant, which of the following identifies the Containment Technical Specifications LCOs that will be affected one hour from now?

A. Both LCO 3.6.4, Containment Pressure, and LCO 3.6.5, Containment Air Temperature, will be exceeded.

B. Only LCO 3.6.4, Containment Pressure, will be exceeded.

C. Only LCO 3.6.5, Containment Air Temperature, will be exceeded.

D. Neither LCO 3.6.4, Containment Pressure, nor LCO 3.6.5, Containment Air Temperature, will be exceeded.

Answer: B

Explanation/Justification: The KA is matched because the operator must demonstrate knowledge of the effect that a loss the PRTS (Rupture Disk Releases) will have on the Containment (i.e. TS Limits are challenged).

A. Incorrect. Plausible if the candidate does not know the cnmt temperature or pressure Tech Spec values or miscalculates the one-hour trend.

B. Correct. After an hour cnmt pressure will rise 1.2 psia, therefore cnmt pressure will be 13.4 + 1.2 psia=14.6 psia. TS 3.6.4 - Containment pressure shall be ≥12.8 psia and ≤14.2 psia. Therefore, the LCO is NOT met at this time.

C. Incorrect. Plausible if the candidate does not know the cnmt temperature Tech Spec values. After the one-hour trend cnmt temp will rise 12°F to 105°F. TS 3.6.5 Containment average air temperature shall be ≥70°F and ≤108°F. Therefore, the LCO is met at this time.

D. Incorrect. Plausible if the candidate does not know the cnmt temperature or pressure Tech Spec values or miscalculates the one-hour trend.

Sys #	System	Category	KA Statement
007	Pressurizer Relief/Quench Tank	K3 Knowledge of the effect that a loss or malfunction of the PRTS will have on the following:	Containment

K/A# K3.01 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** TS 3.6.4 & TS 3.6.5

Question Source: Modified – Robinson 2016 Q35

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

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

34. Initial Event conditions:

- The plant is operating at 50% power.
- A6-1H, Primary Component Cooling Water System Trouble is LIT.
- CCP Surge Tank levels are 14 inches and LOWERING.
- The crew has entered AOP-2.15.1, Loss of Primary Component Cooling Water and taken appropriate actions to makeup to the CCP system.

Current conditions:

- CCP Surge Tank levels had lowered to 1 inch, and are currently 2 inches and RISING

What affect, if any, did this event have on the CCP system alignment?

A. 2CCP-AOV107s, RCPs Thermal Barrier Cooling Water Discharge valves will be CLOSED.

B. 2CCP-MOV177-1, Pri. Comp Cooling Return Hdr B Isolation valve will be CLOSED.

C. 2CCP-MOV151-1, Pri. Comp Cooling Supply Hdr Isolation - Outside Cnmt valve will be CLOSED.

D. No system alignment changes have occurred.

Answer: B

Explanation/Justification: K/A is met by the candidate's ability to monitor the automatic system isolation which occurs when the CCP surge tank lowers to 3 inches to isolate the non-nuclear safety portion of the CCP system in the event of a possible leak in that portion of the system.

A. Incorrect. Plausible because the RCP Thermal Barrier Cooling Water Discharge valves will auto close high flowrate of 58 gpm, or high pressure at 122 psig, but do not auto close due to low-low CCP surge tank level.
B. Correct. 2CCP-MOV177-1, Pri. Component Cooling Return Isolation valve automatically closes when CCP surge tank level reaches 3 inches.
C. Incorrect. Plausible because the Pri. Component Cooling Header Isolation – Outside Cnmt valve will automatically close on a CIB signal, but not low surge tank level.
D. Incorrect. Plausible that the CCP system alignment does not change if the candidate does not know the auto valve closure features of the CCP system.

Sys #	System	Category	KA Statement
008	Component Cooling Water	A4 Ability to manually operate and/or monitor in the control room:	Control of minimum level in the CCWS surge tank

K/A# A4.07 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-53C.4.2.15.1 Rev. 4 pg. 2; 2OM-15.1.D Iss. 4 Rev. 1 pg. 12

Question Source: New

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

Objective: 2SQS-15.1, Rev. 9 Obj. 10. Given a change in CCP system surge tank level, summarize how the system will automatically respond to the change in level.

2SQS-15.1, Rev. 9 Obj. 17. Given a plant low-low surge tank level signal, breakdown how the CCP System valve, pump, flow and/or electrical configuration will change as a result of the signal.

35. The plant is operating at 100% power.

- Pressurizer level channel selector switch as shown.
- 2RCS-LT460, PRZR CH. 2 Level fails LOW.

With no operator action, how will this failure affect charging flow and the PRZR heaters?

1) Charging flow will _____?

2) Pressurizer heaters will _____?

A. 1) rise
   2) energize

B. 1) rise
   2) de-energize

C. 1) lower
   2) energize

D. 1) lower
   2) de-energize

Answer: D

Explanation/Justification: K/A met since candidate must demonstrate knowledge of how a PRZR Level Control channel failure (PZR LCS) can affect the PRZR pressure control system (PZR PCS) by de-energizing the PRZR heaters.

A. Incorrect. First part is incorrect, charging flow will lower. Plausible if the candidate confuses LT459 and LT460, since LT459 failure would cause charging flow to rise. Second part is incorrect, pressurizer heaters would de-energize. Plausible if candidate assumes a rise in pressurizer level causes pressurizer heaters to energize in anticipation of PRZR insurge causing PRZR temperature to lower.

B. Incorrect. First part is incorrect, charging flow will lower. Plausible if the candidate confuses LT459 and LT460, since LT459 failure would cause charging flow to rise. Part 2) is correct, przr level failing to < 14% will isolate letdown and de-energize all przr heaters.

C. Incorrect. Part1 is correct, charging flow will lower as PRZR level rises, as sensed by the upper channel. Part 2) is incorrect, pressurizer heaters would de-energize. Plausible if Candidate assumes rise in pressurizer level causes pressurizer heaters to energize in anticipation of PRZR insurge causing PRZR temperature to lower.

D. Correct. When LT460 fails low, at <14% przr level, letdown will isolate, and all przr heaters will de-energize. When letdown isolates, charging flow will lower because LT459 will sense a rise in przr level indication and reduce charging flow in response.

Sys #	System	Category	KA Statement
010	Pressurizer Pressure Control	K1 Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems:	PZR LCS

K/A# K1.08 **K/A Importance** 3.2 **Exam Level** RO

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

Question Source: Bank – 2016 Surry Q12

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

Objective: 2SQS-6.4, Rev. 17 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.

36. The plant is at 100% power with all systems NSA **EXCEPT**, 2RCS-PCV455A, PRZR Spray Valve is in its FAIL position due to a broken air line.

- RCS Pressure is 2235 psig and STABLE.
- RCS Tavg is 574°F and STABLE.
- 2RCS*PT444, "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. TWO (2) PRZR PORVs will be OPEN.

B. 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 2RCS-PCV455A will fail closed with a broken air line and will not respond when the pressurizer control system demands it to open due to 2RCS*PT444 failing high.

A. Incorrect. These indications are indicative of 2RCS*PT445 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 2RCS*PT444 in the high direction will typically result in 2RCS-PCV455C opening and both PRZR spray Valves failing full open. 2RCS-PCV455A fails closed on a loss of air, therefore only 2RCS-PCV455B 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, 2RCS-PCV455A fails closed on a loss of air, therefore 2RCS-PCV455B 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:** 2OM-6.4.IF, Rev. 13, pg. 16, 19, 24, 25
2OM-53C.4.2.34.1 Rev. 21 pg. 30

Question Source: Bank – 2LOT8 Q36

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

Objective: 2SQS-6.4 Rev. 17 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

37. With the plant operating at 75% power, a Reactor Trip signal will be generated on a Turbine trip provided 2/3 Emergency Trip Header pressures are less than _____(1)_____ psig OR 4/4 _____(2)_____ are CLOSED.

Which of the following completes the statement above?

A. 1) 1013
2) Throttle valves

B. 1) 1600
2) Throttle valves

C. 1) 1013
2) Governor valves

D. 1) 1600
2) Governor valves

Answer: A

Explanation/Justification: The KA is matched because the question requires the applicant to know the turbine trip signals which will provide a signal to RPS to trip the reactor (i.e. cause-effect).

A. Correct. When >P9 (>49% rx power) 2/3 Auto Stop Oil pressures <1013 psig or 4/4 Turbine Throttle valves closed, will generate a Reactor Trip.
B. Incorrect. Plausible because 1600 psig is the Auto start setpoint of stand-by E.H. Fluid Pump. Second part is correct.
C. Incorrect. When >P9 (>49% rx power) 2/3 Auto Stop Oil pressures <1013 psig will generate a Reactor Trip, but the second part is incorrect. Governor valves are plausible because governor valves and throttle valves both isolate Main Steam from the turbine.
D. Incorrect. Plausible because 1600 psig is the Auto start setpoint of stand-by E.H. Fluid Pump. Second part is also incorrect but plausible because both governor valves and throttle valves isolate Main Steam from the turbine.

Sys #	System	Category	KA Statement
012	Reactor Protection	K1 Knowledge of the physical connections and/or cause effect relationships between the RPS and the following systems:	T/G

K/A# K1.06 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 3SQS1.1 PPNT U2 Rev 8 slide 72
2OM-1.5.B.1 Rev 2 pg. 3

Question Source: Bank – McGuire 2012 Q1

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.41.b(6)

Objective: 3SQS-1.1, Rev. 8 Obj. 2. Describe the control, protection and interlock functions for the field 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.

38. An internal fault (short circuit) occurs in 2RCS-LK459F, Master PRZR Level Controller.

Which of the following describes the effect this fault will have on the Reactor Protection System?

The **CONTROLLER** fault will _____.

A. **NOT** feed back into the protection circuit due to the use of isolation devices

B. feed back into the protection circuit, causing the "SELECTED" channels to trip

C. feed back into the protection circuit, preventing the "SELECTED" channels from tripping

D. **NOT** feed back into the protection circuit since separate transmitters are used for control and protection

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Answer: A

Explanation/Justification: K/A is met by demonstrating the knowledge that the pressurizer level transmitters are used for both control and protection, and that a fault on the control circuit will not feed back into the protection circuit due to the installed isolation amplifiers.

A. Correct. Isolation amplifiers maintain the protection and control circuits independent of each other which allows the 3 common przr level instruments to be used for both protection and control.
B. Incorrect. Plausible because this could be the case is there were not isolation amplifiers separating the protection and control circuits.
C. Incorrect. Plausible because this could be the case is there were not isolation amplifiers separating the protection and control circuits.
D. Incorrect. Plausible if the candidate confuses przr pressure control przr level control system.

Sys #	System	Category	KA Statement
012	Reactor Protection	K4 Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following:	Separation of control and protection circuits

K/A# K4.09 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-6.4.IF Rev. 13 pg. 12
3SQS-1.2 Rev. 8 pg. 5

Question Source: Bank – 2LOT17 Audit Exam Q59

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

Objective: 3SQS-1.2, Rev. 8 Obj. 1. Describe the function of the Reactor Protection System Hardware and the associated major components as documented in the Reactor Protection System Hardware associated Operating Manual.

2SQS-6.4, Rev. 17 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.

39. The plant was initially at 100% power when the following sequence of events occurred:

- CNMT pressure transmitter, 2LMS*PT953 fails LOW.
- No actions have been taken in response to the failed pressure transmitter.
- Then, a steam break occurred in Containment causing Containment pressure to rapidly rise.

Complete the following statement describing the required ESF coincidence and ESF Train response:

The combination of 2LMS*PT_____(1)_____ of the remaining Containment pressure channels must sense the Containment high-pressure condition to generate a Safety Injection signal (SIS), and the SIS signal will be sent to SSPS _____(2)_____.

	(1)	(2)
A.	950 & 951	Train 'A' only
B.	950 & 952	Train 'A' and 'B'
C.	951 & 952	Train 'A' only
D.	951 & 952	Train 'A' and 'B'

Answer: D

Explanation/Justification: K/A is met by having the candidates differentiate between ESF actuation inputs of the containment pressure channels and determine that only three of the four cnmt pressure channels are used for Safety Injection actuation. They must also determine that both trains of safety injection will receive a start signal.

A. Incorrect. First part is correct. Second part is plausible if the candidate thinks that only 'A' train will actuate since channel IV (953) is a 'B' Train-powered protection channel.

B. Incorrect. The normal coincidence for High 1 cnmt pressure SIS is 2/3 out of 2LMS-951, 952, 953. With 2LMS-953 failing low, it will require both (2/2) of the remaining channels to actuate a safety injection, however PT 950 does not input to the SIS logic. A SIS signal is sent to both trains of RPS for redundancy.

C. Incorrect. Plausible if the candidate thinks 2LMS*PT953 is not one of the 3 inputs to High 1. This is incorrect because 2LMS*PT950 is one of the four pressure detectors, and it is not used as an input to High 1. Its input is used only for CIB. Second part is plausible if the candidate thinks that only 'A' train will actuate since channel IV (953) is a 'B' Train-powered protection channel.

D. Correct. The normal coincidence for High 1 cnmt pressure SIS is 2/3 out of 2LMS-951, 952, 953. With 2LMS-953 failing low, it will require both (2/2) of the remaining channels to actuate a safety injection. A SIS signal is sent to both trains of RPS for redundancy.

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:	Definitions of safety train and ESF channel

K/A# K5.01 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** UFSAR Figure 7.3-13 Rev 23

Question Source: New

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

Objective: 2SQS-13.1, Rev. 18 Obj.11. Describe the control, protection and interlock functions for the control room components associated with the Containment Depressurization System, including automatic functions, setpoints and changes in equipment status as applicable. 2SQS-13.1, Rev. 18 Obj 12. Given a Containment Depressurization System configuration and without reference material, describe the Containment Depressurization System control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable. b. Safety Injection Signal

40. Given the following conditions:

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

What is the plant response?

The Reactor will trip, and _____

A. all AFW pumps start.

B. no AFW pumps start.

C. only the MDAFW pumps start.

D. only the TDAFW pump starts.

Answer: D

Explanation/Justification: K/A is met by testing the candidates' knowledge of the effect that two SG NR level channels indicating a low-low condition on one SG will have on an Engineered Safety Features Actuation system such as the Auxiliary Feed System.

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

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

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

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

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

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

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

References provided to Candidate None **Technical References:** UFSAR Logic Fig. 7.3-12 Rev 12
UFSAR Logic Fig. 7.3-19 Rev 7
2OM-24.1.D Rev 6 pg 18

Question Source: Bank – 2LOT17 Q41 **(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. 11 - Given a specific plant condition, predict or describe the response of the reactor protection system trip logics & ESFAS 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.

41. Which signal will automatically CLOSE **ALL** of the following valves?

2SWS*MOV152-1, CNMT Air Recirc Clg Coils Outside CNMT Serv Water Inlet Valve
2SWS*MOV152-2, CNMT Air Recirc Clg Coils Inside CNMT Serv Water Inlet Valve
2SWS*MOV155-1, CNMT Air Recirc Clg Coils Outside CNMT Serv Water Outlet Valve
2SWS*MOV155-2, CNMT Air Recirc Clg Coils Inside CNMT Serv Water Outlet 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. 2SWS*MOV152-1/2 and 2SWS*MOV155-1/2 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:** 2OM-53A.1.A-0.5, Rev. 3 pg. 3-5
U2 Fig. 29-4 (RM-0429-004 Rev. 14)

Question Source: Bank – 2LOT7 Q41

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

Objective: 2SQS-29.1 Obj. 3. Describe the control, protection and interlock functions for the field components associated with the Chilled Water System, including automatic functions, setpoints, and changes in equipment status as applicable.

42. The plant experienced a Large Break LOCA and Loss of Offsite Power 30 minutes ago.

The following conditions currently exist:

- 2-2 EDG failed to start.
- 2RSS-MOV155C, Recirc Pump 21C Outside Cnmt Suction Valve has failed closed due to an electrical fault.
- The crew has entered ES-1.3, Transfer to Cold Leg Recirculation.
- RWST level is 365 inches and lowering.
- CNMT pressure peaked at 27 psig and is currently 1 psig and lowering.
- The BOP operator noticed the following indications:

	'A' RSS Pump	'B' RSS Pump	'C' RSS Pump	'D' RSS Pump
Discharge Pressure	120 psig	0 psig	0 psig	0 psig
Discharge Flow	3900 gpm	0 gpm	0 gpm	0 gpm

- A-07 or A-08, Cold Leg Recirculation Actuation / Restoration are in progress.

1) Based on the current RWST level, how many Recirc Spray pumps **should have received** a start signal?
2) How will the crew respond to the current plant conditions?

A. 1) Two
2) Remain in ES-1.3, Transfer to Cold Leg Recirculation.

B. 1) Four
2) Remain in ES-1.3, Transfer to Cold Leg Recirculation.

C. 1) Two
2) Transition to ECA-1.1, Loss of Emergency Coolant Recirculation.

D. 1) Four
2) Transition to ECA-1.1, Loss of Emergency Coolant Recirculation.

Question 42

Answer: D

Explanation/Justification: K/A is met by demonstrating the ability to interpret given plant conditions and determining that the necessary Containment spray (Recirc Spray) pumps are not available due to a loss of power, and a loss of suction to a pump, then transition to ECA-1.1, Loss of Emergency Coolant Recirculation in response to the malfunctions. This is RO level knowledge based on required system knowledge, and the purpose and mitigative strategy of ECA-1.1.

A. Incorrect. First part is plausible because only 'C' and 'D' RSS pumps start on Recirc Mode Initiation signal which occurs at 369 inches with an SI signal present, but an RWST Low level at 381 inches with a CIB signal will start all four pumps. Remaining in ES-1.3 is plausible because we would remain in ES-1.3 if the 'C' or 'D' RSS pump were running because only one flowpath is required but based on the information given only the 'A' RSS pump has flow which is only supplied to the spray header.

B. Incorrect. First part is correct. Remaining in ES-1.3 is plausible because we would remain in ES-1.3 if the 'C' or 'D' RSS pump were running because only one flowpath is required but based on the information given only the 'A' RSS pump has flow which is only supplied to the spray header.

C. Incorrect. First part is plausible because only 'C' and 'D' RSS pumps start on Recirc Mode Initiation signal which occurs at 369 inches with an SI signal present, and RWST Low level at 381 inches with a CIB signal will start all four pumps. Second part is correct.

D. Correct. RWST Low level at 381 inches with a CIB signal will start all four RSS pumps. The stem states that a large break LOCA has occurred and cnmt pressure peaked at 27 psig, therefore the candidate should recognize that a CIB would have occurred. Transition to ECA-1.1 is required because without either 'C' or 'D' RSS pump running, a recirc flowpath cannot be established and ECA-1.1 provides actions to restore emergency coolant recirculation.

Sys #	System	Category	KA Statement
026	Containment Spray	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of containment spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature exceeded cavitation, voiding), or sump level below cutoff (interlock) limit

K/A# A2.07 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None

Technical References:
2SQS-13.1 Iss 1 Rev 18 slides 15, 26
2OM-53A.1.ECA-1.1 Iss 3 Rev 0 pg. 1
2OM-53A.1.ES-1.3 Iss 3 Rev 1 pg. 4
2OM-53B.4.ES-1.3 Iss 3 Rev 1 pg. 13

Question Source: New

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

Objective: 2SQS-13.1, Rev. 18 Obj. 14. Given a specific plant condition, predict the response of the Containment Depressurization 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-53.3, Rev. 5 Obj. 6. Given a set of conditions, locate and apply the proper Emergency Operating Procedures, IAW BVPS-EOP Executive Volume.

43. Given the following conditions:

- A LOCA has occurred
- CNMT pressure is 14.8 psig and lowering
- RWST level is 460 inches and lowering

Based on the conditions given, which of the following indications would be seen in the control room?

	'A' Quench Spray Pump	'A' QS Pump Disch. Valve (2QSS-MOV101A)	'B' Recirc Spray Pump	'B' RS Pump Disch. Valve (2RSS-MOV156B)
A.	Off	Closed	Off	Closed
B.	Running	Open	Running	Open
C.	Running	Open	Off	Open
D.	Running	Open	Off	Closed

Answer: C

Explanation/Justification: K/A is met by demonstrating the ability to identify the expected conditions of the control switch indications of two containment spray system pumps and their associated discharge valves based on a CIB actuation condition, and an RWST level greater than the starting setpoint for the RSS pumps.

A. Incorrect. Plausible if the candidate doesn't recognize that a CIB has occurred which would start the QS pump, and provide an open signal to both discharge valves, even though they are both NSA open. It is correct that the RSS pump will not be running.

B. Incorrect. Plausible if the candidate recognizes that a CIB has occurred and thinks both the QS and RSS pumps will receive a start signal. This is incorrect because the RSS pump needs a CIB signal coincident with a low RWST level of 381 inches. Both discharge valves are NSA open.

C. Correct. With cnmt pressure >11.1 psig (CIB actuation setpoint) the QS pumps will auto start. The RSS pump will not start because even though there is a CIB signal present, the RSS pumps also require an RWST low level of 381 inches to auto start, but the stem states that the RWST is at 460 inches and lowering. Both discharge valves are NSA open, and also receive an open signal on a CIB actuation.

D. Incorrect. First three indications are correct, but the 'B' RSS pump discharge valve is normally open, and also receives an open signal upon a CIB actuation. Plausible because the pump is not running and 2RSS-MOV156B is a cnmt isolation valve.

Sys #	System	Category	KA Statement
026	Containment Spray	A4 Ability to manually operate and/or monitor in the control room:	CSS controls

K/A# A4.01 **K/A Importance** 4.5 **Exam Level** RO

References provided to Candidate None

Technical References:
2OM-13.1.D Rev. 4 pg. 2, 5, 6, 8
2OM-13.3.B.1 Rev. 11 pg. 3
2OM-13.3.B.2 Rev. 8 pg. 8
2SQS-13.1 LP Rev. 18 lss. 1 pg. 39

Question Source: New

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

Objective: 2SQS-13.1, Rev. 18 Obj.14. Given a specific plant condition, predict the response of the Containment Depressurization 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 is operating at 100 % power.
- A4-5A, Radiation Monitoring System Trouble is LIT.
- A4-5C, Radiation Monitoring Level High is LIT.
- 2ARC-RQ100, Condenser Air Ejector Discharge is in alarm.
- All systems function as designed and no operator actions have been taken.

Based on the above conditions, which of the following describes the alignment of the Condenser Air Ejector Discharge, and when it occurs?

A. Air Ejector discharge is AUTO aligned to the containment on an ALERT alarm level.

B. Air Ejector Discharge Blowers will AUTO trip on a HIGH alarm level.

C. Air Ejector discharge is AUTO aligned through the Charcoal Delay Beds on a HIGH alarm level.

D. Air Ejector discharge continues to atmosphere until manual action occurs.

Answer: D

Explanation/Justification: K/A is met by demonstrating the ability to monitor the Air Ejector process rad monitor during an alarm condition and know that there are no auto actions, but that the air ejector discharge will continue to discharge to the atmosphere until it is manually aligned to the Charcoal Delay Beds.

A. Incorrect. Plausible if the candidate confuses Unit 2 with Unit 1 since Unit 1 does AUTO align to the containment.
B. Incorrect. Plausible because the blowers are aligned in the discharge flowpath but are not used for NSA lineup for discharge to atmosphere. They are used when aligning the AE discharge to the Charcoal Delay Beds.
C. Incorrect. The air ejector discharge is aligned to the charcoal delay beds however, this is not an AUTO action.
D. Correct. The ARP directs the air ejector discharge be aligned to the gaseous waste system through the delay beds in accordance with 2OM-19.4.H due confirmed high radiation level from the condenser air removal system which is indicative of a SG tube leak. AOP 2.6.4 also directs the alignment through the delay beds.

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:	Air ejector PRM

K/A# A1.10 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None

Technical References: 2SQS-26.1 PPNT Rev. 10 slide 8
2OM-26.4.H Rev. 20 pg. 2
2OM-43.4.AAD Rev. 5 pg. 2

Question Source: Modified – 2LOT8 Q63

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

Objective: 2SQS-43.1, Rev. 11 Obj. 7. 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.

2SQS-26.1, Rev. 10 Obj. 15. Given a specific plant condition, predict the response of the Main Turbine, Main Condenser, Condenser Air Removal system, and Moisture Separator Reheaters 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.

45. Which of the following conditions, BY ITSELF, would result in automatic closure of the Main Steam Isolation Valves?

A. Raising RCS pressure to 2235 psig with RCS temperature at 460°F.

B. Cooling down the plant to 500°F with RCS pressure at 1950 psig.

C. RCS pressure lowering to 1830 psig due to a stuck open Pressurizer Safety Valve.

D. Containment pressure rising to 5.6 psig due to an RCS leak.

Answer: A

Explanation/Justification: K/A is met by evaluating the knowledge of the auto isolation setpoints of the Main steam isolation valves given various plant conditions

A. Correct. By raising RCS above the P-11 setpoint of 2000 psig will auto enable the auto MSLI for steamline pressure of 500 psig. Saturation pressure for 460F is approx. 451 psig, therefore a MSLI will occur.

B. Incorrect. Plausible distractor because 500F could be confused with the MSLI setpoint of 500 psig, but in this case 500F is approx. 665psig, and the RCS pressure is less than P-11 (2000 psig) which means the 500 psig MSLI may or may not be in effect depending whether it has been blocked. Either way this answer is incorrect.

C. Incorrect. Plausible distractor because the candidate may confuse the SI pressure with the steam line press. setpoint of the Auto MSLI.

D. Incorrect. Plausible distractor if the candidate thinks 5 psig will cause a MSLI. 5 psig is the SI auto initiation setpoint, and 7 psig is the auto MSLI setpoint.

Sys #	System	Category	KA Statement
039	Main and Reheat Steam	A3 Ability to monitor automatic operation of the MRSS, including:	Isolation of the MRSS

K/A# A3.02 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate Steam Tables **Technical References:** Tech Specs pg. 3.3.2-10 & 13

Question Source: Bank – 1LOT18 Q42

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

Objective: 3SQS-1.1, Rev. 8 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.

46. Given the following conditions:

- Reactor power is currently on hold at 15%.
- The Main Generator is synchronized to the grid.
- The crew is manually controlling SG levels at 44%.

Subsequently:

- The Unit Supervisor directs raising power to 30%.
- When the Turbine load increase commenced, the Governor valves opened rapidly.
- Reactor power increases to 25%.
- All SG level deviation from setpoint alarms are LIT.

Which of the following completes these statements?

The initial SG level deviation alarms occur because SG narrow range levels indicate 5% _____(1)_____ than programmed level.

After the transient has stabilized, the Feedwater Regulating Bypass valves will be further _____(2)_____ as compared to the pre-event position.

A. 1) less
   2) open

B. 1) greater
   2) closed

C. 1) less
   2) closed

D. 1) greater
   2) open

Question 46

Answer: D

Explanation/Justification: K/A is met by the candidate demonstrating the ability to operate the BFRVs in manual when power increases, and demonstrating the knowledge of the effect of SG swell will have on the SG indications in the control room.

A. Incorrect: First part is plausible to believe that increasing steam flow would reduce SG inventory and therefore level. The rapid increase in steam demand actually caused swell in the SGs, which would be indicated by a rising level. Second part is correct.

B. Incorrect: 1st part is correct. 2nd part is plausible to believe that since the higher level was due to a transient, feed flow might not need adjusting. Increased steam demand would require increased feed flow to maintain level, in spite of the initial swell in the SGs due to the steam flow increase.

C. Incorrect: First part is plausible to believe that increasing steam flow would reduce SG inventory and therefore level. The rapid increase in steam demand actually caused swell in the SGs, which would be indicated by a rising level. 2nd part is plausible to believe that since the higher level was due to a transient, feed flow might not need adjusting. Increased steam demand would require increased feed flow to maintain level, in spite of the initial swell in the SGs due to the steam flow increase.

D. Correct: The rapid increase in steam demand will cause swell in the SGs, which would be indicated by a rising level causing the SG level deviation higher than the program level. After the transient has stabilized steam flow will be higher than pre-event due to the power increase. This will require increased feed flow to restore and maintain SG level at program, therefore Bypass valves would be further open as compared to pre-event.

Sys #	System	Category	KA Statement
059	Main Feedwater	A4 Ability to manually operate and monitor in the control room:	Feedwater control during power increase and decrease

K/A# A4.03 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 2SQS-24.1 Rev. 26 pg. 47-48
2OM-24.4.AAL Rev. 5

Question Source: Bank – Turkey Point 2015 NRC Exam Q16

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

Objective: 2SQS-24.1, Rev. 26 Obj. 13. List the nominal value of the control room operating parameters associated with the Main Feedwater, Startup Feedwater, Auxiliary Feedwater System and Steam Generator Water Level Control Systems.

2SQS-24.1, Rev. 26 Obj. 17.Given a specific plant condition, predict the response of the Main Feedwater, Startup Feedwater, Auxiliary Feedwater System or Steam Generator Water Level Control System's 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.

47. Given the following:

- The plant was operating at 100% power.
- The Reactor Tripped due to a Loss of all Main Feedwater.
- All RCPs are running.
- Due to equipment malfunctions, **ONLY** 2FWE-P22 Turbine Driven AFW Pump is in service.

Plant conditions have stabilized:

- The TDAFW Pump speed has begun to LOWER due to a malfunctioning governor.

Which of the following describes how the change in AFW flow will affect Pressurizer level, including the reason?

Indicated Pressurizer level will initially _____.

A. rise due to a bubble formation in the Rx Vessel Head

B. rise due to decreased primary to secondary heat transfer

C. lower due to a density decrease in the Pressurizer level

D. lower due to Pressurizer outsurge to RCS

Answer: B

Explanation/Justification: K/A is met with the candidate's knowledge of how reduced AFW flow to the SGs will cause the RCS temperature to rise due to the reduction in heat transfer within the SGs, thus causing PRZR level to rise.

A. Incorrect. Plausible because 'rise" could be correct if such a bubble were to form in the vessel head, but with the RCPs running a bubble should not form in the Reactor Vessel Head.
B. Correct. As AFW Pump speed decreases due to the governor valve closing, less heat is removed from the RCS via less steam to the pump turbine and less feedwater flow to the SGs will occur. PRZR will initially rise as the RCS temperature rises.
C. Incorrect. Since PRZR level will rise, not lower. Plausible because density decrease in Pressurizer level is possible with an insurge of cooler water lowering the saturation temperature of the fluid. This effect would cause the water volume to contract.
D. Incorrect. Since PRZR level will rise, not lower. Plausible because PRZR pressure will increase and could cause an outsurge. If an applicant believes PRZR level will lower, it is logical to believe that the level change is due to an outsurge.

Sys #	System	Category	KA Statement
061	Auxiliary/Emergency Feedwater	K5 Knowledge of the operational implications of the following concepts as the apply to the AFW:	Relationship between AFW flow and RCS heat transfer

K/A# K5.01 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-53B.4.E-0 Iss 3 Rev 0 pg. 19

Question Source: Bank – Turkey Point 2011 Q41

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

Objective: GO-GPF.T7, Rev. 2 Obj. 10. Describe the relationship between heat transfer rate in a heat exchanger and the factors which affect it.

48. Given the following conditions:

- The plant is at full power with all equipment in NSA.
- An electrical malfunction on the 'A' 4KV bus results in a loss of the bus.

Assuming no operator actions, 5 minutes later, what will be the status of the 'A' and 'B' **480VAC** buses?

A. 'A' energized 'B' energized

B. 'A' de-energized 'B' de-energized

C. 'A' de-energized 'B' energized

D. 'A' energized 'B' de-energized

Answer: A

Explanation/Justification: K/A met with the electrical distribution system knowledge of the sources of normal and alternative power supplies for 480 VAC busses. Specifically, that a loss of the 'A' 4KV bus will cause the 'A' 480vac bus to de-energize. The candidate will then determine that the 2A-2B Bus Tie breaker will auto close supplying the 'A' 480 bus from the 'B' 480 bus, which is powered from the 'C' Normal 4KV bus.

A. Correct. 480 Bus 2A will be powered from 480 bus 2B via the auto closure of the Buses 2A and 2B tie ACB on the undervoltage condition. Buses 2A and 2B tie ACB will be closed automatically provided its overcurrent trip is reset and all of the following conditions exist: Control switches for Bus 2A supply ACB and Bus 2B supply ACB in AFTER CLOSE, Bus 2A transformer 2-1A undervoltage, and Bus 2A supply ACB open or racked out or 480 V transformers 2-1A and 2-1B synchronized. All of these requirements are NSA for power operation. Bus 2A supply ACB will be opened provided any of the following conditions exist: 480 VAC Bus 2A transformer 2-1A undervoltage, or 4160 VAC breaker feed to transformer 2-1A open.

B. Plausible if the candidate thinks 480 busses 2A and 2B are both powered from the 'A' 4KV bus, but this in incorrect as busses 2A and 2E are supplied from the 'A' 4KV bus.

C. Plausible if the candidate thinks 480 bus 2A will remain de-energized, and 480 bus 2B will remain energized from 'C' 4KV bus, but this is incorrect because the Buses 2A and 2B tie ACB will auto close to energize the 480 bus 2A on undervoltage.

D. Plausible if the candidate thinks 480 bus 2B is supplied from 4KV bus, and 480 bus 2A remains energized.

Sys #	System	Category	KA Statement
062	AC Electrical Distribution	K4 Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following:	One-line diagram of 4kV to 480V distribution, including sources of normal and alternative power

K/A# K4.07 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-37.1.D Rev. 1 pg. 10-11
3SQS-37.1 U2 PPNT Rev 9 Iss 2 slide 11 & 69

Question Source: Bank – Vision #134785

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

Objective: 3SQS-37.1, Rev. 9 Obj. 9.. Given a change in plant conditions, predict the 480 VAC Distribution System responses, including automatic functions; and changes in system parameters and previously identified components and control systems.

49. The plant is operating at 50% power when a loss of DC Bus 2-2 occurs.

Which of the following DC loads will lose power during this event?

A. 2TML-P215, Main Turb Emer DC Bearing Oil Pump

B. UPS*VITBS2-4 Uninterruptable Power Supply

C. Main Feed Regulating Valves

D. Condenser Steam Dumps

Answer: C

Explanation/Justification: K/A is met with the candidate's knowledge that the Main Feed Reg. Valves are powered from DC bus 2-2.

A. Incorrect. Plausible because 2TML-P215, Main Turb Emer DC Bearing Oil Pump is DC powered, but it receives DC power from DC Bus 2-6.
B. Incorrect. Plausible because UPS*VITBS2-4 Uninterruptable Power Supply is supplied DC power from DC Bus 2-4, but the candidate may think DC Bus 2-2 supplies both Vital Bus UPS 2 & 4 due to their importance to plant operation.
C. Correct. MFRVs are supplied from DC Bus 2-2.
D. Incorrect. Plausible because Condenser Steam Dumps are supplied from DC power, but it is supplied from DC Bus 2-5.

Sys #	System	Category	KA Statement
063	DC Electrical Distribution	K2 Knowledge of bus power supplies to the following:	Major DC loads

K/A# K2.01 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-39.5.B.4 Rev. 19 pg 26; 2OM-53C.4.2.39.1B Rev. 4 pg. 8

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. 19. Given a 125 VDC Distribution System configuration, and without reference material, describe the 125 VDC Distribution System control room response to the following malfunctions, including automatic functions and changes in equipment status: c. Loss of DC Bus

50. Given the following plant conditions:

- The plant is operating at 100% power.
- 2OST-36.1, "Emergency Diesel Generator (2EGS*EG2-1) Monthly Test" is in progress.
- 2-1 EDG is paralleled to the grid, carrying 2000 KW load.
- A grid disturbance causes grid frequency to rise very slightly.
- Grid Voltage remains constant.

Which of the following describes the response of 2-1 EDG?

The response of 2-1 EDG is that _____ .

A. 1) KW output RISES and KVAR output is STABLE

B. 1) KW output LOWERS and KVAR output is STABLE

C. 1) KW output and KVAR output RISES

D. 1) KW output and KVAR output LOWERS

Answer: B

Explanation/Justification: K/A is met by demonstrating the ability to monitor the EDG governor control system operation while the EDG is paralleled to the grid, and a grid disturbance causes frequency to rise.

A. Incorrect. If frequency drops, the EDG would attempt to increase speed, which will pick up real load.

B. Correct. KW output will lower when the EDG sheds more load to the grid as grid frequency rises.

C. Incorrect. KVAR output will remain essentially constant if grid voltage is constant. If it did change it would change in the opposite direction of KW.

D. Incorrect. KVAR output will remain essentially constant if grid voltage is constant.

Sys #	System	Category	KA Statement
064	Emergency Diesel Generator	A3 Ability to monitor automatic operation of the ED/G system, including:	Operation of the governor control of frequency and voltage control in parallel operation

K/A# A3.05 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None

Technical References: GO-GPF.C5 Rev. 2 pgs. 131-135
2OST-36.1, Rev. 76 pg. 11
TS SR 3.8.1.3 & 3.8.1.3 Bases. Rev. 29

Question Source: Modified – 2LOT17 Q48

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

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.

51. 2SSR-RQI100, Steam Generator Blowdown Radiation Monitor senses an elevated radiation level condition.

    Which alarm condition will provide the signal to AUTOMATICALLY CLOSE which of the following valves?

    2BDG*AOV100A1, B1, C1, SG Blowdown Outside CNMT Isolation valves
    2SSR*AOV117A, B, C, SG Blowdown Sample Outside CNMT Isolation valves

A. ALERT alarm level will automatically close 2BDG*AOV100s ONLY.

B. HIGH alarm level will automatically close 2BDG*AOV100s ONLY.

C. ALERT alarm level will automatically close BOTH 2BDG*AOV100s and 2SSR*AOV117s.

D. HIGH alarm level will automatically close BOTH 2BDG*AOV100s and 2SSR*AOV117s.

Answer: D

Explanation/Justification: K/A is met by the candidate's ability to monitor the SG Blowdown isolation and sample valves auto closure when a High Steam Generator Blowdown Radiation Monitor alarms.

A. Incorrect. Plausible because it is logical to isolate the SG Blowdown Outside CNMT Isolation valves to isolate Blowdown when there are elevated radiation levels, but both valves will automatically close, and only when there is a HIGH alarm condition.
B. Incorrect. Plausible because it is logical to isolate the SG Blowdown Outside CNMT Isolation valves to isolate Blowdown when there is a HIGH radiation level, but both valves will automatically close.
C. Incorrect. Plausible because both valves will close, but only when a HIGH alarm condition exists.
D. Correct. Both 2BDG*AOV100s and 2SSR*AOV117s will automatically close when a HIGH alarm condition exists on 2SSR-RQI100. No auto actions occur when an ALERT radiation monitor condition exists on 2SSR-RQI100.

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:** 2OM-43.5.B.3 Rev. 2 pg. 2; 2OM-43.4.AEF Rev. 8 pg. 2, 3

Question Source: New

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

Objective: 2SQS-25.1, Rev. 10 Obj. 8. Describe the control, protection and interlock functions for the control room components associated with the Steam Generator Blowdown System, including automatic functions, setpoints and changes in equipment status as applicable.

52. The plant is operating at 100% power.

- Service Water Pumps 2SWS*P21A **AND** B are **BOTH** in service.
- Service Water Pump 2SWS*P21C is on clearance and unavailable.
- 'A' and 'B' Primary Plant Component Cooling Water Heat Exchangers (HXs) are **BOTH** in service.
- 'A' and 'B' Secondary Plant Component Cooling Water Heat Exchangers (HXs) are **BOTH** in service.

A large Service water leak develops at the inlet to the 'A' Primary Plant Component Cooling Water Heat Exchanger. The leak causes the following Service water header pressure indications:

- Service Water Header Press 2SWS-PI113A is 30 psig and stable.
- Service Water Header Press 2SWS-PI113B is 40 psig and stable.

1) **IF** these Service Water Header Pressures are sustained for greater than 1 minute, what will be the impact on Secondary Plant Component Cooling Water HX operations?

2) IAW AOP-2.30.1, Service Water/Normal Intake Structure Loss, what actions will be **REQUIRED IF BOTH** Service Water Header Pressures drop below 34 psig and cannot be restored above 34 psig?

A. 1) **ONLY** the 'A' Secondary Plant Component Cooling Water HX will be **ISOLATED**.
2) Manually trip the reactor and Go to E-0, Reactor Trip or Safety Injection.

B. 1) **ONLY** the 'A' Secondary Plant Component Cooling Water HX will be **ISOLATED**.
2) Perform a rapid shutdown IAW AOP-2.51.1, Unplanned Power Reduction.

C. 1) **NEITHER** Secondary Plant Component Cooling Water HX will be **ISOLATED**.
2) Manually trip the reactor and Go to E-0, Reactor Trip or Safety Injection.

D. 1) **NEITHER** Secondary Plant Component Cooling Water Heat Exchanger will be **ISOLATED**.
2) Perform a rapid shutdown IAW AOP-2.51.1, Unplanned Power Reduction.

Question 52

Answer: C

Explanation/Justification: K/A is met with the knowledge of how the service water system will respond to a low pressure condition in one service water header, and the automatic features which occur within the system for the indications the candidate was given, then the candidate must apply procedural knowledge to manually trip the plant if service water is lost to the secondary side.

A. Incorrect. Plausible because 2SWS*MOV107A will auto close when pressure is less than 34 psig for greater than 45 seconds. However, this only isolates the "A" SW header to the secondary component cooling HXs. The "B" header will continue to supply BOTH secondary plant component cooling water heat exchangers. Manually tripping the reactor is correct.

B. Incorrect. Plausible because 2SWS*MOV107A will auto close when pressure is less than 34 psig for greater than 45 seconds. However, this only isolates the "A" SW header to the secondary component cooling HXs. The "B" header will continue to supply BOTH secondary plant component cooling water heat exchangers. Manually tripping the reactor is the required action, NOT perform a rapid shutdown. Performing an Unplanned Power Reduction at 5% is a common method of plant shutdown.

C. Correct. Neither Secondary Plant CCW HX will be isolated from service water because the 'B' header will still be supplying the secondary side even when 2SWS*MOV107A auto closes isolating the 'A' header from the secondary side 45 seconds after the 'A' header reaches the low pressure setpoint of 34 psig. If no service water is available to the Secondary Plant CCW HX, then a Reactor trip is required.

D. Incorrect. Neither Secondary Plant CCW HX will be isolated from service water because the 'B' header will still be supplying the secondary side even when 2SWS*MOV107A auto closes isolating the 'A' header from the secondary side 45 seconds after the 'A' header reaches the low pressure setpoint of 34 psig. Manually tripping the reactor is the required action, NOT perform a rapid shutdown. Performing an Unplanned Power Reduction at 5% is a common method of plant shutdown.

Sys #	System	Category	KA Statement
076	Service 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: U2 Vond RM-0430-001 Rev. 37
2OM-53C.4.2.30.1 Rev. 9 pgs. 1, 2
2OM-30.4.AAB Rev. 4 pg. 3

Question Source: Bank – 2LOT6 Q52

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

Objective: 2SQS-30.1, Rev. 23 Obj. 17. Given a specific plant condition, predict the response of the Service Water 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.

2SQS-53C.1, Rev. 11 Obj. 5 Given a set of conditions, apply the correct AOP.

53. The following conditions exist:

- The plant was at 100% power when the Reactor was manually tripped due to a loss of all Station Instrument Air
- Station Instrument Air Header pressure is 0 psig
- AFW throttle valves were throttled to minimize cooldown
- No other actions have been taken by the crew

What will be the approximate equilibrium Tavg value when the plant stabilizes from this event?

A. 541°F

B. 547°F

C. 551°F

D. 556°F

Answer: C

Explanation/Justification: K/A is met by demonstrating the knowledge of how the pneumatic valves which control RCS temperature after a reactor trip will respond when there is a loss of the instrument air system. Specifically, the MSIVs and condenser steam dumps will fail closed on a loss of IAS, and RCS temperature will be maintained by the SG Atmospheric Steam Dump Valves.

A. Incorrect. This is the temperature the condenser steam dumps will close due to P-12 (low-low Tavg). Plausible if the candidate does not recognize that the MSIVs go shut, and the condenser steam dumps will also fail closed on a complete loss of instrument air.

B. Incorrect. This is the temperature the Rx trip controller will maintain with the condenser steam dumps. Plausible if the candidate does not recognize that the MSIVs go shut, and the condenser steam dumps will also fail closed on a complete loss of instrument air.

C. Correct. The Steam Generator Atmospheric Steam Dump Valves [PCV-1MS-101A, B, & C] are electro-hydraulic operated and will operate to maintain 1040 psig (1055 psia) which is approx. 551F.

D. Incorrect. Plausible if the candidate thinks that the SG Atmospheric Steam Dump Valves are air operated also, and the SG safety valves would maintain temp by the lifting of the first at 1075 psig (~1090 psia) which is approx. 556F.

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:** 2OM 53C.4.2.34.1 Rev. 21 pg. 22; 2OM-21.1.D Rev. 1 pg.5, 6; 2OM-21.2.B Rev. 7 pg. 4

Question Source: Bank – 1LOT16 Q54

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

Objective: 2SQS-34.1, Rev. 18 Obj. 15. Given a Unit 2 Compressed Air System configuration and without referenced material, describe the Compressed Air 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

2SQS-21.1, Rev. 23 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

54. Given the following plant conditions:

- The reactor was tripped due to a SG tube rupture.
- The crew has transitioned from E-3 to ECA-3.1 due to RCS subcooling concerns.
- RCS Cooldown to Mode 5 has commenced.
- The crew is currently at step 16 of ECA-3.1, "Check if subcooled recovery is appropriate".
- RWST level, 2QSS-LR100 indicates 440 inches.
- CNMT Sump level, 2RSS-LR151 indicates 35 inches.

1) What is the MAXIMUM allowable RCS Cooldown rate in ECA-3.1?

2) Based on EOP Attachment A-4.8, CNMT Sump Level Versus RWST Level **(References Provided)**, is there sufficient water level in the CNMT Sump to continue in ECA-3.1?

E-3, Steam Generator Tube Rupture
ECA-3.1, SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired
ECA-3.2, SGTR with Loss of Reactor Coolant – Saturated Recovery Desired

A. 1) Cooldown at < 100°F/hour.
   2) Yes, remain in ECA-3.1.

B. 1) Cooldown at < 100°F/hour.
   2) No, transition to ECA-3.2.

C. 1) Cooldown at maximum rate.
   2) Yes, remain in ECA-3.1.

D. 1) Cooldown at maximum rate.
   2) No, transition to ECA-3.2.

Question 54

Answer: A

Explanation/Justification: K/A is met with the candidate's ability to interpret the EOP attachment for containment sump level versus RWST level when determining if the water from the RWST is recoverable to assist in the plants ability to continue RCS cooldown while maintaining subcooling.

A. Correct. Cooldown at < 100°F/hour is correct per ECA-3.1 Major Action Category step 1, and ECA-3.1 step 15. Referring to EOP Attachment A-4.8 and evaluating RWST level of 440 inches with CNMT sump level at 35 inches, the candidate should determine that the sump is in the expected region for continued use of ECA-3.1 per step 16.

B. Incorrect. First part is correct. Transition to ECA-3.2 is plausible if the candidate misinterprets EOP attachment A-4.8 by either RWST or CNMT Sump levels or doesn't understand the reason for transitioning to ECA-3.2 when in ECA-3.1.

C. Incorrect. Max cooldown rate is plausible because it is the cooldown rate in E-3 at step 6, and the candidate may think it is the same in ECA-3.1. Second part is correct.

D. Incorrect. Max cooldown rate is plausible because it is the cooldown rate in E-3 at step 6, and the candidate may think it is the same in ECA-3.1. Transition to ECA-3.2 is plausible if the candidate misinterprets EOP attachment A-4.8 by either RWST or CNMT Sump levels or doesn't understand the reason for transitioning to ECA-3.2 when in ECA-3.1.

Sys #	System	Category	KA Statement
103	Containment	Generic	Ability to interpret reference materials, such as graphs, curves, tables, etc.

K/A# 2.1.25 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate 2OM-53.A.1.ECA-3.1 Iss. 3 Rev. 1 pg. 15; 2OM-53A.1.A-4.8 Iss. 1C Rev. 2

Technical References: 2OM-53B.4.ECA-3.1 Iss. 3 Rev. 1 pg. 9; 2OM-53.A.1.ECA-3.1 Iss. 3 Rev. 1 pg. 15; 2OM-53A.1.A-4.8 Iss. 1C Rev. 2

Question Source: New

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.

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.

55. Given the following plant conditions:

- The Plant is at 100% power.

Which of the following containment conditions and/or malfunctions results in a **one hour or less** technical specification required action?

A. A single Quench Spray Train is inoperable.

B. One containment isolation valve is inoperable.

C. Containment Average Air Temperature is 109 °F.

D. One containment air lock door is inoperable.

Answer: D

Explanation/Justification: K/A is met by the candidate having the knowledge of containment systems tech specs with a one hour or less entry condition which would cause a loss of containment integrity while in Mode 1.

A. Incorrect. Per TS 3.6,6, Two QS trains shall be operable. The TS action to restore one inoperable QS train is 72 hours

B. Incorrect. Per TS 3.6.3, in Mode 1 Each containment isolation valve shall be operable. The TS action for this condition is to isolate the affected penetration within 4 hours.

C. Incorrect. Per TS 3.6.5, In Mode 1 Containment Average Air temperature is to be maintained < 108 F. The TS action to restore temperature within limit is 8 hours.

D. Correct. Per TS 3.6.2, Two air locks shall be operable. With one airlock with a containment air lock door inoperable, the required action to verify the operable door is closed in the effected air lock is 1 hour. Per TS 3.6.1 basis for an operable containment, the isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. (ie; required for containment integrity)

Sys #	System	Category	KA Statement
103	Containment	K3 Knowledge of the effect that a loss or malfunction of the containment system will have on the following:	Loss of containment integrity under normal operations

K/A# K3.02 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** TS 3.6.2, Amend 278/161, pg. 3.6.2 – 1 & 2
TS 3.6.1 Bases, Rev. 0, pg. B3.6.1 - 1

Question Source: Bank – 1LOT8 Q54

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

Objective: 2SQS-47.1-01-05: From memory and for a given set of plant conditions, determine if the given condition meets the criteria for entry into a less than one hour action statement in accordance with the Technical Specifications.

56. The plant is at 100% power when an 'A' Steam Generator Safety Valve fails OPEN.

What effect will this failure have on 'A' RCS Loop ΔT Power, and why?

A. Loop power lowers because Thot lowers

B. Loop power lowers because Tcold lowers

C. Loop power rises because Thot rises

D. Loop power rises because Tcold lowers

Answer: D

Explanation/Justification: K/A is met by introducing an open SG Safety Valve while the plant is at 100% power, and the candidate explaining the effect the SG safety valve will have on RCS Loop ΔT Power by understanding that the additional steam load will lower Tcold.

A. Incorrect. Plausible because Thot will lower but not at the same magnitude as Tcold, therefore RCS Loop ΔT Power will rise because ΔT=Thavg-Tc so ΔT will increase.

B. Incorrect. Plausible because Tcold will lower, but RCS Loop ΔT Power will rise because ΔT=Thavg-Tc so ΔT will increase.

C. Incorrect. Plausible because RCS Loop ΔT Power will rise, but Thot actually lowers.

D. Correct. RCS Loop ΔT Power will rise because ΔT=Thavg-Tc and Thots will lower slightly, but Tcold lowers more due to the additional steam load will be seen on the outlet (cold leg) of the steam generator.

Sys #	System	Category	KA Statement
002	Reactor Coolant	K5 Knowledge of the operational implications of the following concepts as they apply to the RCS:	Relationship between effects of the primary coolant system and the secondary coolant system

K/A# K5.11 **K/A Importance** 4.0 **Exam Level** RO

References provided to Candidate None **Technical References:** 2SQS-6.5 PPNT Rev. 18 slide 12, 35

Question Source: New

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

Objective: GO-3ATA 4.1, Rev. 5 Obj. 2. Explain plant response to the below listed accidents: d. Steam Break (Major and Minor)

2SQS-6.5, Rev. 18 Obj. 7. Given a change in plant conditions, describe the response of the Reactor Coolant System field indication and control loops, including all automatic functions and changes in equipment status.

57. Given the following conditions:

- The plant is operating at 100% power.
- The PRZR heater control switches are positioned as follows:

Group A:	Auto-After Stop
Group B:	Auto-After-Stop
Group C:	On
Group D:	Auto-After-Start
Group E:	Auto-After-Stop

A loss of 480VAC emergency bus 2N then occurs.

Which of the following actions must be taken to manually energize PRZR backup heaters in response to the loss of the 2N bus?

A. Place Group A or D in Auto-After-Start.

B. Place Group B or E in Auto-After-Start.

C. Reset Group C and return to ON.

D. Reset Group D and return to Auto-After-Start.

Answer: B

Explanation/Justification: K/A is met with the knowledge that przr heater groups A & D will lose power on the loss of the 2N bus, and that the control switches for groups B & E must be taken to the Auto-After-Start position to energize these heaters from their normal power source of bus 2P.

A. Incorrect. Plausible if the candidate doesn't know that bus 2N supplies these heaters.
B. Correct. Since B/U groups A & D are deenergized, B/U groups B & E will have to be manually energized from bus 2P by placing the control switches for groups B & E to the Auto-After-Start position to energize these heaters.
C. Incorrect. Plausible if the candidate thinks that group C heaters are deenergized and must be reset to energize from an alternate power source, but this is incorrect because the control heaters are powered from bus 2C which remain energized. Bus 2C could be automatically energized from bus 2D in the event of a loss of power to bus 2C.
D. Incorrect. Plausible because group D heater is the only control switch in Auto-After-Start, and may have to be reset after a power loss, but group D heaters are powered from bus 2N.

Sys #	System	Category	KA Statement
011	Pressurizer Level Control	K2 Knowledge of bus power supplies to the following:	PZR heaters

K/A# K2.02 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-6.3.C Rev 16 pgs. 33-36

Question Source: Bank – 1LOT7 NRC Exam Q36

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

Objective: 2SQS-6.4, Rev. 17 Obj. 19. Given a Pressurizer and Pressurizer Relief System configuration and without referenced material, describe the Pressurizer and Pressurizer Relief System control room response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable. b. Loss of electrical power

58. A Reactor startup is in progress IAW 2OM-50.4.D2, Reactor Startup From Mode 3 To Mode 2.

- Bank 'D' control rods are currently at 16 steps to record 1/M data.
- Estimated Critical position is 100 steps on Bank 'D'.
- Unit Supervisor has directed the RO to continue withdrawing control rods to achieve criticality.

Based on current conditions, IAW 2OM-50.4.D2, a maximum startup rate of _____ is permitted when withdrawing control rods to establish criticality.

A. 0.1 dpm

B. 0.3 dpm

C. 0.5 dpm

D. 1.0 dpm

Answer: C

Explanation/Justification: K/A is met as it requires the candidate to process known procedural requirements for SUR that are followed to prevent an inadvertent SR reactor trip when withdrawing control rods..

A. Incorrect. Plausible because < 0.1 dpm SUR must be maintained during the transition through the POAH iaw 2OM-50.4.D2 P&L 14.
B. Incorrect. Plausible because < 0.3 dpm SUR is identified on attachment 6 "Source Range Checks" of 2OM-50.4.D which is directed by the RO continuous actions when both IR Channels indicate less than 1E-10 AMPS (P6) and the required range is 0.3 dpm.
C. Correct. P&L 14 states maintain the startup rate less than a sustained < 0.5 dpm at all times, unless transition through the POAH (5E-7 amps to 2% power) which has a sustained SUR of 0.1 dpm.
D. Incorrect. Plausible because 1 dpm is possible when the scale for SUR is -1.5 to 5 dpm.

Sys #	System	Category	KA Statement
015	Nuclear Instrumentation	A1 Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the NIS controls including:	SUR

K/A# A1.02 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-50.4.D2 Rev. 5 pg. 26 P&L 14.

Question Source: New

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

Objective: 3LOT-M4D1, Rev. 5 Obj. 1. Explain Precautions and Limitations, Reactor Theory and Kinetics applicable to the startup, in accordance with 2OM-50.4.D2, Reactor Startup from Mode 3 to Mode 2 and BVPS Reactor Theory Manual.

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

- PRZR level control is in automatic controlling PRZR level on program.
- Loop 'A' Tavg is 571°F.
- Loop 'B' Tavg is 568°F.
- Loop 'C' Tavg is 567°F.

Subsequently:

- Loop 'B' Tavg Channel fails to 600°F.

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 Tavg failure and the affect it will have on the input to the Pressurizer level control system to maintain the przr program level setpoint.

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:	Automatic selection of NNIS inputs to control systems

K/A# A3.01 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-6.4.IF Rev 13 pg. 12; 2SQS-6.5 PPNT Rev 18 slide 35, 36

Question Source: New

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

Objective: 2SQS-6.5, Rev. 18 Obj. 22. 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

60. The plant is operating at 100% power when the following takes place:

- Annunciator A6-1B, Spent Fuel Pool Level High/Low ALARMS.
- RO reports SFP level is 170" and lowering slowly.
- Spent Fuel Pool Level High/Low ARP and AOP 2.20.1, Spent Fuel Pool Cooling Trouble have been entered.
- PAB Operator confirms low Spent Fuel Pool Level using the local level indicator and reports SFP water leaking from 'A' Fuel Pool Hx outlet piping.

1) Without Operator action, what is the lowest level the Spent Fuel Pool level will reach?
2) IAW AOP 2.20.1, what is the preferred makeup water to restore the Fuel Pool level?

A. 1) 10 feet above the top of the fuel assemblies.
   2) Fire Protection Hose Racks

B. 1) 23 feet above the top of the fuel assemblies.
   2) Fire Protection Hose Racks

C. 1) 10 feet above the top of the fuel assemblies.
   2) Demin Water

D. 1) 23 feet above the top of the fuel assemblies.
   2) Demin Water

Answer: C

Explanation/Justification: K/A is met by the candidate determining that a leak on the FP Cooling Hx discharge will only drain the fuel pool down to a level no lower than 10 feet above the fuel assemblies due to the piping system design. This provides sufficient shield to allow repair of the ruptured line. Then, based on the candidate's knowledge of the AOP, they recognize that Demin Water is the preferred makeup source to correct the FP low level.

A. Incorrect. First part is correct. Second part is plausible because it is a makeup source to the Fuel Pool, but the preferred method per the AOP is Demin Water because only one valve must be opened, and no pumps need to be operating.

B. Incorrect. Plausible if the candidate confuses the fuel pool tech spec limit with the fuel pool design features. Second part is plausible because it is a makeup source to the Fuel Pool, but the preferred method per the AOP is Demin Water because only one valve must be opened, and no pumps need to be operating.

C. Correct. The spent fuel pool is designed such that the water level in the pool cannot be decreased below 10 feet above the top of the fuel stored in the spent fuel racks because all piping and piping penetrations of the spent fuel pool terminate no lower than 10 feet above the top of the fuel stored in the racks. The preferred method as identified in the AOP for makeup water to the Fuel Pool is the Demin Water supply.

D. Incorrect. Plausible if the candidate confuses the fuel pool tech spec limit with the fuel pool design features. Demin Water is the preferred makeup per the AOP.

Sys #	System	Category	KA Statement
033	Spent Fuel Pool Cooling	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Abnormal spent fuel pool water level or loss of water level

K/A# A2.03 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-20.1.B Rev. 2 pg. 4
2OM-53C.4.2.20.1 Rev. 4 pg. 7

Question Source: New

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

Objective: 2SQS-53C.1, Rev. 11 Obj. 5. Given a set of conditions, apply the correct AOP.
2SQS-20.1, Rev. 13 1. Describe the function of the Fuel Pool Cooling and Purification system and the associated major components as documented in Chapter 20 of the Operating Manual.

61. Given the following plant conditions:

- The plant is in Mode 6 during a refueling outage.
- The plant has been shutdown for 6 days.
- The Containment Equipment Hatch is closed.
- Fuel Movement is in progress.
- Containment Purge is in operation.
- **NO** features have been defeated (Lock out jacks are installed).
- 2HVR*RQ104B, Containment Purge Radiation Monitor fails upscale **HIGH**.
- 2HVR*RQ104A, Containment Purge Radiation Monitor is unaffected.
- All systems function as designed.
- No operator action has occurred.

What will be the impact on Containment Purge AND fuel movement?

Containment Purge will _____(1)_____ AND fuel movement _____(2)_____.

A. (1) automatically isolate
(2) may continue

B. (1) automatically isolate
(2) must be immediately suspended

C. (1) be unaffected and will require manual isolation
(2) must be immediately suspended

D. (1) be unaffected and will require manual isolation
(2) may continue

Question 61

Answer: A

Explanation/Justification: The K/A is met because the malfunctioning radiation monitor results in a loss of containment purge which could have an impact on fuel handling operations.

A. Correct. 2HVR*RQ104B radiation monitor upscale failure high will cause 2HVR-MOD23B & 25B (Cnmt purge suppl and exhaust dampers) to close isolating cnmt purge. This is the only radiation monitor which has any automatic functions associated with fuel handling, there are no direct fuel handling / radiation monitor interlocks at Beaver Valley. The stem of the question states that Containment Purge is in operation and no auto functions have been defeated. This is to alleviate any confusion based on procedure flexibility which allows auto isolation features to be defeated in the plant refueling procedures. TS 3.9.3 requires the Containment Purge to be capable of automatic isolation or isolated, only when moving recently irradiated fuel which BV does not currently do. Since TS 3.9.3 is not applicable, fuel movement may continue.

B. Incorrect. First part is correct because containment purge auto isolates. Second part is plausible since TS 3.9.3 requires the Containment Purge to be capable of automatic isolation or isolated, only when moving recently irradiated fuel which BV does not currently do. Since TS 3.9.3 is not applicable, fuel movement may continue.

C. Incorrect. Plausible if the candidate believes the logic is 2/2 for containment purge to isolate and that manual isolation is required due to impact on fuel handling operations. Second part is plausible since TS 3.9.3 requires fuel movement to be suspended if containment purge is isolated, but only IF we were moving recently irradiated fuel which BV does not currently do.

D. Incorrect. Plausible Refer to C part 1. Since TS 3.9.3 is not applicable as described above, fuel movement may continue.

Sys #	System	Category	KA Statement
034	Fuel-Handling Equipment	K6 Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling System:	Radiation monitoring systems

K/A# K6.02 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None

Technical References: 2OM-43.1.C, Rev. 5, pg 17
2OM-43.5.B.3 Rev. 2 pg. 2
TS 3.9.3 pg. 3.9.3-1 & bases pg. B3.9.3-3

Question Source: Modified – 2LOT8 Q62

Question Cognitive Level: Lower – Memory or Fundamental

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

Objective: 2SQS-44C.1, Rev. 12 Obj. 10. Describe the control, protection and interlock functions for the control room components associated with the Containment Ventilation System, including automatic functions, setpoints and changes in equipment status as applicable.

3SQS-REFUEL ITS, Rev. 1 Obj. 5. From memory, identify a condition concerning the Refueling System Technical Specifications and Licensing Requirements that requires Tech Spec action to be taken in less than or equal to one hour.

62. A leak occurs on a Gaseous Waste Storage Tank (GWST) causing a HIGH radiation alarm on 2RMQ-RQI303, Waste Gas Storage Vault radiation monitor during a GWST discharge. All equipment functioned as designed.

Which of the following will occur because of this HIGH radiation alarm?

A. Decontamination Building Filtered Exhaust fan 2HVQ-FN214A will START and the Gaseous Waste Storage Tank Inlet and Outlet Header Isolation valves will CLOSE.

B. Decontamination Building Normal Exhaust fan 2HVQ-FN214B will be TRIPPED **and** Decontamination Building Filtered Exhaust fan 2HVQ-FN214A will START.

C. Both Decontamination Building Normal and Filtered Exhaust fans 2HVQ-FN214A & B will be TRIPPED.

D. Only the Gaseous Waste Storage Tank Inlet and Outlet Header Isolation valves will CLOSE.

Answer: C

Explanation/Justification: K/A is met by introducing a leak on a Gaseous Waste Storage Tank during a discharge and having the candidate demonstrate their knowledge of the effect that a HIGH radiation monitor alarm will have on the Waste Gas Storage Vault ventilation system.

A. Incorrect. Plausible because on a HIGH radiation alarm it would be expected that a Filtered train of ventilation would align, and the source of the discharge process would be isolated, but both assumptions are incorrect.
B. Incorrect. Plausible because this would be the ventilation lineup if 2RMQ-RQI303 was in ALERT. 2RMQ-RQI303 Alert rad monitor alarm will trip Decontamination Building Normal Exhaust fan 2HVQ-FN214B and start Filtered Exhaust fan 2HVQ-FN214A.
C. Correct. 2RMQ-RQI303 Alert rad monitor alarm will trip Decontamination Building Normal Exhaust fan 2HVQ-FN214B and start Filtered Exhaust fan 2HVQ-FN214A. When the HIGH radiation alarm comes in, it causes the Filtered Exhaust fan 2HVQ-FN214A to trip also.
D. Incorrect. Plausible since there is a discharge in progress, but the candidate doesn't know that 2RMQ-RQI303 monitors the gas and particulate in the Waste Gas Storage Vault, and not the Gaseous Waste discharge process flowpath.

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:** 2OM-43.1.E Rev. 6 pg. 37
2SQS-44B.1 Rev. 9 slides 34, 36
2OM-43.4.ADU Rev.5 pg. 2, 3

Question Source: New

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

Objective: 2SQS-44B.1, Rev. 9 Obj. 3. Describe the control, protection and interlock functions for the field components associated with the Process Cooling System, including automatic functions, setpoints and changes in equipment status as applicable.

63. The plant is operating at 100% power with all systems in NSA.

- Unit 2 Control Room Area Radiation Monitor 2RMC*RQ201 is at background level.
- Testing of the Unit 2 Control Room Area Radiation Monitor 2RMC*RQ202 is in progress IAW 1/2OST-43.17D, Control Room Area Monitor 2RMC*RQ202 Test.
- Per the instructions of 1/2OST-43.17D, a **HIGH** alarm signal is developed for the Unit 2 Control Room Area Radiation Monitor 2RMC*RQ202.

What will be the status of the Control Room Emergency Ventilation Fans after the **HIGH** alarm signal is developed?

A. 2HVC*FN241B will start IN 90 SECONDS, 2HVC*FN241A will REMAIN IN STANDBY.

B. 2HVC*FN241A will start IN 120 SECONDS, 2HVC*FN241B will REMAIN IN STANDBY.

C. 2HVC*FN241B will start IN 120 SECONDS, 2HVC*FN241A will REMAIN IN STANDBY.

D. 2HVC*FN241A will start IN 90 SECONDS, 2HVC*FN241B will start 30 SECONDS LATER (if 2HVC*FN241A flow is low).

Answer: C

Explanation/Justification: K/A is met by the candidate's ability to determine how the Control Room Emergency Ventilation Fans will respond when only 2RMC*RQ202 is manually given a High Radiation signal during testing of the Control Room Area Monitor High alarm testing.

A. Incorrect. Plausible because FN241A starts after 90 seconds and FN241B would remain in standby if 2RMC*RQ201 is in high alarm. FN241B would start 30 seconds later if FN241A failed to start.
B. Incorrect. Plausible because this would be the status if 2RMC*RQ201 were placed in a high alarm condition, however the fan would start in 90 seconds not 120.
C. Correct. Fan 2HVC*FN241B will start approximately 120 seconds after the High radiation signal actuates on 2RMC*RQ202. 2HVC*FN241A will remain in standby since 2RMC*RQ201 is at background level.
D. Incorrect. Plausible because this is what happens when 2RMC*RQ201 is in high alarm and FN241A fails to start.

Sys #	System	Category	KA Statement
072	Area Radiation Monitoring	A4 Ability to manually operate and/or monitor in the control room:	Alarm and interlock setpoint checks and adjustments

K/A# A4.01 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** 1/2OST-43.17D, Rev. 48 pg. 5
2OM-43.1.E Rev. 6 pg. 23, 24
2SQS-43.1 Rev. 11 PPNT slides 86, 87

Question Source: Bank – 2LOT8 Audit Exam Q52

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

Objective: 2SQS-43.1, Rev. 11 Obj. 7. 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.

2SQS-44A.1, Rev. 6 Obj. 12. Describe the control, protection and interlock functions for the Control Room components associated with the Unit-2 Control Area Ventilation System, including automatic functions, setpoints and changes in equipment status as applicable.

64. Which of the following describes the operation of 2SAS-AOV105, Station Air Header to Service Air Header Isolation Valve?

A. Automatically closes at 86 psig decreasing; automatically reopens at 100 psig increasing.

B. Automatically closes at 90 psig decreasing; automatically reopens at 100 psig increasing.

C. Automatically closes at 86 psig decreasing; must be manually reset and reopened.

D. Automatically closes at 90 psig decreasing; must be manually reset and reopened.

Answer: C

Explanation/Justification:

A. Incorrect. 86psig is when 2SAS-AOV105 auto closes. The valve does not automatically reopen at any pressure, but this pressure is plausible because it is when the SACs load.
B. Incorrect. Plausible distractor because 90 psig is when 2SAS-C22, Condensate Polishing compressor auto starts. The valve does not automatically reopen at any pressure, but this pressure is plausible because it is when the SACs load.
C. Correct. 2SAS-AOV105 auto closes when pressure in the station instrument air receiver reaches 86 psig, decreasing, diverting air from service connections to the more important instrument air header. The isolation valve closes automatically on low pressure but must be manually reopened to restore air iaw .2OM-34.4.A section I.
D. Incorrect. Plausible distractor because 90 psig is when 2SAS-C22, Condensate Polishing compressor auto starts. The valve must be manually reset and reopened.

Sys #	System	Category	KA Statement
079	Station Air	K4 Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following:	Cross-connect with IAS

K/A# K4.01 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-34.1.D rev. 5 pg. 5
2OM-34.4.A Rev. 15 pg. 24

Question Source: Bank – 1LOT7 Q38

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

Objective: 2SQS-34.1, Rev. 18 Obj. 13. Describe the control, protection and interlock functions for the control room components associated with the various Unit 2 Compressed Air Systems, including automatic functions, setpoints and changes in equipment status as applicable.

65. Given the following conditions:

- The plant is in Mode 3.
- A11-6C, Process Rack Room Incipient Fire is LIT.
- The Simplex TrueSite Workstation indicates smoke in the Process Rack Room.

Which of the following annunciators will be LIT if an actual fire exists AND the fire protection system actuates as designed?

(Not all expected annunciators are listed.)

A. A11-5A, CO2 FIRE PROTECTION SYSTEM NO. 1 MASTER VLV OPEN

B. A11-5B, CO2 FIRE PROTECTION SYSTEM NO. 2 MASTER VLV OPEN

C. A11-5E, COMMUNICATION ROOM FIRE/HALON FLOW

D. A11-5D, COMPUTER ROOM FIRE/HALON FLOW

Answer: B

Explanation/Justification: K/A is met by the candidate recognizing that the Control Room incipient fire detection system (Simplex TrueSite Workstation – only a detection system) will alert the operators to a possible fire in the process rack room, and that confirmation of an actual fire will be made by the specific fire protection actuation indication annunciator in the control room indicating CO2 system 2 has actuated.

A. Incorrect. Plausible because CO2 is used in the Process Rack Room, but CO2 system 1 provides fire protection for the turbine generator area and acts as a source for purging the MUG. The system 1 7.5-ton unit cannot be cross tied to system 2 which provides fire protection for the Process Rack Room.

B. Correct. A fire in the Process Rack will cause CO2 to be discharged in the Control Building 707' Process Rack Room 60 seconds after the smoke detectors actuate. Annunciator A11-5B, CO2 Fire Protection System No. 2 Master Valve Open will be lit due to the smoke detectors actuating causing the master valves to open on the two 10-ton CO2 units (System 2 CO2).

C. Incorrect. Plausible because Halon is used in areas protecting electrical equipment, and even in Unit 1 Process Rack Room. Halon is used in the Control Building West Communications Room which is located adjacent to the Process Rack Room, therefore the candidate must have knowledge of the room separation and the suppression agents applicable to each.

D. Incorrect. Plausible because Halon is used in areas protecting electrical equipment, and even in Unit 1 Process Rack Room. Halon is used in the Control Building Computer Room, but this room is located at the rear of the Control Room. The candidate must have knowledge of the room locations and the suppression agents applicable to each.

Sys #	System	Category	KA Statement
086	Fire Protection	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** 4.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 2OM-33.4.ACI Rev. 3 pg. 3; 2OM-33.4.ACS Rev. 3 pg. 3; 3SQS-33.1 PPNT Rev. 10 Slide 69

Question Source: New

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

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-33.1-1-04: Given a change in plant conditions, describe the response of the Fire Protection System field indication and control loops, including all automatic functions and changes in equipment status.

66. The plant is in Mode 5 preparing to enter Mode 4.

- Valve alignments are being performed on a Safety-Related system.
- The **REQUIRED** NSA position of a manually operated globe valve is **2 Turns OPEN**.
- The valve must be in this position **PRIOR** to Mode 4 entry.
- The valve has MINIMAL safety significance.
- The valve list **REQUIRES** Concurrent verification for this valve.
- The second verifier will receive 5 mR performing the Concurrent verification.
- The valve has **NO** remote valve indication.
- The valve **CANNOT** be verified in the correct position by the performance of a functional test.

IAW the guidance provided in NOP-OP-1002, Conduct of Operations, how will the Concurrent verification for this valve be addressed?

A. The Shift Manager shall waive the Concurrent verification for this valve based on **MINIMAL** safety significance and **HIGH** radiation exposure to the second verifier.

B. The First verifier places the valve in the required position; **WITHIN** 4 hours the second verifier verifies the valve in the required position.

C. The First verifier places the valve in the required position **WHILE** the second verifier observes the first verifier placing the valve in the required position.

D. The First verifier places the valve in the required position; the second verifier remains **OUTSIDE** the line of sight of the first verifier **THEN** verifies the valve in the required position.

Answer: C

Explanation/Justification: K/A is met by demonstrating the required knowledge on how the operator will perform a concurrent verification on a throttled valve during a valve lineup.

A. Incorrect. Plausible distractor because the SM can waive IV/CV if the verification would result in exposure > 10 mrem.

B. Incorrect. Plausible distractor because these are the requirements for independent verification of Tech Spec related actions that support current plant conditions. Since this valve is required for Mode 4 entry, it is NOT required for the current plant Mode.

C. Correct. In accordance with NOP-OP-1002 and NOBP-LP-2601, Human Performance Program, the performer executes the correct action on the correct component, and the verifier observes the performer before and during execution to confirm the performer takes the correct action on the correct component.

D. Incorrect. Plausible distractor because these are the requirements for independent verification NOT concurrent verification. Additionally, this valve must be concurrently verified since independent verification would negate the original condition

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Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

K/A# 2.1.29 **K/A Importance** 4.1 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-OP-1002 Rev. 13 pg. 81-82; NOBP-LP-2601 Rev. 13 pg. 18, 19

Question Source: Bank – 2LOT6 NRC Exam Q66

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

Objective: 3SQS-48.1, Rev. 24 Obj. 13. From memory, explain the control of valves and plant equipment including definitions of the associated terms for the following: a. valve and equipment operation

67. The plant is at 100% power.

IAW NOP-OP-1002, Conduct of Operations, which of the following lists the UNIT 2 **NORMAL** shift staffing requirements and the **MAXIMUM** time allowed to fill a position due to an **UNEXPECTED** absence?

		# Senior Reactor Operator(s)	# Reactor Operator(s)	Maximum Time to Fill the Position
A.	Condition 1	1	1	1 hour
B.	Condition 2	2	2	1 hour
C.	Condition 3	1	2	2 hour
D.	Condition 4	2	2	2 hour

Answer: D

Explanation/Justification: K/A is met by demonstrating the minimum crew complement required for Mode 1, and the allowable time to replace a crew member due to an unexpected absence.

A. Incorrect. Plausible because 1 SRO and 1 RO are the minimum manning for safe shutdown modes 5,6, and defueled. 1 hr limit is plausible because of the importance of a vacant crew position.
B. Incorrect. Per NOP, 2 SROs and 2 ROs is correct. Second part is incorrect but 1 hr limit is plausible because of the importance of a vacant crew position.
C. Incorrect. Plausible because the SM and US SRO positions are listed separately on attachment 4 and could be mistaken for only 1 SRO required in mode 1. Second part is correct, a 2 hr limit is allowed to fill a vacancy.
D. Correct. Per NOP-OP-1002 attachment 4, in Modes 1-4, 2 SROs and 2Ros are required for minimum staffing. Section 4.1.13.2 states Shift crew composition may be one less than the minimum requirements for a period of time not to exceed-two hours in order to accommodate unexpected absence of on-duty shift crew members.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

K/A# 2.1.5 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-OP-1002 Rev 13 Att 4 pg 98 & Section 4.1.13.2 pg 22

Question Source: Bank – 1LOT14 Q66

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

Objective: 3SQS-48.1, Rev. 24 Obj. 2. From memory, state the operational modes, and the shift staffing requirements during all modes of operation, including overtime limitations.

68. A plant transient has occurred which has required the Unit Supervisor to enter an Abnormal Operating Procedure (AOP).

While performing the AOP, what are the expectations for peer checking panel manipulations and place keeping IAW NOP-OP-1002, Conduct of Operations?

1) Peer checks are _____.
2) Place keeping _____.

A. 1) are required and may be performed by all control room staff.
2) is required during transients.

B. 1) not required during transients.
2) may be waived during transients.

C. 1) are required and may be performed by all control room staff.
2) may be waived during transients.

D. 1) not required during transients.
2) is required during transients.

Answer: D

Explanation/Justification: K/A is met by the candidate's ability to manage their control room responsibilities during a plant transient in regards to peer checking and place keeping.

A. Incorrect. First part is plausible because peer checks are vital to plant control, but during transient periods, actions may be taken without waiting for a peer check, and it is also incorrect because SROs should not be used for peer checking component manipulations. Second part is correct.
B. First part is correct. Second part is plausible because the candidate may think that place keeping during a transient has minimal value for plant control, but place keeping shall be used during plant transients.
C. First part is plausible because peer checks are vital to plant control, but during transient periods, actions may be taken without waiting for a peer check, and it is also incorrect because SROs should not be used for peer checking component manipulations. Second part is plausible because the candidate may think that place keeping during a transient has minimal value for plant control, but place keeping shall be used during plant transients.
D. Correct. Peer checks are not required for panel manipulation iaw section 4.10.6 of Conduct of Operation. Place keeping shall be used during plant transients iaw section 4.10.7 of Conduct of Operations.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to manage the control room crew during plant transients.

K/A# 2.1.6 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-OP-1002 Rev. 13 pg. 64, 65

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. 20. From memory, explain all of the Operations Expectations.

69. In accordance with NOP-OP-1001, Clearance/Tagging Program, a component clearance will require double isolation if the **MINIMUM** fluid temperature or pressure are greater than _____.

(Select the answer that meets BOTH limits)

A. 150°F, 300 psig

B. 150°F, 500 psig

C. 200°F, 300 psig

D. 200°F, 500 psig

Answer: D

Explanation/Justification: K/A is met by demonstrating the knowledge of the site tagging procedure which requires double valve isolation for systems with >500 psig or 200F.

A. Incorrect. Plausible arbitrary values for temperature and pressure.
B. Incorrect. Plausible arbitrary values for temperature and pressure.
C. Incorrect. Plausible arbitrary values for temperature and pressure.
D. Correct. Sect. 3.11 of NOP-OP-1001 states Double Isolation Required if Fluid or gas systems with temperature greater than 200°F (93°C) or pressure greater than 500 psig (35 bar), or involving noxious chemicals, the work area should be isolated using double isolation boundary protection (two closed valves in series), with an open tell-tale vent or drain valve between the double isolation valves.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of tagging and clearance procedures

K/A# 2.2.13 **K/A Importance** 4.1 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-OP-1001 Rev. 27 pg. 7

Question Source: Modified – 1LOT18 Q68

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

Objective: GEN-TAGOVERVIEW_FEN-01 Obj. J. Discuss tagging of the following components: - Control switches, Fuses, Circuit Breakers, Mechanical Systems.

70. You are performing a procedure out in the plant and you note a typographical error in the step you are about to perform. In accordance with NOP-LP-2601, 'Procedure/Work Instruction Use and Adherence', what action are you required to perform.

A. Have a second qualified Operator peer check the typographical error, continue with the activity, and inform your supervisor upon completion.

B. Contact your supervisor, identify the typographical error, have the supervisor annotate the issue in the procedure, and then continue with the activity.

C. Contact your supervisor, identify the typographical error, and perform a Limited Use Change.

D. Contact your supervisor, identify the typographical error, and Revise the procedure.

Answer: B

Explanation/Justification: K/A is met with the knowledge of the expected response when a procedure is found to have a typographical error, and how to make the necessary changes to the procedure prior to completing work.

A. Incorrect. By continuing on in the procedure without discussing it with the authorizing authority or responsible supervisor would be a violation of NOP-OP-2601. A peer check by another qualified Operator does not meet the site expectations.
B. Correct. Per NOP-LP-2601, if a typo is discovered, the performer must stop the work, ensure equipment is in a safe condition, and contact they're supervisor. Clearly identify typo by annotating the procedure and then continue with the activity.
C. Incorrect. A Limited Use Change is not required for a typographical error
D. Incorrect. A procedure Revision is not required for a typographical error.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the process for making changes to procedures.

K/A# 2.2.6 **K/A Importance** 3.0 **Exam Level** RO

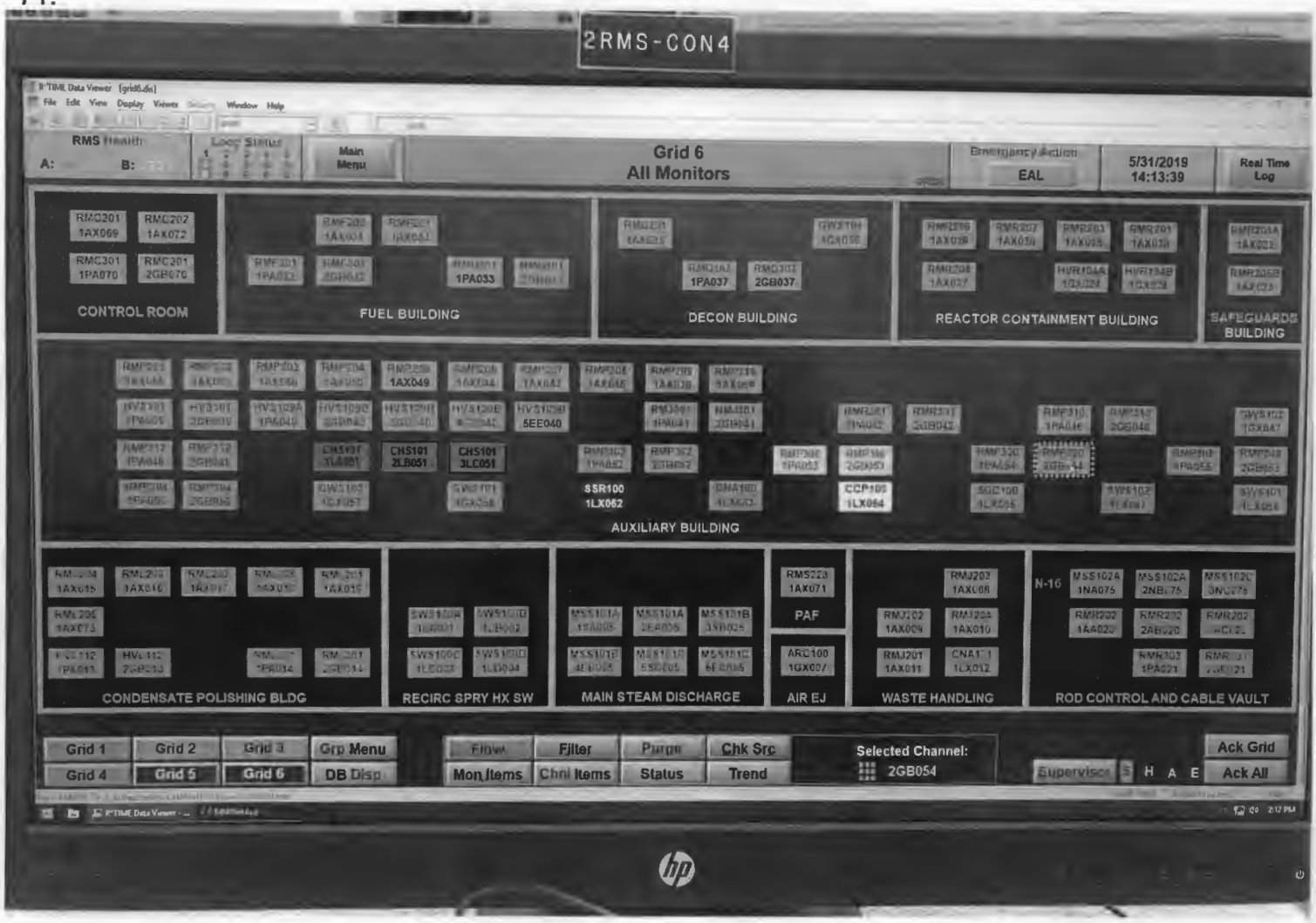
References provided to Candidate None **Technical References:** NOP-LP-2601 rev.6, pg.15 & 16

Question Source: Bank -2LOT15 Q68 **(Last 2 exams)**

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

Objective: 3SSG – Admin Rev. 9 Obj. 5. EXPLAIN the requirements for the use of plant procedures in accordance with NOP-LP-2601, Procedure Use And Adherence

71.



Based on the DRMS screen above, which of the following detectors has had a loss of sample flow to the detector?

A. 2SSR-RQI100, Steam Generator Blowdown Radiation Monitor

B. 2CHS-RQ101A, Reactor Coolant Letdown (Low Range) Radiation Monitor

C. 2RMP-RQI306, Auxiliary Building Ventilation Elev 735A Radiation Monitor

D. 2CCP-RQI100, Component Cooling Water Radiation Monitor

Question 71

Answer: A

Explanation/Justification: K/A is met with the candidate's ability to interpret the abnormal condition of the fixed radiation monitor based on the color code on the Digital Radiation Monitoring System (DRMS) computer screen.

A. Correct. 2SSR-RQI100 is DARK BLUE which indicates a system failure which includes LOSS OF SAMPLE FLOW.
B. Incorrect. Plausible if the candidate is not familiar with the DRMS color coding, but MAGENTA indicates that the monitor has had a COMMUNICATIONS FAILURE.
C. Incorrect. Plausible if the candidate is not familiar with the DRMS color coding, but YELLOW indicates that the monitor is in ALERT.
D. Incorrect. Plausible if the candidate is not familiar with the DRMS color coding, but WHITE indicates that the monitor is OFFLINE.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

K/A# 2.3.5 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate DRMS Picture **Technical References:** 2OM-43.1.D Rev 1 pg. 11
2OM-43.5.B.4 Iss 1 Rev 2 pg. 1

Question Source: New

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

Objective: 2SQS-43.1, Rev. 11 Obj. 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.

72. The following conditions exist at a job site:

- The general area radiation levels are 40 mr/hr.
- Radiation level with shielding is 10 mr/hr.
- Time for **one** worker to install AND remove shielding is 15 minutes.
- Time to conduct the task of **one** worker is 1 hour.
- Time to conduct the task with **two** workers is 20 minutes
- Total time to install AND remove is 15 minutes. Assume 40 mr/hr field for entire 15 minutes.

Assumptions:

- Shielding is installed and removed by one worker.

In order to comply with ALARA practices, which of the following will result in the **LOWEST TOTAL** dose for the job?

Conduct the task with _____

A. One worker with shielding.

B. Two workers with shielding.

C. One worker without shielding.

D. Two workers without shielding.

Answer: B

Explanation/Justification: K/A is met by demonstrating the ability to comply with radiation work permit requirements by implementing As Low As Reasonably Achievable (ALARA) work planning and controls for dose.

A. Incorrect. Dose to install shielding = 10 mr + 10mr/hr = 20 mr

B. Correct. Dose to install shielding = 10 mr + (.33)(10) = (3.3)(2) = 6.6 + 10 = 16.6 mr. In order to comply with radiation work permit requirements of maintaining dose as low as reasonably achievable, the lowest dose derived is by using a worker to install shielding and use two workers to perform the job with the shielding in place.

C. Incorrect. Dose with one worker without shielding is 40 mr x 1hr = 40 mr

D. Incorrect. Dose with two workers without shielding is 40 mr x (.33)(40) = (13.2)(2) = 26.4 mr

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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** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** FEN-RWT Rev 4 Chapter 2 pg. 4

Question Source: Bank – 2LOT7 Q71

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

Objective: 3SSG-Admin 16. Describe the controls for maintaining personnel exposures ALARA IAW NOP-OP-4107, Radiation Work Permit (RWP) FEN-RWT Rev 4 Chapter 5 Obj. 7. Calculate stay time given a dose rate, current dose, and a dose limit.

73. A grid disturbance results in the following indications:

- The plant is stable at 75% power
- 4KV Bus 2A to EMER Bus 2AE ACB 2A10 white light is lit
- 4KV EMER Bus 2AE Bus voltmeter indicates zero volts
- 4KV EMER Bus 2DF to 4KV Bus 2D ACB 2F7 white light is lit
- 2-2 EMER GEN Output Breaker ACB 2F10 white light is lit
- 4KV Bus 2A, 2B, 2C and 2D Volts indicate 122 volts

1) Per the EOP rules of Usage, which procedure will be directly entered based on these indications?

2) The Immediate Actions listed in certain EOPs require that the operator perform from memory _____.

A. 1) ECA-0.0, Loss Of All Emergency 4KV AC Power
2) only the left-hand column

B. 1) ECA-0.0, Loss Of All Emergency 4KV AC Power
2) both columns

C. 1) FR-S.1, Response to Nuclear Power Generation - ATWS
2) only the left-hand column

D. 1) FR-S.1, Response to Nuclear Power Generation - ATWS
2) both columns

Answer: B

Explanation/Justification: K/A is met with the candidate's knowledge that there are only two EOP procedures that may be used to enter the EOP Network IAW the EOP User's Guide, and that once entered, the CR Operators are required to perform both the Action/Expected Response Left column and the Response Not Obtained Right column.

A. Incorrect. First part is correct. Second part is incorrect that an immediate action only requires the operator to execute the left hand step from memory. It is plausible to believe because some of the immediate actions have response not obtained steps which are quite lengthy and difficult to remember.

B. Correct. In accordance with the EOP User's Guide sect. IV.B.1-3, the EOP Network may be entered directly by entering E-0, Reactor Trip or Safety Injection, and ECA-0.0, Loss of All AC Power. ECA-0.0 is a direct entry procedure if both Emergency busses are deenergized, the procedure may also be entered from E-0 if the busses are deenergized in step 3 IOA. Operators are required to perform the Action/Expected Response Left column, and if the action is not met, they are required to perform the Response Not Obtained Right column, both from memory.

C. Incorrect. Plausible since the plant is required to be tripped if power is lost to both emergency busses, and the indications are interpreted that a reactor trip has failed to occur.
Second part is incorrect that an immediate action only requires the operator to execute the left hand step from memory. It is plausible to believe because some of the immediate actions have response not obtained steps which are quite lengthy and difficult to remember.

D. Incorrect. Plausible since the plant is required to be tripped if power is lost to both emergency busses, and the indications are interpreted that a reactor trip has failed to occur. Second part is correct.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of EOP entry conditions and immediate action steps.

K/A# 2.4.1 **K/A Importance** 4.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1/2OM-53B.2 Rev. 9 pg. 6, 12, 13
NOP-OP-1002 Rev. 13 pg 60

Question Source: Bank – Watts Bar 2015 NRC Exam Q73

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

Objective: 3SQS-53.1, Rev. 2 Obj. 1. State from memory "All" of the Emergency Operating Procedures user's guide rules of usage as defined in 1/2OM53B.2.

74. The reactor has tripped, and Safety Injection has actuated due to a LOCA.

- The crew is implementing E-1, Loss of Reactor or Secondary Coolant
- The STA reports the following for Critical Safety Function Status Trees:
  Containment - Orange
  Subcriticality - Orange
  Heat Sink - Red
  Integrity - Red
  All others are Green

Which of the following identifies the required procedure transition AND what it is based on?

A. FR-P.1, Response to Imminent Pressurized Thermal Shock, based on a Severe Challenge to the RPV Integrity

B. FR-Z.1, Response to High Containment Pressure, based on a Severe Challenge to the Containment.

C. FR-S.2, Response to Loss of Core Shutdown, based on a Severe Challenge to the Subcriticality.

D. FR-H.1, Response to Loss of Secondary Heat Sink, based on a Severe Challenge to the Secondary Heat Sink.

Answer: D

Explanation/Justification: K/A is met by evaluating the candidate's knowledge of the CSFSTs priority based on the Safety Function affected, and the severity of the entry condition based on color while in the EOP network performing E-1, Loss of Reactor or Secondary Coolant.

A. Incorrect. Plausible because Integrity FR-I.1 is a Red Path, but a RED Path on Heat Sink takes priority over Integrity IAW the User's Guide.

B. Incorrect. Plausible because Containment FR-Z.1 Orange Path is a high priority, but FR-Z.1 Orange Path does not take priority over the Heat Sink or Integrity Red paths IAW the User's Guide.

C. Incorrect. Plausible because Subcriticality FR-S.1 is the highest priority tree, but an Orange Path does not take priority over the Heat Sink or Integrity Red paths IAW the User's Guide. Another reason it is incorrect is that both Red and Orange Subcriticality paths use FR-S.1, therefore FR-S.2 is plausible but incorrect.

D. Correct. In accordance with the EOP Users Guide, Section III.B determines priority of the CSFSTs in the following order: Subcriticality, Core Cooling, Heat Sink, Integrity, Containment, and Inventory. The same section determines RED as the highest priority followed by ORANGE, YELLOW and GREEN. It also describes how RED paths on lower priority trees must be addressed before ORANGE paths on higher priority trees due to the severe challenge to the safety function. Therefore, FR-H.1 is correct because it is the highest priority Red path CSFST.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

K/A# 2.4.22 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1/2OM-53B.2 Rev. 9 pg. 8-10

Question Source: Bank - Harris 2013 NRC Exam Q73

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

Objective: 3SQS-53.1, Rev. 2 Obj. 2. Concerning critical safety function restoration, IAW BVPS EOP Executive Volume, state from memory the following: a. The CFS in the order of priority.

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.

75. Which of the following parameters are required for Post Accident Monitoring (PAM) instrumentation IAW Tech Spec 3.3.3?

1. Pressurizer Water Level
2. Power Range Neutron Flux
3. Containment Area Radiation (High Range)
4. Reactor Vessel Water Level (RVLIS)
5. RCS Hot Leg Temperature (Wide Range)
6. Containment Sump Water Level (Wide Range)

A. 1, 3, 5 **only**

B. 2, 4, 6 **only**

C. 1, 2, 3, 6 **only**

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

Answer: D

Explanation/Justification: K/A is met by the candidates demonstrating the ability to identify instrumentation used for post-accident monitoring.

A. Incorrect. Plausible distractor because it is not an all-inclusive list of PAM instruments.
B. Incorrect. Plausible distractor because it is not an all-inclusive list of PAM instruments.
C. Incorrect. Plausible distractor because it is not an all-inclusive list of PAM instruments.
D. Correct. All of the instruments are post accident monitoring (PAM) instruments identified in Tech Spec 3.3.3.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to identify post-accident instrumentation.

K/A# 2.4.3 **K/A Importance** 3.7 **Exam Level** RO

References provided to Candidate None **Technical References:** Tech Spec Table 3.3.3-1

Question Source: New

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

Objective: 2SQS-6.7 Rev. 6 Obj. 6. List the nominal value of the control room operating parameters associated with the PSMS.

3SQS-INST ITS, Rev. 1 Obj. 2. State the purpose of each Instrumentation specification as described in the Applicable Safety Analyses section of the Bases

76. The plant is operating at 80% power.

- PRZR Master Pressure controller has failed.
- RCS pressure is 2300 psig and slowly rising.

1) The Unit Supervisor will establish a manual reactor trip setpoint of _____ in accordance with the Transient Response Guidelines.

2) What is the Tech Spec bases for the PRZR Pressure High Reactor Trip setpoint?

A. 1) 2375 psig
2) The setpoint ensures that protection is provided against violating the DNBR limit.

B. 1) 2340 psig
2) The setpoint ensures that protection is provided against violating the DNBR limit.

C. 1) 2375 psig
2) The setpoint minimizes challenges to safety valves while avoiding an unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

D. 1) 2340 psig
2) The setpoint minimizes challenges to safety valves while avoiding an unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

Question 76

Answer: D

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 specific knowledge of the Rx Trip Instrumentation TS bases for the PRZR high pressure Rx trip setpoint. Per the TRGs, Manual reactor trip criteria shall apply only when the command SRO establishes it. The command SRO shall determine manual reactor trip criteria and assign owners to monitor the affected parameters. When the process is returned to automatic control, the Command SRO shall evaluate and, if desired, terminate the set Manual Reactor Trip criteria.

K/A is met with the ability to determine that a manual rx trip is required at 2340 psig due to the przr master pressure controller failing low causing RCS pressure to rise.

A. Incorrect. First part is plausible because it is the setpoint for the PRZR High Pressure Rx trip setpoint which is close to the 2340 psig of the TRGs. Second part is plausible because it is the TS Bases for the przr pressure low pressure Rx trip setpoint.

B. Incorrect. First part is correct. Second part is plausible because it is the TS Bases for the przr pressure low pressure Rx trip setpoint.

C. Incorrect. First part is plausible because it is the setpoint for the PRZR High Pressure Rx trip setpoint which is close to the 2340 psig of the TRGs. Second part is correct.

D. Correct. IAW the Transient Response Guidelines, the manual Rx trip criteria for przr pressure is 2100 psig and dropping for low pressure, and >2340 psig for high pressure. The auto PRZR Pressure High Reactor Trip setpoint is 2375 psig. The Pressurizer Pressure - High LSSS (limiting safety system setting) is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding an unnecessary reactor trip for those pressure increases that can be controlled by the PORVs to prevent RCS overpressure conditions.

Sys #	System	Category	KA Statement
000027	Pressurizer Pressure Control System Malfunction / 3	AA2. Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions:	Conditions requiring plant shutdown

K/A# AA2.06 **K/A Importance** 3.9 **Exam Level** SRO

References provided to Candidate None **Technical References:** BVBP-OPS-0024 Rev. 12 pg. 17 & 18
TS Bases 3.3.1 pg. B3.3.1-19 & 20

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 3SQS-RCS ITS, Rev. 1 Obj. 3 State the purpose of each Instrumentation specification as described in the Applicable Safety Analyses section of the Bases

77. The plant has experienced a Main Steam Line Break inside Containment.

IAW 1/2OM-53B.2, EOP User's Guide, which of the following is the maximum time to isolate Auxiliary Feedwater to the faulted Steam Generator, and what is the reason for this action?

Isolate Auxiliary Feedwater to the faulted Steam Generator within _____

A. 15 minutes to preclude exceeding 10 CFR 50.67 dose limits.

B. 30 minutes to preclude exceeding 10 CFR 50.67 dose limits.

C. 15 minutes to preclude exceeding Containment design pressure and temperature.

D. 30 minutes to preclude exceeding Containment design pressure and temperature.

Question 77

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.A on page 3. Specifically, the SRO must have specific knowledge of the conditions and limitations identified in the UFSAR Safety Analysis in regards to required operator action times for isolating auxiliary feedwater during a MSLB inside containment, and the basis for this action.

K/A is met by demonstrating the ability to isolate a SG which has faulted inside containment, and also demonstrate the ability to explain why the faulted SG must be isolated within the prescribed period of time.

A. Incorrect. 15 minutes is plausible because this is the safety analysis time for AFW isolation during a Main Feedline Break. The reason is plausible because it is the TS primary to secondary leak bases which assumes MSLB will release the activity to the environment.

B. Incorrect. 30 minutes is the safety analysis time for AFW isolation during a Main Steam Line Break inside containment. The reason is plausible because it is the TS primary to secondary leak bases which assumes MSLB will release the activity to the environment.

C. Incorrect. 15 minutes is plausible because this is the safety analysis time for AFW isolation during a Main Feedline Break. Second part is correct.

D. Correct. 30 minutes is the safety analysis time for AFW isolation during a Main Steam Line Break inside containment. The release of high-energy fluid to the containment environment and elevated containment temperatures and pressures assume the auxiliary feedwater will be isolated to the faulted SG within 30 minutes

Sys #	System	Category	KA Statement
000040	Steam Line Rupture - Excessive Heat Transfer / 4	Generic	Ability to explain and apply system limits and precautions.

K/A# 2.1.32 **K/A Importance** 4.0 **Exam Level** SRO

References provided to Candidate None

Technical References: 1/2OM-53B.2 Rev. 9 pg.35-36
U2 UFSAR Rev. 16 pg. 6.2-41c,41h, 41i
2OM-53B.4.E-2 Iss 3 Rev 0 pg. 16

Question Source: Bank – Comanche Peak 2012 Q78

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(1)

Objective: GO-3ATA 4.1, Rev. 5 Obj. 2. Explain plant response to the below listed accidents: d. Steam Break (Major and Minor)

GO-3ATA 4.1, Rev. 5 Obj. 5. Explain plant automatic safety actions and Operator actions which mitigate the consequences of each of the analyzed accidents.

78. The following conditions exist:

- A Station Blackout has occurred.
- ECA-0.0, Loss of All AC Power is in progress.
- Power to at least one 4KV Emergency Bus CANNOT be restored within one hour.

1) Which of the following indications and/or control functions are available to the Operators in the Control Room until AC power is restored to an Emergency Bus after entry into ECA-0.0?
   1) PORV operation
   2) SG Atmospheric Steam Dump valve operation
   3) Plant Safety Monitoring System (PSMS)

2) For the above plant conditions, what procedural actions will be required?

A. 1) 1 and 3 only
   2) Initiate FSG-4, "ELAP DC Bus Load Shed / Management", and continue with procedure ECA-0.0.

B. 1) 1, 2, and 3
   2) Initiate FSG-4, "ELAP DC Bus Load Shed / Management", and continue with procedure ECA-0.0.

C. 1) 1 and 3 only
   2) Continue with procedure ECA-0.0, ELAP conditions are not applicable.

D. 1) 1, 2, and 3
   2) Continue with procedure ECA-0.0, ELAP conditions are not applicable.

Question 78

Answer: A

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 that ECA-0.0 requires that ELAP must be declared in one hour to comply with battery calculations to ensure a load shed of the batteries is completed to conserve battery capacity.

K/A is met by the candidate's ability to identify control room instruments and controls which will be operable during a loss of all AC power in which only the battery will be supplying power to plant equipment. Specifically, the PORVs are DC powered, and the PSMS indication is powered from the vital buses which are being supplied through an inverter by the batteries.

A. Correct. PORV are powered from DC panel DC2-02 & DC2-03, and PSMS is powered from the vital busses which at this time is powered from the Batteries, therefore they are available for use in the control room. The Note prior to ECA step 20 states that ELAP battery coping calculations assume that ELAP must be declared within 1 hour of ECA entry. This will ensure that actions are taken to isolate one train of batteries within 2 hours, and load shedding of the remaining batteries will be completed within 3 hours of the loss of all AC power. Procedure rules of usage require the SRO to commence the load shed FSG and continue with ECA-0.0

B. Incorrect. First part is plausible because the SG Atmospheric Steam Dump valves would normally be used because the condenser steam dumps will be unavailable due to no Circ water pumps running, but the ASD valves are powered from 480VAC (E-05, E13, E14). Second part is correct.

C. Incorrect. First part is correct, PORVs and PSMS will still have power available. Second part is plausible if the candidate does not recall the time limitation for ELAP entry (1 hr).

D. Incorrect. First part is plausible because the SG Atmospheric Steam Dump valves would normally be used because the condenser steam dumps will be unavailable due to no Circ water pumps running, but the ASD valves are powered from 480VAC (E-05, E13, E14). Second part is plausible if the candidate does not recall the time limitation for ELAP entry (1 hr).

Sys #	System	Category	KA Statement
000055	Station Blackout / 6	EA2 Ability to determine or interpret the following as they apply to a Station Blackout:	Instruments and controls operable with only dc battery power available

K/A# EA2.04 **K/A Importance** 4.1 **Exam Level** SRO

References provided to Candidate None

Technical References:
2SQS-6.4 LP Rev 17 pg. 15
2SQS-6.7 LP Rev 6 pg. 7
2SQS-21.1 LP Rev 23 pg.14, 15
2OM-53A.1.ECA-0.0 Iss 3 Rev 2 pg. 11
2OM-53B.4.ECA-0.0 Iss 3 Rev 2 pg. 130
1/2OM-53B.2 pg. 3 & 4

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 3SQS-38.1, Rev. 8 Obj. 13. Given a change in plant conditions, predict the 120 VAC distribution System response, to include automatic functions; and changes in system parameters and previously identified components and control systems.

3SQS-53.3, Rev. 5 Obj. 2. Describe from memory the overall purpose of each procedure, IAW BVPS-EOP Executive Volume.

79. The plant is at 90% power. The following events have occurred on August 2nd:

- At 0130, UPS*VITBS2-2, Vital Bus 2-2 UPS Unit is declared INOPERABLE.
- At 1200, UPS*VITBS2-3, Vital Bus 2-3 UPS Unit is declared INOPERABLE.
- At 1230, UPS*VITBS2-2, Vital Bus 2-2 UPS Unit is declared OPERABLE.

What time must Vital Bus 2-3 UPS Unit be restored to OPERABLE status to meet the Technical Specifications completion time?

(References Provided)

A. 0130 on August 3

B. 1200 on August 3

C. 0130 on August 4

D. 1200 on August 4

Question 79

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B page 3 first bullet. Specifically, the SRO must apply the required completion times (TS Section 1.3) in accordance with the TS Use and Application rules, to a piece of safety related equipment when the LCO had been previously been entered for the same equipment on the other train, and understand that even though an additional 24 hour extension to the completion time was possible, it was not conservative, and therefore, not allowed.

K/A is met by the SRO determining when Vital Bus 2-3 UPS Unit must be restored to operable status when it was declared inoperable at the same time as Vital Bus 2-2 UPS Unit was inoperable.

A. Incorrect – Plausible because this would true for Vital Bus 2-2 UPS Unit if Vital Bus 2-3 UPS Unit were declared operable at 1230.

B. Correct – Tech Spec Use and Application Section 1.3 discusses the use of completion times. All of the criteria for use of the completion time extension apply, specifically that the first inverter and second inverter were concurrently inoperable, and the second inverter remains inoperable after the first inverter is declared operable. Also, there is no exception that allows separate entry conditions for inverters in TS 3.8.7, so therefore the extension time can be applied. Therefore, the more restrictive completion time of 24 hours from the inoperability of Vital Bus 2-3 UPS Unit (second entry) must be used verses the stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours.

C. Incorrect – Plausible because this would be the time from the initial entry of Vital Bus 2-2 UPS Unit completion time plus 24 hours, but that is less restrictive than the correct answer.

D. Incorrect – Plausible if the additional 24 hours was misapplied from the 1200 entry of Vital Bus 2-3 UPS Unit onto the initial completion time of 24 hours.

Sys #	System	Category	KA Statement
000057	Loss of Vital AC Instrument Bus / 6	Generic	Ability to determine operability and/or availability of safety related equipment.

K/A# 2.2.37 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate Tech Spec 3.8.7 pg. 3.8.7-1 with NOTE removed from page.

Tech Spec 1.3 pgs. 1-12

Technical References: Tech Spec 3.8.7 pg. 3.8.7-1
Tech Spec 1.3 pgs. 1.3 -2, 4, 5

Question Source: Bank – DC Cook 2014 NRC Exam Q80

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 3SQS-RULES ITS, Rev. 3 Obj. 5. Given plant conditions, apply the rules of ITS Section 1.3 to ensure compliance with Technical Specifications / LRM

80. The plant is in Mode 3 with maintenance being performed on MCC2-E05 and Battery 2-3.

- Maintenance on MCC2-E05 will de-energize the MCC for 1 hour.
- The Battery maintenance will take 15 minutes.

During the maintenance on battery 2-3, an uninsulated wrench was inadvertently dropped on the battery terminals causing the terminal lead to vaporize.

- The crew has entered AOP 2.39.1C, Loss of 125VDC Bus 2-3.
- All electrical equipment was in NSA prior to the event except for MCC2-E05.
- NO Operator actions have been taken.

1) Based on the above conditions, which of the following indications would be observed on 2RCS-PI457, PRZR CHANNEL 3 PRESS indicator?

   2RCS-PI457 will indicate _____.

45 minutes after the event:

- MCC2-E05 has been restored to service.
- Battery Breaker 2-3 and Battery Breaker Disconnect Switch have been opened.
- Battery Charger 2-3 has been restored to service.

2) Currently, what is the status of Vital Bus 3 Inverter IAW LCO 3.8.7, Inverters – Operating?

   Vital Bus 3 Inverter _____ OPERABLE.

A. 1) failed low
   2) is

B. 1) failed low
   2) is NOT

C. 1) current RCS pressure
   2) is

D. 1) current RCS pressure
   2) is NOT

Question 80

Answer: D

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 specific knowledge of TS bases that are required to analyze the vital bus 3 inverter operability even though the 480 VAC supplies, and the battery charger are available, but the inverter is not operable because the associated Battery must be available as the uninterruptible power supply.

K/A is met with the candidate's ability to determine that a vital bus 3 load (2RCS-PI457) will be indicating properly after a loss of both associated DC supplies (battery and charger), due to a substitute power source which automatically transfers to the inverter via the inverter static switch.

A. Incorrect. First part is plausible if the candidate does not recognize that vital bus 3 will be powered from MCC2-E07 from the static line voltage regulator (SLVR) through the static switch. Second part is plausible because even though the vital bus 3 is restored with normal 480 VAC E05 and the battery charger, it is not operable because the LCO bases states that the 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. The 2-3 battery was removed from service.

B. Incorrect. First part is plausible if the candidate does not recognize that vital bus 3 will be powered from MCC2-E07 from the static line voltage regulator (SLVR) through the static switch. Second part is correct.

C. Incorrect. First part is correct. 2RCS-PI457 will be powered from vital bus 3 and indicating properly. Second part is plausible because even though the vital bus 3 is restored with normal 480 VAC E05 and the battery charger, it is not operable because the LCO bases states that the 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. The 2-3 battery was removed from service.

D. Correct. 2RCS-PI457 will be indicating properly because it is powered from vital bus 3 via MCC2-E07 from the static line voltage regulator (SLVR) through the static switch. Vital bus 3 is NOT operable because even though it is restored with normal 480 VAC E05 and the battery charger, it is not operable because the LCO bases states that the 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. The 2-3 battery was removed from service.

Sys #	System	Category	KA Statement
000058	Loss of DC Power / 6	AA2. Ability to determine and interpret the following as they apply to the Loss of DC Power:	That a loss of dc power has occurred; verification that substitute power sources have come on line

K/A# AA2.01 **K/A Importance** 4.1 **Exam Level** SRO

References provided to Candidate None

Technical References: 3SQS-38.1 U2 PPNT Rev 8 slide 23
3SQS-39.1 U2 PPNT Rev. 9 slide 10
Tech Spec 3.8.7 and Bases. pg. 3.8.7-1, B3.8.7-2

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 3SQS-38.1, Rev. 8 Obj. 13. Given a change in plant conditions, predict the 120 VAC distribution System response, to include automatic functions; and changes in system parameters and previously identified components and control systems.

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.

81. The reactor tripped from 100% power, and the crew is performing step 9, Verify AFW Status of E-0, Reactor Trip or Safety Injection.

- No AFW flow is available.
- The crew has transitioned to FR-H.1, Loss of Heat Sink and established Bleed and Feed.
- The BOP reports that SG WR levels are:
  'A' SG is 11%
  'B' SG is 15%
  'C' SG is 12%

Based on the above conditions, when a flowpath has been established to feed the SGs, which of the following actions are required to be taken by the crew per FR-H.1?

A. Secure Bleed and feed by securing one charging pump and close all PRZR PORVs before feeding any SGs.

B. Commence feeding all SGs at ≤100 gpm.

C. Commence feeding 'B' SG without feed limitation until SG NR level is >12%.

D. Continue Bleed and Feed until Core Exit Thermocouples temperatures begin to lower, then begin feeding SGs at the appropriate feed rate.

Question 81

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 determine that one SG wide range level is >14% WR, therefore it not considered dry, and feed flowrate is not limited during recovery of a loss of heat sink.

K/A is met by demonstrating the candidate's ability to interpret plant conditions while in FR-H.1, Loss of Secondary Heat Sink, and executing the procedural step requiring feedwater addition to the SG which is not classified as dry (<14% WR0 to ensure a more desirable mode of recovery than Bleed and Feed.

A. Incorrect. Plausible because Bleed and Feed is secured starting at step 36 by securing all but one charging pump and closing all PORVs, but only after adequate secondary heat sink is verified with one SG >12% NR in step 31.

B. Incorrect. Plausible because if all SG WRs are <14% and a feed flowpath is established, feed flow not exceeding 100 gpm is established to only one SG due to a possible failure of the SG tubes caused by thermal stresses when feeding a dry SG (<14% WR).

C. Correct. The feed flow rate to 'B' SG does not have to be limited since it is not considered a dry SG (level is 15% WR) and the RNO for controlling feed flow will be used as guidance. The >12% NR is the hold point in the procedure at step 31 to ensure adequate secondary heat sink before moving on to secure Bleed and Feed.

D. Incorrect. Plausible since Bleed and Feed has been established, someone not familiar with the procedure may consider this to be correct since CETs are checked multiple times in FR-H.1 to ensure they are dropping, which the candidate may consider a holding point until feeding the SGs.

Sys #	System	Category	KA Statement
WE05	Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	Generic	Ability to interpret and execute procedure steps.

K/A# 2.1.20 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate None **Technical References:** 2OM-53A.1.FR-H.1 Iss 2 Rev 1 pg. 22-26
2OM-53B.4.FR-H.1 Iss 2 Rev 1 pg. 95-96

Question Source: New

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 plant conditions:

- The crew is raising power from 50% to 100% RTP
- Reactor power is currently at 55% RTP at EOL
- Control Bank D rods are at 171 steps
- 30 minutes ago, a 200 gallon dilution to the RCS was performed
- The ATC pulls rods and releases the switch, then returns rod control to AUTO
- Tavg and Reactor power steadily rise
- A4-3C, TAVG DEV FROM TREF alarms
- A4-6D, DELTA FLUX OUTSIDE TARGET BAND annunciator window ALARMS
- VCT level has remained stable for the last 15 minutes

Which of the following completes the statements below?

Based on the above conditions, the accident the crew is dealing with is a(an) _____(1)_____ accident.

(2) Based on the accident in progress, which of the following actions would be directed by the Alarm Response Procedure for A4-3C, TAVG DEV FROM TREF?

A. 1) uncontrolled rod withdrawal
   2) Place Rod Control in Manual and insert until TAVG= TREF IAW the ARP for A4-3C

B. 1) uncontrolled rod withdrawal
   2) Lower Turbine load IAW 2OM-52.4.B, Load Following

C. 1) dilution
   2) Lower Turbine load IAW 2OM-52.4.B, Load Following

D. 1) dilution
   2) Place Rod Control in Manual and insert until TAVG= TREF IAW the ARP for A4-3C

Question 82

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 specific knowledge of the Axial Flux Difference tech spec bases, and its importance to reactor safety during an uncontrolled rod withdrawal event.

K/A is met by the candidate determining that an uncontrolled rod withdrawal has occurred based on the indications of Tavg and power rising, and delta flux annunciator alarming.

A. Correct. Tavg and Reactor power rising are indications of an uncontrolled rod withdraw. For a Malfunction of Automatic Rod Control, ARP A4-3C directs placing Rod Control to Manual and insert rods until TAVE = TREF.

B. Incorrect. Uncontrolled rod withdraw is correct. Limit the gross radial power distribution is incorrect. ARP A4-3C directs adjusting turbine load for other causes of the alarm, such as a decrease in steam demand.

C. Incorrect. Dilution is incorrect. Plausible if the student thinks that 200 gallon dilution is too much. A normal dilution at power for temperature control is roughly 20 gallons. However, 200 gallons would be an appropriate amount for raising power. The candidate should also realize that VCT level is constant, during a dilution, level would be rising. Limit the amount of axial power distribution is correct. ARP A4-3C directs adjusting turbine load for other causes of the alarm, such as a decrease in steam demand.

D. Incorrect. Dilution is incorrect. Plausible if the student thinks that 200 gallon dilution is too much. A normal dilution at power for temperature control is roughly 20 gallons. However, 200 gallons would be an appropriate amount for raising power. The candidate should also realize that VCT level is constant, during a dilution, level would be rising. For a Malfunction of Automatic Rod Control, ARP A4-3C directs placing Rod Control to Manual and insert rods until TAVE = TREF.

Sys #	System	Category	KA Statement
000001	Continuous Rod Withdrawal / 1	AA2. Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal	Uncontrolled rod withdrawal, from available indications

K/A# AA2.05 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate None **Technical References:** 2OM-2.4.AAV Rev 6 pg. 4

Question Source: Modified – Robinson 2013 NRC Exam Q82

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

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

83. Given the following conditions:

- The plant has just shutdown for a refueling outage.
- Currently in Mode 3.
- An accident has occurred causing a Waste Gas Charcoal Delay bed to be damaged releasing the tank contents to the Auxiliary Building.

1) Which of the following detectors would indicate that a gaseous release has occurred in the Auxiliary Building?

2) What is the required frequency of the Channel Operational Test for this detector to maintain operability IAW 1/2-ODC-3.03?

(Reference Provided)

A. 1) 2HVS-RQ101, Ventilation Vent Radiation Monitor
   2) Monthly

B. 1) 2HVS-RQ101, Ventilation Vent Radiation Monitor
   2) Quarterly

C. 1) 2HVS-RQ109, Elevated Release Radiation Monitor
   2) Monthly

D. 1) 2HVS-RQ109, Elevated Release Radiation Monitor
   2) Quarterly

Question 83

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B first and fourth bullets on page 3. Specifically, the SRO must have specific knowledge of the ODCM to determine that a channel operational test must be completed on a quarterly basis to maintain the Elevated Release Radiation Monitor, 2HVS-RQ109 operable.

K/A is met by the candidate's ability to determine that an accidental gaseous release in the Auxiliary Building will be detected by the Leak Collection Filtered exhaust system radiation monitor at the Elevated Release discharge, and to maintain this radiation monitor operable, a channel operational test must be performed quarterly per the ODCM.

A. Incorrect. Plausible because Leak Collection Normal and Filtered exhaust systems are connected. The candidate must know that the Auxiliary Building is exhausted by Filtered exhaust fans and not the Normal exhaust which discharges out the Ventilation Vent. Second part is plausible because the channel source check must be completed monthly.

B. Incorrect. Plausible because Leak Collection Normal and Filtered exhaust systems are connected. The candidate must know that the Auxiliary Building is exhausted by Filtered exhaust fans and not the Normal exhaust which discharges out the Ventilation Vent. COT performed quarterly is correct.

C. Incorrect. Elevated Release Radiation Monitor is correct. Second part is plausible because the channel source check must be completed monthly.

D. Correct. The Auxiliary Building is exhausted by Filtered exhaust fans which discharge out the Elevated Release which is monitored by 2HVS-RQ109. The ODCM states that a Channel Operational test must be completed on a quarterly basis to maintain 2HVS-RQ109 operable.

Sys #	System	Category	KA Statement
000060	Accidental Gaseous Radwaste Release / 9	Generic	Ability to determine operability and/or availability of safety related equipment.

K/A# 2.2.37 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate 1/2-ODC-3.03 Rev. 17 with Table 1.1 removed. **Technical References:** 2SQS-16.1 PPNT Rev. 12 Slide 6; 1/2-ODC-3.03 Rev. 17 pg. 48

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 2SQS-43.1, Rev. 11 Obj. 13. Using a copy of the Technical Specifications or Offsite Dose Calculation Manual, analyze a given set of plant conditions for compliance with the licensing requirements, including the determination of equipment operability and applicable action statements.

84. The plant is in Mode 6 with core reload in progress.

- Vital Bus 1 failure has occurred
- US has entered AOP 2.38.1A, Loss of Vital Bus 1

Which of the following describes the effect of this issue on core reload?

A. Core reload may commence after operators verify SR channel N32 is operable, and one Gamma Metrics detector is operable providing indication in the control room.

B. Core reload may NOT continue until SR channel N31 is returned to operable status and Boron concentration has been verified.

C. Core reload may commence after operators verify that the SR Audio Count Rate Drawer is aligned to SR channel N32 and verify continuous audible indication in the control room.

D. Core reload may commence after operators verify that SR channel N32 is operable providing continuous visual indication in the control room.

Question 84

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 specific knowledge that two source range nuclear instruments are required to be operable in mode 6 to move fuel, and that the Tech Spec bases states that an alternate monitor may be used in place of the normal SR monitors. In this case, source range channel N31 has lost power and become inoperable, therefore, a Gamma Metrics monitor may be used in its place as a second SR monitor.

K/A is met by the loss of vital bus 1 which powers SR detector N31 making the candidate determine what available SR detectors are needed to recommence the core reload. During refueling, two redundant SR detectors are required to detect changes in core reactivity.

A. Correct. Vital bus 1 loss will de-energize the N31 SR detector making it inoperable. When in Mode 6, TS 3.9.2 requires two operable SR detectors providing visual indication in the control room. The operable detectors may consist of N31, N32, or either of the alternate monitors (gamma metrics) when in mode 6. Since N31 is de-energized (inoperable), core reload may not commence until N32 and a gamma metrics detector are verified operable.

B. Incorrect. Plausible if the candidate knows that 2 SR detectors are required for fuel movement but does not recognize that the Tech Spec 3.9.2 background permits the gamma metrics to replace N31 or N32.

C. Incorrect. Plausible if the candidate is thinking of TS 3.3.8, Boron Dilution Detection which requires one SR detector to be operable and audible indication in the control room and containment, but TS 3.3.8 is applicable in modes 3-5. TS 3.9.2 is the applicable TS in mode 6. This is an L5 log reading that the candidate should be familiar with during refueling.

D. Incorrect. Plausible if the candidate confuses TS 3.3.8 which only requires one SR detector to be operable, but TS 3.3.8 is only applicable in modes 3-5. TS 3.9.2 is applicable in mode 6 and requires two operable SR detectors.

Sys #	System	Category	KA Statement
000032	Loss of Source Range Nuclear Instrumentation / 7	AA2. Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation:	Normal/abnormal power supply operation

K/A# AA2.01 **K/A Importance** 2.9 **Exam Level** SRO

References provided to Candidate None **Technical References:** TS 3.9.2 pg. 3.9.2-1
TS 3.9.2 Bases pg. B 3.9.2-1 & 2

Question Source: Modified – 2016 Indian Point Q83

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

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

85. Given the following plant conditions:

- A Large Break LOCA has occurred.
- The Unit Supervisor has completed ES-1.3, "Transfer to Cold Leg Recirculation" and transitions back to E-1, "Loss of Reactor or Secondary Coolant", procedure step in effect.
- Containment Pressure is 10 psig and slowly DROPPING.
- No containment Quench Spray Pumps are running.
- Containment Sump Level is 190 inches and slowly RISING.

Based on these plant conditions, (1) what Functional Restoration Procedure (FRP) transition is required AND (2) what actions will be taken by the Unit Supervisor?

A. (1) FR-Z.2, Response to Containment Flooding
(2) remain in FR-Z.2 until the orange condition has cleared

B. (1) FR-Z.1, Response to High Containment Pressure
(2) remain in FR-Z.1 until the orange condition has cleared

C. (1) FR-Z.2, Response to Containment Flooding
(2) after completing FR-Z.2 procedural steps, transition back to E-1

D. (1) FR-Z.1, Response to High Containment Pressure
(2) after completing FR-Z.1 procedural steps, transition back to E-1

Question 85

Answer: C

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 specific knowledge of diagnostic steps and decision points in FR-Z.2, Response to Containment Flooding that involve the transition from FR-Z.2 at the end of the procedure even if conditions are not restored to a GREEN priority.

K/A is met by requiring the candidate to determine that an Orange path has been met for Containment Flooding while in E-1, Loss of Reactor or Secondary Coolant, and that a transition should be made to FR-Z.2. Then, when the steps of FR-Z.1 are completed, transition back to E-1 whether the condition has cleared or not. The question demonstrates integrated plant procedure use in the EOP network.

A. Incorrect. It is correct that an orange path FR-Z.2 condition does exist. However, the SRO should know the rules of usage and background information which states that once all actions of this procedure are completed, the operator shall return to procedure and step in effect.

B. Incorrect. Although plausible, an orange condition exists only if containment pressure is > 11 psig with no quench spray pumps. In the stated conditions, containment pressure is 10 psig and dropping. Again the procedural usage portion is incorrect.

C. Correct. According to 2OM-53A.1.F-0.5, containment sump level > 187 inches is an orange path condition and warrants entry into FR-Z.2. 2OM-53B.4.FR-Z.2 background document states that once all actions of this procedure are completed that the containment status tree function may not be restored to a green condition. In this case, the appropriate FRP does not need to be implemented again since all actions have already been performed in order to adhere to facility license and amendments.

D. Incorrect. Incorrect procedure but correct procedural usage for the actions of this procedure as explained above.

Sys #	System	Category	KA Statement
WE15	Containment Flooding / 5	Generic	Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A# 2.1.23 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** 2OM-53A.1.F-0.5, Iss. 3 Rev. 0
2OM-53B.4.FR-Z.2, Iss. 3 Rev. 0 step 4

Question Source: Bank – 2LOT7 Q85

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 EOPs IAW BVPS-EOP Executive Volume

86. The following plant conditions exist:

- A Small Break LOCA has occurred
- The Reactor **failed** to trip
- Reactor Power is 10% and LOWERING
- RCS pressure is 1200 psig and LOWERING
- SG pressures are 1000 psig and STABLE
- SG WR levels are 40% and LOWERING
- SI has actuated properly
- AFW failed to start manually or automatically

What action is required regarding the RCPs, and why?

Procedure titles:
E-0, Reactor Trip or Safety Injection
FR-H.1, Response to Loss of Secondary Heat Sink
FR-S.1, Response to Reactor Restart/ATWS

A. Trip RCPs because FR-H.1 entry criteria is met.

B. Trip RCPs because of E-0 left hand page RCP trip criteria.

C. Leave RCPs in service for core cooling due to FR-S.1 entry.

D. Leave RCPs in service for heat removal due to bleed and feed criteria not met.

Question 86

Answer: C

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 assess the given plant conditions and determine that a transition to FR-S,1 is required, as well the FR-S.1 background knowledge that the RCPs are required for core cooling when reactor power is >5% even if all normal running conditions are not met.

K/A is met by interpreting the plant conditions given and determining that the Reactor Coolant Pumps must remain running based on FR-S.1, Response to Nuclear Power Generation -ATWS procedural guidance requiring temporary core cooling.

A. Incorrect. Plausible because a loss of AFW leads to FR-H.1, however FR-S.1 guidance takes priority over FR-H.1.

B. Incorrect. Plausible because the candidate may recognize the E-0 RCP trip criteria of SG/RCS DP of <205 psid is met, but should also recognize that the first immediate operator action of E-0, step 1 RNO will transition the crew to FR-S.1 which requires the RCPs not to be tripped when reactor power is >5% because they are providing temporary cooling of the core.

C. Correct. Conditions for entry into subcriticality red path FR-S.1 have been met, and a caution before step 1 of FR-S.1 states the RCP's should not be tripped with reactor power greater than 5%. FR-S.1 background document states "During an ATWS, RCP operation could be beneficial by temporarily cooling the core under voided RCS conditions. If reactor power is greater than 5%, the RCPs should not be tripped even if all normal running conditions are not satisfied.

D. Incorrect. Plausible because FR-H.1 Bleed and Feed criteria of two SGs WR levels <14% has not been met, therefore they may consider leaving the RCPs running because the RCPs are tripped in FR-H.1 after attempts to restore feedwater have been tried. The answer is also incorrect because FR-S.1 is a higher priority that FR-H.1.

Sys #	System	Category	KA Statement
003	Reactor Coolant Pump	Generic	Ability to interpret and execute procedure steps.

K/A# 2.1.20 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate None **Technical References:** 2OM-53A.1.FR-S.1 Iss. 2 Rev. 2 pg. 2
2OM-53B.4.FR-S.1 Iss. 2 rev. 2 pg. 60

Question Source: Bank – 2018 Ginna Q79

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

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

87. Given the following conditions:

- The plant has tripped due to a small break LOCA.
- The crew is performing ES-1.2, Post LOCA Cooldown and Depressurization.
- A6-4A, Primary Plant Demin Water Storage Tank Level Low annunciator alarms.
- The A6-4A ARP directed the crew to perform EOP Att. A-1.8, MAKEUP TO PPDWST [2FWE*TK210].

1) What is the setpoint for the Primary Plant Demin Water Storage Tank Level Low alarm?
2) IAW EOP Att. A-1.8, when is the crew directed to align service water to the AFW pump suction?

A. 1) 85 inches
   2) 25 inches

B. 1) 85 inches
   2) 50 inches

C. 1) 200 inches
   2) 25 inches

D. 1) 200 inches
   2) 50 inches

Question 87

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E 1st bullet on page 7. Specifically, the SRO must have specific knowledge of EOP attachment A-1.8, Makeup to PPDWST (2FWE*TK210) and know that when PPDWST lowers to 25 inches, that service water must be aligned to the AFW pump suction.

K/A is met with the ability to verify the AFW pump normal suction supply tank, Primary Plant Demin Water Storage Tank (PPDWST), low level setpoint alarm, and recognize that if the PPDWST level goes below 25 inches, that they must align service water to the AFW pump suction iaw. the alarm response procedure directed EOP attachment A-1.8.

A. Correct. Primary Plant Demin Water Storage Tank low level alarm setpoint is 85 inches. The note prior to step 9 of EOP att. A-1.8 states if no other source of water is available, then service water should be used, and step 9 states if PPDWST level drops to 25 inches, then align service water to the AFW pump suction.

B. Incorrect. First part is correct. Second part is plausible because it is an arbitrary value which is low enough to consider aligning service water to AFW based on both the 85 & 200 inch levels given for the low-level values of the question.

C. Incorrect. 200 inches is plausible because it is greater than the EOP procedure left hand page value of 150 inches to perform attachment A-1.8, but the low level setpoint is 85 inches. Second part is correct.

D. Incorrect. 200 inches is plausible because it is greater than the EOP procedure left hand page value of 150 inches to perform attachment A-1.8, but the low level setpoint is 85 inches. Second part is plausible because it is an arbitrary value which is low enough to consider aligning service water to AFW based on both the 85 & 200 inch levels given for the low-level values of the question.

Sys #	System	Category	KA Statement
061	Auxiliary/Emergency Feedwater	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:** 2OM-24.4.AAI Rev. 12 pg. 3
2OM-53A.1.A-1.8 Rev. 7 pg. 3

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 2SQS-24.1, Rev. 26 Obj. 20. Given a Main Feedwater, Startup Feedwater, Auxiliary Feedwater System or Steam Generator Water Level Control System alarm condition and using the Alarm Response Procedure(s), determine the appropriate alarm response, including automatic and operator actions in the control room.

88. Given the following plant conditions:

- The plant is at 100% power.
- Vertical Board 'C' indicator for No. 1 DC Bus Grd Volts meter indicates a +110 VDC ground.
- The crew performs 2OM-39.4.F, Clearing Grounds (125 VDC Busses 2-1 and 2-2).
- When [BAT*BKR2-1], Battery Breaker for 2-1 battery, was opened IAW 2OM-39.4.F, the No. 1 DC Bus Grd Volt meter indicated 0 VDC.

As compared to the as found conditions, No. 1 DC Bus Voltage indication will be _____(1)_____ when the battery breaker is opened.

Based on these conditions, entry into Technical Specification 3.8.9, Distribution Systems – Operating, _____(2)_____ required.

A. 1) lower
   2) is

B. 1) lower
   2) is not

C. 1) unaffected
   2) is

D. 1) unaffected
   2) is not

Question 88

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B page 3 third bullet. Specifically, the SRO must apply the applicable tech specs when a DC power source is lost and have sufficient knowledge of the tech spec bases to determine that given the plant conditions, the Electrical distribution TS remains met because an OPERABLE DC electrical distribution subsystem is required to be energized to the correct voltage from EITHER the associated battery OR charger.

K/A is met with the ability to determine the impact on DC bus voltage when a battery is removed from service due to a ground.

A. Incorrect. Plausible if the candidate doesn't understand that the charger maintains a "float" charge on the battery which keeps DC bus voltage higher than if the battery was supplying the DC bus. TS 3.8.4 is applicable for an inoperable Battery, however, since the charger remains operable, the DC bus will remain powered with its correct voltage, so TS 3.8.9 entry is NOT required.

B. Incorrect. Plausible if the candidate doesn't understand that the charger maintains a "float" charge on the battery which keeps DC bus voltage higher than if the battery was supplying the DC bus. Second part is correct.

C. Incorrect. First part correct since the charger continues to supply the DC bus voltage. TS 3.8.9 entry is not required since the charger remains operable, the DC bus will remain powered with its correct voltage.

D. Correct. The charger will continue to supply the DC bus with the correct voltage and TS 3.8.9 entry is not required since the charger remains operable, the DC bus will remain powered with its correct voltage.

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** 3.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** TS 3.8.9 bases pg. B3.8.9-2
2OM-39.4F Rev 5

Question Source: Modified – 1LOT16 Q48

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 3SQS-39.1, Rev. 9 Obj. 25. Using a copy of the Technical Specifications or Licensing Requirements Manual, analyze a given set of plant conditions for compliance with the licensing requirements, including the determination of equipment operability and applicable action statements. Using a copy of the Technical Specifications or Licensing Requirements Manual, analyze a given set of plant conditions for compliance with the licensing requirements, including the determination of equipment operability and applicable action statements.

89. The plant is at 100% power with all systems in normal alignment **except** 2SWE-P21B, 'B' Standby Service Water pump is on clearance.

- 2SWS-P21A, 'A' Service Water pump tripped on overcurrent due to a shaft seizure
- A1-4G, Service Water Header Pressure Low is LIT
- 2SWE-P21A, Standby Service Water pump failed to auto start
- All other equipment operated as designed

1) Which of the following Tech Spec LCOs are required to be entered when 2SWS-P21A, 'A' Service Water pump tripped?

The crew has entered AOP-2.30.1, Service Water/Main Intake Structure Loss, and taken the appropriate Control Room actions to restore Service Water Header pressures to normal. No Field Operator actions have been taken.

2) Is LCO 3.7.8 met once the Service Water Header pressures are restored to normal?

TS 3.7.8, Service Water System
TS 3.8.1, AC Sources - Operating

A. 1) Tech Spec 3.7.8 only.
   2) Yes, LCO 3.7.8 is met.

B. 1) Tech Spec 3.7.8 only.
   2) No, LCO 3.7.8 is not met.

C. 1) Tech Spec 3.7.8 and 3.8.1.
   2) Yes, LCO 3.7.8 is met.

D. 1) Tech Spec 3.7.8 and 3.8.1.
   2) No, LCO 3.7.8 is not met.

Question 89

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B page 3 third bullet. Specifically, the SRO must apply the applicable tech specs when a train of SW is lost and have sufficient knowledge of the tech spec bases to determine that given the plant condition, the TS may not be exited.

K/A is met with the ability to predict the impact that a loss of one service water pump will have on the SWS system other associated systems involving equipment operability. Then based on system and tech spec knowledge, the candidate must know that to mitigate the consequences of the standby service water being cross tied to the SWS, that TS 3.7.8 must remain in effect.

A. Incorrect. TS 3.7.8 only is plausible if they do not remember the note stating to enter TS 3.8.1, or don't remember that the SWS is required for EDG operation. Exiting TS 3.7.8 is plausible because the standby service water pump will operate and maintain the header pressure as required, but that portion of the system is not qualified to maintain SWS operable.

B. Incorrect. TS 3.7.8 only is plausible if they do not remember the note stating to enter TS 3.8.1, or don't remember that the SWS is required for EDG operation. Second part is correct. TS 3.7.8 cannot be exited until 'C' SW pump is placed in service, and 2SWEMOV116A is closed isolating the non-qualified piping of the standby service water system.

C. Incorrect. Both TS 3.7.8 and TS 3.8.1 must be entered is correct. Exiting TS 3.7.8 is plausible because the standby service water pump will operate and maintain the header pressure as required, but that portion of the system is not qualified to maintain SWS operable.

D. Correct. Both TS 3.7.8 and TS 3.8.1 must be entered. TS 3.7.8, because the 'A' service water pump is inoperable and standby service water piping is not qualified. TS 3.8.1 must be entered because the note on condition A states it must be entered for the diesel generator made inoperable due to inoperable SWS. TS 3.7.8 cannot be exited until 'C' SW pump is placed in service, and 2SWEMOV116A is closed isolating the non-qualified piping of the standby service water system. By stating no field actions have been taken it ensures that the 'C' SWS is not available.

Sys #	System	Category	KA Statement
076	Service Water	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of SWS

K/A# A2.01 **K/A Importance** 3.7 **Exam Level** SRO

References provided to Candidate None

Technical References:
TS 3.7.8 pg. 3.7.8-1
TS 3.7.8 bases pg. B3.7.8-2
2OM-30.4.AAB rev. 4 pg. 3
2SQS-30.1, Rev. 23 pg. 58, 59

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 3SQS-PLTSYS ITS, Rev. 2 Obj. 3. Given plant conditions, determine the criteria necessary to ensure compliance with each Section Plant Systems System LCO and Licensing Requirement in accordance with the Bases, Surveillance Requirements, and the Applicability.

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.

90. 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 2FWS*FCV479(489)(499).
- Annunciator A6-3C, Station Instrument Air Receiver Tank Trouble is received.
- Station Instrument Air Header Pressure is 80 psig and slowly dropping.
- A local operator reports that the in-service station instrument air dryer has malfunctioned, and both towers are venting.

1) IAW AOP 2.34.1, 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 Instrument Air Bypass filters in service, **THEN** isolate the Instrument Air dryers.
2) **OPEN**

B. 1) Place the Instrument Air Bypass filters in service, **THEN** isolate the Instrument Air dryers.
2) **CLOSED**

C. 1) Place the Standby Instrument Air dryer in service, **THEN** isolate the malfunctioned Instrument Air dryer.
2) **OPEN**

D. 1) Place the Standby Instrument Air dryer in service, **THEN** isolate the malfunctioned Instrument Air dryer.
2) **CLOSED**

Question 90

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 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 demonstrating how a malfunction of an instrument air dryer could affect the equipment controlled by instrument air, and based on knowledge of the loss of instrument air AOP, direct the field operator to place the bypass filters in service, and isolate the air dryers to minimize plant response.

A. 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.

B. Correct. IAW AOP-2.34.1 step 8 refers to attachment D which places the bypass filters in service and isolates the air dryers. A note at the beginning of attachment E for actions for imminent loss of station air states that the BFRVs fail closed.

C. 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 D 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.

D. 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 D directs placing the bypass filters in service and isolating both air dryers. Second part is correct.

Sys #	System	Category	KA Statement
078	Instrument Air	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Air dryer and filter malfunctions

K/A# A2.01 **K/A Importance** 2.9 **Exam Level** SRO

References provided to Candidate None **Technical References:** 2OM-53C.4.2.34.1 rev. 21 pg. 4, 10, 11

Question Source: Bank – 2LOT6 Q90

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 2SQS-53C.1, Rev. 11 Obj. 4 Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.

2SQS-34.1, Rev. 18 Obj. 1. Describe the function of the various Unit 2 Compressed Air Systems and the associated major components as documented in 2OM-34, the associated Operating Manual.

91. The plant was operating at full power when a reactor trip occurred.
The following was noted during the IOAs of E-0, Reactor Trip or Safety Injection:

- Power range NI's are 3% and lowering.
- All control rods indicate 0 steps except for the following:
  H14 – 18 steps
  F6 – 48 steps
- A complete loss of Off-site power occurred when the auto bus transfer occurred.
- Both Diesel generators started and loaded.

The crew is currently performing ES-0.1, Reactor Trip Response at step 6, Verify all control rods inserted.

1) What is the status of Digital Rod Position Indication (DRPI)?

2) What action will the US direct the crew to perform at this step and why?

A. 1) Energized
2) Continue to the next step of ES-0.1 because shutdown margin is met.

B. 1) Energized
2) Emergency borate to account for the reactivity worth of the stuck rods.

C. 1) De-energized
2) Continue to the next step of ES-0.1 because shutdown margin is met.

D. 1) De-energized
2) Emergency borate to account for the reactivity worth of the stuck rods.

Question 91

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 ES-0.1 step which verifies all control rods are fully inserted, and recognize that with two rods stuck out, the crew must emergency borate per the step Response Not Obtained column.

K/A is met with the candidate's ability to determine that Digital Rod Position Indication (DRPI) will be de-energized after a loss of offsite power because it is powered from normal, not emergency 480 VAC busses, then recognize that when verifying all control rods are fully inserted in ES-0.1, that emergency boration of the RCS is required, because DRPI is de-energized, and they must use previously reported indications.

A. Incorrect. Plausible to think DRPI is energized after the loss of offsite power if the candidate thinks it is powered from the diesel generators. Second part is plausible if the candidate does not know that emergency boration is required for two or more stuck rods due to shutdown margin requirements.

B. Incorrect. Plausible to think DRPI is energized after the loss of offsite power if the candidate thinks it is powered from the diesel generators. Second part is correct.

C. Incorrect. First part is correct. Second part is plausible if the candidate does not know that emergency boration is required for two or more stuck rods due to shutdown margin requirements.

D. Correct. DRPI is powered from 480 bus 2E which is supplied from the 'A' 4KV bus, 'A' 4KV bus is de-energized on a loss of offsite power. On a Bus 2E undervoltage condition, bus 2E auto crossties to bus 2F which is power from 'C' 4KV bus, but in the case of a complete loss of offsite power, 'A' 4KV bus is de-energized also. Based on the information given in the stem regarding two stuck rods prior to DRPI de-energizing, the US would direct emergency boration because with more than one rod failing to fully insert, the shutdown margin must be made up through emergency boration to account for reactivity worth of the stuck rods.

Sys #	System	Category	KA Statement
014	Rod Position Indication	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of offsite power

K/A# A2.01 **K/A Importance** 3.3 **Exam Level** SRO

References provided to Candidate None **Technical References:** 2SQS-1.4 Rev. 9 pg. 7
2OM-53A.1.ES-0.1 Iss. 3 Rev. 0 pg. 7
2OM-53B.4.ES-0.1 Iss. 3 Rev. 0 pg. 13

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 2SQS-1.4, Rev. 9 Obj. 4. Explain the impact on the DRPI system of losing any one of the following power supplies: a. 4 KV Bus 2A b. 480 VAC Bus 2E, c. MCC 2-19-1, d. MCC 2-19-2

92. Plant is operating at 100% power.

- 2MSS-PT496, Ch. IV, 'C' SG Steam Pressure fails HIGH

Which of the following completes these statements?

1) If **NO** operator actions are taken, the Reactor will automatically trip _____(1)_____ the Main Turbine trips.

2) Based on the given conditions, per NOP-OP-1015, Event Notifications, the NRC Operations Center is required to be notified in no later than _____(2)_____ hours.

(Reference Provided)

A. 1) before
   2) 4 hrs.

B. 1) before
   2) 8 hrs.

C. 1) after
   2) 4 hrs.

D. 1) after
   2) 8 hrs.

Question 92

Answer: C

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 NRC notifications after a plant event occurs, and the required times to make the notification.

K/A is met by the steam pressure transmitter failing high causing the steam flow to fail high. This steam flow/feed mismatch will cause the feed flow to increase until a Full Feedwater Isolation occurs. The FWI will trip the turbine, which trips the Reactor requiring the notification of the NRC within 4-hours iaw NOP-OP-1015, Event Notifications.

A. Incorrect. Plausible for the reactor to trip first if the candidate thinks that the failed steam pressure transmitter will cause steam flow and feed flow to decrease causing a low SG water reactor trip at 20.5%. 4-hour notification is correct for the reactor trip.

B. Incorrect. Plausible for the reactor to trip first if the candidate thinks that the failed steam pressure transmitter will cause steam flow and feed flow to decrease causing a low SG water reactor trip at 20.5%. 8-hour notification is plausible if the candidate thinks a safety limit has been violated, or a FWI falls under 10CFR50.72 actuation, but neither are the case. A reactor trip requires a 4-hour notification.

C. Correct. The turbine will trip before the reactor. This is due to the pressure transmitter failing high which is used to calculate steam flow which will fail high also, causing a steam flow/feed mismatch. This mismatch will cause the MFRVs to open and feed the SG until the level increases to 92.2% which causes a Full Feedwater Isolation (P-14) which trips the turbine. Since the plant was at 100% power, the turbine trip will then cause the reactor trip. The NRC Ops Center must be notified in any event that results in the actuation of the Reactor Protection System (RPS) when the reactor is critical per pg. 34 of NOP-OP-1015.

D. Incorrect. The turbine will trip before the reactor is correct. 8-hour notification is plausible if the candidate thinks a safety limit has been violated, or a FWI falls under 10CFR50.72 actuation, but neither are the case. A reactor trip requires a 4-hour notification.

Sys #	System	Category	KA Statement
035	Steam Generator	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the SG; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Steam flow/feed mismatch

K/A# A2.04 **K/A Importance** 3.8 **Exam Level** SRO

References provided to Candidate NOP-OP-1015 Rev. 7 **Technical References:** 2SQS-24.1 PPNT Rev. 26 Slides 58, 100 NOP-OP-1015 Rev. 7 pg. 34

Question Source: Modified – Votgle 2015 Q88

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 3SQS-1.1, Rev. 8 Obj. 2. Describe the control, protection and interlock functions for the field 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.

2SQS-24.1, Rev. 26 Obj. 5. Given a change in plant conditions, describe the response of the Main Feedwater, Startup Feedwater, Auxiliary Feedwater System or Steam Generator Water Level Control system's field indication and control loops, including all automatic functions and changes in equipment status.

93. A load rejection has just occurred, and Operators are performing AOP 2.35.2, Load Rejection. The following conditions currently exist.

- Reactor power is 75% and STABLE.
- Bank D Control rods are at 145 steps.
- Annunciator A4-8D, Rod Control Bank D Low is LIT.
- Annunciator A4-6D, Delta Flux Outside Target Band is LIT.
- The RO is monitoring Delta Flux due to the transient.
- Axial Flux Difference Values are as follows:

N41	-18%
N42	-14%
N43	-15%
N44	-19%

1) Is Axial Flux Difference (AFD) within Tech Spec limits?

2) What action (if any) is required per the Alarm Response Procedures?

(Reference Provided)

A. 1) The AFD limits are NOT met.
2) Use 2OM-52.4.B, Load Following, reduce power to less than 50% within 30 minutes.

B. 1) The AFD limits are NOT met.
2) Use 2OM-7.4.Q, Emergency Boration, the Control Rods are below the Rod Insertion Limit.

C. 1) The AFD limits are met for the current Reactor power.
2) No action is required, maintain Reactor power ≤ 75%.

D. 1) The AFD limits are met for the current Reactor power.
2) No action is required, Reactor Power can be raised to ≤ 94%.

Question 93

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 specific knowledge of TS AFD LCO and bases so that they can analyze the data given and determine whether the Axial Flux Difference (AFD) Technical Specification had been violated after the control rods drove in on a load rejection.

K/A is met by demonstrating the candidate's ability to interpret excore AFD values after the control rods drove in due to a plant load rejection which generated a Delta Flux Outside Target Band alarm.

A. Correct. Two of the excore channels are outside Tech Spec limit of -16.5 to +15 at 75% power iaw CB-14. The candidate must know the associated note that states AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits. N41 & N44 are both outside the TS limit. Per ARP A4-6D when two channels are outside the target band, the direction is to restore or reduce power to <50 per 2OM-52.4.B.

B. Incorrect. First part is correct, AFD is outside TS limits. Plausible, ARP A4-6D has direction to restore AFD by adjusting boron, however the reason is not because the control rods are below RIL. The RIL limit at 75% power is 137 steps on Control Bank D.

C. Incorrect. Plausible if the applicant does not understand that the AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits. Plausible, Since the other AFD channel are not outside the limit they may believe that power is restricted to the current level, to avoid going outside the limit as power is increased.

D. Incorrect. Plausible if the applicant does not understand that the AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside the TS limits. Plausible, if the limit is confused with the QPTR limit, with one channel out by 2%, then the QPTR TS would limit power to 94% (2 %*3% = 6% from RTP).

Sys #	System	Category	KA Statement
001	Control Rod Drive	Generic	Ability to verify that the alarms are consistent with the plant conditions.

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

References provided to Candidate U2 Curve Book CB-14 Rev. 7; COLR Cycle 21 Figure 5.1-2 LRM Rev 94

Technical References: 2OM-2.4.AAL Rev. 6 pg.3; 2OM-1.4.AAN Rev. 4 pg 4; U2 Curve Book CB-14 Rev. 7; COLR Cycle 21 Figure 5.1-2 LRM Rev 94; Tech Spec 3.2.3 and bases pg. 3.2.3-1, B 3.2.3-2

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

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

94. Which of the following refueling activities must be performed by the Refueling Senior Reactor Operator (SRO) in accordance with 1/2RP-1.1, Refueling Administrative Section?

1. Coordinate plant operations with the Refueling Lead and Refueling Engineer to ensure that refueling operations are carried out in an orderly fashion.
2. Be stationed inside the containment during core alterations.
3. Assists the Refueling Lead and Refueling Engineer in the performance of their duties.
4. Inform the SM of any plant system or component condition that is not in accordance with the refueling operations limiting conditions for operation.
5. Responsible for Vendor oversight.
6. Closely monitor all operations involving raising and lowering fuel assemblies with the manipulator crane.

A. 1, 3, and 5 only

B. 2, 4, and 6 only

C. 1, 2, and 4 only

D. 3, 5, and 6 only

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.G first bullet on page 9. Specifically, the SRO must have specific knowledge of the refuel floor SRO responsibilities during plant refueling.

K/A met by requiring the SRO candidate to have knowledge of the fuel-handling SRO responsibilities iaw BVPS Refueling administrative procedures.

A. Incorrect. Plausible because these responsibilities are performed by other refuel positions. The SM coordinate plant operations with the Refueling Lead and Refueling Engineer to ensure that refueling operations are carried out in an orderly fashion. Refueling Assistant assists the Refueling Lead and Refueling Engineer in the performance of their duties. The Refueling Lead and Refueling Engineer are Responsible for Vendor oversight.

B. Correct. IAW 1/2RP-1.1 the Refuel SRO must be stationed inside the containment during core alterations, must inform the SM of any plant system or component condition that is not in accordance with the refueling operations limiting conditions for operation, and must closely monitor all operations involving raising and lowering fuel assemblies with the manipulator crane.

C. Incorrect. Plausible because coordinate plant operations with the Refueling Lead and Refueling Engineer to ensure that refueling operations are carried out in an orderly fashion sounds like a responsibility of the Refuel SRO but this is a SM responsibility. The other two are Refuel SRO responsibilities.

D. Incorrect. Plausible because assisting the Refueling Lead and Refueling Engineer in the performance of their duties, and Responsible for Vendor oversight sound like Refuel SRO responsibilities, but they are responsibilities of the Refueling Assistant and Refueling Lead and Refueling Engineer.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the fuel-handling responsibilities of SROs

K/A# 2.1.35 **K/A Importance** 3.9 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1/2RP-1.1 Iss 0 Rev 33 pg. 7-10

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(7)

Objective: 3SQS-6.14, Rev. 8 Obj. 2. STATE the refueling responsibilities of the following BVPS personnel: b. Refueling SRO

95. Which of the following are responsibilities to be performed by an Operations SRO per NOP-WM-4300, Order Execute Process?

1. Responsible for preparing, processing and revising Order packages.
2. Authorizing all Orders that are NOT pre-authorized.
3. Ensuring that the "scope of work" is adequately defined.
4. Ensuring all required Return to Service Testing has been completed and details are recorded as required in the Order hard copy.
5. Signing the Order hardcopy in the Operations Closure Review section.
6. Responsible for printing, assembling and statusing Order packages as necessary.

A. 1, 2, 3 only

B. 1, 3, 6 only

C. 2, 4, 5 only

D. 4, 5, 6 only

Question 95

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 administrative procedure for work authorization and the responsibilities of the Operations SRO in the work execution process.

K/A is met with the knowledge required to review, process, and authorize maintenance work orders iaw. NOP-WM-4300, Order Execution Process procedure.

A. Incorrect. Plausible because this answer is a mixture of responsibilities. #2 is a responsibility of the Operations SRO, but #1 is the responsibility of the Planner, and #3 is the responsibility of the Project Coordinator.

B. Incorrect. Plausible because this answer is a mixture of responsibilities, but #1 is the responsibility of the Planner, #3 is the responsibility of the Project Coordinator, and #6 is the responsibility of the Work Support Center/Planning and Support Personnel.

C. Correct. Per section 4.1.3, Operations SRO (i.e., Shift Manager, Unit Supervisor, Shift Engineer) are responsible for 2, 4, and 5.

D. Incorrect. Plausible because this answer is a mixture of responsibilities, #4 and #5 are responsibilities of the Operations SRO, but #6 is the responsibility of the Work Support Center/Planning and Support Personnel.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of maintenance work order requirements.

K/A# 2.2.19 **K/A Importance** 3.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** NOP-WM-4300 Rev 15 pg. 5-7

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 3SSG Admin Rev 9 Obj. 8. EXPLAIN the process for initiating and processing a Work Request and Work Order in accordance with: d. NOP-WM-4300, Order Execute Process

96. Given the following plant conditions:

- It is currently 0700 hours.
- A plant heatup from a refueling outage is currently in progress.
- Highest available RCS temperature is 325°F and RISING at a constant 5°F/hr.
- A Field Operator reports that the 'A' MDAFW Pump electrical breaker has a Ground Overcurrent Relay flagged.
- Engineering is performing a risk assessment; the evaluation will be complete by 1500 hours.

Which of the following completes the statements below?

1) The current Technical Specification operational MODE is _____(1)_____.

2) In accordance with Technical Specifications, the plant startup is _____(2)_____ to continue for the rest of the shift?

A. 1) Mode 3
   2) NOT allowed

B. 1) Mode 4
   2) NOT allowed

C. 1) Mode 3
   2) allowed

D. 1) Mode 4
   2) allowed

Question 96

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .B page 3 first and second bullet. Specifically, the SRO must make a determination of the current plant mode, and based on knowledge of TS 3.7.5 and LCO 3.0.4, make a determination that a mode change to mode 3 is not allowed iaw LCO 3.0.4.b because the mode change would occur before the risk assessment was completed.

K/A is met by giving plant conditions and determining the applicable Technical Specification Mode of Operation. This alone is RO knowledge, therefore the second part of the question was added to test knowledge of the specific system LCO and LCO 3.0.4 for the SRO to make a determination of whether a Mode change could be made while starting up the plant with a MDAFW pump inoperable. The SRO must make the determination that the 'A' MDAFW pump is inoperable due to the Ground Overcurrent Relay being flagged.

A. Incorrect. Incorrect Mode. Mode 3 is ≥350F. Mode change to Mode 3 is not allowed under LCO 3.0.4.b. (see correct answer)

B. Correct. Mode 4 is 350F>Tavg>200F. Mode change to Mode 3 is not allowed under LCO 3.0.4.b because the plant will reach mode 3 at 1200 hours at the current heatup rate of 5F/hr, and the risk assessment won't be complete until 1500 hours. (When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications. TS 3.7.5 does not identify any exceptions.

C. Incorrect. Incorrect Mode. Mode 3 is ≥350F. Mode change to Mode 3 is not allowed under LCO 3.0.4.b. Plausible if the candidate thinks that a mode change can occur iaw LCO3.0.4.b because it is in the process of being completed.

D. Incorrect. Mode 4 is correct for the given conditions. Mode change to Mode 3 is not allowed under LCO 3.0.4.b. Plausible if the candidate thinks that a mode change can occur iaw LCO3.0.4.b because it is in the process of being completed.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to determine Technical Specification Mode of Operation.

K/A# 2.2.35 **K/A Importance** 4.5 **Exam Level** SRO

References provided to Candidate None **Technical References:** TS Table 1.1-1 pg. 1.1-7; Tech Spec LCO 3.0.4 pg. 3.0-1 & 2; Tech Spec 3.7.5 pg. 3.7.5.1 & 2

Question Source: Bank – 1LOT16 Q96

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(2)

Objective: 3SQS-RULES ITS-01-01 Rev.3 - Given plant conditions, apply the rules of ITS Section 3.0 to ensure compliance with Technical Specifications

3SQS-RULES ITS-01-03 Rev.3 - Given plant conditions, determine the plant MODE in accordance with the ITS.

97. Unit 1 and Unit 2 are at 100% power with all systems in NSA.

- RWDA-L has been prepared for a Unit 1 Liquid Waste Discharge to Unit 1 Cooling Tower Blowdown

In accordance with 1/2OM-17.4A.D, Unit 1 Liquid Waste Discharge To Unit 1 Cooling Tower Blowdown, which of the items below does the 'APPROVED BY UNIT 2 SM/US' signature block on the RWDA-L denote?

1. Dilution flow will not be altered.
2. Midland Water Treatment Plant has been notified.
3. Unit 2 liquid Waste Discharge is not in progress.

A. 1 ONLY

B. 1 and 2 ONLY

C. 1 and 3 ONLY

D. 2 and 3 ONLY

Question 97

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .A page 17 fourth bullet. Conditions and limitations in the facility license. National Pollutant Discharge Elimination System (NPDES) requirements, if applicable. Specifically, what the SRO is signing for when approving a RWDA-L for a liquid waste discharge.

K/A is met by demonstrating the knowledge of the specific conditions which must be met for the Unit 2 Shift Manager/Unit Supervisor to sign the approval for the discharge release permits.

A. Incorrect. Plausible distractor because adequate dilution flow is correct, but the verification that no Unit 2 discharge is in progress or will not be started is also denoted by this signature.

B. Incorrect. Plausible distractor because Midland Water Treatment Plant has to be notified when an offsite radioactive release occurs as part of the offsite protective action recommendations, but not for discharging. Adequate dilution flow is correct, alarms must be properly adjusted also.

C. Correct. Per 1/2OM-17.4A.D, the note prior to the, procedural step for the U2 SM to sign the block states "Unit 2 SM/US signature on the RWDA-L denotes the following: 1. No Unit 2 liquid waste discharge is in progress, 2. No Unit 2 liquid waste discharge will begin UNTIL the Unit 1 RWDA-L has been terminated OR completed, 3. The proper dilution flow as provided by Unit 2 will not be altered..

D. Incorrect. Plausible distractor because Midland Water Treatment Plant has to be notified when an offsite radioactive release occurs as part of the offsite protective action recommendations, but not for discharging. The verification that no Unit 2 discharge is in progress or will not be started is denoted by this signature.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to approve release permits.

K/A# 2.3.6 **K/A Importance** 3.8 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1/2OM-17.4A.D Rev. 15 pg. 16

Question Source: Bank BV1LOT16 Q98

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.13 / 43.4 / 45.10)

Objective: 2SQS-17.1, Obj. 12 - 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.

3SSG-ADMIN-01-23: EXPLAIN the approval requirements for radiological release permits in accordance with 1/2-ENV-05.04, Radioactive Waste Discharge Authorization - Liquid. 1/2-ENV-05.05, Radioactive Waste Discharge Authorization - Gas.

98. The plant is in Mode 1. The crew is performing an emergency entry into Containment IAW 2OM-47.4.F, Emergent Containment Entry.

    1) IAW 2OM-47.4.F, who must authorize this Containment entry?
    2) Where must the Movable Incore Detectors (MIDS) be located for this entry prior to personnel entering Containment?

    _____(1)_____ authorization is required to make an emergent containment entry.

    The Movable Incore Detectors are required to be _____(2)_____.

A. 1) Shift Manager
   2) fully inserted in the core **only**

B. 1) Shift Manager
   2) fully inserted in the core **or** in their shielded storage location

C. 1) Radiation Protection Manager
   2) fully inserted in the core **only**

D. 1) Radiation Protection Manager
   2) fully inserted in the core **or** in their shielded storage location

Question 98

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.D second bullet on page 6. Specifically, the SRO must have specific knowledge of the content of the Emergent Containment Entry procedure identifying who must authorize the emergent containment entry, and understand to minimize the entrants radiation dose, the incore flux detectors must be shielded in their storage location or fully inserted in the core..

K/A is met by demonstrating the knowledge of who must authorize an emergent containment entry in accordance with plant procedures, and knowledge of the procedural requirements to protect the entrants from the radiological hazards associated with the movable incore detectors.

A. Incorrect. First part is correct. Second part is plausible because it would make sense that the activated detectors be fully inserted into the core to provide shielding for plant personnel when entering containment, but they may also be stored in their shielded storage location during entry.

B. Correct. The Shift Manager/Unit Supervisor must approve emergent containment entry. The P&Ls of 2OM-47.4.F, Emergent Containment Entry requires the Incore Detectors be fully inserted in the core or be stored in their shielded storage location during entry.

C. Incorrect. First part is incorrect, but plausible because the Radiation Protection Manager would be informed of work in containment and responsible for providing necessary Radiation Protection support for RBC entries during normal cnmt entries, but the RPM does not have to give authorization for entry. Second part is plausible because it would make sense that the activated detectors be fully inserted into the core to provide shielding for plant personnel when entering containment, but they may also be stored in their shielded storage location during entry.

D. Incorrect. First part is incorrect, but plausible because the Radiation Protection Manager would be informed of work in containment and responsible for providing necessary Radiation Protection support for RBC entries during normal cnmt entries, but the RPM does not have to give authorization for entry. Second part is correct.

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** 3.7 **Exam Level** SRO

References provided to Candidate None **Technical References:** 2OM-47.4.F Rev. 1 pg. 3,4

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(4)

Objective: 2SQS-47.1-01-02: Given a set of plant conditions and the appropriate procedure(s), summarize the operational sequence, parameter limits, precaution and limitations, and cautions and notes applicable necessary to complete the task.

Beaver Valley Unit 2 NRC Written Exam (2LOT19)
(SRO ONLY)

99. A plant transient has occurred.

    1) What procedure provides the guidance for Alarm Response Procedure use during this transient period?
    2) What are the requirements for returning to normal annunciator response?

A. 1) NOP-OP-1014, Plant Status Control
   2) Normal annunciator response will be resumed only after off-normal/emergency procedures are exited.

B. 1) NOP-OP-1014, Plant Status Control
   2) Normal annunciator response will be resumed when plant conditions permit at the Command SRO direction.

C. 1) NOP-OP-1002, Conduct of Operations
   2) Normal annunciator response will be resumed only after off-normal/emergency procedures are exited.

D. 1) NOP-OP-1002, Conduct of Operations
   2) Normal annunciator response will be resumed when plant conditions permit at the Command SRO direction.

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E page 7 third bullet. Specifically, the SRO must have knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. This question evaluates this knowledge by identifying when a transient is in progress, the need for Alarm Response may be put on hold until the Command SRO restores normal annunciator response.

K/A is met with the knowledge when normal alarm response may be suspended during transient conditions as set for in NOP-OP-1002, Conduct of Operations, and restored at the direction of the Command SRO.

A. Incorrect. Plausible because NOP-OP-1014 describes the process that Operations uses for plant equipment control, however the process for annunciator use during abnormal conditions is described in NOP-OP-1002. The second part is plausible but incorrect, exiting the abnormal or emergency procedures is not required to resume normal annunciator response.

B. Incorrect. First part is plausible as stated above. Second part is correct.

C. Incorrect. First part is correct. The second part is plausible but incorrect, exiting the abnormal or emergency procedures is not required to resume normal annunciator response.

D. Correct. Per NOP-OP-1002, when the plant is operating in AOPs or EOPs, the Transient annunciator's policy becomes the alarm response mode. The operators will only report significant annunciators to the Command SRO, which will be used by the Command SRO to evaluate potential escalation into higher E-Plan conditions. Once plant conditions permit, the Command SRO will inform the crew and they will then validate the current annunciators.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of abnormal condition procedures.

K/A# 2.4.11 **K/A Importance** 4.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** NOP-OP-1002 Rev. 13 pg. 62-63

Question Source: Bank – Browns Ferry 2015 NRC Exam Q99

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** 55.43.b(5)

Objective: 3SSG – Admin Rev. 9 Obj. DESCRIBE the process for disseminating operating shift instructions in accordance with NOP-OP-1002, Conduct of Operations.

Beaver Valley Unit 2 NRC Written Exam (2LOT19)
(SRO ONLY)

100 Given the following conditions:

- A general emergency is in progress.
- A site evacuation has just been ordered due to high airborne radiation levels on site.

Where do personnel in the Operations Support Center (OSC) go?

A. Personnel remain in the OSC in the Outage Central Area.

B. Personnel relocate to the Alternate OSC in the Unit 1 Process Rack area.

C. Personnel relocate to the Alternate TSC at the EOF in Chippewa Township.

D. Personnel relocate to the Beaver County Community College Golden Dome.

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E page 6 first paragraph. Specifically, the SRO must have knowledge of the content of the EPP procedures versus knowledge of the procedure's overall mitigative strategy or purpose. In this case, the SRO must have knowledge of where the Operations Support Center personnel will be located to in the event the emergency renders the OSC unusable due to radiological conditions.

K/A is met by demonstrating the knowledge that the Operations Support Center personnel will be directed to go to the alternate OSC in the event of a high airborne condition requiring a site evacuation.

A. Incorrect. Plausible because the Operations Support Center (OSC) is located in the Outage Central Complex above the BV 1 and 2 Control Rooms, but this area is not protected against radiological accidents.

B. Correct. The BV 1/2 Alternate Operations Support Center (OSC) is in the Process Instrumentation and Rod Position Instrumentation Area located below the BV-1 Control. This area is supplied by CR Emergency Ventilation System which contains HEPA and charcoal filters in the event of an emergency.

C. Incorrect. Plausible because the alternate TSC located offsite at the EOF is located approx. 10 miles from site. This area would not support a rapid response to mitigate site damage, therefore personnel with EP assignments remain on site.

D. Incorrect. Plausible because in the event of the site is being evacuated due to radiological conditions, the Beaver County Community College Golden Dome is a remote assembly area approx. 10 miles from site. OSC personnel would muster in the front of the EOF. This area would not support a rapid response to mitigate site damage, therefore personnel with EP assignments remain on site.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of emergency response facilities.

K/A# 2.4.42 **K/A Importance** 3.8 **Exam Level** SRO

References provided to Candidate None **Technical References:** EPP-PLAN-SECTION-7 Rev 31 sect. 7.1.2
1/2-EPP-IP-1.5 Rev 24 pg. 6

Question Source: Bank – 2LOT17 Q99 **(Last 2 exams)**

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** 55.43.b(5)

Objective: Epp-9250 Rev. 13 Obj. 12. State the purpose and locations of the on-site emergency response facilities at Beaver Valley Power Station.