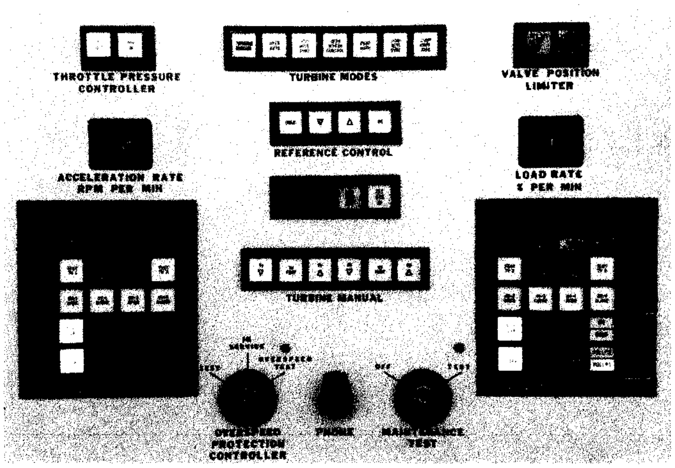
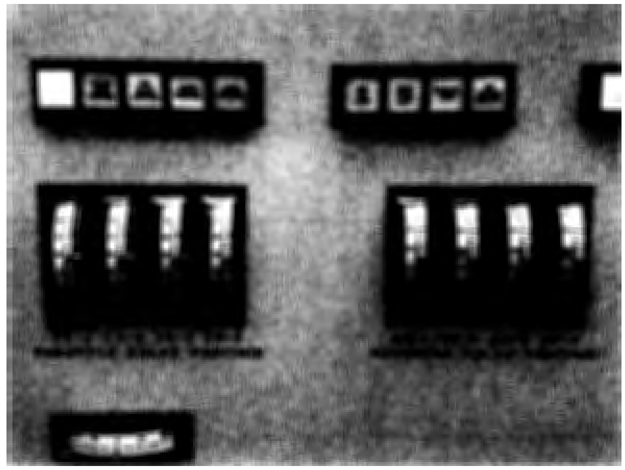
Beaver Valley Unit 1 NRC Written Exam (1LOT16)

1. Given the following:
   - A reactor trip has occurred from 100% power
   - E-0, "Reactor Trip or Safety Injection", is in progress at Step 2 - Verify Turbine Trip





Based on the above indications, in accordance with E-0, which of the following is the NEXT REQUIRED action?

A. Manually Trip the Turbine.

B. Initiate a Main Steam Line Isolation.

C. Verify power to at least one AC emergency bus.

D. Verify Main Generator Output breakers open.

2. The plant was at 100% power when a PRZR Safety valve failed open.

   - The crew is performing the actions of E-0, Reactor Trip or Safety Injection
   - PRZR pressure is 1150 psig and lowering
   - HHSI flow is 540 gpm and stable
   - SG pressures are:
     - 'A' SG - 950 psig
     - 'B' SG - 980 psig
     - 'C' SG - 970 psig

   Based on the above conditions, the RCPs should ______(1)______, and the basis for this requirement is ______(2)______.

A. 1) remain running
   2) to provide core heat removal via forced two-phase coolant flow through the core

B. 1) remain running
   2) to provide core heat removal via the break and the Steam Generators

C. 1) be tripped
   2) to prevent excessive depletion of RCS water inventory

D. 1) be tripped
   2) to prevent excessive depletion of the Steam Generator inventory

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Answer: C

Explanation/Justification: K/A met by giving a PRZR Safety valve failing open (small break LOCA) and requiring the interpretation of given RCS and SG pressures, and making a determination of whether or not to trip the RCPs as is required by the Left Hand Page of E-0, Rx Trip or Safety Injection.

A. Incorrect. RCPs are required to be tripped due to SG/RCS D/P being <200 psid, Second part is plausible since the RCPs could be restarted in FR-C.1 Response to Inadequate Core Cooling to provide decay heat removal.

B. Incorrect. RCPs are required to be tripped due to SG/RCS D/P being <200 psid. Second part is plausible since this would be correct if the HHSI pumps were not operating during a small break LOCA and the RCPs were left operating.

C. Correct. Trip criteria of E-0 LHP page has been met with HHSI flow and RCS/Highest SG D/P being <200 psid. (170 psid per the stem), therefore RCPs would be tripped to prevent excessive depletion of RCS water inventory through a small break in the RCS. RCP trip criteria applies for LOCA events of varying sizes and locations in the coolant system. Conditions could exist where core voiding occurs and a column of fluid is raised into the pressurizer with two phase flow through the safety valve, this would result in depleting RCS inventory. Tripping the RCPs when the criteria is met would limit the inventory loss through the valve. There is no restriction on RCP trip criteria based upon the location of the leak in the RCS.

D. Incorrect. RCPs would be tripped. Second part is plausible with this being the reason of tripping the RCPs in FR-H.1 Loss of Secondary Heat Sink.

Sys #	System	Category	KA Statement
000008	Pressurizer Vapor Space Accident / 3	Generic	Ability to interpret and execute procedure steps.

K/A# 2.1.20 **K/A Importance** 4.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.E-1 Iss 3 rev 0 LHP
1OM-53B.5.GI-6 Iss 2 Rev 0 pg. 10

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.12)

Objective: 3SQS-53.2, Rev. 2 Obj. 1. State from memory the basis for RCP trip criteria, IAW BVPS EOP Executive Volume.

3. The plant was operating at 100% power when a 1,000 gpm LOCA occurs. All systems operate as designed.

   10 minutes after the Reactor Trip, which of the choices below describes the trend of the following parameters?

   (Assume NO operator actions)

	SG Water Levels	Boron Concentration	CNMT Pressure	CNMT Temperature
A.	Rising	Stable	Rising	Rising
B.	Rising	Rising	Rising	Rising
C.	Lowering	Rising	Lowering	Lowering
D.	Lowering	Stable	Lowering	Lowering

Answer: B

Explanation/Justification: K/A is met by having the candidate interpret how cnmt temperature, cnmt pressure, boron conc, and SG water level will respond after a small break LOCA occurs.

A. Incorrect. Plausible distractor if it is not recognized that Safety Injection will automatically initiate on low pressurizer pressure and raise Boron concentration. All other trends are correct.

B. Correct. SGWL will rise due AFW injection. Boron will rise due to Safety Injection injecting from the RWST. Cnmt pressure and temperature will rise due to the 1000 gpm high energy small break LOCA inside containment.

C. Incorrect. Plausible distractor if it is not recognized that on the Rx trip SGWLs will drop below the auto start setpoints for AFW pumps, and they think that Quench Spray has initiated due to cnmt high pressure (11.1 psig) and the cnmt pressure and temperature is lowering due to spray.

D. Incorrect. Plausible distractor if it is not recognized that on the Rx trip SGWLs will drop below the auto start setpoints for AFW pumps. Boron conc stable if it is not recognized that Safety Injection will automatically initiate on low pressurizer pressure or cnmt pressure of 5 psig. Pressure and Temp lowering if they think that Quench Spray has initiated due to cnmt high pressure.

Sys #	System	Category	KA Statement
000009	Small Break LOCA / 3	EA2. Ability to determine or interpret the following as they apply to a small break LOCA:	Containment temperature, pressure, and humidity

K/A# EA2.11 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** GO-3ATA-4.2, Rev. 8 pg. 70; 1OM-53B.4.E-1 Iss. 3 Rev. 0 pg. 3-5

Question Source: New

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

Objective: GO-ATA 4.2, Rev. 8 Obj. 1 - Predict and explain plant automatic actions which mitigate the consequences of each type of LOCA.

4. 1) How is Containment Operability maintained with the 'A' Outside Recirc Spray Pump in STANDBY?

   2) IF the Outside Recirc Spray pumps were running due to a Large break LOCA, and A1-108, "Outside Recirc Spray PP 2A Seal Water Level Low" alarm was received, which of the following actions are required to be taken in accordance with A1-108 ARP?

   The ______(1)______ will maintain containment operability when the 'A' Outside Recirc Spray Pump is in standby.

   ARP A1-108 will direct the crew to ______(2)______.

A. 1) Suction and Discharge Isolation valves being CLOSED
   2) fill the pump seal system by using plant demineralized water

B. 1) Suction and Discharge Isolation valves being CLOSED
   2) shutdown and isolate the Outside Recirc Spray 2A Pump when allowed by the EOPs

C. 1) Pump Seal
   2) fill the pump seal system by using plant demineralized water

D. 1) Pump Seal
   2) shutdown and isolate the Outside Recirc Spray 2A Pump when allowed by the EOPs

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Answer: D

Explanation/Justification: K/A is met by identifying that the Outside RSS pump seal is used to maintain containment integrity in order to comply with TS 3.6.1 due to the suction and discharge valves being NSA open. Further knowledge is required to identify that the RSS pump must be shut down and isolated as soon as permissible when the Seal Water Level Low alarms due to the potential release of radioactive fluid to the safeguards area.

A. Incorrect. Plausible distractor as both valves are containment isolation valves, but when the RSS pumps are in standby, both valves are NSA open. Second part is plausible because normal response to fill the seal, but with a large break LOCA in progress, a note in the ARP states with CIB in progress access to the seal area is not possible, so the normal action of filling the seal is not possible.

B. Incorrect. Plausible distractor as both valves are containment isolation valves, but when the RSS pumps are in standby, both valves are NSA open. Second part is correct.

C. Incorrect. Pump seal does maintain the containment integrity. Second part is plausible because normal response to fill the seal, but with a large break LOCA in progress, a note in the ARP states with CIB in progress access to the seal area is not possible, so the normal action of filling the seal is not possible.

D. Correct. Pump seal does maintain the containment integrity since the RSS pump suction and discharge valves are NSA open. Securing and isolating the RSS pump when permitted is directed by the ARP due to the potential loss of containment integrity during a CIB event.

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:	That equipment necessary for functioning of critical pump water seals is operable

K/A# EA2.07 **K/A Importance** 3.2? **Exam Level** RO

References provided to Candidate None **Technical References:** 1SQS-13.1 Rev. 15 pg. 14 & 32; 1OM-13.4.AAM Rev. 2 pg.2

Question Source: New

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

Objective: 1SQS-13.1 Rev 15 Obj. 10 - Given a Containment Depressurization System alarm condition and using the Alarm Response Procedure(s), determine the appropriate alarm response, including automatic and operator actions in the field.

1SQS-13.1 Rev 15 Obj. 14 - In the field, locate all of the components identified on the Normal-System-Arrangement System Flow path drawings that are accessible during normal operation within the ALARA considerations.

5. Given the following conditions:

- Plant startup is in progress
- Rx power is 7% and slowly RISING
- Annunciator A3-78, Reactor Cool Pump Seal Injection Flow Low is LIT
- Annunciator A3-87, Reactor Cool Pump Seal Leakoff Flow High is LIT
- FI-1CH-124, 'C' Seal Injection Flow indicates 0 gpm
- FR-1CH-154A, 'C' Seal Leakoff Flow is 5.7 gpm and STABLE
- TV-1CC-107C, 'C' RCP Thermal Barrier CCR Outlet Isol Vlv is CLOSED

1) Which of the following is the correct sequence of actions required for this event IAW AOP 1.6.8, Abnormal RCP Operation?

1) Manually trip 'C' RCP
2) Manually trip Reactor, enter E-0, and perform IOAs
3) Close PCV-1RC-455B, Loop 'C' PRZR Spray Valve
4) Close MOV-1CH-303C, RCP 'C' No. 1 Seal Leakoff Isol Vlv (3-5 minutes after RCP S/D)

2) Based on the above conditions, what is the reason for the valve manipulation at the end of these required actions?

A. 1) 1, 3 ONLY
2) Prevent the active loop spray flow from flowing back through the idle loop spray line rather than the pressurizer.

B. 1) 1, 4 ONLY
2) Allow the pump to stop rotating which helps to prevent debris from damaging the #2 seal, and isolate the low pressure piping.

C. 1) 2, 1, 3
2) Prevent the active loop spray flow from flowing back through the idle loop spray line rather than the pressurizer.

D. 1) 2, 1, 4
2) Allow the pump to stop rotating which helps to prevent debris from damaging the #2 seal, and isolate the low pressure piping.

QUESTION 5

Answer: C

Explanation/Justification: K/A is met evaluating parameters to determine which RCP malfunction has occurred, and testing the knowledge of the reason for the required sequence of events that will occur to place the reactor and RCP in a safe condition.

A. Incorrect. Plausible with reactor power <P-7 (auto block of 2/3 RCS loop low flow) candidate could think that tripping the 'C' RCP and closing the 1RC-455C without tripping the reactor is allowed. BV1 procedures do not allow this course of action without tripping the reactor. To prevent backflow is the correct reason for closing the spray valve.

B. Incorrect. Plausible with reactor power <P-7 (auto block of 2/3 RCS loop low flow) candidate could think that tripping the 'C' RCP and closing MOV-1CH-303C without tripping the reactor is allowed. This would be partial actions for a high seal leakoff (≥6 gpm), but the stem states that seal leakoff is only 5.7 gpm and stable. The high seal leakoff annunciator alarming helps with the distractor (setpoint is 4.7 gpm). BV1 procedures do not allow this course of action without tripping the reactor. If high seal leakoff was the correct reason for tripping the RCP, this would be the correct reason.

C. Correct. In accordance with AOP-1.6.8, a loss of both seal injection and thermal barrier flow are immediate RCP shutdown criteria and warrant tripping the reactor and taking these actions. To prevent backflow is the correct reason for closing the spray valve.

D. Incorrect. Plausible distractor if high seal leakoff was the actual failure because these are to the required actions of AOP-1.6.8 without closing the spray valve. If high seal leakoff was the correct reason for tripping the RCP, this would be the correct reason.

Sys #	System	Category	KA Statement
015/017	RCP Malfunctions / 4	AK3. Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow) :	Sequence of events for manually tripping reactor and RCP as a result of an RCP malfunction

K/A# AK3.03 **K/A Importance** 3.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.6.8. Rev 18 step 1
1OM-53B.5.GI-6 Iss. 2 Rev. 0 pg. 44-45

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. 6 - Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.

6. An event has occurred which has caused RCS pressure to rise.

   - RCS pressure is 2800 psig and STABLE
   - FCV-1CH-122, CHG Flow to Regen Hx Inlet Control Vlv controller is in AUTO
   - PRZR level is 70%

   Based on the above conditions, which of the following choices complete the statements below?

   FI-1CH-122A will indicate ___(1)___ gpm.

   The RCS Pressure ___(2)___ been maintained below the RCS Pressure Safety Limit.

A. 1) 0
   2) has NOT

B. 1) 0
   2) has

C. 1) 15
   2) has NOT

D. 1) 15
   2) has

Answer: A

Explanation/Justification: K/A is met by requiring the candidate to recall the shutoff head of the charging pumps (2600 psig), and determining that a at 2800 psig in the RCS, a loss of Reactor Coolant Makeup has occurred. The pressure differential between charging and the RCS will cause charging flow to indicate zero gpm.

A. Correct. Charging pump shutoff head is 2600psig, and RCS pressure is 2800 psig, therefore indicated flow will be zero. The RCS pressure safety limit of ≤2735 psig has been violated.

B. Incorrect. It is correct that flow is zero, but RCS SL has been violated.

C. Incorrect. 15 gpm is a plausible distractor with the minimum charging flow set at 15 gpm in auto to prevent flashing downstream of the L/D orifices. It is correct that the RCS SL has been violated.

D. Incorrect. 15 gpm is a plausible distractor with the minimum charging flow set at 15 gpm in auto to prevent flashing downstream of the L/D orifices. The RCS SL has been violated.

Sys #	**System**	**Category**	**KA Statement**
000022	Loss of Rx Coolant Makeup / 2	AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup	Relationship of charging flow to pressure differential between charging and RCS

K/A# AK1.02 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-7.1.C Rev.8 pg. 3
SPD-FCV-1CH-122 Rev 00
Tech Specs pg. 2.0-1 Admend. 278/161

Question Source: New

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

Objective: 1SQS-7.1, Rev. 20 Obj. 9 - Given a set of plant conditions for the Chemical and Volume Control System describe the appropriate operating procedure(s) operational sequence, and the applicable parameter limits, precaution and limitations, and cautions & notes used to complete the task activities in the field.

3SQS-SL ITS, Rev. 0 obj. 2 - Given plant conditions, determine the criteria necessary to ensure compliance with each ITS Safety Limit.

7. Given the following initial conditions:
   - Plant is in Mode 5
   - 1RH-P-1A, 'A' RHR pump is running
   - 1CC-P-1A, 'A' CCR pump is running
   - DF bus is cleared for maintenance

   Final conditions:
   - 1CC-P-1A, 'A' CCR pump trips on overcurrent
   - 1CC-P-1C, 'C' CCR pump failed to start
   - Crew is performing AOP 1.15.1, Loss of Primary Component Cooling Water in conjunction with AOP 1.10.1, Loss of Residual Heat Removal Capability

   1) Which of the following describes the RHR components that have lost cooling water flow?

   2) What is the reason for monitoring RHS inlet temperature at this time?

A. 1) RHR Hx and RHR pump seal cooler **only.**
   2) If temperature exceeds 180°F, the RHR pumps must be tripped to prevent cavitation.

B. 1) RHR Hx and RHR pump seal cooler **only.**
   2) If temperature exceeds 180°F, the RHR pump must be tripped to prevent seal damage.

C. 1) RHR Hx, RHR pump seal cooler, and RHR pump motor lube oil cooler.
   2) If temperature exceeds 180°F, the RHR pumps must be tripped to prevent cavitation.

D. 1) RHR Hx, RHR pump seal cooler, and RHR pump motor lube oil cooler.
   2) If temperature exceeds 180°F, the RHR pump must be tripped to prevent seal damage.

Answer: B

Explanation/Justification: K/A is met by requiring knowledge of the RHR components which will lose cooling water flow when the closed cooling water pumps (CCR) are lost. The loss of CCR will also enter into a loss of RHR and require monitoring the RHR inlet temperature to minimize RHR pump mechanical seals.

A. Incorrect. CCR is supplied to only the Hx and the seal coolers. Tripping the RHR pumps due to cavitation is plausible since RHR will be heating up, but 180F is stated in the AOP for mechanical seal damage.

B. Correct. CCR is supplied to only the Hx and the seal coolers. The RHR temperatures are required to be monitored iaw AOP-1.10.1 step 9 to ensure the mechanical seals are not damaged if RHR inlet temperatures reach 180F.

C. Incorrect. Plausible distractor of RHR pump motor lube oil coolers with several other pumps having supplied cooling to the motors. Tripping the RHR pumps due to cavitation is plausible since RHR will be heating up, but 180F is stated in the AOP for mechanical seal damage.

D. Incorrect. Plausible distractor of RHR pump motor lube oil coolers with several other pumps having supplied cooling to the motors. The RHR temperatures are required to be monitored iaw AOP-1.10.1 step 9 to ensure the mechanical seals are not damaged if RHR inlet temperatures reach 180F.

Sys #	System	Category	KA Statement
000025	Loss of RHR System / 4	AK2. Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following:	Service water or closed cooling water pumps

K/A# AK2.03 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** U1 RM-0415-005 Rev 14; 1OM-53C.4.1.10.1 Rev 15 pg 15

Question Source: New

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

Objective: 1SQS-10.1 Rev. 18, Obj. 21-Given a set of plant conditions for the Residual Heat Removal System, describe the appropriate operating procedure(s) operational sequence, and the applicable parameter limits, precaution and limitations, and cautions & notes used to complete the task activities from the control room.

1SQS-15.1, Rev. 14, Obj. 2-Outline the general flowpath of the CCR system including the CCR pumps and heat exchangers, the supply headers, the CCR major loads and the return header

8. The plant is at 75% power with all systems in normal alignment for this power level **EXCEPT** MOV-1RC-535, PORV 455C MOTOR OPERATED ISOL VLV is **CLOSED** due to PORV seat leakage.

- PC-1RC-444A, PRZR PRESS CONTROL output fails to 10% in Automatic
- No Operator action is taken

What is the status of the plant 15 minutes after this event?

A. The plant will trip due to a PRZR Pressure LOW reactor trip.

B. The plant will trip due to a PRZR Pressure HIGH reactor trip.

C. The plant will remain at power and RCS pressure will cycle around the PORV setpoint.

D. The plant will remain at power and RCS pressure will stabilize at a lower reference pressure.

Answer: C

Explanation/Justification: K/A is met by the knowledge required to determine that the purpose of PC-1RC-444A is to control PRZR pressure and how the controller will function when it fails to 10%. This knowledge will encompass the response of the PORVs in the non-affected portion of the Pressurizer Pressure Control System, and the control function side with the heaters and spray valves, with the 455C PORV being isolated.

A. Incorrect. Plausible if PC-1RC-444A output failed high, causing both spray valves to open fully and depressurize the RCS until LP Rx trip occurs.

B. Incorrect. Plausible distractor because pressure would rise to 2375 psig and cause a HP Rx trip if the PORVs didn't lift at 2335 psig. Candidate must know that the Rx trip setpoint is higher than the PORV opening setpoint, and that the PORV will prevent the trip setpoint from being reached.

C. Correct. The plant will remain at power and the pressure will cycle between the PORV auto open setpoint of 2335 psig and close setpoint of 2315 psig. This will occur only on in-service PORVs 456 & 455D since 455C is manually isolated.

D. Incorrect. Plausible distractor because of a misconception that the pressure controller failing at 10% affects the reference pressure setting resulting in a lower pressure control setpoint. Pressure is maintained by the controller pot setting (normally set at 667 units for 2235 psig). This is incorrect because the reference setting will remain at 2235 psig, and the controller will close all spray valves and actually cause pressure to rise due to the controller output failing to 10% resulting in both spray valves closing and heaters energizing.

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

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

References provided to Candidate None **Technical References:** 1OM-6.4.IF Rev 11 Att. 2

Question Source: Bank - 2LOT15 NRC Exam (Q7)

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

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.

9. The plant is operating at 100% power when a turbine trip occurs.
   - The Rx failed to trip
   - All Control rods are fully withdrawn
   - Safety Injection has NOT initiated
   - The crew has entered FR-S.1, Response to Nuclear Power Generation – ATWS

The BOP initiates Emergency Boration with the indications given on the following page.

In regards of borating the RCS, what is the correct response to these indications in accordance with FR-S-1?

A. Initiate Safety Injection.

B. Align RWST to the Charging Pumps.

C. Align Blender to the Charging Pump suction.

D. Continue Emergency Borating as aligned.

Answer: B

Explanation/Justification: K/A is met by the candidate being required to interpret pump and valve position indications, and indications of emergency boration flow as given in the above pictures, then determine the correct boration flowpath as directed by FR-S.1.

A. Incorrect. Plausible distractor, but not warranted in this situation since FI-1CH-122A shows >75 gpm flowrate. This indicates that a charging pump is running, therefore there is no need for Safety Injection because other sequenced boration options are possible iaw FR-S.1.

B. Correct. As indicated in the pictures MOV-1CH-350 failed to open and FI-1CH-110 indicates zero, therefore iaw the RNO, aligning the RWST to the Charging pumps is correct.

C. Incorrect. Plausible distractor but not directed by FR-S.1. This method is an alternate iaw 1OM-7.4.S "Emergency Boration" procedure, but it also requires local valve manipulations in the blender cubicle.

D. Incorrect. Plausible distractor if MOV-1CH-350 indicated open and/or FI-1CH-110 indicated >30 gpm Emergency Boration flow as required by FR-S.1.

Sys #	System	Category	KA Statement
000029	ATWS / 1	EA2 Ability to determine or interpret the following as they apply to a ATWS:	System component valve position indications

K/A# EA2.05 **K/A Importance** 3.4* **Exam Level** RO

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

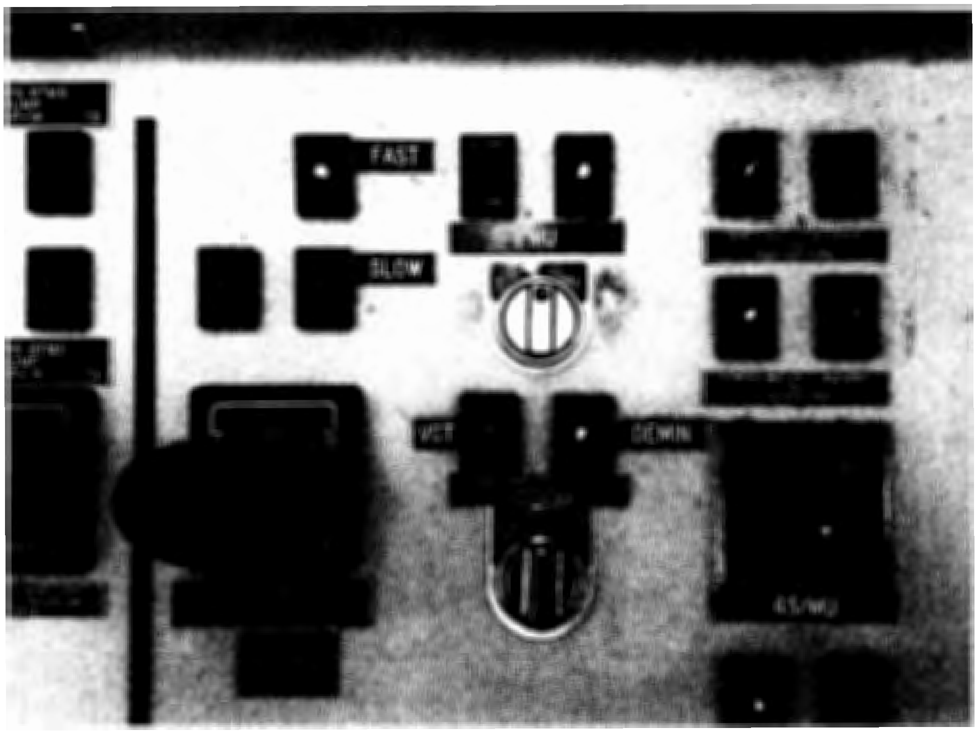
Question Source: New

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

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

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.

9. (continued)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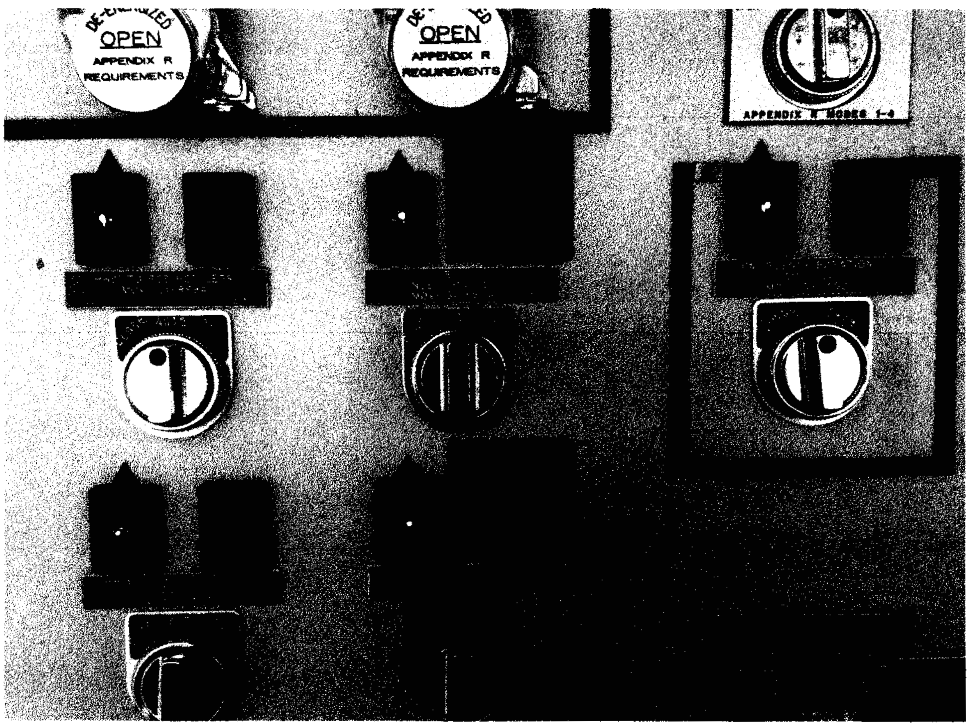
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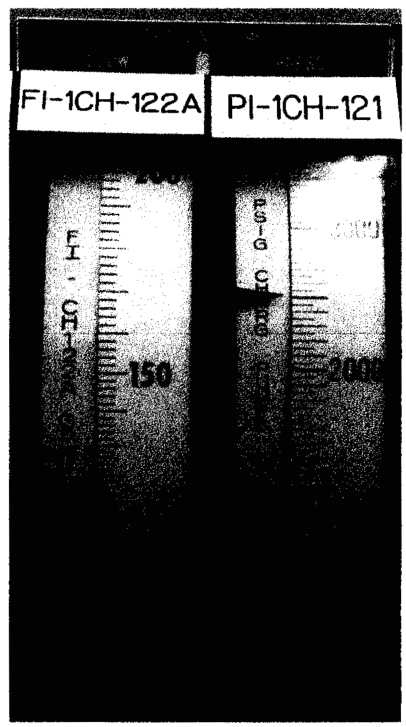
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10. The following conditions exist:

- 'B' SG has a 200 gpm tube rupture
- The crew is performing E-3, Steam Generator Tube Rupture
- Both Motor Driven AFW pumps are RUNNING
- MOV-1MS-105, AFW Turbine Steam Isol Vlv was CLOSED to secure the TDAFW pump

In accordance with E-3, which of the following actions must be completed prior to restarting the Turbine Driven AFW pump?

A. Close 1MS-16, 1B SG Steam Supply to 1FW-P-2 Isolation.

B. Close TV-1BD-100B, 1B SG Blowdown CNMT Isolation Valve.

C. Close TV-1MS-101B, Loop 1B Main Steam Trip Valve and NRV-1MS-101B, B SG NRTRN Vlv.

D. Reset the Turbine Driven AFW pump Trip Throttle Valve.

Answer: A

Explanation/Justification: K/A is met by the candidate demonstrating procedural knowledge of plant system lineups during a Steam Generator Tube rupture, and identifying a local valve manipulation required to isolate the ruptured SG and restore the use of the Turbine Drive AFW pump.

A. Correct. IAW E-3 step 4.c.4, 1MS-16 must be closed to isolate the ruptured SG steam supply to the TDAFW pump prior to restarting the TDAFW pump.
B. Incorrect. Plausible distractor if it is thought that the interlock between the TDAFW pump and the SG Blowdown Isolation valve would prevent the TDAFW pump from starting. The interlock actually closes TV-1BD-100A, B, C to preserve steam generator inventory when the AFW pumps start.
C. Incorrect. The MSIVs for the ruptured SG are isolated, but have no bearing on the TDAFW pump. Plausible distractor is the candidate thinks that closing the MSIVs will isolate the ruptured SG from the TDAFW pump.
D. Incorrect. The stem of the question states MOV-1MS-105 was closed thus isolating all steam to the TDAFW. This does not trip the TDAFW pump trip throttle valve. E-3 step 4.c.4 states to reopen MOV-1MS-105 after isolating 1MS-16.

Sys #	System	Category	KA Statement
000038	Steam Gen. Tube Rupture / 3	Generic	Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

K/A# 2.4.34 **K/A Importance** 4.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.E-3 Iss 3 Rev 0 pg 6

Question Source: New

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

11. The plant was operating at 100% when a Loss of ALL AC Power occurred. The following conditions exist:

    - The Emergency Diesel Generators failed to energize the Emergency busses.
    - The operating crew is conducting a secondary depressurization per ECA-0.0, Loss of All Emergency 4KV AC Power.
    - SG depressurization is in progress with the following pressures:

      1A SG: 245 psig
      1B SG: 247 psig
      1C SG: 244 psig

    Which of the following choices complete the statements below?

    Based on the above conditions, Per ECA-0.0, this secondary pressure reduction is required to ______(1)______.

    The basis for stopping the secondary pressure reduction at the specified pressure in ECA-0.0 is to prevent ______(2)______.

A. 1) be STOPPED
   2) a challenge to the Integrity Critical Safety Function

B. 1) CONTINUE
   2) a challenge to the Integrity Critical Safety Function

C. 1) be STOPPED
   2) injection of accumulator nitrogen into the RCS

D. 1) CONTINUE
   2) injection of accumulator nitrogen into the RCS

Question 11

Answer: D

Explanation/Justification: K/A is met by this question presenting a scenario where a Loss of Offsite Power occurs and the Emergency DGs fail to energize the emergency 4kv busses. The candidate is required to know the reason that the secondary depressurization is stopped at 210 psig (reasons for the actions contained in the EOP). Beaver Valley does not have Abnormal or Emergency procedure actions for a loss of offsite power, the procedure actions address a loss of the emergency busses only. Normal operating procedures or attachments are used to recover from a loss of offsite power. This is a bank question from an exam given at another site.

A. Incorrect. The pressures are not, <210 psig as required by ECA-0.0. Plausible distractor since this is the reason for the Tcold temperature limit of 330°F during the pressure reduction, but NOT the reason for stopping at 210 psig.

B. Incorrect. The SG depressurization should continue until SGs are <210 psig. Plausible distractor since this is the reason for the Tcold temperature limit of 330°F during the pressure reduction, but NOT the reason for stopping at 210 psig.

C. Incorrect: The pressures are not, <210 psig as required by ECA-0.0. To prevent injection of Nitrogen is the correct reason for stopping the SG depressurization.

D. Correct. The SG depressurization should continue until SGs are <210 psig, this reduces RCS pressure and temperature to minimize seal leakage and Inventory loss. To prevent the injection of accumulator Nitrogen into the RCS is the correct reason for stopping SG depressurization at 210 psig in ECA-0.0.

Sys #	System	Category	KA Statement
000056	Loss of Off-site Power / 6	AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power:	Actions contained in EOP for loss of offsite power

K/A# AK3.02 **K/A Importance** 4.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-0.0 Iss. 2 Rev. 1 step 24
1OM-53B.4.ECA-0.0 Iss. 2 Rev. 1 step 24

Question Source: Bank – Farley 2013 NRC Exam (Q37)

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

Objective: 1LOT-M5D21-01-05: While performing ECA-0.0 Depressurize the RCS using the S/Gs at the max rate PRA Important Operator Action # 8

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

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

- Battery Breaker #1-2 is **OPEN** for maintenance
- Vital Bus #2 Manual Bypass switch is in the ALTERNATE SOURCE TO LOAD position
- Battery Charger #2A is in service

Battery Charger #2A AC Input Breaker [B301] **TRIPS** open.

What impact will this have on the control room indications for #2 DC Bus Volts and N42 Power Range Channel?

Control room indication for #2 DC Bus Volts will ______(1)______, and N42 Power Range Channel indication will be ______(2)______.

A. 1) remain as is
2) energized

B. 1) remain as is
2) de-energized

C. 1) drop to ZERO
2) energized

D. 1) drop to ZERO
2) de-energized

Answer: C

Explanation/Justification: K/A is met by requiring knowledge of the initial DC system lineup with the battery breaker open for maintenance, and the in-service battery charger AC input breaker tripping causing a loss of #2 DC voltage. Knowledge must be used to identify that the standby battery charger must be manually aligned to supply the DC bus (No automatic swap over exists), and #2 DC volts instrumentation will indicate zero volts at this time. The candidate must also display knowledge of the status of #2 vital bus being energized and providing CR indication at this time based on the current electrical line-up.

A. Incorrect. Part 1 is wrong. The dual battery chargers will not automatically swap over (like the vital bus will swap over) when voltage drops. Part 2 is correct. Having the manual bypass switch in alternate source to load position will keep the vital bus energized which powers N42 Power Range Channel indication.
B. Incorrect. Part 1 is wrong. The dual battery chargers will not automatically swap over (like the vital bus will swap over) when voltage drops. The Part 2 is wrong. Having the manual bypass switch in alternate source to load position will keep the vital bus energized supplying N42 PR indication.
C. Correct. The dual battery chargers will not automatically swap over (like the vital bus will swap over) when voltage drops. Therefore the DC bus will be de-energized. Having the manual bypass switch in alternate source to load position will keep the vital bus energized which powers N42 Power Range Channel indication.
D. Incorrect. Part 1 is correct. Part 2 is wrong. Having the manual bypass switch in alternate source to load position will keep the vital bus energized supplying N42 PR indication.

Sys #	System	Category	KA Statement
000058	Loss of DC Power / 6	AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power:	Battery charger equipment and instrumentation

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

References provided to Candidate None **Technical References:** LP 3SQS-38.1 rev. 8 Unit 1 ppt slide 28
1OM-39.4.AAL rev.6 page 2 of 5 step 5
LP 3SQS-39.1 rev. 9 Unit 1 ppt slide 7

Question Source: Bank – 1LOT14 NRC Exam Q13 Modified

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

Objective: 3SQS-39.1 Rev. 9 Obj. 10 Given a 125 VDC Distribution System configuration, and without reference material, describe the 125 VDC Distribution System field response to the following malfunctions, including automatic functions and changes in equipment status.
a. Loss of AC power, b. Loss of Station Battery, c. Loss of DC Bus

13. The plant is in Mode 3 with all systems in normal alignment for plant startup.
    - 1WR-P-1A, 'A' River Water Pump is RUNNING
    - 1WR-P-1B, 'B' River Water Pump is in STBY

    Which of the following system indications below describe the Reactor Plant River Water System lineup **five minutes** after an inadvertent **Train 'A' CIB** actuation?
    (Assume NO Operator action)

	'A' RPRW PUMP	'B' RPRW PUMP	RSS HX FLOWRATE	CCR HX FLOWRATE
A.	Running	Off	Pre-event	Pre-event
B.	Off	Running	Increase	Decrease
C.	Running	Running	Increase	Decrease
D.	Running	Running	Pre-event	Pre-event

Answer: C

Explanation/Justification: K/A is met by the candidate analyzing the effects of an inadvertent Train 'A' CIB actuation will have on the Reactor Plant River Water System. When the inadvertent Train 'A' CIB actuation occurs, a loss of River Water flow will occur to the CCR heat exchangers, and flow will increase to the RSS heat exchangers because the inlet valve will open. .When the CCR Hxs are isolated, a low pressure condition on the header will cause an auto start of the 'B' RPRW pump.

A. Incorrect. Both RPRW pumps will be running, and RSS Hx flowrate will increase, and CCR Hx flowrate will decrease to zero. Plausible distractor if the candidate thinks both trains of CIB must be actuated for the RPRW system to realign.
B. Incorrect. Plausible distractor if the candidate think that the 'A' RPRW pump trips on a Train 'A' CIB actuation, causing 'B' RPRW pump to auto start. RSS Hx flowrate does increase, and CCR Hx flowrate does decrease to zero.
C. Correct. Train A CIB will close MOV-1RW-114A & B which will isolate both trains of RPRW to the CCR Hxs causing CCR Hxs flowrate to decrease to zero. This will cause a low pressure condition on PS-1RW-113 A & B which causes an auto start of the 'B' RPRW pump. MOV-1RW-103A & B will open supplying RW to the RSS Hxs causing the flowrate to increase.
D. Incorrect. Both RPRW pumps will be running, but RSS Hx flowrate will increase, and CCR Hx flowrate will decrease to zero. Plausible distractor if the candidate thinks both trains of CIB must be actuated for the RPRW system to realign.

Sys #	System	Category	KA Statement
000062	Loss of Nuclear Svc Water / 4	AA1. Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water (SWS):	Flow rates to the components and systems that are serviced by the SWS; interactions among the components

K/A# AA1.07 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** UFSAR Fig. 7.2-1 Rev. 22
Unit 1 LSK-017-001E Rev. 7
1OM-30.1.D Iss. 4 Rev. 3 pg. 5
1SQS-30.2 PPNT Rev. 18 Slide 47

Question Source: New

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

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. b. Containment Isolation Signal – Phase B (CIB)

14. The crew has entered AOP 1.34.2, Loss of Containment Instrument Air due to a large air leak inside containment.
    - TV-1IA-400, Inst Air to CNMT Inst Air Isol valve has been CLOSED
    - 1IA-90, Bypass Valve for TV-1IA-400 is CLOSED
    - CNMT Inst Air pressure is 60 psig and lowering

    The leak has been repaired and air must be restored to containment.

    1) Based on the above conditions, what actions must be taken to restore Instrument Air to Containment IAW AOP 1.34.2?

    2) When restoring Instrument Air to Containment, why must the air to containment be restored slowly?

A. 1) Open TV-1IA-400 ONLY
   2) Main Steam Trip Valves may CLOSE

B. 1) Open 1IA-90 then Open TV-1IA-400
   2) Main Steam Trip Valves may CLOSE

C. 1) Open TV-1IA-400 ONLY
   2) CNMT Instrument Air Receiver Relief Valve may LIFT

D. 1) Open 1IA-90 then Open TV-1IA-400
   2) CNMT Instrument Air Receiver Relief Valve may LIFT

Answer: B

Explanation/Justification: K/A is met by evaluating the ability to restore Station instrument Air to Cnmt Instrument Air after it has been isolated iaw the Loss of Instrument Air AOP and the loss of IA has been corrected so that rapid depressurization of IA does not cause the MSIVs to close.

A. Incorrect. Plausible to only open TV-1IA-400 if Cnmt IA header pressure was ≥ 95 psig, but the stem stated pressure was 60 psig. Therefore, equalization around TV-1IA-400 must be done slowly to prevent closure of the MSIVs. (Caution in the AOP). Second part is correct.

B. Correct. With Cnmt IA header pressure at < 60 psig, 1IA-90 is opened to slowly equalize around TV-1IA-400 to prevent closure of the MSIVs. Second part correct because AOP-1.34.2 has a caution stating attempting to restore Instrument Air to CNMT by immediately opening TV 1IA 400 or by opening 1IA-90 too quickly may reduce Station Instrument Air pressure enough to close a Main Steam Line Trip Valve.

C. Incorrect. Plausible to only open TV-1IA-400 if Cnmt IA header pressure was ≥ 95 psig, but the stem stated pressure was 60 psig. Therefore, equalization around TV-1IA-400 must be done slowly to prevent closure of the MSIVs. Second part is plausible if the candidate thinks that supplying air to the cnmt air receiver too quickly will cause the 120 psig relief valve to lift.

D. Incorrect. First part is correct. Second part is plausible if the candidate thinks that supplying air to the cnmt air receiver too quickly will cause the 120 psig relief valve to lift.

Sys #	System	Category	KA Statement
000065	Loss of Instrument Air / 8	AA1. Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air:	Restoration of systems served by instrument air when pressure is regained

K/A# AA1.03 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.34.2 Rev. 9 pg. 2, 3, 5, 10

Question Source: New

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

Objective: 1SQS-53C.1, Rev. 12 Obj. 4 - Explain the basis for cautions and notes of each AOP.
1SQS-53C.1, Rev. 12 Obj. 5 - Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.

15. Given the following conditions:
    - The plant was at 100% power
    - A Loss of Coolant Accident (LOCA) outside containment results in RCS Subcooling dropping to 0°F
    - The crew is performing the actions of ECA-1.2, LOCA Outside Containment
    - Actions are being taken to isolate the leak

    In accordance with ECA-1.2, which of the following flowpaths are isolated, and what indication is used to confirm the LOCA has been isolated?

    The ______(1)______ pumps are isolated from the RCS.

    ______(2)______ rising will be used to confirm the LOCA has been isolated.

A. 1) Low Head Safety Injection
   2) Pressurizer level

B. 1) Low Head Safety Injection
   2) RCS pressure

C. 1) Outside Recirculation Spray
   2) Pressurizer level

D. 1) Outside Recirculation Spray
   2) RCS pressure

Answer: B

Explanation/Justification: K/A is met with the knowledge of the procedural steps of the LOCA Outside Containment procedure, as well as having knowledge of which low pressure piping which is isolated, and how it is determined that the leak is isolated by checking RCS pressure.

A. Incorrect. LHSI is correct. Pressurizer level is a plausible distractor because it is one of the indications used in the EOP network to determine whether SI flow is required

B. Correct. Low Head Safety Injection is specifically call out in step 2, as a low pressure flowpath which could be the cause of the LOCA outside cnmt. RCS pressure is the correct indication because if the break is isolated, a significant RCS pressure increase will occur due to the SI flow filling up the RCS with break flow stopped.

C. Incorrect. Outside RSS pumps are a plausible distractor since they are located outside cnmt and in other circumstances, could be aligned for cold leg recirc if the LHSI pumps are not available. This requires local operation of manual valves which would not be operated in E-1 or ECA-1.2. Pressurizer level is a plausible distractor because it is one of the indications used in the EOP network to determine whether SI flow is required.

D. Incorrect. Outside RSS pumps are a plausible distractor since they are located outside cnmt and in other circumstances, could be aligned for cold leg recirc if the LHSI pumps are not available. This requires local operation of manual valves which would not be operated in E-1 or ECA-1.2. RCS pressure is correct.

Sys #	System	Category	KA Statement
W/E04	LOCA Outside Containment / 3	EK3. Knowledge of the reasons for the following responses as they apply to the (LOCA Outside Containment)	Normal, abnormal and emergency operating procedures associated with (LOCA Outside Containment).

K/A# EK3.2 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-1.2 Iss. 2 rev. 1 pg. 1-3
RM-0411-001 Rev. 27
RM-0411-002 Rev. 13

Question Source: Bank – Sequoyah 2012 NRC Exam (Q15)

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

16. The following conditions exist:

- The crew has entered FR-H.1, Response to Loss of Secondary Heat Sink
- A8-115, 4160V Emergency Bus 1DF ACB 1F7 Overcurrent Trip is LIT
- A7-7, Steam Unavailable to Turbine Driven Feed Pump is LIT
- 'A' MDAFW tripped on startup
- Dedicated AFW pump failed to start
- All RCPs have been secured
- SG levels A – 14% WR
  B – 11% WR
  C – 13% WR

1) What is the next required action of FR-H.1?

2) When feedwater is restored, what is the MAXIMUM feed flowrate to the 'B' Steam Generator in accordance with FR-H.1?

A. 1) Attempt to start 1 MFP
2) up to 50 gpm

B. 1) Attempt to start 1 MFP
2) up to 100 gpm

C. 1) Initiate Bleed and Feed
2) up to 50 gpm

D. 1) Initiate Bleed and Feed
2) up to 100 gpm

Answer: D

Explanation/Justification: K/A is met by placing the candidate into FR-H.1 with various annunciators requiring the candidate demonstrate knowledge that other means of feed flow are unavailable, and Bleed and Feed requirements have been met. The question also tests the remedial actions of limiting feed to a dry (≤14%) to ≤100gpm.

A. Incorrect. SG WR levels are ≤14%, therefore bleed and feed is required. 50 gpm is a plausible distractor since it is a low volume, but it is the required flowrate for ECA-2.1, Uncontrolled depressurization of All SGs.

B. Incorrect. SG WR levels are ≤14%, therefore bleed and feed is required. It is correct that the feed rate to a dry SG (14%WR) is ≤100gpm.

C. Incorrect. SG WR levels are ≤14%, therefore bleed and feed is required. 50 gpm is a plausible distractor since it is a low volume, but it is the required flowrate for ECA-2.1, Uncontrolled depressurization of All SGs.

D. Correct. With all 3 SGs ≤14% FR-H.1 continuous action step 3 requires Bleed and Feed to be performed. If 2 of the SGs were >14%, then starting the MFP would be appropriate. It is correct that the feed rate to a dry SG (14%WR) is ≤100gpm.

Sys #	System	Category	KA Statement
W/E05	Inadequate Heat Transfer - Loss of Secondary Heat Sink / 4	EK1. Knowledge of the operational implications of the following concepts as they apply to the (Loss of Secondary Heat Sink)	Annunciators and conditions indicating signals, and remedial actions associated with the Loss of Secondary Heat Sink

K/A# EK1.3 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-H.1 Iss 2 Rev 2 pg. 2, 13, &19

Question Source: New

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

Objective: 1LOT-M5D12, Rev. 15 Obj 2-5 Given a Loss Of Heat Sink with Bleed and Feed Criteria met, conduct a RCS Bleed and Feed in accordance with FR-H.1 Loss of Secondary Heat Sink. Obj 2-6 Given a Loss Of Heat Sink with Dry SG Criteria met, feed the Steam Generators in accordance with FR-H.1.

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

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

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

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

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

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

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

Answer: A

Explanation/Justification: K/A is met by the knowledge of the actions taken in ECA-1.1, Loss of Emergency Coolant Recirculation to reduce the RCS leakage. This includes operation of the High Head Safety Injection system to reduce to the RWST outflow, and the effect that depressurizing the RCS will have on the RCS leak rate. Both of these actions will help to delay depletion of the RWST and minimize break flow.

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

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

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

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

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

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

References provided to Candidate None **Technical References:** 1OM-53B.4.ECA-1.1 Iss. 3 Rev. 0 pgs. 3 & 47

Question Source: Bank – 1LOT7 NRC Exam (Q65) Modified

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

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

18. Initial conditions:

- Plant was operating at 100% power with all systems in normal alignment
- All 3 steam lines ruptured in the Main steam Valve area upstream of the MSIVs

Based on the above accident, which of the following coincidence and logics will generate the Reactor Trip signal?

A. 2/3 Steamline Pressure Low on 1/3 SG

B. 2/3 High Steamline Pressure Rate on 2/3 SG

C. 2/4 Power Range High

D. 2/3 HI-HI SG level on 1/3 SG

Answer: A

Explanation/Justification: K/A is met demonstrating knowledge of the expected Rx trip and E-0, Reactor Trip and Safety Injection entry which would occur due to an uncontrolled depressurization of all 3 SGs. The candidate will have to demonstrate knowledge of the P-11 interlock, and determine that at NOP (>P-11) the plant will trip due to steamline pressure low, which is one of the automatic actuations of Safety Injection which is a Rx trip. PR high at 109% will not cause the Rx trip due to the rods inserting with rod control in automatic. Actual trip setpoints were not used in the answer choices to ensure no other exam question was compromised.

A. Correct. Based on the above conditions, the SI initiation signal of 2/3 Steamline Pressure Low on 1/3 S/G at 500 PSIG + rate will trip the reactor.
B. Incorrect. Plausible distractor because 2/3 High Steamline Pressure Rate on 1/3 SG MSLI is instated when MSLI 2/3 Steamline Pressure Low on 1/3 S/G is blocked when <P-11 (2000 psig), therefore pressure rate will not cause a reactor trip with these conditions.
C. Incorrect. Plausible because this is a reactor trip which could trip the reactor based on these conditions if Rod Control was in Manual. The initial conditions state normal alignment for this power.
D. Incorrect. Plausible distractor because swell will occur in the SG, but this coincidence is for a FWI which will not occur based on these conditions, although a FWI would cause the reactor to trip, this is not the cause.

Sys #	System	Category	KA Statement
W/E12	Uncontrolled Depressurization of all Steam Generators	Generic	Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

K/A# 2.4.2 **K/A Importance** 4.5 **Exam Level** RO

References provided to Candidate None **Technical References:** UFSAR Logic Diagrams Figure 7.2-1 sh. 7 & 8
1OM-1.5.B.1, Rev. 3, pg. 2
1OM-1.5.B.4 rev. 17 pg. 2

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7 / 45.7 / 45.8)

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

19. A fuel assembly has been dropped in Containment during core off-load.

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

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

A. 1) RM-1RM-219A, Containment High Range Area Monitor
   2) Control Room

B. 1) RM-1RM-219A, Containment High Range Area Monitor
   2) CNMT Personnel Air Lock

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

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

Answer: C

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

A. Incorrect. Plausible distractor because high radiation in CNMT would be just cause for evacuating, but 1RM-219A does not provide input to the automatic CNMT evacuation alarm. CR is the correct location for manual actuation.

B. Incorrect. Plausible distractor because high radiation in CNMT would be just cause for evacuating, but 1RM-219A does not provide input to the automatic CNMT evacuation alarm. CNMT Personnel Air Lock is incorrect. It is a plausible distractor because the PAL is a central location for communications, and contains a red revolving warning light for incore moveable detector movement.

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

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

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

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

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

Question Source: New

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

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

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

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

20. On the RADAREA IPC printout, which of the following Area Radiation Monitor indications are **NOT** NORMAL when the plant is operating at 100% power?

SEE ATTACHED RADAREA IPC PRINTOUT

1) RM-1RM-201, Containment High Range Area Monitor
2) RM-1RM-204, Incore Instrument Transfer Device Area Monitor
3) RM-1RM-207, Fuel Pool Bridge Area Monitor
4) RM-1RM-218A and 218B, Control Room Area Gamma Radiation Monitors

A. 1 and 4

B. 2 and 3

C. 1 and 3

D. 2 and 4

Answer: B

Explanation/Justification: K/A is met by the candidate evaluating the Radiation Monitor In Plant Computer printout to determine which of the area radiation monitors are indicating normal radiation intensity, and which radiation monitors are in an alarm or an out of service condition.

A. Incorrect. RM-1RM-201 is a plausible distractor if the candidate does not know the normal radiation levels for containment. 1.0E-01 is normal for this detector. RM-1RM-218A and 218B - plausible distractor because it is a normal plant value, but it is also just below the High Rad alarm of 0.357 mr/hr.

B. Correct. RM-1RM-204 is in high alarm at 23 mr/hr. (Setpoint is 20 mr/hr). This is not a normal condition during plant operations. RM-1RM-207 indicates out of service which is not normal for plant operations. LR 3.1.11 requires RM-207 to be functional when fuel is in the spent fuel pool.

C. Incorrect. RM-1RM-201 is a plausible distractor if the candidate does not know the normal radiation levels for containment. 1.0E-01 is normal for this detector. Second part is correct. RM-1RM-207 indicates out of service which is not normal for plant operations.

D. Incorrect. First part is correct. RM-1RM-204 is in high alarm at 23 mr/hr. (Setpoint is 20 mr/hr). This is not a normal condition during plant operations. Second part is not correct. RM-1RM-218A and 218B - plausible distractor because it is a normal plant value, but it is also just below the High Rad alarm of 0.357 mr/hr.

Sys #	System	Category	KA Statement
000061	ARM System Alarms / 7	AA2. Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms:	Normal radiation intensity for each ARM system channel

K/A# AA2.02 **K/A Importance** 2.9 **Exam Level** RO

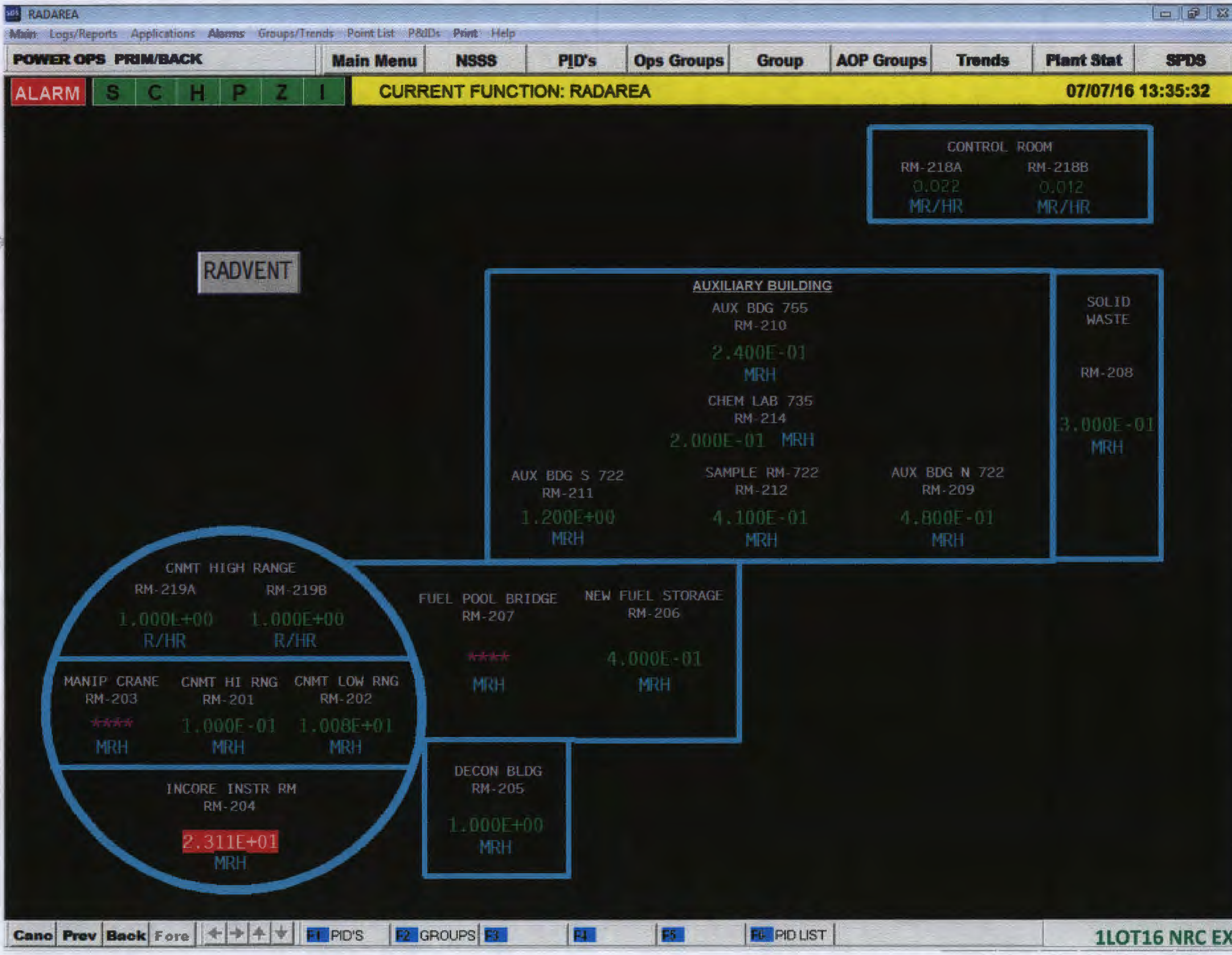
References provided to Candidate RADAREA IPC PRINTOUT **Technical References:** Unit 1 LRM Rev. 71 pg. 3.3.11-1
SPD-RM-1RM-204 Rev. 00 (setpoint document)
1OM-54.3.L5 Rev. 88 pgs. 33 & 37

Question Source: New

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

Objective: 3SQS-54.1, Rev. 7 Obj. 6 - Describe the functions of the Surveillance (L-5) logs, and the logs role in verifying compliance with requirements, as documented in Operating Manual chapter 1OM-54, 2OM-54, and NOP-OP-1002 and in accordance with the applicable Units' Licensing Documents (e.g. Technical Specifications, License Requirement Manual, Offsite Dose Calculation Manual, and 1/2ADM-1900 items).

3SQS-54.1, Rev. 7 Obj. 7 - Given a datalogger and a PC, be able to perform this additional function: Enter Shift Plant Parameters



1LOT16 NRC EXAM
Q20

21. Given the following plant conditions:

- AOP 1.33.1A, Control Room Inaccessibility has just been entered
- PRZR level is 17% and slowly LOWERING

In accordance with AOP 1.33.1A, from which of the following locations will FCV-1CH-122, Charging Flow to Regen HX Inlet Control Valve be operated to throttle charging flow?

A. Locally

B. Control Room

C. Backup Indicating Panel

D. Emergency Shutdown Panel

Answer: D

Explanation/Justification: K/A is met by placing the candidate into a situation which requires entry into the Control Room Evacuation AOP, and based on this entry, display the ability to operate the RCS charging header FCV from the Emergency Shutdown Panel.

A. Incorrect. Plausible distractor but FCV-1CH-122 is capable of being operated from the SDP, and local operation is not referenced in AOP-1.33.1A. Local operation only allows for full open or full closed valve positions.

B. Incorrect. Plausible distractor, but the FCV-1CH-122 is not one of the components operated prior to the CR evacuation.

C. Incorrect. Plausible distractor with the BIP having several control and indications available, but FCV-1CH-122 is not operated from the BIP, nor is the BIP referenced in AOP-1.33.1A. BIP is used iaw 1OM-56C for a large fire in the CR.

D. Correct. AOP-1.33.1A references the operation of FCV-1CH-122 from the SDP by placing the control selector switch in SDP, then controlling PRZR level between 22-54%.

Sys #	System	Category	KA Statement
000068	Control Room Evac. / 8	AA1. Ability to operate and / or monitor the following as they apply to the Control Room Evacuation:	Flow control valve for RCS charging header

K/A# AA1.22 **K/A Importance** 4.0 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.33.1A Rev 13 pg. 10 & 11

Question Source: New

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

Objective: 1SQS-45.1-01: From memory and/or using supplied reference materials, explain the Miscellaneous Safety-Related Systems design basis, major components, flow paths, controls and interlocks, transient response, and normal & abnormal operations, in accordance with established design basis, station policies, technical specifications and procedures.

22. 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 L/D rad monitor is in alarm.
Beaver Valley does not have any actions for High Reactor Coolant Activity contained in the EOPs, therefore the question was directed towards the High Reactor Coolant Activity AOP which is the procedure BV would use.

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 / 9	AK3. Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity :	Actions contained in EOP for high reactor coolant activity

K/A# AK3.06 **K/A Importance** 3.2 **Exam Level** RO

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

Question Source: New

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.

23. Initial conditions:
    - Plant is at 100% power
    - SSST 1A is on Clearance for relay maintenance

    Current Conditions:
    - 'B' RCP tripped due to a locked rotor
    - Safety Injection actuated on the Rx Trip
    - The crew has transitioned to ES-1.1, SI Termination and are attempting to start an RCP

    In reference to starting a RCP in accordance with ES-1.1, based on the above conditions, which of the following statements is correct and why?

A. Start 'C' RCP to provide normal PRZR spray flow.

B. Start 'A' RCP to provide normal PRZR spray flow.

C. Verify Natural Circulation because no RCPs can be started.

D. Reinitiate Safety Injection because no RCPs can be started.

Answer: A

Explanation/Justification: K/A is met by demonstrating knowledge of the note referring to RCP priority for PRZR spray in ES-1.1, SI Termination, and with the knowledge to determine the 4KV electrical lineup status for available RCP power supplies.

A. Correct. Starting 'C' RCP is the priority for PRZR spray in accordance with the note preceding the step. The candidate must determine that the pump tripped due to the A and B pumps receiving an under frequency, and causing the C RCP to trip also. The auto bus transfer will reenergize the C and D normal 4KV busses after the Rx trip, therefore power will be available to restart C RCP.

B. Incorrect. This is plausible distractor because the note in SI term states to start C, A, B RCPs in that priority to provide normal spray flow, but with SSST 1A on clearance for relay maintenance, there is no power available to the normal 4KV 'B' bus.

C. Incorrect. Plausible distractor because if no RCPs can be started, this would be the correct action in accordance with the RNO of the RCP starting step of ES-1.1. But, the 'C' bus will be energized in this case and it is the priority pump for PRZR spray.

D. Incorrect. There is no reason to assume that SI re-initiation criteria of ES-1.1 LHP have been met since the SI was due to a locked RCP rotor, and no conditions pertaining to PRZR level or subcooling were given in the stem. Plausible if the candidate thinks this is the only means of core cooling.

Sys #	System	Category	KA Statement
W/E02	SI Termination / 3	Generic	Knowledge of 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.ES-1.1 Iss. 3 rev. 0 pg. 13
3SQS-36.1 PPNT Rev 12 Iss 1 slide 10

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

3SQS-36.1, Rev. 12 Obj. 16 - Given a specific plant condition, predict the response of the 4KV Distribution System control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off-normal condition.

24. Given the following plant conditions:

- The STA reports a Yellow path on **CORE COOLING** exists
- The Unit Supervisor announces a transition to FR-C.3, Response to Saturated Core Cooling
- 'C' Reactor Coolant Pump (RCP) is running

Which of the following is a mitigating strategy in FR-C.3?

A. Stop 'C' RCP and open all RCS vent paths.

B. Depressurize SGs to depressurize the RCS.

C. Start 'A' & 'B' RCPs and open all RCS vent paths.

D. Establish SI flow to maintain minimum RCS subcooling.

Answer: D

Explanation/Justification: K/A is met by the candidates knowledge of the interrelation of the plants emergency coolant (SI) requirements during a Saturated Core Cooling event and the importance of operating Safety Injection to maintain minimal RCS subcooling.

A. Incorrect. Opening RCS vents is a major action category for FR-C.1. It is plausible that the running RCP is stopped before reducing RCS pressure to minimum. In the situation of FR-C.3 Saturated Core Cooling, the procedure requires the vent paths to be closed.
B. Incorrect. This is a major action category for FR-C.1 & 2, but not for FR-C.3.
C. Incorrect. This is one of the major action categories for FR-C.1 versus FR-C.3. FR-C.3 checks for open paths and requires open vent paths to be closed to minimize inventory loss.
D. Correct. A major action category for FR-C.3 is to establish SI flow and maintain minimum RCS subcooling.

Sys #	System	Category	KA Statement
W/E07	Saturated Core Cooling / 4	EK2. Knowledge of the interrelations between the (Saturated Core Cooling) and the following:	Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility

K/A# EK2.2 **K/A Importance** 3.5 **Exam Level** RO

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

Question Source: Bank – 2LOT8 NRC Exam (Q24)

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

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

25. Initial Conditions:
    - A reactor trip occurred from 100% power
    - GV-1 and TV-4 have failed OPEN
    - E-0 IOAs were completed, and the crew transitioned to ES-0.1, Reactor Trip response

The STA reported a **YELLOW** condition on the Heat Sink status tree, and the crew transitioned to FR-H.2, Response To Steam Generator Overpressure.

The following conditions exist:
- 'A' SG pressure indicates 1150 psig
- 'B' and 'C' SG pressures indicate 1010 psig
- 'A' SG NR level is 79%
- No ruptured SG exists

Which of the following actions will be required to mitigate the SG overpressure condition?

A. Initiate AFW flow to the 'A' SG to facilitate cooldown.

B. Open condenser steam dump valves to reduce SG pressures.

C. Go to FR-H.3, Response to SG High Level to reduce pressure by reducing SG level.

D. Operate the residual heat release valve to reduce SG pressure.

Answer: D

Explanation/Justification: K/A is met demonstrating the knowledge of which components are available to depressurize the overpressure SG when automatic equipment failures such as main turbine trip occurs, and the crew has to manually Main Steam Isolate to minimize cooldown after a Rx trip.

A. Incorrect. Plausible distractor to think that AFW will cause a cooldown and SG pressure will lower, but in fact AFW is isolated to the affected SG due to it being a potential source of SG pressurization, and SGWL being high.

B. Incorrect. Plausible distractor because this would be the primary choice for dumping steam, but the initial conditions stated that GV-1 & TV-4 failed open, therefore the crew would have to isolate the MSIVs to prevent an SI, and continue in ES-0.1.

C. Incorrect. Plausible distractor with 89% NR SG being an entry into FR-H.3, and that step 3 of FR-H.2 checks affected SG level <96% directing a transition to FR-H.3.

D. Correct. With no ruptured SG, using the RHR control valve is the appropriate action iaw FR-H.2 step 4 under the major action category of controlling the affected SG pressure based on the current plant conditions.

Sys #	System	Category	KA Statement
W/E13	Steam Generator Over-pressure / 4	EK2. Knowledge of the interrelations between the (Steam Generator Overpressure) 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** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-H.2 Iss. 3 Rev. 0 pg. 1; 3SQS53.3 FRH Series Att. 4 Rev.2

Question Source: Bank - BV2 2005 NRC Exam Q64 (Modified)

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

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

26. The plant is at 75% power with all systems in normal alignment for this power level.

- A Large Break LOCA occurred
- E-1, "Loss of Reactor or Secondary Coolant" is in progress
- Containment pressure peaked at 38 psig and is now 9 psig and SLOWLY LOWERING
- RWST Level is 29 ft. 5 in. and LOWERING
- Assume no operator action related to equipment stated below

Based on the above conditions, which statement below identifies the equipment which will be operating at this time for reducing containment pressure?

A. Recirc Spray pumps **ONLY**.

B. Quench Spray pumps **ONLY**.

C. Recirc Spray **AND** Quench Spray pumps **ONLY**.

D. Quench Spray **AND** Containment Air Recirculation Fans **ONLY**.

Answer: B

Explanation/Justification: K/A is met by the candidate recognizing the cnmt and RWST conditions, and the fact that CIB has actuated causing ONLY the Quench Spray pumps to start to meet their intended function of reducing cnmt pressure. The candidate must also recognize that with the volume available in the RWST, the Recirc Spray pumps have not started at the current level.

A. Incorrect. Recirc Spray pumps will not start until RWST level reaches 27' 7.5" along with the CIB signal.
B. Correct. Only the Quench Spray pumps will be operating due to the CIB actuation.
C. Incorrect. QS Pumps will be running, but Recirc Spray pumps will not start until RWST level reaches 27' 7.5" along with the CIB signal.
D. Incorrect. QS Pumps will be running, but the CAR fans will not be operating due the 480VAC bus supply breakers opening on the CIB signal.

Sys #	System	Category	KA Statement
W/E14	Loss of CTMT Integrity / 5	EK1. Knowledge of the operational implications of the following concepts as they apply to the (High Containment Pressure)	Components, capacity, and function of emergency systems

K/A# EK1.1 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-13.1.D Rev 5 pg. 3

Question Source: Bank – 2LOT8 NRC Exam (Q27) Modified

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

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

27. Given the following plant conditions:

- A Loss of Coolant Accident (LOCA) occurred.
- Safety Injection was lost and containment radiation level increased to 3E+5 R/hr.
- Safety Injection has been re-established and containment radiation is now 2E+3 R/hr and trending DOWN.

Which of the following describes the correct use of Adverse Containment parameter values for this event?

A. **NOT** required during this transient.

B. Required as soon as the dose rate limit was exceeded, but are no longer required because the dose rate is now below the limit.

C. Required as soon as the dose rate limit was exceeded, and remain in effect for the duration of the event because total integrated dose is unknown.

D. Required as soon as the dose rate limit was exceeded, and remain in effect for the duration of the event, because since the dose rate was exceeded, the integrated dose rate was also exceeded.

Answer: C

Explanation/Justification: K/A is met by requiring the candidate to interpret the containment radiation values at various timelines of the event, and determine whether adverse conditions would still be applicable in the EOP network to operate within the limitations in the facility's license and amendments.

A. Incorrect. Containment Radiation levels exceeded 1E + 5 R/hr, so therefore adverse parameters are required.
B. Incorrect. Although it is true that the limit of 1E + 5 R/hr was exceeded and also true that the radiation levels are now below this limit, 1OM-53B.5.GI-2 requires that integrated dose remained less than 1E +6 R/hr. This value is not known in the stated plant conditions and until it is known, the operator must continue to use adverse parameters.
C. Correct. IAW 1OM-53B.5.GI-2, and in conjunction with justifications above.
D. Incorrect. There is no way of determining if integrated dose was exceeded based on stated plant conditions. Additionally, it is not true that whenever dose rate is exceeded that the integrated dose is exceeded.

Sys #	System	Category	KA Statement
W/E16	High Containment Radiation / 9	EA2. Ability to determine and interpret the following as they apply to the (High Containment Radiation)	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

K/A# EA2.2 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53B.5.GI-2 Iss. 2 Rev. 0 pgs. 13 & 14

Question Source: Bank - 2LOT7 NRC Exam Q27

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

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

28. The plant was operating at 100% power.
    - 1CC-P-1A CS is in AFTER-START
    - 1CC-P-1B is on clearance for maintenance
    - 1CC-P-1C is racked on to DF bus with the CS in AFTER-STOP
    - An inadvertent Reactor Trip has occurred
    - During the transfer to Off-site power, SSST 1B received a differential protection alarm

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

____(1)____ Reactor Plant Component Cooling Water pump(s) will be running.

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

A. 1) One
   2) No

B. 1) One
   2) Two

C. 1) Two
   2) No

D. 1) Two
   2) Two

Answer: D

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

A. Incorrect. Both parts are plausible if the candidate assumes that SSST 1B supplies 'A' & "B' Normal 4KV busses. Therefore the 'A' CCR pump losses power, and they recognizes that the 'C' CCR pump will auto start. Second part is plausible if they think that SSST 1B supplies 'A' & "B' Normal 4KV busses which would cause 2 RCPs to trip on undervoltage/under frequency. In this case, the third RCP would also trip due to 2/3 RCP under frequency trip.

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

C. Incorrect. Two CCR pumps will be running. Second part plausible if candidate thinks that SSST 1B supplies 'A' & "B' Normal 4KV busses which would cause 2 RCPs to trip on undervoltage/under frequency. In this case, the third RCP would also trip due to 2/3 RCP under frequency trip.

D. Correct. SSST 1B supplies the C & D Normal 4KV busses. This will cause 'C' RCP to trip on undervoltage due to the loss of 'C' bus, leaving 'A' & 'B' RCPs running. It will also cause #2 EDG to auto start due to undervoltage on 'DF' bus. When the #2 EDG sequences, the 'C' CCR pump will auto start, and 2 CCR pumps will be running. The candidate must know that a SSST 1B differential protection alarm will cause the 'C' & 'D' busses to de-energize due to an internal transformer fault.

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

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

References provided to Candidate None **Technical References:** 1OM-36.4.AAE Rev. 1 pg. 2, 1OM-6.3.C Rev. 11 pg. 8; 1OM-15.1.D Iss. 4 Rev. 1 pg. 3; 3SQS36.1 PPNT Unit 1 Rev 12 Iss. 1 slide 10

Question Source: New

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

Objective: 3SQS-36.1, Rev. 12 Obj. 16 - Given a specific plant condition, predict the response of the 4KV Distribution System control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off-normal condition.

1SQS-15.1, Rev. 14 Obj. 3 - Describe the electrical configuration of the CCR pumps and the special considerations that are required if that configuration is changed.

29. The crew has entered AOP 1.6.8, Abnormal RCP Operation due to A3-109, Reactor Cool Pump 1B Seal Vent Pot Level High alarm.

The following indications exist:

- VCT Pressure, PI-1CH-117 is 27 psig and STABLE
- Primary Drains Transfer Tank level, LI-1DG-107A is 20.1 inches and slowly RISING
- RCP 1B Seal Leak Off, FR-1CH-154A is 1.21 gpm and STABLE

Based on these alarms and indications, which 'B' RCP seal has failed?

A. #1 seal

B. #2 seal

C. #3 seal

D. Low pressure seal

Answer: B

Explanation/Justification: K/A is met because the candidate must analyze both Control Room indications and annunciators of RCP seal leakage to determine which RCP seal has failed.

A. Incorrect. If #1 seal had failed seal leak-off flow would be high NOT low. Added VCT pressure to ensure that normal backpressure was felt on #1 seal leakoff.
B. Correct. With seal vent pot level high, Seal leakoff indicating lower than normal (~2.8 gpm), and TK1 level rising, these all indicate a failure of #2 seal.
C. Incorrect. If #3 seal had failed the seal vent pot level low would be indicated, NOT high.
D. Incorrect. The low pressure seal is not functional when the motor is coupled to the pump.

Sys #	System	Category	KA Statement
003	Reactor Coolant Pump	A4 Ability to manually operate and/or monitor in the control room:	RCP seal leakage detection instrumentation

K/A# A4.05 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.4.ABG Rev 6
1SQS6.3 PPNT Rev 15 Iss 1 slide 43
1OM-53C.4.1.6.8 Rev. 18 pg. 29

Question Source: Bank – 2LOT15 (Q3)

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

Objective: 1SQS-6.3, Rev. 15 Obj. 21. Given a change in plant conditions due to system or component failure, analyze the Reactor Coolant Pump and support system to determine what failure has occurred.

30. The plant is operating at 100% power when the air supply to FCV-1CH-122, CHG Flow Regen Hx Inlet Control Valve actuator is severed.

Which of the following completes the statement below?

FCV-1CH-122 ______(1)______, and RCP seal injection flow will ______(2)______.

	(1)	(2)
A.	closes	decrease
B.	closes	increase
C.	opens	decrease
D.	opens	increase

Answer: C

Explanation/Justification: K/A is met with the knowledge of the effects of a severed air line malfunction on FCV-1CH-122, charging line flow control valve, and the affects it will have on the RCP seal injection flowrate.

A. Incorrect. First part is incorrect. FCV-1CH-122 fails open. Plausible if the candidate does not recall the fail position of FCV-1CH-122. Second part is correct. Logical connection to the first part if the applicant believes that FCV-1CH-122 is upstream of the seal inj. line and the closure of FCV-1CH-122 would stop seal injection flow.

B. Incorrect. First part is incorrect. FCV-1CH-122 fails open. Plausible if the applicant does not recall the fail position of FCV-1CH-122. Second part is incorrect. Logical connection to the first part if the applicant thought that FCV-1CH-122 failed closed since it would be the correct seal inj. response for this condition.

C. Correct. FCV-1CH-122 does fail open. With FCV-1CH-122 being located downstream of the Seal inj. supply line, it will reduce flow to the RCP seal inj.

D. Incorrect. First part is correct, but the second part is incorrect. Logical connection to the first part if the applicant assumes that more charging flow equates to more seal injection flow.

Sys #	System	Category	KA Statement
004	Chemical and Volume Control	K3 Knowledge of the effect that a loss or malfunction of the CVCS will have on the following:	RCP seal injection

K/A# K3.08 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** U1 RM-0407-001 Rev. 34, 1OM-7.4.IF rev. 3 pg. 12

Question Source: Bank – Farley 2013 NRC Exam (Q3)

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7/45/6)

Objective: 1SQS-7.1, Rev. 20 Obj. 20. Given a Chemical and Volume Control System configuration and without referenced material, describe the Chemical and Volume Control 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

31. The plant is in Mode 4 with RCS temperature at 210°F.

- RHR HX 1RH-E-1A is on clearance
- Train "B" of RHR is in service and being used for an RCS cooldown at 25°F/hr
- All Train "B" RHR components are arranged in their normal alignment for plant cooldown
- MOV-1RH-758, RHR HX FCV is 30% OPEN
- MOV-1RH-605, RHR HX Bypass FCV is 50% OPEN
- MOV-1CH-142, RH LTDN to Non Regen Hx Inlet FCV is 50% OPEN

During the cooldown, a partial blockage of the "B" RHR HX tubes occur.

IF the RCS cooldown is to CONTINUE at 25°F/hr, the reactor operator will be required to ______________.

A. **CLOSE** MOV-1RH-605, RHR HX Bypass FCV and allow MOV-1RH-758, RHR HX FCV to automatically throttle **OPEN** to maintain total RHS system flow

B. **OPEN** MOV-1CH-142, RH LTDN to Non Regen Hx Inlet FCV and allow MOV-1RH-758, RHR HX FCV to automatically throttle **OPEN** to maintain total RHS system flow

C. **CLOSE** MOV-1CH-142, RH LTDN to Non Regen Hx Inlet FCV and allow MOV-1RH-605, RHR HX Bypass FCV to automatically throttle **CLOSED** to maintain total RHS system flow

D. **OPEN** MOV-1RH-758, RHR HX FCV and allow MOV-1RH-605, RHR HX Bypass FCV to automatically throttle **CLOSED** to maintain total RHS system flow

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Answer: D

Explanation/Justification: K/A is met by evaluating the candidates knowledge of how the RHR heat exchanger flow control system will be operated in the event of a partial loss of flow through the heat exchanger were to occur.

A. Incorrect. Closing MOV-1RH-605 will force more water through the HX. However, MOV-1RH-758 does not have automatic flow control and will NOT automatically throttle open to maintain RHR system flow.

B. Incorrect. Opening MOV-1CH-142 will actually lower the flow indicated at FT605, which will cause MOV-1RH-605 to open further to balance out system preset flow. 1CH-142 is located between the HXs and MOV-758, therefore no change of flow through the HX will occur. MOV-1RH-758 does not have automatic flow control and will NOT automatically throttle open to maintain RHR system flow.

C. Incorrect. Closing MOV-1CH-142 will actually raise the flow indicated at FT605, which will cause MOV-1RH-605 to close further to balance out system preset flow. 1CH-142 is located between the HXs and MOV-758, therefore no change of flow through the HX will occur. MOV-1RH-605 will automatically throttle closed to maintain the preset RHR system flow.

D. Correct. IAW 1OM-10.4.A cooldown rate is adjusted by throttling open MOV-1RH-758 to control flow through the RHR HX, and MOV-1RH-605 will automatically throttle closed to maintain the preset RHR system flow.

Sys #	System	Category	KA Statement
005	Residual Heat Removal	K6 Knowledge of the effect of a loss or malfunction on the following will have on the RHRS:	RHR heat exchanger

K/A# K6.03 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1SQS-10.1 PPNT Rev. 18 slide 21
1OM-10.4.A Rev. 34 pg. 27
Unit 1 RM-0410-001 rev. 15

Question Source: Bank - 2LOT6 NRC Exam (Q30)

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

Objective: 1SQS-10.1, Rev. 18 Obj. 17 - Describe the control, protection and interlock functions for the control room components associated with the Residual Heat Removal System, including automatic functions, setpoints and changes in equipment status as applicable.

32. The following conditions exist:

- A Large Break LOCA has occurred
- All systems have functioned as designed

At what RWST level and signal will the automatic Transfer to Cold Leg Recirculation occur?

A. 1) 14' 0.5" with a CIB signal

B. 1) 14' 0.5" with a SI signal

C. 1) 27' 7.5" with a CIB signal

D. 1) 27' 7.5" with a SI signal

Answer: B

Explanation/Justification: K/A is met with the knowledge of the ECCS design feature which causes a Transfer to cold leg recirculation to automatically occur when the RWST level reaches 14' 0.5" coincident with an SI signal. This transfer lineup will have the LHSI pumps take suction from the CNMT sump, and discharge to the suction to the HHSI pumps and also discharge to the LHSI cold leg injection lines.

A. Incorrect. Plausible because the level is correct but the coincidence is with the SI actuation instead of the CIB. RWST of 27' 7.5" coincident with CIB signal will start the RSS pumps.
B. Correct. 2/4 RWST level at 14' 0.5" coincident with a safety Injection signal will automatically align Cold Leg Recirculation. Recirc alignment shifts the LHSI pump suction to the containment sump, and the HHSI pump suction will be supplied by the LHSI pumps discharge before the RWST is isolated. The LHSI pumps will still be aligned to the RCS Cold Legs via MOV-1SI-890C, LHSI to RCS Cold Legs valve which is NSA Open.
C. Incorrect. Plausible because a RWST level of 27' 7.5" coincident with CIB signal will start the RSS pumps which is an ESF actuation, but it is not correct for the Transfer to Cold Leg Recirculation.
D. Incorrect. Plausible because a RWST level of 27' 7.5" coincident with CIB signal will start the RSS pumps which is an ESF actuation, but it is not correct for the Transfer to Cold Leg Recirculation.

Sys #	System	Category	KA Statement
006	Emergency Core Cooling	K4 Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following:	Cross-connection of HPI/LPI/SIS

K/A# K4.14 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-13.2.B Rev. 12 pg. 2
1OM-1.5.B.4 Rev. 17 pg 21
1OM-11.3.B.1 Rev. 20 pg. 26
1SQS-11.1 PPNT Rev. 14 Slide 108

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. 18 - Describe the control, protection and interlock functions for the control room components associated with the Safety Injection System, including automatic functions, set points and changes in equipment status as applicable including the following: c. Transfer to Cold Leg Recirculation Logic.

33. Given the following:

- A reactor trip has occurred due to low RCS pressure.
- The crew is performing E-0, reactor Trip or Safety Injection.
- A PRZR Safety Valve is failed open.
- Containment Pressure is 13.7 psia and stable.
- PRT pressure is 26 psig and rising.

Which of the following describes the highest indicated PRT pressure that will exist just prior to Containment pressure rising due to this event?

A. 84 psig

B. 98.7 psig

C. 100 psig

D. 113.7 psig

Answer: A

Explanation/Justification: K/A is met demonstrating the knowledge of the PRT rupture disc setpoint and the concept that the setpoint is based on pressure inside the tank and containment atmospheric pressure.

A. Correct. PRT pressure indication is relative to containment atmospheric pressure. The rupture discs actuate at 85 psid between internal tank pressure and containment atmospheric pressure. If containment is at 13.7 psia (-1 psig), then PRT pressure (indicated) would be at 84 psig just prior to rupture discs actuating.

B. Incorrect. Plausible distractor if candidate thinks the setpoint is 85 psig + 13.7psia. Misconception of psig and psia.

C. Incorrect. Plausible distractor if candidate thinks setpoint is 100 psig, and doesn't take cnmt pressure into consideration.

D. Incorrect. Plausible distractor if candidate thinks the setpoint is 100 psig + 13.7psia. Misconception of psig and psia.

Sys #	**System**	**Category**	**KA Statement**
007	Pressurizer Relief/Quench Tank	K1 Knowledge of the physical connections and/or cause effect relationships between the PRTS and the following systems:	Containment system

K/A# K1.01 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.2.B Rev. 17 pg. 9; 1OM-6.1.C Rev. 7 pg.28

Question Source: Bank – 1LOT7 NRC Exam (Q11)

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

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 in Mode 5, preparing to draw a bubble in the PRZR.

- PRZR is solid
- Tcold is 175 F and STABLE
- RCS pressure is 290 psig and STABLE
- 'C' RCP is running
- Both Trains of RHR are in service

MOV-1CH-142, RH LTDN to Non-Regen Hx Inlet Flow Control Valve fails CLOSED.

Based on the above conditions, which of the choices below complete the following statements?

1) PRT parameters will be rising due to ______(1)______ discharging to the PRT.

2) The ____(2)____ will automatically prevent the PRT from exceeding design pressure.

A. 1) PCV-1RC-455C, PRZR PORV Relief Valve
2) Rupture Discs

B. 1) PCV-1RC-455C, PRZR PORV Relief Valve
2) PRT Spray Valve, MOV-1RC-516

C. 1) RV-1RH-721, RHR PP SUCT HDR Relief Valve
2) Rupture Discs

D. 1) RV-1RH-721, RHR PP SUCT HDR Relief Valve
2) PRT Spray Valve, MOV-1RC-516

Answer: A

Explanation/Justification: K/A is met by the candidate determining which component will be discharging to the PRT based on plant conditions and setpoint knowledge, and recognize that the automatic rupture disks will release pressure from the PRT at 85 psig.

A. Correct. The candidate must know that OPPS is in service when RHR is in service, and must know the different setpoints for OPPS (~400 psig) and the RHR relief (600 psig). The candidate must also know that only 2 PORVs are used for OPPS (455C and 455D), so even if they know OPPS is in service they must recognize that 455C is OPPS. If this condition is not corrected the PRT Rupture Disks will blow out at 85 psig, and release the PRT pressure and contents to the containment.

B. Incorrect. It is correct that the OPPS PORV is the source into the PRT. The PRT Spray Valve is a plausible distractor because primary grade water is manually used to lower pressure on the PRT, but there are no automatic features associated with the PRT Spray Valve.

C. Incorrect. It is plausible because the RHR relief does discharge to the PRT, but this relief setpoint is 600 psig, and with OPPS in service, PORV 455C will lift at ~400 psig. If this condition is not corrected the PRT Rupture Disks will blow out at 85 psig.

D. Incorrect. It is plausible because the RHR relief does discharge to the PRT, but this relief setpoint is 600 psig, and with OPPS in service, PORV 455C will lift at ~400 psig. The PRT Spray Valve is a plausible distractor because primary grade water is manually used to lower pressure on the PRT, but there are no automatic features associated with the PRT Spray Valve.

Sys #	System	Category	KA Statement
007	Pressurizer Relief/Quench Tank	A3 Ability to monitor automatic operation of the PRTS, including:	Components which discharge to the PRT

K/A# A3.01 **K/A Importance** 2.7* **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.2.B Rev. 17 pg. 10
1OM-6.1.C Rev. 7 pg. 26 & 28
1OM-10.2.B rev. 5 pg. 3

Question Source: New

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

Objective: 1SQS-6.4, Rev. 14 Obj. 5 - List the nominal value of the field operating parameters associated with the Pressurizer and Pressurizer Relief System.

1SQS-6.4, Rev. 14 Obj. 5 - Describe the control, protection and interlock functions for the field components associated with the Pressurizer and Pressurizer Relief System, including automatic functions, setpoints and changes in equipment status as applicable.

35. The plant is at 65% power with all systems in normal alignment for this power level when the crew enters AOP 1.28.1, Loss Of Secondary Component Cooling Water.

Which of the following conditions, and required action will be directed by AOP 1.28.1 due to the high Secondary Component Cooling Water temperature condition?

A. Main Feedwater Pump Bearing temperature is 224°F, trip the Main Feedwater pump.

B. Turbine Lube Oil Cooler Outlet temperature is 177°F, trip the Reactor.

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

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

Answer: A and B are correct responses

Explanation/Justification: K/A is met by the candidate determining that the Main Feedwater Pump Bearing temperature has exceeded its high temperature setpoint due to the high Secondary Component Cooling Water temperature, and requires the MFW pump to be tripped per the loss of secondary cooling water AOP.

A. Correct. In accordance with AOP-1.28.1, if the Main Feedwater Pump Bearing temperature is >220°F, then a MFP trip is required. The AOP doesn't have the operator look at Rx power, but it does state if all MFW is lost, then trip Rx. The power level of 65% in the stem was given so that the candidate would not think they had to trip the Rx due to being >70% as required in the loss of feedwater AOP.

B. Incorrect. Plausible distractor since AOP-1.28.1 requires a Rx trip if Turbine Lube Oil Cooler Outlet temperature is ≥180°F with Rx power >P9 (49%).

C. Incorrect. Plausible distractor since AOP-1.28.1 requires a Rx trip if Turbine Journal Bearing Metal temperature is ≥225°F with Rx power >P9 (49%).

D. Incorrect. Plausible distractor since AOP-1.28.1 requires the pump tripped if Main Condensate Pump Bearing temperature is ≥180°F

Sys #	System	Category	KA Statement
008	Component Cooling Water	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	High/low CCW temperature

K/A# A2.03 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.28.1 Rev. 3 pg. 7 & LHP.

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-24.1, Rev. 20 Obj. 19 - 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.

36. Given the following conditions after a small break LOCA:
- TI-1RC-453, PRZR Liquid temperature, indicates 545°F
- TI-1RC-454, PRZR Vapor temperature, indicates 561°F
- Pressurizer pressure is 1085 psig

Based on the above temperature indications, which of the following statements is correct?

A. The steam is at saturation.

B. The steam is superheated by approximately 5°F.

C. The water is at saturation.

D. The water is subcooled by approximately 5°F.

Answer: B

Explanation/Justification: K/A is met by the candidate taking actual plant parameters of PZR pressure, PZR vapor and liquid temperature, and determining the condition of the PZR fluid.

A. Incorrect. Saturation temperature of 556°F at 1100 psia, therefore the steam is 5F superheated. The saturation pressure is based on the 1085 psig (1100 psia) which has a corresponding saturation temperature of 556°F. Based on the liquid temperature, the water is subcooled by approximately 10F.

B. Correct. The saturation pressure is based on the 1085 psig (1100 psia) which has a corresponding saturation temperature of 556°F. Therefore, the steam is superheated by 5F.

C. Incorrect. The water is subcooled approximately 11°F.

D. Incorrect. The water is subcooled approximately 11°F.

Sys #	System	Category	KA Statement
010	Pressurizer Pressure Control	K5 Knowledge of the operational implications of the following concepts as the apply to the PZR PCS:	Determination of condition of fluid in PZR, using steam tables

K/A# K5.01 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate Steam Tables **Technical References:** Steam tables

Question Source: Bank – Diablo Canyon 2012 NRC Exam (Q8) Modified

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

Objective: GO-GPF.T3, Rev. 0 Obj. 8 - Apply saturated and superheated steam tables in solving liquid-vapor problems.

37. The plant is at 100% power, when the PRZR Spray Valve, PCV-1RC-455B controller output fails to 100%.

1) Which of the following annunciators will be in alarm FIRST?
2) How will RCS Subcooling respond to this event?

Annunciator ______(1)______ will alarm first, and RCS subcooling will ______(2)______.

A. 1) A4-23, Pressurizer 2/3 Press Relief Block
2) decrease

B. 1) A4-23, Pressurizer 2/3 Press Relief Block
2) increase

C. 1) A4-11, Pressurizer Control Pressure Low
2) decrease

D. 1) A4-11, Pressurizer Control Pressure Low
2) increase

Answer: C

Explanation/Justification: K/A is met by evaluating the candidates knowledge of the annunciators associated with lowering RCS pressure due to a spray valve controller output failing to 100%, and which annunciator will alarm first. Then determine the RCS subcooling indications will be decreasing with the lowering RCS pressure.

A. Incorrect. Pressurizer 2/3 Press Relief Block alarms at 2000 psig after A4-11. Plausible distractor because both of these alarms will come in as pressure is lowering. Subcooling will be decreasing.
B. Incorrect. Pressurizer 2/3 Press Relief Block alarms at 2000 psig after A4-11. Plausible distractor because both of these alarms will come in as pressure is lowering. RCS subcooling will be decreasing as RCS pressure lowers. Plausible distractor if the candidate does not understand how subcooling is calculated.
C. Correct. When the spray valve controller fails to 100%, the spray valve travels to the full open position. RCS pressure will continue to lower until a Rx trip occurs. Pressurizer Control Pressure Low alarms at 2185 psig, and Pressurizer 2/3 Press Relief Block alarms at 2000 psig. As pressure is lowering the RCS subcooling will also be lowering.
D. Incorrect. Pressurizer Control Pressure Low will alarm first, but RCS subcooling will be decreasing as RCS pressure lowers. Plausible distractor if the candidate does not understand how subcooling is calculated.

Sys #	System	Category	KA Statement
010	Pressurizer Pressure Control	Generic	Knowledge of annunciator alarms, indications, or response procedures.

K/A# 2.4.31 **K/A Importance** 4.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.4. ABU Iss. 3 rev. 0 pg. 1
1OM-6.4. AAM rev. 2 pg. 2
GO-GPF.T3 Rev. 0 pg. 49

Question Source: New

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

GO-GPF.T3, Rev 0 Obj. 11 - Define the following terms: Subcooling

38. Given the following conditions:
- The plant is operating at 100% power.
- DC-SWBD-1 Breaker 8-7, "Reactor Trip Breaker Switchgear (RTA, BYB)" tripped OPEN.

Which of the following statements describe the effect that this loss of DC power will have on the 'A' Reactor Trip Breaker?

A. Reactor will trip due to a shunt trip of 'A' Reactor Trip Breaker.

B. Reactor will trip due to an undervoltage trip of 'A' Reactor Trip Breaker.

C. A shunt trip signal will NOT be capable of opening 'A' Reactor Trip Breaker.

D. An undervoltage trip signal will NOT be capable of opening 'A' Reactor Trip Breaker.

Answer: C

Explanation/Justification: K/A is met by demonstrating knowledge of the 125 VDC power supply in RPS, and knowing that the shunt trip coil is normally de-energized, that in the event of a loss of 125VDC, the coil will be unable to actuate a RX trip.

A. Incorrect. Plausible because most actuations are de-energized to actuate, but the shunt trip requires that 125 VDC be applied to the shunt trip coil to cause the breaker to open.

B. Incorrect. Plausible because the UV trip receives power from a DC power supply, but the power supply is 48 VDC within SSPS.

C. Correct. Bkr 8-7 supplies 125 VDC power to the shunt trip coil for Train A Trip Breaker. UV coils and shunt trip relays are supplied from 48 VDC from SSPS. The shunt trip coil is normally de-energized and without power available, a shunt trip of Train A Trip Breaker is not possible.

D. Incorrect. Plausible because the UV trip receives power from a DC power supply, but the power supply is 48 VDC within SSPS and is de-energized to actuate.

Sys #	System	Category	KA Statement
012	Reactor Protection	K1 Knowledge of the physical connections and/or cause effect relationships between the RPS and the following systems:	125V dc system

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

References provided to Candidate None **Technical References:** 3SQS-1.2 U1 LP PPT rev. 8 slide 9
U1 RE-0021TZ Rev. AA
1OM-53C.4.1.39.1A Rev. 8 pg. 30

Question Source: Bank - Comanche Peak 2013 NRC Exam (Q10)

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-1.2, Rev. 7 Obj. 11 - Given a Reactor Protection System Hardware configuration and without referenced material, describe the Reactor Protection System Hardware control room response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable: Loss of electrical power

39. Given the following plant conditions:

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

Which of the following identifies the ESF logic response to the subsequent containment instrument failure?

CIB ______(1)______ actuate, and Safety Injection ______(2)______ actuate.

	CIB (1)	Safety Injection (2)
A.	will	will
B.	will	will NOT
C.	will NOT	will
D.	will NOT	will NOT

Answer: D

Explanation/Justification: K/A is met by demonstrating the knowledge to determine that the initial cnmt channel was placed in trip/bypass, and the ability to determine that a second failure will not cause an ESF actuation based on the specific input channels and logic.

A. Refer to correct answer explanation.

B. Refer to correct answer explanation.

C. Refer to correct answer explanation.

D. 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.

Sys #	System	Category	KA Statement
013	Engineered Safety Features Actuation	A3 Ability to monitor automatic operation of the ESFAS including:	Input channels and logic

K/A# A3.01 **K/A Importance** 3.7* **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.4.IF Rev. 8 pgs. 5 & 7; U1 USFAR Fig. 7.2-1 Rev. 22

Question Source: Bank – Vision 254545 Modified

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

Objective: 3SQS-1.2-01-07: Given a Reactor Protection System Hardware alarm condition and using the Alarm Response Procedure(s), determine the appropriate alarm response, including automatic and operator actions in the field.

40. Given the following initial conditions:
    - Plant is at 100% power
    - 'A' and 'B' CNMT Air Recirc (CAR) Fans are RUNNING
    - 'C' CNMT Air Recirc Fan is aligned to Bus 1N, with the control switch GREEN targeted

    Subsequent conditions:
    - A11-25, Containment Air Recirc Fan Auto Stop is in alarm
    - 'A' CNMT Air Recirc Fan indicating light is Bright WHITE

    Based on the above conditions, complete the following statements.

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

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

A. 1) auto start
   2) is

B. 1) auto start
   2) is NOT

C. 1) remain shutdown
   2) is

D. 1) remain shutdown
   2) is NOT

Answer: D

Explanation/Justification: K/A is met by the candidate knowing that the 'C' CAR fan will not automatically start, and that the Alarm Response Procedure for the overcurrent trip of the 'A' CAR fan will have the crew refer to 1OM-44C.4.D to manually start the stby CAR fan. The startup procedure will also align cooling water if it is necessary for containment temperature control.

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

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

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

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

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

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

References provided to Candidate None **Technical References:** 1OM-44C.4.D Rev. 13 pg. 8-10; 1OM-44C.4.AAD Rev. 13 Iss. 2

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-44C.1, Rev.10 Obj. 4 - Describe the control, protection and interlock functions for the field components associated with the Containment Ventilation System, including automatic functions, setpoints and changes in equipment status as applicable.

41. The plant has tripped from 100% power due to a LOCA.

- The Crew has transitioned to ES-1.3, "Transfer to Cold Leg Recirculation".
- A1-25, "2/4 RWST LO LEVEL & SI AUTO XFR SI INJ TO RECIRC" is LIT.
- Attachment 1-G, "Cold Leg Recirculation Actuation" is in progress.

Which of the following valves will automatically **CLOSE** to prevent radioactive release from the containment to the RWST? (disregard system check valves)

1. MOV-1SI-885A, B, C, D, LHSI Pump Mini Flow Isol Valves
2. MOV-1CH-115B, D, RWST Discharge to Chg Pumps Suct Valves
3. MOV-1SI-862A, B, LHSI Pump RWST Suction Valves

A. 1 and 2 ONLY.

B. 2 and 3 ONLY.

C. 1 and 3 ONLY.

D. 1, 2, and 3.

Answer: D

Explanation/Justification: K/A is met by demonstrating knowledge of the interrelation between the RWST (Cnmt Spray System) and the Charging/Safety Injection system, and the automatic swapover features which are designed to prevent escape of radioactivity from containment to the outside after a Transfer to Cold Leg Recirculation.

A. Incorrect. MOV-1SI-862A, B also close to prevent cnmt sump water from reaching the RWST.
B. Incorrect. MOV-1SI-885A, B, C, D also close to prevent cnmt sump water from reaching the RWST.
C. Incorrect. MOV-1CH-115B, D also close to prevent cnmt sump water from reaching the RWST.
D. Correct. The candidate must know that the 885s, 115s, and 862s all automatically close on Transfer to Cold Leg Recirculation to isolate the RWST from Cnmt. The question is focused on automatic interlocks, therefore 'disregard system check valves' is included in the question stem to eliminate confusion since MOV-1CH-115A & B are closed to provide redundant isolation (UFSAR) due to check valve SI-27 being in the flowpath.

Sys #	System	Category	KA Statement
026	Containment Spray	K4 Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following:	Prevention of path for escape of radioactivity from containment to the outside (interlock on RWST isolation after swapover)

K/A# K4.09 **K/A Importance** 3.7* **Exam Level** RO

References provided to Candidate None **Technical References:** U1 UFSAR Rev 24 pg. 6.3-27
1OM-53A.1.1-G Rev. 2
U1 RM-0413-001 Rev. 25
U1 RM-0407-001 Rev. 34
U1 RM-0411-001 Rev. 27

Question Source: Bank – 2LOT8 NRC Exam (Q40) Modified

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

Objective: 1SQS-13.1 Rev 15 Obj. 21 - Given a specific plant condition, predict the response of the Containment Depressurization System control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off-normal condition.

1SQS-11.1 Rev. 14 obj.19 - Given a Safety Injection System configuration and without referenced material, describe the Safety Injection System control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable: c. Transfer to Cold Leg Recirculation

42. Which of the following conditions does **NOT** meet the applicable Limiting Condition for Operation (LCO)?

A. TV-1MS-101C, Main Steam Trip Valve lost DC control power in MODE 4.

B. HYV-1FW-100B, 1B Main Feedwater CNMT Isolation Valve has a broken stem in MODE 4.

C. SV-1MS-105A, 'A' Steam Generator Safety Valve lift setpoint is out of tolerance in MODE 3.

D. HCV-1MS-104, Residual Heat Release Control Valve broken air line on actuator in MODE 1.

Answer: C and B are correct responses

Explanation/Justification: K/A is met by the candidate demonstrating knowledge of various Limiting Conditions for Operation and Mode of applicability and making a determination which Main Steam component does not meet the LCO.

A. Incorrect. IAW LCO 3.7.2 Three MSIVs shall be operable in Modes 1, and MODES 2 and 3 except when all MSIVs are closed and de-activated. Plausible answer since the candidate must know that the LCO is not applicable in Mode 4.

B. Incorrect. IAW LCO 3.7.3, Main feedwater Isolation Valves (MFIVs), MFRVs, and MFRV Bypass valves shall be operable in Modes 1, 2, and 3 unless closed and de-activated. This distractor put the condition in Mode 4 so that the exception is not applied if chosen.

C. Correct. IAW LCO 3.7.1 Five MSSV per SG shall be operable when in Modes 1, 2, and 3. A Safety valve lift setpoint out of tolerance does make it inoperable in Mode 3.

D. Incorrect. IAW LCO 3.7.4, three ADV lines shall be OPERABLE in MODES 1, 2, and 3, and MODE 4 when steam generator is relied upon for heat removal. Plausible answer because some candidates may consider the RHR valve an ADV but at Unit 1 it is not considered as such.

Sys #	**System**	**Category**	**KA Statement**
039	Main and Reheat Steam	Generic	Knowledge of conditions and limitations in the facility license.

K/A# 2.2.38 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** Tech Specs pgs. 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-PLTSYS ITS, Rev. 2 Obj. 3 - Given plant conditions, determine the criteria necessary to ensure compliance with each Section Plant Systems System LCO and Licensing Requirement in accordance with the Bases, Surveillance Requirements, and the Applicability.

1SQS-21.1, Rev. 16 Obj. 7 - Identify the Main Steam Supply System field instruments, subsystems and components that are required to be operable by the Technical Specifications & Licensing Requirements Manual.

43. Reactor power is 11%, with a plant startup in progress.

- T_{avg} is 551°F and STABLE
- RCS pressure is 2235 psig and STABLE
- STM Dump Control Mode Selector switch is in "STM PRESS" position
- Cooldown Vlvs Control AM-1MS-464B is in AUTO and set for 1005 psig

Main Steam Manifold pressure transmitter PT-1MS-464 suddenly fails to zero.

How will the steam dumps respond to this failure?

All steam dump valves will________________________

A. initially CLOSE, then the cooldown valves will modulate OPEN to control pressure at 1005 psig.

B. CLOSE and remain CLOSED until the operator places AM-1MS-464B in "MAN" and increases the output signal.

C. initially CLOSE, then the Load rejection controller will "Arm" and the first 2 banks will modulate OPEN to control T_{avg} at 553°F.

D. CLOSE and remain CLOSED until the operator places the STM Dump Control Mode Selector switch in the "T_{avg}" position.

Answer: B

Explanation/Justification: K/A is met by demonstrating knowledge of the SDS response to a loss of Main Steam Pressure transmitter, and how it will be responded to in the control room.

A. Incorrect. The steam dumps will close, but with PT464 indicating 0 psig, and SDs in steam pressure mode, they will not open to try to maintain 1005 psig.
B. Correct. In Steam Pressure mode, PT464 controls the steam dump actions. When steam pressure indicates 0 psig on PT464, the SDS will have no demand to open the steam dumps.
C. Incorrect. Valves will close, but they will NOT re-open. The Load rejection controller would arm if PT446 failed low, but this is not indicated in the stem.
D. Incorrect. It is correct the steam dumps will close. Even if the Steam Dump Control Mode selector switch is taken to the T_{avg} position, the dumps will NOT open since no arming signal exist.

Sys #	System	Category	KA Statement
039	Main and Reheat Steam	K3 Knowledge of the effect that a loss or malfunction of the MRSS will have on the following:	SDS

K/A# K3.06 **K/A Importance** 2.8* **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-21.5.A.24 Rev 4, 1OM-21.5.A.25 Rev 4

Question Source: Bank – 1LOT14 Audit Exam (Q35)

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

Objective: 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.

44. A plant startup is in progress in accordance with 1OM-52.4.B, Load Following.
    - Rx power is 48% and RISING
    - 2 Condensate pumps are RUNNING
    - 1FW-P-1B, Main FW 1B Pump is RUNNING

    In accordance with 1OM-52.4.B, Load Following, when should the second main feedwater pump be started?

A. reactor power is raised to between 51-52 percent

B. A7-31, SG Feed Pump Disch Flow Low-Stop One Pump CLEARS

C. FI-1FW-150, Main FW Pump Flow is 12K-13K GPM

D. FCV-1FW-150B, SG Main FW Pump 1B Recirc Valve indicates CLOSED

Answer: A

Explanation/Justification: K/A is met by demonstrating the ability to start a second MFW pump during a plant startup when required by plant procedures based on the plant power level of 51-52%.

A. Correct. The Load Following procedure directs the starting of the second MFW pump at 51-52% power or when annunciator A7-39, 'SG Feed Pump Discharge Flow HI Start 2nd Pump' is in alarm.

B. Incorrect. Plausible distractor because A7-31, Steam Generator Feed Pump Disch Flow Low-Stop One Pump annunciator alarms when 2 MFW pumps are running and feedwater flow lowers to <14000 gpm. This alarm alerts the operator to secure one MFW pump. By stating that the alarm clears in this selection, it makes the answer plausible.

C. Incorrect. Plausible distractor because FI-1FW-150 is the input to A7-39, 'SG Feed Pump Discharge Flow HI Start 2nd Pump' annunciator which alerts the operator to start a second MFW pump. A7-39 doesn't alarm until FI-1FW-150 indicates >15000 gpm with one MFW pump running.

D. Incorrect. Plausible distractor if the candidate thinks the FCV-1FW-150B, 1B Recirc Valve closes when the MFW pump is supplying maximum flow capable, when in fact FCV-1FW-150B closes when either main feed pump is operating and flow to the steam generators is greater than 8,000 gpm or if both main feed pumps are shutdown.

Sys #	System	Category	KA Statement
059	Main Feedwater	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including:	Power level restrictions for operation of MFW pumps and valves.

K/A# A1.03 **K/A Importance** 2.7* **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-52.4.B Rev. 53 pg. 93

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.5 / 45.5)

Objective: 1SQS-24.1, Rev. 20 Obj. 13 - List the nominal value of the control room operating parameters associated with the Main Feedwater, Dedicated Auxiliary Feedwater, Auxiliary Feedwater System and Steam Generator Water Level Control Systems.

1SQS-24.1, Rev. 20 Obj. 19 - 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.

1SQS-24.1, Rev. 20 Obj. 20 - Given a Main Feedwater, Dedicated Auxiliary Feedwater, Auxiliary Feedwater System or Steam Generator Water Level Control System alarm condition and using the Alarm Response Procedure(s), determine the appropriate alarm response, including automatic and operator actions in the control room.

45. The plant is operating at 100% power.

- The breaker for SOV-1MS-105A, Control Solenoid for TV-1MS-105A, AFW Turbine Steam Supply A Train Trip Valve, has tripped OPEN

Based on the above conditions, which of the following completes the statement below.

The Turbine Driven AFW Pump will ______________________.

A. NOT automatically start under any condition

B. immediately start and operate at full design capacity

C. **only** start if an Auto Start initiation signal occurs, and will operate at full design capacity

D. **only** start if an Auto Start initiation signal occurs, but insufficient steam is available to operate at full design capacity

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Answer: B

Explanation/Justification: K/A is met by the knowledge which is demonstrated by identifying the failure mode of the control solenoid for the steam admitting valve for the TDAFW pump, then determining that design flow will be obtained.

A. Incorrect. Plausible distractor is the candidate thinks TV-1MS-105A & B are in series, and the control solenoid failure prevents the valve from opening with or without an Auto Start initiation signal.

B. Correct. On a loss of power the solenoid will de-energize causing TV-1MS-105A to open. It is a fail open valve which is in parallel with TV-1MS-105B. The plant operating at 100% power, steam is supplied from 2 SGs, up through NSA open MOV-1MS-105, and normally is isolated from the NSA open 1MS-465 (trip throttle valve) by TV-1MS-105A & B. One open train of steam supplied to the TDAFW is sufficient for design flow.

C. Incorrect. This is plausible if the candidate thinks the control solenoid de-energizing would cause TV-1MS-105A to fail close, and knowing that TV-1MS-105B is in parallel and capable of supplying sufficient steam for full design flow.

D. Incorrect. This is plausible if the candidate thinks the control solenoid de-energizing would cause TV-1MS-105A to fail close, and knowing that TV-1MS-105B is in parallel and with one valve closed not capable of supplying sufficient steam for full design flow.

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:	Controllers and positioners

K/A# K6.01 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-24.1.D Rev 6 pg 3; U1 RM-0421-001 Rev 25; 1OM-21.3.B.1 Rev 22 pg 13

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 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.

46. Given the following conditions.

- The plant is in Mode 3 with all systems in normal alignment for this mode
- ACB-41A trips on overcurrent causing a loss of the "A" 4KV bus
- No operator actions have been taken

What will be the status of the 'A' and 'B' 480v buses five minutes after ACB-41A trips?

	480v Bus **'A'**	480v Bus **'B'**
A.	De-energized	De-energized
B.	Energized	Energized
C.	De-energized	Energized
D.	Energized	De-energized

Answer: B

Explanation/Justification: K/A is met by testing the knowledge of the existence of the automatic bus transfer between 480v busses 1A and 1B on an undervoltage condition of the bus. The time delay of 45 seconds, as well as how an overcurrent condition on 'A' 4KV will affect the 480v bus 1A, and the interlocks/auto design feature of the supply and tie brks work with each other are required knowledge to answer the question.

A. Incorrect. Plausible if the candidate thinks both busses are powered from 'A' 4KV bus A. 480 bus 1B is supplied from 4KV bus 1C.

B. Correct. Both bus 1A and 1B will be energized. The 'A' 4KV bus will be locked out on overcurrent causing the 'A' 480v bus to be initially de-energized. The 1A-1B bus tie bkr (1A10) will close automatically to carry the loads on bus 1A from bus 1B. The ABT requires bus 1B to be energized, and there is approximately a 45 second delay after bus 1A receives the undervoltage condition. All 480v bkr control switch are in the normal alignment as stated in the stem, and the stem states 5 minutes after ACB41 opens causing the undervoltage condition.

C. Incorrect. Plausible if the candidate knows that 480 bus 1B is supplied from 4KV bus 1C and energized, but does not know of the automatic bus transfer that supplies 480v bus 1A from 1B on a loss of power.

D. Incorrect. Plausible if the candidate thinks that 480 bus 1A is supplied from 4KV bus 1C and energized, and thinks 480v bus 1B is de-energized due to the loss of 4KV bus 'A', and does not know of the automatic bus transfer that supplies each 480v bus 1A or 1B on a loss of power.

Sys #	System	Category	KA Statement
062	AC Electrical Distribution	K4 Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following:	Interlocks between automatic bus transfer and breakers

K/A# K4.03 **K/A Importance** 2.8* **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-37.1.D Iss. 4 Rev. 0 pgs. 3-5
U1 RE-0100A rev. 9
3SQS-37.1 U1 PPNT Rev. 9 iss. 2 Slide 44

Question Source: Bank – Vision 257139

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

Objective: 3SQS-37.1, Rev. 9 Obj. 3 - Describe the control, protection, and interlock functions associated with the 480 VAC Distribution System operation for the following. Include automatic functions, set points, and changes in equipment status. 480 VAC bus supply breakers & 480 VAC bus tie breakers

47. The plant is operating at 75% power with all systems in normal alignment for this power level.
    - A loss of DC Bus 1 has occurred
    - The crew has entered AOP 1.39.1A, Loss of 125VDC Bus 1

    What is the status of 4 KV Bus 1A and 1AE **two minutes** after the Loss of DC Bus 1?

    4 KV Bus 1A will ____(1)____, and 4 KV Bus 1AE will ____(2)____.

	(1)	(2)
A.	be energized	be energized
B.	be energized	NOT be energized
C.	NOT be energized	be energized
D.	NOT be energized	NOT be energized

Answer: D

Explanation/Justification: K/A is met by demonstrating the ability to monitor the operation of major plant busses after a loss of DC Bus 1. The operator will have to determine what the status of the busses will be two minutes after the event. Two minutes is sufficient time for the Rx to trip due to a loss of feedwater, and for the fast bus transfer to occur. On the fast bus transfer the candidate will have to recognize that Offsite breaker ACB41A will fail to close due to the failure of the fast bus auto transfer due to no DC control power, and the EDG will fail to start due to a loss of DC power to the air start solenoids.

A. Incorrect. Plausible answer if the candidate doesn't recognize that a Rx trip will occur due to a loss of DC bus 1. The Busses will remain energized until the fast bus transfer occurs after the Rx trip.
B. Incorrect. Plausible answer if the candidate thinks the fast bus transfer will occur and energize the 'A' 4KV bus, but the 'AE' bus will not be energized due to the loss of DC power to the air start solenoids.
C. Incorrect. Plausible answer if the candidate recognizes that the fast bus transfer will not energize the 'A' 4kv bus, but thinks that the EDG will start on the loss of power to the AE bus.
D. Correct. Within 2 minutes after the loss of DC Bus 1, the Rx will trip due to the FRVs failing closed, and the auto transfer to Off-site power occurs. Bus 'A' power is lost due to ACB41A not closing on the fast bus transfer due to ACB41C failing to open on the loss of DC control power. Bus 'AE' will not be energized because of two reasons, first the #1 EDG will not start due to the loss of DC power to the air start solenoids, and second, no dc control power is available to close ACB 1E9, EDG Output Breaker if the EDG were able to start.

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:** 1OM-53C.4.1.39.1A Rev. 8 pgs. 1, 20, 29

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. 16 – Describe the control, protection, and interlock functions for the control room components associated with the 125 VDC Distribution System including automatic functions, set points, and changes in equipment status as applicable.

3SQS-39.1, Rev. 9 Obj. 19 - Given a 125 VDC Distribution System configuration, and without reference material, describe the 125 VDC Distribution System control room response to the following malfunctions, including automatic functions and changes in equipment status. c. Loss of DC Bus

48. Given the following plant conditions:

- The plant is at 100% power
- Vertical Board 'C' indicator for No. 2 DC Bus Grd Volts meter indicates a +110 VDC ground
- The crew performs 1OM-39.4.E, Clearing Grounds (125 VDC Busses 1-1 and 1-2)
- The ground is discovered to be on battery charger 1-2

How will No. 2 DC Bus Voltage indication in the Control Room respond when battery charger 1-2 is removed from service?

What action must be taken to ensure continued plant operation in regards to maintaining No. 2 DC Bus Voltage?

No. 2 DC Bus Voltage indication will be ______(1)______ when the battery charger is removed from service.

Plant operation may continue by ______(2)______.

A. 1) lower
2) placing the redundant battery charger in service

B. 1) lower
2) closing DC Bus 1-2A to Bus 5 crosstie breaker

C. 1) unaffected
2) placing the redundant battery charger in service

D. 1) unaffected
2) closing DC Bus 1-2A to Bus 5 crosstie breaker

Question 48

Answer: A

Explanation/Justification: K/A is met by the candidates ability to predict that when the battery charger is removed from service due to grounds, that the CR indication of DC bus voltage will lower to battery voltage. The candidate also has to know that continued plant operation cannot continue on the battery beyond its design of 2hrs, therefore the redundant charger for bus 1-2 must be placed in service.

A. Correct. With the charger supplying a float charge to the bus, the voltage is normally 132 VDC (129-138 vdc). When the charger is removed from service, bus voltage will lower to approximately 125 vdc. DC Bus 1-2 has 2 battery chargers available to maintain DC bus voltage, battery charger 1-2A and 1-2B. In the event one fails, the redundant battery charger will be placed in service to maintain DC loads.

B. Incorrect. First part is correct. Second part is a plausible distractor because DC bus 1-2 can be aligned to supply DC bus 5 in the event battery charger 5 fails.

C. Incorrect. Plausible if the candidate thinks the CR indication is the battery voltage, or that the charger maintains the same voltage as the battery. Second part is correct.

D. Incorrect. Plausible if the candidate thinks the CR indication is the battery voltage, or that the charger maintains the same voltage as the battery. Second part is a plausible distractor because DC bus 1-2 can be aligned to supply DC bus 5 in the event battery charger 5 fails.

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

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

References provided to Candidate None **Technical References:** 1OM-39.4.E rev. 7 pg. 5
1OM-39.4.C rev. 10 pg. 13
1OM-39.1.B rev. 1 pg. 3
Unit 1 RE-0001V rev. 33

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: 3SQS-39.1, Rev. 9 Obj. 2 - Describe the distribution paths for the 125 VDC Distribution System from the batteries and chargers to the 125 VDC Distribution Panels.

3SQS-39.1, Rev. 9 Obj. 8 - State the following NSA parameters for the 125 VDC Distribution System. a. Battery Float Voltage

49. What is the power supply to the #1 EDG starting air compressor 1EE-C-2A?

A. DC-SWBD-1

B. DC-SWBD-2

C. MCC1-E7

D. MCC1-E8

Answer: C

Explanation/Justification: K/A is met by demonstrating knowledge of the 480 VAC power supplied to #1 EDG air compressors.

A. Incorrect. Plausible answer since 125 VDC DC-SWBD-1 is supplied to #1 EDG for air compressor control circuit and starting air solenoids. 125VDC does run some lube oil pumps, therefore it's plausible that it could supply the air compressors for the EDG operation.

B. Incorrect. Plausible answer since 125 VDC DC-SWBD-2 is supplied to #2 EDG for air compressor control circuit and starting air solenoids. 125VDC does run some lube oil pumps, therefore it's plausible that it could supply the air compressors for the EDG operation. The candidate may think that the second air compressor is supplied by the opposite train of DC power.

C. Correct. MCC1-E7 supplies power to both #1 EDG air compressors 1EE-C-1A and 1EE-C-2A.

D. Incorrect. Plausible answer because MCC1-E8 supplies both #2 EDG air compressors. Candidate must know that both air compressors on the EDG are supplied by the same bus, but they may think that the second air compressor is supplied from the other train as a reliable source.

Sys #	System	Category	KA Statement
064	Emergency Diesel Generator	K2 Knowledge of bus power supplies to the following:	Air compressor

K/A# K2.01 **K/A Importance** 2.7* **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-36.1.C rev. 6 pg. 9; RM-0436-001 Rev. 11

Question Source: Bank – Vision #123486

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

Objective: 1SQS-36.2, Rev. 18 Obj. 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.)

50. What type of radiation detectors are used for the RM-1MS-102 Steam Generator Leak Monitoring System, and the leak rate indications are not valid until reactor power is above what reactor power?

    RM-1MS-102 Steam Generator Leak Monitoring System uses a ______(1)______ detector.

    Leak rate indications are **NOT** valid until reactor power reaches at least ______(2)______.

A. 1) Beta Scintillation
   2) 20 % power

B. 1) Beta Scintillation
   2) 30 % power

C. 1) Gamma Scintillation
   2) 20 % power

D. 1) Gamma Scintillation
   2) 30 % power

Answer: C

Explanation/Justification: K/A is met by demonstrating that Gamma Scintillation detector are used for the SG N-16 detectors to provide an indication of a primary-secondary leak, and that there are not sufficient N-16 gammas produced when Rx power is <20% for the detector to provide accurate indications.

A. Incorrect. Plausible distractor as there are only two types of detectors used in the process radiation monitoring system. Gamma and Beta Scintillation detectors. Second part is correct.
B. Incorrect. Plausible distractor as there are only two types of detectors used in the process radiation monitoring system. Gamma and Beta Scintillation detectors. Second part is a plausible value for Rx power but it is too high.
C. Correct. Steam Generator Leak detectors use Gamma Scintillation detectors to detect N-16 gammas in the main steam lines when Rx power is ≥20%. When Rx power is <20%, N-16 production in the core reduces significantly.
D. Incorrect. Gamma Scintillation detector is correct. Second part is a plausible value for Rx power but it is too high.

Sys #	System	Category	KA Statement
073	Process Radiation Monitoring	K5 Knowledge of the operational implications as they apply to concepts as they apply to the PRM system:	Radiation theory, including sources, types, units, and effects

K/A# K5.01 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-43.1.E Rev. 12 pg. 7
1OM-43.4.O Rev. 4 pg. 2
1OM-43.1.C Rev. 8 pg. 6
1OM-43.1.D Rev. 10 pg. 2

Question Source: New

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

Objective: 1SQS-43.1-01-1: Describe the function of the Radiation Monitoring systems and the associated major components.

1SQS-43.1-01-7: Describe the control, protection and interlock functions for the control room components associated with the Radiation Monitoring System, including automatic functions, and changes in equipment status as applicable.

51. Given the following plant conditions:

- The plant was operating at 100% power
- An event occurred that caused containment pressure to peak at 6 psig
- Offsite Power has remained available for the duration of the event
- All System functions as designed

Based on these plant conditions, which of the following combinations of reactor and turbine building components will have River Water flow for temperature control?

CCR HX's = Reactor Plant Component Cooling Water Heat Exchangers
CCT HX's = Turbine Plant Component Cooling Water Heat Exchangers
EDG's = Emergency Diesel Generators
RSS HX's = Recirculation Spray Heat Exchangers

	CCR HX's	CCT HX's	EDG's	RSS HX's
A.	YES	YES	YES	YES
B.	YES	YES	YES	NO
C.	NO	NO	NO	NO
D.	YES	NO	YES	NO

Answer: B

Explanation/Justification: K/A is met by the candidate predicting which of the listed components will have Service/River water supplied after an SI and CIA signal occur. The K/A statement is met by identifying that the CCR, CCT, and EDG HXs will have temperature control capabilities, and the RSS HXs will not have Service/River Water available for temperature control after a Safety Injection.

A. Incorrect. Plausible if the candidate thinks MOV-1RW-103A-D (RSS Hx) open on a Safety injection signal.

B. Correct. At >5 psig containment pressure, SI and CIA have actuated. CCR and CCT Hxs are not affected by the SI signal, and RW will remain available to the HXs. The SI signal will start EDGs and open MOV-1RW-113A-D, therefore providing cooling to EDG's. RSS Hx valves, MOV-1RW-103A-D do not automatically open until a CIB signal is received at 11.1 psig, therefore they will remain closed.

C. Incorrect. Plausible if the candidate thinks a CIB has occurred which would close MOV-1RW-106A & B (CCR) and MOV-1RW-114A & B (CCR) and the candidate may think this isolates both the CCR and CCT HX's, and candidate may think that NSA closed EDG and RSS Hx valves would have to be manually opened.

D. Incorrect. Plausible if the candidate thinks the CCR Hx will remain in service, the RSS Hx will remain isolated, and the EDG Hxs are aligned with a SI signal, but thinks TPRW system either trips on SI or CIA, or isolates the Hxs due to the SI.

Sys #	System	Category	KA Statement
076	Service Water	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including:	Reactor and turbine building closed cooling water temperatures

K/A# A1.02 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-30.1.D Iss 4 Rev. 3 pgs. 1, 9-11
1SQS-30.2 PPT, Rev. 18 Slide 31

Question Source: Bank – 2LOT15 NRC Exam (Q52) Modified

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

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. (SI, CIB, EDG Start).

52. Which of the following components are capable of being supplied by the Reactor Plant River Water system?

1) Charging Pump Lube Oil Coolers
2) Steam Generator Blowdown Heat Exchanger
3) Control Room Air Conditioning Condensers
4) Recirculation Spray Heat Exchangers
5) Steam Generator Auxiliary Feedwater Pumps
6) Main Steam Valve Area Air Handling Unit

A. 1, 2, and 3

B. 4, 5, and 6

C. 1, 3, and 5

D. 2, 4, and 6

Answer: C

Explanation/Justification: K/A is met by evaluating the function of the Reactor Plant River Water system by identifying various loads of the system during both normal and abnormal conditions.

A. Incorrect. Plausible distractor because Charging Pump Lube Oil Coolers and the Control Room Air Conditioning Condensers are supplied from RPRW, but the candidate must know that the Steam Generator Blowdown Heat Exchanger is cooled by Condensate.
B. Incorrect. Plausible distractor because Recirculation Spray Heat Exchangers and the Steam Generator Auxiliary Feedwater Pumps are, or can be supplied by RPRW, but the candidate must know that the Main Steam Valve Area Air Handling Unit is cooled by Chilled Water.
C. Correct. Charging Pump Lube Oil Coolers, Control Room Air Conditioning Condensers, and Steam Generator Auxiliary Feedwater Pumps (emergency use via 6" hard piped section to the AFW system) are all capable of being supplied by RPRW.
D. Incorrect. Plausible distractor because Recirculation Spray Heat Exchangers are supplied by RPRW, but the candidate must know that the Steam Generator Blowdown Heat Exchanger is cooled by Condensate, and the Main Steam Valve Area Air Handling Unit is cooled by Chilled Water.

Sys #	**System**	**Category**	**KA Statement**
076	Service Water	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-30.1.A Iss. 4 Rev. 1 pg. 1
1OM-30.1.C Rev. 12 pg. 8

Question Source: New

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

Objective: 1SQS-30.2-01-24: Describe the function of the Reactor Plant River Water system and the associated major components as documented in Chapter 30 of the associated Operating Manual.

53. The plant is at 100% power.

- 1SA-C-1A, 'A' Station Air Compressor is RUNNING
- 1SA-C-1B, 'B' Station Air Compressor is on clearance
- 1SA-TK-1A, Station Air Receiver 1A ruptures and develops a 5000 scfm air leak
- The crew is performing AOP 1.34.1, "Loss of Station Instrument Air"
- PI-1IA-106, Instrument Air Header Press is 87 psig and LOWERING
- All system components function as designed

Which of the following represents the air pressures indicated in the Main Control Room for the Instrument Air Receiver and Instrument Air Header 10 minutes after the event?

	PI-1IA-106B Instrument Air RECEIVER PRESS	PI-1IA-106 Instrument Air HEADER PRESS
	NOP - Normal Operating Pressure	
A.	NOP	NOP
B.	Less than NOP	NOP
C.	NOP	Less than NOP
D.	Less than NOP	Less than NOP

Answer: B

Explanation/Justification: K/A is met by evaluating the candidates knowledge of how the Instrument Air system will respond to a large air leak, and the automatic start of the Diesel driven air compressor to maintain the Instrument Air Header. This knowledge will be required to assist in monitoring the air system gauges in the control room to verify proper system response.

A. Incorrect. Plausible if the candidate thinks that TV-1SA-105 closing at 95 psig isolates the SA receiver from the IA receiver, and the diesel driven air compressor maintains the IA receiver. This would not be correct because there is a check valve (1IA-11) isolating the IA receiver from the IA header which is being maintained by the diesel driven air compressor which started at 93 psig.

B. Correct. IA Header pressure will be between 105 – 115 psig since the Diesel driven air compressor (1IA-C-4) will be carrying all of the IA loads. 1IA-C-4 auto starts at 93 psig. PI-1IA-106B, IA Receiver pressure will be approximately 0 psig due to 1SA-C-1A only has a capacity of 570 scfm, and the IA receiver is isolated from the IA header by an installed check valve 1IA-11 preventing backflow from the IA header.

C. Incorrect. Plausible if the candidate thinks that the Diesel driven air compressor will maintain the IA receiver, and the IA header will depressurize due to the SA receiver rupturing. The Diesel driven air compressor supplies the Standby IA receiver and the IA header.

D. Incorrect. Plausible if the candidate does not recognize that the Diesel driven air compressor will start at 93 psig and maintain the IA header pressure at 105-115 psig.

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.U Rev. 9 pg.2; 1OM-34.1.C Rev. 3 pg 2; 1SQS-34.1 PPNT Rev. 15 slide 9

Question Source: New

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

Objective: 1SQS-34.1, Rev. 15 Obj. 10 - List the nominal value of the control room operating parameters associated with the Compressed Air System.

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.

54. The following conditions exist:

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

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

A. 541°F

B. 547°F

C. 551°F

D. 556°F

Answer: D

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

A. Incorrect. This is the temperature the condenser steam dumps will close due to P-12 (low-low Tavg). Plausible if the candidate does not recognize that the MSIVs go shut, and the condenser steam dumps will also fail closed on a complete loss of instrument air.
B. Incorrect. This is the temperature the Rx trip controller will maintain with the condenser steam dumps. Plausible if the candidate does not recognize that the MSIVs go shut, and the condenser steam dumps will also fail closed on a complete loss of instrument air.
C. Incorrect. Plausible if the candidate thinks the Steam Generator Atmospheric Steam Dump Valves [PCV-1MS-101A, B, & C] will operate to maintain 1060 psig (not taking into account 1075 psia), but this is not accurate because these valves fail closed on loss of instrument air.
D. Correct. The loss of instrument air causes the MSIVs, SG Atm steam dump valves, and RHR valve to fail closed. Therefore, Tavg will be maintained by the lifting of the first SG safety valve at 1075 psig (~1090 psia).

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

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

References provided to Candidate Steam Tables **Technical References:** 1OM-21.1.C Rev. 6 pg. 3
1OM-53C.4.1.34.1 rev. 21 pg. 17-18

Question Source: Bank – Vision 257146

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

Objective: 1SQS-34.1, Rev. 15 Obj. 12 - Given a Compressed Air System configuration and without referenced material, describe the Compressed Air System control room response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable: Loss of instrument air

1SQS-21.1 Rev. 16 Obj. 12: Given a Main Steam Supply System configuration and without referenced material, describe the Main Steam Supply System control room response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable: Loss of instrument air

55. If a loss of the Chilled Water System occurs, ______(1)______ must be aligned to the Containment Air Recirculation (CAR) Cooling Coils in order to prevent containment temperature from exceeding the Technical Specification limit of ______(2)______.

A. 1) Primary Plant Component Cooling Water
   2) 108° F

B. 1) Primary Plant Component Cooling Water
   2) 105° F

C. 1) Reactor Plant River Water
   2) 108° F

D. 1) Reactor Plant River Water
   2) 105° F

Answer: C

Explanation/Justification: K/A is met with knowledge that the Reactor Plant River Water is a an alternative source of cooling for the Containment Air Recirculation (CAR) Cooling Coils if a loss of the normal cooling supply (Chilled Water) were to occur to ensure the Containment temperature remains less than the tech spec limits.

A. Incorrect. CCR is a plausible distractor since use of the alternate source (service water) is rarely performed and CCR cools several loads inside cnmt. TS 3.6.5 states cnmt average air temp shall be ≥70F and ≤108F.

B. Incorrect. CCR is a plausible distractor since use of the alternate source (service water) is rarely performed and CCR cools several loads inside cnmt. Plausible distractor with 1OM-29.4.H, Initiating River Water Backup Cooling to the Containment Air Recirculation Cooling Coils stating 105F in the purpose and P&Ls.

C. Correct. Reactor Plant River Water is the alternate method of cooling containment air when the chilled water system is unable to maintain average air temperature. TS 3.6.5 states cnmt average air temp shall be ≥70F and ≤108F.

D. Incorrect. Reactor Plant River Water is the alternate method. Plausible distractor with 1OM-29.4.H, Initiating River Water Backup Cooling to the Containment Air Recirculation Cooling Coils stating 105F in the purpose and P&Ls.

Sys #	System	Category	KA Statement
103	Containment	K1 Knowledge of the physical connections and/or cause-effect relationships between the containment system and the following systems:	CCS

K/A# K1.01 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-29.4.H Rev. 4 pg. 2
Tech Spec 3.6.5 pg. 3.6.5-1

Question Source: Bank – North Anna 2010 NRC Exam (Q25)

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-44C.1, Rev 10 Obj. 15 - Given a set of plant conditions and the appropriate procedure(s), apply the operational sequence, parameter limits, precaution and limitations, and cautions & notes applicable to the completion of the task activities in the control room.

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

56. The plant is at 100% power.
    - PRZR Level transmitter, LT-1RC-461 failed LOW earlier in the shift
    - All actions of 1OM-6.4.IF, Instrument Failure Procedure and Tech Spec 3.3.1 are complete
    - PRZR Level Control Channel Selector is in Channel I & II (LT459 & 460)
    - PRZR level is on program

    PRZR Level transmitter, LT-1RC-459 bellows ruptures causing an instantaneous failure.

    1) How will PRZR level indicator, LI-1RC-459A respond to this failure?

    2) What procedure will the crew use to respond to this event?

A. 1) rise
   2) E-0, Reactor Trip or Safety Injection

B. 1) lower
   2) E-0, Reactor Trip or Safety Injection

C. 1) rise
   2) AOP 1.4.1, Process Control Failure

D. 1) lower
   2) AOP 1.4.1, Process Control Failure

Answer: A

Explanation/Justification: K/A is met by placing a przr level instrument LT-1RC-461 bistables in trip iaw the Instrument Failure procedure and Tech Specs, Then rupturing the bellows on LT459, causing its level to fail high. With 2/3 PRZR level transmitters >92% the reactor will trip and E-0 will used to mitigate the event.

A. Correct. A bellows rupture will cause LT459 to fail high because the transmitter will sense a D/P of zero indicating the PRZR is full. Therefore, with all actions for LT461 failing low being complete, the bistables for LT461 are tripped. With 2/3 PRZR level indicators being > 92%, the Reactor will automatically trip. Proper procedure for the Reactor tripping is E-0.

B. Incorrect. Plausible if the candidate thinks a ruptured bellows makes the instrument fail low. Second part is plausible because if the instrument fails low, letdown will isolate causing the przr level to rise until 2/3 przr level indicators were > 92%, in which case a reactor trip would occur.

C. Incorrect. It is correct that level will rise. Second part is plausible if the candidate doesn't recognize that the bistables for LT461 have been placed in the trip condition. AOP-1.4.1 is the correct procedure to respond to a failed PRZR level instrument if the Rx hadn't tripped.

D. Incorrect. Plausible if the candidate thinks a ruptured bellows makes the instrument fail low. AOP-1.4.1 is the correct procedure to respond to a failed PRZR level instrument if the Rx hadn't tripped.

Sys #	System	Category	KA Statement
011	Pressurizer Level Control	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Failure of PZR level instrument - high

K/A# A2.10 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** GO-GPF.C7 Rev. 4 pg.54 & 55; 1OM-6.4.IF rev. 11 pg. 7 & 11; 1OM-1.5.B.1 Rev. 3 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: GO-GPF.C7, Rev. 4 Obj. 9 - Given a potential failure mode for a differential pressure cell used for level indication, describe how indicated level will be affected. c. Diaphragm leak

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

57. Given the following:

- The plant is operating at 75% power.
- Rod Control is in MANUAL.
- Control Bank D rods are at 190 steps.
- All Tavg channels are approximately 4°F higher than Tref.

Which of the following describes the rod control response if the Control Rod Bank Selector Switch is placed in AUTO prior to matching Tave and Tref?

1) Control rods will step at ______(1)______.

2) ______(2)______ Individual Rod Position Indication (IRPI) meters will indicate rod movement in response to this action?

A. 1) 40 SPM
2) 4

B. 1) 40 SPM
2) 8

C. 1) 56 SPM
2) 4

D. 1) 56 SPM
2) 8

Answer: B

Explanation/Justification: K/A is met by the candidate knowing that Control Bank D rods will drive inward at 40 steps per minutes when the control rod selector switch is manually selected to Auto, and displays the ability to monitor the Individual Rod Position Indication meters on VB-B when 8 rod position indication meters will indicate movement in response to the 4F Tavg mismatch.

A. Incorrect. 40 SPM is the rod speed when the mismatch is 4F. Second part is a plausible distractor because Control bank D has 8 Control rods broken into 2 groups of 4 rods. The candidate must recognize that both groups of rods (8) will be moving inward.

B. Correct. 40 SPM is the rod speed when the mismatch is 4F. Rod speed rises from a minimum of 8 SPM form 1.5-3F, to a maximum of 72 SPM when the mismatch is 5F or greater. 8 Bank D control rods will move (2 groups of 4 rods) when the selector switch is taken to Auto to correct the mismatch.

C. Incorrect. Plausible distractor if the candidate thinks the mismatch band starts at 1F and scales up to 5F with rod speed at 8 SPM at 1F, and 72 SPM at 5F, then 4F would be 56 SPM. This is incorrect because rod control has a 1.5F deadband in which rods do not drive until Tavg is 1.5F greater than Tref, and rods will move inward at 8 SPM until 3F is reached. Plausible distractor because Control bank D has 8 Control rods broken into 2 groups of 4 rods. The candidate must recognize that both groups of rods (8) will be moving inward.

D. Incorrect. Plausible distractor if the candidate thinks the mismatch band starts at 1F and scales up to 5F with rod speed at 8 SPM at 1F, and 72 SPM at 5F, then 4F would be 56 SPM. This is incorrect because rod control has a 1.5F deadband in which rods do not drive until Tavg is 1.5F greater than Tref, and rods will move inward at 8 SPM until 3F is reached. Second part is correct.

Sys #	System	Category	KA Statement
014	Rod Position Indication	A4 Ability to manually operate and/or monitor in the control room:	Rod selection control

K/A# A4.01 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.1D rev.3 Pg. 15
3SQS-1.3 Rev 7 Iss 1 PPNT slide 136
1OM-1.5.B.5 Iss. 2 Rev. 10 Table 1-5

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-1.3, Rev. 7 Obj. 23 - Discuss the Control Room indications and controls that are available to manipulate and monitor the Rod Control System.

58. The plant is at 100% power.

- Tref is 578°F
- Loop 1 Protection Tavg has failed to 620°F

Subsequently, TRB-1RC-432B1, Loop 3 Hot Leg RTD fails to 630°F.

- Loop Protection Tavgs now indicate as follows:
  - Loop 1 TI-1RC-412D is 620°F
  - Loop 2 TI-1RC-422D is 577°F
  - Loop 3 TI-1RC-432D is 580°F

When Loop 3 Hot Leg RTD fails to 630°F, what will be the HIGHEST Rod Speed indicated on SI-1RC-408?

A. 0 SPM

B. 8 SPM

C. 48 SPM

D. 72 SPM

Answer: B

Explanation/Justification: K/A is met by understanding of how individual loop Tavg is used, and that Median Tavg is used for auto Rod Control. In this question individual loop Tavg indications are given in the stem, and knowledge that the individual loop Tavgs undergo Median Select where the high and low Tavgs are rejected, and the Median passes through for use of the automatic Rod Control.

A. Incorrect. Plausible distractor if the 2F Tavg>Tref mismatch is recognized, but the candidate thinks that there is a 3F deadband instead of 1.5F deadband. At BV after the 1.5F deadband, rod speed is 8 SPM.

B. Correct. Due to median Tavg selector Tavg used by Rod Control will be 580F. Therefore the Tavg-Tref mismatch is 2F. Rod speed is 8 SPM between 1.5-3F mismatch. The candidate will have to know that Tavg median select is used by Rod Control, and the variable rod speeds in auto.

C. Incorrect. Plausible distractor if Tavg > Tref is recognized, but variable rod speed is not known. 48 SPM is the manual rod speed.

D. Incorrect. Plausible distractor if the Tavgs are averaged (620+577+580=1777/3=592F. This would be 14F mismatch. When Tavg > Tref by ≥5F the rod speed is 72 SPM.

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

K/A# A3.02 **K/A Importance** 2.9* **Exam Level** RO

References provided to Candidate None **Technical References:** 3SQS-1.3 Rev. 7 Iss. 1 PPNT slides 131 & 136
1OM-1.5.A.52 Rev. 1
1OM-6.4.IF Rev. 11 pg. 40

Question Source: New

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

Objective: 3SQS-1.3, Rev. 7 Obj. 10 - State the three modes of rod control, the rod step speeds for each, and how and when each mode is selected.

3SQS-1.3, Rev. 7 Obj. 20 - Determine how automatic rod control is affected when any of the process control input signals fail.

59. The plant has experienced a Reactor Trip and SI due to a LOCA and the following conditions exist:
    - The crew has transitioned to E-1, Loss of Reactor or Secondary Coolant
    - The Core Exit Thermocouples (CETCs) are reading as follows:
        - TWO CETCs are short circuited
        - THREE CETCs are 1204°F and rising
        - All other CETCs are reading between 950°F and 1150°F and rising

Which of the following completes the statements below?

The indication for the short circuited CETCs fail ______(1)______.

The ______(2)______ CETC(s) will be used to evaluate entry into FR-C.2, Response To Degraded Core Cooling.

	(1)	(2)
A.	high	hottest
B.	high	five hottest
C.	low	hottest
D.	low	five hottest

Answer: D

Explanation/Justification: K/A is met by demonstrating the knowledge of the effect that a shorted CETC will have on the incore temperature monitoring system.

A. Incorrect. Plausible if the candidate doesn't recall which direction a shorted CETC fails. Second part is incorrect. Plausible if the candidate doesn't recall that the FIVE hottest CETCs are selected to allow for failed high thermocouples. This is a common misconception.

B. Incorrect. Plausible if the candidate doesn't recall which direction a shorted CETC fails. Second part is correct.

C. Incorrect. First part is correct. Second part is incorrect. Plausible if the candidate doesn't recall that the FIVE hottest CETCs are selected to allow for failed high thermocouples. This is a common misconception.

D. Correct. Thermocouples that are shorted fail low. The status tree for Core Cooling uses the FIVE hottest CETCs for making a determination of which FRP is applicable for the conditions.

Sys #	System	Category	KA Statement
017	In-core Temperature Monitor	K6 Knowledge of the effect of a loss or malfunction of the following ITM system components:	Sensors and Detectors

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

References provided to Candidate None **Technical References:** 3SQS-3.1 Rev. 6 pg. 16; 1OM-53A.1.F-0.2 Iss. 3 Rev. 0

Question Source: Bank – Farley 2013 NRC Exam (Q19)

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

Objective: 3SQS-3.1 Rev 6 Obj. 10 - Describe the response of a thermocouple readout to open and short circuits.

3SQS-53.1, Rev. 2 Obj. 2 - Concerning critical safety function restoration, IAW BVPS EOP Executive Volume, state from memory the following: a. The CFS in the order of priority, b. The priorities of the color-coded end points of the CSF status trees, c. The red path summary conditions from the EOPs.

60. Which of the following components contribute to Iodine removal during the first 24 hours of a design basis accident?

    1) Sodium Tetraborate (NaTB) Baskets
    2) Quench Spray Pumps
    3) Outside Recirculation Spray Pumps
    4) Inside Recirculation Spray Pumps

A. 1 ONLY

B. 1 and 2 ONLY

C. 1, 3, and 4 ONLY

D. 1, 2, 3, and 4

Answer: D

Explanation/Justification: K/A is met by demonstrating the knowledge that Iodine is removed from the cnmt by Quench Spray using the RWST upon CIB actuation, and aided by both Inside and Outside Recirculation Spray Pumps after the CIB actuation and the RWST reaches the low level setpoint which starts the Recirculation Spray Pumps. The Sodium Tetraborate is dissolved in the sump water to raise the pH and sprayed by the RSS pump to assist in Iodine removal.

A. Incorrect. The Sodium Tetraborate is vital in the long term Iodine removal and corrosion control, but it alone does not remove the Iodine.
B. Incorrect. RWST water via the Quench spray pumps does remove Iodine, and Sodium Tetraborate aids in the removal, but Recirc spray pumps also play a vital role in Iodine removal.
C. Incorrect. Recirc spray pumps do spray the cnmt sump water containing Sodium Tetraborate to remove Iodine, but Iodine removal is also completed with the RWST water being sprayed via the Quench spray pumps.
D. Correct. The RWST is sprayed via the QS pumps. This water is acidic, but Iodine removal is successful. As cnmt sump level rises during a DBA, the Sodium Tetraborate dissolves in the water, and raises the pH to enhance the Iodine absorption capacity of the spray during recirculation from the cnmt sump. When RWST level reaches the low level setpoint (27'7.5") both the Inside and Outside RSS pumps will start and spray down cnmt with the alkaline sump water. The QS system is designed to cool and depressurize

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Sys #	System	Category	KA Statement
027	Containment Iodine Removal	K1 Knowledge of the physical connections and/or cause effect relationships between the CIRS and the following systems:	CSS

K/A# K1.01 **K/A Importance** 3.4* **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-13.1.B Rev. 3 pg. 2
1SQS-13.1 Rev. 15 pg. 3
Tech Spec 3.6.8 Bases pg.B3.6.8-1

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

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

1SQS-13.1 Rev 15 Obj. 7 - Given a Containment Depressurization System configuration and without reference material, describe the Containment Depressurization System field response to the following actuation signals, including automatic functions and changes in equipment status. Containment Isolation – Phase B

61. The plant has just entered Mode 5 for a refueling shutdown.

- A work party entered Containment to evaluate pump vibrations
- The Personnel Airlock CNMT side door manual equalizing valves 1VS-170 and 1VS-179 were left OPEN
- Containment Purge to the Ventilation Vent is in progress IAW 1OM-44C.4.A
- 1VS-D-5-5A, CNMT Isol Purge Supply Damper failed CLOSED due to an electrical fault

How will Containment pressure be affected, and how would the Emergency Squad enter Containment if needed?

CNMT will be at a ______(1)______ pressure.
Emergency Squad will enter Containment using the ______(2)______.

A. 1) negative
2) 18-inch Emergency Manway

B. 1) negative
2) Equipment Hatch Emergency Air Lock

C. 1) positive
2) 18-inch Emergency Manway

D. 1) positive
2) Equipment Hatch Emergency Air Lock

Answer: B

Explanation/Justification: K/A is met with the knowledge that a cnmt purge supply damper failing closed with the purge exhaust fan operating will cause cnmt pressure to go negative. With this differential pressure across the cnmt PAL (Normal egress), that the Equipment Hatch Emergency Air Lock will have to be used for cnmt entry.
Containment Purge to the Ventilation Vent may be accomplished with only the purge exhaust fan running, or with both the purge exhaust and supply fans running. The stem is set up that knowledge of the supply fan operation is not necessary since the supply damper has failed closed. The candidate will have to have knowledge of the cnmt purge flowpath including the existence of a supply fan and an exhaust fan since neither are mentioned in the stem, and whether the fans have any interlocks with the supply or exhaust dampers.

A. Incorrect. Negative pressure is correct. 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. Equalization is through a 1.5" line.

B. Correct. Prior to starting the cnmt purge system, cnmt vacuum is broken, and return to atmospheric pressure. When 1VS-D-5-5A, CNMT Isol Purge Supply Damper closes, the cnmt exhaust fan will continue to run which will take cnmt pressure sub-atmospheric (negative). There are no interlocks to trip the purge exhaust fan on cnmt pressure or damper position. With the Personnel Airlock CNMT Door Manual Equalizing valves open, personnel will not be able to equalize pressure across the PAB side of the air lock via the normal access door or the 18-inch Emergency Manway. Since pressure cannot be equalized, the only way into CNMT is through the Equipment Hatch Emergency Air Lock which has two doors and equalization capabilities to pass through the airlock.

C. Incorrect. Plausible distractor if the candidate is not familiar with the cnmt purge lineup, and believes the supply fan connects between the two cnmt supply dampers where the cnmt vacuum breaker connects. 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. Equalization is through a 1.5" line.

D. Incorrect. Plausible distractor if the candidate is not familiar with the cnmt purge lineup, and believes the supply fan connects between the two cnmt supply dampers where the cnmt vacuum breaker connects. Equipment Hatch Emergency Air Lock is correct.

Sys #	System	Category	KA Statement
029	Containment Purge	K3 Knowledge of the effect that a loss or malfunction of the Containment Purge System will have on the following:	Containment entry

K/A# K3.02 **K/A Importance** 2.9* **Exam Level** RO

References provided to Candidate None **Technical References:** 1SQS-44C.1 Rev. 10 Iss. 1 PPNT slide 46
U1 RM-0447-001 Rev. 11

Question Source: New

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

Objective: 1SQS-47.1, Rev. 9 Obj. 1 - Describe the control and interlock functions for the containment airlocks.

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

62. Given conditions:
   - Annunciator A7-4, Condenser Vacuum Low is LIT
   - PR-1CN-103, Condenser Backpressure Recorder is reading 5.2 In. Hg. Abs
   - PI-1AS-100, Auxiliary Steam Header Pressure is reading 105 psig and FLUCTUATING
   - The crew is performing AOP 1.26.2, Loss of Condenser Vacuum
   - AOP 1.26.2 directs the performance of ARP A2-81, Aux Steam Local Panel Trouble
   - No Vacuum Priming system annunciators are in alarm

What actions will be taken in ARP A2-81 to restore Condenser Vacuum?

A. Place a second set of Air Ejectors in service.

B. Place the Priming Ejectors (Hoggers) in service.

C. Start an additional Vacuum Priming Pump.

D. Take manual control of PCV-1AS-100, Main Steam Reducer.

Answer: D

Explanation/Justification: K/A is met with knowledge of the actions taken outside the control room iaw AOP and ARP procedures to restore condenser vacuum due to a malfunction of the auxiliary steam pressure control valve during a loss of Condenser Vacuum event.

A. Incorrect. Plausible distractor because two sets of air ejectors would assist in correcting a lowering vacuum issue, but it is not incorporated into the AOP or ARP, and the P&Ls of 1OM-26.4.F state that placing all 4 sets of air ejectors in service will overload the drain system and the air ejectors will not be able to maintain condenser vacuum.

B. Incorrect. Plausible distractor because the Priming Ejectors (Hoggers) are used for drawing initial vacuum in the condenser, but they are not used to assist in the recovery of lowering vacuum.

C. Incorrect. Plausible distractor as this would assist in restoring vacuum by raising water level in the circulating water lines and waterboxes, but it is an action taken in the Loss of Vacuum AOP only if there are Vacuum Priming system annunciators in alarm.

D. Correct. Normal auxiliary steam pressure is 135-160 psig. Loss of Vacuum AOP directs the use of ARP-A2-81 when auxiliary steam pressure is not normal. The stem does not state that 105 psig is low, therefore the candidate must know that this is not a normal aux steam pressure and that ARP A2-81 directs the Operator to take manual control of PCV-100 and raise pressure.

Sys #	System	Category	KA Statement
055	Condenser Air Removal	Generic	Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

K/A# 2.4.34 **K/A Importance** 4.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-27.4.AAJ Rev. 3 pg. 2-3
1OM-54.3.CRO1 Rev. 46 pg. 44-45

Question Source: New

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

Objective: 1SQS-26.1, Rev. 12 Rev. 12 Obj. 2 - List the nominal value of the field operating parameters associated with the Main Turbine, Main Condenser, Condenser Air Removal system, and Moisture Separator Reheaters.

1SQS-26.1, Rev. 12 Rev. 12 Obj. 8 - Given a Main Turbine, Main Condenser, Condenser Air Removal system, and Moisture Separator Reheaters alarm condition and using the Alarm Response Procedure(s), determine the appropriate alarm response, including automatic and operator actions in the field.

63. What is the basis for maintaining a minimum of 14 inches of water in LG-1GW-101, Loop Seal to Waste Gas Header?

A. Prevent an unmonitored radioactive release from the Degasifiers to the Gaseous Waste system.

B. Prevent Oxygen from entering the relief valve header and forming an explosive mixture with Hydrogen.

C. Prevent water from the Liquid Waste system from entering the Gaseous Waste system.

D. Prevent vacuum collapse of the Gaseous Waste Disposal Blower suction piping.

Answer: B

Explanation/Justification: K/A is met by demonstrating knowledge of the design feature of the loop seal which separates the Degasifiers, evaporators, and Unit 2 Gaseous Waste components from the Gaseous Waste Disposal Blower discharge piping to prevent oxygen from entering the tank relief valve header and forming an explosive mixture with hydrogen.

A. Incorrect. Plausible distractor with the Degasifiers (Boron Recovery System) relief valves discharging to the GW Header, but the release would be monitored if the relief valve lifted, and the loop seal is for preventing O2 from entering the relief header and mixing with Hydrogen.
B. Correct. Per 1OM-19.4.A P&L, Water level in [LG-1GW-101], Loop Seal to Waste Gas Header, shall be maintained at all times to prevent oxygen from entering the relief valve header and forming an explosive mixture with hydrogen.
C. Incorrect. Plausible distractor if it is not known that only the gasses from the Evaporators (Liquid Waste system) are discharged to the Gaseous Waste discharge header.
D. Incorrect. Plausible distractor if the loop seal location is not known in reference to the system flowpath. Preventing vacuum collapse of suction piping is a function of a loop seal, but in this case the location of the loop seal, and the purpose are both incorrect.

Sys #	System	Category	KA Statement
071	Waste Gas Disposal	K4 Knowledge of design feature(s) and/or interlock(s) which provide for the following:	Tank loop seals

K/A# K4.03 **K/A Importance** 2.5* **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-19.4.A Rev. 12 pg. 2
1SQS-19.1 Rev. 17 pg. 59
Unit 1 RM-0419-001 rev. 19

Question Source: New

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

Objective: 1SQS-19.1, Rev. 17 Obj. 22 - Given a set of plant conditions and the appropriate procedure(s), apply the operational sequence, parameter limits, precaution and limitations, and cautions & notes applicable to the completion of the task activities in the control room.

64. 1) Which of the following Radiation Monitors are used to assess Adverse Containment Criteria?

2) The meter on this Radiation Monitor reads in which of the following units?

A. 1) RM-1RM-219A, Containment High Range Area Monitor
2) R/HR

B. 1) RM-1RM-219A, Containment High Range Area Monitor
2) MR/HR

C. 1) RM-1RM-204, Incore Instrument Transfer Device Area Monitor
2) R/HR

D. 1) RM-1RM-204, Incore Instrument Transfer Device Area Monitor
2) MR/HR

Answer: A

Explanation/Justification: K/A is met by evaluating the candidates knowledge of which area radiation monitor is used to assess adverse conditions for the proper implementation of the EOPs, and what unit of measure is used in the radiation monitor.

A. Correct. Rad monitors RM-1RM-219A/B, cnmt pressure, and Integrated cnmt radiation are used in the EOP network to determine adverse cnmt conditions. The meter on RM-1RM-219A/B reads in R/Hr.
B. Incorrect. RM-1RM-219A is correct, but the meter reads in R/hr. Plausible distractor if the candidate doesn't know the detector type or scale.
C. Incorrect. Plausible distractor because this is a radiation monitor which alarms when RCS leaks are present inside cnmt, however the meter scale upper limit is 10^4 MR/HR. Second part is correct.
D. Incorrect. Plausible distractor because this is a radiation monitor which alarms when RCS leaks are present inside cnmt, however the meter scale upper limit is 10^4 MR/HR MR/HR is incorrect.

Sys #	System	Category	KA Statement
072	Area Radiation Monitoring	K5 Knowledge of the operational implications of the following concepts as they apply to the ARM system:	Radiation theory, including sources, types, units, and effects

K/A# K5.01 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.E-0 Iss. 3 Rev. 0 LHP
1OM-43.1.E Rev. 12 pg. 16

Question Source: New

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

Objective: 3SQS-53.2, Rev. 2 Obj. 15 - Define from memory adverse containment conditions, IAW BVPS EOP Executive Volume.
1SQS-43.1-01-1 - Describe the function of the Radiation Monitoring systems and the associated major components

65. To comply with 1/2-ADM-1900, Fire Protection Program, the Fire Suppression Water Distribution System must be maintained at ______(1)______ , and when the main fire header pressure lowers to ______(2)______, the Diesel Engine Driven Fire Pump, 1FP-P-2 will auto-start.

A. 1) 115 psig
   2) 95 psig

B. 1) 115 psig
   2) 105 psig

C. 1) 125 psig
   2) 95 psig

D. 1) 125 psig
   2) 105 psig

Answer: C

Explanation/Justification: K/A is met by demonstrating the knowledge of the Technical requirements of ½-ADM-1900 to maintain the fire header at 125 psig, and the ability to predict when the Diesel Engine Driven Fire Pump will automatically start to restore the fire header pressure to the design limit.

A. Incorrect. 115 psig is a plausible distractor because it is 10 psig less than the required pressure, and 10 psig above the auto start setpoint for the Motor driven Fire pump, which is the first fire pump to start as headed pressure lowers. 95 psig is the auto start for Engine Driven Fire Pump.
B. Incorrect. 115 psig is a plausible distractor because it is 10 psig less than the required pressure, and 10 psig above the auto start setpoint for the Motor driven Fire pump, which is the first fire pump to start as headed pressure lowers. 105 psig is plausible because it is the auto start setpoint for the Motor Driven Fire Pump.
C. Correct. 125 psig is correct as required by ADM-1900 Att. B sect. 2. The Diesel Engine Driven Fire Pump does auto start at 95 psig.
D. Incorrect. 125 psig is correct as required by ADM-1900. 105 psig is plausible because it is the auto start setpoint for the Motor Driven Fire Pump.

Sys #	System	Category	KA Statement
086	Fire Protection	A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Fire Protection System operating the controls including:	Fire header pressure

K/A# A1.01 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1/2-ADM-1900 Rev. 38 pg. 45
1OM-33.2.B Rev. 14 pg. 5

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.5 / 45.5)

Objective: 3SQS-33.1-1: From memory and/or using supplied reference materials, summarize the Fire Protection System design basis, major components, flow paths, controls and interlocks, transient response, and normal and abnormal operations, in accordance with established design basis, station policies, 1/2-ADM-1900 and operations procedures.

66. Which of the following describes the system or component status for a value displayed in YELLOW on the In-plant Computer?

A. Denotes a static OR reference value.

B. Denotes a dynamic OR important value.

C. Indicates that the data quality value is NOT good.

D. Indicates that a process value has exceeded an ALARM setpoint.

Answer: C

Explanation/Justification: K/A is met demonstrating the ability to interpret the In-plant Computer parameter color coding to determine that the input data is not good for a parameter.

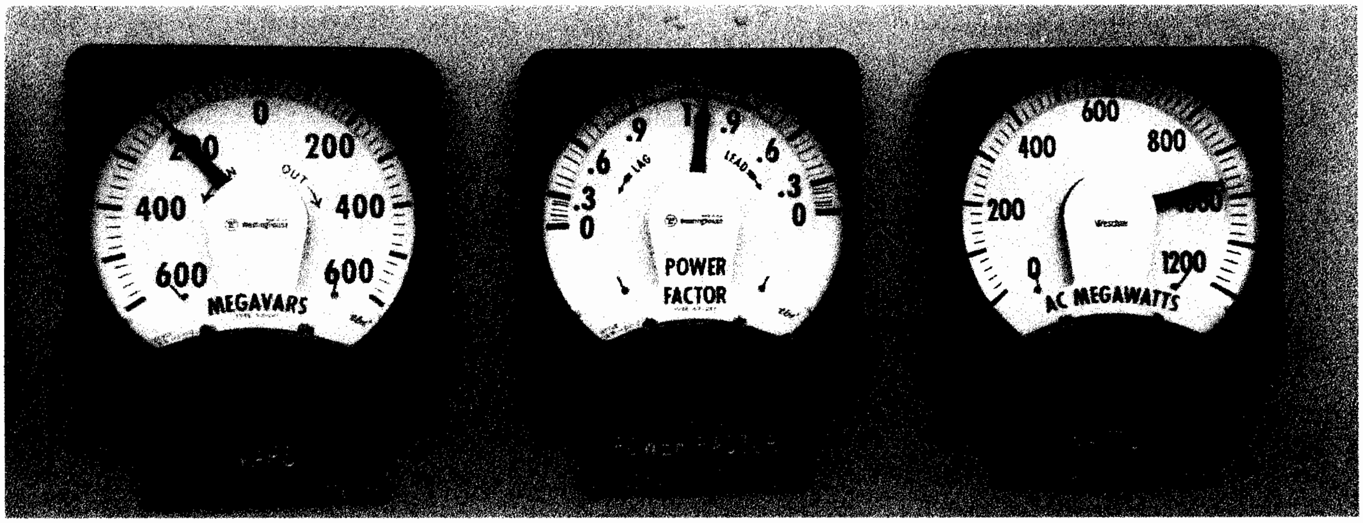
A. Incorrect. This would be a blue indication.
B. Incorrect. This would be a green indication.
C. Correct. Yellow indicates that the data quality value is not good.
D. Incorrect. This would be a red indication.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to use plant computers to evaluate system or component status.

K/A# 2.1.19 **K/A Importance** 3.9 **Exam Level** RO
References provided to Candidate None **Technical References:** 1OM-5C.1.B Rev. 0 Iss. 4 pg 2
Question Source: Bank – 1LOT7 NRC Exam (Q66)
Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 45.12)

Objective: 1SQS-5A.2, Rev. 2 Obj. 6 - Given an IPC SDS, determine when a computer point changes state, or goes into or out of alarm per guidance provided.

67. Given the following conditions:
    - Plant is operating at 100% power
    - Main Generator hydrogen gas pressure is 60 psig and STABLE
    - Main Generator Voltage Regulator is in service



Based on the above indications, and using Figure 35-14, "Main Generator Capability Curve, what actions, if any, need to be taken?

(Reference Provided)

A. No actions are required. The Generator is within Normal Operating Limits.

B. Raise Hydrogen Gas pressure to 75 psig.

C. Adjust the Main Generator Voltage Adjuster control switch to 100 VARS OUT.

D. Reduce Generator Load to less than 940 MW.

Question 67

Answer: C

Explanation/Justification: K/A is met by demonstrating the ability to interpret the main generator capability curve given plant parameters requiring adjustment to the generator voltage adjust to raise MVARs into the overexcited region of the curve. The candidate must evaluate actual plant meters, gas pressure, and MW loading to make a determination of proper generator operation.

A. Incorrect. Plausible if the candidate doesn't recognize that the generator is underexcited, and plots it on the capability curve as a lagging /overexcited generator.
B. Incorrect. Plausible because raising Hydrogen gas pressure to 75 psig would put it inside the operating curve, but the generator is still underexcited and this is not permitted by procedure.
C. Correct. Adjusting the Main Generator Voltage Adjuster control switch to 100 VARS OUT would put the generator within the capability curve and in the overexcited region of the curve.
D. Incorrect. Plausible because by lowering generator load, it would bring the generator within the operating limits of the gas curve, but the generator is underexcited.

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

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

References provided to Candidate 1OM-35.5.A.14 Rev. 5 **Technical References:** 1OM-35.4.B Rev. 16 pg. 2 & 4; 1OM-35.5.A.14 Rev. 5

Question Source: New

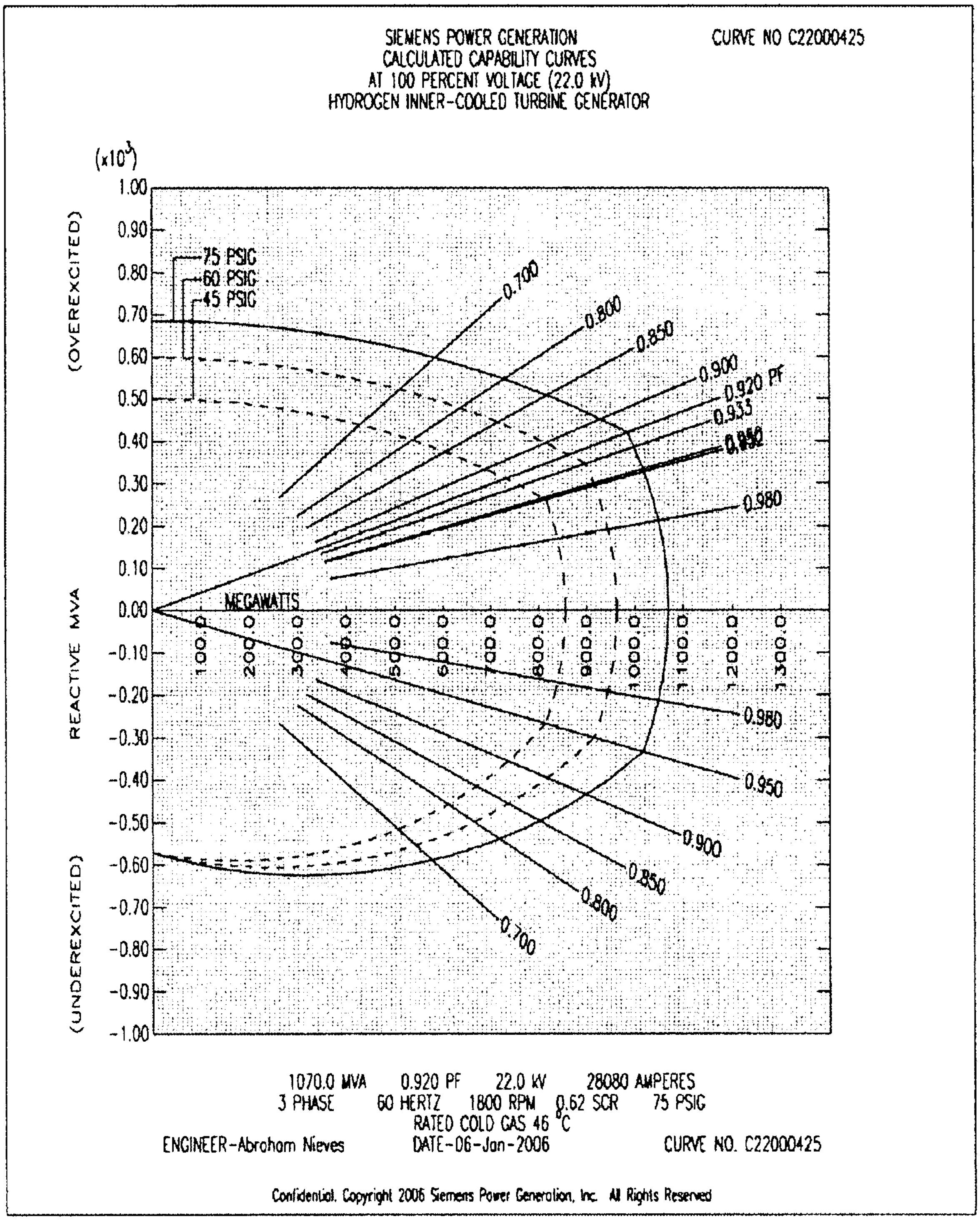
Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 45.12)

Objective: 1SQS-35.3, Rev. 9 Obj. 6 - 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.

Q67

Beaver Valley Power Station Main Generator and Transformer (GO) Figures and Tables	**Unit 1**	1OM-35.5.A.14 Revision 5 Page 2 of 2

Figure 35-14 - Main Generator Capability Curve



68. When in Modes 1 and 2, the Reactor Core Safety Limits in section 2.0 of Technical Specifications include limits on ______(1)______.

Automatic enforcement of the Reactor Core Safety Limits is provided by the appropriate operation of the ______(2)______.

A. 1) Highest Loop Tave
2) Main Steam Safety Valves AND Reactor Protection System

B. 1) Average of the 5 highest Core Exit Thermocouples
2) ONLY the Main Steam Safety Valves

C. 1) Highest Loop Tave
2) ONLY the Main Steam Safety Valves

D. 1) Average of the 5 highest Core Exit Thermocouples
2) Main Steam Safety Valves AND Reactor Protection System

Answer: A

Explanation/Justification: K/A is met by requiring the knowledge of what parameters are includes in the Safety Limits, and the operational components/systems required to prevent exceeding the SLs as described in the Tech Spec Bases.

A. Correct. Safety Limits 2.1.1 states "In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR". Core SLs are maintained by RPS and MSSVs.
B. Incorrect. CETs are a plausible distractor with peak fuel centerline temperature SL at ≤4700 F, and BOTH RPS and MSSVs are credited for maintaining the Safety Limits.
C. Incorrect. Highest loop Tavg is included in the SLs, but BOTH RPS and MSSVs are credited for maintaining the Safety Limits.
D. Incorrect. CETs are a plausible distractor with peak fuel centerline temperature SL at ≤4700 F. It is correct that both RPS and MSSVs are credited for maintaining the Safety Limits.

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

K/A# 2.2.25 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** Tech Specs - SLs 2.0 Amend 278/161 pg 2.0-1
Tech Spec Bases SL 2.0 Rev. 0 pg. B 2.1.1 - 2

Question Source: Bank – Braidwood 2013 NRC Exam (Q69)

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.5 / 41.7 / 43.2)

Objective: 3SQS-SL ITS-01-01: State the purpose of each ITS Safety Limit as described in the Applicable Safety Analyses section of the ITS Bases.

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

- The plant is in Mode 6 with core reload in progress
- The crew has completed Train swap to the "A" Train after completing the operability run on the #1 EDG
- #2 EDG has been placed on clearance for maintenance activities
- Several hours later the control room receives a report that after reviewing the maintenance work order for #1 EDG, incorrect gasket material installation makes #1 EDG inoperable

Which of the following TS 3.8.2, "AC Sources – Shutdown" LCO action(s) is(are) **immediately** required?

1) Suspend Core Alterations.
2) Suspend operations involving positive reactivity additions that could result in a loss of shutdown margin or boron concentration.
3) Initiate action to restore required EDG to operable status.

A. 1 ONLY

B. 1 and 2 ONLY

C. 2 and 3 ONLY

D. 1, 2, and 3.

Answer: D

Explanation/Justification: K/A met by recognizing that in mode 6, at least 1 DG must be available to supply one train of power, then, from memory identify the required actions taken since the limiting conditions for operations is not met.

A. Incorrect. (refer to correct answer explanation)
B. Incorrect. (refer to correct answer explanation)
C. Incorrect. (refer to correct answer explanation)
D. Correct. The RO candidate must be able to analyze the effect of maintenance activities on the EDG and determine the status of LCOs for TS 3.8.2. The ROs are expected to know the LCO statements and associated applicability information (ie: the information above the double lines separating actions from the LCO and associated applicability statements). ROs are also required to know ≤ 1 hour action statements. Based on stated plant conditions TS 3.8.2 requires one EDG capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystems during modes 5 & 6. Since there are no operable EDGs, TS 3.8.2 Condition B requires all of the actions above to be immediately performed.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

K/A# 2.2.36 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** TS 3.8.2 Amend. 278/161, pg. 3.8.2-1 & 3

Question Source: Bank – 2LOT8 NRC Exam (Q70)

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 43.2 / 45.13)

Objective: 3SQS-ELECT ITS Rev. 1- Obj. 4. Given plant conditions that constitute non-compliance with any electrical power systems LCO, or LRM, determine the applicable condition(s), required action(s), and associated completion times.

70. Given the following plant conditions:

- The Shift Manager has determined that an Annunciator alarm should be disabled due to a failure
- The corresponding knife switch for this alarm has been repositioned OPEN
- Repairs are scheduled to take 36 hours

Which of the following are required by NOP-OP-1014, Plant Status Control to track this inoperable alarm?

1. Write a Notification for the condition that required the Annunciator alarm to be disabled.
2. Post a Maintenance Deficiency (green) sticker on the alarm window.
3. Hang a Caution Tag on the opened knife switch stating what alarm was removed from service.

A. 1 ONLY

B. 1 and 2 ONLY

C. 2 and 3 ONLY

D. 1, 2, and 3

Answer: D

Explanation/Justification: K/A is met by demonstrating the knowledge of the requirements of NOP-OP-1014 that must be applied to track inoperable alarms.

A. Incorrect. Only partially correct. Plausible distractor if the candidate thinks the shift manager can authorize an annunciator disabled with no configuration controls in place.
B. Incorrect. Only partially correct. Plausible distractor if the candidate thinks that short term configuration control does not require a caution tag unless the switch is out of position for 48 hrs.
C. Incorrect. Only partially correct. Plausible distractor if the candidate thinks that only configuration control is required to disable an annunciator.
D. Correct. In accordance with NOP-OP-1014 section 4.12.1, all of these choices are required.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Generic	Knowledge of the process used to track inoperable alarms.

K/A# 2.2.43 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-OP-1014, Rev. 4, pg. 26

Question Source: Bank – 1LOT8 NRC Exam (Q70)

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

Objective: 3SSG-ADMIN Rev.9 Term. Obj. - Using the appropriate FENOC Administrative Procedures, apply knowledge gained to improve the performance of the BVPS Units.

71. You are going into a contaminated area, which has the following radiological characteristics to perform a valve lineup.

- Your current exposure for the year is 938 mrem
- The RWP states:
  - General area dose rate = 30 mrem/hr
  - Airborne contamination concentration = 10.0 DAC
- The valve lineup will take you 2 hours if you wear a full-face respirator.
- The valve lineup will only take you 1 hour if you do **NOT** wear the respirator.

1) Which of the following choices for completing this job would maintain your exposure within the station administrative requirements and the principles of ALARA?

2) Why is this action appropriate?

A. 1) You should **NOT** wear the respirator.
2) Your calculated TEDE dose received will be less than if you do wear a respirator.

B. 1) You should **NOT** wear the respirator.
2) Your dose received wearing a respirator will exceed the site annual personnel dose limits.

C. 1) You must wear the respirator.
2) You will exceed DAC limits if you do **NOT** wear a respirator.

D. 1) You must wear the respirator.
2) Your calculated TEDE dose received will be less than if you do **NOT** wear a respirator.

Answer: A C is the correct answer, not A

Explanation/Justification: K/A is met by demonstrating the ability to comply with an RWP to determine dose received with or without a respirator to achieve the lowest possible dose for a job.

A. Correct. Without respirator: TEDE = 30 mrem/hr x 1 hr = 30 mrem, From airborne contamination: TEDE = 10 DACx1 hr x 2.5 mrem/DAC-hr = 25 mrem, TEDE = 30 + 25 = 55 mrem from job, Total exposure for year = 938 + 55 = 993 mrem
With respirator, TEDE = 30 mrem/hr x 2 hr = 60 mrem TEDE = 60 mrem, Total exposure for year = 938 + 60 = 998 mrem
TEDE = 60 mrem-vs-55 mrem = do not use a respirator.

B. Incorrect. This answer is plausible if the applicant miscalculates the dose.

C. Incorrect. This answer is plausible if the applicant does not understand the concept of DAC-hours and DAC-hour limits.

D. Incorrect. This answer is plausible if the applicant incorrectly calculates the exposure.

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-4201 Rev. 2 pg. 20
FENRWT Rev 3 CNRR 08-08-14 Handout pg.26 & 64

Question Source: Bank – 2LOT15 NRC Exam (Q71)

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.12 / 45.10)

Objective: FENRWT Rev 3, Chap 5 Obj. 7 - Calculate stay time given a dose rate, current dose, and a dose limit, and Chap. 8 Obj. 6 - State the relationship among DACs, ALIs, CEDE, and TEDE (DAC and mrem/hr relationship).

72. A liquid waste discharge is in progress from 1BR-TK-4B, Coolant Recovery Tank, to the cooling tower blowdown, when RM-1LW-104, Liquid Waste Effluent Monitor, fails HIGH.

Which of the following describes the resulting positions of the following liquid waste system components?

TV-1LW-105, Liquid Waste Effluent Trip Valve
FCV-1LW-104-2, Liquid Waste Effluent Flow Control to Cooling Tower Blowdown

	TV-1LW-105	FCV-1LW-104-2
A.	OPEN	OPEN
B.	OPEN	CLOSED
C.	CLOSED	OPEN
D.	CLOSED	CLOSED

Answer: D

Explanation/Justification: K/A is met by demonstrating the ability to verify which components will close to terminate a discharge in the event of a failed radiation monitor.

A. Incorrect. Plausible distractor is the candidate thinks there are no automatic actions which occur on RM-1LW-104 High-High setpoint, or does recognize that a RM failing High will actuate the auto valve closures.
B. Incorrect. Plausible if the candidate thinks that only the FCV will isolate on High-High alarm, and the Trip valve (105) will remain open.
C. Incorrect. Plausible if the candidate thinks that only the Trip valve (105) will isolate on High-High alarm, and the 104-2 will remain open.
D. Correct. When RM-1LW-104 fails high, the High-High alarm setpoint will initiate a signal to close TV-1LW-105, FCV-1LW-104-1, and FCV-1LW-104-2 to isolate discharge flow to the Cooling Tower Blowdown.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Generic	Ability to control radiation releases.

K/A# 2.3.11 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-43.1.E Rev. 12 pg. 4 & 5

Question Source: Bank – 1LOT7 NRC Exam (Q50)

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.11 / 43.4 / 45.10)

Objective: 1SQS-17.1, Rev. 15 Obj. 16 - Describe the control, protection and interlock functions for the control room components associated with the Liquid Waste Disposal System, including automatic functions, setpoints and changes in equipment status as applicable.

73. Given the following conditions, you are required to post a clearance inside the VCT Cubicle.

- The VCT Cubicle is posted a "Locked High Radiation Area".
- Your present annual exposure is 700 mrem.
- The general area you will be working in is at the MINIMUM radiation level for LHRA posting.

What would be your maximum calculated stay time in this area in order to avoid exceeding the Site Administrative Control Level (ACL) Limit for radiation exposure per NOP-OP-4201, Routine External Exposure Monitoring?

A. 18 minutes

B. 1 hour

C. 3 hours

D. 4 hours 18 minutes

Answer: A

Explanation/Justification: K/A is met by requiring the candidate to know the minimum dose rate for a Locked High Radiation Area, and the site ACL of 1000 mr, then apply the radiological safety principle of time when calculating stay time for a Locked High Radiation Area entry for posting a clearance.

A. Correct. Stay time= (Dose Limit – Current Dose)/Dose rate in the work area. Site ACL is 1000mr. Locked High Radiation Area (LHRA) - radiation levels could result in dose rates ≥1,000 mr/hr.
B. Incorrect. Plausible distractor if it is not recognized that 700 mr has already been received for the year, and has not been subtracted from the 1000 mr allowed for the year.
C. Incorrect. Plausible distractor if 100 mr/hr is used for the dose rate in the work area. This would be the minimum dose rate for a High Radiation Area posting.
D. Incorrect. Plausible distractor if 5000mr is used for the ACL. This is the 10CFR20 limit for TEDE, not the site ACL.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

K/A# 2.3.12 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-OP-4201 Rev. 2 pgs. 3 & 20
NOP-OP-4102 Rev. 11 pg. 5
FEN-RWT-05 Rev. 4 pg. 5

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.12 / 45.9 / 45.10)

Objective: FEN-RWT-05 Rev. 4 Obj. 3 - Explain how time may be used to reduce dose, and state methods to implement this concept.

FEN-RWT-05 Rev. 4 Obj. 7. Calculate stay time given a dose rate, current dose, and a dose limit.

FEN-RWT-05 Rev. 4 Obj. 8. Define and recognize the following radiological area postings: d. Locked High Radiation Area

74. The plant is operating at 100% power.

- A Small Break LOCA occurs
- The crew is performing the actions of ES-1.2, "Post LOCA Cooldown and Depressurization
- All SI pumps are running
- All RCPs are running
- RCS cooldown via Condenser Steam Dumps is ongoing
- RCS Tcold is 510°F and lowering at a rate of 50°F/Hr
- RCS pressure is 1350 psig and stable
- Pressurizer (PRZR) level indicates 38% and rising

Which of the following describes the **NEXT MAJOR** action to be implemented in the EOP to mitigate the current conditions?

A. Stop RCP's NOT needed for PRZR Spray and begin the SI flow reduction sequence by stopping ECCS pumps.

B. Transition to ES-1.1, SI Termination and begin the SI flow reduction sequence by stopping ECCS pumps.

C. Repressurize the RCS to maximize RCS subcooling.

D. Stop the cooldown. Energize all PRZR heaters to collapse voids and stabilize PRZR level.

Answer: A

Explanation/Justification: K/A is met by demonstrating the knowledge of the major action steps of ES-1.2, Post LOCA Cooldown and Depressurization to mitigate the event.

A. Correct. The depressurization to raise przr level to >31% is complete (major action step 2), therefore the next major action step is to stop all but one RCP and then reduce RCS injection flow (steps 3 & 4).

B. Incorrect. ES-1.1 is a plausible distract since it terminates SI. In the case of ES-1.2, the steps to terminate Si are incorporated in EOP steps 18-21.Candidate must know that reducing SI is a major action of the procedure.

C. Incorrect. Subcooling is sufficient as identified in the stem, but the candidate will have to make that determination. Plausible because several steps in ES-1.2 check subcooling, and if it is not sufficient, then the procedure directs realigning safety Injection (step 26).

D. Incorrect. ES-1.2 cools the plant down to mode 5 conditions so there is no need to stop cooldown unless 100F/hr was exceeded. No voids exist at the current time with RCPs running. This would be performed if a void existed in ES-0.2 or ES-0.3.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of EOP mitigation strategies.

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

References provided to Candidate None **Technical References:** 1OM-53A.1.ES-1.2 Iss. 3, Rev. 1 pgs. 1, 10-12

Question Source: Bank – 2LOT15 NRC Exam (Q73)

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

75. Given the following plant conditions:

- The plant is at 100% power
- A3-53, VOLUME CONTROL TANK LEVEL HIGH-LOW is LIT
- LI-1CH-115 is 85% and RISING
- LI-1CH-112 is 25% and LOWERING

Based on the above indications, which VCT Level transmitter has failed, and what action is required to be taken to stabilize the VCT Level?

1) ____(1)____ has failed.

2) VCT level is stabilized by placing the ____(2)____.

A. 1) LT-1CH-112
2) Volume Control Tank Level Control Selector Switch in the VCT position

B. 1) LT-1CH-112
2) Boric Acid Blender Control Switch in the STOP position

C. 1) LT-1CH-115
2) Volume Control Tank Level Control Selector Switch in the VCT position

D. 1) LT-1CH-115
2) Boric Acid Blender Control Switch in the STOP position

Answer: C

Explanation/Justification: K/A is met by demonstrating the knowledge of the VCT level high (76%) and Low (15%) alarm setpoints and the ability to stabilize the VCT as the system automatically responds to LT-115 failing high by operating the Volume Control Tank Level Control Selector Switch as required by the Alarm Response Manual.

A. Incorrect. First part is plausible if the candidate thinks that LT-112 failed low causing auto makeup to occur. This is incorrect because auto makeup is initiated from LT-115 at 20%. Second part is correct because LT-115 is above the emergency divert level of 81%.

B. Incorrect. . First part is plausible if the candidate thinks that LT-112 failed low causing auto makeup to occur. This is incorrect because auto makeup is initiated from LT-115 at 20%. Second part is plausible if the candidate thinks the auto makeup is causing LT-115 to rise.

C. Correct. LT-115 has failed high based on annunciator A3-52, VCT level high triggered at 76% (this alarm is only activated by LT-115, LT-112 has no input to the annunciator). Indications given indicate LT-115 failing high which in turn will emergency divert to the Degasifiers at 81% on LT-115, as the VCT is diverted, LT-112 will lower due LCV-112 & 115A diverting all flow to the Degasifiers , and the charging pump still maintaining normal flow from the VCT. Selecting the Volume Control Tank Level Control Selector Switch in the VCT position will stabilize the VCT level.

D. Incorrect. First part is correct. Second part is plausible if the candidate thinks the auto makeup is has started due to low level on LT-112. This would not be correct because LT-115 starts auto makeup at 20%.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

K/A# 2.4.50 **K/A Importance** 4.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-7.4.AAX Rev. 4 pg. 4; 1OM-7.4.IF Rev. 3 pg. 9

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.3)

Objective: 1SQS-7.1, Rev. 20 Obj. 6 - Given a change in plant conditions, describe the response of the Chemical and Volume Control System field indication and control loops, including all automatic functions and changes in equipment status.

1SQS-7.1, Rev. 20 Obj. 10 - Given a Chemical and Volume Control System alarm condition, and using the Alarm Response Procedure(s), determine the appropriate alarm response, including automatic and operator actions in the field.

76. The plant was operating at 100% when the crew tripped the Reactor due to indications of a SG tube rupture.

- Safety Injection is actuated
- All RCPs are running
- The crew has transitioned to E-3, Steam Generator Tube Rupture
- The crew is performing step 4, Isolating flow from the ruptured SG

Annunciator A3-76, Reactor Cool Pump Motor Bearing Temp High Alarms

- TI-1RC-417A, 'A' RCP Upper Thrust Bearing Temperature indicates 198ºF and slowly RISING
- The crew has taken the required actions to address the 'A' RCP Upper Thrust Bearing Temperature

Based on the above conditions, which of the following combinations of PRZR spray flow and PORVs will be used to depressurize the RCS to restore inventory?

	Spray Valve(s)	PORV(s)
A.	1	1
B.	1	3
C.	2	1
D.	2	3

Question 76

Answer: A

Explanation/Justification: Meets NUREG-1021 Rev. 10, Att.2 Sect. II E page 21 third bullet. SRO is required to have knowledge of the content of the procedure versus knowledge of the overall mitigative strategy or purpose Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures. Specifically the SRO must make a determination that the RCP trip criteria in AOP-1.6.8 have been exceeded, and know the pump is no longer running, then have detailed knowledge of E-3 depressurization step to decide that 1 spray valve and 1 PORV is procedurally required with these plant conditions.

K/A is met by giving a CCW cooled component ('A' RCP Upper Thrust Bearing) a high temperature alarm with exceeds the immediate shutdown criteria of AOP-1.6.8. Then they must determine that 'A' RCP will no longer be running when the E-3 depressurization continues which removes one spray valve for use of depressurization.

A. Correct. The SRO must recognize that AOP 1.6.8 must be entered due to the 'A' RCP motor bearing temperature being >195F. This meets the RCP shutdown criteria of AOP-1.6.8. With the 'A' RCP shutdown, only PCV-1RC-455B ('C' RCP) spray valve is available. PCV-1RC-455A ('A' RCP) spray valve is not available and would slow down depressurization if opened to spray the PRZR. Only 1 PORV is used to augment the Spray valve depressurization in E-3 because of the loss of RCS coolant to the PRT which could rupture during depressurization.

B. Incorrect. First part is correct. Second part is plausible distractor because 3 PORVs are opened to depressurize the RCS when establishing RCS Bleed and Feed in FR-H.1.

C. Incorrect. First part is plausible if the candidate doesn't recognize that AOP-1.6.8 RCP trip criteria for 'B' RCP has been exceeded, or doesn't know which spray valves are for the RCPs. Second part is correct.

D. Incorrect. First part is plausible if the candidate doesn't recognize that AOP-1.6.8 RCP trip criteria for 'B' RCP has been exceeded, or doesn't know which spray valves are for the RCPs. Second part is plausible distractor because 3 PORVs are opened to depressurize the RCS when establishing RCS Bleed and Feed in FR-H.1.

Sys #	System	Category	KA Statement
000026	Loss of Component Cooling Water / 8	AA2. Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water:	The normal values and upper limits for the temperatures of the components cooled by CCW

K/A# AA2.04 **K/A Importance** 2.9* **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.6.8 Rev. 18 pg. 3 & LHP
1OM-53A.1.E-3 Iss. 3 Rev. 0 pg. 15

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

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

77. The plant was operating at 100% power when an event occurred which caused the following indications 10 minutes after the automatic Reactor Trip.

CNMT pressure – 14.4 psig and RISING
PRZR level – 27.9% and RISING
CETs – 504°F and LOWERING
CNMT sump level – 11.7 inches and RISING
AFW flow 'A' SG- 300 gpm 'B' SG - 300 gpm 'C' SG - 300 gpm and all are STABLE
SG 'A' pressure – 673 psig and LOWERING WR level – 50.9% and RISING
SG 'B' pressure – 603 psig and LOWERING WR level – 14.8% and LOWERING
SG 'C' pressure – 684 psig and LOWERING WR level – 51.4% and RISING

Based on the above indications, what EAL classification level should be declared by the Shift Manager?

(REFERENCE PROVIDED)

A. FU1

B. FA1

C. HU4

D. HA4

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II E page 21 third bullet. SRO is required to have knowledge of the Emergency Classifications. This is a SRO position function only.

K/A is met by requiring the candidate to interpret the plant conditions and apply their knowledge of the EAL classifications to determine the correct classification for a Steam Line break inside of Containment.

A. Incorrect. Plausible because the candidate will evaluate the CT column of the Fission Product Barrier Degradation and see that cnmt pressure > 11 psig with less than 1 full train of depressurization equipment available is a potential loss. Nothing in the question stem infers that containment depressurization equipment is not available.
B. Incorrect. Plausible because the candidate may feel that heat sink is lost with NR level in all SGs is <50%, or since SI actuated a leak greater than the capacity of one charging pump has occurred, or they may question if core cooling Orange path has been met.
C. Incorrect. Plausible because a steam line break is entry criteria for HU4 based on the note, but in the case of this question, the break is inside containment, which is why this would be an incorrect classification.
D. Correct. Alert under Fire/Explosion. Indications show that a steam line break is inside containment, therefore, since containment is an area identified on Table H-1, this would be the correct classification.

Sys #	System	Category	KA Statement
000040	Steam Line Rupture – Excessive Heat Transfer / 4	Generic	Knowledge of the emergency action level thresholds and classifications.

K/A# 2.4.41 **K/A Importance** 4.6 **Exam Level** SRO
References provided to Candidate EPP Chart **Technical References:** EPP-PLAN-SECTION-4 Rev. 30 pgs. 4-98 & 4-99
EPP-I-1a Att. A Rev. 17 pg. 21 of 28

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.11)

Objective: EPP-9281-01-11:Given specific plant conditions, classify the condition in accordance with EPP I-1 a & b.

78. The plant is at 37% power.

- FCV-1FW-478, 1A Steam Generator Main Feedwater Regulating Valve, fails and sticks full open
- NO Operator actions have been taken

Ten minutes after this failure, the Reactor ______(1)_______ be tripped.

The bases for the automatic closure of the Main Feedwater Regulating valves is to prevent ______(2)_______.

A. 1) will NOT
2) an excessive mass addition to the Steam Generator during a Steamline Break event

B. 1) will NOT
2) a radiological release through the feedline penetration during a Steam Generator Tube Rupture event

C. 1) will
2) an excessive mass addition to the Steam Generator during a Steamline Break event

D. 1) will
2) a radiological release through the feedline penetration during a Steam Generator Tube Rupture Event

Question 78

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II B page 17 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 Main Feedwater Regulating Valves provide secondary isolation to the Main Feedwater Isolation valves in the event of a High Energy Line Break thereby limiting mass and energy release for Steamline Breaks.

K/A is met by the candidate determining that the 'B' MFRV failing open will raise the Steam Generator water level to the high setpoint (P-14) and generate a Feedwater Isolation Signal (FWI), causing the Turbine to trip, Main Feedwater Pumps to trip and close all Feedwater Isolation Valves. The loss of main feedwater will generate an auto start of the Aux Feedwater pumps however there is inadequate feedwater flow to maintain SG water levels at power. Steam Generator water level to lower to the Reactor Trip Setpoint.

A. Incorrect. Plausible if the candidate thinks that the MFRV failing open generates only a Turbine trip when power is <49% (P-9) and does not result in a Reactor Trip due to lowering Steam Generator water level. Second part distractor is correct per the Tech Spec 7.7.3 Bases.

B. Incorrect. Plausible if the candidate thinks that the MFRV failing open generates only a Turbine trip when power is <49% (P-9) and does not result in a Reactor Trip due to lowering Steam Generator water level. Second part distractor is plausible based on the incorrect assumption that valves that connect to the Steam Generator must be isolated during a SGTR event due to potential backflow through the feed lines.

C. Correct. Due to the MFRV failing open, Steam Generator water level will rise to the P-14 setpoint (89.7%) causing the MFWPs to trip, the Turbine to trip and FWIVs to close. The motor driven Aux Feedwater Pumps will start when the Main Feedwater pump trips, however they do not have the capacity to maintain SG water level above the SG low level trip with the plant remaining at power. The Tech Spec Bases for the MFRV closure is to prevent an excessive mass addition to the Steam Generator during a Steamline Break event, reducing the cooldown

D. Incorrect. The reactor will trip following the opening of the MFRV due to low SG water level. Second part distractor is incorrect but plausible based on the incorrect assumption that valves that connect to the Steam Generator must be isolated during a SGTR event due to potential backflow through the feed lines.

Sys #	System	Category	KA Statement
000054	Loss of Main Feedwater / 4	AA2. Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW):	Occurrence of reactor trip and/ or turbine trip

K/A# AA2.01 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-24.4.IF Rev.14 pg. 14
1OM-24-4-AAA Rev 4 pg. 2
T.S. Bases 3.7.3 pg. B3.3.7.3-1

Question Source: Bank - Farley 2013 NRC Exam (Q84) Modified

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

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

3SQS-PLTSYS ITS Rev. 2 Obj. 2 - State the purpose of each Plant Systems System specification as described in the Applicable Safety Analyses section of the Bases.

79. The following conditions exist:
    - A Station Blackout has occurred
    - ECA-0.0, Loss of All Emergency 4KV AC Power is in progress
    - The crew has commenced feeding and depressurizing the SGs after a significant delay due to a TDAFW pump malfunction
    - RCS pressure is 1675 psig and very slowly RISING
    - ALL CETCs are 735°F and RISING
    - Hot Leg temperatures are 612°F and RISING
    - Cold Leg temperatures are 585°F and LOWERING
    - SG pressures are 885 psig and LOWERING

    Which one of the following completes the statements below?

    Natural circulation ______(1)______ established.

    The crew is required to implement FR-C.2, Response to Degraded Core Cooling, ______(2)______.

	(1)	(2)
A.	is	immediately upon exit from ECA-0.0
B.	is	after restarting equipment in ECA-0.2, Loss of All AC Power Recovery With SI Required
C.	is NOT	immediately upon exit from ECA-0.0
D.	is NOT	after restarting equipment in ECA-0.2, Loss of All AC Power Recovery With SI Required

QUESTION 79

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II E page 21 second bullet. SRO is required to have knowledge of specific procedure steps, notes, and cautions which help in making decisions for procedure transitions. This is a SRO position function only.

K/A is met by requiring the candidate to interpret control room indications of Natural Circulation verification during an ECA-0.0 event, and that a consequence of a loss of natural circulation is Core Cooling being challenged. They also must understand how operator actions of implementing ECA-0.0 defeats the automatic loading of major equipment, and steps 1-13 of ECA-0.2 starts ESF equipment as needed and has priority over the Core Cooling FRP. Transition to Procedure ECA-0.2 is required since subcooling requirements are not met.

A. Incorrect. Plausible to think Natural circulation is established with SG pressures lowering and Cold leg temperatures lowering, but other requirements of Natural Circulation are not met. Plausible to transition to FR-C.2 after ECA-0.0 is complete, but this would be incorrect because ECA-0.0 defeats the automatic loading of major equipment, and steps 1-13 of ECA-0.2 starts ESF equipment as needed and have priority over the FRPs.

B. Incorrect. Plausible to think Natural circulation is established with SG pressures lowering and Cold leg temperatures lowering, but other requirements of Natural Circulation are not met. Per ECA-0.0 Step 1 NOTE, and ECA-0.2 Step 1 NOTE, FRPs are not implemented, only monitored. FRPs should not be implemented until after all safety equipment has been restarted after the completion of step 13 of ECA-0.2, in which a NOTE then states FRPs may now be implemented.

C. Incorrect. Natural circulation is NOT established based on CETs and Hot legs rising. Plausible to transition to FR-C.2 after ECA-0.0 is complete, but this would be incorrect because ECA-0.0 defeats the automatic loading of major equipment, and steps 1-13 of ECA-0.2 starts ESF equipment as needed and have priority over the FRPs.

D. Correct. Natural circulation is NOT established based on CETs and Hot legs rising. Per ECA-0.0 Step 1 NOTE, and ECA-0.2 Step 1 NOTE, FRPs are not implemented, only monitored. FRPs should not be implemented until after all safety equipment has been restarted after the completion of step 13 of ECA-0.2, in which a NOTE then states FRPs may now be implemented.

Sys #	System	Category	KA Statement
000055	Station Blackout / 6	Generic	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

K/A# 2.2.44 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None

Technical References:
1OM-1.2-G Iss. 1C rev. 1
1OM-53B.4.ECA-0.0 Iss. 2 Rev 1 pg 71
1OM-53A.1.ECA-0.0 Iss. 2 Rev 1 pg 24
1OM-53B.4.ECA-0.2 Iss. 2 Rev 0 pg 6

Question Source: Bank – Farley 2011 NRC Exam (Q84)

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.5 / 43.5 / 45.12)

Objective: 3SQS-53.2, Rev. 2 Obj. 12 - State from memory five (5) conditions which indicate that natural circulation is occurring, IAW BVPS EOP Executive Volume.

3SQS-53.3, Rev. 5 Obj. 4 - Explain from memory the basis for ALL Cautions and Notes, 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.

80. Given the following conditions:

0530 - LI-1SI-928, 'C' SI Accumulator Level Indicator failed LOW
0830 - L5 Surveillance Verification Logs were completed
1030 - Loss of Vital Bus 3 occurs causing a loss of power to LI-1SI-930, 'C' SI Accumulator Level Indicator and PI-1SI-931, 'C' SI Accumulator Pressure Indicator

The Vital 3 bus fault has been isolated to Primary Process Rack 23 (RK-PRI-PROC-23), and Vital Bus 3 has been restored.
LI-1SI-930 and PI-1SI-931 remain de-energized due to the fault isolation.

Per Technical Specifications, what is the latest time the 'C' Safety Injection Accumulator must be declared Inoperable?

(REFERENCES PROVIDED)

A. 0530

B. 1630

C. 2030

D. 2330

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II B page 17 first and second bullets. SRO knowledge of the Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1), and Application of generic Limiting Condition for Operation (LCO) requirements (LCO 3.0.1 thru 3.0.7; SR 4.0.1 thru 4.0.4). Specifically, the SRO must be able to determine that the SI accumulator will not need to be declared inoperable until 12 hours after the last performance of the surveillance, plus 3 hours based on SR 3.0.2 interval extension.

K/A is met by determining that with a loss of one 'C' SI accumulator level indicators due to failing low, and a loss of vital bus 3 causing the second 'C' SI accumulator level indicator and pressure indicator to be lost, the accumulator must be declared inoperable 15 hours after the last surveillance was complete.

A. Incorrect. Plausible if the candidate does not recognize that the accumulator remains operable with one level and one pressure indicator available.

B. Incorrect. Plausible if the candidate thinks the surveillance requirement must be completed by 1630 based on the L5 logs time requirements. This is incorrect because the surveillance does not need to be completed until 2330 based on the 12 hour surveillance requirement plus 3 hours.

C. Incorrect. Plausible if the candidate thinks the surveillance requirement must be completed by 2030 based on SR.3.5.1.2 time of 12 hours from the last time it was completed. This is incorrect because the surveillance does not need to be completed until 2330 based on the 12 hour surveillance requirement plus 3 hours for SR 3.0.2 allowance of 1.25 times the interval..

D. Correct. The surveillance does not need to be completed until 2330 based on the 12 hour surveillance requirement of SR 3.5.1.2, plus 3 hours for SR 3.0.2 allowance of 1.25 times the interval specified in the frequency. The surveillance was last completed at 0830.

Sys #	System	Category	KA Statement
000057	Loss of Vital AC Inst. Bus / 6	AA2. Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus:	Core flood tank pressure and level indicators

K/A# AA2.02 **K/A Importance** 3.8* **Exam Level** SRO

References provided to Candidate
1OM-54.3.L5 Rev. 88 pg.19
T.S. 3.5.1 pg.3.5.1-1 &2
BVPS Unit 1 Surv. Test Intervals Rev. 0

Technical References:
T.S. 3.5.1 pg.3.5.1-1 & 2
BVPS Unit 1 Surv. Test Intervals Rev. 0
T.S SR 3.0.2 pg. 3.0-4
1OM-54.3.L5 Rev. 88 pg.19

Question Source: New

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

Objective: 3SQS-RULES ITS-01-05: Given plant conditions, apply the rules of ITS Section 1.3 to ensure compliance with Tech Specs / LRM.
3SQS-RULES ITS-01-01: Given plant conditions, apply the rules of ITS Section 3.0 to ensure compliance with Technical Specifications.

81. Given the following:

- The plant is in Mode 4 preparing to enter Mode 3.
- All systems are in normal configurations for this mode.
- The following alarms are received:
  - A8-27, SYS STA SERV TRANS 1A UNDERVOLTAGE
  - A8-31, SYS STA SERV TRANS 1B UNDERVOLTAGE
- Voltages on all 4160V normal and emergency busses are fluctuating between 80% and 98% of their normal values.
- The DLC System Dispatcher has informed the Unit Supervisor that the grid is EXTREMELY unstable.
- NO automatic or manual actions have occurred.
- 'A' Charging Pump, 1CH-P-1A is RUNNING
- The crew has entered AOP 1/2.35.1, Degraded Grid.

Per AOP 1/2.35.1, which of the following describes the actions that are required for this condition?

A. OPEN BOTH emergency bus tie breakers, allow the EDGs to auto start, and verify proper load sequencing on bus AE and DF.

B. Manually start #2 EDG, OPEN the emergency bus tie breaker, and verify proper load sequencing on bus DF. Then, manually start #1 EDG, OPEN the emergency bus tie breaker, and verify proper load sequencing on bus AE.

C. OPEN the emergency bus tie breaker for bus DF, allow #2 EDG to auto start, and verify proper load sequencing on bus DF. Then, OPEN emergency bus tie breaker for bus AE, allow #1 EDG to auto start, and verify proper load sequencing on bus AE.

D. Manually start BOTH EDGs, parallel the EDGs to its respective emergency bus. Then OPEN the emergency bus tie breakers, and verify proper load sequencing on bus AE and DF

Question 81

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .E page 21 third bullet. SRO is required to have knowledge of the content of the procedures and implementation of the Abnormal procedure. The SRO must evaluate the plant conditions and determine the required actions for this situation. Specifically, the SRO must determine the specific sequence of starting the Diesel Generators and energizing the emergency busses while the electrical grid is unstable, which is SRO knowledge of the AOP procedure content.

K/A is met by evaluating the plant annunciators and indications, and determine how to preemptively energize the emergency busses in the event of grid instability or voltage degradation.

A. Incorrect. This would start and sequence both EDGs, and divorce them from the grid, but it is not the correct action per the AOP. The bus with the non-running charging pump is started and sequenced first to minimize the effect of a loss of charging.

B. Correct. The Degraded Grid AOP states the EDG for the bus with the non-running charging pump should be started first, and the procedure requires the EDG manually started, then the emergency bus tie breaker opened to divorce the EDG from the grid, and allow it to sequence on the required loads.

C. Incorrect. This is the proper bus sequence, but it is not the correct sequence for starting the EDG and loading the emergency busses.

D. Incorrect. This would work for energizing the emergency busses, but it is not correct per the degraded Grid AOP. There is a specific step in the AOP which states if the DG is paralleled to offsite power, unload the DG and open the output breaker.

Sys #	System	Category	KA Statement
000077	Generator Voltage and Electric Grid Disturbances / 6	Generic	Knowledge of annunciator alarms, indications, or response procedures.

K/A# 2.4.31 **K/A Importance** 4.1 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1/2OM-53C.4A.35.1 Rev. 9 pg. 8-9

Question Source: Bank – 1LOT7 NRC Exam (Q44) Modified

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

Objective: 1SQS-35.2, Rev. 9 Obj. 12 - Given a set of plant conditions and the appropriate procedure(s), apply the operational sequence, parameter limits, precaution and limitations, and cautions & notes applicable to the completion of the task activities in the control room.

82. Given the following conditions:

The plant is at 100% power.
A gaseous waste decay tank ruptured.
No gaseous waste system evolutions were in progress.
The following Radiation Monitors are in High-High alarm reading as indicated:

- RM-1VS-107B, Reactor Bldg. and SLCRS Vent Gas Monitor is 9.1E+03 CPM
- RM-1VS-106, Waste Gas Tank Vault Ventilation Exhaust Gas Monitor is 1.2E+06 CPM
- RM-1VS-110, Reactor Bldg/SLCRS Exhaust (PING) Monitor is 9.1E+05 µCi/s

Automatic actions of RM-1VS-106 in High-High alarm have failed to occur.

1) What actions must be taken in accordance with the ARP for RM-1VS-106 in High-High alarm?

    1) Close 1VS-D-4-1A, Leak Collection Filter Bank Bypass Damper
    2) Open 1VS-D-4-2A, Leak Collection to Main Filter Bank Damper
    3) Stop 1VS-F-7A, Aux. Bldg. Exhaust Fan

2) The SLCRS Elevated Release path radiation monitor, RM-1VS-107B Channel Operational Test is required at which of the following frequencies?

(REFERENCE PROVIDED)

A. 1) 1 and 2 ONLY
   2) Monthly

B. 1) 1, 2, and 3
   2) Monthly

C. 1) 1 and 2 ONLY
   2) Quarterly

D. 1) 1, 2, and 3
   2) Quarterly

Question 82

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II B page 17 first bullet. SRO is required to have knowledge of the application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1). Specific knowledge of the use of the ODCM is an SRO position function.

K/A is met by demonstrating the ability to verify that the Leak Collection Filter is aligned properly after a Gaseous Waste Decay Tank ruptures and actuates a High-High alarm on RM-1VS-106. The question states that the automatic actions of RM-1VS-106 in High-High alarm have failed to occur, making the candidate determine what actions should have occurred.

A. Incorrect. First part is correct. Second part is plausible because RM-1VS-107B is identified in attachment D (pg. 24) as a second backup for the Rx Building/SLCRS high range noble gas detector RM-1VS-110 which has a COT frequency of monthly. The candidate will have to read note (a) at the bottom of the page to recognize that ODCM surveillance requirements do not apply to PMM (Preplanned Method of Monitoring) instruments and determine that RM-1VS-107B is the primary Rx Building/SLCRS noble gas detector on page 46.

B. Incorrect. First part is plausible because 1VS-D-4-1A closed and 1VS-D-4-2A opened are auto actions of RM-1VS-106, but RM-1VS-102A, Aux Bldg Gas monitor trips 1VS-F-7A. Second part is plausible because RM-1VS-107B is identified in attachment D as a second backup for the Rx Building/SLCRS high range noble gas detector RM-1VS-110 which has a COT frequency of monthly. The candidate will have to read note (a) at the bottom of the page to recognize that ODCM surveillance requirements do not apply to PMM (Preplanned Method of Monitoring) instruments and determine that RM-1VS-107B is the primary Rx Building/SLCRS noble gas detector on page 46.

C. Correct. 1OM-43.4.AEJ (ARP for RM-1VS-106) states to verify the following automatic actions have occurred, 1VS-D-4-1A closed and 1VS-D-4-2A opened. Second part is correct per ODCM attachment F (page 10 of 13) (pg. 46) which states that the Rx Building/SLCRS noble gas activity monitor is required to be tested Quarterly.

D. Incorrect. First part is plausible because 1VS-D-4-1A closed and 1VS-D-4-2A opened are auto actions of RM-1VS-106, but RM-1VS-102A, Aux Bldg Gas monitor trips 1VS-F-7A. Second part is correct.

Sys #	System	Category	KA Statement
000060	Accidental Gaseous Radwaste Rel. /9	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 1/2ODC-3.03 Rev 13 (pgs. 1- 84) **Technical References:** 1OM-43.4.AEJ Rev. 4 pg. 2
1OM-43.5.B.2 Rev. 4 pg.3
Unit 1 RM-0416-001 Rev. 16
1/2ODC-3.03 Rev 13 pg. 46 of 84

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.3)

Objective: 1SQS-16.1, Rev. 9 Obj. 12 - Describe the control, protection and interlock functions for the control room components associated with the SLCRS, including automatic functions, setpoints and changes in equipment status as applicable.

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.

83. The following conditions exist:
    - A Small break LOCA has occurred
    - CETs are 730 °F and RISING
    - The crew has transitioned to FR-C.1, Response to Inadequate Core Cooling, and are preparing to depressurize the Steam Generators
    - Low Steamline Pressure SI has been blocked

In accordance with FR-C.1, step 14, "Depressurize all Intact SGs to 110 psig", at what rate are the SGs depressurized, and why was 110 psig chosen as the target pressure?

1) The SGs are depressurized at ______(1)______ rate using the Condenser Steam Dumps.

2) The SG pressure of 110 psig was chosen because it ______(2)______.

A. 1) less than 100°F/hr cooldown
   2) maximizes SI accumulator water injection and minimizes Nitrogen injection to the RCS

B. 1) less than 100°F/hr cooldown
   2) corresponds to RCS hot leg temperature of 410°F, which ensures the RCS pressure is below the shutoff head of the LHSI pumps

C. 1) the maximum
   2) maximizes SI accumulator water injection and minimizes Nitrogen injection to the RCS

D. 1) the maximum
   2) corresponds to RCS hot leg temperature of 410°F, which ensures the RCS pressure is below the shutoff head of the LHSI pumps

Question 83

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .E page 21 third bullet. SRO is required to have knowledge of the content of the procedures and implementation of the Emergency procedures. The SRO must evaluate the plant conditions and determine the required actions for this situation. Specifically, the SRO must determine the required steam generator depressurization rate and target pressure to ensure the SI accumulators will provide maximum water to the RCS while minimizing Nitrogen injection.

K/A is met by the candidates ability to determine that depressurizing the steam generators using the condenser steam dumps at maximum rate will reduce RCS temperature and pressure sufficiently enough to ensure the SI accumulators will deliver maximum SI water accumulator water, while minimizing nitrogen delivery to the RCS while performing FR-C.1, Response To Inadequate Core Cooling.

- **A.** Incorrect. First part is plausible because 100F/hr is used often within the EOP network and it is the Tech Spec limit for plant cooldown. Second part is correct.
- **B.** Incorrect. First part is plausible because 100F/hr is used often within the EOP network and it is the Tech Spec limit for plant cooldown. Second part is plausible because 410F hot leg temperature is discussed in FR-C.1, but it is the temperature which corresponds to SI accumulator water delivery, not the LHSI shutoff head pressure. 410F corresponds to approximately 260 psig, and the LHSI shutoff head pressure is 178 psig.
- **C.** Correct. Step 14d has the SGs depressurized at maximum rate since it is the most effective way to reduce RCS pressure. The bases of the 110 psig target for SG pressure is to ensure maximum SI water accumulator water delivery while minimizing nitrogen delivery to the RCS.
- **D.** Incorrect. First part is correct. Second part is plausible because 410F hot leg temperature is discussed in FR-C.1, but it is the temperature which corresponds to SI accumulator water delivery, not the LHSI shutoff head pressure. 410F corresponds to approximately 260 psig, and the LHSI shutoff head pressure is 178 psig.

Sys #	System	Category	KA Statement
000074	Inad. Core Cooling / 4	EA2 Ability to determine or interpret the following as they apply to Inadequate Core Cooling:	The effect of turbine bypass valve operation on RCS temperature and pressure

K/A# EA2.08 **K/A Importance** 4.6* **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-C.1 Iss. 3 Rev. 0 step 14 pg. 7-9; 1OM-53B.4.FR-C.1 Iss. 3 Rev. 0 pg. 25

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. 2 - Describe from memory the overall purpose of each procedure, IAW BVPS-EOP Executive Volume.

84. ES-0.0, Rediagnosis, has been entered following a reactor trip and safety injection.

The following conditions exist:

- Containment pressure 11.2 psig and LOWERING
- Containment temperature 165°F and LOWERING
- Steam Generator level 44% WR and LOWERING in all SGs
- Steam Generator pressure 350 psig and LOWERING in all SGs
- Pressurizer level 0%
- RCS pressure 1680 psig and LOWERING
- Atm Steam Dump and RHR valves are CLOSED
- All MSIVs are CLOSED

Which of the following describes the proper transition to be made from ES-0.0 to respond to the above plant conditions?

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

B. ES-1.2, Post LOCA Cooldown and Depressurization

C. E-2, Faulted Steam Generator Isolation

D. ECA-2.1, Uncontrolled Depressurization of All Steam Generators

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Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II E page 21 second bullet. SRO is required to have additional knowledge of the procedure's content, and knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures.

K/A is met by requiring a detailed knowledge of ES-0.0, Rediagnosis, and interpreting plant conditions to select the appropriate procedure to respond to the event.

A. Incorrect. Plausible because some indications of a LOCA are present in the stem (cnmt pressure, przr level, RCS pressure), but before getting to the E-1 evaluation in ES-0.0 (last evaluation), the correct transition is identified as ECA-2.1.

B. Incorrect. Plausible because some indications of a LOCA are present in the stem (cnmt pressure, przr level, RCS pressure), but the first step of ES-0.0 checks SG pressure and directs to ECA-2.1 based on the given conditions. The E-1/ECA-1 series evaluation in ES-0.0 is the last step..

C. Incorrect. Plausible because this would be the transition from the first step in ES-0.0 if MSIVs were open, but with the MSIVs closed, ECA-2.1 the correct transition.

D. Correct. With all 3 SGs pressure lowering, and MSIVs being closed, the ECA-2.1 transition is correct iaw ES-0.0 step 1 RNO.

Sys #	System	Category	KA Statement
W/E01	Rediagnosis / 3	EA2. Ability to determine and interpret the following as they apply to the (Reactor Trip or Safety Injection Rediagnosis)	Facility conditions and selection of appropriate procedures during abnormal and emergency operations

K/A# EA2.1 **K/A Importance** 4.0 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.ES-0.0 Iss. 3 Rev. 0 pg. 2 & 3

Question Source: Bank – Callaway 2013 NRC exam (Q85)

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

Objective: 3SQS-53.3-01-06: Given a set of conditions, locate and apply the proper Emergency Operating Procedures, IAW BVPS-EOP Executive Volume

85. Cooldown is in progress per ES-0.2, Natural Circulation Cooldown.

The crew is at the step in ES-0.2 to check that a steam void in Reactor Vessel does NOT exist.

The following data is noted:

Time	RVLIS Upper Range	PRZR Level
10:00	100	25
10:15	100	25
10:30	98	28
10:45	84	58

Which of the following choices correctly completes the statement below?

Based on the given conditions, the MAXIMUM cooldown rate allowed in ES-0.2 is ______(1)______ **and** the NEXT action required by ES-0.2 is to ______(2)______.

A. 1) 25°F per hour
2) repressurize the RCS within the limits of EOP Att. 5-C, "CRDM Fans Running - Natural Circulation Cooldown Subcooling Requirements" to collapse potential voids in the system

B. 1) 100°F per hour
2) repressurize the RCS within the limits of EOP Att. 5-C, "CRDM Fans Running - Natural Circulation Cooldown Subcooling Requirements" to collapse potential voids in the system

C. 1) 25°F per hour
2) actuate Safety Injection and transition to E-0, Reactor Trip or Safety Injection

D. 1) 100°F per hour
2) actuate Safety Injection and transition to E-0, Reactor Trip or Safety Injection

Question 85

Answer: A

Explanation/Justification: Meets NUREG-1021 Rev. 10, Att.2 Sect. II.E pg 21 second bullet which requires the knowledge of diagnostics steps and decision points in EOPs that involve transitions to event specific sub-procedures. The SRO must be aware of whether to transition to a sub-procedures for Natural Circulation Cooldown or repressurize the RCS and remain in the current procedure, and know the required cooldown rates for the various Natural Circulation sub procedures. Detailed procedure knowledge is required for the 25F/hr cooldown rate and ES-0.2, Natural Circulation Cooldown procedure RNO step to repressurize the RCS.

K/A is met by the candidate interpreting that voids exist in the vessel while performing ES-0.2, Natural Circulation Cooldown, and determines that the RCS must be repressurized IAW EOP att. 5-C per the RNO step of ES-0.2.

A. Correct. 25F/hr cooldown rate is correct per step 6.b. Step 19 checks for voids, and if present has the operator repressurize the RCS IAW attachment 5-C.

B. Incorrect. First part is plausible because 100F/hr is BV Tech Spec C/D limit, and is used in several EOP procedures. Second part is correct.

C. Incorrect. First part is correct. Second part is plausible because actuating SI and transitioning to E-0 is a Left Hand Page item in ES-0.2, and even though it is not one of the transition criteria identified, it sound plausible as a way of removing RCS voids.

D. Incorrect. First part is plausible because 100F/hr is BV Tech Spec cooldown limit, and is used in several EOP procedures. Second part is plausible because actuating SI and transitioning to E-0 is a Left Hand Page item in ES-0.2, and even though it is not one of the transition criteria identified, it sound plausible as a way of removing RCS voids.

Sys #	System	Category	KA Statement
W/E10	Natural Circ. / 4	Generic	Ability to interpret and execute procedure steps.

K/A# 2.1.20 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.ES-0.2 Iss. 3 Rev. 0 pg. 6, 7, 19

Question Source: Bank - Vogtle 2012 NRC Exam (Q100)

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.12)

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.

86. Given the following conditions:

- An inadvertent Reactor Trip has occurred
- Crew has transitioned to ES-0.1, Reactor Trip Response
- FCV-1CH-122, Chg Flow to Regen Hx Inlet Control Vlv is stuck OPEN due to a bent stem
- 1CH-P-1B, 'B' Charging Pump is RUNNING
- FI-1CH-150, Letdown Flow is 0 gpm
- FI-1CH-122A, Charging Flow is 180 gpm
- PRZR level is 32% and RISING

1) To mitigate the consequences of Letdown flow being lost, step 3 of ES-0.1 directs the crew to restore letdown using which of the following procedures?

2) How is PRZR level controlled with FCV-1CH-122 stuck open?

A. 1) AOP 1.7.1, Loss of Charging or Letdown
2) Start/Stop 1CH-P-1B, 'B' Charging Pump

B. 1) AOP 1.7.1, Loss of Charging or Letdown
2) Open/Close MOV-1CH-289, Charging Pump Disch Header to Regen HX in CNMT Isolation

C. 1) 1OM-7.4.H, Excess Letdown Heat Exchanger Operation
2) Start/Stop 1CH-P-1B, 'B' Charging Pump

D. 1) 1OM-7.4.H, Excess Letdown Heat Exchanger Operation
2) Open/Close MOV-1CH-289, Charging Pump Disch Header to Regen HX in CNMT Isolation

Question 86

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II E page 21 second & third bullets. SRO is required to have additional knowledge of the procedure's content, and knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures. Specifically, the SRO must use the Loss of Letdown AOP in parallel with ES-0.1, to restore Letdown flow. The SRO must also have specific AOP knowledge for controlling Charging Flow when the FCV is stuck open.

K/A is met by requiring knowledge of specific EOP guidance directing the use of the Loss of Charging or Letdown AOP in parallel with the EOP to restore Letdown flow.

A. Incorrect. AOP-1.7.1 is correct. Second part is plausible because the charging are stopped and started when realigning HHSI flowpaths under certain conditions per the EOP attachments.

B. Correct. Step 3 of EOP ES-0.1 has the crew check letdown flow. If letdown flow is not indicated the crew will perform AOP-1.7.1 in parallel with the EOP to restore normal letdown. The AOP then references using MOV-1CH-289 to maintain przr level by opening and closing as needed.

C. Incorrect. 1OM-7.4.H is plausible because it is referred to in multiple EOPs, and if normal letdown cannot be restored, AOP-1.7.4 references the use of 1OM-7.4.H. Second part is plausible because the charging pumps are stopped and started when realigning HHSI flowpaths under certain conditions per the EOP attachments. Second part is plausible because the charging pumps are stopped and started when realigning HHSI flowpaths under certain conditions per the EOP attachments.

D. Incorrect. 1OM-7.4.H is plausible because it is referred to in multiple EOPs, and if normal letdown cannot be restored, AOP-1.7.4 references the use of 1OM-7.4.H. Second part is correct.

Sys #	System	Category	KA Statement
004	Chemical and Volume Control	Generic	Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

K/A# 2.4.8 **K/A Importance** 4.5 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.ES-0.1 Iss. 3 Rev. 0 pg. 2
1OM-53C.4.1.7.1 Rev. 11 pg. 11-12

Question Source: New

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

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

87. The plant is operating at 100% power.
    - Annunciator A6-37, 'PRI COMP COOL WTR SURGE TANK LEVEL HIGH-LOW' is in alarm
    - LI-1CC-100, CCR Surge Tank Level is offscale LOW
    - LCV-1CC-100A, CCR Surge TK Level Control Valve is OPEN with M/U water available
    - The crew has entered AOP 1.15.1, Loss Of Primary Component Cooling Water

    Given the above conditions, which of the following actions are required to be performed at this time?

A. Continue in AOP 1.15.1, remove non-essential CCR loads from service, and monitor for additional CCR component alarms.

B. Trip the reactor, enter and perform E-0 IOAs, trip the RCPs, isolate Letdown, transfer charging pump suction to RWST, stop all CCR pumps.

C. Continue in AOP 1.15.1, go to Attachment 1, "CCR System Out-Leakage" and attempt to isolate the leak.

D. Trip the reactor, enter and perform E-0 IOAs, trip the RCPs, close PRZR spray valves.

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .E page 21 third bullet. SRO is required to have knowledge of the content of the procedures and transitions between Abnormal and EOPs. The SRO must evaluate the plant conditions and determine if a plant parameter has been exceeded, and the required actions for this situation. Then the SRO must determine the specific sequence of actions to take after tripping the reactor, which are listed as sub-steps in the Abnormal Operating Procedure. Additionally directing the action to secure the RCPs and other equipment is to occur following completion of the IOAs, which is SRO knowledge of the AOP procedure content.

K/A is met by the candidate determining that a CCR Surge tank level offscale low, will require a reactor trip, and transition to E-0, and detailed knowledge of the additional actions required by the AOP to mitigate the consequences of a loss of CCR.

A. Incorrect. Plausible distractor because removing non-essential CCR loads from service would reduce the heat load on the system, and possibly delay additional alarms from coming in while working through Loss of CCR AOP.
B. Correct. Step 1 of AOP-1.15.1 checks the surge tank level onscale, if level is offscale low as stated in the stem, then these actions are correct per step 1 RNO.
C. Incorrect. Plausible distractor if Surge tank level was onscale and stable with the makeup valve open, or level was dropping.
D. Incorrect. Plausible distractor if an RCP trip criteria were met since these actions are required per AOP-1.6.8, Abnormal RCP Operation AOP, but there is no indication in the stem indicating that an RCP alarm exists at this time.

Sys #	System	Category	KA Statement
008	Component Cooling Water	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	High/low surge tank level

K/A# A2.02 **K/A Importance** 3.5 **Exam Level** SRO
References provided to Candidate None **Technical References:** 1OM-53C.4.1.15.1 Rev. 6 pg.3
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-53C.1, Rev. 12 Obj. 5 - Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.

88. Given the following conditions:

- A total loss of Main Feedwater and Auxiliary Feedwater occurs
- The crew is performing the actions of FR-H.1, Response to Loss of Secondary Heat Sink
- Bleed and Feed has been initiated
- All steam generators are "dry"

The capability to feed all steam generators using the TDAFW pump has been restored.

- The crew is currently feeding 'A' SG at 100 gpm
- Core Exit Thermocouples are 555°F and slowly RISING

Which of the following actions will be used to recover a secondary heat sink, and why?

A. Establish MAXIMUM flow to the 'A' steam generator due to the urgent need to restore a heat sink.

B. Establish MAXIMUM flow to ALL steam generators due to the urgent need to restore a heat sink.

C. Establish 100 gpm flow to ALL steam generators to meet minimum heat sink requirements.

D. Maintain 100 gpm flow to the 'A' steam generator because minimum heat sink has been restored.

Question 88

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II E page 22 Fig. 2 First bullet, Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. Specifically the SRO must recognize that SI has been initiated, and feed flow has been establish, but the CETs are still rising. This will require the RNO action of feeding the one SG at maximum feed flow.

K/A is met with the knowledge of the mitigation strategy of FR-H.1 when SI and feed flow have been established to a dry SG, but the RCS temperature continues to rise requiring procedural action to feed one SG at maximum rate due to the urgent need to restore a heat sink.

A. Correct. With CET temperatures rising with SI flow initiated, and feed flow to one SG, then the appropriate action per FR-H.1 step 28 RNO is to feed at maximum rate to one SG due to the urgency of the plant conditions. Limited to one SG to limit possible thermal failure to only one SG.

B. Incorrect. Plausible distractor with the given urgency of no heat sink based on the plant conditions, but maximum flow is only limited to one SG due to possible thermal failure of the SGs.

C. Incorrect. Plausible distractor because the bases for FR-H.1 states "to ensure the total AFW flow rate for all combinations of flow will satisfy the "minimum AFW flow requirement for heat removal plus allowances for normal channel accuracy", a minimum of 115 GPM flow to each SG is necessary." This is not applicable based on the stated conditions of the CET temperatures rising with RCS Bleed and Feed in progress.

D. Incorrect. Plausible distractor because step 5 of FR-H.1 checks if there is 370 gpm AFW flow to the SGs, if not, the RNO checks that feed flow is established to at least one SG (stated conditions state 100 gpm flow to 'A' SG), in which case the step is to continue feeding the SG until NR level is >31%. This is not applicable based on the stated conditions of the CET temperatures rising with RCS Bleed and Feed in progress.

Sys #	System	Category	KA Statement
059	Main Feedwater	Generic	Knowledge of EOP mitigation strategies.

K/A# 2.4.6 **K/A Importance** 4.7 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-H.1 Iss 2 Rev 2 pg. 19-20
1OM-53B.4.FR-H.1 Iss. 2 Rev. 2 pg 73-74

Question Source: Bank - Diablo Canyon 2012 draft (Q9) Modified

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

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

89. The plant is at 100% power.

- TV-1MS-105A, AFW Turb Steam Supply 'A' Train Trip Valve air supply line ruptures.

1) Based on the above failure, what is the status of the 1FW-P-2, Turbine-Driven Auxiliary Feed Pump?
2) What actions, if any, will the crew take to mitigate this event?

The Turbine-Driven Auxiliary Feed Pump is ____(1)____.
The crew will ____(2)____

A. 1) shutdown
2) enter LCO 3.7.5, Cond. B for the Inoperable AFW pump, which allows 72 hours for the airline to be repaired.

B. 1) shutdown
2) continue power operations because the TDAFW remains operable due to the availability of the opposite train steam supply.

C. 1) running
2) reduce power using 1OM-52.4.B.1, Turbine Load Changes until the overpower condition has cleared.

D. 1) running
2) use 1OM-24.4.K, Steam Generator Level Control With Auxiliary Feedwater until the TDAFW pump is shutdown.

Answer: C

Explanation/Justification: KA is met by identifying that a loss of air due to a broken air supply line to the Turbine-Driven Auxiliary Feed Pump will cause the pump to start, then predict that an overpower condition will occur when operating at 100% power and cold water is fed to the SGs. They will have to determine that power will need to be reduced using the Turbine Load Change procedure.

A. Incorrect. Plausible distractor if the candidate does not know how TV-1MS-105A fails, or thinks the 2 steam supply valves are in series. Second part is plausible because LCO 3.7.5 would be entered for this failure.

B. Incorrect. Plausible distractor if the candidate does not know how TV-1MS-105A fails, or thinks the 2 steam supply valves are in series. Second part is plausible if the candidate thinks TV-1MS-105A fails open, but is in series, in which case a second valve being available would allow the TDAFW pump to operate.

C. Correct. The TDAFW will be running due to TV-1MS-105A failing open on the loss of air and admitting steam to the turbine. An overpower event will occur due to the addition of cold water to the SGs while at 100% power. This will require a down power to reduce Rx power <100% using Turbine Load Changes procedure 1OM-52.4.B.1.

D. Incorrect. The TDAFW pump will be running. Second part is plausible because the candidate may not recognize that an overpower event will occur, and this procedure sounds plausible for the conditions.

Sys #	System	Category	KA Statement
061	Auxiliary/Emergency Feedwater	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of air to steam supply valve

K/A# A2.02 **K/A Importance** 3.6* **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.34.1 rev. 21 pg. 18
1OM-52.4.B.1 Rev. 1 pg. 3
Unit 1 RM-0421-001 Rev. 25

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 Obj. 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.

1SQS-53C.1, Rev. 12 Obj. 2 - State the conditions or symptoms that would require entry into the AOPs.

90. Given the following conditions:

- Plant startup is in progress after a refueling outage
- Plant is in Mode 5
- Engineering has reported that the Containment Type 'A' overall Integrated Leak Rate Test calculation was re-verified, and determined L_a is 1.2

1) What Design Basis Accident is the Integrated Leak Rate Test maximum allowable leakage based upon?

2) Based on the Integrated Leak Rate Test results, is plant startup permitted to continue into Mode 4?

A. 1) LOCA
2) is permitted

B. 1) Steam Line Break
2) is permitted

C. 1) LOCA
2) is NOT permitted

D. 1) Steam Line Break
2) is NOT permitted

Question 90

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .B page 17 first, second, and third bullets. SRO is required to have knowledge of the TS bases, generic Limiting Condition for Operation (LCO) requirements, and application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements. Specifically the SRO must have detailed knowledge of the TS bases, and application of TS 3.6.1 to determine that the Integrated Leak Rate Test does not meet the LCO, and plant startup may not continue.

K/A is met by the candidates knowledge that a LOCA is the basis for the Integrated Leak Rate test of containment to ensure the offsite dose consequences are bounded by the assumptions of the safety analysis for containment describes the impact of a failed ILRT. Uses Tech Specs and determines that the Integrated Leak Rate test results do not meet Tech Spec 3.6.1 or T.S. 5.5.12 acceptance criteria for Containment Operability, and determines that TS 3.6.1 is not met. Then demonstrates the understanding that a transition into Mode 4 is not allowed, therefore plant startup must be halted.

A. Incorrect. A design basis LOCA is the basis for the ILRT of containment. Second part is a plausible distractor if the candidate thinks that transition into Mode 4 would be allowed due to not knowing that the modes of applicability for TS 3.6.1 are modes 1-4, or they think LCO 3.0.4 criteria is met.

B. Incorrect. Plausible because Steam Line Break is a DBA evaluated for the Containment Tech Spec, but the ILRT is performed to ensure the offsite dose consequences are bounded by the assumptions of the safety analysis. Second part is a plausible distractor if the candidate thinks that transition into Mode 4 would be allowed due to not knowing that the modes of applicability for TS 3.6.1 are modes 1-4, or they think LCO 3.0.4 criteria is met.

C. Correct. A design basis LOCA is the basis for the ILRT of containment to ensure the offsite dose consequences are bounded by the assumptions of the safety analysis. Plant startup is not permitted into Mode 4 because the containment LCO modes of applicability is Modes 1-4, and none of the criteria of LCO 3.0.4 are stated or implied in the question stem.

D. Incorrect. Plausible because Steam Line Break is a DBA evaluated for the Containment Tech Spec, but the ILRT is performed to ensure the offsite dose consequences are bounded by the assumptions of the safety analysis. Second part is correct.

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	Integrated leak rate test

K/A# A2.01 **K/A Importance** 2.6* **Exam Level** SRO

References provided to Candidate None

Technical References:
T.S. 3.6.1 pg 3.6.1-1
T.S. B3.6.1 pg B3.6.1-2
T.S 5.5.12 pg. 5.5-19 & 20
T.S. LCO 3.0.4 pg. 3.0-1

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

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).

91. The plant is at 100% power with all plant conditions stable.

1) How will the Overtemperature ΔT (OTΔT) SETPOINT respond if Power Range N41 UPPER detector fails HIGH?

2) Regarding Technical Specification Safety Limits, which of the following core limitations does the Overtemperature ΔT (OTΔT) Reactor Trip prevent exceeding?

A. 1) Increase
2) Departure from Nucleate Boiling (DNB)

B. 1) Increase
2) Power Density (KW/ft)

C. 1) Decreases
2) Departure from Nucleate Boiling (DNB)

D. 1) Decreases
2) Power Density (KW/ft)

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .B page 17 third bullet. SRO is required to have knowledge of the TS bases. Specifically the SRO must have detailed knowledge of the TS bases to determine that the Overtemperature ΔT reactor trip is to ensure the DNBR design limit is met.

K/A is met by demonstrating knowledge of how the OTΔT setpoint will decrease when the upper detector of N41 fails High, then identifying the Technical Specifications bases for the Overtemperature ΔT limiting conditions for operations and safety limits.

A. Incorrect. Plausible distractor if it is not known how to calculate ΔI, and it is determined that ΔI becomes a smaller value in the OTΔT setpoint equation, therefore the OTΔT setpoint would increase. Second part is correct.
B. Incorrect. Plausible distractor if it is not known how to calculate ΔI, and it is determined that ΔI becomes a smaller value in the OTΔT setpoint equation, therefore the OTΔT setpoint would increase. Second part is plausible because Overpower ΔT trip provides protection for power density
C. Correct. ΔI is the power at the top core, minus power at the bottom of the core. With the upper detector failing high, ΔI becomes a larger value in the OTΔT setpoint equation, therefore the OTΔT setpoint will decrease. When ΔI limits are exceeded, the OTΔT setpoint is always reduced. The Overtemperature ΔT trip Function is provided to ensure that the design limit DNBR is met.
D. Incorrect. First part is correct. Second part is plausible because Overpower ΔT trip provides protection for power density.

Sys #	System	Category	KA Statement
015	Nuclear Instrumentation	Generic	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

K/A# 2.2.25 **K/A Importance** 4.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** Tech Spec Bases Rev. 0 pg. B3.3.1-16
Tech Spec Table 3.3.1-1 (pg. 5 of 9)

Question Source: Bank – 1LOT5 NRC Exam (Q36) Modified

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.5 / 41.7 / 43.2)

Objective: 3SQS-1.1, Rev. 8 Obj. 17 - Describe the design bases for the Reactor Protection System Trip Logics & Engineered Safeguards Features Actuation Signals and the associated major components as documented in the UFSAR.

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

92. The plant was at 100% power when a small break Loss of Coolant Accident occurred.

Conditions 30 minutes after the accident:

- RCS pressure is 1225 psig and STABLE
- PRZR level is 15% and STABLE
- RCS subcooling is 15°F
- All SG pressures are 950 psig and SLOWLY LOWERING
- All SG Narrow Range levels are 22% and RISING
- Containment pressure is 3.4 psig and SLOWLY RISING

Based on the above conditions, which of the following actions are required after transitioning to E-1, Loss of Reactor or Secondary Coolant?

A. Remain in E-1, Loss of Reactor or Secondary Coolant, to remove decay heat and cooldown the RCS using long term recirculation.

B. Enter E-2, Faulted Steam generator Isolation, to isolate all 3 Steam Generators.

C. Enter ES-1.1, SI Termination, to terminate Safety Injection to prevent RCS overfill.

D. Enter ES-1.2, Post LOCA Cooldown and Depressurization, to remove decay heat and cooldown the RCS using the Steam Generators.

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .E page 21 third bullet. SRO is required to have knowledge of the content of the procedures. Specifically the SRO must evaluate the plant conditions and determine that plant cooldown and depressurization after the SBLOCA will be accomplished using ES-1.2, Post LOCA Cooldown and Depressurization using the available steam generators. Detailed knowledge of the procedural content is required to select the correct procedural direction.

K/A is met by determining that a SBLOCA which only lowers the RCS pressure to a pressure greater than the Low Head SI pump shutoff head of 275 psig will meet the requirements of transitioning to ES-1.2, and plant cooldown will be performed using the steam generators.

A. Incorrect. Plausible distractor if the transition to ES-1.2 requirement of RCS pressure >275 psig is not known, and it is thought that long term recirculation is the only option for cooling the RCS.

B. Incorrect. Plausible distractor because E-1 step 4 checks if any SGs are faulted. With the indications given, SG pressures are lowering, but this is due to Safety Injection cooling the RCS, not a faulted SG.

C. Incorrect. Plausible distractor because E-1 step 9 checks for SI Term criteria. Based on the conditions given it can be determined that SI term criteria is not based on przr level being <17%, and subcooling being 15F.

D. Correct. E-1 step 20 checks if RCS cooldown and depressurization is required prior to continuing in E-1 for long term recirculation. With RCS pressure >275 psig, the transition to ES-1.2 is warranted.

Sys #	System	Category	KA Statement
035	Steam Generator	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the SG; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Small break LOCA

K/A# A2.06 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.E-1 Iss. 3 Rev. 0 pg. 15
1OM-53B.4.ES-1.2 Iss. 3 Rev. 1 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-53.3, Rev. 5 Obj. 6 - Given a set of conditions, locate and apply the proper Emergency Operating Procedures, IAW BVPS-EOP Executive Volume.

93. Given the following conditions:

- Plant startup is in progress IAW 1OM-50.4.D, Reactor Startup From Mode 3 To Mode 2
- Rx power is 2.5% and slowly RISING
- AM-1MS-464B, Cooldown Vlvs Controller is in Steam Pressure Mode set at 1005 psig
- PS-1MS-101A, Atmospheric Steam Dump SG 1A pressure switch fails HIGH

1) How will the Condenser Steam Dump valves respond?

The crew takes manual control of the 'A' SG Atmospheric Steam Dump IAW Att. 2-U, Local Operation of SG Atmospheric Steam Dump Valves.

2) While in Local-Manual control, is the 'A' SG Atmospheric Steam Dump Valve Operable per Tech Spec 3.7.4, Atmospheric dump Valves?

A. 1) throttle closed
2) yes

B. 1) throttle closed
2) no

C. 1) not be affected
2) yes

D. 1) not be affected
2) no

Question 93

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .B page 17 third bullet. SRO is required to have knowledge of the TS bases. Specifically the SRO must evaluate Local-Manual operation of the Atmospheric Steam Dump and determine that the valve is inoperable based on the Tech Spec bases stating that the ASD must be capable of BOTH remote and local operation to be operable. Detailed knowledge of the bases is required to determine that the ASD is inoperable.

K/A is met by demonstrating the ability to predict the impact of an atmospheric steam dump failing open during a plant startup, and the response of the condenser steam dump system. When the ASD is locally closed to mitigate the event, the SRO must determine that the ASD is not operable.

A. Incorrect. The First part is correct. The second part is plausible because TS 7.4.1 bases states the ASDs must be capable of BOTH local and remote operation to be considered operable.

B. Correct. When the 'A' ASD pops open due to the pressure switch failing high, the condenser steam dumps will throttle closed due to the condenser steam dump controller being in steam pressure mode, and PT-1MS-464 sensing a lower pressure on the steam header. The 'A' ASD is Inoperable based on TS 7.4.1 bases which states that in order to meet the assumptions of the operational assessment used to evaluate single failure concerns, the Unit 1 ADVs must be capable of being operated locally as well as from the control room in order to be considered OPERABLE.

C. Incorrect. First part is plausible if the candidate thinks that AM-1MS-464B is in manual, in which case the condenser steam dumps would not respond, but the stem states that the controller is in steam pressure mode set at 1005 psig. The second part is plausible because TS 7.4.1 bases states the ASDs must be capable of BOTH local and remote operation to be considered operable.

D. Incorrect. First part is plausible if the candidate thinks that AM-1MS-464B is in manual, in which case the condenser steam dumps would not respond, but the stem states that the controller is in steam pressure mode set at 1005 psig. Second part is correct.

Sys #	System	Category	KA Statement
041	Steam Dump/Turbine Bypass Control	A2 Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations:	Steam valve stuck open

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

References provided to Candidate None **Technical References:** Unit 1 RM-0421-001 rev. 25 & 002 Rev. 20
1OM-21.5.A.24 Rev. 4 pg. 2
T.S. 3.7.4 Bases pg B3.7.4-1

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-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.

3SQS-PLTSYS ITS, Rev. 2 Obj. 3 - Given plant conditions, determine the criteria necessary to ensure compliance with each Section Plant Systems System LCO and Licensing Requirement in accordance with the Bases, Surveillance Requirements, and the Applicability.

94. The plant is operating at 100% power when an Operator filling a Tech Spec required watch position becomes ill and leaves the site.

Per TS 5.2.2 and 10CFR50.54, the crew composition may remain less than the minimum for a period of time not to exceed ______(1)______.

If the vacant position is NOT refilled within the required time, the crew will ______(2)______.

A. 1) 1 hour
   2) take action to place the unit in MODE 5

B. 1) 1 hour
   2) maintain current power level

C. 1) 2 hours
   2) take action to place the unit in MODE 5

D. 1) 2 hours
   2) maintain current power level

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .A page 17 third bullet. SRO is required to have knowledge of the TS section 5 and 6 actions related to plant staffing. Additionally the SRO is required to know the administrative procedure content related to not meeting staffing requirements.

K/A is met with the knowledge of how long an on-duty shift position may be unfilled, and the requirements of Tech Specs and Plant Procedures to maintain safe plant operation.

A. Incorrect. See correct answer.
B. Incorrect. See correct answer.
C. Incorrect. See correct answer.
D. Correct. In accordance with TS 5.2.2, 2 hours is the expected time to fill the required position provided immediate action is taken to restore shift composition to minimum. BVPS has incorporated Licensing position on TS 5.2.2 into NOP-OP-1002, which states it is not conservative to place the plant into a transient due to staffing, therefore maintain the unit in a steady state condition and continue calling out personnel.

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:** T.S. 5.2.2 Amend 278/161 pg. 5.2-1
NOP-OP-1002 Rev. 11 sect. 4.1.13

Question Source: Bank – 2LOT15 NRC Exam (Q94)

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 43.2)

Objective: 3SQS-48.1 Obj. 3 From memory, describe the required actions if less than the minimum shift staffing complement exists.

95. The plant was at 100% power when the following occurs:
    - The reactor failed to trip after receiving a valid trip signal
    - SRO transitioned from E-0, Reactor Trip or Safety Injection, to FR-S.1, Response to Nuclear Power Generation/ATWS

    Current conditions:
    - Emergency Boration was initiated
    - Safety injection did not actuate
    - Reactor power is 3% and LOWERING
    - Intermediate range channels indicate negative SUR
    - Operators are verifying the reactor subcritical at step 7 of FR-S.1

    Based on the current plant conditions:

    (1) Boration _______________ required to continue after verifying the reactor is subcritical.

    (2) Which of the following describes the required procedural flowpath?

A. 1) is
   2) Return to E-0.

B. 1) is
   2) Remain in FR-S.1

C. 1) is **not**
   2) Return to E-0.

D. 1) is **not**
   2) Remain in FR-S.1

Question 95

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .E page 21 second bullet. SRO is required to have knowledge of the content of the procedures. Specifically the SRO must evaluate the plant conditions and determine which the procedure transition based upon the existing power level and SUR. This evaluation requires detailed knowledge of the EOP procedure content and transition criteria. Additionally, the SRO must decide what action is required related to continuing the boration flow. Knowledge of the procedure steps is required to make the decision and select the correct transition.

K/A is met with the EOP background knowledge that emergency boration is required to continue to ensure adequate shutdown margin during future cooldown. The candidate must also determine if conditions are satisfied to transition back to E-0, or stay in FR-S.1.

A. Correct: In FR-S.1, after verifying the Rx is subcritical in step 7, step 7c states "Continue boration as necessary to obtain adequate shutdown margin during subsequent actions." Per the background this is to ensure adequate S/D margin during the future plant cooldown. When power < 5% and negative IR SUR is achieved in FR-S.1, step 7d directs returning to the procedure and step in effect which is E-0.

B. Incorrect: Boration is required to continue to obtain adequate shutdown margin during subsequent actions. It is not required to remain in FR-S.1 once it has been verified that the reactor is subcritical. Step 7d directs returning to the procedure and step in effect which is E-0.

C. Incorrect: step 7c states "Continue boration as necessary to obtain adequate shutdown margin during subsequent actions". Returning to E-0 is correct since the reactor is subcritical.

D. Incorrect: step 7c states "Continue boration as necessary to obtain adequate shutdown margin during subsequent actions". It is not required to remain in FR-S.1 once it has been verified that the reactor is subcritical. Step 7d directs returning to the procedure and step in effect which is E-0.

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** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-S.1 Iss. 3 Rev. 0 pg. 5
1OM-53B.4.FR-S.1 Iss. 3 Rev. 0 pg. 82

Question Source: Bank – 2LOT15 NRC Exam (Q83)

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, Rev. 12 Obj. 5 - Discuss the general flowpath of each procedure including the importance of step sequencing, where applicable.

96. 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
    - A Field Operator reports that the 'A' MDAFW Pump electrical breaker has a Ground Overcurrent Relay flagged

    Which of the following completes the statements below?

    1) The current Technical Specification operational MODE is ______(1)______.

    2) In accordance with Technical Specifications, a change to the next higher MODE based on the conditions given is ______(2)______.

A. 1) Mode 3
   2) allowed if a risk assessment is performed

B. 1) Mode 4
   2) allowed if a risk assessment is performed

C. 1) Mode 3
   2) NOT allowed

D. 1) Mode 4
   2) NOT allowed

Answer: B

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .B page 17 first bullet. Specifically, the SRO must make a determination of the current plant mode, and based on knowledge of TS 3.7.5 and LCO 3.0.4, make a determination that a mode change to mode 3 is allowed iaw LCO 3.0.4.b.

K/A is met by giving plant conditions and determining the applicable Technical Specification Mode of Operation. This alone is RO knowledge, therefore the second part of the question was added to test knowledge of the specific system LCO and LCO 3.0.4 for the SRO to make a determination of whether a Mode change could be made while starting up the plant with a MDAFW pump inoperable. The SRO must make the determination that the 'A' MDAFW pump is inoperable due to the Ground Overcurrent Relay being flagged

A. Incorrect. Incorrect Mode. Mode 3 is ≥350F. Mode change to Mode 3 is allowed under LCO 3.0.4.b. (see correct answer)

B. Correct. Mode 4 is 350F>Tavg>200F. Mode change to Mode 3 is allowed under LCO 3.0.4.b. (When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made: b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications.

C. Incorrect. Incorrect Mode. Mode 3 is ≥350F. Mode change to Mode 3 is allowed under LCO 3.0.4.b. Plausible because there is a note on TS 3.7.5 stating that LCO 3.0.4.b is not applicable when entering MODE 1.

D. Incorrect. Mode 4 is correct for the given conditions. Mode change to Mode 3 is allowed under LCO 3.0.4.b. Plausible because there is a note on TS 3.7.5 stating that LCO 3.0.4.b is not applicable when entering MODE 1.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to determine Technical Specification Mode of Operation.

K/A# 2.2.35 **K/A Importance** 4.5 **Exam Level** SRO

References provided to Candidate None **Technical References:** TS Table 1.1-1 pg. 1.1-7
Tech Spec LCO 3.0.4 pg. 3.0-1 & 2
Tech Spec 3.7.5 pg. 3.7.5.1 & 2

Question Source: Bank – Robinson 2011 (Q96)

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.7 / 41.10 / 43.2 / 45.13)

Objective: 3SQS-RULES ITS-01-01 Rev.3 - Given plant conditions, apply the rules of ITS Section 3.0 to ensure compliance with Technical Specifications

3SQS-RULES ITS-01-03 Rev.3 - Given plant conditions, determine the plant MODE in accordance with the ITS.

97. Per Technical Specifications, which of the following describes a Surveillance Requirement (including the required Frequency and MODE) that is applicable to the Feedwater Isolation "SG Water Level – High-High" initiation function?

(REFERENCES PROVIDED)

A. A COT (CHANNEL OPERATIONAL TEST) must be performed every 184 days while in MODE 1.

B. A TADOT (TRIP ACTUATING DEVICE OPERATIONAL TEST) must be performed every 18 months while in MODE 3.

C. A CHANNEL CHECK must be performed every 12 hours while in MODE 4.

D. A CHANNEL CALIBRATION must be performed every 92 days while in MODE 2.

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .B page 17 first bullet. SRO must understand the application of Required Actions and Surveillance Requirements in accordance with rules of application requirements. Specifically, SRO must know how to apply the TS Surveillance requirements for an instrument, and have an understanding of the required modes of applicability, and surveillance frequencies.

K/A is met demonstrating the ability to determine the required TS surveillance, applicable modes, and required frequency to ensure the SG Water Level – High-High" initiation function is operable.

A. Correct. The Channel Operational Test (COT) (SR 3.3.2.4) is required every 184 days when in modes 1-3 per TS Table 3.3.2-1, function 5.b. Tech Spec Table 3.3.2-1 function 5.b identifies the applicable modes as 1-3, and the following surveillances as being required to maintain the function operable. (SR 3.3.2.1, 3.3.2.4, 3.3.2.8, 3.3.2.9).

B. Incorrect. A Trip Actuating Device Operational Test (TADOT) (SR 3.3.2.7) is not a required surveillance function 5.b. Plausible because a TADOT is a required surveillance of TS 3.3.2, and the frequency and modes do coincide with other instrumentation, but not SG Water Level – High-High.

C. Incorrect. A Channel Check (SR 3.3.2.1) is required every 12 hours in modes 1-3. Plausible distractor because the frequency is correct, but mode 4 is incorrect.

D. Incorrect. A Channel Calibration (SR 3.3.2.8) is required in modes 1-3, but the frequency is 18 months. Plausible distractor because the mode is correct, but 92 days is incorrect.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to determine operability and/or availability of safety related equipment.

K/A# 2.2.37 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate TS 3.3.2 pgs. 3.3.2-(1-13) Unit 1 Surveillance Test Intervals Rev. 0

Technical References: TS 3.3.2 Amend. 292/179 pg. 3.3.2-(5, 6, 11) Unit 1 Surveillance Test Intervals Rev. 0 pg. 5

Question Source: Bank – Vision 254264

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

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

98. Unit 1 and Unit 2 are at 100% power with all systems in NSA.

- RWDA-L has been prepared for discharging 1BR-TK-4B, Coolant Recovery Tank
- After the RWDA-L is approved by Radiation Protection, the Unit 1 SM or US is then required to review the RWDA-L to confirm the status of various items as part of the approval process

In accordance with 1OM-17.4.AN, Discharging a Coolant Recovery Tank to Cooling Tower Blowdown, which of the items below does the 'APPROVED BY UNIT 1 SM' signature block on the RWDA-L denote?

1. Adequate dilution flow exists.
2. Midland Water Treatment Plant has been notified.
3. Appropriate alarms have been properly adjusted.

A. 1 ONLY

B. 1 and 2 ONLY

C. 1 and 3 ONLY

D. 2 and 3 ONLY

Answer: C

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .A page 17 fourth bullet. Conditions and limitations in the facility license. National Pollutant Discharge Elimination System (NPDES) requirements, if applicable. Specifically what the SRO is signing for when approving a RWDA-L for a liquid waste discharge.

K/A is met by demonstrating the knowledge of the specific conditions which must be met for the Unit 1 Shift Manager/Unit Supervisor to sign the approval for the discharge release permits.

A. Incorrect. Plausible distractor because adequate dilution flow is correct, but appropriate alarms have been properly adjusted is also covered under this approval signature.

B. Incorrect. Plausible distractor because Midland Water Treatment Plant has to be notified when an offsite radioactive release occurs as part of the offsite protective action recommendations, but not for discharging. Adequate dilution flow is correct, alarms must be properly adjusted also.

C. Correct. Per 1OM-17.4.AN, the note prior to the, procedural step for the U1 SM to sign the block states "Unit 1 SM/US signature on the RWDA-L denotes the following: 1. Authorization to discharge the tank, 2. Only one batch RWDA-L is being discharged from the site at one time, 3. Adequate dilution flow exists, and 4. Appropriate alarms have been properly adjusted.

D. Incorrect. Plausible distractor because Midland Water Treatment Plant has to be notified when an offsite radioactive release occurs as part of the offsite protective action recommendations, but not for discharging. Alarms have been properly adjusted is correct, but adequate dilution flow is also covered under this approval signature.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Generic	Ability to approve release permits.

K/A# 2.3.6 **K/A Importance** 3.8 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-17.4.AN Rev. 7 pg. 20

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.13 / 43.4 / 45.10)

Objective: 1SQS-17.1, Rev. 15 Obj. 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.

3SSG-ADMIN-01-23: EXPLAIN the approval requirements for radiological release permits in accordance with 1/2-ENV-05.04, Radioactive Waste Discharge Authorization - Liquid. 1/2-ENV-05.05, Radioactive Waste Discharge Authorization - Gas.

99. The plant has been in Mode 6 for 3 weeks, and the following conditions exist:

- Fuel movement is in progress
- The following annunciators associated with this event are LIT:
  - A6-3, spent fuel pool level low
  - A6-30, Refueling Cavity Level Low
  - A1-49, Containment Sump Level High
- Refuel SRO reports Refuel Cavity level is below minimum Tech Spec level and LOWERING
- Both CNMT Sump Pumps are running and level is RISING
- RWST level is 21' 2" and STABLE
- Leakage around the Reactor Cavity Seal has been reported from Containment
- A used fuel assembly has just been unlatched and is in the Upender in the upright position
- 1FH-1, Fuel Transfer Tube Gate Valve is CLOSED

In accordance with ARP A6-30, Refueling Cavity Level Low, which of the following completes the statements below?

The fuel assembly must be placed in ______(1)______.

The initial method of makeup to the reactor cavity during the leak will be the ______(2)______.

A. 1) a horizontal position
   2) RWST via the LHSI system per 1OM-11.4.D, Filling Reactor Refueling Cavity

B. 1) a horizontal position
   2) Blender per 1OM-7.4.L, Blender Boration Operation

C. 1) an open area inside the core
   2) RWST via the LHSI system per 1OM-11.4.D, Filling Reactor Refueling Cavity

D. 1) an open area inside the core
   2) Blender per 1OM-7.4.L, Blender Boration Operation

QUESTION 99

Answer: A

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .G page 23 first bullets. Fuel handling facilities and procedures. Refuel floor SRO responsibilities. Specifically, the candidate must have knowledge that the irradiated fuel must be placed in the horizontal position to maximize the time during a refuel cavity leak to prevent a high radiation condition on the refuel deck. Also meets SRO only guidance of ES-401 Attachment 2 per section II .E page 21 third bullet. SRO is required to have knowledge of the content of the procedures. Specifically the SRO must know that makeup water to the refuel cavity is from the RWST per the ARP for lowering refuel cavity level. Detailed knowledge of the content is required to select the correct procedural direction.

K/A is met with the knowledge that the fuel assembly must be placed horizontal during fuel movement when a leak occurs. By placing the assembly horizontal, more time is available to resolve the cavity leak to prevent high radiation condition on the refuel deck.

A. Correct. With the fuel assembly in the Upender in the upright position, the ARP states to place the fuel assembly and upender in the horizontal position. The preferred method of makeup to the Refuel cavity per the ARP if the RWST via the LHSI pump due to the available makeup capabilities of the RWST and low head system.

B. Incorrect. First part is correct. Second part is plausible because it could be an available makeup source to the cavity, but the ARP does not direct this as a makeup source. The ARP clearly states to commence makeup to the cavity by performing 1OM-11.4.D using the RWST

C. Incorrect. Plausible distractor since the upender is upright and the candidate may feel that using the manipulator crane and placing the assembly in an open area of the core would be better, but laying it horizontal is specifically addressed in the ARP. Second part is correct.

D. Incorrect. Plausible distractor since the upender is upright and the candidate may feel that using the manipulator crane and placing the assembly in an open area of the core would be better, but laying it horizontal is specifically addressed in the ARP. Second part is plausible because it could be an available makeup source to the cavity, but the ARP does not direct this as a makeup source. The ARP clearly states to commence makeup to the cavity by performing 1OM-11.4.D using the RWST

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

K/A# 2.3.13 **K/A Importance** 3.8 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-20.4.AAP Rev. 16 pg. 2-5; 1OM-11.4.D Rev. 16 pg. 2

Question Source: Bank – Farley 2011 NRC Exam (Q96)

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.12 / 43.4 / 45.9 / 45.10)

Objective: 3SQS-6.12, Rev. 7 Obj. 9 - IDENTIFY the safe Fuel Assembly storage locations in the event of loss of water resulting from a Refueling Cavity seal failure during fuel movement. (SOER 85-1 Rx Cavity Seal Failure Rec. 5a, 5b).

3SQS-6.13, Rev. 6 Obj. 8 - Given a Fuel Handling System alarm condition and using the Alarm Response Procedure(s), determine the appropriate alarm response, including automatic and operator actions in the control room.

100. The following conditions exist:

- A manual Rx trip and Safety Injection was initiated 10 minutes ago based on criteria in AOP 1.6.7, Excessive Primary Plant Leakage
- Pressurizer level is off-scale LOW
- Pressurizer pressure is 1500 psig and LOWERING
- All SG levels are 5% NR and slowly RISING
- All SG pressures are 1005 psig and STABLE
- All main steam line radiation monitors are reading 20 cpm
- RM-1VS-105, Leak Collection Area Gas Monitor is reading 1.8 E2 cpm in High alarm.
- PI-1LM-100B, Containment pressure is 13.5 psia
- A11-28, Safeguards Area Sump Level High alarm is LIT

Upon exiting E-0, which of the following is the correct procedure transitions for the event in progress?

A. Go to E-1, Loss of Reactor or Secondary Coolant, **then** ECA-1.2, LOCA Outside Containment, **then** ECA-1.1, Loss of Emergency Coolant Recirculation.

B. Go to E-1, Loss of Reactor or Secondary Coolant, **then** ECA-1.1, Loss of Emergency Coolant Recirculation, **then** ECA-1.2, LOCA Outside Containment.

C. Go to ECA-1.1, Loss of Emergency Coolant Recirculation, **then** ECA-1.2, LOCA Outside Containment.

D. Go to ECA-1.2, LOCA Outside Containment, **then** ECA-1.1, Loss of Emergency Coolant Recirculation.

Answer: D

Explanation/Justification: Meets the requirements of the SRO only guidance of ES-401 Attachment 2 per section II .E page 21 second bullet, Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific sub-procedures or emergency contingency procedures. The procedure selection knowledge required by this question is beyond the entry conditions of major EOPs and FRPs; thereby testing the procedure selection at the SRO level.
K/A is met by requiring knowledge of how the different subsets of emergency procedures are used in conjunction with each other to combat an event.

A. Incorrect. Plausible distractor with PRZR level off scale low, but no criteria for transition to E-1 have been met (CNMT radiation, pressure, and sump levels are normal). ECA-1.2 and ECA-1.1 are in the proper procedural sequence, but ECA-1.2 is a transition from step 20 of E-0.
B. Incorrect. Plausible distractor with PRZR level off scale low, but no criteria for transition to E-1 have been met (CNMT radiation, pressure, and sump levels are normal). ECA-1.2 and ACA-1.1 are also out of procedural sequence.
C. Incorrect. ECA-1.2 and ECA-1.1 are the correct procedures, but are not in the proper procedural sequence. ECA-1.2 is a transition from step 20 of E-0, and ECA-1.1 transition occurs at step 4 of ECA-1.2 when the leak is not isolated.
D. Correct. With a LOCA outside containment that cannot be isolated, the correct transitions following E-0 are as stated in this choice. See supporting copy of E-0 which shows the correct flow path through ECA-1.2 to ECA-1.1.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.

K/A# 2.4.5 **K/A Importance** 4.3 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.E-0 Iss. 3 rev. 0 pg. 17
1OM-53A.1.ECA-1.2 Iss. 2 rev. 1 pg. 3

Question Source: Bank – Surry 2010 NRC Exam (Q100)

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.