Beaver Valley Unit 1 NRC Written Exam (1LOT18)

1. As power decays after a reactor trip, which of the following describes (1) the MINIMUM time that it takes for the Source Range instruments to energize, and (2) the condition that would **prevent** the Source Range instruments from energizing automatically when required?

A. 1) 5 - 10 minutes
2) Undercompensated Intermediate Range

B. 1) 5 - 10 minutes
2) Overcompensated Intermediate Range

C. 1) 11 - 15 minutes
2) Undercompensated Intermediate Range

D. 1) 11 - 15 minutes
2) Overcompensated Intermediate Range

Answer: C

Explanation/Justification: K/A is met with the candidates' knowledge of the approximate time after a reactor trip in which reactor power will decay low enough for the intermediate range power detectors to energize the source range detectors.

A. Incorrect. Time is too short. With prompt drop to about 5% power and negative 1/3 DPM thereafter, it should take about 13-15 minutes to get to the source range from 100%. Undercompensated causes the reading to be higher than actual, which could prevent SR from energizing.
B. Incorrect. Same as A above, except that an overcompensated channel would cause one input to SR energization to be active earlier, not later than required. Credible because there is a common misconception about the behavior of IR compensation, and the applicant must know the actuation logic for energizing SR instruments.
C. Correct. With prompt drop to about 5% power and negative 1/3 DPM thereafter, it should take about 13-15 minutes to get to the source range from 100%. Undercompensated causes the reading to be higher than actual, which could prevent SR from energizing.
D. Incorrect. Correct time, but incorrect compensation.

Sys #	System	Category	KA Statement
007	Reactor Trip	EK1 Knowledge of the operational implications of the following concepts as they apply to the reactor trip:	Decay power as a function of time

K/A# EK1.05 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** 3SQS-2.1 Rev 9, section VI.B.7

Question Source: Bank – 1LOT7 Q10

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** CFR 41.8 / 41.10 / 45.3

Objective: 3SQS-2.1 Obj 15 - Given an NIS failure, predict the NIS and interrelated system response for the given failure.

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 likely occurred?

A. Steam Generator feedwater line break outside Containment.

B. Charging flow control valve, FCV-1CH-122, failed open.

C. Steam Generator steamline break outside Containment.

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. a SG steamline 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.

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 20psig.

Sys #	**System**	**Category**	**KA Statement**
000008	Pressurizer (PZR) Vapor Space Accident	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:** 1LOT-M5D11 LP Rev 11 pgs. 9-11
1LOT-M5D11 Rev 11 Scenario #2

Question Source: Bank – South Texas 2011 NRC Exam Q66

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR 41.7 / 45.7)

Objective:

GO-ATA 4.2, Rev. 8	14. Compare and contrast a Pressurizer steam space LOCA with a small break LOCA in the cold leg. Include predicted plant response.
1LOT-M5D11, Rev. 11	2-1. Given a Pressurizer Safety failed open, respond in accordance with E-1 Loss of Reactor or Secondary Coolant

3. Given the following conditions:

- The plant was manually tripped due to a loss of Station Instrument Air
- A small break LOCA occurred after the reactor was tripped
- The crew is currently in ES-1.2, Post LOCA Cooldown and Depressurization, and AOP-1.34.1, Loss of Station Instrument Air
- PI-1IA-106, Station Instrument Air Header Pressure is 0 psig
- Safety Injection is in operation
- RCS pressure is 800 psig and stable

Which of the following valves will be used to dump steam from the Steam Generators during the RCS cooldown in ES-1.2?

A. All Condenser Steam Dumps

B. Only three Condenser Steam Dumps

C. SG Atmospheric Steam Dumps

D. Steam Generator Safety Valves

Answer: C

Explanation/Justification: K/A is met by the identifying that during a small break LOCA coincident with a loss of air, the plant can still be cooled down by alternate methods of operating the SG ASDs to reduce SG pressure (Secondary pressure).

A. Incorrect. Plausible distractor, but the MSLI valves will close on a loss of instrument air, therefore the Condenser Steam Dumps are not available.
B. Incorrect. Plausible distractor, but the MSLI valves will close on a loss of instrument air, therefore the Condenser Steam Dumps are not available.
C. Correct. SG Atmospheric Steam Dumps must be manually operated locally using nitrogen to cooldown the RCS due to the loss of instrument air as directed in AOP 1.34.1 step 12 and ES-1.2 step 13.d.
D. Incorrect. Plausible distractor because station instrument air is used by all the other choices, but the SG Atmospheric Steam Dumps can be manually operated when air is not available.

Sys #	**System**	**Category**	**KA Statement**
000009	Small Break LOCA / 3	EA1 Ability to operate and monitor the following as they apply to a small break LOCA:	Secondary pressure control

K/A# EA1.14 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.34.1 Rev 23 step 12

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR 41.7 / 45.5 / 45.6)

Objective: 1SQS-21.1, Rev. 16 Obj. 4. Given a Main Steam Supply System configuration and without referenced material, describe the Main Steam Supply System field response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable. a. Loss of instrument air

4. Given the following plant conditions:

- A large break LOCA has occurred
- AE bus is de-energized due to an electrical fault
- RCS pressure is 10 psig and stable
- All other equipment operated as designed

Complete the statements below.

1) The LHSI pump design flowrate at this pressure is approximately ______________.
2) The HHSI pump design flowrate at this pressure is approximately ______________.

A. 1) 2000 gpm
2) 150 gpm

B. 1) 2000 gpm
2) 550 gpm

C. 1) 4000 gpm
2) 150 gpm

D. 1) 4000 gpm
2) 550 gpm

Answer: D

Explanation/Justification: K/A is met by the operators required knowledge of the High Head and Low Head pumps design flowrate capability during a large break LOCA versus design flowrate at normal operating pressure.

A. Incorrect Plausible distractor since each Low Head pump flow on VB-A is approx. 2500 gpm per pump with two pumps running, due to minimal RCS pressure and both pump discharge headers combine into the injection header. One LHSI pump flow raises to ~3600 gpm when a pump is lost during a DBA LOCA. 150 gpm is plausible because it is the design flowrate of the HHSI pump at 2500 psig, not runout flowrate.

B. Incorrect. Plausible distractor since each Low Head pump flow on VB-A is approx. 2500 gpm per pump with two pumps running, due to minimal RCS pressure and both pump discharge headers combine into the injection header. One LHSI pump flow raises to ~3600 gpm when a pump is lost during a DBA LOCA. 550 gpm is correct for one HHSI pump operating at design runout flowrate.

C. Incorrect. 4000 gpm is correct. 150 gpm is plausible because it is the design flowrate of the HHSI pump at 2500 psig, not runout flowrate.

D. Correct. The LHSI pump design runout flowrate is approximately 4000 gpm, and the HHSI pump design runout flowrate is approximately 550 gpm.

Sys #	System	Category	KA Statement
011	Large Break LOCA	EK2 Knowledge of the interrelations between the and the following Large Break LOCA:	Pumps

K/A# EK2.02 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1SQS-11.1 Rev 14 PPNT Slides 49 & 83

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR 41.7 / 45.7)

Objective: 1SQS-11.1 Rev. 14 Obj. 17. List the nominal value of the control room operating parameters associated with the Safety Injection System.

5. The plant is at 100% power when TV-1CC-103C1, 1C RCP CCR Inlet Cnmt Isolation valve fails to 5% open.

   The following conditions exist:

   - A3-75, React Cool PP Lower Brg Lube Oil Cool Water Flow Low is LIT
   - A3-77, React Cool PP Stator Winding Cool Water Flow Low is LIT
   - A3-83, React Cool PP Upper Brg Lube Oil Cool Water Flow Low is LIT

   Given the following RCP temperatures, which one would warrant a trip of the RCP at this time?

A. Seal Leakoff temperature is 205°F and rising.

B. Stator Winding temperature is 325°F and rising.

C. Motor Bearing temperature is 185°F and rising.

D. Lower Radial Bearing temperature is 195°F and rising.

Answer: B

Explanation/Justification: K/A is met with the candidates' knowledge of the ARP trip criteria for the RCP on high stator temperature caused by a loss of primary component cooling water flow to the RCP.

A. Incorrect. Plausible because it is an immediate RCP trip criterion identified on the LHP of AOP 1.6.8, but the required trip temperature is >225F.

B. Correct. ARP A3-77 states that unless directed otherwise by SM/US, IF stator temperature > 248F, perform the following in mode 1, trip the Rx and stop the RCP. >325 was selected as the correct choice because of the following note in the ARP. RCP operation may continue until stator winding temperature increases to 248F (280F with RCS temperature < 350F) SM/US may authorize continued operation with stator temperature to 310F for several hours.

C. Incorrect. Plausible because it is an immediate RCP trip criterion identified on the LHP of AOP 1.6.8, but the required trip temperature is >195F.

D. Incorrect. Plausible because it is an immediate RCP trip criterion identified on the LHP of AOP 1.6.8, but the required trip temperature is >210F.

Sys #	System	Category	KA Statement
000015	Reactor Coolant Pump Malfunctions / 4	AA2. Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow):	When to secure RCPs on high stator temperatures

K/A# AA2.09 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.4.AAT Rev 3 pg. 2
1OM-53C.4.1.6.8 Rev. 19 LHP

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR 43.5 / 45.13)

Objective: 1SQS-6.3, Rev. 15 16. List the nominal value of the control room operating parameters associated with the Reactor Coolant Pump and support systems.

1SQS-53C.1, Rev. 12 1. State all Immediate Operator Actions associated with the AOPs

6. The plant is in Mode 4, cooling down for a refueling outage.

- RHR is in service with both pumps and heat exchangers in service.
- MOV-1RH-605, RHR HX Bypass FCV is in MANUAL.
- RCS pressure is 300 psig.
- TE-RH-604, RHR COOLER INLET temperature is 205°F.

'A' RHR pump trips due to an electrical fault.

- Component cooling water flow to the heat exchangers remains constant.
- The Operator adjusts MOV-1RH-605 to reduce system flow to 3900 gpm.

1) How will TE-RH-606, RHR COOLER OUTLET temperature respond to this failure?

2) How will the RCS cooldown rate be affected by this failure?

RHR COOLER OUTLET temperature will ___(1)___, and RCS cooldown rate will ___(2)___.

A. 1) rise
2) increase

B. 1) rise
2) decrease

C. 1) lower
2) increase

D. 1) lower
2) decrease

Answer: D

Explanation/Justification: K/A is met by requiring the candidate to determine how the RHR heat exchanger outlet temperature will respond after one RHR pump trips causing RHR system flow to be reduced, thus RHR Cooler Outlet temperature indicator lowering.

A. Incorrect. Plausible distractor if the candidate doesn't recognize that the CCR flowrate remains unchanged, and the RHR flowrate has been reduced by half. Second part is incorrect.

B. Incorrect. Plausible distractor if the candidate doesn't recognize that the CCR flowrate remains unchanged, and the RHR flowrate has been reduced by half. Second part is correct.

C. Incorrect. Hx outlet temperature will lower. Second part is plausible if the candidate thinks that the outlet temp goes down that it will cause the cooldown rate to go up, but with the RHR flowrate reduced by ½, this is not the case.

D. Correct. RHR loop return is the outlet of the Hxs, and by reducing the flow through the Hxs due to the pump loss, the outlet temperature will lower during this event. $m1cp1\Delta T1 - m2cp2\Delta T2$, knowing that cp is assumed constant, and inlet temp will initially remain approx. the same. Cooldown rate will also decrease because the volume of water that is mixing with the RCS has decreased.

Sys #	System	Category	KA Statement
025	Loss of Residual Heat Removal System (RHRS)	AA1. Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System:	RHR cooler inlet and outlet temperature indicators

K/A# AA1.08 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** GO-GPF.C3 Rev 1 Iss 1 pg. 23-26
U1 RM-0410-001 Rev. 15

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR 41.7 / 45.5 / 45.6)

Objective: GO-GPF.C3, Rev. 1 Obj. 10. Identify the relationship between flow rates and temperature.

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

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

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

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

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

Answer: A

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

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

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

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

References provided to Candidate None **Technical References:** 1OM-53B.4.FR-Z.1 Iss 3 Rev 0 Step 3

Question Source: Bank - Comanche Peak 2012 NRC Exam Q46

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR 41.5,41.10 / 45.6 / 45.13)

Objective: 3SQS-1.1, Rev. 8 1. Describe the function of the Reactor Protection System Trip Logics & Engineered Safeguards Features Actuation Signals and the associated major components as documented in Operating Manual, 1(2)OM-1.

8. Given the following:

- Reactor power is 100%
- PRZR Master Pressure Controller output drifts low

Five minutes after the failure, RCS subcooling will be ______(1)______ AND PRZR subcooling will be ______(2)______.

A. same, higher

B. higher, same

C. same, same

D. higher, higher

Answer: B

Explanation/Justification: K/A is met by failing the PRZR Master Pressure Controller output low and having the candidates determine the effect on the RCS subcooling, and the PRZR which is at saturation temperature.

A. Incorrect. Plausible if the student reverses the change in pressure and temperature between the RCS and przr.

B. Correct - output lowering will cause heaters to turn on to raise pressure and sprays would not respond to control pressure since the whole system is controlled by the output of the master pressure controller. A PORV will open after pressure gets to setpoint to stop the pressure rise. Since pressure is now higher and temperature is the same RCS subcooling is higher and the przr pressure and temperature is higher so subcooling would be same.

C. Incorrect. Plausible if the student confuses the fact that the PZR will not change subcooling and thinks the RCS will follow what the przr does.

D. Incorrect. Plausible if the student confuses pressure and temperature in the przr since that system will stay at saturation.

Sys #	System	Category	KA Statement
000027	Pressurizer Pressure Control System Malfunction	AK1. Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions:	Definition of saturation temperature

K/A# AK1.01 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** GO-GPF.T3, Rev. 0 App. A pgs. 49-50; 1OM-6.4.IF Rev 11 pg. 23 & 24

Question Source: Bank – Wolf Creek 2015 NRC Exam Q7

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR 41.8 / 41.10 / 45.3

Objective:

GO-GPF.T3, Rev. 0	11. Define the following terms: n. Subcooled and compressed liquids & o. Subcooling
1SQS-6.4 Rev 14	17. Describe the control, protection and interlock functions for the control room components associated with the Pressurizer and Pressurizer Relief System, including automatic functions, setpoints and changes in equipment status as applicable.

9. The Reactor was manually tripped due to a Steam Generator Tube Rupture.
   - A Loss of Offsite power occurred during the Rx trip
   - The crew is performing step 7, RCS cooldown IAW E-3, Steam Generator Tube Rupture.
   - The STA reports an Orange path on FR-P.1, Vessel Integrity.

   Which of the following is the correct action to be taken by the crew?

A. Stop the RCS cooldown and transition to FR-P.1. Perform all actions of FR-P.1.

B. Complete the RCS cooldown to target temperature, then transition to FR-P.1.

C. Complete the RCS cooldown. Re-evaluate Vessel Integrity after SI Termination in E-3.

D. Stop the RCS cooldown until the FR-P.1 Orange path condition clears, then recommence the RCS cooldown.

Answer: C

Explanation/Justification: K/A is met with the knowledge of the bases to the CAUTION prior to step 7 of E-3, Steam Generator Tube Rupture. The bases require that the candidate must know that during a cooldown with no RCPs running, the ruptured loop Tcold indication is not accurate due to the redirection of SI flow in the loop.

A. Incorrect. Plausible distractor is the candidate does not recall the caution and transitions to the Orange path as is expected under the EOP rules of usage. Normal EOP rules of usage do not apply in this case because of the specific caution for the Integrity status tree.

B. Incorrect. Plausible distractor is the candidate recalls a caution prior to the RCS cooldown, but thinks the caution only applies during the cooldown to target temperature, then the transition can be made.

C. Correct. The candidate must recognize that with the loss of offsite power, that no RCPs will be running. Therefore, with a natural circ cooldown, reverse flow through the ruptured loop during the cooldown or when a PORV is opened, the Safety Injection flowpath in the ruptured loop could change making Tcold in that loop inaccurate. The caution before step 7 states to disregard the ruptured loop Tcold ad vessel integrity until after SI termination later in the SGTR procedure.

D. Incorrect. Plausible distractor is the candidate recalls a caution prior to the RCS cooldown, but doesn't remember if a transition is warranted.

Sys #	System	Category	KA Statement
000038	Steam Generator Tube Rupture (SGTR)	Generic	Knowledge of the operational implications of EOP warnings, cautions, and notes.

K/A# 2.4.20 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.E-3 Iss 3 Rev 2 pg. 9
1OM-53B.4.E-3 Iss 3 Rev 2 pg. 66

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.13)

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

10. The plant was operating at 100% power when a loss of all 4KV power occurs.
    - ECA-0.0, Loss of All Emergency 4KV AC Power is in progress
    - Station Instrument Air header is 0 psig
    - CNMT Instrument Air header is 20 psig and lowering

    Which of the following components is available for RCS or Secondary pressure control from the control room?

A. PRZR PORVs **only**.

B. SG Atmospheric Steam Dump Valves **only**.

C. **Both** PRZR PORVs **and** SG Atmospheric Steam Dump Valves.

D. **Neither** the PRZR PORVs **or** SG Atmospheric Steam Dump Valves.

Answer: A

Explanation/Justification: K/A is met by demonstrating the ability to determine which valves can be operated from the control room during a station blackout with a loss of instrument air.

A. Correct. The PORVs will still be able to be cycled from the control room due to DC power will be available to the solenoids, and N2 Accumulators interconnected with the air supply to the PORVS is supplied as a backup.
B. Incorrect. Plausible because the candidate may recognize that the ASDs still have power (DC) to operate the valves, but may not know that they are air operated, or they may think there is an automatic nitrogen backup for these valves instead of a local manual lineup of nitrogen.
C. Incorrect. Plausible if the candidate thinks there is a nitrogen backup for both of these valves.
D. Incorrect. Plausible because the candidate may not recall the nitrogen supply to the przr porvs.

Sys #	System	Category	KA Statement
000055	Station Blackout	EA2 Ability to determine or interpret the following as they apply to a Station Blackout:	Existing valve positioning on a loss of instrument air system

K/A# EA2.01 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.1.D Iss 4 Rev 1 pg. 9
U1 RM-0406-002 Rev 24
U1 RM-0411-002 Rev 17
U1 RM-0421-001 Rev 28

Question Source: New

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

Objective: 1SQS-6.4, Rev. 14 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. a. Loss of instrument air, b. Loss of electrical power

11. Reactor power is at 35% with all systems in normal alignment **EXCEPT**:
    - Rod control is in MANUAL

    If a loss of Vital Bus 2 occurs, which of the following instruments will lose indication?

    1) PI-1LM-100C, Containment Pressure Instrument
    2) PI-1RC-445, Pressurizer Pressure Instrument
    3) NI-1NI-36, Intermediate Range Detector
    4) LI-1RC-460, PRZR Level Instrument

A. 1 and 2 only

B. 3 and 4 only

C. 1 and 3 only

D. 2 and 4 only

Answer: B

Explanation/Justification: K/A is met by demonstrating how the function of supplying power from vital bus 2 to the various instruments is lost during a loss of vital bus 2.

A. Incorrect. PI-1LM-100C and PI-1RC-445 are both channel III instruments powered from Vital bus 3.
B. Correct. NI-1NI-36 and LI-1RC-460 are both channel II instruments powered from Vital bus 2.
C. Incorrect. NI-1NI-36 is powered from Vital bus 2, but PI-1LM-100C is powered from Vital bus 3.
D. Incorrect. LI-1RC-460 is powered from Vital bus 2, but PI-1RC-445 is powered from Vital bus 3.

Sys #	**System**	**Category**	**KA Statement**
000057	Loss of Vital AC Instrument Bus	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:** 1OM-2.3.C iss 4 Rev 3 pg. 2
1OM-6.3.C Rev 11 pg. 50

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7)

Objective: 3SQS-38.1, Rev. 8 Obj. 2. From memory, describe the Normal System Arrangement for the Emergency 120 VAC Distribution Systems, including distribution paths, status of feeder breakers, loads, bus transfer switches, power train, and bus designation.

Obj. 6. From memory, list the controls and instrumentation associated with the 120 VAC Distribution System located in the Field.

12. The plant is in Mode 3 with all systems in normal alignment for this Mode.

- 125V DC Bus 1 Voltage is ZERO
- Subsequently, a Loss of Offsite power occurs

How will these conditions impact #1 EDG start capabilities?

EDG #1 _____(1)_____ be **MANUALLY** started from the control room by depressing both STOP pushbuttons, then selecting EXERCISE, then depressing the START pushbutton

AND

EDG #1 _____(2)_____ be **LOCALLY** started by placing the AUTO/LOCAL switch to LOCAL, then depressing the ENGINE START pushbutton.

A. (1) CAN
(2) CANNOT

B. (1) CAN
(2) CAN

C. (1) CANNOT
(2) CANNOT

D. (1) CANNOT
(2) CAN

Answer: C

Explanation/Justification: K/A is met by the candidate demonstrating the knowledge to understand the effect on the DGs when a loss of control power occurs due to a loss of 125V DC Bus 1.

A. Incorrect. Plausible if candidate believes that only the local manual start circuitry is impacted by these conditions

B. Incorrect. Plausible if candidate believes that only the Auto start feature is impacted by these conditions.

C. Correct. The EDG start solenoids will not have power and cannot be started from the CR or locally. The sequence for all of the manual /local starts are all correct for starting the EDG if there was power available to the solenoids.

D. Incorrect. Plausible if candidate believes that only the CR manual start circuitry is impacted by these conditions.

Sys #	System	Category	KA Statement
000058	Loss of DC Power	AK3. Knowledge of the reasons for the following responses as they apply to the Loss of DC Power:	Use of dc control power by D/Gs

K/A# AK3.01 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.39.1A Att. 1 Rev 9 page 20

Question Source: Bank – 1LOT14 Q50 **(LAST 2 EXAMS)**

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR 41.5,41.10 / 45.6 / 45.1)

Objective: 1SQS-36.2, Rev. 19

3. Identify the power supplies for the EDG components which are powered from the class 1E electrical distribution system. (For the 4160v system, include the power train and bus designation. For the 480v system, include only the power train.)

6. Given an EDG configuration and without referenced material, describe the EDG field response to the loss of electrical power, including automatic functions and changes in equipment status as applicable.

13. Given the following conditions:

- The plant is at 75% power with all systems in normal alignment for this power level.
- A barge has struck the Intake Structure causing severe damage to the components inside Cubicles A, B, and C.
- 1WR-P-6B, TP RW 6B Pump is running with normal discharge pressure.
- The crew has entered AOP 1.30.2, River Water/Main Intake Structure Loss.

Which of the following actions must be taken by the crew IAW AOP 1.30.2?

A. Manually trip the reactor and enter E-0, Reactor Trip or Safety Injection because NO River Water is available for equipment cooling.

B. Manually trip the reactor and enter E-0, Reactor Trip or Safety Injection, and crosstie Circulating Water system to the RPRW header.

C. Check that both Auxiliary River Water pumps automatically started on low pressure, and MOV-1RW-116A & B, Aux RW Pump Supply to A(B) Header RP RW automatically opened.

D. Manually start an Auxiliary River Water pump and verify open MOV-1RW-116A(B), Aux RW Pump Supply to A(B) Header RP RW.

Answer: D

Explanation/Justification: K/A is met with the knowledge that if the Rx Plant River Water is lost, the AOP gives guidance to manually start and align the Aux River Water pumps which are in the alternate intake structure. BV1 EOPs do not address loss of nuclear service water, therefore the AOP was used for the question.

A. Incorrect. Plausible because if no RW was available iaw AOP step 2 the Rx is to be tripped, but the candidate must know the purpose of the Aux RW system, and know that it is available.

B. Incorrect. Plausible because the Rx would be tripped if no RW was available, and cross tying the Circ Water system to the Turb Plant RW is identified as step 7 in the AOP if the Rx is tripped and RW was not recovered.

C. Incorrect. Plausible because the Aux RW pumps will supply the RW loads, but there are no automatic features associated with the Auxiliary River Water pumps. The associated MOV-1RW-116A(B) would open automatically on a pump start.

D. Correct. The Aux RW pumps must be manually operated because there are no auto starts, and the associated MOV-1RW-116A(B) will open automatically on pump start. Knowing that the auxiliary river water system will be put into operation whenever the river water system is unable to supply the required cooling water when the intake structure is disabled is the reason for the AOP starting the Aux RW pumps and establishing cooling for plant operation. They must also know that there is an Auxiliary Intake Structure which is separate, and upriver of the Intake Structure.

Sys #	System	Category	KA Statement
000062	Loss of Nuclear Service Water	AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water:	Guidance actions contained in EOP for Loss of nuclear service water

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

References provided to Candidate None **Technical References:** 1OM-53C.4.1.30.2 Rev 9 pg. 2
U1 RM-0430-001 Rev 37

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR 41.4, 41.8 / 45.7)

Objective: 1SQS-30.2, Rev. 18 Obj. 17. Given a specific plant condition, describe the response of the Reactor Plant River 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.

14. The plant is operating at 100% power.

    - A Loss of Station Air occurs
    - You are performing AOP-1.34.1, Loss of Station Instrument Air

    IF Station Instrument Air header pressure continues to lower, AOP-1.34.1 requires a manual reactor trip.

    IAW AOP-1.34.1, which of the following conditions **REQUIRES** you to perform a manual reactor trip?

A. As soon as air pressure drops below 65 psig.

B. As soon as air pressure drops below 75 psig.

C. Annunciator A1-56 "STEAMLINE STOP VALVE NOT FULLY OPEN" – LIT.

D. Annunciator A6-99, "STA AIR COMPR 1A RCVR TANK PRESS LOW" – LIT.

Answer: C

Explanation/Justification: K/A is met with the knowledge that the Loss of Instrument Air AOP directs the manual tripping of the reactor (plant shutdown) when air pressure has lowered top the point that the MSIVs are not full open which causes Annunciator A1-56, STEAMLINE STOP VALVE NOT FULLY OPEN to be in alarm.

A. Incorrect. This is the Unit 2 Air pressure that requires a reactor trip
B. Incorrect. This is a pressure selected to be plausible with 65 psig.
C. Correct. According to AOP 1.34.1 step 10.
D. Incorrect. This is an entry condition annunciator but not a trip criterion.

Sys #	System	Category	KA Statement
000065	Loss of Instrument Air	AA2. Ability to determine and interpret the following as they apply to the Loss of Instrument Air:	When to commence plant shutdown if instrument air pressure is decreasing

K/A# AA2.05 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.34.1 Rev 23 pg. 6

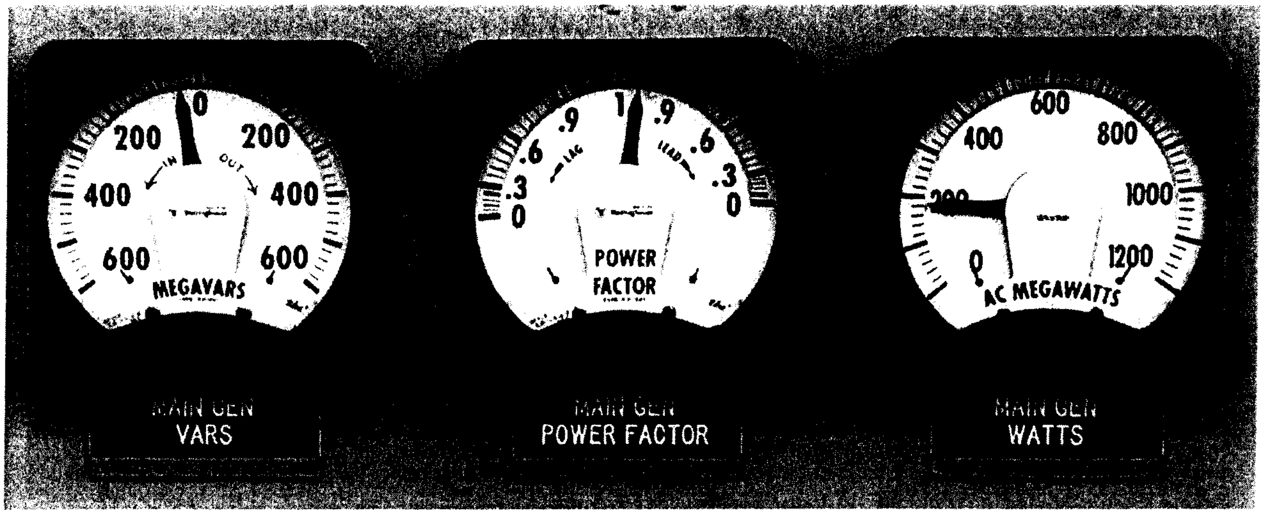
Question Source: Bank - BV1LOT16 AUDIT Exam Q50

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 43.5 / 45.13)

Objective: 1SQS-34.1, Rev. 15 Obj. 16. Given a Compressed Air System alarm condition and using the Alarm Response Procedure(s), determine the appropriate alarm response, including automatic and operator actions in the control room.

15. The crew has entered AOP-1/2.35.1, Degraded Grid after observing grid instabilities. The grid is currently stable, and the crew is performing step 7, checking the Main Generator is operating properly.

The plant is at 25% power, and Hydrogen pressure is 60 psig and stable.



Based on the above indications, what, if anything, must be done to the Main Generator to ensure proper operation IAW OM-35, Main Generator and Transformer?

A. Raise Voltage

B. Lower Voltage

C. Raise Megawatts

D. Maintain current conditions

Question 15

Answer: A

Explanation/Justification: K/A is met by requiring the candidate to demonstrate knowledge of main generator under-excitation indications and demonstrating the ability to correct the condition while in the Degraded Grid AOP.

A. Correct. MVARS IN and Power Factor leading are indications that the generator is under-excited. By raising voltage using the voltage adjuster, the Megavars will raise to MVARS OUT and the generator will be operating in the over-excited region iaw the Main Generator Capability Curve. This is RO operational knowledge and is reinforced by general fundamentals and 1OM-35.4.B P&L #3.
B. Incorrect. Lowering voltage will raise MVARS in the IN direction and make the generator more under-excited.
C. Incorrect. Raising generator load will not clear the under-excited condition but is plausible if the candidate thinks that Megawatts must be raised to maintain the generator on or above the Power Factor line due to a misunderstanding of the Main Generator Capability Curve.
D. Incorrect. Plausible if the candidate does not recognize that the generator is operating in the MVARS IN region of the Main Generator Capability Curve indicating that the generator is under-excited or thinks that this is the expected way of operating the generator.

Sys #	System	Category	KA Statement
000077	Generator Voltage and Electric Grid Disturbances	AK1. Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances:	Under-excitation

K/A# AK1.03 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None

Technical References: 1/2OM-53C.4A.35.1 Rev 10 pg. 5
1OM-35.5.A.14 Rev. 5
1OM-35.4.B Rev. 17 pg. 2

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)

Objective: 1SQS-35.3, Rev. 10 Obj. 3 Describe the control, protection and interlock functions for the control room components associated with the Main Generator and Exciter System, including automatic functions, setpoints and changes in equipment status as applicable.

16. Given the following:

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

Which of the following indications is used to determine if the leak has been isolated in accordance with ECA-1.2, and why?

A. RCS pressure, because when the break is isolated, SI flow will repressurize the RCS.

B. Pressurizer level, because when the break is isolated, PRZR level will be the first indication of the increase in RCS inventory.

C. Safety injection flow, because when the break is isolated, it is the first parameter that will change.

D. RVLIS indication, because when the break is isolated, vessel head voiding will immediately be reduced.

Answer: A

Explanation/Justification: K/A is met with the knowledge that monitoring the RCS pressure for a pressure rise during ES-1.2 will indicate that the desired results of the EOP are obtained when the leak is isolated.

A. Correct. RCS pressure is the primary means of determining whether the leak is isolated per ECA-1.2 step 4, and Major Action Category 3 as described in ECA-1.2 background.
B. Incorrect. RCS inventory will increase, but may not immediately show up on przr level.
C. Incorrect. SI Flow is a good confirmatory indication when RCS pressure rises, because it will be reduced, but RCS pressure rise is the only immediate indication.
D. Incorrect. RVLIS may indicate 100% at the start, so may not provide indication of isolation at all.

Sys #	**System**	**Category**	**KA Statement**
WE04	LOCA Outside Containment	EA1. Ability to operate and / or monitor the following as they apply to the (LOCA Outside Containment)	Desired operating results during abnormal and emergency situations.

K/A# EA1.3 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None

Technical References: 1OM-53A.1.ECA-1.2, Iss 3 Rev 0 step 4
1OM-53B.4.ECA-1.2, Iss 3 Rev 0 pg. 2

Question Source: Bank – 1LOT7 Q61

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 / 45.6)

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.

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

Plant conditions are as follows:

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

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

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

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

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

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

Answer: C

Explanation/Justification: K/A is met with the knowledge of the sequence of restoring feedwater, including resetting interlocks, iaw FR-H.1, Response to Loss of Secondary Heat Sink to restore SG heat sink.

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

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

C. Correct. To establish MFW after the SI has been reset requires the FWI to be reset to allow the BFRVs to be opened to have feedwater aligned to the SGs. FR-H.1 step 7b RNO and step 7c identifies the sequence.

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

Sys #	System	Category	KA Statement
WE05	Inadequate Heat Transfer—Loss of Secondary Heat Sink	EK2. Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following:	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

K/A# EK2.1 **K/A Importance** 3.7 **Exam Level** RO

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

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 45.7)

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

18. Given the following conditions:

- A Large Break LOCA has occurred.
- Loss of AE Emergency Bus occurred during the transfer to offsite power.
- The crew are currently performing ES-1.3. Transfer to Cold Leg Recirculation.

Which of the following conditions in ES-1.3 will require a transition to ECA-1.1, Loss of Emergency Coolant recirculation?

A. A loss of all offsite power.

B. LI-RS-151A/B, CNMT Sump Level is 58 inches.

C. MOV-1SI-860B, B LHSI Pump Rx CNMT Sump Suction valve will not open.

D. 'B' LHSI pump is cavitating after the transfer to Recirculation.

Answer: C

Explanation/Justification: K/A is met with the candidates' ability to determine the need to transition to ECA-1.1, Loss of Emergency Coolant Recirculation when required conditions are not met during ES-1.3, transfer to Cold Leg Recirculation.

A. Incorrect. Plausible if the candidate doesn't recognize that a loss of offsite power to AE bus does not de-energize the AE bus because the diesel is still available to power the required single train of SI equipment.

B. Incorrect. Plausible distractor because the ES-1.3 step 3 entry into ECA-1.1 is containment sump level less than 48", not 58".

C. Correct. Step 5 verifies SI system aligned for recirculation, and with 'A' train being unable to align due to the loss of AE bus, and 'B' LHSI Pump Rx CNMT Sump Suction valve closed, there will be no flowpath established from the sump and ECA-1.1 must be entered. This is RO level knowledge based on required entry conditions for ECA-1.1.

D. Incorrect. Plausible because ES-1.3 step 6 states if the LHSI pumps are cavitating, then stop charging pumps and if signs of screen blockage, transition to SBCRG-1, Sump Blockage Control Room Guideline.

Sys #	System	Category	KA Statement
WE11	Loss of Emergency Coolant Recirculation	EA2. Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation)	Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A# EA2.1 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.ES-1.3 Iss 3 Rev 0
1OM-53A.1.ECA-1.1 Iss 3 Rev 0 pg 1
1OM-53A.1.1-G Rev 2

Question Source: New

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

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.

19. Initial conditions:

- Power Range N-41 is reading 60% power
- Main Turbine is in in '1st STG OUT'
- AFD is +0.9%
- Control Bank 'D' is at 161 steps

The RO pulls rods 2 steps out for Tavg control, when he releases the switch, the Control Bank 'D' rods continue to step out to 181 steps with NO operator action.

What effect will this continuous rod withdrawal have on Axial Flux Distribution, and Power Range N-41 (if any), 5 minutes after the event?

1) Axial Flux Distribution will become ______________.

2) Power Range N-41 will ______________.

A. 1) more positive
   2) remain at 60%

B. 1) more positive
   2) increase

C. 1) more negative
   2) remain at 60%

D. 1) more negative
   2) increase

Answer: B

Explanation/Justification: K/A is met by the candidate demonstrating an understanding of the effects that a continuous rod withdrawal will have on both axial flux distribution and power level.

A. Incorrect. First part is correct. Second part is plausible if the candidate thinks that rod movement at power will only change Tavg, and not power.

B. Correct. AFD will become more positive because the power (flux) will be higher in the top of the core, and the candidate must know that AFD is Power top half of core – Power Bottom half of core. With the rods being pulled 20 steps, AFD will shift to the right in the doghouse which is more positive. N-41 power will increase due to the positive reactivity added by the control rods and the turbine controls being in 1st stage OUT which does not use 1st stage pressure as an input to control the turbine load.

C. Incorrect. Plausible if the candidate doesn't understand AFD, and thinks of it in reverse. Second part is plausible if the candidate thinks that rod movement at power will only change Tavg, and not power.

D. Incorrect. Plausible if the candidate doesn't understand AFD, and thinks of it in reverse. Second part is correct.

Sys #	System	Category	KA Statement
000001	Continuous Rod Withdrawal	AK1. Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal:	Effects of power level and control position on flux

K/A# AK1.07 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 3SQS-2.1 PPNT Rev 9 slides 145-148
GO-3ATA-4.1 U1 Rev 5 Iss 1 Slide 25

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR 41.8 / 41.10 / 45.3)

Objective: 3SQS-2.1, Rev. 9 Obj. 12. Define Quadrant Power Tilt Ratio (QPTR) and Axial Flux Difference (AFD).

GO-3ATA 4.1, Rev. 5 Obj. 2. Explain plant response to the below listed accidents: a. RCCA Bank/Rod Withdrawal

20. Given the following plant conditions:

- The plant is in Refueling Mode with systems aligned for core off-load.
- While lowering a spent fuel assembly into the Spent Fuel Pool, the assembly ruptures and releases **ALL** of the gasses from **ALL** of the rods in that assembly **ONLY**.
- RM-1RM-207, Fuel Pool Bridge Area Radiation Monitor goes into **HIGH** alarm.

Based on these plant conditions, what will be the status of RM-1VS-103B, Fuel Building Ventilation Exhaust B Monitor **AND** the reason why?

RM-1VS-103B ______(1)______ be in **HIGH** alarm because ______(2)______.

A. 1) will
   2) the iodine and xenon released from the fuel assembly will **NOT** be sufficiently scrubbed out by the water above the assembly

B. 1) will
   2) RM-1VS-103B is designed to detect gamma radiation (GM tube)

C. 1) will **NOT**
   2) the iodine and xenon released from the fuel assembly **WILL** be sufficiently scrubbed out by the water above the assembly

D. 1) will **NOT**
   2) RM-1VS-103B is designed to detect beta radiation (scintillation)

Answer: A

Explanation/Justification: K/A is met with the knowledge that in a fuel handling accident that causes the area rad monitor to alarm, the ventilation rad monitor will also alarm due to the noble gasses released from the accident.

A. Correct. When the area rad monitor RM-207 alarms due to a fuel handling accident, RM103B will also alarm because the noble gas activity released from the damaged rods is all released to the building with no removal by pool water scrubbing (The majority of Iodine will be scrubbed by the water). The rad monitor RM-1VS-103B will detect the airborne activity and alarm also. The UFSAR analysis states quite clearly that activity will be released into the buildings (Fuel or Containment) for the a fuel handling accident of this type. Therefore both detectors will be in HIGH alarm.

B. Incorrect. Correct RM-1VS-103B response, however, the detector type is incorrect. RM-1VS-103B is NOT a GM tube, it is a Beta scintillation.

C. Incorrect. The UFSAR analysis states quite clearly that activity will be release into the buildings for a fuel handling accident of this type. Second part is incorrect because the noble gasses will not be scrubbed by the fuel pool water.

D. Incorrect. First part is incorrect. The second part is plausible because even though the listed monitor type is correct (RM-1VS-103B is a scintillation detector), the fact that the area monitor went into an alarm condition implies that enough activity was released into the area to raise the activity level sensed in the ventilation duct.

Sys #	System	Category	KA Statement
000036	Fuel-Handling Incidents	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:** GO-3ATA-4.3 U1 PPT Rev 6 Slides 78-83
1SQS43.1 PPNT Rev 16 slide 112
GO-3ATA-4.3 SHO Rev 6 pg. 24

Question Source: Bank – 2LOT8 NRC Exam Q19

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 45.3)

Objective: GO-3ATA 4.3, Rev. 6 Obj. 2. Identify the "worst" case of initial conditions or, given a parameter, identify which direction of its magnitude would be "worse" for initial conditions for each listed accident.

21. The plant is operating at 100% power and multiple radiation alarms indicate a tube leak.

- The crew is in AOP-1.6.4, Steam Generator Tube Leakage
- Currently performing step 7 Check VCT level Trend
- RCS temperature and PRZR level are stable
- VCT level has lowered 1.5% over a 3-minute trend

Based on the above conditions, answer the following questions.

1) What is the Primary to secondary leak rate?
2) What procedure does AOP-1.6.4 direct using to shutdown the plant for these conditions?

A. 1) 7 gpm
2) Trip the Reactor, enter E-0, Reactor Trip or Safety Injection.

B. 1) 14 gpm
2) Trip the Reactor, enter E-0, Reactor Trip or Safety Injection.

C. 1) 7 gpm
2) Shutdown the plant using AOP-1.51.1, Unplanned Power Reduction.

D. 1) 14 gpm
2) Shutdown the plant using AOP-1.51.1, Unplanned Power Reduction.

Answer: C

Explanation/Justification: K/A is met by the knowledge of the SG tube leak procedure evaluating the water inventory balance and knowing the basic system knowledge of the VCT, then, based on the leak rate determination, decide what procedure will be used to shutdown the plant.

A. Incorrect. 7 gpm is correct. Rx trip is plausible if the candidate feels that a 7 gpm tube leak warrants it due to excessive RCS leakage. Neither the tube leak procedure or the excessive RCS leakage procedure calls for a Rx trip.
B. Incorrect. 14 gpm is plausible since it is the volume of 1% of the VCT. Rx trip is plausible if the candidate feels that a 7 gpm tube leak warrants it due to excessive RCS leakage. Neither the tube leak procedure or the excessive RCS leakage procedure calls for a Rx trip.
C. Correct. VCT trend of 1.5% over 3 minutes, is 0.5% per minute, therefore, based on the note prior to step 7 which states 1% VCT equals 14 gallons, 7gpm is the correct answer. At a leak rate of 7 gpm, the charging pump can maintain przr level, which was stable during the evaluation. The AOP states in action level 3 to commence a prompt and controlled shutdown. Step 11 identifies AOP-1.51.1 if >20% power.
D. Incorrect. 14 gpm is plausible since it is the volume of 1% of the VCT. AOP-1.51.1 is correct.

Sys #	System	Category	KA Statement
000037	Steam Generator Tube Leak	AK3. Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak:	Actions contained in procedures for radiation monitoring, RCS water inventory balance, S/G tube failure, and plant shutdown

K/A# AK3.05 **K/A Importance** 3.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.6.4 Rev 29 pgs. 6-8, 36

Question Source: New

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

Objective: 1SQS-53C.1, Rev. 12 Obj. 2. State the conditions or symptoms that would require entry into the AOPs.
Obj. 4. Explain the basis for cautions and notes of each AOP.

22. Given the following conditions:

- The plant is operating at 60% power when air leakage into the condenser resulted in a rising condenser backpressure.
- A load reduction is initiated at a rate of 5% per minute in accordance with AOP-1.51.1, Unplanned Power Reduction.
  - Three minutes after the load reduction was commenced, condenser backpressure has risen to 8.1 In. Hg. Abs. and is continuing to rise.
  - Eight minutes after the load reduction was commenced, condenser backpressure has risen to 9.1 In. Hg. Abs. and is continuing to slowly rise.

What operator action is required by AOP-1.26.2, 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-1.26.1, 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 < 8" HgA 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 8.1" HgA. Therefore, tripping the turbine and entering the Turbine and Generator Trip AOP is the correct course of action since vacuum was not restored to < 8' HgA 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 < 8" HgA 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. The Turbine trip setpoint has been reached (10" HgA), therefore the turbine should be tripped

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, the turbine trip setpoint has been reached (10" HgA), therefore the turbine should be tripped.

Sys #	System	Category	KA Statement
000051	Loss of Condenser Vacuum	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:** 1OM-53C.4.1.26.2 Rev 5 pg. 2

Question Source: Modified - 1LOT5 NRC Exam Q50

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

Objective: 1SQS-53C.1, Rev. 12 Obj. 5. Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.

Obj. 6. Given a set of conditions, apply the correct AOP.

23. The plant was operating at 100% power when the Shift Manager entered AOP-1.33.1A, Control Room Inaccessibility due to heavy smoke in the Control Room.

- All control room actions have been taken IAW AOP-1.33.1A.
- The SM is at step 23, Establish Boration Control.
- The RO does not remember seeing MOV-1CH-350, Emergency Boration Isolation valve indicate OPEN prior to leaving the Control Room.

1) How can the position of MOV-1CH-350 be determined when establishing Boration Control at the Emergency Shutdown Panel (SDP) in AOP-1.33.1A?

2) How many Boric Acid Transfer Pumps can have control transferred to the Emergency Shutdown Panel (SDP)?

A. 1) Valve position indication on the Emergency Shutdown Panel
2) One

B. 1) Local valve position at MOV-1CH-350.
2) One

C. 1) Local valve position at MOV-1CH-350.
2) Two

D. 1) Valve position indication on the Emergency Shutdown Panel
2) Two

Answer: C

Explanation/Justification: K/A is met by evaluating the candidates' ability to monitor the emergency boration valve MOV-350 position and control the Boric Acid Transfer Pumps from the shutdown panel after the Control Room has been evacuated due to a fire.

A. Incorrect. First part is plausible because several valve positions are indicated on the SDP, but MOV350 is not. Second part is plausible because the candidate may think of the other CR evac procedure which only requires one train of equipment.

B. Incorrect. First part is correct. Second part is plausible because the candidate may think of the other CR evac procedure which only requires one train of equipment.

C. Correct. Since the CR is inaccessible, the only way to verify the position of MOV-1CH-350 is locally in the blender room. The candidate must know that the valve is outside of cnmt. Both boric acid transfer pumps are capable of being transferred to the SDP iaw AOP-1.33.1A pg 11.

D. Incorrect. First part is plausible because several valve positions are indicated on the SDP, but MOV350 is not. Second part is correct.

Sys #	System	Category	KA Statement
000068	Control Room Evacuation	AA1. Ability to operate and / or monitor the following as they apply to the Control Room Evacuation:	Emergency borate valve controls and indicators

K/A# AA1.11 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.33.1A Rev 13 pg 11
U1 RE-0025M
1SQS-7.1 Rev 20 pg. 12

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR 41.7 / 45.5 / 45.6)

Objective: 1SQS-7.1, Rev. 20 Obj. 13. In the field, locate all of the components identified on the Nuclear Operator Normal-System-Arrangement System Flow path drawing. Locate those components accessible during normal operation within the ALARA considerations.

Obj. 14. In the field, locate the following control functions for the Chemical and Volume Control System. Locate those controls accessible during normal operation within the ALARA considerations. a. Local controls as indicated in 1OM-7.3.C, Power Supply and Control Switch List.

24. The crew is responding to High RCS Activity in accordance with AOP 1.6.6, High Reactor Coolant System Activity.
    - Plant is at 100% power
    - Chemistry is currently sampling the RCS, no results have been reported
    - RM-1CH-101B, Reactor Coolant Letdown Low Range Monitor is in High-High alarm
    - RM-1CH-101A, Reactor Coolant Letdown High Range Monitor is in High-High alarm

    What action will be taken by the crew due to the Letdown Radiation monitors being in High-High alarm, and what is the reason for this action?

A. Increase Letdown flow to accelerate RCS activity cleanup through the mixed bed demineralizers.

B. Commence RCS Degasification to remove radioactive gases from the RCS.

C. Perform Unplanned Power Reduction IAW AOP 1.51.1, then manually initiate CIA to preclude the release of potentially high airborne and elevated radiation levels to the environment.

D. Stop the CNMT Sump pumps and CNMT Vacuum pumps to prevent the release of potentially high airborne and elevated radiation levels to the Auxiliary Building.

Answer: D

Explanation/Justification: K/A met by requiring the candidate to know the action required by the High Reactor Coolant Activity AOP, and the reason for this action, if the High or Low Letdown radiation monitor is in alarm.

A. Incorrect. Plausible distractor if the candidate thinks increasing letdown flow will clean up the activity faster, but at 100% power 2 letdown orifices are currently in service. P&L for letdown states L/D flow shall not exceed 120gpm, and 2 60gpm orifices should not be placed in service at the same time. AOP step 4 recommends evaluating reducing Letdown flow.

B. Incorrect. Plausible if the candidate thinks removal of the radioactive gases from the RCS will minimize the effect of the High activity in the RCS, but in this case it would actually increase the radiological effects in the Aux building, and not keep it contained in the RCS or cnmt. AOP step 4 recommends evaluating stopping of RCS degasification.

C. Incorrect. Plausible if candidate feels that the plant must be shutdown, but without actual activity and chemistry results available to compare to the Tech Spec and LRM limits, this would not be the appropriate action. Initiating a CIA after shutdown is plausible if the candidate feels that both of these alarms will lead to a release which could be detrimental to the public and containment isolation is required. CIA is not directed in the ARPs or AOPs.

D. Correct. IAW the AOP, if either the LOW or HIGH Letdown rad monitors are in High-High alarm, the sump pumps and vacuum pumps should be stopped to minimize the radiological effects to the Aux building.

Sys #	**System**	**Category**	**KA Statement**
000076	High Reactor Coolant Activity	AK3. Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity	Corrective actions as a result of high fission-product radioactivity level in the RCS

K/A# AK3.05 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.6.6, Rev. 4 step 6

Question Source: Bank – 1LOT16 Q22 **(LAST 2 EXAMS)**

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR 41.5,41.10 / 45.6 / 45.13)

Objective: 1SQS-53C.1 Rev. 12 Obj. 5 - Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.

25. Given the following plant conditions:
    - A Small Break LOCA occurred.
    - ES-1.2, Post LOCA Cooldown and Depressurization is in progress.
    - 'A' RCP is running **only**.
    - Both HHSI pumps are running in injection mode.
    - Operators have been directed by ES-1.2 to depressurize the RCS to refill the Pressurizer.

    Which of the following will provide the most effective depressurization based on current plant conditions?

A. Open MOV-1CH-311, PRZR AUX SPRAY ISOLATION VALVE **only**.

B. Open PCV-1RC-455A, PRZR SPRAY VALVE **only**.

C. Open PCV-1RC-455B, PRZR SPRAY VALVE **only**.

D. Open **both** PCV-1RC-455A and PCV-1RC-455B, PRZR SPRAY VALVES.

Answer: B

Explanation/Justification: K/A is met by demonstrating the ability to determine which depressurization method is best for the plant conditions to obtain the desired results during post LOCA Cooldown and Depressurization.

A. Incorrect. Plausible because Auxiliary Spray can be used for depressurization, but incorrect because Auxiliary Spray cannot be used for depressurization unless HHSI flow is secured and normal charging is in service.

B. Correct. With "A" RCP running, PCV-1RC-455A provides the most effective spray flow.

C. Incorrect. Plausible because PCV-1RC-455B would be the most effective method with the 'C' RCP running, but incorrect because with only the 'A' RCP running, there will be no driving head across the 455B. candidates must also know which spray valve is associated with which loop.

D. Incorrect. Plausible because if 'C' RCP was the only RCP running, opening both PCV-1RC-455A and 455B would provide spray through both of the spray valves due to the surge line being connected to the 'C' loop. With only the 'A' RCP running, the spray valve associated with the inactive loop must remain shut to prevent robbing flow from the valve from an active loop. Incorrect because opening both PCV-455A and PCV-455B is less effective than opening PCV-1RC-455A only.

Sys #	System	Category	KA Statement
WE03	LOCA Cooldown-Depressurization	EA1. Ability to operate and / or monitor the following as they apply to the (LOCA Cooldown and Depressurization)	Desired operating results during abnormal and emergency situations.

K/A# EA1.3 **K/A Importance** 3.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.ES-1.2 Iss 3 Rev 1 pg 11; 1OM-53B.4.ES-1.2 Iss 3 Rev 1 pg 31; 1OM-6.2.A Rev 20 P&L 52

Question Source: Bank – VC Summer NRC Exam 2013 Q70

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 / 45.6)

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.

Obj. 4. Explain from memory the basis for ALL Cautions and Notes, IAW BVPS-EOP Executive Volume.

26. The crew is performing FR-P.1, Response to Imminent Pressurized Thermal Shock Condition, and are performing step 24, Establish Letdown. It has been determined that Normal Letdown can not be established, and Excess Letdown must be placed in service.

    1) IAW 1OM-7.4.H, what valve must be energized to place Excess Letdown in service?

    Excess Letdown has been placed in service, and the crew is performing a required RCS temperature soak IAW step 30.

    2) Which of the following component/system actions is **permitted** to be performed during this soak period?

A. 1) MOV-1RC-556A,B, **or** C; 1A,B,C RCL Fill Valve.
   2) Energize Pressurizer Heaters.

B. 1) MOV-1RC-556A,B, **or** C; 1A,B,C RCL Fill Valve.
   2) Place Auxiliary Spray in service.

C. 1) MOV-1RC-557A,B, **or** C; 1A, B, or C RCL Drain Valve.
   2) Energize Pressurizer Heaters.

D. 1) MOV-1RC-557A,B, **or** C; 1A, B, or C RCL Drain Valve.
   2) Place Auxiliary Spray in service.

Answer: D

Explanation/Justification: K/A is met by evaluating the candidates' knowledge of local field operator actions to place excess letdown in service as required by FR-P.1 when normal letdown cannot be established.

A. Incorrect. Plausible distractor because the RCL Fill valves are used as an alternate charging path in FR-P.1 step 17. Second part is plausible if they think it is required to raise pressure but cooling down or raising pressure is not permitted during the soak period.
B. Incorrect. Plausible distractor because the RCL Fill valves are used as an alternate charging path in FR-P.1 step 17. Second part is correct.
C. Incorrect. RCL Drain Valve is correct for excess letdown. Second part is plausible if they think it is required to raise pressure but cooling down or raising pressure is not permitted during the soak period.
D. Correct. One RCL Drain Valve must be energized locally at MCC1-17, 18, or 19, then must be opened from the control room to provide a flowpath for excess letdown. Auxiliary Spray may be placed in service to reduce pressure while maintaining RCS pressure and cold leg temperatures within the limits of Att. 5-F.

Sys #	System	Category	KA Statement
WE08	RCS Overcooling-Pressurized Thermal Shock	Generic	Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

K/A# 2.4.35 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-P.1 Iss 3 Rev 0 pgs. 15, 16, 19
1OM-53B.4.FR-P.1 Iss 3 Rev 0 pg 50
1OM-7.4.H Rev 6 pg 3

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.13)

Objective: 1SQS-7.1, Rev. 20 — 14. In the field, locate the following control functions for the Chemical and Volume Control System. Locate those controls accessible during normal operation within the ALARA considerations. a. Local controls as indicated in 1OM-7.3.C, Power Supply and Control Switch List.

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

27. A Large Break LOCA has occurred. The crew is performing ES-1.3, Transfer to Cold Leg Recirculation.

The following conditions exist:

- RX Cnmt Sump Pumps DA-P-4A(B) are both in AUTO
- Containment Sump Discharge Cnmt Isolation Valves TV-1DA-100A(B) are both in AUTO
- Cnmt sump level is 75 inches and rising
- Cnmt pressure is 6.2 psig and lowering

1) Which of the following will cause the RX Cnmt Sump Pumps to stop pumping down the cnmt sump?

2) Entry conditions into FR-Z.2, Response to Containment Flooding ___________ been met?

A. 1) CIA closes the Containment Sump Discharge Cnmt Isolation Valves.
2) have

B. 1) RX Cnmt Sump Pumps motor breakers shunt trip on Cnmt Sump level high.
2) have

C. 1) CIA closes the Containment Sump Discharge Cnmt Isolation Valves.
2) have NOT

D. 1) RX Cnmt Sump Pumps motor breakers shunt trip on Cnmt Sump level high.
2) have NOT

Answer: C

Explanation/Justification: K/A is met by the candidate demonstrating knowledge of the controls and interlocks of the containment sump pumps during containment flooding. Specifically, that the cnmt sump pump discharge valves automatically close with a CIA actuation (SI signal generated), thereby not allowing the sump pumps to lower cnmt level.

A. Incorrect. First part is correct. Plausible distractor if the candidate does not know that entry into FR-Z.1 is directed at ≥ 81" in the cnmt sump.

B. Incorrect. Plausible distractor is the candidate thinks that cnmt sump pump breakers trip on sump level. This is not correct as the pumps will still operate, but the discharge valves will be closed due to CIA actuation and cnmt pump down will not occur. Plausible distractor if the candidate does not know that entry into FR-Z.1 is directed at ≥ 81" in the cnmt sump.

C. Correct. Since cnmt pressure is 6.2 psig in the stem it is known that CIA has occurred due to SI actuation. Both TV-1DA-100A & B, cnmt sump pump valves are CIA valves and will be closed. The candidate must know that entry conditions into FR-Z.1 Cnmt Flooding requires >81 inches.

D. Incorrect. Plausible distractor is the candidate thinks that cnmt sump pump breakers trip on sump level. This is not correct as the pumps will still operate, but the discharge valves will be closed due to CIA actuation and cnmt pump down will not occur. Second part is correct.

Sys #	System	Category	KA Statement
WE15	Containment Flooding	EK2. Knowledge of the interrelations between the (Containment Flooding) and the following:	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

K/A# EK2.1 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-9.1.D Iss 4 Rev 0 pg. 2
1OM-53A.1.F-0.5 Iss 3 Rev 0
1OM-9.1.C Rev 1 pg.4

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** CFR: 41.7 / 45.7

Objective: 1SQS-9.1, Rev. 11 — Obj. 14. Given a Containment Isolation Actuation (CIA) signal, evaluate the response of the Primary Vents and Drains System.

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.

28. Which of the following describes a function of the flywheel on the RCPs?

A. Prolongs RCP coastdown time to aid in maintaining loop flow thus maintaining hot channel factors at an acceptable level during certain loss of RCS flow events.

B. Prolongs RCP coastdown time to aid in maintaining loop flow thus maintaining DNBR within acceptable limits during certain loss of flow events.

C. Maintains constant RCP speed, minimizing the potential for spurious RCS low flow reactor trips and maintaining hot channel factors at an acceptable level during power operation.

D. Maintains constant RCP speed, minimizing the potential for spurious RCS low flow reactor trips and maintaining DNBR within limits during power operation.

Answer: B

Explanation/Justification: K/A is met by demonstrating the knowledge of the effects that the RCP flywheel has during RCP coast down and how this assists in maintaining DNBR within acceptable limits during certain loss of flow events.

A. Incorrect. Flywheel designed to provide inertia to aid DNBR, not specifically for hot channel factors. Hot channel factors are affected by control rods.
B. Correct.
C. Incorrect. Flywheel more important for loss of flow, where RCP coastdown time is important for heat removal. Hot channel factors are affected by control rods.
D. Incorrect. RCS flow is a consideration for DNBR, but at rated RCP speed, the flywheel inertia is insignificant in performing the function of maintaining flow.

Sys #	System	Category	KA Statement
003	Reactor Coolant Pump	K5 Knowledge of the operational implications of the following concepts as they apply to the RCPS:	Effects of RCP coastdown on RCS parameters

K/A# K5.02 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1SQS-6.3 rev 15 Iss 1 pgs. 41-42

Question Source: Bank – BV2 2005 NRC Exam Q1

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.5 / 45.7)

Objective: 1SQS-6.3, Rev. 15 Obj. 19. Given a Reactor Coolant Pump and support system configuration and without referenced material, describe the Reactor Coolant Pump and support 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

29. The plant is in Mode 3 preparing to enter Mode 2. All systems are in normal alignment for this condition.

    The following indications are present in the control room:

    - Charging Pump Press, PI-1CH-121 is indicating 2500 psig and stable
    - RCP1A Seal Injection, FI-1CH-130 is indicating 9.5 gpm and stable
    - RCP1B Seal Injection, FI-1CH-127 is indicating 8.75 gpm and stable
    - RCP1C Seal Injection, FI-1CH-124 is indicating 9.25 gpm and stable
    - RCS pressure is 2235 psig and stable
    - HCV-1CH-186, RCP Seal Supply valve is full open

    Based on these conditions, is Tech. Spec. LCO 3.5.5 Seal Injection Flow being met, and why?

A. Yes, all parameters meet LCO 3.5.5

B. No, Charging Pump Discharge Pressure is below minimum LCO pressure.

C. No, Seal Injection Flow exceeds the LCO flowrate.

D. No, Charging Pump Discharge Pressure exceeds the LCO pressure.

Answer: A

Explanation/Justification: K/A is met by determining the operability of the CVCS seal injection flowrate in accordance with LCO 3.5.5.

A. Correct. LCO 3.5.5 requires reactor coolant pump seal injection flow shall be ≤ 28 gpm with charging pump discharge pressure ≥ 2457 psig and the seal injection flow control valve full open. The total from the stem is 27.5 gpm.

B. Incorrect. LCO 3.5.5 has been met. Plausible if the candidate does not know that the required pump discharge pressure must be ≥ 2457 psig.

C. Incorrect. LCO 3.5.5 has been met. Plausible if the candidate does not know that total seal injection flow must be ≤28 gpm, or adds the injection flowrates incorrectly. The total from the stem is 27.5 gpm.

D. Incorrect. LCO 3.5.5 has been met. Plausible if the candidate thinks that the required pump discharge pressure must be ≤ 2457 psig.

Sys #	System	Category	KA Statement
004	Chemical and Volume Control	Generic	Ability to determine operability and/or availability of safety related equipment.

K/A# 2.2.37 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** TS 3.5.5

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 43.5 / 45.12)

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

30. What is the basis for sampling the RCS for Fluoride and Oxygen, and how are these chemistry parameters controlled when the plant is operating at 100% power?

1) The basis for sampling the RCS for Fluoride and Oxygen is to ______________.
2) Fluorides are controlled by the ______________ Demineralizers, and Oxygen is controlled by maintaining ________________ cover gas on the VCT.

A. 1) minimize corrosion in the RCS which could lead to RCS system failure.
2) Mixed Bed; Hydrogen

B. 1) minimize corrosion in the RCS which could lead to RCS system failure.
2) Cation Bed; Nitrogen

C. 1) minimize activity in the RCS which could lead to increased dose rates.
2) Cation Bed; Nitrogen

D. 1) minimize activity in the RCS which could lead to increased dose rates.
2) Mixed Bed; Hydrogen

Answer: A

Explanation/Justification: K/A is met by evaluating the knowledge of basis for sampling the RCS for Fluorides and Oxygen, and identifying how the Fluorides and Oxygen are controlled by the Chemical and Volume Control system.

A. Correct. RCS chemistry is sampled and controlled to ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion (LRM). Mixed bed demineralizers will remove ionic isotopes such as Cl, Fl, and Na. RCS Oxygen created by the dissociation of water is scavenged by the Hydrogen cover gas in the VCT.

B. Incorrect. First part is correct. Second part is plausible because the cation bed (de-lithiating demin) is normally used at chemistry's request to remove Lithium, but negative ions such as CL and Fl will not be removed by a cation bed, and Nitrogen gas on the VCT is not used for oxygen control at power. Nitrogen is only used as a cover gas when shutting down to displace Hydrogen.

C. Incorrect. First part is plausible if the candidate thinks Fluoride and Oxygen are fission products leading to higher RCS activity. Second part is plausible as stated above.

D. Incorrect. First part is plausible if the candidate thinks Fluoride and Oxygen are fission products leading to higher RCS activity. Second part is correct.

Sys #	**System**	**Category**	**KA Statement**
004	Chemical and Volume Control	K5 Knowledge of the operational implications of the following concepts as they apply to the CVCS:	Reason for sampling for chloride, fluoride, sodium and solids in RCS

K/A# K5.29 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-7.1.C Rev 8 pg. 10 & 13
LRM B3.4.2-1 Rev 56
1SQS-7.1 LP Rev 20 pg. 4 & 16

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.5/45.7)

Objective: 1SQS-7.1, Rev. 20 Obj. 1 - Describe the function of the Chemical and Volume Control System and the associated major components as documented in Operating Manual Chapter 7.

31. Given the following:

- The plant is in Mode 4, cooling down for refueling.
- 'A' RHR Pump and heat exchanger are in service.
- The auto setpoint on MOV-1RH-605, RHR Heat Exchanger Bypass Flow Control Valve drifts LOW.

Which of the following describes the effect on the RCS cooldown rate?

The RCS cooldown rate...

A. increases, due to the increased flow through the RHR heat exchanger.

B. lowers, due to the decreased total flow through the RHR system.

C. lowers, due to the decreased flow through the RHR heat exchanger.

D. increases, due to the increased total flow through the RHR system.

Answer: A

Explanation/Justification: K/A is met by demonstrating the knowledge of how the cooldown rate is affected when MOV-1RH-605, RHR Heat Exchanger Bypass Flow Control Valve drifts low causing the more RHR flow to flow through the RHR HX.

A. Correct. Less bypass flow with a lower setpoint for total flow, results in higher percentage of RHR flow through the heat exchanger, which results in a lower average temperature of RHR returning to RCS.
B. Incorrect. Rate rises, but total flow decreases, not increases. Opposite effect for both parameters
C. Incorrect. Cooldown rate will rise. If setpoint drifts low, less total flow will result in more RHR heat exchanger flow.
D. Incorrect. Cooldown rate increases. Reason is incorrect; Higher flow through the heat exchanger, but not through the RHR system.

Sys #	System	Category	KA Statement
005	Residual Heat Removal	K4 Knowledge of RHRS design feature(s) and/or interlock(s) which provide or the following:	RHR heat exchanger bypass flow control

K/A# K4.03 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM 10.1.B, pg 2

Question Source: Bank – 1LOT7 NRC Exam Q6

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7)

Objective: 1SQS-10.1 Obj 18

32. Given the following:

- 480VAC MCC1-E3 is deenergized due to a fault.
- All other equipment is in the at-power NSA configuration when the reactor automatically trips on low Pressurizer pressure.
- Safety Injection has actuated.

Based on the above conditions, which of the following describes the status of High Head Safety Injection (HHSI) pumps suction supply, and Cold Leg Injection alignments after SI alignment is complete?

A. Charging pumps will taking suction from both the VCT and RWST, and discharging to the RCS Cold legs.

B. Charging pumps will taking suction from the VCT ONLY and discharging to the RCS Cold legs.

C. Charging pumps will taking suction from the RWST ONLY and discharging to the RCS Cold legs.

D. Charging pumps will taking suction from the RWST ONLY and the BIT will remain isolated.

Answer: C

Explanation/Justification: K/A is met with the knowledge of the power supplies to the High Head Safety Injection pumps suction supply valves and system flowpaths.

A. Incorrect. Plausible if the candidate knows E3 powers the 115B & 115C, but doesn't recognize that VCT isolations are in series.

B. Incorrect. Plausible if the candidate knows E3 powers the 115B & 115C, but thinks VCT charging pump suction is aligned in parallel, and the RWST is aligned in series.

C. Correct. The candidate must know that MOV-1CH-115B and MOV-1CH-115C will be de-energized but that VCT suction line is isolated by 115C & E in series, and 115B & D are in parallel. They must also know that the MOV867s are powered from E5 & E6, and will not be affected.

D. Incorrect. Plausible if the candidate knows E3 powers the 115B & 115C, and how they are aligned, but doesn't know that E3 does not affect the 867 valves because they're powered from E5 & E6.

Sys #	System	Category	KA Statement
006	Emergency Core Cooling	K2 Knowledge of bus power supplies to the following:	ESFAS-operated valves

K/A# K2.04 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None

Technical References: 1OM-7.3.C Rev 13 pg. 3
1OM-11.3.C Rev 8 pgs. 3-4
1SQS-11.1 LOIT PPNT Rev 14 slide 53

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7)

Objective: 1SQS-11.1 Rev. 14

Obj. 5. Identify the power supplies for the components identified on the Normal-System-Arrangement System Flow-path drawing which are powered from the class 1E electrical distribution system. (For the 4160v system, include the power train and bus designation. For the 480v system, include only the power train.)

Obj. 20. Given a Safety Injection System configuration and without referenced material, describe the Safety Injection System control room response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable. a. Loss of electrical power

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

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

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

A. 1) automatically opens
   2) automatically opens

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

C. 1) is NSA open
   2) automatically opens

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

Answer: D

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

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

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

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

References provided to Candidate None **Technical References:** 1OM-6.4.AAB Rev 2 pg. 2; 1OM-6.1.C Rev 7 pg. 28

Question Source: Bank – 2LOT7 NRC Exam Q33

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7)

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

34. The plant is operating at 100% power when an inadvertent CIA occurs.

What effect will the CIA have on the following CCR flow indicators?

FI-1CC-118, Reactor Plant Component Cooling **8 INCH** Header Inlet Flow will ____(1)____.
FI-1CC-107A, 'A' RCP Thermal Barrier Flow will ____(2)____.

A. 1) lower
2) lower to ZERO

B. 1) lower
2) remain the same

C. 1) remain the same
2) lower to ZERO

D. 1) remain the same
2) remain the same

Answer: B

Explanation/Justification: K/A is met by requiring the candidate to predict what happens to Component Cooling Water (CCR) flowrates in one of the three system headers, and the RCP thermal barrier flowrate if an inadvertent CIA occurs.

A. Incorrect. First part is correct. The second part is plausible because the thermal barrier flow would isolate on a CIB.

B. Correct. The 8" header flow will lower due to several components being isolated, only leaving the NRHX and Seal Water HX as loads on the 8" header. The A RCP Thermal Barrier Flow will remain the same because only a CIB would affect the isolation valves TV-103A, 103A1, 107E1, and 107E2.

C. Incorrect. First part is plausible if the candidate does not know that several loads on the 8" header are isolated on a CIA. The second part is plausible because the thermal barrier flow would isolate on a CIB.

D. Incorrect. First part is plausible if the candidate does not know that several loads on the 8" header are isolated on a CIA. Second part is correct.

Sys #	**System**	**Category**	**KA Statement**
008	Component Cooling Water	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including:	CCW flow rate

K/A# A1.01 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.1-B Rev 2
U1 RM-0415-001 Rev 23
U1 RM-0415-002 Rev 13
U1 RM-0415-005 Rev 14
1SQS-15.1 PPNT Rev. 14 slide 16

Question Source: New

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

Objective: 1SQS-15.1, Rev. 14 Obj. 15 - Given a plant CIA signal, breakdown how the CCR System valve, pump, flow and/or electrical configuration will change as a result of the signal.

35. The plant is at 75% power with all systems in normal alignment for this power level when a Loss of Vital Bus III occurs.

Which of the following components will lose power during the Loss of **Vital Bus III**?

No Operator actions have occurred.

A. Control Room Annunciator Panels

B. Pressurizer Valve Position Monitoring System (VMS)

C. LT-1QS-100D, Refueling Water Storage Tank level instrument

D. PT-1MS-447, HP Turbine First Stage pressure instrument

Answer: B

Explanation/Justification: K/A is met by demonstrating the knowledge of the power supplies to the Pressurizer Valve Position Monitoring System which provides CR indication for the PRZR Code Safety Valves by requiring the candidates to identify what component will lose power during a loss of vital bus III.

A. Incorrect. Control Room Annunciator Panels are powered from Vital Bus II. Plausible because it is powered from one of the four vital busses.

B. Correct. The Pressurizer Valve Position Monitoring System (VMS), which provides indication of PRZR Code Safety Valves on VB-B is NSA powered from Vital Bus III. There is a transfer switch to transfer power from VB-IV which is why the statement of no operator actions have occurred is in the question stem.

C. Incorrect. LT-1QS-100D, Refueling Water Storage Tank level instrument is powered from Vital Bus II. Plausible because it is powered from one of the four vital busses and the letters do not correspond to the vital bus.

D. Incorrect. PT-1MS-447, HP Turbine First Stage pressure instrument is powered from Vital Bus IV. Plausible because it is powered from one of the four vital busses.

Sys #	System	Category	KA Statement
010	Pressurizer Pressure Control	K2 Knowledge of bus power supplies to the following:	Indicator for code safety position

K/A# K2.04 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.38.1C Rev. 5 pg. 18
1SQS-6.4, Rev. 14 pg. 28 - 29
1OM-6.3.C Rev 11 pg 5

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7)

Objective: 3SQS-38.1, Rev. 8 Obj. 2 - From memory, describe the Normal System Arrangement for the Emergency 120 VAC Distribution Systems, including distribution paths, status of feeder breakers, loads, bus transfer switches, power train, and bus designation.

1SQS-6.4 Rev 14 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

36. The plant is operating at 100% power, with Reactor Trip Breaker testing in progress, and initial conditions are as follows:

- Reactor Trip Breakers (RTB) "A" and "B" are closed.
- Operators have just closed Reactor Trip Bypass Breaker (BYB) "A".

The following sequence of events occurs:

1. The "C" Main Feed Reg Valve fails closed, resulting in a valid automatic reactor trip signal.
2. The "B" Train of RPS fails to actuate.

Assuming the "A" Train of RPS operates as designed, and no operator action is taken, how do RTB "B" and BYB "A" initially respond?

	Reactor Trip Breaker (RTB) "B"	Bypass Breaker (BYB) "A"
A.	Trips	Does NOT trip
B.	Does NOT trip	Trips
C.	Does NOT trip	Does NOT trip
D.	Trips	Trips

Answer: C

Explanation/Justification: K/A is met by evaluating the candidate's knowledge of the RTBs and BYBs and where the trip signals for each is generated. Specifically, with the BYB 'A' closed for testing, and only the 'A' train of RPS actuates, the CRDS will remain powered.

A. Incorrect. Neither the Reactor Trip Breaker 'B' or the Bypass Breaker 'A' receive an auto trip signal from the 'A' RPS train.
B. Incorrect. It is correct that the Reactor Trip Breaker does not trip. The Bypass Breaker "A" does not receive an auto trip signal from the 'A' train
C. Correct. Since there was an AUTO trip signal due to low-low SG level, the 'A' Train RPS sends trip signals to the 'A' Trip Breaker and the 'B' Bypass Breaker, but RTB 'B' and BYB 'A' do not receive a trip signal.
D. Incorrect. Reactor Trip Breaker "B" does not receive an auto trip signal from the "A" train.

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

K/A# K3.01 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.1.B Rev 10 pg. 3; 1OM-1.5.A.1 rev 8

Question Source: Bank – 2013 Millstone 3 NRC Exam Q37

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 45.6)

Objective: 3SQS-1.2, Rev. 8 Obj. 4 - Given a Reactor Protection System Hardware configuration and without referenced material, describe the Reactor Protection System Hardware field response to the following actuation signals, including automatic functions and changes in equipment status: Reactor Trip

37. Given the following plant conditions:

- The plant is stable in Mode 3 following a reactor trip.
- Containment Pressure Transmitter PT-1LM-100A has failed high.
- All required actions directed by the Instrument Failure Procedure were completed.
- Subsequently, PT-1LM-100B fails high.

What will be the CIB and Safety Injection response, if any?

A. Both CIB and Safety Injection actuate.

B. CIB actuates but Safety Injection doesn't.

C. Neither CIB or Safety Injection actuate.

D. Safety Injection actuates but CIB doesn't.

Answer: C

Explanation/Justification: K/A is met by testing the candidates' knowledge of the effect of two failed cnmt pressure detector will have on Engineered Safety Features Actuation systems.

A. Incorrect. Refer to correct answer explanation.
B. Incorrect. Refer to correct answer explanation.
C. Correct. The candidate must recognize that the initial failure was on CH-I. The stem states that all required actions of the IF procedure have been completed, which means that the CH-I input to the CIB actuation circuitry has been bypassed which changes the actuation logic from 2/4 to 2/3. Upon a subsequent failure of a 2nd channel, (CH-II), NO CIB actuations will occur because only 1 of 3 Channels have seen the failure. CH I does not provide input to Safety Injection actuation circuitry, so therefore when CH II fails it does not satisfy the 2/3 logic required for Safety Injection to actuate, therefore no SI actuation occurs. All distractors are plausible if the candidate does not know SSPS logics or impacts of these failures upon the system.
D. Incorrect. Refer to correct answer explanation.

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:** U1 UFSAR Figure 7.2-1 Sheet 8 Rev 22
1OM-1.4.IF Rev 8 pgs. 5-7

Question Source: Bank – 2LOT8 NRC Exam Q38

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 to 45.8)

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.

38. If a loss of Chilled Water occurs, ______(1)______ may be aligned to the Cnmt Air Recirculation Fans (CARF) in order to prevent containment temperature from exceeding Tech Spec 3.6.5 limit of ______(2)______.

A. 1) Primary Component Cooling Water (CCR)
2) 105°F

B. 1) Primary Component Cooling Water (CCR)
2) 108°F

C. 1) Reactor Plant River Water (RPRW)
2) 105°F

D. 1) Reactor Plant River Water (RPRW)
2) 108°F

Answer: D

Explanation/Justification: K/A is met by demonstrating the ability to provide cooling to the Cnmt Air Recirculation Fans in the event that chilled water is lost to prevent the Tech Spec Cnmt temperature from being exceeded due to no cnmt cooling flow.

A. Incorrect. Plausible since use of the alternate source (service water) is rarely performed, thus the candidate who does not have detailed knowledge of the system may conclude that since CCR supplies other loads in CNTMT such as CRDM shroud coolers that it would be the natural source to backup chilled water. 105F is plausible because it is the cnmt temperature mentioned in the purpose of the RPRW, but it is not the TS limit.

B. Incorrect. Plausible since use of the alternate source (service water) is rarely performed, thus the candidate who does not have detailed knowledge of the system may conclude that since CCR supplies other loads in CNTMT such as CRDM shroud coolers that it would be the natural source to backup chilled water. Max temperature for containment avg air temp is 108F based on TS 3.6.5.

C. Incorrect. River Water is the backup cooling source for the CARFs. 105F is plausible because it is the cnmt temperature mentioned in the purpose of the RPRW, but it is not the TS limit.

D. Correct. River Water is the backup cooling source for the CARFs. Max temperature for containment avg air temp is 108F based on TS 3.6.5.

Sys #	**System**	**Category**	**KA Statement**
022	Containment Cooling	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including:	Containment temperature

K/A# A1.01 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-29.4.H rev 4 pg. 2
TS 3.6.5

Question Source: Bank – North Anna NRC Exam 2010 Q25

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

Objective: 1SQS-29.1, Rev. 12 — Obj. 5. Given a Chilled Water System configuration and without referenced material, describe the Chilled Water System field response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable. b. Loss of electrical power

3SQS-CONT ITS, Rev. 1 — Obj. 4. Given plant conditions that constitute non-compliance with any Containment Systems LCO, or Licensing Requirement determine the applicable Condition(s), Required Action(s), and associated Completion Time(s).

39. Given the following plant conditions:

- The Unit is operating at Full Power with all systems in NSA.
- 1VS-F-1A, "CNMT Air 1A Recirc Fan" is running.
- 1VS-F-1B, "CNMT Air 1B Recirc Fan" is secured for maintenance.
- 1VS-F-1C, "CNMT Air 1C Recirc Fan" is running.
- 1VS-F-2C, "CRDM Shroud Fan" is aligned to "B" Train.
- A Loss of Bus 1P1 occurs.
- No operator action has occurred and all systems function as designed.

Which of the following describes the **CURRENT** status of 1VS-F-1A & C, Containment Air Recirculation Fans?

	1VS-F-1A	**1VS-F-1C**
A.	RUNNING	RUNNING
B.	RUNNING	**NOT** RUNNING
C.	**NOT** RUNNING	**NOT** RUNNING
D.	**NOT** RUNNING	RUNNING

Answer: B

Explanation/Justification: K/A is met by the candidate demonstrating knowledge of the power supplies to the containment cooling fans when in a non-NSA alignment because the 'C' Cnmt cooling fan can be aligned to either 1N1 or 1P1 480 bus.

A. Incorrect. Correct that 1VS-F-1A is running, however, 1VS-F-1C is tripped. Plausible because 1VS-F-1C can be selected to either power supply.

B. Correct. 1VS-F-1A is powered from Bus 1N1, 1VS-F-1B is powered from Bus 1P1 and 1VS-F-1C can be powered from either 1N1 or 1P1. In the stated plant conditions 1VS-F-1C is running. Since 1VS-F-1A is being supplied from 1N1, NSA would dictate that 1VS-F-1C would be aligned to the 1P1 bus to allow train separation. If 1P1 is lost, then 1VS-F-1A will be the only running containment air recirc fan. To further clarify NSA, the question stem states that 1VS-F-2C is aligned to the B Train. Procedurally, 1VS-F-1C is aligned to the same Bus as 1VS-F-2C to ensure train separation.

C. Incorrect. Correct that 1VS-F-1C is not running. Incorrect that 1VS-F-1A is not running.

D. Incorrect. Opposite of the correct fan status.

Sys #	**System**	**Category**	**KA Statement**
022	Containment Cooling	K2 Knowledge of power supplies to the following:	Containment cooling fans

K/A# K2.01 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-44C.3.C, Rev. 6, pg. 4 - 6
1SQS-44C.1 PPNT, Rev. 10, Iss 1 Slides 8
1SQS-44C.1, Rev. 10 Iss 1, Pg. 17 -18

Question Source: Bank – 1LOT8 NRC Exam Q39

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7)

Objective: 1SQS-44C.1 Obj. 3 - Identify the power supplies for the components identified on the Normal System Arrangement System flowpath drawing which are powered from the class 1E electrical distribution system.

40. The plant was operating at 100% power when SG 'A' faulted. The following conditions exist:

- The Reactor tripped
- Annunciator A5-13, Low Stm Line Press Reactor Trip SI & Stm Line Isol is LIT
- Containment pressure is 15 psig and RISING
- RWST level is 31' 6" and LOWERING
- All equipment operated as designed

1) Based on the above conditions, which containment spray systems will be in operation?

2) The Containment Spray systems are designed to maintain the peak containment liner temperature below what temperature?

A. 1) Quench Spray pumps **only**
   2) 280°F

B. 1) Quench Spray pumps **only**
   2) 350°F

C. 1) Quench Spray pumps **and** Recirc Spray pumps
   2) 280°F

D. 1) Quench Spray pumps **and** Recirc Spray pumps
   2) 350°F

Answer: A

Explanation/Justification: K/A is met by determining which Cnmt spray pumps will be operating based on the given conditions, and identifying that the Quench spray pumps are designed to limit cnmt liner temperature to ,280F during a DBA SLB.

A. Correct. Quench Spray pumps will be the only cnmt spray pumps running. They start on a CIB actuation at 11.1 psig in cnmt. Maintaining the containment liner < 280F is correct.

B. Incorrect. Quench Spray pumps will be the only cnmt spray pumps running. They start on a CIB actuation at 11.1 psig in cnmt. 350F is plausible because it is a higher value than the actual liner temperature, and it is a temperature that is familiar to modes of operation.

C. Incorrect. Plausible because both need a CIB to start, but Recirc Spray pumps will not start until 2 of 3 RWST low level signals are in at <27' 7.5". Maintaining the containment liner < 280F is correct.

D. Incorrect. Plausible because both need a CIB to start, but Recirc Spray pumps will not start until 2 of 3 RWST low level signals are in at <27' 7.5". 350F is plausible because it is a higher value than the actual liner temperature, and it is a temperature that is familiar to modes of operation.

Sys #	System	Category	KA Statement
026	Containment Spray	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including:	Containment temperature

K/A# A1.02 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-13.1.D Rev 5 pgs. 3, 5, 6
1OM-13.2.B Rev 12 pg 2
TS 3.6.6 bases rev 16 B 3.6.6 – 2

Question Source: New

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

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

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

41. Given the following conditions:

    - The plant was at 100% power when all 3 Steam Generators Faulted in Containment
    - Containment Pressure is 31 psig and RISING
    - 1QS-P-1A and 1B Quench Spray Pumps failed to start

    What are the **MINIMUM** required Engineered Safety Features Actuation System (ESFAS) switch manipulations required to start **BOTH** Quench Spray Pumps per the electrical logic diagram?

A. Two pushbuttons consisting of one pushbutton per SI Train.

B. Two pushbuttons consisting of two Train 'A' CIB pushbuttons **OR** two Train 'B' CIB pushbuttons.

C. Four pushbuttons consisting of two pushbuttons each from both Train 'A' **AND** Train 'B'.

D. Six pushbuttons consisting of one pushbutton per SI Train **AND** two pushbuttons each from both Train 'A' **AND** Train 'B'.

\nswer: C

Explanation/Justification: K/A is met by the candidates recognizing that CIB did not actuate and the knowledge of how many ESF actuation switches must be operated to ensure both trains of Quench Spray Pumps start and prevent CNMT pressure from exceeding design pressure.

A. Incorrect. Plausible if the candidate does not know that the QS pumps start off a CIB signal at 11.1psig instead of an SI signal.

B. Incorrect. Plausible if the candidate thinks actuating one train of CIB will actuate both trains of CIB. This is not the case as unit 1 requires 2 pushbuttons to actuate one train, and they are train specific.

C. Correct. Each train consists of 2 pushbuttons being required to actuate the train, and the pushbuttons are train specific for the equipment associated with that train.

D. Incorrect. Plausible if the candidate thinks both trains of SI and CIB must be actuated to start the QS pumps.

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:** USFAR Fig. 7.2-1 sheet 8 Rev 22

Question Source: Modified – 2LOT15 NRC Exam Q41

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 to 45.8)

Objective: 1SQS-13.1 Rev 15 19. 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. a. Containment Isolation Phase B

42. 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 a 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 500psig. 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 – South Texas 2014 NRC Exam Q26 Modified

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

Objective: 3SQS-1.1, Rev. 8 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.

43. The plant is at 20% power with the following conditions:

- A7-45, Steam Generator 1A Level Deviation From Setpoint, is in alarm.
- An Operator reports that the air supply line to FCV-1FW-479, 1A SG FW BYPASS FCV, has blown off.
- AOP-1.4.1, Process Control failure has been entered.

Which of the following completes the statements below?

1) FCV-1FW-479 will fail _______________.

2) In accordance with the Transient Response Guidelines the reactor will be tripped if SGWL reaches ________.

A. 1) open
 2) > 85%

B. 1) open
 2) > 90%

C. 1) closed
 2) < 5%

D. 1) closed
 2) < 25%

Answer: D

Explanation/Justification: K/A is met by testing the feedwater bypass flow control valve failure on a loss of air to the valve, and testing the knowledge of the plant response due to the failure and when a reactor trip is directed by the Transient Response Guideline procedure which give the crew setpoints for manual reactor trips.

A. Incorrect. Plausible if the candidate thinks the FCV fails open. 85% is plausible because if the FCV did fail open, this is the TRG Manual Reactor Trip setpoint for SGWL level high.
B. Incorrect. Plausible if the candidate thinks the FCV fails open. 90% is plausible because the FCV failing open would lead to a high SGWL, but the 90% is the TRG Manual Reactor Trip setpoint for PRZR level high.
C. Incorrect. The FCV does fail closed. 5% is plausible because the FCV failing closed will lead to a low SGWL, but the 5% is the TRG Manual Reactor Trip setpoint for PRZR level low.
D. Correct. The FCV will fail closed on a loss of air. <25% is the TRG Manual Reactor Trip setpoint for SGWL level low.

Sys #	System	Category	KA Statement
059	Main Feedwater	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Failure of feedwater regulating valves

K/A# A2.12 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** U1 RM-0424-001 Rev 20
BVBP-OPS-0024 Rev. 11 pg. 16

Question Source: New

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

Objective: 1SQS-24.1, Rev. 20 16. Given a Main Feedwater, Dedicated Auxiliary Feedwater, Auxiliary Feedwater System or Steam Generator Water Level Control System configuration and without referenced material, describe the associated system's control room response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable. a. Loss of instrument air

44. The crew is preparing to start the 'B' Main Feed Pump (MFP) in accordance with 1OM-24.4.A, Steam Generator Feedwater System Startup.

The procedure directs placing the auxiliary oil pump, 1LO-P-3B to MANUAL.

Complete the following statements.

1) The control switch for the MFP auxiliary oil pump is located ________________.
2) During the MFP startup, the purpose of the MFP auxiliary oil pump is to ________________ to allow the MFP breakers to close.

A. 1) on BB-C near the MFP control switches
2) pressurize the MFP lube oil system

B. 1) on BB-C near the MFP control switches
2) warm up the MFP lube oil to normal operating temperature

C. 1) locally near the MFPs
2) pressurize the MFP lube oil system

D. 1) locally near the MFPs
2) warm up the MFP lube oil to normal operating temperature

Answer: C

Explanation/Justification: K/A is met by demonstrating the ability to locate the MFP auxiliary oil pump control switch locally at the MFP, and understand the purpose the purpose of the auxiliary oil pump during MFP startup.

A. Incorrect. First part is plausible because the indication for the AOPs are located above the MFP control switches. Second part is correct.
B. Incorrect. First part is plausible because the indication for the AOPs are located above the MFP control switches. The second part is plausible since the MFP is being started lube oil temperatures may be low, the aux oil system will eventually raise the temperature of the lube oil, but this is not why the AOP is started in this case.
C. Correct. The AOP switch is located in the TB 693' near the MFPs. The AOP is started to raise lube oil pressure in order to meet the MFP starting interlock of Lube oil pressure >10 PSIG to allow the MFP breakers to close.
D. Incorrect. First part is correct. Second part is plausible since the MFP is being started lube oil temperatures may be low, the aux oil system will eventually raise the temperature of the lube oil, but this is not why the AOP is started in this case.

Sys #	**System**	**Category**	**KA Statement**
059	Main Feedwater	Generic	Ability to locate and operate components, including local controls.

K/A# 2.1.30 **K/A Importance** 4.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-24.4.A Rev 14 pg. 9
1OM-24.1.C Rev 5 pg. 2

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7 / 45.7)

Objective: 1SQS-24.1, Rev. 20 4. Describe the control, protection and interlock functions for the field components associated with the Main Feedwater, Dedicated Auxiliary Feedwater, Auxiliary Feedwater System and Steam Generator Water Level Control Systems, including automatic functions, setpoints and changes in equipment status as applicable.

45. Given the following plant conditions:
- The plant is in Mode 3 following a reactor trip from 100%
- The crew has completed ES-0.1, Reactor Trip Response, and transitioned to 1OM-52.4.R.1.F, Station Shutdown from 100% Power to Mode 5
- A plant cooldown of 50°F/hr is commenced using Condenser Steam Dumps
- Steam Generator (SG) Narrow Range levels are maintained stable at 65% during the cooldown, using the Auxiliary Feedwater pumps

Which one of the following describes the trend of Auxiliary Feedwater flow requirements as the plant cools down to Mode 5?

A. AFW flow requirements are constant as long as SG level remains constant.

B. AFW flow requirements are constant as long as the cooldown rate remains constant.

C. More AFW flow is required to maintain SG level due to a rise in SG water density as it cools.

D. Less AFW flow is required to maintain SG level because heat input to the SGs lowers as the cooldown continues.

\nswer: D

Explanation/Justification: K/A is met by determining that the required AFW flow will be less as the cooldown continues due to the RSC decay heat dropping over time from the reactor trip.

A. Incorrect. Plausible if heat input to the SG did not change. Heat input lowers due to less decay heat as the cooldown progresses.
B. Incorrect. Plausible if the effects of less decay heat are not considered.
C. Incorrect. Water density does not have any impact at this temperature in the SG.
D. Correct. This is an operational fundamental question that requires the candidate to simply understand that decay heat rate drops over time. Therefore less AFW flow is required as the amount of heat transfer from the RCS to S/Gs drops.

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:** GOGPF.R8 Appendix A Rev 1 pg. 81-82

Question Source: Bank – 2LOT15 NRC Exam Q47

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.5 / 45.7)

Objective:
1SQS-24.1, Rev. 20 Obj. 17. Given a specific plant condition, predict the response of the Main Feedwater, Dedicated Auxiliary 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.

GOGPF.R8 Rev 1 Obj. 25. Explain the relationship between decay heat generation and: a. Power level history/b. Power production/c. Time since reactor shutdown

46. Given the following conditions:

- The plant is at 100% power
- 'B' MDAFW pump is on clearance for corrective maintenance
- All applicable Tech Spec actions have been completed

An inadvertent Rx trip occurs, concurrent with a complete loss of AC power.

What is the current flowpath of the AFW system with NO operator action?

A. Only the TDAFW running but unable to feed the SGs through either train of AFW throttle valves.

B. Only the TDAFW feeding the SGs through both trains of AFW throttle valves.

C. Only the TDAFW feeding the SGs through the 'A' Train of AFW throttle valves.

D. Only the TDAFW feeding the SGs through the 'B' Train of AFW throttle valves.

Answer: D

Explanation/Justification: K/A is met with the knowledge that when the 'B' MDAFW pump is on clearance, the turbine driven AFP will be aligned to the 'B' header, therefore when the loss of all AC occurs, and no motor driven AFW pumps are available, the TDAFW pump will inject through the 'B' train AFW throttle valves.

A. Incorrect. Plausible if the candidate thinks that the AFW throttle valves are NSA closed, and there is no power to operate the valves on the AFW initiation signal. This is not correct because even though power is lost and the indicating lights will lose power, the NSA position of the MOV AFW throttle valves at 100% power is Open.

B. Incorrect. Plausible if the candidate thinks that the TDAFW pump is aligned to both AFW headers, but this is not correct.

C. Incorrect. Plausible if the candidate does not know about the TS requirements to realign the TDAFW pump which is NSA to 'A' header.

D. Correct. With the B MDAFW on clearance, TSs require that with one inoperable AFW pump (B MDAFW), the remaining two AFW pumps will be aligned to separate redundant headers capable of supplying flow to each steam generator. This means that the TDAFW pump would be aligned to the 'B' AFW header.

Sys #	System	Category	KA Statement
061	Auxiliary/Emergency Feedwater	K6 Knowledge of the effect of a loss or malfunction of the following will have on the AFW components:	Pumps

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

References provided to Candidate None **Technical References:** Tech Specs 3.7.5 Cond B
1OM-24.3.B.1 rev 23 pg. 8
U1 RM-0424-002 Rev. 20

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 45.7)

Objective: 1SQS-24.1, Rev. 20 17. Given a specific plant condition, predict the response of the Main Feedwater, Dedicated Auxiliary 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. The following conditions exist:

- The plant has tripped due to a loss of all 4Kv power
- The crew is performing ECA-0.0, "Loss of All Emergency 4KV AC Power"
- The BOP Operator is placing equipment in PTL in accordance with step 15 of ECA-0.0

Which of the following components will **NOT** be placed in PTL during this step, and what is the basis for not removing this component from service?

A. Auxiliary Feedwater pump; provide sufficient water to maintain an effective secondary heat sink.

B. Component Cooling Pump; provide cooling to the Reactor Coolant Pump thermal barrier.

C. Charging Pump; provide cooling to the Reactor Coolant Pump seals.

D. River Water Pump; provide cooling to the Emergency Diesel Generator.

Answer: D

Explanation/Justification: K/A is met by the knowledge that ECA-0.0 places equipment to PTL to defeat automatic loading of large loads on the AC emergency bus with the exception of River Water Pumps which is stated as a caution prior to step 15. The knowledge of the bases of the RW pump remaining available to load on a diesel start to provide diesel cooling is expected RO knowledge.

A. Incorrect. MDAFW pumps are not required immediately after power restoration and are considered a large load. The goal of this step is to avoid potential overload of the energized emergency bus. During ECA-0.0, sufficient AFW flow is provided by the turbine driven AFW pump, so heat sink is not a concern.

B. Incorrect. CCP pumps are not required immediately after power restoration and are considered a large load. The goal of this step is to avoid potential overload of the energized emergency bus. Providing cooling flow to the hot thermal barrier is not required at this time.

C. Incorrect. Charging pumps are not required immediately after power restoration and are considered a large load. The goal of this step is to avoid potential overload of the energized emergency bus. Providing cooling flow to the hot RCP seals could cause thermal shock to the seals & shaft.

D. Correct. River water pump auto start is desired to provide cooling to the EDG in the event it is locally restored. This is stated as a caution prior to step 15 and switches are placed to Auto in step 1 of Att. 2-E for starting the diesel locally.

Sys #	System	Category	KA Statement
062	AC Electrical Distribution	Generic	Knowledge of the operational implications of EOP warnings, cautions, and notes.

K/A# 2.4.20 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-0.0 Iss 3 Rev 2 step 15
1OM-53B.4.ECA-0.0 Iss 3 Rev 2 pg. 124

Question Source: Bank – 2LOT15 NRC Exam Q11

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.13)

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

48. The plant is at 100% power.

Based on the indications provided, and assuming all bulbs are functioning as expected:

1) A loss of which 480V bus would cause Battery charger 2 & 4 indications?
2) What is the cause of Battery Breaker #3 indication?

(Reference attached drawing)

A. 1) 480V Bus 1C is de-energized.
2) DC Battery Switchgear No. 3 control power is de-energized.

B. 1) 480V Bus 1C is de-energized.
2) Vital Bus 3 is de-energized.

C. 1) 480V Bus 1P is de-energized.
2) DC Battery Switchgear No. 3 control power is de-energized.

D. 1) 480V Bus 1P is de-energized.
2) Vital Bus 3 is de-energized.

Answer: C

Explanation/Justification: K/A is met by evaluating the ability of the candidate to determine that the DC control power has been lost for Battery breaker #3, and the power supply to battery chargers 2 & 4 have been lost causing the given control room indications.

A. Incorrect. Plausible because bus 1C is a normal 480V bus powered from 4160V bus B, this could lead the candidate to think Train B for battery chargers 2 & 4. DC Battery Switchgear No. 3 control power is correct.

B. Incorrect. Plausible because bus 1C is a normal 480V bus powered from 4160V bus B, this could lead the candidate to think Train B for battery chargers 2 & 4. Vital bus 3 is plausible because it provides power to multiple indications on the control room panels, and powers the dc bus volt meters in the CR.

C. Correct. 480V Bus 1P is correct because it supplies emergency MCC-1 -E10, which supplies both battery chargers 2 & 4. DC Battery Switchgear No. 3 control power is correct.

D. Incorrect. 480V Bus 1P is correct because it supplies emergency MCC-1 -E10, which supplies both battery chargers 2 & 4. Vital bus 3 is plausible because it provides power to multiple indications on the control room panels, and powers the dc bus volt meters in the CR..

Sys #	System	Category	KA Statement
063	DC Electrical Distribution	A4 Ability to manually operate and/or monitor in the control room:	Major breakers and control power fuses

K/A# A4.01 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** U1 RE-0001V Rev 33
3SQS39.1 PPNT (U-1) Rev 9 slide 32
3SQS37.1 U1 PPNT Rev 9 Iss 2 Slide 13

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 to 45.8)

Objective: 3SQS-39.1, Rev. 9

Obj. 15. Describe the instrumentation associated with the 125 VDC Distribution System located in the Main Control Room.

Obj. 21. Given a change in plant conditions due to system/component failure, analyze the 125 VDC Distribution System to determine what failure has occurred.

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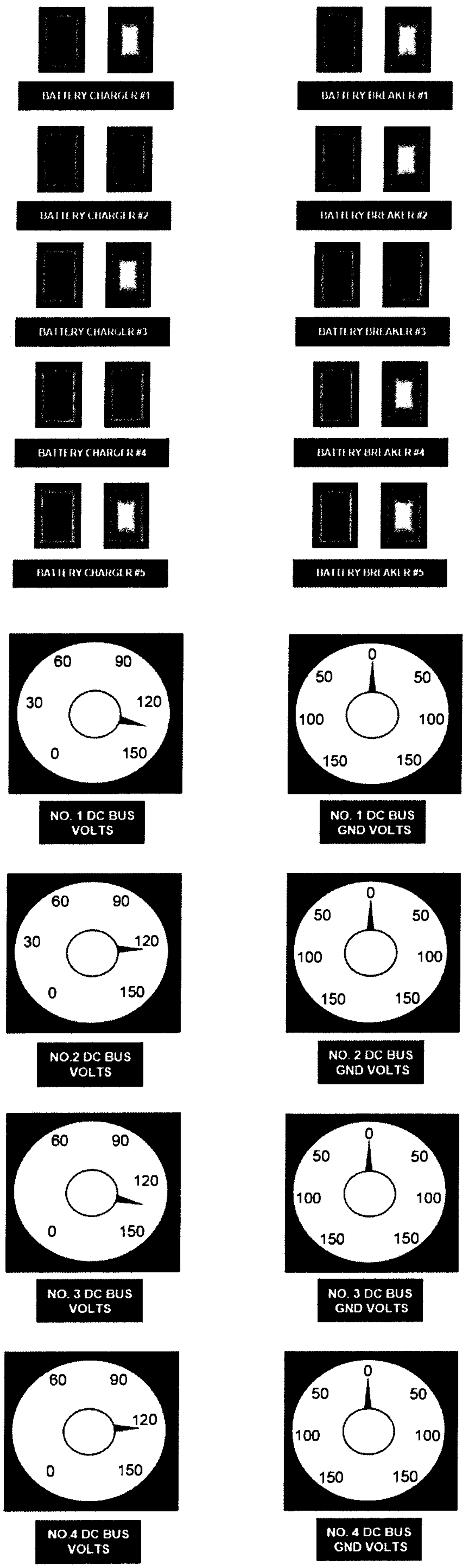
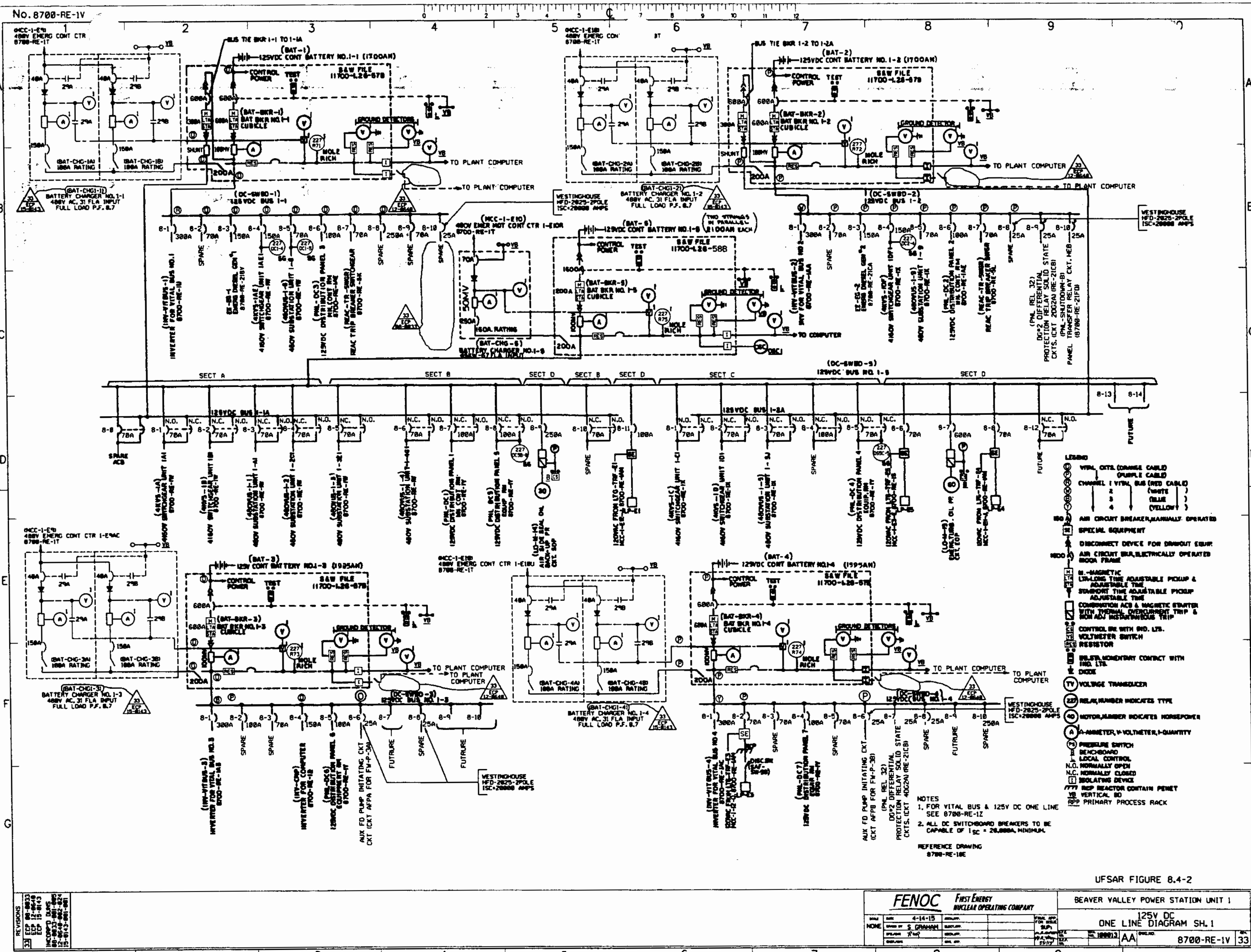

BATTERY CHARGER #1
BATTERY BREAKER #1
BATTERY CHARGER #2
BATTERY BREAKER #2
BATTERY CHARGER #3
BATTERY BREAKER #3
BATTERY CHARGER #4
BATTERY BREAKER #4
BATTERY CHARGER #5
BATTERY BREAKER #5
60 90 30 120 0 150
NO. 1 DC BUS VOLTS
0 50 50 100 100 150 150
NO. 1 DC BUS GND VOLTS
60 90 30 120 0 150
NO.2 DC BUS VOLTS
0 50 50 100 100 150 150
NO. 2 DC BUS GND VOLTS
60 90 120 0 150
NO. 3 DC BUS VOLTS
0 50 50 100 100 150 150
NO. 3 DC BUS GND VOLTS
60 90 120 0 150
NO.4 DC BUS VOLTS
0 50 50 100 100 150 150
NO. 4 DC BUS GND VOLTS


No. 8700-RE-1V
TO PLANT COMPUTER
125VDC CONT BATTERY NO.1-1 (1700AH)
125VDC CONT BATTERY NO.1-2 (1700AH)
125VDC CONT BATTERY NO.1-5
125V CONT BATTERY NO.1-3 (1995AH)
125VDC CONT BATTERY NO.1-4 (1995AH)
GROUND DETECTORS
WESTINGHOUSE HFD-2025-2POLE ISC=20000 AMPS
SECT A
SECT B
SECT C
SECT D
125VDC BUS 1-1A
125VDC BUS 1-2A
125VDC BUS NO.1-5
FUTURE
SPARE
NOTES
1. FOR VITAL BUS & 125V DC ONE LINE SEE 8700-RE-1Z
2. ALL DC SWITCHBOARD BREAKERS TO BE CAPABLE OF I_{SC} = 20,000A, MINIMUM.
REFERENCE DRAWING 8700-RE-10E
UFSAR FIGURE 8.4-2
FENOC
FirstEnergy NUCLEAR OPERATING COMPANY
BEAVER VALLEY POWER STATION UNIT 1
125V DC ONE LINE DIAGRAM SH. 1
4-14-15
S GRAHAM
NONE
AA
8700-RE-1V
33

49. When transferring 125VDC battery chargers for Battery #1 in accordance with 1OM-39.4.U, Swapping 125V DC Battery Chargers, which of the following identifies:

1) The correct order of breaker operation when ENERGIZING a charger?
2) The required sequence for transferring chargers?

A. 1) Close the DC output, then the AC input
2) Remove the in-service charger from service, then place the standby charger in service on the battery.

B. 1) Close the AC input, then the DC output
2) Place the standby charger in service on the battery, then remove the in-service charger from service.

C. 1) Close the DC output, then the AC input
2) Place the standby charger in service on the battery, then remove the in-service charger from service.

D. 1) Close the AC input, then the DC output
2) Remove the in-service charger from service, then place the standby charger in service on the battery.

Answer: D

Explanation/Justification: K/A is met with the knowledge that when swapping battery chargers for the battery, that the AC input must be closed first to verify the proper charger DC voltage exists, and that there is a kirk key interlock on the charger DC disconnect to prevent having both switches closed at the same time. This will ensure the A and B chargers in the dual unit are not operated in parallel.

A. Incorrect. Plausible distractor since there are only AC and DC breakers on the charger. Second part is correct.

B. Incorrect. Correct breaker order. Second part is plausible if the candidate does not remember about the kirk key interlock.

C. Incorrect. Plausible distractor since there are only AC and DC breakers ion the charger. Second part is plausible if the candidate does not remember about the kirk key interlock.

D. Correct. AC input is closed first so that DC voltage can be verified between 129-138 vdc (TS minimum is 127,8V), then the DC output is closed. This readies the charger to be placed in service so that when the in-service charger is removed from service, and the battery is supplying the DC loads, the standby charger can be placed in service with minimal drain on the battery. There is a placard on the front of the battery chargers to remind the operators. When swapping chargers, the DC Disconnect Switches are provided with a "Kirk Key" interlock to prevent having both switches closed at the same time, therefore, one is removed before the other one is placed in service.

Sys #	System	Category	KA Statement
063	DC Electrical Distribution	K1 Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems:	Battery charger and battery

K/A# K1.03 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-39.4.U rev 0 pg. 3-5
3SQS39.1 PPNT (U-1) Rev 9 slide 41

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Objective: 3SQS-39.1, Rev. 9 Obj. 6. Describe the controls and instrumentation associated with the 125 VDC Distribution System located in the field.

50. The Emergency Diesel Generator is maintained warm when it is not running by which of the following?

A. An immersion heater mounted in the lube oil sump heats the lube oil for the Auxiliary Lube Oil Circulating Pump to circulate through the engine.

B. An immersion heater mounted in the cooling water system heats the lube oil in the lube oil cooler for the Auxiliary Lube Oil Circulating Pump to circulate through the engine.

C. Cylinder jacket strip heaters mounted outside of the engine cylinders maintains the engine block warm.

D. Electric wall-mounted heaters maintain the Diesel Generator Room ambient air temperature above 90°F.

Answer: B

Explanation/Justification: K/A is met with the knowledge that the DG cooling water immersion heater will heat the lube oil in the lube oil cooler to maintain the lube oil temperature between 125-155F in a standby diesel.

A. Incorrect. Plausible because it is similar to the actual design, but immersion heaters are not used in the DG oil system.
B. Correct. Water, heated by the cooling water immersion heaters, circulates through the lube oil cooler by convection flow to maintain the lube oil being circulated at 125F to 155F when the EDG is shutdown.
C. Incorrect. Plausible because there are strip heaters located directly underneath each diesel generator winding. These strip heaters are designed to keep the windings dry while the generators are not operating, but they do not maintain the EDG warm.
D. Incorrect. Plausible because the DG monthly surveillance has the heaters set 65F (iaw minimum USFAR room temp), and the room fans set at 90F. 90F was chosen because it is the minimum oil temperature.

Sys #	System	Category	KA Statement
064	Emergency Diesel Generator	K1 Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems:	D/G cooling water system

K/A# K1.02 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-36.1.C Rev 7 pg. 10

Question Source: Bank – Vision #123487

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Objective: 1SQS-36.2, Rev. 19 1. Describe the function of the EDG and the associated major components and subsystems as documented in chapter 36 of the Unit 1 Operating Manual.

51. A liquid waste discharge is in progress from 1BR-TK-4A, Coolant Recovery Tank, to the cooling tower blowdown IAW 1OM-17.4.AN, Discharging a Coolant Recovery Tank to Cooling Tower Blowdown.

An hour after the discharge begins, RM-1LW-104, Liquid Waste Effluent Monitor power supply fails.

What must be done with the current discharge in progress, and what actions are required for continuing the discharge?

A. Verify automatic isolation of the discharge path has occurred. Re-establish flow only when 1/2-ODC-3.03, ODCM requirements have been met.

B. Manually stop the discharge. Re-establish flow only when RM-1LW-104, Liquid Waste Effluent Monitor is restored to service.

C. Manually stop the discharge. Re-establish flow when the system is aligned to process through RM-1LW-116, Liquid Waste Contaminated Drain Monitor.

D. Allow the discharge to continue with increased monitoring of RM-1LW-116, Liquid Waste Contaminated Drain Monitor. Verify 1/2-ODC-3.03, ODCM requirements have been met.

Answer: A

Explanation/Justification: K/A is met by evaluating the knowledge of the automatic actions which occur due to a failed power supply for RM-1LW-104 during a liquid waste discharge, and the procedural requirements to re-establish the discharge.

A. Correct. The candidate must know that RM-1LW-104 is one of three rad monitors that will cause automatic actions to occur when the power supply fails to the rad monitor. All three rad monitors detect activity in the discharge flowpath of liquid or gaseous waste. The discharge may continue after ODCM requirements of RM-1LW-104 restored, or 2 independent samples, 2 independent verifications of the release rate calculations, Independent verification of discharge valving. The ODCM specifics are SRO knowledge which is why it is not part of the answers.

B. Incorrect. The discharge path is automatically isolated on RM power failure. Plausible distractor that flow could be re-established when RM-1LW-104 is restored, but it is not the only way to restore the discharge.

C. Incorrect. The discharge path is automatically isolated on RM power failure. Plausible distractor because RM-1LW-116 is the other LW RM which has auto actions, but the coolant recovery tanks cannot be lined up through this rad monitor.

D. Incorrect. Plausible because the candidate may not know that the auto isolation features occur on a loss of power supply, and may think that RM-1LW-116 may be the ODCM alternate for the coolant recovery tank discharge.

Sys #	System	Category	KA Statement
073	Process Radiation Monitoring	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Erratic or failed power supply

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

References provided to Candidate None **Technical References:** 1SQS-43.1 Rev 16 LP, pg. 15
1OM-43.4.ACQ Rev 5 pg 2
1OM-17.2.A Rev 5 pg. 2
1OM-17.4.AN Rev 9 pg. 5

Question Source: New

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

Objective: 1SQS-17.1, Rev. 15 Obj. 18. Given a Liquid Waste Disposal System configuration and without reference material, describe the Liquid Waste Disposal 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

21. 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.

52. System Engineering suspects tube fouling of the Primary Component Cooling Water (CCR) Heat Exchangers. Which of the following would provide corroborating evidence for this condition?

A. Rising River Water differential pressure across the CCR Heat Exchanger with CCR temperature control valve, TCV-1CC-100, closed further than normal.

B. Rising CCR differential pressure with CCR temperature control valve, TCV-1CC-100, open further than normal.

C. Lowering River Water differential pressure across the CCR Heat Exchanger with CCR temperature control valve, TCV-1CC-100, open further than normal.

D. Lowering CCR differential pressure with CCR temperature control valve, TCV-1CC-100, closed further than normal.

Answer: A

Explanation/Justification: K/A is met by the Primary Component Cooling water (CCR) heat exchangers River Water (RW) side becoming fouled causing CCR temperature to rise and determining how the CCR Temp Control Valve will respond.

A. Correct. River Water flows through the CCR Hx tubes. Fouling would be indicated by both rising RW ΔP and rising CCR Hx outlet temperatures to which TCV-100 would close forcing more CCR flow through the Hx.

B. Incorrect. Plausible if candidate does not remember which system flows through the tubes and through the shell side of the CCR Hxs, and further reasons that TCV-100 would close, not open, in response to rising CCR Hx outlet temperature.

C. Incorrect. Plausible if candidate does not remember that TCV-100 is a Hx bypass valve not a Hx outlet valve and that it therefore closes in response to rising CCR Hx outlet temperature; reasoning there is more CCR flow through the CCR side of the Hx the candidate further misinterprets that RW flow would lower, lowering the RW side ΔP.

D. Incorrect. Plausible if candidate reasons that tube fouling will cause TCV-100, as an outlet not a bypass valve, to close thereby lowering flow and ΔP across the CCR side of the Hx.

Sys #	System	Category	KA Statement
076	Service Water	K3 Knowledge of the effect that a loss or malfunction of the SWS will have on the following:	Reactor building closed cooling water

K/A# K3.03 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-15.2.A Rev 4 pg 2
U1 RM-0415-001 Rev. 23

Question Source: Bank – Vision 264461

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 45.6)

Objective: 1SQS-15.1, Rev. 14 Obj. 15. Summarize the Precaution and Limitations as outlined in the Operating Manual 1OM-15.1 procedures.

53. 1) Which of the ESF actuations will cause River Water (RW) to be isolated to the CCR Hxs?

2) If **only** the Train 'B' ESF actuation occurs, what is the status of the RW headers supplying the CCR Hxs?

A. 1) Containment Isolation Phase A (CIA)
2) Both RW headers will be isolated.

B. 1) Containment Isolation Phase A (CIA)
2) Only the 'B' RW header will be isolated.

C. 1) Containment Isolation Phase B (CIB)
2) Both RW headers will be isolated.

D. 1) Containment Isolation Phase B (CIB)
2) Only the 'B' RW header will be isolated.

Answer: C

Explanation/Justification: K/A is met with the knowledge that a 'B' Train CIB signal will automatically close the RW header isolation valves in both trains to isolate the CCR Hxs by closing MOV-1RW-106A and 106B.

A. Incorrect. Wrong ESF signal. Correct that MOV-1RW-106A and 106B will close to isolate both RW headers to CCR Hxs.
B. Incorrect. Wrong ESF signal. Second part is plausible because it would make sense that 'B' Train CIB would isolate the 'B' Train of RW.
C. Correct. CIB will isolate the CCR Hxs. A Train 'B' CIB will isolate both RW headers by closing MOV-1RW-106A and 106B which are isolation vales in both the 'A' & 'B' RW headers.
D. Incorrect. First part is correct. Second part is plausible because it would make sense that 'B' Train CIB would isolate the 'B' Train of RW.

Sys #	System	Category	KA Statement
076	Service Water	K4 Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following:	Conditions initiating automatic closure of closed cooling water auxiliary building header supply and return valves

K/A# K4.01 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.5.B.4 rev. 17 pgs. 12-14
1SQS-30.2 PPNT Rev 18 slide 17

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41/7)

Objective: 1SQS-30.2, Rev. 18 Obj. 15 - Given a Reactor Plant River Water System configuration and without referenced material, describe the Reactor Plant River Water System control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable. a. Safety Injection Signal (SIS), b. Containment Isolation Signal – Phase B (CIB).

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

- The plant is operating at 100% power.
- The following control room annunciators are received:
    - A6-100, STA INSTR AIR RCVR TANK PRESS LOW
    - A6-109, STA INSTR AIR RCVR TANK DISCH PRESS LOW
- PI-1IA-106, "INSTR AIR HEADER" Pressure Indicator is 92 psig and RISING.
- PI-1IA-106B, "INSTR AIR RCVR" Pressure Indicator is 90 psig and slowly LOWERING.
- All systems function as designed.
- Assume NO operator actions.

Based on **these** air pressure readings, what will be the status of TV-1SA-105, Station Air Header Trip Valve **AND** 1IA-C-4, Diesel Driven Air Compressor?

TV-1SA-105, Station Air Header Trip Valve will be ____(1)____.
1IA-C-4, Diesel Driven Air Compressor will ____(2)____.

A. 1) OPEN
2) be RUNNING

B. 1) CLOSED
2) be RUNNING

C. 1) OPEN
2) **NOT** be RUNNING

D. 1) CLOSED
2) **NOT** be RUNNING

Answer: B

Explanation/Justification: K/A is met by demonstrating the candidates' ability to monitor control room pressure gauges and determine if the automatic instrument air system lineups have occurred.

A. Incorrect. Plausible if the candidate does not know the pressure at which TV-1SA-105 closes (95 psig). Second part is correct.

B. Correct. TV-1SA-105 closes at 95 psig. Based on control board pressure gauge readings TV-1SA-105 will be closed. The Diesel Driven Air Compressor starts at 93 psig and running based on pressure stated plant conditions. The TV-1SA-105 and Diesel Air Comp knowledge tested in this question relates to the IA pressure at which they operate.

C. Incorrect. Plausible if the candidate does not know the pressure at which TV-1SA-105 closes (95 psig). Second part is plausible if the candidate does not know that the diesel air compressor starts at 93 psig, and thinks the check valve at the outlet of the air receiver has isolated the leak, and 105 supplies the instrument air header.

D. Incorrect. TV-1SA-105 closes at 95 psig. Second part is plausible if the candidate does not know that the diesel air compressor starts at 93 psig, and thinks the check valve at the outlet of the air receiver has isolated the leak, and the station air compressors are carrying the inst. air header.

Sys #	**System**	**Category**	**KA Statement**
078	Instrument Air	A4 Ability to manually operate and/or monitor in the control room:	Pressure gauges

K/A# A4.01 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-34.4.AAI, Rev. 7 pg. 5
1SQS-34.1, Rev. 15, PPNT Slide 9

Question Source: Modified – 1LOT8 NRC Exam Q53

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 to 45.8)

)bjective: 1SQS-34.1 Rev 15 Obj. 13. Given a specific plant condition, predict the response of the Compressed Air 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.

55. The plant is in Mode 3, and the crew has entered the containment IAW 1OM-47.4.G, Emergent Containment entry, in search of an RCS leak.

When opening the inner Containment door on the Personnel Access, the hinge broke, and the door could not be closed.

1) Which of the following procedures will be used to exit containment?

2) To comply with TS 3.6.2, Containment Air Locks, the **maximum** time to verify the operable airlock door closed is?

A. 1) 1OM-47.4.C, 18-Inch Escape Manway Operations.
2) 30 minutes

B. 1) 1OM-47.4.C, 18-Inch Escape Manway Operations.
2) 1 hour

C. 1) 1OM-47.4.D, Equipment Hatch Emergency Air Lock Operation.
2) 30 minutes

D. 1) 1OM-47.4.D, Equipment Hatch Emergency Air Lock Operation.
2) 1 hour

Answer: D

Explanation/Justification: K/A is met by performing an emergency containment entry with a failure of the inner door to close. The candidate has to determine which procedural exit is possible, and using Tech Spec knowledge determine that the operable air lock door must be closed within 1 hour to mitigate the consequence of the door failure.

A. Incorrect. 18-inch Emergency Manway is plausible if it is not known that there is only one hatch, and no way of equalizing cnmt to the Aux Building in a timely manner. 30 minutes is plausible because it is ≤1hour and 30 minutes are used to satisfy other LCOs.

B. Incorrect. 18-inch Emergency Manway is plausible if it is not known that there is only one hatch, and no way of equalizing cnmt to the Aux Building in a timely manner. 1 hour is correct iaw TS 3.6.2 Cond A.

C. Incorrect. First part is correct. 30 minutes is plausible because it is ≤1hour and 30 minutes are used to satisfy other LCOs.

D. Correct. Since pressure cannot be equalized on the personnel access hatch, the only way into and out of CNMT is through the Equipment Hatch Emergency Air Lock which has two doors and equalization capabilities to pass through the airlock. 1 hour is correct iaw TS 3.6.2 Cond A.

Sys #	System	Category	KA Statement
103	Containment	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations	Emergency containment entry

K/A# A2.05 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** TS 3.6.2 Cond A
1OM-47.4.D Rev 5 pg. 2

Question Source: New

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

Objective: 1SQS-47.1-01-14: Outline the components necessary to operate the Containment Airlocks.

1SQS-47.1-01-15: From memory and for a given set of plant conditions, determine if the given the condition meets the criteria for entry into a less than one hour action statement in accordance with the Technical Specifications.

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

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

A. 480 VAC Substation 8N & 9P, RDMG's, RTB's, Power Cabinets.

B. 480 VAC Substation 8N & 9P, Power Cabinets, RDMG's, RTB's.

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

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

Answer: C

Explanation/Justification: K/A is met by demonstrating knowledge of the power supply flowpath to the reactor trip breakers.

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

Sys #	System	Category	KA Statement
001	Control Rod Drive	K2 Knowledge of bus power supplies to the following:	One-line diagram of power supply to trip breakers

K/A# K2.02 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.3.C Rev 6 pg. 8; 3SQS-1.3 PPNT Rev 7 Iss 1 slide 46

Question Source: Bank - 2LOT8 NRC EXAM Q56

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7)

Objective: 3SQS-1.3 Rev. 7 3. Describe how power is supplied to the Rod Drive Motor Generator (RDMG) sets, Logic/Power Cabinets, DC Hold Cabinet, and the Control Rod Drive Mechanism (CRDM) coils.

57. A plant startup is in progress IAW 1OM-52.4.B, Load Following.
- Rx is at 6% power and RISING
- LT-1RC-461 fails LOW
- The crew has completed all required actions of the appropriate Instrument Failure procedure

30 minutes later the following conditions exist:
- Rx power is 8% and STABLE
- LT-1RC-460 fails off scale HIGH
- NO OPERATOR actions have been taken

Complete the following statements.

1) The Reactor ______________ remain critical.

2) The pressurizer level high reactor trip bistables are set at ______________.

A. 1) will
2) 90%

B. 1) will
2) 92%

C. 1) will NOT
2) 90%

D. 1) will NOT
2) 92%

Answer: B

Explanation/Justification: K/A is met by testing the knowledge of the RPS Rx trip setpoints associated with the PZR level inputs, specifically that these inputs to RPS will be blocked when Rx power is <10%.

A. Incorrect. The reactor will remain critical because the przr coincidence is 2/3 przr levels >92% when greater than P-7 (10% power). 90% is plausible because it is the 2/3 RCS Loop Low Flow Rx trip setpoint.

B. Correct. The reactor will remain critical because the przr coincidence is 2/3 przr levels >92% when greater than P-7 (10% power). In the stem Rx power is 8% when LT460 fails high, and the bistables for LT461 had already been placed in trip, therefore the plant is <P-7 and no trip occurs. 92% is the pressurizer level high reactor trip bistable setpoint.

C. Incorrect. The Rx will remain critical due to being <P-7 setpoint. 90% is plausible because it is the 2/3 RCS Loop Low Flow Rx trip setpoint.

D. Incorrect. The Rx will remain critical due to being <P-7 setpoint. 92% is the pressurizer level high reactor trip bistable setpoint.

Sys #	**System**	**Category**	**KA Statement**
011	Pressurizer Level Control	K4 Knowledge of PZR LCS design feature(s) and/or interlock(s) which provide for the following:	PZR level inputs to RPS

K/A# K4.05 **K/A Importance** 3.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.4 IF Rev. 11 pgs. 6, 7, 12
1OM-1.5.B.1 Rev 3 pg 2
1OM-1.5.B.2 Rev 9 pg 2, TS table 3.3.1-1 func. 17

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7)

Objective: 1SQS-6.4, Rev. 14 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.

58. Given the following conditions:

- 1OM-50.4.D2, Reactor Startup From Mode 3 To Mode 2 is in progress
- Shutdown Bank A and B rods are at the all-rods out position
- SR N-31 is reading 1.94E+2 CPS
- SR N-32 is reading 1.85E+2 CPS
- Both SR channels High Flux At Shutdown selector switches have been placed in the BLOCK position

Subsequently, after Control Bank withdrawal commences, the Instrument power fuse for N-31 blows.

The blown N-31 Instrument power fuse will cause the N-31 Level Trip bistable to be in a ____(1)____ condition, and an automatic Reactor Trip ____(2)____ occurred.

A. 1) tripped
   2) has

B. 1) tripped
   2) has **not**

C. 1) **not**-tripped
   2) has

D. 1) **not**-tripped
   2) has **not**

Answer: A

Explanation/Justification: K/A is met by demonstrating the knowledge of the nuclear instrumentation system response when the instrument power fuse for the source range nuclear Instrument N31 blows. The instrument power fuse is an interconnection between 120VAC vital bus power and the SR detector.

A. Correct. The level trip bistable will be tripped due to the loss instrument power and the Level Trip switch being in normal for this power condition (Below P-6 >1X10^{-10} Amps). Due to N-31 being de-energized causing the level trip bistable to be tripped, the1/2 SR high Rx trip has been met, and the Rx will be tripped.

B. Incorrect. The level trip bistable will be tripped. Second part is plausible if the candidate thinks the level trip bistable is tripped, but the Rx trip is blocked by the High Flux At Shutdown Block. This would be correct if the Level Trip Switch for N-31 was in Bypass but the switch must be in normal for operability prior to energizing the rod drive system.

C. Incorrect. Plausible if the candidate thinks that a loss of instrument power causes only the N-31 meters to fail downscale, but forgets that the power loss will cause the Level Trip bistable to be tripped. Rx trip is plausible if the candidate confuses the Instrument power with the Control power and thinks the Rx trip relay picks up.

D. Incorrect. Plausible if the candidate thinks that a loss of instrument power causes only the N-31 meters to fail downscale, but forgets that the power loss will cause the Level Trip bistable to be tripped. which would cause a Rx trip.

Sys #	System	Category	KA Statement
015	Nuclear Instrumentation	K6 Knowledge of the effect of a loss or malfunction on the following will have on the NIS:	Component interconnections

K/A# K6.03 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 3SQS-2.1 PPNT Rev 9 slide 31, 52, 114
U1 UFSAR Figure 7.2-1 sheet 3 Rev 17

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 45.7)

Objective: 3SQS-2.1, Rev. 9 Obj. 6 - Describe the required positions of all NIS control switches for a given power level.
3SQS-2.1, Rev. 9 Obj. 15 - Given an NIS failure, predict the NIS and interrelated system response for the given failure.

59. The plant is at 50% power with all systems in normal alignment for this power level.

- 1GW-TK-1A, Gaseous Waste Decay Tank discharge is in progress.
- RM-1VS-106, Waste Gas Tank Vault Ventilation Exhaust Gas Monitor is in HIGH-HIGH alarm.

Based on the above information, complete the following statements.

1) The Gaseous Waste Tank Vault ventilation flow path will be automatically ____________.
2) IAW RM-1VS-106 High-High ARP, the Gaseous Waste Decay Tank discharge will be ____________ terminated.

A. 1) isolated
2) automatically

B. 1) aligned to the main filter banks
2) automatically

C. 1) isolated
2) manually

D. 1) aligned to the main filter banks
2) manually

Answer: D

Explanation/Justification: K/A is met by demonstrating the ability to predict the response of the Waste Gas Disposal ventilation system to a Waste Gas Tank Vault Ventilation Exhaust Gas Monitor in HIGH-HIGH alarm.

A. Incorrect. Plausible distractor because isolating the Gaseous Waste Tank Vault sounds reasonable especially during a discharge to prevent the spread of activity. Plausible distractor because if RM-1GW-108B gaseous waste gas monitor were in high-high alarm, both the GW Decay Tank Bleed Valves and GW Decay Tank Disch to CTWR valve would close automatically.

B. Incorrect. RM-1VS-106 in High-high alarm will automatically align the MFBs. Plausible distractor because if RM-1GW-108B gaseous waste gas monitor were in high-high alarm, both the GW Decay Tank Bleed Valves and GW Decay Tank Disch to CTWR valve would close automatically.

C. Incorrect. Plausible distractor because isolating the Gaseous Waste Tank Vault sounds reasonable especially during a discharge to prevent the spread of activity. The GWDT discharge is required to be manually iaw 1OM-43.4.AEJ step 5. Automatic isolation of the GWDT discharge would occur if RM-1GW-108B were in high-high alarm.

D. Correct. RM-1VS-106 in High-high alarm will automatically align the MFBs by closing VS-D-4-1A MFB bypass damper, and opening VS-D-4-2A MFB inlet damper. The GWDT discharge is required to be manually iaw 1OM-43.4.AEJ step 5. Automatic isolation of the GWDT discharge would occur if RM-1GW-108B were in high-high alarm.

Sys #	System	Category	KA Statement
071	Waste Gas Disposal System (WGDS)	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Waste Gas Disposal System operating the controls including:	Ventilation system

K/A# A1.06 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-43.4.AEJ Rev 4 pg 2; 1OM-43.5.B.2 Rev 4 pg. 2 & 3

Question Source: New

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

Objective: 1SQS-16.1, Rev. 9 Obj. 13. Given a SLCRS configuration and without referenced material, describe the SLCRS control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable. c. High-high radiation condition.

1SQS-19.1, Rev. 17, Obj. 18. Given a Gaseous Waste Disposal System configuration and without reference material, describe the Gaseous Waste Disposal System control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable. a. High Radiation

60. The plant is at 100% power with all systems in normal alignment for this power level, **except** for the following:

- Control Rod Bank select Switch is in MANUAL.
- The Instrument air supply to the condenser steam dumps on Condenser 1B has been isolated due to an air leak.
- Steam Dump Control Mode Selector Switch is in STM PRESS.
- Cooldown Valves Controller AM-1MS-464B is in MANUAL with zero demand.

Subsequently, a 30% load rejection occurs with NO Operator actions taken.

1) How will the load rejection effect the fuel cladding temperature?
2) How will the RCS temperature be controlled?

A. 1) Fuel cladding temperature will RISE.
2) Available condenser steam dumps and the atmospheric steam dumps ONLY.

B. 1) Fuel cladding temperature will RISE.
2) Atmospheric steam dumps and the SG safety valves ONLY.

C. 1) Fuel cladding temperature will LOWER.
2) Available Condenser steam dumps ONLY.

D. 1) Fuel cladding temperature will LOWER.
2) Atmospheric steam dumps and the SG safety valves ONLY.

Answer: B

Explanation/Justification: K/A is met with the knowledge that with control rods and the steam dump controller in manual when a 30% load rejection occurs, that the fuel cladding temperature will rise because the condenser steam dumps will not operate but the atmospheric steam dumps and the SG safeties will control RCS temperature at a higher value.

A. Incorrect. Fuel cladding temperature will rise, but the condenser steam dumps will not open to control RCS temperature. Plausible because the condenser steam dumps will normally lower Tavg to within 3F of Tref, but not when the controller is in manual steam pressure control.

B. Correct. Fuel cladding temperature will rise as seen in the CETC temperatures rising. The Condenser steam dumps will not operate because the controller is in manual steam pressure mode, therefore, the atmospheric steam dumps will open at 1060 psig, and the first SG safeties will open at 1075 psig.

C. Incorrect. Plausible if the candidate thinks that the fuel cladding temperature will be lower due to the lower power after the load rejection. Available steam dumps are plausible because the condenser steam dumps will normally lower Tavg to within 3F of Tref, but not when the controller is in manual steam pressure control, and rods would also auto insert to restore Tavg, but not when in manual.

D. Incorrect. Plausible if the candidate thinks that the fuel cladding temperature will be lower due to the lower power after the load rejection. Atmospheric steam dumps and the SG safety valves is correct.

Sys #	System	Category	KA Statement
041	Steam Dump/Turbine Bypass Control	K5 Knowledge of the operational implications of the following concepts as the apply to the SDS:	Effect of power change on fuel cladding

K/A# K5.06 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-21.5.A.24 Rev 4; 1OM-21.2.B Rev. 6 pg. 2, 3

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.5 / 45.7)

Objective: 1SQS-21.1 Rev 16 Obj. 10 - Describe the control, protection and interlock functions for the field components associated with the Main Steam Supply System, including automatic functions, setpoints and changes in equipment status as applicable.

1SQS-21.1 Rev 16 Obj. 13 - Given a specific plant condition, predict the response of the Main Steam Supply 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.

61. The plant was operating at 100% power when a Load Rejection resulted in the following conditions:

- Reactor Power is 85%.
- Turbine Power is 800 MWe.
- A4-116, ROD CONTROL BANK D LOW is in alarm
- Tavg is 576°F and stable.
- Tref is 572°F and stable.

Which of the following completes the statements below?

1) The Control Rod Insertion Limit of the Core Operating Limits Report (COLR) _______________ been exceeded.

2) The next action that the operating crew is required to perform is _______________.

A. 1) has
2) initiate emergency boration IAW 1OM-7.4.S, Emergency Boration, and withdraw Control Rods

B. 1) has
2) trip the Reactor and enter E-0, Reactor Trip or Safety Injection

C. 1) has NOT
2) raise turbine load IAW 1OM-52.4.B.1, Turbine Load Changes to match Tavg/Tref, then withdraw Control Rods

D. 1) has NOT
2) initiate normal boration IAW 1OM-7.4.L, Blender Boration Operation, and withdraw Control Rods

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Answer: D

Explanation/Justification: K/A is met by demonstrating the ability to determine based on plant annunciators that the Control rod insertion limits were not exceeded during a load rejection, and demonstrates the knowledge that a normal boration must be completed iaw 1OM-7.4.L.

A. Incorrect. Plausible distractor if the candidate does not know that there are 2 different annunciators, and that control bank low is not exceeding RIL. Control bank low-low would indicate RIL was exceeded. If RIL were exceeded, emergency boration would be the correct response iaw ARP A4-124, Rod Control Bank D low-low.

B. Incorrect. Plausible distractor if the candidate does not know that there are 2 different annunciators, and that control bank low is not exceeding RIL. Control bank low-low would indicate RIL was exceeded. E-0 is plausible if the candidate thinks the Rx must be tripped when RIL is exceeded.

C. Incorrect. Correct that RIL has not been exceeded. Plausible distractor is the candidate thinks that Tavg/Tref mismatch/turbine load adjustment is the primary concern before the control rods can be withdrawn to clear A4-116 annunciator.

D. Correct. A4-116 alarms at 10 steps above A4-124 Bank D low-low alarm which indicates that RIL has been exceeded. The candidate must know that there are 2 different annunciators, and that control bank low is not exceeding RIL. Normal boration and rod withdrawal is the expected corrective actions iaw A8-116 ARP and AOP 1.35.2.

Sys #	System	Category	KA Statement
045	Main Turbine Generator	A2 Ability to (a) predict the impacts of the following malfunctions or operation on the MT/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Control rod insertion limits exceeded (stabilize secondary)

K/A# A2.12 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.4.ABA Rev 4 pg 2

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Objective: 3SQS-1.3, Rev. 7 Obj. 23. Discuss the Control Room indications and controls that are available to manipulate and monitor the Rod Control System.

62. Given the following plant conditions:

- The plant is operating at 100 % power with all systems in NSA.
- A 10 gpm Steam Generator Tube Leak occurs.
- A Hi-Hi Radiation signal is confirmed on RIS-1SV-100, "Condenser Air Ejector Discharge".
- All systems function as designed.

With no operator action, which of the following describes the plant/component response to this set of plant conditions?

TV-1SV-100A, Condenser Air Ejector to Containment Trip Valve ____(1)____ **AND**
TV-1SV-100B, Condenser Air Ejector to Gaseous Waste Trip Valve ____(2)____.

A. 1) OPENS
2) CLOSES

B. 1) CLOSES
2) OPENS

C. 1) REMAINS OPEN
2) REMAINS CLOSED

D. 1) REMAINS CLOSED
2) REMAINS OPEN

Answer: A

Explanation/Justification: K/A is met by demonstrating the knowledge of how the condenser air removal system air ejector discharge lineup is changed when a high-high alarm on the Condenser Air Ejector Discharge process rad monitor is in.

A. Correct. If a Hi-Hi radiation level is reached (stated in question stem), then TV-1SV-100A opens and TV-1SV-100B closes to reposition air ejector off gas from the gaseous waste system to the containment. This design in unique to Unit 1.
B. Incorrect. This is opposite of the expected response.
C. Incorrect. This is plausible if the candidate is not familiar with NSA and believes the Hi signal already caused the alignment to occur.
D. Incorrect. This would be the response for a Hi signal versus Hi-Hi signal.

Sys #	System	Category	KA Statement
055	Condenser Air Removal	K1 Knowledge of the physical connections and/or cause-effect relationships between the CARS and the following systems:	PRM system

K/A# K1.06 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-26.1.B, Rev. 11, pg. 34, 35
U1 RM-0426-006 Rev 16

Question Source: Bank – 1LOT8 NRC Exam Q61

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Objective: 1SQS-26.1 Rev 12 Obj. 13. Given a Main Turbine, Main Condenser, Condenser Air Removal system and MSR configuration and without referenced material; describe the Main Turbine, Main Condenser, Condenser Air Removal system and MSR control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable: c. Air Ejector Air Discharge Hi-Hi Radiation.

63. The plant is operating at 100% power when Feedwater Heater Bypass valve, TV-1CN-100 inadvertently OPENS.

What effect will this have on the plant operations?

A. Feedwater Inlet Temperature to the Steam Generators will **LOWER**.

B. Condenser Hotwell level will **RISE**.

C. Main Feed Pump Discharge pressure will **LOWER**.

D. Turbine Plant Demineralized Water Storage Tank will **RISE**.

Answer: A

Explanation/Justification: K/A is met by the candidate demonstrating an understanding of the effect that the Condensate Feedwater Heater Bypass valve opening will have on the Main Feedwater temperature entering the Steam Generators.

A. Correct. TV-1CN-100 open will bypass the feedwater heaters which is being preheated by extraction steam, with the condensate flowing through the heaters, feedwater inlet temp will lower.
B. Incorrect. This would be true if the bypass around the normal LCV was failed open. (LCV-1CN-103).
C. Incorrect. This would be true if the condensate pump Recirc valve (FCV-1CN-101) was failed partially open, but when TV-1CN-100 opens, MFP discharge pressure will rise.
D. Incorrect. This would be true if the bypass around the normal condensate pump reject MOV was failed open. (LCV-1CN-101).

Sys #	System	Category	KA Statement
056	Condensate	K1 Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following systems:	MFW

K/A# K1.03 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-22.1.D Rev 4 pg. 4
U1 RM-0422-002 Rev. 18

Question Source: Bank – 2LOT6 NRC Exam Q63

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Objective: 1SQS-22.1, Rev. 11 11. Given a Condensate System configuration and without reference material, describe the Condensate 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. The plant is at 100% power.

Which of the following actions will occur if Control Room Area Radiation Monitor, RM-1RM-218A spuriously fails HIGH?

Assume only this channel has failed.

Control Room Air Intake Damper, 1VS-D-40-1A will ____(1)____.
Control Room Air Intake Damper, 1VS-D-40-1B will ____(2)____.
Control Room Air Exhaust Damper, 1VS-D-40-1C will ____(3)____.
Control Room Air Exhaust Damper, 1VS-D-40-1D will ____(4)____.

A. 1) close
2) close
3) close
4) close

B. 1) remain as is
2) close
3) remain as is
4) close

C. 1) close
2) remain as is
3) close
4) remain as is

D. 1) remain as is
2) remain as is
3) remain as is
4) remain as is

Answer: C

Explanation/Justification: K/A is met by demonstrating the ability to determine how the Control Room ventilation lineup will automatically align when only one train of the control room area rad monitor spuriously fails high.

A. Incorrect. The High alarm from RM 218A will close the Train A Components, dampers 1VS-D-40-1A & 1C only.

B. Incorrect. The High alarm from RM 218A will close the Train A Components, not dampers 1VS-D-40-1B & 1D (Train B).

C. Correct. Must evaluate the action from a single Area RM in High, and whether a single train of dampers or both trains are affected, as well as the Automatic action for the intake and/or exhaust damper. The High alarm from RM 218A will close the Train A Components, which are dampers 1VS-D-40-1A & 1C.

D. Incorrect. The High alarm from RM 218A will close the Train A Components only, One intake damper 1VS-D-40-1A and one exhaust damper 1VS-D-40-1C receive the High signal. If student assumes one train Rad Monitor actuates both trains, then this answer is plausible.

Sys #	System	Category	KA Statement
072	Area Radiation Monitoring	A3 Ability to monitor automatic operation of the ARM system, including:	Changes in ventilation alignment

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

References provided to Candidate None **Technical References:** 1OM-43.5.B.2 Rev 4 Pg 3, 1OM-43.4.ADP Rev 3 pg 3, 1OM-44A.1.D Rev 7Pg 10

Question Source: Bank – 1LOT14 NRC Exam Q23 **(LAST 2)**

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5)

Objective: 1SQS-44A.1, Rev. 7 Obj. 13. Given a Unit-1 Control Area Ventilation System configuration and without referenced material, describe the System's Control Room response to the following actuation signals, including automatic functions and changes in equipment status as applicable. b. Control Room Area Radiation Monitor Actuation

65. The plant is at 100% power.

- Annunciator A6-65, Cooling Tower Blowdown Flow Low _10,000_ GPM is in alarm

Per the Alarm Response Procedure, which of the following pump operations will cause Cooling Tower Blowdown flow to **DECREASE**?

1. Reactor Plant River Water Pump [1WR-P-1A] Start
2. Reactor Plant River Water Pump [1WR-P-1A] Stop
3. Turbine Plant River Water Pump [1WR-P-6A] Start
4. Turbine Plant River Water Pump [1WR-P-6A] Stop
5. Cooling Tower Pump [1CT-P-1A] Start
6. Cooling Tower Pump [1CT-P-1A] Stop

A. 1, 2, 3, 4, 5, 6

B. 2, 4, 5, 6 ONLY

C. 1, 3, 5 ONLY

D. 2, 3, 6 ONLY

Answer: B

Explanation/Justification: K/A is met by understanding how a shutdown of a RPRW, TPRW, or startup/shutdown of a Cooling Tower Pump, will cause the Cooling Tower Blowdown flow to decrease. Cooling Tower blowdown is part of the Circulating Water System.

A. Incorrect. RP and TP RW pump starts will increase overall CW flow which increases blowdown flow. The RW pump stops will reduce blowdown flow, as will the starting or stopping of the Cooling Tower pumps. It will reduce CW basin causing blowdown flow to reduce.
B. Correct. Stopping the RW pumps will reduce blowdown flow by reducing overall CW flow. Cooling Tower pump starts or stops will also reduce blowdown flow since the system is open to atmosphere at the cooling tower and pump operation will affect basin level, reducing the amount of blowdown flow.
C. Incorrect. River Water pump starts will increase overall CW flow which increases blowdown flow. Cooling Tower pump start is correct. It will reduce CW basin causing blowdown flow to reduce.
D. Incorrect. TP River Water pump start will increase overall CW flow which increases blowdown flow. RP RW pump stop and CT pump stop will reduce blowdown flow.

Sys #	System	Category	KA Statement
075	Circulating Water	A4 Ability to manually operate and/or monitor in the control room:	Emergency/essential SWS pumps

K/A# A4.01 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-31.4.AAA Rev 2 pg 2

Question Source: Bank – 1LOT14 NRC Exam Q 64 **(LAST 2 EXAMS)**

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 to 45.8)

Objective: 1SQS-31.1, Rev. 14 12. Given a specific plant condition, predict the response of the Circulating 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.

66. The plant has experienced an event and the crew has entered the EOP network. The Unit Supervisor has assigned you as the RO to monitor two Continuous Action steps from E-1, Loss of Reactor or Secondary Coolant. The crew now transitions to FR-C1, Response to Inadequate Core Cooling.

The Continuous Actions from E-1 _______________.

A. remain applicable until superseded by directed actions of FR-C1.

B. remain applicable throughout the performance of FR-C1.

C. are NOT applicable upon entering FR-C1.

D. may be performed at the Unit Supervisors discretion upon entry into FR-C1.

Answer: C

Explanation/Justification: KA is met with knowledge of the conduct of ops in which procedure use is one of the RO knowledges from 1/2OM-48.2.C Conduct of Operations - Adherence and Familiarization to Operating Procedures. Sect VII.C.1 requires knowledge of procedure rules of usage as an admin requirement for RO's to understand and be able to adhere to without the procedure in hand.

A. Incorrect - Plausible if the student misunderstands the requirement for continuous action steps in the FRGs.

B. Incorrect. Plausible if the student confuses the transition to FRGs and other ORGs since normally they do apply.

C. Correct - Optimal Recovery Guideline actions are not to be performed while a Critical Safety Function is being restored from a RED or ORANGE condition iaw 1/2OM-53B.2 pg. 7.

D. Incorrect. Plausible because this could be correct if the transition was to another ORG, but 1/2OM-53B.2 states, Optimal Recovery Guideline actions are not to be performed while a Critical Safety Function is being restored from a RED or ORANGE condition.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of conduct of operations requirements.

K/A# 2.1.1 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1/2OM-48.2.C Rev 24 pg. 9; 1/2OM-53B.2 rev 9 pg. 6-7

Question Source: Bank – Wolfe Creek 2015 NRC Exam Q66

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 45.13)

Objective: 3SQS-53.1, Rev. 2 Obj. 1 - 1. State from memory "All" of the Emergency Operating Procedures user's guide rules of usage as defined in 1/2OM53B.2.

67. Given the following:

- A MANUAL SI was initiated in response to a LOCA.
- The operating crew is performing ES-1.2, Post LOCA Cooldown and Depressurization.
- No RCP's are running.
- BOTH HHSI pumps are running, injecting through the BIT.
- The crew is depressurizing the RCS to refill the pressurizer.
- RCS Pressure is 1000 PSIG.
- RCS Subcooling is at 25°F (20°F is currently required in accordance with Attachment 6-A, for SI Reinitiation criteria).

Per ES-1.2, which of the following describes the required action if subcooling decreases below 20°F during the depressurization to refill the pressurizer?

A. Continue the depressurization - subcooling will be restored after the depressurization.

B. Continue the depressurization - the goal is to maintain the RCS at saturated conditions.

C. Terminate the depressurization - to avoid bubble formation in the reactor vessel head.

D. Terminate the depressurization - to preclude uncontrolled SI accumulator discharge.

Answer: A

Explanation/Justification: K/A is met by demonstrating the knowledge of specific procedural steps during the performance of ES-1.2, Post LOCA Cooldown and Depressurization, and determining that that RCS depressurization does not need to be stopped due to subcooling requirements.

A. Correct. The background on ES-1.2 mitigating step 2 'Depressurize RCS to refill PRZR' states "if subcooling is lost during the depressurization, it should be restored as the cooldown continues."
B. Incorrect. Goal is not to maintain saturated conditions, it is to maintain minimum subcooling for cooldown and depressurization.
C. Incorrect. Would not terminate depressurization if subcooling was too low, part of strategy to continue. A note prior to the depressurization states that a void may occur, resulting in rapid przr level rise.
D. Incorrect. Would not terminate, and accumulator discharge would not occur until pressure was lower.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A# 2.1.23 **K/A Importance** 4.3 **Exam Level** RO
References provided to Candidate None **Technical References:** 1OM-53B.4.ES-1.2(ISS3) Rev 1 pg. 2 & 31
Question Source: Bank – 1LOT7 NRC Exam Q14
Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.2 / 45.6)
Objective: 3SQS-53.3, Rev. 5 3. State from memory the basis and sequence for the Major Action Steps of each EOP procedure, IAW BVPS-EOP Executive Volume.

68. In accordance with NOP-OP-1001, Clearance/Tagging Program, a component clearance will require double isolation if the fluid temperature or pressure are greater than ___________.

(Select the answer that meets BOTH limits)

A. 150°F or 1000 psig

B. 180°F or 750 psig

C. 200°F or 500 psig

D. 225°F or 300 psig

Answer: C

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. 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.

D. Incorrect. Plausible arbitrary values for temperature and pressure.

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. 25 pg. 7
GENTAGOVERVIEW_FEN PPNT Rev 3 slide 24

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 45.13)

Objective: GENTAGOVERVIEW_FEN Discuss tagging of the following components: - Control switches, Fuses, Circuit Breakers, Mechanical Systems.

69. Using Operations Manual (OM) Figure 21-2 as a reference, which of the following describes information that can be obtained from this mechanical print?

1. Valves which can be operated from the Main Control Boards
2. System parameter monitoring instrumentation
3. System flow direction
4. Piping size
5. Plant Computer input
6. Normal system operating setpoints

(See attached drawing)

A. 1, 2, & 3

B. 4, 5, & 6

C. 1, 3, 4 & 6

D. 2, 3, 4, & 5

Answer: D

Explanation/Justification: K/A is met by demonstrating the ability to interpret the Main Steam mechanical drawing to determine available information found on the print.

A. Incorrect. System parameter monitoring instrumentation and System flow direction are identified on the print. Valves which can be operated from the Main Control Boards are not identified any different than a valve controlled locally.

B. Incorrect. Piping size and Plant Computer inputs are identified on the print. Normal system operating setpoints are not available on the print.

C. Incorrect. System flow direction and Piping size are identified on the print. Valves which can be operated from the Main Control Boards are not identified any different than a valve controlled locally. Normal system operating setpoints are not available on the print.

D. Correct. Pressure, temperature, and flow transmitters are designated by mark numbers and some have identified symbols. System flow direction is designated with black arrow heads. Piping size can be obtained by the numbers and arrows followed by the inch sign. Plant computer inputs are identified by triangles with a 'C' inside.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to obtain and interpret station electrical and mechanical drawings.

K/A# 2.2.41 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate OP Manual Fig. 21-2, Rev. 20 **Technical References:** Fig 21-2 – U1 RM-0421-002 Rev 20

Question Source: Modified – 2LOT7 NRC Exam Q70

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 45.12 / 45.13)

Objective:

70. Which of the following plant conditions/evolutions can result in significantly higher radiation levels in the Safeguards Building?

A. Venting an idle charging pump IAW 1OM-7.4.AV, Charging Pump Suction Header Venting.

B. Performing the Low Head SI Pump Test IAW 1OST-11.1, Safety Injection Pump Test - 1SI-P-1A.

C. Transferring to Cold Leg Recirculation IAW ES-1.3, Transfer to Cold Leg Recirculation.

D. Placing the deborating demineralizer in operation IAW 1OM-7.4.AW, Mixed Bed and Deborating Demineralizer Operations.

Answer: C

Explanation/Justification: K/A is met with the knowledge that radiation levels will rise in the safeguards building when Transfer to Cold Leg Recirculation occurs in ES-1.3 due to cnmt sump water will flow through the Recirc piping in the safeguards building.

A. Incorrect. This is a plausible evolution which is a radiation hazard and requires RP assistance due to the potential for high radioactive gaseous release. This hazard is in the PAB as opposed to the safeguards area.
B. Incorrect. LHSI Pumps are located in Safeguards and this evolution recirculates the RWST through the safeguards which makes this distractor plausible. However, this evolution should not increase radiation levels in safeguards.
C. Correct. The candidate must have knowledge of radiation or contamination hazards that may arise during any plant activity. Specifically, they must sort through a list of valid situations and determine that transfer to cold leg recirculation during a LOCA has the greatest potential to increase Safeguards and/or PAB radiation levels. ES-1.3 has a caution prior to step 1 that warns the operator of this hazard.
D. Incorrect. This evolution has a potential to increase radiation levels, however, the procedure is more concerned with the potential reactivity event which could occur as a result of this evolution. Increased radiation levels would be more of a concern in the PAB as opposed to Safeguards.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

K/A# 2.3.14 **K/A Importance** 3.4 **Exam Level** RO

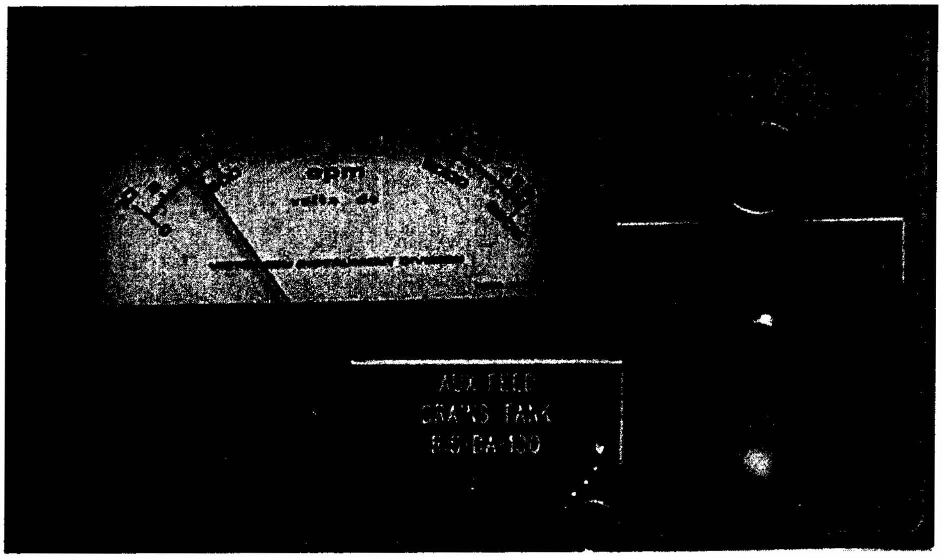
References provided to Candidate None **Technical References:** 1OM-53A.1.ES-1.3 Iss 3 Rev 0 (step 1 caution)

Question Source: Bank – 2LOT8 NRC Exam Q71

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.12 / 43.4 / 45.10)

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

71.



Refer to the above photograph of the AUX FEED DRAINS TANK monitor control module.

The High-High setpoint will be displayed on the meter when the high-high pushbutton is depressed concurrent with what other action?

A. The c.s. pushbutton is depressed.

B. The rotary switch is placed in the cal. position.

C. The rotary switch is placed in the h.v. position.

D. The rotary switch is placed in the oper. position

Answer: B

Explanation/Justification: K/A is met by demonstrating the knowledge of how to test the high-high alarm setpoint on the Auxiliary Feedwater Area Drain Tank fixed radiation monitor.

A. Incorrect. The c.s (check source) pushbutton when pushed will unshield a source and provide a meter deflection to ensure the meter is properly functioning. This is more plausible than the fourth position (off) on the rotary cal switch but not correct.

B. Correct. Placing the switch in the cal position will allow the high radiation and high-high radiation alarm setpoints to be displayed when the respective high or high-high pushbutton is pressed.

C. Incorrect. This position will not cause the setpoint to be displayed. Plausible since it is one of the four positions

D. Incorrect. This position will not cause the setpoint to be displayed. Plausible since it is one of the four positions.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

K/A# 2.3.15 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-43.4.C, Rev. 8, pg. 4

Question Source: Bank – 1LOT8 NRC Exam Q72

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.12 / 43.4 / 45.9)

Objective: 1SQS-43.1-01-1: Describe the function of the Radiation Monitoring systems and the associated major components.

72. Given the following plant conditions:

- You are directed to post a clearance in the Primary Auxiliary Building (PAB).
- This task requires entry into an Area where General Radiation Levels are 110 mr/hr.

In addition to reviewing your approved RWP and radiological conditions, which of the following item(s) will be required prior to entry into this area IAW NOP-OP-4101 **AND** NOP-OP-4107?

(NOP-OP-4107, "Radiation Work Permit")
(NOP-OP-4101, "Access Controls for Radiologically Controlled Areas")
(Note this is **NOT** an all-inclusive list)

1. TLD and a Direct Reading Dosimeter.
2. TLD and an Alarming Direct Reading Dosimeter with appropriate augmentation.
3. Self Briefing documented on NOP-OP-4101-06, "RA Request and Briefing Form".
4. A Trip Ticket that has been initialed by a Radiation Protection Technician.

A. 1 **ONLY**.

B. 1 & 3 **ONLY**.

C. 2 & 4 **ONLY**.

D. 2, 3, & 4 **ONLY**.

Answer: C

Explanation/Justification: K/A is met by demonstrating the ability to comply with the radiation work permit requirements during clearance post in the PAB, and identify that a TLD and an Alarming Direct Reading Dosimeter, and a Trip Ticket is required for entry.

A. Incorrect. Incorrect since this is a HRA. Plausible and correct for RA entry. (refer to correct answer explanation).

B. Incorrect. Incorrect because NOP-OP-4101 does not allow self-briefing for entry into a HRA. Plausible if the candidate does not recognize this is a HRA does not know the Access Control & Briefing requirements do not allow self-briefs into a RCA when dose is >25mrem. (refer to correct answer explanation).

C. Correct. NOP-OP-4101 defines a HRA an accessible area in which radiation levels could result in an individual receiving a deep-dose equivalent in excess of 100 mr/hr at a distance of 30 cm or more from any surface that the radiation penetrates. The candidate must recognize that they are entering a HRA versus RA and then differentiate the briefing and access requirements in order to comply with RWP requirements. NOP-OP-4107, "Radiation Work Permit" requires radiation protection briefings to ensure compliance with RWP requirements when entering HRAs (sect 4.5.1). For entry into radiation areas considered to be low risk, operators are allowed to self brief and do not need formal briefings. According to NOP-OP-4101 sect. 4.5.4, for entry into a HRA, a trip ticket and dose alarm augmentation device are required. This is a higher level question because it goes beyond simple recall and requires comprehension of the operationally relevant task to be performed.

D. Incorrect. Incorrect because NOP-OP-4107 allows self briefing for entry into low risk areas ONLY. Since this is a HRA it is considered higher risk. Plausible if the candidate recognizes this is a HRA but does not know the RWP briefing requirements. (refer to correct answer explanation).

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:** NOP-OP-4101, Rev. 14, pgs. 4, 7, 11
NOP-OP-4107, Rev. 18, pgs. 16 & 17

Question Source: Bank – 1LOT8 NRC Exam Q73

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.12 / 45.10)

Objective: 3SSG-Admin Obj. 16 - Describe the controls for maintaining personnel exposures ALARA in accordance with NOP-OP-4107, Radiation Work Permit (RWP).

73. A Rx trip has occurred. The following conditions exist.

- The Immediate Operator Actions of E-0, Reactor Trip or Safety Injection are complete.
- 'A' SG has a SGTR.
- 'B' SG Main Steam Safety Valve is stuck OPEN.
- Aux Feedwater flow is 800 gpm.
- CNMT pressure is -1.2 psig and stable.
- SG NR levels are as follows:

A	22%
B	28%
C	27%

IAW the EOP Users Guide, which, if any, of the following actions may be taken prior to the EOP step directing isolation of feedwater flow to the SGs?

A. Isolate 'A' SG, but NOT 'B' SG

B. Isolate 'B' SG, but NOT 'A' SG

C. Isolate BOTH 'A' and 'B' SGs

D. Do not isolate any SGs

Answer: B

Explanation/Justification: K/A is met by demonstrating knowledge of general guidelines for EOP usage, specifically for the use and applicability of preemptive action steps which may be taken to isolate feedwater to a faulted SG or a SG with a tube rupture.

A. Incorrect. Plausible distractor is the candidate confuses the steps to isolate the tube rupture without concern for SG NR, and feels that the faulted SG level must be >31%.

B. Correct. The preemptive steps identified in the EOP Users Guide state that feedwater may be isolated to a faulted SG to prevent excessive cooldown without concern of SG NR level provided heat sink requirements are maintained. A SG with a tube rupture must have a minimum SG NR level of 31% (50% adverse) prior to isolating feedwater.

C. Incorrect. Plausible distractor if the candidate knows feedwater must be isolated to both SGs but does not know that the tube rupture requires >31% prior to isolating.

D. Incorrect. Plausible distractor if the candidate thinks both the faulted SG, and the ruptured SG require >31% NR level.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Generic	Knowledge of general guidelines for EOP usage.

K/A# 2.4.14 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1/2OM-53B.2 Rev. 9 pgs. 7, 29, 30

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 45.13)

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. a. For a given event, apply the EOP users guide rules of usage as defined in OM-53B.2.

74. Related to Emergency Operating Procedures (EOPs) rules of usage, which of the following completes the statements below?

1) The step below is a (an) ______________ action step.

3. **Check**

2) Steps which may be performed in any order are designated by ______________.

A. 1. immediate
   2. asterisks

B. 1. immediate
   2. bullets

C. 1. continuous
   2. asterisks

D. 1. continuous
   2. bullets

Answer: B

Explanation/Justification: The K/A is matched because it requires the candidate to have knowledge of symbols used in the Emergency Operating Procedures (EOPs).

A. Incorrect. First part is correct. Second part is plausible because asterisks are another form of non-sequential characters which could be used as step designators, but they are not in the EOPs. Asterisks are used to distinguished CAUTIONS by a line of asterisks above and below them.

B. Correct. Immediate Action Steps are located at the beginning of Procedures E-0, ECA-0.0 and FR-S.1. These steps are required to be performed by memory when the EOPs are entered. These types of steps are identified by a circle around the action/expected condition step number. If sequence of performance is not important, the subtasks are designated by bullets (•).

C. Incorrect. First part is plausible because continuous action steps are use throughout the EOPs, but they are identified by triangle around the action step number. Second part is plausible because asterisks are another form of non-sequential characters which could be used as step designators, but they are not in the EOPs. Asterisks are used to distinguished CAUTIONS by a line of asterisks above and below them.

D. Incorrect. First part is plausible because continuous action steps are use throughout the EOPs, but they are identified by triangle around the action step number. Second part is correct.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Generic	Knowledge of EOP layout, symbols, and icons.

K/A# 2.4.19 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1/2OM-53B.2 Rev. 9 pgs. 3 & 6

Question Source: Bank – McGuire 2015 NRC Exam Q73

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 45.13)

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. a. For a given event, apply the EOP users guide rules of usage as defined in OM-53B.2.

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

- A Plant Startup in accordance with 1OM-52.4.B, Load Following.
- Reactor power is 5% and slowly RISING.
- A4-126, ROD BOTTOM ROD DROP is received.
- The Reactor Operator verifies only ONE Rod Bottom light is LIT.
- Before the crew takes any actions, additional alarms are received in the Control Room:

- A4-23, PRESSURIZER 2/3 PRESS RELIEF BLOCK
- A4-105, ROD CONTROL SYSTEM URGENT ALARM
- A4-41, LOOP TAVG LOW-LOW

Which of these plant conditions **REQUIRES** a manual reactor trip?

A. A4-126, ROD BOTTOM ROD DROP

B. A4-41, LOOP TAVG LOW-LOW

C. A4-23, PRESSURIZER 2/3 PRESS RELIEF BLOCK

D. A4-105, ROD CONTROL SYSTEM URGENT ALARM

Answer: B

Explanation/Justification: K/A is met by the candidate demonstrating the ability to prioritize and interpret annunciators received after a control rod drops during a plant startup, and determine which alarm will require a reactor trip.

A. Incorrect. According to ARP for A4-126F, a reactor trip is only required if two or more rod bottom indications are received. The ARP refers the operators to AOP 1.1.8.

B. Correct. According to the Transient Response Guidelines and AOP 1.1.8, when Tavg drops to 541 F, the reactor shall be tripped. 1OM-1.4.ACC Alarm Response Procedure for Loop Tavg low-low states that this alarm is received loop Tavg input sensing < 541 F. AOP 1.1.8 step 3 furthermore requires a reactor trip if RCS TAvg < 541 F.

C. Incorrect. According to 1OM-6.4.AAM, this alarm is received at 2000 psig. This is a possible alarm on a dropped rod due to decreasing RCS pressure. The low pressure reactor trip does not occur until 1945 psig and the ARP does not require a reactor trip.

D. Incorrect. This alarm could potentially have resulted in the dropped rod. The candidate could mistake this alarm for a General Warning Alarm. The candidate needs to understand that this alarm will block auto and manual rod motion but does not require a reactor trip.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to prioritize and interpret the significance of each annunciator or alarm.

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

References provided to Candidate None **Technical References:** 1OM-53C.4.1.1.8 Rev 5 pg. 2, 3; 1OM-1.4.AAC Rev 8 pg. 2; BVBP-OPS-0024 Rev 11 pg. 16

Question Source: Bank – 2LOT7 NRC Exam Q75

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.3 / 45.12)

Objective: 3SQS-53.5, Rev. 4 Obj. 14 - Describe the actions for a rod position indication malfunction.

76. An ATWS has occurred from 100% power.

- The crew is performing FR-S.1, Response to Nuclear Power Generation/ATWS.
- The RO is inserting control rods in MANUAL.
- SI has actuated.
- All SG pressures are approximately 550 psig and slowly LOWERING.
- RCS Temperature is approximately 490°F and slowly LOWERING.
- Reactor Power indicates approximately 4% and slowly LOWERING.
- Intermediate Range Startup Rate is slightly NEGATIVE.

The Unit Supervisor is currently reading Step 7, Check if Reactor is subcritical.

Which of the following describes the mitigation strategy for the current conditions?

A. Remain in FR-S.1 and isolate the faulted SGs. Transition to E-0, Reactor Trip or Safety Injection when isolation is complete.

B. Remain in FR-S.1 and continue inserting control rods until adequate subcriticality is obtained. Transition to E-0, Reactor Trip or Safety Injection when Shutdown Margin is verified.

C. Exit FR-S.1. Transition to E-0, Reactor Trip or Safety Injection, and subsequently isolate the faulted SG using E-2, Faulted Steam Generator Isolation.

D. Exit FR-S.1. Transition to E-2, Faulted Steam Generator Isolation, and isolate the faulted SG.

uestion 76

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 knowledge of diagnostic steps and decision points in the emergency operating procedures (EOPs) that involve transitions to event-specific sub-procedures or emergency contingency procedures. For this question the SRO must diagnose that the reactor is subcritical, and a transition back to E-0 will be made, then the faulted SG will be dealt with iaw E-2.

K/A is met by evaluating plant conditions during an ATWS, and determine that the Rx is subcritical, and a transition out of FR-S.1 back to E-0 is warranted because the Rx is shutdown.

A. Incorrect. Plausible distractor because FR-S. 1 step 13 identifies, then isolates a faulted SG, and step 17 at the end of FR-S.1 would transition back to E-0.
is below 5%

B. Incorrect. Plausible if the candidate does not know the FR-S.1 criteria for subcriticality, and thinks SDM validation is the criteria for transitioning back to E-0. This is incorrect because the Rx power is 4% and there is a negative SUR, therefore the Rx is subcritical.

C. Correct. Step 7 of FR-S.1 is a continuous action step that checks for subcriticality by observing PR <5%, negative SUR, continued boration which is accomplished with SI injecting. If these criteria are met a transition back to E-0 is directed. Once the crew identifies the faulted SG in E-0, a transition will be made to E-2.

D. Incorrect. Would not transition to E-2 directly from FR-S. 1. Must go back to E-0 first.

Sys #	System	Category	KA Statement
000007	Reactor Trip, Stabilization, Recovery / 1	EA2 Ability to determine or interpret the following as they apply to a reactor trip:	If reactor should have tripped but has not done so, manually trip the reactor and carry out actions in ATWS EOP

K/A# EA2.04 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-S.1 Iss 3 rev 1 step 7
1/2OM-53B.2 Rev 9 pgs. 4-5 & 12-14

Question Source: Bank - 1LOT7 NRC Exam Q82

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR 41.7 / 45.5 / 45.6)

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.

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.

77. Given the following conditions:

- A LOCA has occurred from 50% power 35 minutes ago
- The crew is performing E-1, Loss of Reactor or Secondary Coolant
- RCS pressure is 90 psig and lowering
- RCS cold leg temperatures are all < 230°F
- HHSI to BIT Flow is 690 gpm and STABLE
- Total LHSI flow is 4850 gpm and STABLE
- RWST level is 20' 4"

What action is required by the crew, and why?

(Reference provided)

FR-P.1 - Response to Imminent Pressurized Thermal Shock Condition
E-1 - Loss of Reactor or Secondary Coolant
ES-1.3 - Transfer to Cold Leg Recirculation

A. Transition to FR-P.1, perform applicable steps, then return to procedure and step in effect when directed because for a large break LOCA, PTS is not a serious concern.

B. Transition to FR-P.1, perform ALL steps, then return to procedure and step in effect because RCS conditions may result in thermal shock to the vessel causing a small flaw to grow.

C. Continue in E-1 until directed to transition to another procedure because for a large break LOCA, PTS is not a serious concern.

D. Transition to ES-1.3, perform applicable steps until directed to transition to another procedure because Cold Leg Recirculation Switchover Criterion has been met.

uestion 77

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E page 6. Specifically, the knowledge of diagnostic steps and decision points in the emergency operating procedures (EOPs) that involve transitions to event-specific sub-procedures or emergency contingency procedures. For this question the SRO must diagnose that Vessel Integrity has been violated and a transition must be made, but also must know that the second step of the procedure sends them back to E-1 due to the large break LOCA.

K/A met by evaluating plant conditions after a large break LOCA, and determining that PTS operation curves have been violated and a transition to FR-P.1 must be made, and also must know that for a large break LOCA PTS is not a serious concern and transition back to E-1 will be required.

A. Correct. A Red path for Integrity (FR-P.1) is present, therefore EOP rules of usage requires a transition to FR-P.1. FR-P.1 step 2 checks to see if a LBLOCA has occurred by checking RCS pressure, then LHSI flow. At this point, FR-P.1 will direct a transition back to E-1 SIE. The reason is correct also because a LBLOCA drops RCS pressure, therefore PTS is not a serious concern.

B. Incorrect. Plausible distractor if the candidate does not recognize that a LBLOCA has occurred or does not know that FR-P.1 step 2 checks the RCS pressure and LHSI flow.

C. Incorrect. Plausible distractor because it is true that FR-P.1 step 2 will direct procedure use back to E-1 because PTS is not a concern when a LBLOCA has occurred, but EOP rules us usage requires transition to FR-P.1 from E-1.

D. Incorrect. Plausible distractor because transition to ES-1.3 is an E-1 LHP step, but RWST level of 19' was not met in the question stem.

Sys #	System	Category	KA Statement
000011	Large Break LOCA / 3	EA2 Ability to determine or interpret the following as they apply to a Large Break LOCA:	Actions to be taken if limits for PTS are violated

K/A# EA2.14 **K/A Importance** 4.0 **Exam Level** SRO

References provided to Candidate 1OM-53A.1.5-D Iss 1C Rev 0

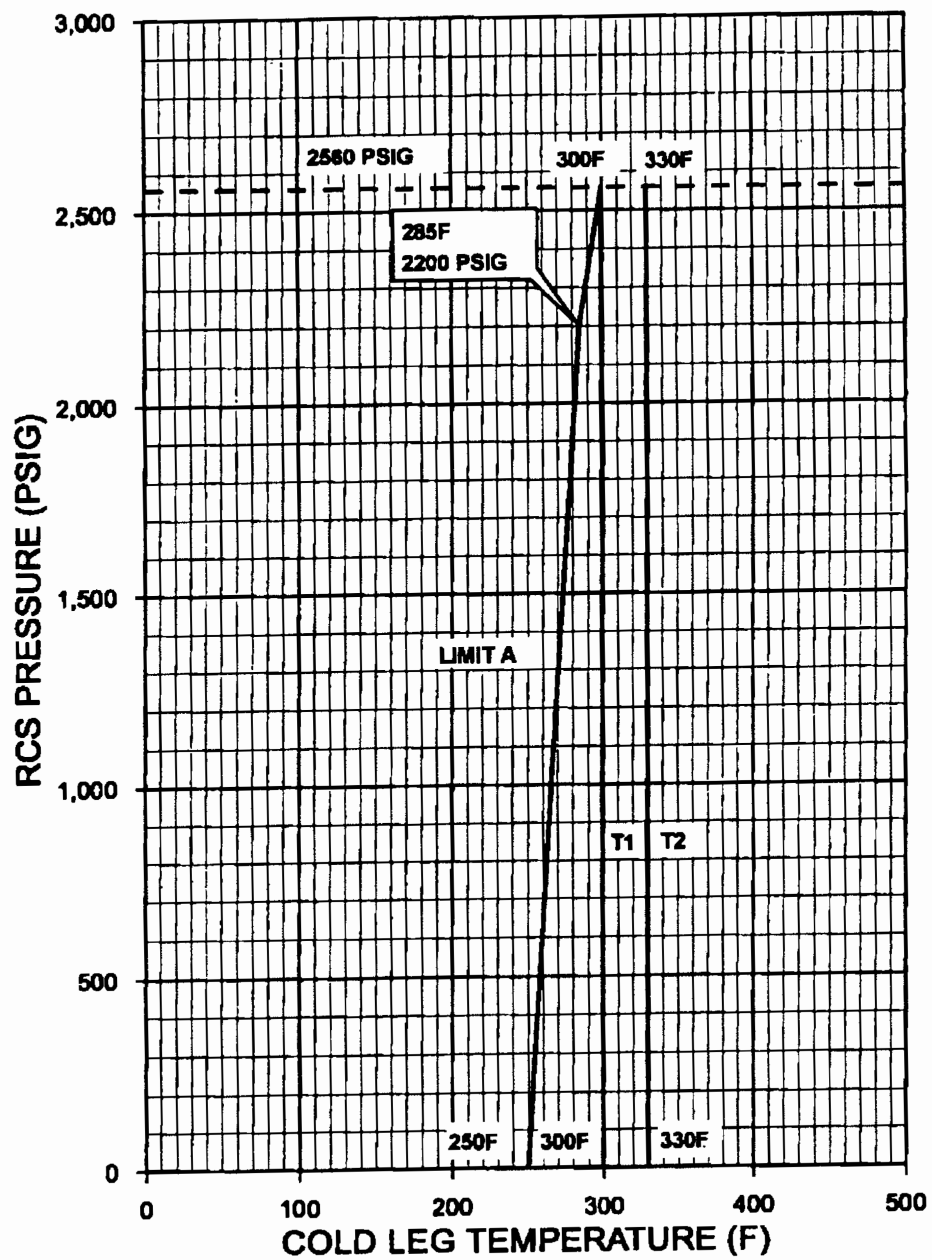
Technical References: 1OM-53A.1.F-0.4 Iss 3 Rev 0 pg. 1
1OM-53A.1.5-D Iss 1C Rev 0 pg. 1
1OM-53A,1,FR-P.1 Iss 3 Rev 0 step 2

Question Source: New

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

bjective:

Number	Title	
5-D	PTS - Operational Limits Curve	Issue 1C Revision 0



78. The plant is cooling down for a refueling outage.
    - RHR is in service with 'A' RHR pump running
    - 'B' RHR pump is on clearance
    - All RCS loops are operable
    - RCS Cold leg temperature is 310°F
    - RCS pressure is 315 psig

    A loss of RHR has occurred due to 'A' RHR pump tripping on overcurrent.
    - AOP-1.10.1, Loss of Residual Heat Removal Capability is in progress
    - The crew is at step 15, Pressurize or Makeup to RCS in preparation to startup a RCP

    If pressure continues to rise during this step, in what order will the following overpressure protection components operate?

    1) PCV-RC-456, Power Operated Relief Valve
    2) MOV-RH-700, 701, 720A and 720B, RHR Suction and Discharge valve auto closure interlock
    3) PCV-RC-455C and 455D, Power Operated Relief Valves
    4) RV-1RH-721, Residual Heat Removal PP Suction Header Relief

A. 1, 3, 4

B. 3, 4, 1

C. 3, 4, 2, 1

D. 1, 2, 4, 3

uestion 78

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E page 6. Specifically, the SRO must have specific knowledge of the content of the procedure that is being used. In this case the SRO must have specific knowledge that the loss of RHR AOP will pressurize the RCS and check for OPPS to be in service. The SRO will also know that OPPS is placed in service, and the auto closure features of the RHR isolation valves is defeated in step 8 of the RHR startup procedure. Therefore, for the conditions stated in the stem, procedurally OPPS must be in service, and the interlocks must be defeated.

K/A is met by determining what methods of RHR overpressure protection are available when in the loss of RHR procedure at the point of preparing to start an RCP and checking RCS pressure.

A. Incorrect. Plausible since 1) PCV-RC-456 is a PORV, but it is not OPPS and the setpoint is 2335 psig. 3) OPPS uses PORVs PCV-RC-455C and 455D being set at ≤ 397 psig, and 4) RV-1RH-721 is set at 600 psig. Therefore, this is the incorrect order of components.

B. Correct. Step 15 of the loss of RHR AOP pressurizes the RCS in preparation for starting an RCP and has the crew check that OPPS is in service. 1) OPPS consists of both PCV-RC-455C and 455D being set at ≤ 397 psig, 4) RV-1RH-721 is set at 600 psig, and 1) PCV-RC-456 is a PORV, but it is not OPPS and the setpoint is 2335 psig. This is the correct order. Item 2 interlocks are defeated after OPPS is placed in service.

C. Incorrect. Plausible because this would be the correct order if the auto closure interlocks were not defeated during the RHR startup procedure, when OPPS is placed in service.

D. Incorrect. Plausible if the candidate doesn't know that PCV-RC-456 is not OPPS and confuses the order of the components.

Sys #	**System**	**Category**	**KA Statement**
000025	Loss of Residual Heat Removal System / 4	AA2. Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System:	Existence of proper RHR overpressure protection

K/A# AA2.06 **K/A Importance** 3.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** U1 LRM pg. 5.2-14
1OM-6.2.B Rev. 18 pg. 9, 10
1OM-10.2.B Rev. 5 pg. 3
1OM-10.4.A Rev 36 pg 28

Question Source: New

uestion Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 43.5 / 45.13)

Objective: 3SQS-RCS ITS, Rev. 1

Obj. 2. State the purpose of each RCS specification as described in the Applicable Safety Analyses section of the Bases.

Obj. 3. Given plant conditions, determine the criteria necessary to ensure compliance with each RCS LCO and Licensing Requirement in accordance with the Bases, Surveillance Requirements, and the Applicability.

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

- The plant is at 80% power.
- 1CH-P-1A, 'A' Charging pump is in service.
- 1CC-P-1A, Reactor Plant Comp Cooling Water pump is in service.
- A Loss of Emergency Bus 1AE occurs.
- The crew entered AOP 1.36.2, "Loss of 4KV Emergency Bus".
- 1CH-P-1B, 'B' Charging pump was started.
- 1CC-P-1B, Reactor Plant Comp Cooling Water pump trips on overcurrent.

Based on these plant conditions and in accordance with the applicable procedure, what sequence of actions is required?

A. Trip Reactor, complete IOAs of E-0, then trip RCPs, Close PRZR Spray Valves.

B. Trip Reactor, complete IOAs of E-0, then trip RCP's, Isolate letdown, Transfer charging pump suction to the RWST.

C. Trip Reactor, complete IOAs of E-0, Isolate letdown, Minimize Charging flow.

D. Isolate letdown, Minimize Charging flow, Rack 'C' CCR pump on to DF bus and restore CCR flow.

Question 79

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E page 8 first bullet. Specifically, the SRO must make an assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed. Specifically recognize that a loss of CCR event has occurred, and actions within the Loss of CCR AOP requires a reactor trip, and other specific actions due to the loss of cooling water.

K/A is met by demonstrating the ability to recognize that along with the loss of the emergency 4KV bus, Component Cooling Water was also lost during the event, and the loss of CCR AOP must be entered and complied with in a timely manner due to loss of cooling to the RCPs.

A. Incorrect. Plausible actions for losing CCR to the RCPs, but these are not the only actions, and closing the przr spray valves have no affect on the RCS.

B. Correct. When the candidate recognizes that there had been a loss of Primary Component Cooling Water (CCR), it is expected that the loss of CCR AOP would be entered. The AOP step 2 RNO states if no CCR pumps are running, trip rx, enter E-0, trip RCPs, isolate letdown, transfer charging suction to RWST.

C. Incorrect. Plausible actions because for these conditions the reactor would be tripped. Isolating letdown due to the loss of CCR to the NRHX would make sense, as would minimizing charging flow when letdown is isolated.

D. Incorrect. Plausible because isolating letdown due to the loss of CCR to the NRHX would make sense, as would minimizing charging flow when letdown is isolated, and racking in 'C' CCR pump makes logical sense since it is available to the plant but racking in the breaker would take too long and would violate the AOP. 'C' CCR pump cannot be racked to either bus in NSA due to trip interlocks with the CCR pump on the bus.

Sys #	System	Category	KA Statement
000026	Loss of Component Cooling Water / 8	Generic	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

K/A# 2.4.47 **K/A Importance** 4.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.15.1 Rev. 6 step 2.

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 / 45.6)

Objective: 1SQS-53C.1, Rev. 12 Obj. 5. Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.

Obj. 6. Given a set of conditions, apply the correct AOP.

80. The plant is in Mode 3
A loss of Switchyard 138KV Bus #2 has occurred
Annunciator A8-31, SYS STA SERV TRANS 1B UNDERVOLTAGE is in ALARM
All automatic actions occur as designed.

1) Which of the following actions must be completed IAW ARP A8-31?

2) After the initial 1OST-36.7, Offsite to Onsite Power Distribution System Breaker Alignment Verification is completed, what is the **maximum** time until 1OST-36.7 must be completed again if the condition is not corrected?

PCB-83 1B + 3B STA SERVICE FDR 138KV BREAKER
ACB-241B 4160V BUS 1C SUPPLY BREAKER
ACB-341B 4160V BUS 1D SUPPLY BREAKER

A. 1) Verify ACB-241B and ACB-341B are OPEN.
2) 8 hours from the initial completion.

B. 1) Verify ACB-241B and ACB-341B are OPEN.
2) 10 hours from the initial completion.

C. 1) CLOSE PCB-83 to satisfy the interlock to permit closing ACB-241B and ACB-341B.
2) 8 hours from the initial completion.

D. 1) CLOSE PCB-83 to satisfy the interlock to permit closing ACB-241B and ACB-341B.
2) 10 hours from the initial completion.

QUESTION 80

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 surveillance requirements (TS Section 4) in accordance with rules of application requirements, and understand that surveillance requirement SR 3.0.2 allows for the extension.

K/A is met by giving the candidate a loss of offsite power condition and the candidate must demonstrate the ability to verify the normal 4160VAC supply breakers open iaw the ARP for Sys Sta Serv Trans 1B Undervoltage.

A. Incorrect. First part is correct. 8 hours is plausible because it is the time associated with TS 3.8.1 Cond A, but the stem asks for maximum time which adds on 2 hours per SR 3.0.2.

B. Correct. The ARP verifies that ACB-241B & 341B have both opened automatically. 10 hours is correct because 8 hours is the time associated with TS 3.8.1 Cond A, but the stem asks for maximum time which adds on 2 hours per SR 3.0.2.

C. Incorrect. Plausible if the candidate thinks that closing PCB-83 to prepare for closing 241B & 341B to reenergize the busses. This is an actual interlock but not applicable in this condition, or directed by the ARP A8-31. 8 hours is plausible because it is the time associated with TS 3.8.1 Cond A, but the stem asks for maximum time which adds on 2 hours per SR 3.0.2.

D. Incorrect. Plausible if the candidate thinks that closing PCB-83 to prepare for closing 241B & 341B to reenergize the busses. This is an actual interlock but not applicable in this condition, or directed by the ARP A8-31. 10 hours is correct.

Sys #	System	Category	KA Statement
000056	Loss of Offsite Power / 6	Generic	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

K/A# 2.4.50 **K/A Importance** 4.0 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-36.4.AAV Rev 1 pg. 2
Tech Spec pg. 1.3-10 & 3.0-4

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.3)

Objective:

3SQS-36.1, Rev. 12	Obj. 14. Describe the control, protection and interlock functions for the control room components associated with the 4KV Distribution System, including automatic functions, setpoints and changes in equipment status as applicable.
3SQS-ELEC ITS, Rev. 1	Obj. 4. Given plant conditions that constitute non-compliance with any Electrical Power Systems LCO, or Licensing Requirement determine the applicable Condition(s), Required Action(s), and associated Completion Time(s).
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

81. The plant was operating at 100% power when a barge impacted the intake structure resulting in a loss of all River Water.
    - The crew has implemented AOP 1.30.2, "River Water Main Intake Structure Loss".
    - Attempts to restore River Water have failed.
    - The reactor has been manually tripped and E-0 implemented in parallel with AOP 1.30.2

    Per **AOP 1.30.2**, which of the following actions will be taken?

    1) Secure all Charging/ HHSI pumps
    2) Align the Fire Protection header to the River Water system
    3) Secure all RCPs
    4) Secure all CCR pumps
    5) Isolate Letdown

A. 1, 2, 3, ONLY

B. 2, 3, 4 ONLY

C. 1, 3, 5, ONLY

D. 1, 4, 5 ONLY

Beaver Valley Unit 1 NRC Written Exam (1LOT18)
(SRO ONLY)

QUESTION 81

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E page 8 first bullet. Specifically, the SRO must make an assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed. In this case the SRO must determine which actions are applicable to the loss of River Water procedure based on detailed knowledge of the AOP determine that the correct equipment actions are taken.

K/A is met by understanding of the action steps of the response not obtained steps for a loss of both the River Water and Aux River water pumps. This requires a detailed knowledge of the procedure and comprehension of the actions contained within.

A. Incorrect. Plausible Securing the RCPs and Charging pumps are correct actions, cross connect ing the Fire Protection system is not an action in the procedure. The EOPs will direct cross connect Fire Protection to RW per EOP Attachment 2-H, or procedure 1OM-33.4.L directs the cross tie when normal and alternate RW are not available. The Fire Pumps are impacted due to the loss of the intake structure.

B. Incorrect. Plausible since the cross tie to Fire Protection is not available, refer to A distractor explanation. Securing the RCPs is a correct action. Securing the CCR pumps is plausible since the RW system is the cooling medium for the CCR system, but there is no directions to secure the CCR pumps in the RW AOP. CCR Pumps are secured in AOP 1.15.1 if CCR Surge tank level is lost.

C. Correct. Per AOP 1.30.2, the Charging pumps are secured due to the loss of cooling water (RW), the RCPs are secured since RW cools the CCR heat exchangers. Letdown is isolated since changing flow is isolated and CCR cooling to the regenerative heat exchanger is lost due to the loss of RW.

D. Incorrect. Plausible, Securing the Charging pumps and isolating Letdown are correct actions, securing the CCR pumps is not an action per AOP 1.30.2, refer to the B distractor explanation.

Sys #	System	Category	KA Statement
000062	Loss of Nuclear Service Water / 4	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:** 1OM-53C.4.1.30.2 Rev. 9 Step 2 RNO

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.2 / 45.6)

Objective: 1SQS-53C.1-01-07 Obj 7 Given a set of conditions, apply the correct AOP.

1SQS-30.2-01-41 Obj 41 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

82. The Plant is stable at 55% power with all systems in normal alignment for this condition.
    - Tavg is 565°F and stable
    - Control Bank D is at 190 steps.
    - Control Rod Group Selector Switch is in the "AUTO" position.

    Plant Parameters are **NOW** as follows:
    - A4-76, Computer Alarm Rod Deviation/SEQ NIS Power Range Tilts
    - A4-69, NIS Power Range Neutron Flux Rate High is in alarm.
    - Tavg is 562 °F and slowly dropping.
    - RCS Pressure is 2230 psig and slowly dropping.
    - Reactor power has dropped to 51% and is slowly rising.
    - Control Bank D Demand step counters remain at 190 steps.
    - CERPI indication for Rod D4 is 12 steps.

    Based on these conditions:

    What procedure entry is **REQUIRED** and what operator action will be **REQUIRED** by this procedure?

A. AOP 1.1.7, Rod Position Indication Malfunction, and reduce thermal power to $\leq$ 50% within 8 hours.

B. AOP 1.1.7, Rod Position Indication Malfunction, and place Control Rod Group Selector Switch to "MAN".

C. AOP 1.1.8, Rod Inoperability, and reduce thermal power to $\leq$ 50% within 8 hours.

D. AOP 1.1.8, Rod Inoperability, and place Control Rod Group Selector Switch to "MAN".

82. The Plant is stable at 55% power with all systems in normal alignment for this condition.
    - Tavg is 565°F and stable
    - Control Bank D is at 190 steps.
    - Control Rod Group Selector Switch is in the "AUTO" position.

    Plant Parameters are **NOW** as follows:
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    - A4-69, NIS Power Range Neutron Flux Rate High is in alarm.
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    What procedure entry is **REQUIRED** and what operator action will be **REQUIRED** by this procedure?

A. AOP 1.1.7, Rod Position Indication Malfunction, and reduce thermal power to $\leq$ 50% within 8 hours.

B. AOP 1.1.7, Rod Position Indication Malfunction, and place Control Rod Group Selector Switch to "MAN".

C. AOP 1.1.8, Rod Inoperability, and reduce thermal power to $\leq$ 50% within 8 hours.

D. AOP 1.1.8, Rod Inoperability, and place Control Rod Group Selector Switch to "MAN".

˜uestion 82

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .E page 8 first bullet. Specifically, the SRO must make an assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed. In this case the SRO must diagnose that a control rod dropped and based on detailed knowledge of the AOP determine that the Control Rod Group Selector Switch must be placed in manual.

K/A is met by validating NIS Power Range Neutron Flux Rate High and Computer Alarm Rod Deviation alarm are expected alarms for the given plant conditions indicating that Control rod D4 has dropped.

A. Incorrect. Rod position deviation is listed as symptom/entry condition for AOP 1.1.7, however this it be accompanied by no corresponding change in power or Tavg. The stem has power and Tavg changing. Additionally, reducing power to below 50% is only an option if the demand counters are inoperable.

B. Incorrect. Rod position deviation is listed as symptom/entry condition for AOP 1.1.7, however this it be accompanied by no corresponding change in power or Tavg. The stem has power and Tavg changing. Additionally, placing rods in manual is not a required action of AOP 2.1.7.

C. Incorrect. Correct procedure entry, wrong required action.

D. Correct. IAW AOP 1.1.8 symptoms the alarms and plant response are consistent with a dropped rod. AOP 1.1.8 addresses a dropped rod in Part A. and directs rod to be placed in manual. First part of the question can be answered with RO knowledge since it involves AOP entry conditions. 2nd part of the question requires SRO knowledge of procedure content. SRO candidate must evaluate the given conditions, and select what actions will be necessary to address the given situation.

Sys #	System	Category	KA Statement
000003	Dropped Control Rod	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 None **Technical References:** 1OM-53C.4.1.1.8 Rev 5 pg. 1 & 2

Question Source: Bank – 2LOT8 NRC Exam Q82

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.3 / 45.12)

›bjective: 1SQS-53C.1, Rev. 12 Obj. 2. State the conditions or symptoms that would require entry into the AOPs.

Obj. 5. Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable

83. Given the following plant conditions:
    - The plant is at 12% power
    - PRZR Level Instrument, LT-461, has been isolated due to a leak on the transmitter, and all required actions have been completed.
    - PRZR Level Controller Selector switch is selected to 459/460
    - FCV-1CH-122, Charging Flow to Regen HX Inlet Control Valve is in AUTO

    The ATC reports the following indications:
    - A4-4, PRESSURIZER CONTROL LOW LEVEL DEVIATION is in ALARM
    - A4-3, PRESSURIZER CONTROL LEVEL LOW is in ALARM
    - PRZR level is lowering on PRZR Level Instrument LT-459
    - PRZR level on LT-460 is slowly RISING

    1) Which of the following required actions would the Unit Supervisor direct?
    2) What is the required action IAW with Technical Specifications?

A. 1) Take manual control and CLOSE FCV-1CH-122.
   2) Reduce reactor power to < 10%.

B. 1) Take manual control and CLOSE FCV-1CH-122.
   2) Place the failed PRZR level channel bistable to TRIP.

C. 1) Take manual control of FCV-1CH-122 and RAISE Charging flow.
   2) Reduce reactor power to < 10%.

D. 1) Take manual control of FCV-1CH-122 and RAISE Charging flow.
   2) Place the failed PRZR level channel bistable to TRIP.

Question 83

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B page 3 first and second bullets. Specifically, the SRO must make an assessment of TS 3.3.1 when 2 PRZR level instruments are inoperable, and applying the modes of operability, determine that the correct action for this condition is to enter LCO 3.0.3 because no applicable condition exists in TS 3.3.1 for both of these failures, and reduce power to less than P-7 which is where the condition not longer applies.

K/A is met by giving plant conditions which will cause przr level to raise due to an instrument failure which will cause letdown to isolate and FCV122 to raise charging flow due to level error in the control scheme. The candidates will have to recognize the przr level issue and respond in accordance to plant procedures.

A. Correct. The candidate must recognize that PRZR Control Level Low setpoint is 14% which is the setpoint for Letdown isolating and all heaters deenergizing under our przr level control scheme. When entering AOP-1.4.1, Process Control Failure, there is a note stating if letdown has isolated, go to AOP-1.7.1, Loss of Charging and Letdown. IAW the AOP, FCV122 must be closed due to letdown being isolated. Knowing this setpoint and the actions required by the ARP/IF and AOP 1.4.1 and 1.7.1, this is the correct response. Second part is correct because LT459 failing will make two przr level transmitters inoperable for which there are no Actions provided by TS 3.3.1, therefore LCO 3.0.3 is in effect. The SRO must know that they will remain in TS 3.0.3 until the power is <10% (P-7), and the Mode of applicability is no longer applicable.

B. Incorrect. First part is correct, see correct answer explanation. Second part is plausible if the candidate doesn't recognize that LT-461 has been isolated and placed in trip by the statements in the stem. When LT459 fails, and the crew places the bistable to trip, the 2/3 coincidence will cause a reactor trip since power is >P-7.

C. Incorrect. Plausible because LT459 is falling low, and ARP A4-3 step 4 states that if przr level drops to 5% then initiate a manual Reactor trip. The candidate may get confused by the LT failure and the information within the stem and think that FCV122 must be opened to restore level, but letdown will isolate and FCV122 will open on LT459 failing low. Second part is correct, see correct answer explanation.

D. Incorrect. Plausible because LT459 is falling low, and ARP A4-3 step 4 states that if przr level drops to 5% then initiate a manual Reactor trip. The candidate may get confused by the LT failure and the information within the stem and think that FCV122 must be opened to restore level, but letdown will isolate and FCV122 will open on LT459 failing low. Second part is plausible if the candidate doesn't recognize that LT-461 has been isolated and placed in trip by the statements in the stem. When LT459 fails, and the crew places the bistable to trip, the 2/3 coincidence will cause a reactor trip since power is >P-7.

Sys #	System	Category	KA Statement
00028	Pressurizer (PZR) Level Control Malfunction / 2	Generic	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

K/A# 2.4.50 **K/A Importance** 4.0 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-6.4.IF Rev 11 pg. 12
1OM-6.4.ABQ Rev 6
1OM-53C.4.1.4.1 Rev 1 pg.1
1OM-53C.4.1.7.1 Rev 12 pg.11
Tech Spec 3.3.1 pg 1, 4 &14 and pg. 3.0-1

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.3)

Objective:

1SQS-6.4, Rev. 14	Obj. 17. Describe the control, protection and interlock functions for the control room components associated with the Pressurizer and Pressurizer Relief System, including automatic functions, setpoints and changes in equipment status as applicable.
3SQS-RULES ITS, Rev. 3	Obj. 1. Given plant conditions, apply the rules of ITS Section 3.0 to ensure compliance with Technical Specifications.
3SQS-INST ITS, Rev. 1	Obj. 4. Given plant conditions that constitute non-compliance with any Instrumentation LCO, or Licensing Requirement determine the applicable Condition(s), Required Action(s), and associated Completion Time(s).

84. Given the following conditions:
    - The plant was operating at 100% power
    - A serious fire in the Cable Spreading Room has occurred.
    - The crew has entered 1OM-56C, Alternate Safe Shutdown From Outside Control Room.
    - The Backup Indicating Panel (BIP) is in service IAW 1OM-56C.4.F-1, BIP Activation".
    - The crew is currently cooling down the plant at <50°F/hr

1) Review Data Sheet 1 (provided), and determine if the crew stayed within the acceptable RCS pressure limits of Figure 56C-1.

   The crew _________ remained within the acceptable RCS pressure limit of Figure 56C-1.

2) If the crew must lower RCS pressure during the plant cooldown, 1OM-56C directs the Shift Manager to use the ____________ from the Backup Indicating Panel.

A. 1) has
   2) PORV

B. 1) has NOT
   2) PORV

C. 1) has
   2) Pressurizer Vent

D. 1) has NOT
   2) Pressurizer Vent

Question 84

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E page 6. Specifically, the SRO must have specific knowledge of the content of the procedure that is being used. In this case the SRO must have specific knowledge of the Alternate Safe Shutdown from Outside Control Room procedures and know that the PORVS are isolated during the implementation of the procedure, therefore depressurization is completed using the Przr Vent.

K/A is met by evaluating the RCS pressure and temperatures taken at the Backup Indicating Panel after the control room has been evacuated, and determine if the plant has remained within RCS pressure limit of Figure 56C-1.

A. Incorrect. First part is incorrect because pressure limit was violated. Second part is plausible as a means to reduce pressure, but the 56C procedure isolates the PORVS and uses the Pressurizer Vent from the BIP.

B. Incorrect. The pressure/temperature limit was violated at the 1850 psig reading. Second part is plausible as a means to reduce pressure, but the 56C procedure isolates the PORVS and uses the Pressurizer Vent from the BIP.

C. Incorrect. First part is incorrect because pressure limit was violated. Second part is correct.

D. Correct. The pressure/temperature limit was violated at the 1850 psig reading. The SRO must know the Intent and Methodology of Alternate Safe Shutdown from Outside Control Room and understand that the PORVs are isolated during 56C, and pressure control is maintained with the pressurizer vent and pressurizer heaters (1OM-56C.4.A step 18).

Sys #	System	Category	KA Statement
000068	Control Room Evacuation / 8	AA2. Ability to determine and interpret the following as they apply to the Control Room Evacuation:	RCS pressure

K/A# AA2.06 **K/A Importance** 4.3 **Exam Level** SRO

References provided to Candidate 1OM-56C.5.A.1 Rev 5
1OM-56C.4.F-1 Rev. 22 pg 13 filled out with plant parameters.

Technical References: 1OM-56C.5.A.1 Rev 5
1OM-56C.4.F-1 Rev. 22 pg 13
1OM-56C.4.B Rev 46 pg. 13
1OM-56C.4.A Rev 11 pg 5

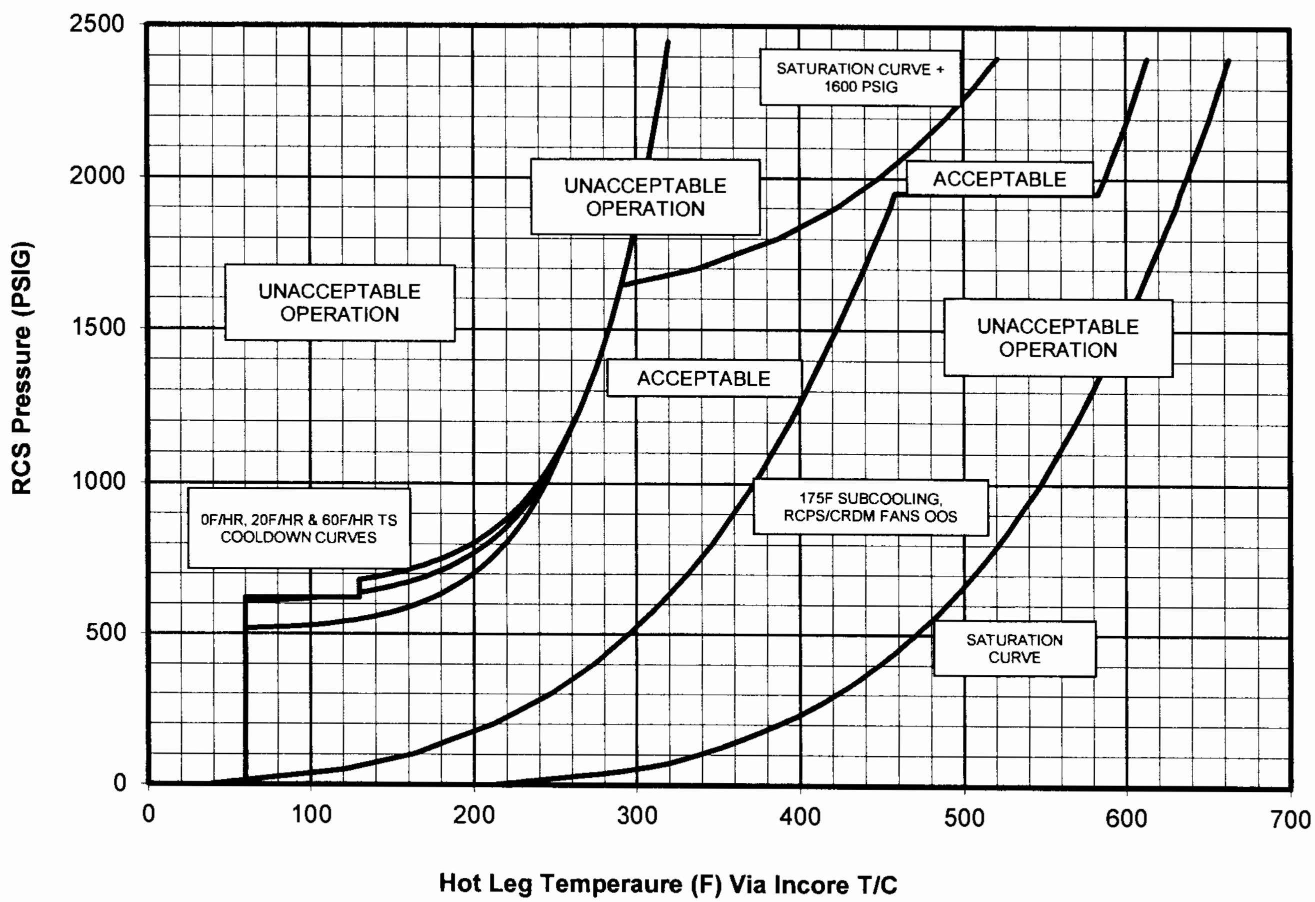
Question Source: New

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

Objective: 1SQS-56C.1, Rev. 10 4. Given a set of plant conditions and the appropriate procedure(s), apply the operational sequence, parameter limits, precautions & limitations, cautions and notes applicable to the completion of the task activities in the field.

Figure No. 56C-1 - RCS Pressure Limits

Figure 56C-1



DATA SHEET 1

	INSTRUMENT	READING	READING	READING	READING
1.	RCS Pressure PI-RC403	2100 PSIG	2000 PSIG	1850 PSIG	1750 PSIG
2.	PZR Level (actual) LI-RC460	35 %	31 %	27 %	23 %
3.	RCS C Cold Leg Temp TI-RC430 Note 1	552 °F	510 °F	465 °F	421 °F
4.	RCS A Cold Leg Temp TI-RC410 Note 1	551 °F	508 °F	465 °F	422 °F
5.	RCS B Cold Leg Temp TI-RC420 Note 1	552 °F	509 °F	464 °F	423 °F
6.	Thermocouple TI-RC 029	575 °F	530 °F	485 °F	440 °F
7.	Thermocouple TI-RC 038	575 °F	530 °F	485 °F	440 °F
8.	Thermocouple TI-RC 040	575 °F	530 °F	485 °F	440 °F
9.	Thermocouple TI-RC 043	575 °F	530 °F	485 °F	440 °F
10.	Steam Pressure A Loop [PI-MS-501]	1047 PSIG	715 PSIG	471 PSIG	305 PSIG
11.	Steam Pressure B Loop [PI-MS-502]	1047 PSIG	715 PSIG	471 PSIG	305 PSIG
12.	Steam Pressure B Loop [PI-MS-503]	1047 PSIG	715 PSIG	471 PSIG	305 PSIG
13.	SG Level A Loop LI-FW475	60 %	54 %	56 %	52 %
14.	SG Level B Loop LI-FW485	60 %	54 %	56 %	52 %
15.	SG Level C Loop LI-FW495	60 %	54 %	56 %	52 %
16.	Source Range Indication	1000 CPS	900 CPS	800 CPS	675 CPS
	Time	Time Ø	Time 1hr	Time 2hr	Time 3hr
	Date	Today	Today	Today	Today
	Initials	[illegible]	[illegible]	[illegible]	[illegible]

Note 1 If RCS cold leg temperature indication is not available, use saturated temperature for SG pressure recorded above in accordance with Table 56C-2, Saturated Steam Table.

8

85. Initial conditions:

- The plant had a complete loss of switchyard
- The crew was performing steps in ES-0.2, Natural Circulation Cooldown
- Station management recommended a rapid cooldown due to secondary inventory concerns
- The crew transitioned to ES-0.3, Natural Circulation Cooldown with Steam Void in the Vessel

Current conditions:

- Pressurizer level is 92% and RISING
- RVLIS Full Range level is 70% and decreasing
- The STA notes a YELLOW path on INVENTORY and confers with the SM regarding the need to transition to FR-I.3, Response to Voids in Reactor Vessel

1) The SRO is required to direct the crew to perform which of the following actions in order to control void growth without interrupting natural circulation?

2) Which procedure will direct the crew to perform this action?

A. 1) Open reactor vessel head vents
   2) FR-I.3, Response to Voids in Reactor Vessel

B. 1) Open reactor vessel head vents
   2) ES-0.3, Natural Circulation Cooldown with Steam Void in the Vessel

C. 1) Energize pressurizer heaters
   2) FR-I.3, Response to Voids in Reactor Vessel

D. 1) Energize pressurizer heaters
   2) ES-0.3, Natural Circulation Cooldown with Steam Void in the Vessel

Question 85

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E second bullet, page 7. Specifically, the SRO must have specific knowledge of the content of the procedure that is being used. This question requires analysis of the accident conditions, determination of what action will control vessel head void growth, without interrupting natural circulation, and then evaluating which procedure to use to accomplish the action. The procedure selected is NOT a major EOP but is a specific sub-procedure (ES-0.3). The question does require the applicant to assess plant conditions and then select a sub-procedure with which to proceed.

K/A Match: The candidate is presented with plant conditions involving a loss of the switchyard. Then, based on the resulting operating characteristics, including the instrument indications for reactor vessel water level and pressurizer level must evaluate whether a bubble is forming in the vessel head. Based on that assessment, the SRO applicant then determines the appropriate actions using ES-0.3, Natural Circulation with Steam Void in Vessel with/without RVLIS.

A. Incorrect. Plausible, since this is an action in FR-I.3 to vent voids in the reactor vessel, but the Caution prior to step 1 states "If a controlled natural circulation cooldown is in progress and a void in the reactor vessel upper head is expected, this procedure should not be performed". FR-I.3 is plausible because it is used as a response to voids in the reactor vessel under, but not under these circumstances.

B. Incorrect. Plausible, since this is an action in FR-I.3 to vent voids in the reactor vessel, but the Caution prior to step 1 states "If a controlled natural circulation cooldown is in progress and a void in the reactor vessel upper head is expected, this procedure should not be performed". ES-0.3 is the correct procedure which directs the use of pressurizer heaters to control pressure.

C. Incorrect. Plausible because pressurizer heaters are the method used to control void growth in ES-0.3, but not in FR-I.3. FR-I.3 vents the reactor vessel using the head vents, but ES-0.3 is not applicable with the current conditions.

D. Correct. ES-0.3 is implemented if the 25°F/hr cooldown of ES-0.2 limit is not fast enough. ES-0.3 is designed to perform a plant cooldown on natural circulation, assuming that a void will develop in the reactor vessel head region. The operator monitors the void growth and the procedure requires that level in the vessel head be maintained greater than 71% Full range level. The void level is controlled by the use of pressurizer heaters (to control subcooling) and charging and letdown.

Sys #	**System**	**Category**	**KA Statement**
WE10	Natural Circulation/4	EA2. Ability to determine and interpret the following as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS)	Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments.

K/A# EA2.2 **K/A Importance** 3.9 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.ES-0.3 Iss 3 Rev 0 step 5
1OM-53B.4.ES-0.3 Iss 3 Rev 0 pg. 17
1OM-53A.1.FR-I.3 Iss 3 Rev 0 pg. 2

Question Source: Bank – Catawba 2014 NRC Exam Q 85

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

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

Obj. 6. Given a set of conditions, locate and apply the proper Emergency Operating Procedures, IAW BVPS-EOP Executive Volume.

86. Given the following conditions:

- Rx at 7% performing startup
- AOP 1.6.8, Abnormal RCP Operation, has been entered for 'B' RCP high vibrations
- The vibration immediate trip setpoint has been exceeded
- The Rx failed to trip from BB 'B' and BB 'A'
- ATC is inserting the Control Rods in manual

1) What is the required Emergency classification?
2) How long do you have to notify the NRC of the plant event?

(References provided)

A. 1) Unusual Event
2) 15 minutes

B. 1) Alert
2) 15 minutes

C. 1) Unusual Event
2) 1 hour

D. 1) Alert
2) 1 hour

Beaver Valley Unit 1 NRC Written Exam (1LOT18)
(SRO ONLY)

uestion 86

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E 4th bullet on page 8. Specifically, the SRO must have knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. In this case the SRO must demonstrate the ability to make an EPP declaration and determine the NRC notification time limit.

K/A is met by requiring the candidate to recognize that an Alert must be declared when a manual reactor trip is required based on RCP high vibrations, and the reactor fails to trip from the control room, and notify the NRC within the 1hr time requirement identified in NOP-OP-1015, Event Notifications.

A. Incorrect. UE is plausible is the candidate reads SU6 and doesn't understand that a subsequent automatic trip or manual trip action taken at the Control Room Benchboards (reactor trip breaker switch or pushbutton or tripping the turbine) is successful in shutting down the reactor (Note 8). The misunderstanding would occur if Note 8 is not reviewed and they feel that inserting the control rods until rx power is <5% with a negative SUR would constitute a Rx trip. Note 8 specifically states that it does not include manually driving in control rods or implementation of boron injection strategies. Plausible distractor because 15 minutes is identified for notification of Declaration of an Emergency Classification to state and local governments.

B. Incorrect. Correct that it is classified as an Alert based on SA6. Plausible distractor because 15 minutes is identified for notification of Declaration of an Emergency Classification to state and local governments.

C. Incorrect. UE is plausible is the candidate reads SU6 and doesn't understand that a subsequent automatic trip or manual trip action taken at the Control Room Benchboards (reactor trip breaker switch or pushbutton or tripping the turbine) is successful in shutting down the reactor (Note 8). The misunderstanding would occur if Note 8 is not reviewed and they feel that inserting the control rods until rx power is <5% with a negative SUR would constitute a Rx trip. Note 8 specifically states that it does not include manually driving in control rods or implementation of boron injection strategies. 1 hour is correct.

D. Correct. Classified as an Alert based on SA6, Auto or manual trip fails to shut down the reactor AND Manual trip actions taken at the Control Room Benchboards (reactor trip breaker switch or pushbutton or tripping the turbine) are not successful in shutting down the reactor (Note 8). IAW NOP-OP-1015 att. 2 pg. 1 the NRC must be notified immediately via ENS to NRC after the State and local notifications but not more than 1 hour after declaration.

Sys #	System	Category	KA Statement
003	Reactor Coolant Pump	Generic	Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

K/A# 2.4.30 **K/A Importance** 4.1 **Exam Level** SRO

References provided to Candidate NOP-OP-1015 rev 6
U1 EAL Wallboard

Technical References: EPP-PLAN-SECTION-4 Rev 32 pg. 4-170 & 171
U1 EAL Wallboard (EPP-I-1a.F01 Rev 4)
NOP-OP-1015 rev 6 pg. 17

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.11)

Objective: EPP-9281, Rev. 13 Obj. 11. Given specific plant conditions, classify the condition in accordance with EPP I-1a & b.

Obj. 15. Describe the Emergency Director's responsibilities relative to notification of the offsite agencies.

S

- Automatic turbine runback ≥ 25% thermal power
- Electrical load rejection > 25% full electrical load
- Safety Injection actuation

6

None

SS6.1
An automatic or manual trip fails to shut down the reactor
AND
All actions to shut down the reactor are not successful
AND EITHER:
- Core Cooling-RED Path conditions met
- Heat Sink-RED Path conditions met

SA6.1
An automatic or manual trip fails to shut down the reactor
AND
Manual trip actions taken at the Control Room Benchboards (reactor trip breaker switch or pushbutton or tripping the turbine) are not successful in shutting down the reactor (Note 8)

SU6.1
An automatic trip did not shut down the reactor after ANY RPS setpoint is exceeded
AND
A subsequent automatic trip or manual trip action taken at the Control Room Benchboards (reactor trip breaker switch or pushbutton or tripping the turbine) is successful in shutting down the reactor (Note 8)

SU6.2
A manual trip did not shut down the reactor after ANY manual trip action was initiated
AND
A subsequent automatic trip or manual trip action taken at the Control Room Benchboards (reactor trip breaker switch or pushbutton or tripping the turbine) is successful in shutting down the reactor (Note 8)

6

7

Table 1S-4
Communications Methods

System	Onsite	ORO	NRC
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

None

None

1 2 3 4

SU7.1
Loss of ALL Table 1S-4 onsite communication methods

SU7.2
Loss of ALL Table 1S-4 offsite response organization (ORO) communication methods

SU7.3
Loss of ALL Table 1S-4 NRC communication methods

7

8

None

None

1 2 3 4

SU8.1
ANY penetration is not isolated within 15 min. of a VALID containment isolation signal
OR
Containment pressure > 11 psig AND < one full train of depressurization equipment operating per design for ≥ 15 min.
(Note 1)

8

Table 1S-5
Hazardous Events

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard

1 2 3 4

SA9.1
The occurrence of ANY Table 1S-5 hazardous event

BEAVER VALLEY UNIT 1

46.00 x 46.00 in

one

EPP-I-1a.F01
Rev 4

00

87. Given the following conditions:

- A PORV opened due to an instrument failure
- The PORV has been isolated
- PI-1RC-472, PRT Pressure is 75 psig and stable
- LI-1RC-470, PRT Level is 70% and stable
- TI-1RC-471, PRT Temperature is 105°F and stable

Based on the above conditions:

1) The crew is required to perform which of the following actions?
2) What procedure will be used to conduct this evolution?

A. 1) Drain the PRT to the CNMT Sump to prevent a Hydrogen explosion in the Waste Disposal System vent header.
2) ARP A4-36, Pressurizer Relief Tank Level High-Low

B. 1) Spray down the PRT to reduce pressure prior to venting to the Waste Disposal System to prevent overpressurization of the vent header.
2) ARP A4-37, Pressurizer Relief Tank Press High

C. 1) Vent the PRT to the Waste Disposal System slowly to prevent a Hydrogen explosion in the vent header.
2) ARP A4-37, Pressurizer Relief Tank Press High

D. 1) Drain the PRT to 1DG-TK-1, Primary Drains Transfer Tank to prevent overpressurization of the Waste Disposal System vent header and prevent a Hydrogen explosion.
2) ARP A4-36, Pressurizer Relief Tank Level High-Low

Question 87

Answer: B

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 make an assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed. In the question, the SRO must have knowledge that the vent path (waste gas vent header) is connected to the primary drains transfer tank which has a design pressure rating of 50 psig. Since the PRT is at 75 psig, it must be sprayed down to to at least 60 psig based on an OM-6 precaution and limitation. Therefore, this is detailed procedural and system knowledge.

K/A is met by demonstrating the ability to spray down the PRT iaw the ARP to reduce pressure low enough to prevent overpressurization of the waste gas vent header which is connected to the Primary Drains tank which has a design pressure of 50 psig.

A. Incorrect. Venting the PRT to TK-1 appears plausible since TK-1 in inside cnmt, but it is actually vented to the vent header to TK-2 outside cnmt.

B. Correct. Spraying down the PRT will reduce pressure, then the PRT is vented to the Waste Disposal System vent header. This is to prevent overpressurization of the vent header which is connected to Primary Drains Transfer Tank TK-2 which has a design pressure of 50 psig.

C. Incorrect. Venting to the Waste Disposal System is plausible, but the ARP states to spray down first, and the PRZR P&L #64 states that the PRT must not be vented to the Waste Disposal System vent header if the pressure exceeds 60 psig, and the stem states pressure is 75 psig.

D. Incorrect. Draining is plausible if the candidate thinks the open PORV raised level in the PRT causing pressure to rise to 75 psig, but PRT level annunciator was not lit in the stem and is not the correct action iaw the ARP.

Sys #	System	Category	KA Statement
007	Pressurizer Relief/Quench Tank	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Overpressurization of the waste gas vent header

K/A# A2.04 **K/A Importance** 2.9 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-6.4.AAC rev 6 pg. 2
1OM-6.2.A Rev 20 pg. 8

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Objective: 1SQS-9.1, Rev. 11 Obj. 5. Summarize the operational aspects of the Non-Aerated Liquid Drain System headers located inside and outside of containment.

1SQS-6.4, Rev. 14 Obj. 22. Given a set of plant conditions and the appropriate procedure(s), apply the operational sequences, parameter limits, precautions and limitations, and cautions & notes applicable to the completion of the task activities in the control room. a. 1OM-6.2.A, Precautions and Limitations

88. Which of the following components are assumed to operate at their setpoints to ensure that RCS pressure remains below the Technical Specification RCS Pressure Safety Limit?

1. Pressurizer Safety Valves
2. Pressurizer PORVs
3. Main Steam Safety Valves

A. 1 only

B. 1 and 2 only

C. 1 and 3 only

D. 1, 2, and 3

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II B page 5 third bullet. SRO is required to have specific Knowledge of TS bases that are required to analyze TS required actions and terminology. Specifically, the SRO must know that the Pressurizer Safety Valves and the Main Steam Safety Valves have settings established to ensure that the RCS pressure Safety Limit will not be exceeded.

K/A is met by testing the knowledge of conditions and limitations of the RCS pressure safety limit, and that the Pressurizer Safety Valves and Main Steam Safety Valves have setpoints which ensure the SL is not exceeded.

A. Incorrect. Plausible distractor, but the MSSVs are also depended on to maintain RCS pressure less than the safety limit.

B. Incorrect. Plausible distractor because both the przr safety valves and the PORVs relieve RCS pressure, but the PORVs are not depended on to maintain pressure less than the SL. The PORVS are used to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available.

C. Correct. TS 2.1.2 bases states that the RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL of 2735 psig will not be exceeded.

D. Incorrect. Plausible distractor, but the PORVs are not credited for maintaining the RCS pressure less than the SL.

Sys #	**System**	**Category**	**KA Statement**
010	Pressurizer Pressure Control	Generic	Knowledge of conditions and limitations in the facility license.

K/A# 2.2.38	**K/A Importance** 4.5	**Exam Level**	SRO
References provided to Candidate	None	**Technical References:**	Tech Spec bases 2.1.2 pgs. 1 & 2 Tech Spec bases pg. 3.4.10-1 Tech Spec bases pg. 3.7.1-1

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7 / 41.10 / 43.1 / 45.13)

Objective: 3SQS-SL ITS Rev 0 Obj. 2 - Given plant conditions, determine the criteria necessary to ensure compliance with each ITS Safety Limit.

89. A large break LOCA has occurred from 100% power.

- The crew is currently performing ECA-1.1, Loss of Emergency Coolant Recirculation due to not being able to verify cold leg recirculation capability in E-1, Loss of Reactor or Secondary Coolant.
- Cnmt pressure peaked at 37 psig, and is currently 6 psig and lowering

1) IAW ECA-1.1, at what containment pressure must the quench spray pumps be secured?
2) What is the bases for securing the Quench Spray pumps at this pressure?

A. 1) < -4 psig
2) Minimize the demand on the RWST.

B. 1) < -4 psig
2) Provide sufficient NPSH for the pumps taking suction from the containment sump.

C. 1) < 11 psig
2) Minimize the demand on the RWST.

D. 1) < 11 psig
2) Provide sufficient NPSH for the pumps taking suction from the containment sump.

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E page 6. Specifically, the SRO must have specific knowledge of the content of the procedure that is being used. In this case the SRO must have specific knowledge of when to secure the quench spray pumps based on cnmt pressure iaw ECA-1.1 in order to preserve the RWST level.

K/A is met by the candidate having the knowledge of securing the Containment Quench Spray pumps at 11 psig in ECA-1.1, Loss of Emergency Coolant Recirculation in order to preserve the RWST level.

A. Incorrect. -4 psig is plausible because it is the pressure the QS pumps would be secured in E-1 step 10 if there were no faulted SGs. The second part is correct.

B. Incorrect. -4 psig is plausible because it is the pressure the QS pumps would be secured in E-1 step 10 if there were no faulted SGs. The bases for securing the QS pump is plausible because QS does provide flow to the cnmt sumps to improve the NPSH available to the RSS pumps, but this is not the reason for securing the QS pumps in ECA-1.1.

C. Correct. ECA-1.1 step 7 states if cnmt pressure is < 11 psig, secure the Quench Spray pumps in order to minimize RWST depletion.

D. Incorrect. 11 psig is correct. The bases for securing the QS pump is plausible because QS does provide flow to the cnmt sumps to improve the NPSH available to the RSS pumps, but this is not the reason for securing the QS pumps in ECA-1.1.

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:	Safe securing of containment spray when it can be done)

K/A# A2.08 **K/A Importance** 3.7 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-1.1 Iss 3 Rev 0 pg. 3
1OM-53B.4.ECA-1.1 Iss 3 Rev 0 pg. 15

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

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.

Obj. 6. Given a set of conditions, locate and apply the proper Emergency Operating Procedures, IAW BVPS-EOP Executive Volume.

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

- Unit 1 is at Full Power with all systems in NSA.
- RM-1VS-107B, "Reactor Building and SLCRS Vent Release Particulate Gas Monitor" power supply fails.
- The US declares this monitor non-functional and consults 1/2-ODC-3.03, "ODCM: Controls for RETS and REMP Programs" to determine the impact of this failure, if any.
- Assume all other Radiation Monitors are functional.

Which of the following actions according to 1/2-ODC-3.03 will be taken based on the failed power supply, if any? **(Reference Provided)**

A. Restore non-functional channel within 72 hours.

B. No action is necessary, elevated discharge can continue.

C. Immediately suspend the release of radioactive gaseous effluents.

D. Take grab samples at least every 12 hours and analyze for gross activity.

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B page 3. The SRO must demonstrate the ability to use the ODCM to determine if an elevated release may continue when RM-1VS-107B, Reactor Building and SLCRS Vent Release Particulate Gas Monitor detector fails.

K/A is met by giving the candidate a failed process rad monitor, and applying the requirements of the ODCM to determine that the elevated release may continue because it is not the primary rad monitor.

A. Incorrect. This is a plausible alternative if the candidate refers to Table 3.3-6 and incorrectly applies action 35.

B. Correct. RM-1VS-107B is referenced several places in the ODCM. Table 3.3-6 1a makes reference to this monitor as a 2nd PMM, so there is no impact on the elevated release because the Primary Instrument (RM-1VS-110) is still operable. Table 4.3-3 shows the required surveillance for this instrument. Table 3.3-13 3a requires RM-1VS-107B or RM-1VS-110 as the alternate. Since 1 is the minimum channels operable, Action 29 does not need to be performed. Action 30a is N/A since a purge from containment is not in progress. Note that the candidate must understand that in NSA an elevated release from Safeguards is in progress.

C. Incorrect. Plausible if the candidate misunderstands or incorrectly applies the ODCM, but not correct.

D. Incorrect. This is a plausible alternative if there were one less than the minimum required channels. Only one channel is required to be operable per Table 3.3-13, 3a.

Sys #	System	Category	KA Statement
073	Process Radiation Monitoring	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Detector failure

K/A# A2.02 **K/A Importance** 3.2 **Exam Level** SRO

References provided to Candidate 1/2-ODC-3.03, Rev. 14 **Technical References:** 1/2-ODC-3.03, Rev. 14, pg. 20, 24, 39, 46; 1SQS-43.1, Rev. 16, PPNT Slide # 4

Question Source: Bank – 1LOT8 NRC Exam Q90

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

Objective: 1SQS-43.1-01-13: Analyze a given set of plant conditions for compliance with the licensing requirements, including the determination of equipment operability and applicable action statements.

ATTACHMENT D
Page 2 of 8
ODCM CONTROLS: RADIATION MONITORING INSTRUMENTATION

TABLE 3.3-6

BV-1 RADIATION MONITORING INSTRUMENTATION

Pri = Primary Instruments, PMM = Preplanned Method of Monitoring[(a)]

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT[(1) (b)]	NOMINAL MEASUREMENT RANGE	ACTION
1. Noble Gas Effluent Monitors - PINGS[(3)]					
a. Reactor Building/SLCRS (CV-1; Also called Elevated Release)					
High Range Noble Gas	(1)	1, 2, 3, & 4			35
Pri: (RM-1VS-110 HRNG Rel)			≤ 1.56E+7 uCi/s	1E-4 to 1E+5 uCi/cc[(2)]	
1st PMM: (RM-1VS-110 LRNG Rel)					
2nd PMM: (RM-1VS-107B)					
3rd PMM: Grab Sampling every 12 hours					
b. Auxiliary Building Ventilation System (VV-1; Also called Ventilation Vent)					
High Range Noble Gas	(1)	1, 2, 3, & 4			35
Pri: (RM-1VS-109 HRNG Rel)			≤ 1.18E+7 uCi/s	1E-4 to 1E+5 uCi/cc[(2)]	
1st PMM: (RM-1VS-109 LRNG Rel)					
2nd PMM: (RM-1VS-101B)					
3rd PMM: Grab Sampling every 12 hours					
c. Gaseous Waste/Process Vent System (PV-1/2)					
High Range Noble Gas	(1)	1, 2, 3, & 4			35
Pri: (RM-1GW-109 HRNG Rel)			≤ 7.84E+9 uCi/s	1E-4 to 1E+5 uCi/cc[(2)]	
1st PMM: (RM-1GW-109 LRNG Rel)					
2nd PMM: (RM-1GW-108B)					
3rd PMM: Grab Sampling every 12 hours					

(a) Instruments or actions shown as PMM are the preplanned methods to be used when the primary instrument is inoperable. SINCE the PMM instruments shown are not considered comparable alternate monitoring channels, THEN the ODCM Surveillance Requirements do not apply to the PMM. Therefore, the reporting requirement of Action 35b would still apply when inoperability of the primary instrument exceeds 30 days.

(b) Setpoints are calculated in calculation package ERS-MPD-93-007.

ATTACHMENT D
Page 6 of 8
ODCM CONTROLS: RADIATION MONITORING INSTRUMENTATION

TABLE 4.3-3 (Continued)

BV-1 RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Pri = Primary Instruments, PMM = Preplanned Method of Monitoring[(a)]

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Noble Gas Effluent Monitors - PINGS				
a. Reactor Building/SLCRS (CV-1; Also called Elevated Release)				
High Range Noble Gas Pri: (RM-1VS-110 HRNG Rel) 1st PMM: (RM-1VS-110 LRNG Rel) 2nd PMM: (RM-1VS-107B) 3rd PMM: Grab Sampling every 12 hours	S	R	M	1, 2, 3, & 4
b. Auxiliary Building Ventilation System (VV-1; Also called Ventilation Vent)				
High Range Noble Gas Pri: (RM-1VS-109 HRNG Rel) 1st PMM: (RM-1VS-109 LRNG Rel) 2nd PMM: (RM-1VS-101B) 3rd PMM: Grab Sampling every 12 hours	S	R	M	1, 2, 3, & 4
c. Gaseous Waste Process Vent System (PV-1,2)				
High Range Noble Gas Pri: RM-1GW-109 HRNG Rel) 1st PMM: (RM-1GW-109 LRNG Rel) 2nd PMM: (RM-1GW-108B) 3rd PMM: Grab Sampling every 12 hours	S	R	M	1, 2, 3, & 4

(a) Instruments or actions shown as PMM are the preplanned methods to be used when the primary instrument is inoperable. SINCE the PMM instruments shown are not considered comparable alternate monitoring channels, THEN the ODCM Surveillance Requirements do not apply to the PMM. Therefore, the reporting requirement of Action 35b would still apply when inoperability of the primary instrument exceeds 30 days.

ODCM CONTROLS: RETS INSTRUMENT FOR GASEOUS RELEASES

TABLE 3.3-13 (continued)

BV-1 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Pri = Primary Instruments Alt = Alternate Instruments [(a)]

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
2. Auxiliary BuildingVentilation System (VV-1; Also called Ventilation Vent)			
a. Noble Gas Activity Monitor Pri: [RM-1VS-109 LRNG Rel], or Alt: [RM-1VS-101B]	(1)	*	29,30A
b. Particulate and Iodine Sampler Pri: Filter Paper & Charcoal Cartridge for [RM-1VS-109], or 1st Alt: Filter Paper & Charcoal Cartridge for [RM-1VS-111] [(b)], or 2nd Alt: Continuous collection via RASP Pump	(1)	*	32
c. System Effluent Flow Rate Measuring Device Pri:, [FR-1VS-101] or Alt: [RM-1VS-109 LR STK FL]	(1)	*	28A
d. Sampler Flow Rate Measuring Device Used for Particulate and Iodine Sample Collection (see 2.b) Pri: [ABPM-1VS-109-PISFlow], or Alt: RASP Pump Flow Instrument	(1)	*	28B
3. Reactor Building/SLCRS (CV-1; Also called Elevated Release)			
a. Noble Gas Activity Monitor Pri: [RM-1VS-110 LRNG Rel], or Alt: [RM-1VS-107B]	(1)	*	29,30A
b. Particulate and Iodine Sampler Pri: Filter Paper & Charcoal Cartridge for [RM-1VS-110], or 1st Alt: Filter Paper & Charcoal Cartridge for [RM-1VS-112] [(b)], or 2nd Alt: Continuous collection via RASP Pump	(1)	*	32
c. System Effluent Flow Rate Measuring Device Pri:, [FR-1VS-112] [(b)] or Alt: [RM-1VS-110 LR STK FL]	(1)	*	28A
d. Sampler Flow Rate Measuring Device Used for Particulate and Iodine Sample Collection (see 3.b) Pri: [ABPM-1VS-110-PISFlow], or Alt: RASP Pump Flow Instrument	(1)	*	28B

* During Releases via this pathway.

(a) Condition Report generation and reporting in the Radioactive Effluent Release Report (per Control 3.3.3.10 Action b) do not apply when using an alternate to satisfy inoperability of the primary instrument beyond 30 days.

(b) May be used as a sampled flow path provided flow is verified. Radiation monitor operability is not required.

ATTACHMENT F
Page 10 of 13
ODCM CONTROLS: RETS INSTRUMENT FOR GASEOUS RELEASES

TABLE 4.3-13

BV-1 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Pri = Primary Instruments   Alt = Alternate Instruments

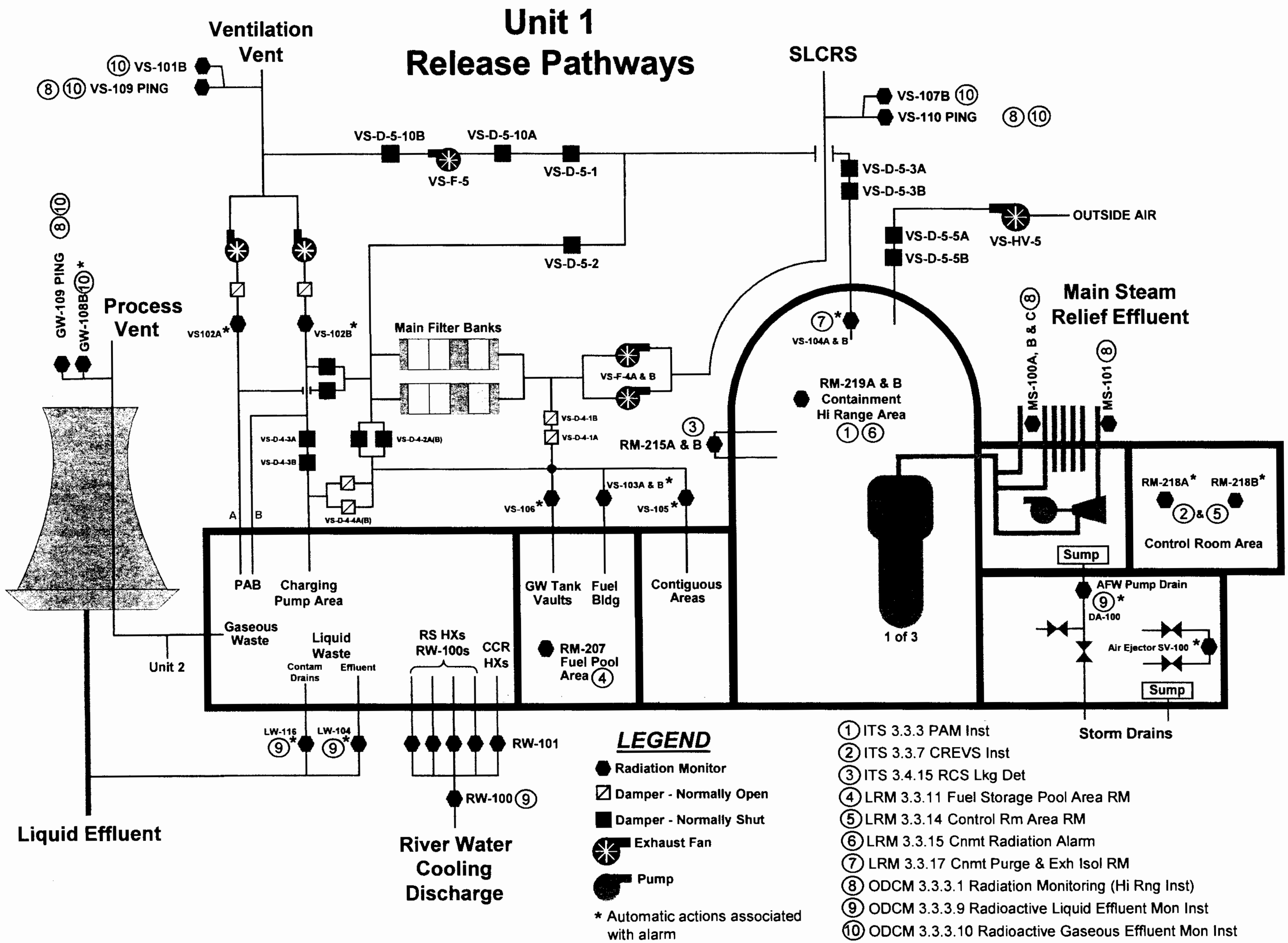
INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL OPERATIONAL TEST
3. Reactor Building/SLCRS (CV-1; Also called Elevated Release)				
a. Noble Gas Activity Monitor Pri: [RM-1VS-110 LRNG Rel], or Alt: [RM-1VS-107B]	D	M(4), P(4)***	R(3)	Q(2)
b. Particulate and Iodine Sampler Pri: Filter Paper & Charcoal Cartridge for [RM-1VS-110], or 1st Alt: Filter Paper & Charcoal Cartridge for [RM-1VS-112] (b), or 2nd Alt: Continuous collection via RASP Pump	W	NA	NA	NA
c. System Effluent Flow Rate Measuring Device Pri: [FR-1VS-112] (b), or Alt: [RM-1VS-110 LR STK FL]	D	NA	R	Q
d. Sampler Flow Rate Measuring Device Used for Particulate and Iodine Sample Collection (see 3.b) Pri: [ABPM-1VS-110-PISFlow], or Alt: RASP Pump Flow Instrument	D	NA	R	Q

* During releases via this pathway.

*** During purging of Reactor Containment via this pathway.

(b) May be used as a sampled flow path provided flow is verified. Radiation monitor operability is not required.

Unit 1 Release Pathways



15QS-43.1 PPNT

9

(SRO ONLY)

Beaver Valley Unit 1 NRC Written Exam (1LOT8)

90.

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

- Unit 1 is at Full Power with all systems in NSA.
- RM-1VS-107B, "Reactor Building and SLCRS Vent Release Particulate Gas Monitor" power supply fails.
- The US declares this monitor non-functional and consults ½-ODC-3.03, "ODCM: Controls for RETS and REMP Programs" to determine the impact of this failure, if any.
- Assume all other Radiation Monitors are functional.

Which ONE of the following actions according to ½-ODC-3.03 will be taken based on the failed power supply, if any? **(Reference Provided)**

A. Restore non-functional channel within 72 hours.

B. No action is necessary, elevated discharge can continue.

C. Immediately suspend the release of radioactive gaseous effluents.

D. Take grab samples at least every 12 hours and analyze for gross activity.

Answer: B

Explanation/Justification:

A. Incorrect. This is a plausible alternative if the candidate refers to Table 3.3-6 and incorrectly applies action 35.

B. Correct. RM-1VS-107B is referenced several places in the ODCM. Table 3.3-6 1a makes reference to this monitor as a 2nd PMM, so there is no impact on the elevated release because the Primary Instrument is still operable. Table 4.3-3 shows the required surveillance for this instrument. Table 3.3-13 3a requires RM-1VS-107B or RM-1VS-110 ch 5 as the alternate. Since 1 is the minimum channels operable, Action 29 does not need to be performed. Action 30a is N/A since a purge from containment is not in progress. Note that the candidate must understand that in NSA an elevated release from Safeguards is in progress.

C. Incorrect. Plausible if the candidate misunderstands or incorrectly applies the ODCM, but not correct.

D. Incorrect. This is a plausible alternative if there were one less than the minimum required channels. Only one channel is required to be operable per Table 3.3-13, 3a.

Sys #	System	Category	KA Statement
073	Process Radiation Monitoring	Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Erratic or failed power supply.

K/A# A2.01 **K/A Importance** 2.9 **Exam Level** SRO

References provided to Candidate ½-ODC-3.03, Rev. 11 **Technical References:** ½-ODC-3.03, Rev. 11, pg. 20, 24, 37, 39, 43, 46 1SQS-43.1, Rev. 13, PPNT Slide # 2

Question Source: Modified Bank – Vision # 598

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(2)&(4)

Objective: 1SQS-43.1 12. Using a copy of the Technical Specifications, Licensing Requirements Manual, or Offsite Dose Calculation Manual, analyze a given set of plant conditions for compliance with the licensing requirements, including determination of the equipment operability and applicable action statement.

91. With the Interlock Override Key switch is in Normal; Hoist motion, raise or lower, **can** occur when:

A. Raising a fuel assembly in Open Water with the load cell indicating 2500 lbs.

B. Raising the hoist with the gripper disengaged, load cell is indicating 1750 pounds.

C. Lowering a fuel assembly on index, load cell indicating 2325 lbs., and the Hoist Underload red light illuminates.

D. Raising a fuel assembly on index, load cell indicating 2625 lbs., and the Hoist Overload red light illuminates.

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.G page 25. The SRO must demonstrate the ability to monitor refueling activities as the Refueling Floor SRO. Specifically, the SRO must determine system if functioning as designed and that the required equipment load interlocks actuate appropriately. The SROs are the responsible Operations personnel at the site for Refueling, the reactor operators have limited Refueling responsibilities.

K/A is met by the candidate determining that which set of load conditions would exceed the limits of the manipulator crane hoist operation.

A. Correct. The load cell reading is within the tolerance limits to allow the hoist to be raised or lowered. An unrodded assembly is ~2475 lbs and a rodded assembly is ~2625 lbs.
B. Incorrect. Plausible since the gripper is disengaged from the assembly, however the load is above the <1200 lbs. limit to raise the hoist.
C. Incorrect. Plausible since the load is within the limit to operate the hoist however the underload limit is interlocked to stop lowering the hoist.
D. Incorrect. Plausible since the load is within the limit to operate the hoist however the overload limit is interlocked to stop raising the host.

Sys #	System	Category	KA Statement
034	Fuel-Handling Equipment	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fuel Handling System controls including:	Load limits

K/A# A1.01 **K/A Importance** 3.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** 3SQS-6.13 Rev. 6 pgs. 11-12; 1RP-3.3 Rev. 9 pgs. 8-9

Question Source: Modified – Vision 124583

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.5 / 45.5)

Objective: 3SQS-6.13-01-03 Given a specific plant condition, predict the response of the Fuel Handling System indication and control, including all automatic functions and changes in equipment status.

3SQS-6.13-01-09: Describe the control, protection and interlock functions for the fuel handling equipment, including automatic functions, setpoints and changes in equipment status as applicable.

92. The plant was performing a Tech Spec shutdown due to high RCS activity caused by fuel damage.

During the shutdown a LOCA occurred. The following conditions exist:

- The crew is performing the actions of E-1, Loss of Reactor or Secondary Coolant.
- RCS pressure is 1100 psig and LOWERING
- SG pressures are 'A' – 800 psig, 'B' – 850 psig, 'C' – 825 psig
- CNMT pressure 4 psig and RISING
- CNMT radiation is 1.2×10^5 R/Hr and STABLE
- HHSI flow is 600 gpm and slowly RISING

Based on the above conditions:

1) The RCPs should _______________?
2) How many Fission Product Barriers have exceeded a threshold for Loss or Potential Loss of a barrier?

A. 1) be tripped
2) Only 2 barriers

B. 1) remain running
2) All 3 barriers

C. 1) be tripped
2) All 3 barriers

D. 1) remain running
2) Only 2 barriers

.uestion 92

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E 4th bullet on page 8. Specifically, the SRO must have knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. In this case the SRO must demonstrate the ability to determine the number of fission product barriers have been lost or potentially lost due to containment radiation readings with the containment iaw EEP Wallboard table 1F-2.

K/A is met by demonstrating the ability to recognize that the containment area radiation monitor is indicating adverse conditions in containment and must interpret the RCS/ highest SG D/P RCP trip criteria from the pressures given and determine that the RCPs are required to be tripped.

A. Incorrect. First part is correct because RCS/SG D/P is < 350 psid. Second part is incorrect but plausible if the candidate does not recognize the potential loss of the cnmt barrier.

B. Incorrect. First part is plausible if the candidate does not recognize that cnmt is adverse due to radiation levels and determines the RCS/SG D/P is 250 psid which is greater than 200 psid required for non-adverse RCP trip criteria and does not trip the RCPs. Second part is correct.

C. Correct. The RCS/ highest SG D/P RCP trip criteria of <200 psid (350 psid adverse) has been exceeded. With cnmt radiation levels >1 x10^5 R/Hr, the cnmt is adverse. RCS/ highest SG D/P is 250 psid, therefore the trip criteria of <350 psid has been exceeded. All 3 Fission Product Barriers have exceeded a threshold for Loss or Potential Loss of a barrier due to cnmt radiation levels iaw EPP table 1F-2.

D. Incorrect. First part is plausible if the candidate does not recognize that cnmt is adverse due to radiation levels and determines the RCS/SG D/P is 250 psid which is greater than 200 psid required for non-adverse RCP trip criteria and does not trip the RCPs. Second part is incorrect but plausible if the candidate does not recognize the potential loss of the cnmt barrier.

Sys #	System	Category	KA Statement
072	Area Radiation Monitoring	Generic	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

K/A# 2.2.44 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate U1 EAL Wallboard

Technical References: 1OM-53A.1.E-1 Iss. 3 Rev. 0 LHP
1OM-53A.1.E-0 Iss. 3 Rev. 1 LHP
U1 EAL Wallboard (EPP-I-1a.F01 Rev 4)

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.5 / 43.5 / 45.12)

Objective: EPP-9281, Rev. 13 Obj. 11. Given specific plant conditions, classify the condition in accordance with EPP I-1a & b.

93. A Heat Actuating Device (HAD) in the Cable Tray Mezzanine has short circuited, causing an actuation of the fire suppression system.

    1) Which of the following annunciators do you expect to alarm in the control room?

    2) Per 1/2-ADM-1900, what type of Fire Watch is required if the fire suppression system in this area were isolated due to the inadvertent actuation?

A. 1) A11-113, ENGINE DRIVEN FIRE PUMP RUNNING
   2) Hourly fire watch patrol

B. 1) A11-113, ENGINE DRIVEN FIRE PUMP RUNNING
   2) Continuous fire watch patrol

C. 1) A11-79, MAIN PLANT CO2 STORAGE TANK PRESS HIGH-LOW
   2) Hourly fire watch patrol

D. 1) A11-79, MAIN PLANT CO2 STORAGE TANK PRESS HIGH-LOW
   2) Continuous fire watch patrol

¿uestion 93

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.A page 3. The SRO must demonstrate the ability to administer the fire protection program requirements, such as compensatory actions associated with inoperable sprinkler systems and fire doors. Specifically, the SRO must determine that the HAD is inoperable, and using ADM-1900, Fire Protection Program, station a continuous fire watch in the area.

K/A is met by the candidate determining that when a Heat Actuating Device (HAD) in the Cable Tray Mezzanine short circuits, a CO2 discharge will occur causing annunciator A11-79 to alarm, and the detector will no longer provide automatic fire protection in the area. Then based on this inability to perform its function, determines that a continuous fire watch is required iaw ADM-1900, Fire Protection Program.

A. Incorrect. Plausible because several areas in the plant are covered by the water suppression system, but the cable tray mezz is protected by CO2. Second part is plausible if the candidate thinks the cable mezzanine is listed under the hourly fire watch areas.

B. Incorrect. Plausible because several areas in the plant are covered by the water suppression system, but the cable tray mezz is protected by CO2. Second part is correct iaw ADM-1900 Att. B.

C. Incorrect. First part is correct. Second part is plausible if the candidate thinks the cable mezzanine is listed under the hourly fire watch areas.

D. Correct. The candidate must know that the Cable Tray Mezzanine which is the largest volume to be protected by Main Plant (10 ton) CO2 unit and will lower the level and pressure of the CO2 tank and bring in annunciator A11-79. Att B of ADM-1900 states that a continuous fire watch must be stationed within 1 hour if the system is not capable of automatic discharge to the protected area, or CO2 tank parameters not within spec.

Sys #	System	Category	KA Statement
086	Fire Protection	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Inadvertent actuation of the FPS due to circuit failure or welding.

K/A# A2.03 **K/A Importance** 2.9 **Exam Level** SRO

References provided to Candidate None

Technical References: 1OM-33.1.C Rev 11 pg. 4
1OM-33.4.AAZ Rev 4 pg. 2
1/2-ADM 1900 Rev 41 pgs. 57-58

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Objective: 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.

3SQS-33.1-1-06: Identify the Fire Protection System field instruments, subsystems and components that are required to be operable by the 1/2-ADM-1900 "Fire Protection".

Beaver Valley Unit 1 NRC Written Exam (1LOT18)
(SRO ONLY)

94. Given the following conditions:

- The plant is in MODE 3 preparing for a startup.
- #1 RDMG set motor AND generator breakers are closed, and BOTH Reactor Trip Breakers 'A' and 'B' are shut for rod control testing.
- Lift coil disconnect switches are closed.

IAW 1OM-1.2.A, Rx Control and Protection P&Ls, which of the following identifies how many Reactor Coolant loops are REQUIRED to be in operation, and why?

A. Two loops in operation to meet the accident analysis limits of an inadvertent control rod withdrawal.

B. Two loops in operation to meet the safety analysis acceptance criteria for DNB.

C. Three loops in operation to meet the accident analysis limits of an inadvertent control rod withdrawal.

D. Three loops in operation to meet the safety analysis acceptance criteria for DNB.

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B page 3, third bullet. The SRO must have knowledge of TS bases that are required to analyze TS-required actions and terminology. Specifically, the SRO must correlate the precaution and limitation of the control rod system into the bases derived from the Tech Spec bases of RCS loop operation in mode 3.

K/A is met by applying knowledge of the number of RCS loops required to be in operation during rod control testing and explain the bases for this requirement.

A. Correct. 2 loops are required whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized and the Rod Control System is capable of withdrawing rods. This is due to the postulation of a power excursion because of an inadvertent control rod withdrawal.

B. Incorrect. First part is correct. DNB is plausible because it is the bases for TS 3.4.4, RCS loops Modes 1& 2.

C. Incorrect. Plausible if the candidate thinks that mode 3 requires 3 loops whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized and the Rod Control System is capable of withdrawing rods. 3 loops are required in modes 1 & 2. Second part is the reason for 2 loops in operation as stated in TS 3.4.5 bases.

D. Incorrect. Plausible if the candidate thinks that mode 3 requires 3 loops whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized and the Rod Control System is capable of withdrawing rods. 3 loops are required in modes 1 & 2. DNB is plausible because it is the bases for TS 3.4.4, RCS loops Modes 1& 2.

Sys #	System	Category	KA Statement
N/A	N/A	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:** 1OM-1.2.A Rev 10 pg. 5 P&L 25; Tech Spec pg. B3.4.5-1 & 2

Question Source: Modified - Salem 2012 NRC Exam SRO 19

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 43.2 / 45.12)

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

95. NOP-LP-4011, FENOC Work Hour Control requires the Shift Manager to ensure that no personnel exceed 10 CFR 26 work hour limits without appropriate prior authorization.

Which of the below listed items are 10 CFR 26 work hour limits?

(Assume both Units are at 100% power with all systems in NSA)

1. No more than 20 work hours in any 32-hour period.
2. No more than 16 work hours in any 24-hour period.
3. No more than 26 work hours in any 48-hour period.
4. No more than 72 work hours in any 7-day.
5. No more than 72 work hours in any 168-hour period.
6. A 34-hour break in any 9-day period.

A. 1, 2, 3, & 4, ONLY

B. 1, 2, 5, & 6 ONLY

C. 2, 3, 4, 5, & 6 ONLY

). 1, 3, 4, 5, & 6 ONLY

Answer: C

Explanation/Justification: This meets the SRO only requirement for meeting conditions and limitations in the facility license as defined in 10CFR 55.43(b)(1). This is also an SRO responsibility at BVPS as stated in the NOPLP-4011 section 4.1.4.

K/A is met by demonstrating the knowledge of the work hour rules and ensuring the crew personnel adhere to the work hour rules as set forth in NOP-LP-4011 and 10 CFR 26.

A. Incorrect. Incorrect. Item 1 is not required and item 6 is required.
B. Incorrect. Item 1 is not required and item 4 is required.
C. Correct. IAW NOP-LP-4011 Rev. 12 pages 15 and 16.
D. Incorrect. Item 1 is not required and item 2 is required.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

K/A# 2.1.4 **K/A Importance** 3.8 **Exam Level** SRO

References provided to Candidate None **Technical References:** NOP-LP-4011 Rev. 13 pgs 10, 15, 16

Question Source: Bank – 2LOT8 NRC Exam Q94

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.2)

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.

96. Given the following plant conditions:

- The Unit is operating at 100% power with all systems in NSA.
- Scaffolding is scheduled to be built around an operating Heater Drain Pump in preparation for upcoming outage.
- No grid activities are scheduled to be performed.
- You are asked as part of your Senior Reactor Operator (SRO) responsibilities to perform a risk assessment in accordance with NOP-OP-1007, "Risk Management".

Using the **reference provided**, which of the following describes the **HIGHEST** classification of risk associated with the stated plant conditions?

A. Green

B. Yellow

C. Orange

D. Red

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E page 6. The SRO must demonstrate the ability to assess facility conditions and the selection of appropriate procedures during normal, abnormal, and emergency situations. Specifically, the SRO must determine that building scaffolding near the heater drains pump while at power is a yellow risk work item iaw the sites Risk Management procedure.

K/A is met by the candidate determining that building scaffolding around the Heater Drain Pump is assessed as yellow risk due to work in the Power Block near sensitive equipment in accordance with NOP-OP-1007, Risk Management.

A. Incorrect. Refer to correct answer explanation.

B. Correct. According to NOP-OP-1007, "Risk Management", section 2.3 specifies that exempt activities are listed in Attachment 1. Exempt activities shall be conducted as Green Risk activities. Attachment 1, #3 states that for the work to be exempt is must be a maintenance activity where work is not within the power block which is not the case (Heater Drain Pump is within power block), so therefore this is NOT a green risk exempt activity. Section 2.8 states that for plant conditions an SRO shall perform risk assessment using Attachment 3. Attachment 3, #4 states that any physical activity performed near protected train equipment or trip sensitive equipment that would cause a plant transient (e.g., building a scaffold over a protected diesel or an operating feed pump at-power.) is a yellow risk activity.

C. Incorrect. Refer to correct answer explanation.

D. Incorrect. Refer to correct answer explanation.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

K/A# 2.2.17 **K/A Importance** 3.8 **Exam Level** SRO

References provided to Candidate NOP-OP-1007, Rev. 27 (Procedure Body and Attachment 1,2, 3 ONLY) **Technical References:** NOP-OP-1007, Rev. 27 pg. 38, 40

Question Source: Bank – 2LOT7 NRC Exam Q96

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.13)

Objective: 3SSG-ADMIN-01-09 Explain the process for planning and scheduling maintenance and PRA integration into the 12 week rotating schedule in accordance with NOP-OP-1007.

ATTACHMENT 1: EXEMPT ACTIVITIES

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1. Routine station Activities including Chemistry and Radiation Protection samples performed on a weekly or more frequent basis, pump oil additions, and operator rounds.
2. Work on: spare parts, bench work, shop fabrication, and staging of materials, equipment, components, supplies and fittings.
3. Facility Maintenance Activities meeting ALL of the following criteria:
   - Work scope is not Power Block Equipment. (Reference NOP-WM-0001) ⇒ NOT EXEMPT
   - The work does not affect Safety, Regulatory, Tech Spec, Code, or Environmental Qualifications.
4. Toolpouch Maintenance Activities in accordance with Repair Identification and Toolpouch Maintenance, NOP-WM-9001.
5. Fire Protection Activities:
   - Fire hose, extinguisher, fire fighting equipment and plant door inspections.
   - Fire Detector Surveillances that are detection only and are not interlocked with any other system.
6. Non-Tech Spec required crane and manlift Surveillances.
7. Work Activities on Security equipment.
8. Non Intrusive Activities performed without physical impact (e.g., engineering calculations, run time data gathering, setpoint verifications, visual inspections, etc.).
9. Lubrication Activities not requiring additional equipment operation.
10. Removing dried boron off piping and equipment.
11. Maintenance Activities under the following conditions:
    - local instrumentation when isolated from the process stream,
      - in an isolation/thermal well or that is isolated from the process stream,
      - there is no interface with control or remote indicators or other equipment and,
      - no common process taps with other instruments unless they are also not included in this process.
    - Activities on instrumentation loops, which only provide local, remote panel or Control Room alarms, computer point or indication changes with no actuation or equipment control.
    - cleaning or replacing external screens and filters (for example, Main Feedwater Pump motors, Inverters, etc.)
    - shop work.
12. Non-safety related thermal insulation (non-asbestos) removal/modification and installation Activities.
13. Operations surveillances which have no impact to grid or unit reliability.

ATTACHMENT 3: RISK ASSESSMENT WORKSHEET
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ALL "NO" answers to the questions indicate the Activity poses Green Risk to Nuclear Safety, Personal Safety, Radiological Safety, Environmental Safety and Generation. See NORM-OP-1007 for Risk Assessment Question Basis.

An asterisk * signifies items that also need to be classified as Generation Risk Activities.

A.	**NUCLEAR SAFETY RISK ASSESSMENT**	
1.*	Activity, if performed incorrectly (i.e., any single unrecoverable error), that would cause a reactor trip, unplanned power change >0.5%, unplanned start/stop/ actuation of trip sensitive Engineered Safety Features, actuation of Turbine Trip circuitry or cause a RPS trip signal and the channel is not bypassed. If answer is yes, this Activity will be Yellow Risk.	YES/NO
2.	Activity, if performed incorrectly (i.e., any single unrecoverable error), that would cause a loss of, or defeat of a safety system. If answer is yes, this Activity shall be Yellow Risk.	YES/NO
3.*	Activity, if performed incorrectly (i.e., any single unrecoverable error), or a planned evolution that would cause a loss or unavailability of an Offsite Power Source or Emergency Diesel Generation. If answer is yes, this Activity shall be Yellow Risk.	YES/NO
4.*	Any physical Activity performed on or near trip sensitive equipment or within a protected train boundary that would cause a plant transient or loss of an engineered safety feature. (e.g., building a scaffold over a protected diesel or an operating feed pump at-power.) If answer is yes, this Activity will be Yellow Risk.	YES/NO
5.	Activity that will affect reactivity sensitive or reactivity monitoring systems in a manner that increases the likelihood of a Reactivity Event Level 1-4. Refer to NOP-OP-1004 for Reactivity Sensitive systems. If answer is yes, this Activity will be Yellow Risk.	YES/NO
6.*	Activity, if performed incorrectly (i.e., any single unrecoverable error), would cause a loss of a safety electrical bus. If answer is yes, this Activity will be Yellow Risk.	YES/NO
7.*	Activity places the plant in a single-point vulnerability for a Reactor trip, Turbine Generator trip, Main Feed Pump trip, or unplanned power change of >0.5%. Single point vulnerabilities are activities where one redundant component or train of equipment is nonfunctional. e.g. operation with one Generator Output Breaker open, one Control Rod Drive power train out of service, one Electrohydraulic (EHC) pump out of service, or one Main Feed Pump Lube Oil Pump out of service. If answer is yes, this Activity will be Yellow Risk.	YES/NO

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97. Given the following plant conditions:

- A plant heatup from a refueling outage is currently in progress.
- Highest available RCS temperature is 325°F and rising.
- The 'A' Quench Spray Pump has been declared INOPERABLE.

Which of the following completes the statements below?

1) The current technical specification operational MODE is ______________.
2) Assuming all other requirements for the next higher Mode have been met, IAW Tech Spec 3.6.6, Quench Spray System, a change to the next higher MODE is permitted __________.

A. 1) 3
   2) with no additional actions required

B. 1) 4
   2) with no additional actions required

C. 1) 3
   2) ONLY if an additional risk assessment is performed

D. 1) 4
   2) ONLY if an additional risk assessment is performed

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B 2nd bullet page 3. The SRO must be able to apply the generic limiting condition for operation (LCO) requirements (LCO 3.0.1 through 3.0.7; SR 4.0.1 through 4.0.4). Specifically, the SRO must recognize that both trains of the QSS are required to be operable, and the LCO is not met. Therefore TS 3.0.4 must be addressed which requires a risk assessment to be performed.

K/A is met by demonstrating the candidates' ability to determine the plant is in Mode 4 based on given plant conditions iaw Tech Spec definitions.

A. Incorrect. Mode 3 is incorrect, plausible if it is not known that mode 3 is ≥350F. Second part is plausible if the candidate thinks that LCO 3.6.6 permits continued operation for an unlimited period of time since the LCO background states only one train is required to perform the system function, or doesn't remember that a risk assessment is required to change modes IAW LCO 3.0.4,

B. Incorrect. Mode 4 is correct. Second part is plausible if the candidate thinks that LCO 3.6.6 permits continued operation for an unlimited period of time since the LCO background states only one train is required to perform the system function, or doesn't remember that a risk assessment is required to change modes IAW LCO 3.0.4,

C. Incorrect. Mode 3 is incorrect, plausible if it is not known that mode 3 is ≥350F. Second part is correct IAW LCO 3.0.4.

D. Correct. Mode 4 is 350F>Tavg>200F. Second part is correct because LCO 3.6.6 is not met because both trains of the QSS are required to be operable in mode 1-4. This requires knowledge of LCO 3.0.4 to allow a risk assessment to be performed to permit a change to a higher mode. LCO 3.0.4 allows a mode change with LCO unlimited operation time, risk assessment, or if the LCO has a specific allowance stated within the LCO.

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 LCO 3.6.6 and 3.0.4 TS Table 1.1-1
Question Source:	New				
Question Cognitive Level:		Higher – Comprehension or Analysis		**10 CFR Part 55 Content:**	(CFR: 41.7 / 41.10 / 43.2 / 45.13)
Objective:	3SQS-CONT ITS, Rev. 1	4. Given plant conditions that constitute non-compliance with any Containment Systems LCO, or Licensing Requirement determine the applicable Condition(s), Required Action(s), and associated Completion Time(s).			

98. Given the following:

- A Refueling SRO is entering the Spent Fuel Pool area.
- While reviewing his RWP prior to beginning work, he notices an ALARA briefing is required.
- The dose rate is 1100 mrem/hr due to damaged fuel assemblies.
- The Refueling SRO will also exceed 2000 mrem Annual TEDE limits while in the area.

Which of the following completes the following statements?

1) Based on the area dose rate, the Refueling SRO will be required to receive an ALARA briefing prior to ________________ during the work shift.

2) Per NOP-OP-4201, "Routine External Exposure Monitoring," the ________________ is the MINIMUM authority level required to exceed the limit.

A. 1) the first entry only
2) Radiation Protection Manager

B. 1) the first entry only
2) Site Vice President

C. 1) each entry
2) Radiation Protection Manager

D. 1) each entry
2) Site Vice President

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‹uestion 98

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .D page 6 second bullet. Specifically, the SRO must make an analysis and interpretation of radiation and activity readings as they pertain to the selection of administrative, normal, abnormal, and emergency procedures. In this question the SRO is required to have specific procedural knowledge of whether an ALARA brief is required for every entry into the spent fuel pool during refueling, and whose authorization is required to exceed his site Administrative Control Level (ACL).

K/A is met with the knowledge that after reviewing the RWP the Refueling SRO recognize that an ALARA briefing is required for every entry into the area, and knows whose authorization is needed to get an Administrative Control Level (ACL) extension.

A. Incorrect. Plausible because this appears to be acceptable for an ALARA brief, but since it is a LHRA, subsequent briefs are required for subsequent entries. RPM is correct.
B. Incorrect. Plausible because this appears to be acceptable for an ALARA brief, but since it is a LHRA, subsequent briefs are required for subsequent entries. Site VP is plausible because they are required to approve Administrative Control Level (ACL) in excess of 4000 mrem, and the Site VP is also required to approve emergency exposure.
C. Correct. ALARA briefing is required for each entry since this is a LHRA iaw NOP-OP-4101 and 4201. RPM approval is required for any ACL extension iaw form NOP-OP-4201-02.
D. Incorrect. ALARA briefing is required for each entry since this is a LHRA iaw NOP-OP-4101 and 4201. Site VP is plausible as stated above.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to comply with radiation work permit requirements during normal or abnormal conditions.

K/A# 2.3.7 **K/A Importance** 3.6 **Exam Level** SRO

References provided to Candidate None **Technical References:** NOP-OP-4101 rev. 15 sect. 4.15
NOP-OP-4201 rev 3 pg 14
NOP-OP-4201-02 (form) Rev. 2

Question Source: Bank – Vogtle 2013 NRC Exam Q96

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.12 / 45.10)

Objective: FEN-RWT Rev 4 Chapter 4 Obj. 5. State the actions to be taken if administrative dose control levels are being approached.

99. The plant is at 100% power with all systems in NSA EXCEPT #2 EDG is out of service for maintenance.
    - Annunciator A5-17, 1/3 Reactor Cool Pump Loop Flow Low Reactor Trip is received.
    - 'A' RCS Loop flow indicates 0% flow.
    - The reactor failed to automatically trip and manual trip from the control room was unsuccessful.
    - The crew is performing FR-S.1, Response to Nuclear Power Generation/ATWS.

    Current Conditions:

    - The RO is manually inserting rods
    - Reactor power is 45% and lowering.
    - A Loss of both emergency 4KV busses occurred during transfer to Offsite power.
    - #1 EDG failed to start.

    1) Based on the given conditions, and the **Transient Response Guidelines**, what is the procedural flowpath to mitigate the loss of all 4KV power?
    2) What is the bases for the Reactor Coolant Flow - Low reactor trip?

A. 1) Continue with FR-S.1, and enter AOP 1.36.2, Loss of 4KV Emergency Bus.
   2) It ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops.

B. 1) Continue with FR-S.1, and enter AOP 1.36.2, Loss of 4KV Emergency Bus.
   2) It ensures that protection is provided against violating Heat Flux Hot Channel Factor FQ(Z) limits due to low flow in one or more RCS loops.

C. 1) Exit FR-S.1, and enter ECA-0.0, Loss of All Emergency 4KV AC Power.
   2) It ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops.

D. 1) Exit FR-S.1, and enter ECA-0.0, Loss of All Emergency 4KV AC Power.
   2) It ensures that protection is provided against violating Heat Flux Hot Channel Factor FQ(Z) limits due to low flow in one or more RCS loops.

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Question 99

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.B page 3, third bullet. The SRO is required to have specific Knowledge of TS bases that are required to analyze TS required actions and terminology. Specifically, the SRO must know that the TS bases for loss of flow reactor trip ensures that protection is provided against violating the DNBR limit.

K/A is met with the knowledge of the required procedure entry conditions during a loss of all 4KV power while in FR-S.1. The NRC Exam Chief authorized deviation from the second part of the K/A due to it being RO level knowledge. The question was taken to SRO level knowledge with the knowledge of the Tech Spec bases for loss of flow reactor trip.

A. Correct. The Transient Response Guidelines states that the preferred method of addressing a loss of Emergency 4KV power is via AOP-36.2 or E-0, and entry into ECA-0.0 should not be performed until after the Command SRO reads the IOAs of AOP-36.2 or E-0. TS 3.3.1 Function 10 bases states "The Reactor Coolant Flow - Low trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow.

B. Incorrect. First part is correct. Heat Flux Hot Channel Factor is a plausible distractor because it sounds like a probable basis for a loss of RCS flow, but the Heat Flux Hot Channel Factor limit is designed to ensure that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition during a loss of flow event.

C. Incorrect. Entering ECA-0.0 is plausible but does not meet BV expectations iaw the Transient Response Guidelines. The TRGs states that the preferred method of addressing a loss of Emergency 4KV power is via AOP-36.2 or E-0. Second part is correct.

D. Incorrect. Entering ECA-0.0 is plausible but does not meet BV expectations iaw the Transient Response Guidelines. The TRGs states that the preferred method of addressing a loss of Emergency 4KV power is via AOP-36.2 or E-0. Heat Flux Hot Channel Factor is a plausible distractor because it sounds like a probable basis for a loss of RCS flow, but the Heat Flux Hot Channel Factor limit is designed to ensure that the hot fuel rod in the core does not experience a departure from nucleate boiling (DNB) condition during a loss of flow event.

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** 4.8 **Exam Level** SRO

References provided to Candidate None **Technical References:** BVPS-OPS-0024 Rev. 11 pg. 11 (sect. 4.10)
Tech Spec Bases 3.3.1 Function 10 pg. B3.3.1-21

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.13)

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.

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.

100 Initial plant conditions:

- Plant is in MODE 5 on the 'A' Train of RHR
- 4160V Bus DF is de-energized for maintenance, and cannot be immediately restored
- Vacuum refill of the RCS has been completed
- RCS temperature is 180°F
- RCS pressure is 175 psi
- All Steam Generators are in Wet-Layup
- No RCPs are running

'A' RHR Pump fails due to a motor bearing failure.

Which of the following actions are required in accordance with AOP-1.10.1, Loss of Residual Heat Removal Capability?

A. Establish feed and bleed; using one Charging Pump from the RWST and open all PORVs.

B. Establish feed and bleed; using two Charging Pumps from the RWST and open all PORVs.

C. Establish Natural Circulation; control SG level and open the steam generator atmospheric dump valves.

D. Start one Reactor Coolant Pump, control SG level and open the steam generator atmospheric dump valves.

QUESTION 100

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II.E page 6. Specifically, the SRO must have specific knowledge of the content of the procedure that is being used. In this case the SRO must have specific knowledge of AOP 1.10.1, Loss of RHR, and determine which condition is permitted for RCS heat removal during a loss of both RHR pumps in Mode 5.

K/A is met by the candidate having the knowledge of the procedural requirements of the Loss of RHR AOP and determining the correct mitigation strategy for RCS heat removal after a loss of both RHR pumps.

A. Incorrect. Plausible because iaw AOP-1.10.1, step 15 states if cooling cannot be established using SGs, then establish RCS feed and bleed using one charging pump aligned to the RWST, and only one PORV.

B. Incorrect. Plausible because iaw AOP-1.10.1, step 15 states if cooling cannot be established using SGs, then establish RCS feed and bleed using one charging pump aligned to the RWST, and only one PORV.

C. Correct. The SGs are available since they are in wet layup and can be used as a heat sink since the RCS vacuum refill is completed. AOP 1.10.1 step 15f states establish natural circulation.

D. Incorrect. Plausible because the procedure directs these actions if conditions exist that support starting an RCP, but pressure at 175 psig is too low to start the RCP. AOP-1.10.1 requires the RCS to be pressurized to >215 psig.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

K/A# 2.4.9 **K/A Importance** 4.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.10.1 Rev 16 pgs. 18-21

Question Source: Bank – Indian Point 3 2013 NRC Exam Q81

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.13)

Objective: 1SQS-53C.1, Rev. 12 Obj. 5 - Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.