Given the following plant conditions and sequence of events:

- The plant is operating at 100% power.
- 1MSP-1.05-I, "Reactor Protection System Train B Test" is being performed.
- Reactor Trip Bypass Breaker (BYB) has been closed.
- "B" Train SSPS input error inhibit switch is in the INHIBIT position.
- A turbine trip occurs.
- All systems function as designed.

Which ONE of the following breakers will receive a trip signal?

RTA – (Reactor Trip Breaker "A")
RTB – (Reactor Trip Breaker "B")
BYA – (Reactor Trip Bypass Breaker "A")
BYB – (Reactor Trip Bypass Breaker "B")

A. RTA **ONLY**.

B. RTA and BYB.

C. RTB and BYA.

D. RTB and BYB.

Answer: B

Explanation/Justification:

A. Incorrect. Train A RTB will open, however, it is not the only breaker which will open. BYB will also open.

B. Correct. Since reactor power is > P-9 (49%), a turbine trip will result in a reactor trip signal to RPS. With Train B SSPS input error inhibit switch in INHIBIT, upon reactor trip, the Train A reactor protection system de-energizes the Train A Reactor trip Breaker and the Train B Bypass breaker UV coils. BYB is closed to allow testing on the "B" Train.

C. Incorrect. Train B RTB will not open due to Train B SSPS input error inhibit switch in the INHIBIT position. BYA will not open.

D. Incorrect. Train B RTB will not open due to Train B SSPS input error inhibit switch in the INHIBIT position. BYB will open.

Sys #	**System**	**Category**	**KA Statement**
007	Reactor Trip	Ability to determine or interpret the following as they apply to a reactor trip:	Reactor trip breaker position

K/A# EA2.03 **K/A Importance** 4.2 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-1.1B, Rev. 10, pg. 3
1SQS-1.2 U1 LP PPT, Rev. 5 Issue 1, pg. 7 & 22
1MSP-1.05-I, Issue 4, Rev. 49, pg. 26 & 29

Question Source: Bank – Vision 8266

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

Objective: 3SQS-1.2-01-03 Describe the control, protection, and interlock functions for the field components associated with RPS hardware, including automatic functions, setpoints and changes in equipment status as applicable.

Given the following plant conditions:

- Reactor Power is 85%, steady state, all systems in NSA.
- Pressurizer (PRZR) pressure control is in its normal configuration.
- Pressurizer Relief Valve (PCV-1RC-455C) inadvertantly lifts and does **NOT** fully reseat.
- A4-11, "PRESSURIZER CONTROL PRESSURE LOW", annunciates.
- PRZR Pressure is 2170 psig and slowly lowering.

How will the PRZR Spray Valves (PCV-1RC-455A and PCV-1RC-455B) and PRZR PORV Block Valves (MOV-1RC-535, 536, 537) respond to these plant conditions, with no operator action?

PRZR Spray Valves ___ (1) ___.
PRZR PORV Block Valves ___ (2) ___.

A. (1) remain open
(2) remain open

B. (1) close
(2) remain open

C. (1) remain open
(2) close

D. (1) close
(2) close

Answer: B

Explanation/Justification:

A. Incorrect. Spray Valves auto close on lowering PRZR pressure versus remain open. If candidate does not understand PZR Pressure Control scheme, they may have a misconception on how this control works. Correct that block valves will remain open (NSA Normal).

B. Correct. A lowering PRZR pressure due to the vapor space leak caused by PORV leakage results in a low pressure alarm when PRZR pressure drops to 2185 psig. This lowering demand signal closes spray valves as shown on technical references. The PRZR PORV Block valves will remain open due to control switches NSA Normal until the operator takes action to close these valves.

C. Incorrect. PRZR Spray Valves close on lowering PRZR pressure versus remain open. Plausible that PRZR Block Valves will automatically close when 2/3 PRZR pressures channels are @ 2185 psig if the control switch was in AUTO versus OPEN since this is the Unit 2 design.

D. Incorrect. Correct Spray Valve Position. Plausible that PRZR Block Valves will automatically close when 2/3 PRZR pressures channels are @ 2185 psig if the control switch was in AUTO versus OPEN since this is the Unit 2 design.

Sys #	System	Category	KA Statement
008	Pressurizer Vapor Space Accident (Relief Valve Stuck Open)	Knowledge of the interrelationship between the pressurizer vapor space accident and the following:	Valves

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

References provided to Candidate None **Technical References:** 1OM-6.4.IF, Rev. 11, pg. 23 & 24
1SQS-6.4, Rev. 11, pg. HO-54
1OM-6.3.C, Issue 4, Rev. 8, Pg. 14
1OM-6.4.ABU, Issue 3, Rev. 0, pg. 1

Question Source: Bank – Vision # 82001 (2LOT7 NRC Exam – Q#2)

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

ʲjective: 1SQS-6.4 19. Given a specific plant condition, predict the response of the pressurizer and pressure relief system control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off normal condition: (Excessive Primary Plant Leakage, RCS voiding, process instrument failure)

Given the following plant conditions:

- Unit 1 experienced a Small Break Loss of Coolant Accident (SBLOCA).
- A manual Reactor Trip and Safety Injection was initiated.
- RCS pressure is 785 psig and stable.
- The hottest Loop THot indication is 471°F and stable.
- The average of the FIVE hottest CET's is 483°F and stable.
- Total Feed Flow is 600 gpm and stable.
- Pressurizer Level is 15% and stable.
- Containment Pressure is 5.5 psig and stable.
- Operators are determining whether conditions are present to allow a transition to ES-1.1, "SI Termination", from E-0, "Reactor Trip or Safety Injection".

The Unit Supervisor asks the Reactor Operator to determine RCS subcooling using Steam Tables. Which of the following identifies current RCS subcooling, and whether a transition to ES-1.1 is appropriate? (**Reference Provided**)

Subcooling is approximately ___ (1) ___, and the transition to ES-1.1 ___ (2) ___ be made.

A. (1) 35°F
(2) shall

. (1) 47°F
(2) shall

C. (1) 35°F
(2) shall **NOT**

D. (1) 47°F
(2) shall **NOT**

Answer: C

Explanation/Justification:

A. Incorrect. Correct subcooling margin @ 35°F. Incorrect that transition criteria is met. (refer to correct answer explanation)

B. Incorrect. If the candidate uses Thot as opposed to CET to calculate subcooling, it will come out to 47°F which makes this a plausible number. however, the transition criteria is still not met based on PRZR level (refer to correct answer explanation).

C. Correct. 785 psig equals 800 psia. Saturation temperature for 800 psia is 518.21°F IAW Steam Tables. Subcooling is 35.21°F. The subcooling criteria > 54°F is NOT met. Since subcooling criteria is NOT met, the RNO requires the use of Attachment 6-A which is provided. At 785 psig, the 35 F would be met based on adverse criteria , however, the required 38% PRZR level is not met since adverse containment numbers apply due to containment pressure > 5 psig. Based on these numbers, transition criteria is NOT met and a transition to ES-1.1 shall NOT occur. It is reasonable that the candidate know SI termination criteria from memory, so therefore this criteria is NOT provided.

D. Incorrect. Incorrect subcooling value. Correct that transition shall not be made. (refer to correct answer explanation).

Sys #	System	Category	KA Statement
009	Small Break LOCA	Knowledge of the interrelations between the small break LOCA and the following:	S/Gs

K/A# EK2.03 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate Steam Tables (Red Book) E-1 Attachment 6-A **Technical References:** 1OM-53A.1.E-1, Rev. 14, pg. 6, 1OM-53B.4.E-1, Rev. 14, pg. 53 1OM-53B.5.GI-11, Issue 2, Rev. 0, pg. 2 & 5,Steam Tables (Red Book),

Question Source: Bank – Vision #82002 (2LOT7 NRC – Q#3)

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

Objective: 3SQS-53.2 3. State from memory the basis for SI termination criteria IAW BVPS –EOP Executive Volume.
3SQS-53.3 6. Given a set of plant conditions, locate and apply the proper EOP IAW BVPS –EOP Executive Volume.

Given the following plant conditions:

- A beyond design basis earthquake occurs resulting in a LBLOCA & Station Blackout.
- The Control Room Team is performing actions contained in ECA-0.0, "Loss of All Emergency 4KV AC Power" to restore power.
- RCS Subcooling based on Core Exit TC's is 2 °F.
- Core Exit TC's are 735 °F and slowly RISING.
- RVLIS Full Range indication is 38% and slowly DROPPING.

Which ONE of the following describes the status of the Core Cooling Critical Safety Function Status Tree, AND what method will be used to determine their status?

A. RED Path; **may** be determined using IPC.

B. ORANGE Path; **may** be determined using IPC.

C. RED Path; **must** be determined using MCB indications because IPC is unavailable.

D. ORANGE Path; **must** be determined using MCB indications because IPC is unavailable.

Answer: A

Explanation/Justification:

A. Correct. Conditions present indicate a RED CSF Status. (ie; > 719 F and < 40% RVLIS) IPC is available during a station blackout to monitor core cooling status. The IPC is powered by Battery Backed DC-SWBD-3 since normal MCC1-E9 supply is lost due to station blackout.

B. Incorrect. Incorrect CSF conditions. Conditions present indicate a RED versus ORANGE CSF Status. Correct IPC status.

C. Incorrect. Correct CSF Status. It is incorrect that MCB indications must be used although they are available because IPC is still available. This is plausible if the candidate does not know the status of IPC.

D. Incorrect. Incorrect CSF conditions. It is incorrect that MCB indications must be used although they are available because IPC is still available. This is plausible if the candidate does not know the status of IPC.

Sys #	System	Category	KA Statement
011	Large Break LOCA	Ability to determine and interpret the following as they apply to a Large Break LOCA:	Verification of adequate core cooling.

K/A# EA2.10 **K/A Importance** 4.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.F-0.2, Issue 1C, Rev. 1, pg 1
1OM-5A.3.C, Rev. 10 pg. 4 & 6
3SQS-39.1, Rev. 8 PPNT Slides

Question Source: New

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

Objective: 3SQS-53.1 2. Concerning critical safety function restoration, IAW BVPS EOP Executive Volume, state from memory the following, The CFS in order of priority, The priorities of the color-coded end points of the CSFSTs, and the red path summary conditions from EOPs.

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

Given the following plant conditions:

- The Unit was operating at 60% when all charging flow was lost.
- The control room crew entered AOP 1.7.1, "Loss of Charging Or Letdown".
- Letdown has been isolated and charging restoration is being investigated.
- The Reactor Operator reports Pressurizer (PRZR) level is lowering at a rate of 1 % every five (5) minutes.
- PRZR Level was 4% below reference when letdown was isolated.

If charging flow is **NOT** restored, which ONE of the following is the **LONGEST** time that PRZR heater operation can be maintained?

A. 40 minutes

B. 85 minutes

C. 125 minutes

D. 145 minutes

Answer: C

Explanation/Justification:

A. Incorrect. Value reflects 22% - 14% = (8%) (5) = 40 minutes. Incorrect initial level.

B. Incorrect. Value reflects 39% - 22% = (17) (5) = 85 minutes Incorrect final level

C. Correct. At 60% power, PRZR Level is 60% of full range (0% power = 547 F = 22% PRZR Level / 100% power = 578 F = 57 % PRZR Level) Therefore at 60% PRZR level = 43%. The stem of the question indicates PRZR Level was 4% less than reference which = 39%. PRZR Level cutout automatically occurs when PRZR level reaches 14%. Therefore with level lowering 1% every 5 minutes, it will take 125 minutes to reach 14%. 125 minutes is the longest time that PRZR heater operation can be maintained based on these plant conditions. All distractors are plausible based on misconceptions or common errors.

D. Incorrect. Value reflects 43% - 14% = (29%) (5) = 145 minutes. Did not subtract 4% below reference level.

Sys #	**System**	**Category**	**KA Statement**
022	Loss of Reactor Coolant Makeup	Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup:	How long PZR level can be maintained within limits.

K/A# AA2.04 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1SQS-6.4 PPNT Slide # 56, Rev. 11; 1OM-52.4.B, Rev. 40, pg 141 & 142

Question Source: New

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

Objective: 1SQS-7.1 20. Given a specific plant condition, predict or describe the response of the Chemical and Volume Control System control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant conditions or for an off normal condition (ie. Excessive Primary Plant /CVCS Leakage)

Beaver Valley Unit 1 NRC Written Exam (1LOT8)

Given the following plant conditions:

- The Unit is in Mode 5 with "A" Train of Residual Heat Removal (RHR) in service.
- "B" RHR Train is OPERABLE but **NOT** in operation.
- No Reactor Coolant Pumps (RCPs) are in operation, however, loops are filled.
- All Steam Generators are > 30% NR level.
- Annunciator A1-127, "RESIDUAL HEAT REMOVAL PP AUTO STOP" is received.
- No operator action has been taken.
- All systems function as designed.

Based on these plant conditions, which ONE of the following describes the operational implications?

RHR system flow will ___ (1) ___ and a Technical Specification entry will ___ (2) ___.

A. (1) drop to zero
(2) be required

B. (1) drop to zero
(2) **NOT** be required

(1) drop and then return to original value
(2) be required

D. (1) drop and then return to original value
(2) **NOT** be required

Answer: A

Explanation/Justification:

A. Correct. The candidate must recognize that a Loss of "A" RHR Pump has occurred based on stated plant conditions. Since there is no auto start of the "B" RHR Pump, system flow will drop to zero and remain there until operator action is taken to start the standby pump. The candidate must also recognize that TS LCO 3.4.7 requires a TS entry (< 1hour actions statement) applies to the given set of plant conditions.
B. Incorrect. Correct system flow response. Incorrect that there is no TS implication.
C. Incorrect. Plausible that the candidate may believe there is an auto start of the "B" RHR pump. Correct that a TS entry is required.
D. Incorrect. Plausible that the candidate may believe there is an auto start of the "B" RHR pump. It is also plausible that if the standby pump auto started that with all S/Gs > 28%, no TS entry is required.

Sys #	System	Category	KA Statement
025	Loss of RHR System	Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System:	Loss of RHRS during all modes of operations.

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

References provided to Candidate None **Technical References:** 1OM-10.4.AAC, Rev. 2, pg. 2
1OM-10.1.D, Rev 1, pg. 3
BVPS Unit 1 & 2 TS 3.4.7, Amend 278/161, pg 3.4.7-1 & 2

Question Source: New

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

ective: 1SQS-10.1 18. Given a specific plant condition, predict the response of the RHR system control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant conditions or for an off normal condition.
22. From memory, and for a given set of plant conditions, determine if the conditions meets the criteria for entry into a less than or equal to one hour action statement in accordance with the Technical Specifications.

Given the following plant conditions:

- The plant is operating at 100% power with all systems in NSA.
- [FI-1CC-107A], "RCP 1A Thermal Barrier Flow" has increased to 55 GPM and continues to slowly RISE.
- [LI-1CC-100], "CCR Surge Tank Level" is slowly DROPPING.

Which ONE of the following is the cause of these conditions **AND** what automatic action will occur if these trends continue?

The cause of these conditions is a ____ (1) ____ **AND** with no operator action the ___ (2) ___.

A. (1) 1A RCP Thermal Barrier Leak
(2) 1A RCP thermal barrier outlet isolation valve will CLOSE.

B. (1) CCR Leak downstream of 1A RCP Thermal Barrier
(2) 1A RCP thermal barrier outlet isolation valve will CLOSE.

C. (1) 1A RCP Thermal Barrier Leak
(2) inlet and outlet containment isolation valves for 1A RCP will CLOSE.

(1) CCR Leak downstream of 1A RCP Thermal Barrier
(2) inlet and outlet containment isolation valves for 1A RCP will CLOSE.

Answer: B

Explanation/Justification:

A. Incorrect. Plausible that 1A RCP thermal barrier could be leaking based on increasing FI-1CC-107A flow, however, the candidate must differentiate out leakage from the system versus in leakage based on CCR Surge Tank level change. Correct that a high flow condition at 58 gpm will close the 1A RCP thermal barrier outlet valve.

B. Correct. Increasing flow on FI-1CC-107A combined with a dropping CCR Surge Tank level is indicative of a leak downstream of 1A RCP thermal barrier. TV-1CC-107A will auto close when CCR Flow as sensed by FT-1CC-107A increases to 58 GPM. Dropping Surge Tank level is an entry condition for Loss of CCR AOP.

C. Incorrect. Refer to discussion in A. Incorrect that the inlet and outlet CI valves close on this condition. They close on CIA only.

D. Incorrect. Correct condition. Incorrect auto action (refer to above discussion)

Sys #	System	Category	KA Statement
026	Loss of Component Cooling Water	Ability to operate and/or monitor the following as they apply to the Loss of Component Cooling Water:	Loads on the CCWS in the control room.

K/A# AA1.02 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None

Technical References:
1SQS-15.1, Rev. 11 PPNT Slides
1OM-53C.4.1.15.1, Rev. 4, Pg. 1, 2, & 4
1OM-15.1.E, Rev. 4, Pg. 5 & 8
1OM-15.1.D, Issue 4, Rev. 1, Pg. 5

Question Source: New

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

Objective: 1SQS-15.1 17. Given in-leakage or out-leakage to/from the CCR system, describe all the means by which the leakage can be detected.

Given the following plant conditions and sequence of events:

- The Plant is operating at 50% power.
- Control rods are in MANUAL.
- Pressurizer (PRZR) 2A & 2B Backup Heaters are in the ON position.
- The Pressurizer (PRZR) Master pressure controller output failed **AS IS.**
- A secondary load rejection (step load decrease of 10%) occurs.
- No operator action is taken.

Based on these plant conditions, what will be the impact of the secondary load rejection on PRZR water level **AND** the two groups of energized PRZR Backup Heaters (2A & 2B)?

PRZR Water Level will **INITIALLY** ___ (1) ___ AND energized PRZR heaters will ___ (2) ___.

A. (1) decrease
(2) de-energize

B. (1) decrease
(2) remain energized

C. (1) increase
(2) de-energize

D. (1) increase
(2) remain energized

Answer: D

Explanation/Justification:

A. Incorrect. PRZR level decrease reflects a load increase vs. decrease. Plausible if candidate has concepts backwards or if they are confusing normal downpower response (lowering PRZR level). PRZR heaters will remain energized. Plausible if the candidate believes PRZR level drops to 14% which cuts off PRZR heaters by interlock.

B. Incorrect. PRZR level reflects a load increase vs. decrease. Correct PRZR Backup Heater response.

C. Incorrect. Correct PRZR level response. PRZR heaters will remain energized. Plausible because PRZR heaters are designed to turn off with increasing PRZR pressure.

D. Correct. A load rejection results in an increase in RCS temperature. (Plant will not trip due to reactor power level) The Tavg increase will cause an expansion of water into the PRZR (Insurge) compressing the vapor space which in turn will increase PRZR pressure. Backup PRZR heaters 2A & 2B will remain energized because they are energized on and the master pressure controller has failed at a setpoint which will not cause them to turn off regardless of what happens to PRZR pressure following the transient.

Sys #	System	Category	KA Statement
027	Pressurizer Pressure Control System Malfunction	Knowledge of the operational implications of the following concepts as they apply to Pressurizer pressure Control Malfunctions:	Expansion of liquids as temperature increases.

K/A# AK1.02 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.1.C, Rev. 7, pg.23 & 24
1OM-6.4.IF, Rev. 11, pg. 12 & 23
1SQS-6.4, Rev. 11, Issue 1, pg HO 18 & 74 & 75

Question Source: New

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

jective: 1SQS-6.4 19. Given a specific plant condition, predict the response of the PRZR and Pressure Relief System control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant conditions or for an off-normal condition (ie: Process instrument failure).

Given the following plant conditions:

- The Unit has been operating at 100% power for 345 days.
- A **VALID** high pressurizer pressure reactor trip signal is received and the reactor **DOES NOT** automatically trip, and it **CANNOT** be tripped manually from the control room.
- The Control Room Team is performing FR-S.1, "Response to Nuclear Power Generation - ATWS" actions.

For these conditions, WHAT is the basis for tripping the turbine?

Turbine trip ______________________________

A. removes a large source of positive reactivity addition.

B. prevents the main feed pumps from tripping on low suction pressure.

C. provides an additional reactor trip signal to the reactor protection system.

D. maintains the pressurizer pressure relief system within its relief capability.

Answer: A

Explanation/Justification:

A. Correct. IAW the bases for step 1 and 5 of FR-S.1. The turbine removes a potential RCS cooldown which would add positive reactivity from the negative MTC.

B. Incorrect. Tripping the turbine should improve the feed pump suction pressure. However, this is not the basis for tripping the turbine during an ATWS event. Tripping the turbine also conserves SG water inventory. In the event of a loss of feed induced ATWS conserving water inventory is a primary purpose for tripping the turbine. The candidate may link the loss of feed to the low suction pressure trip on the main feed pumps.

C. Incorrect. The candidate will need to understand the fundamentals of a negative MTC in order to arrive at the correct answer. The Turbine trip will send an additional Rx trip signal to RPS. However, this is not the basis for tripping the turbine during an ATWS event.

D. Incorrect. For the various analyzed ATWS events, RCS pressure does rise, and the pressure relief system will function to keep RCS pressure within acceptable limits. However, this is not the basis for tripping the turbine during an ATWS event. One of the design criteria for the pressurizer is to keep it operable for a variety of analyzed events. The candidate may believe that this is one of the events that challenges these design criteria and tripping of the turbine is necessary to keep the pressurizer operable.

Sys #	System	Category	KA Statement
029	Anticipated Transient Without Scram (ATWS)	Knowledge of the reasons for the following responses as they apply to the ATWS:	Actions contained in EOP for ATWS.

K/A# EK3.12 **K/A Importance** 4.4 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-S.1, Issue 1C, Rev. 5, pg. 2 & 4
1OM-53B.4.FR-S.1, Issue 1C, Rev. 5, pg. 57 & 62

Question Source: Bank – Vision # 82007 (2LOT7 NRC Exam – Q#8)

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 1. State from memory the basis and sequence of major action steps of each EOP IAW BVPS-EOP Executive Volume.

. During power operation, Steam Generator (S/G) Tube leakage was detected and estimated at 100 gpm when RCS pressure was 2200 psig and S/G pressure was 800 psig.

Which ONE of the following is the approximate current leak rate if RCS pressure is 1350 psig and S/G pressure is 1000 psig? (Assume break size has **NOT** changed)

S/G Tube leak rate __________

A. decreases to approximately 25 gpm.

B. decreases to approximately 50 gpm.

C. increases to greater than 100 gpm.

D. remains equal to initial leak rate of 100 gpm.

Answer: B

Explanation/Justification:

A. Incorrect. Plausible if candidate misunderstands fundamentals and/or miscalculates.

B. Correct. The leak rate corresponds to the square root of the differential pressure. 2200 – 800 = 1350 Square root = 37 / 1350 -1000 = 350 Square root = 18. Therefore the leak rate drops by about half. (ie: 18/37 = .486)
Incorrect. Plausible if candidate misunderstands fundamentals and/or miscalculates.

D. Incorrect. Plausible if candidate misunderstands fundamentals and/or miscalculates.

Sys #	System	Category	KA Statement
038	Steam Generator Tube Rupture	Knowledge of the operational implications of the following concepts as they apply to the SGTR:	Leak rate vs. pressure drop.

K/A# EK1.02 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** GO-GPF.T6, Rev. 2

Question Source: New

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

Objective: GO-GPF.T6 27. Describe different type of fluid measuring devices.

Given the following plant conditions:

- The Unit is operating at Full Power with all systems in NSA.
- A Large Steam Line Break occurs outside containment in the turbine building.
- All systems function as designed.
- No operator action occurs.

Which ONE of the following describes the status of the following components?

1FW-P-1A ("A" Main Feedwater Pump)
HYV-1FW-100B (1B S/G Main FW CNMT Isol Vlv)
FCV-1FW-499 (1C S/G FW Bypass FCV)
FCV-1FW-478 (1A S/G Main FW Reg Vlv)

	1FW-P-1A	HYV-1FW-100B	FCV-1FW-499	FCV-1FW-478
A.	RUNNING	OPEN	OPEN	CLOSED
B.	RUNNING	CLOSED	CLOSED	CLOSED
C.	STOPPED	CLOSED	CLOSED	OPEN
	STOPPED	CLOSED	CLOSED	CLOSED

Answer: D

Explanation/Justification:

A. Incorrect. This configuration reflects a partial FWI which occurs on a reactor trip with 2 of 3 Tavg signals < 554 F.
B. Incorrect. MFP trips on Full FWI.
C. Incorrect. MFRV closes on a Full FWI.
D. Correct. This is the correct configuration for a Full FWI which has occurred due to initiation of SI on low steam line pressure as a result of the SLB.

Sys #	System	Category	KA Statement
040	Steam Line Rupture Excessive Heat Transfer	Ability to operate and / or monitor the following as they apply to the Steam Line Rupture:	Feedwater isolation.

K/A# AA1.02 **K/A Importance** 4.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-24.1.D, Rev. 5, pg. 7
1OM-53A.1.1-C, Issue 1C, Rev. 1, pg 3
Unit 1 ECCS Setpoints

Question Source: New

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

Objective: 1SQS-24.1 16. Given a specific plant condition, predict the response of the Main Feedwater, Dedicated Auxiliary Feedwater System, Auxiliary Feedwater System or SGWLC System control room indication and control loops, including all automatic functions and changes in equipment status for either a change in plant condition or off normal condition.

Given the following plant conditions:

- The Unit is operating at 75% power with all systems in NSA.
- One Main Feedwater Pump has tripped.
- SG 1A NR levels are ALL indicating 23% and lowering.
- SG 1B NR levels are ALL indicating 22% and lowering.
- SG 1C NR levels are ALL indicating 21% and lowering.
- Pressurizer pressure is indicating 2310 psig and rising on all protection pressure channels.
- Plant Operation at power continues and no operator action has been taken.

Based on the **current** conditions, which ONE of the following reflects reactor status **AND** required operator action, if any?

A reactor trip ____ (1) ____ automatically occurred.
AOP 1.24.1, "Loss of Main Feedwater", ____ (2) ____ a reactor trip.

A. (1) should have
(2) requires

B. (1) should have
(2) does **NOT** require

C. (1) should **NOT** have
(2) requires

D. (1) should **NOT** have
(2) does **NOT** require

Answer: C

Explanation/Justification:

A. Incorrect. 2 of 3 NR S/G water levels < 19.6% on any S/G will automatically function to trip the reactor. Since these setpoints have currently NOT been reached, the reactor should NOT have automatically tripped. AOP-1.24.1 does require an immediate action to trip the reactor if one MFP is lost > 70% power. The candidate may be unaware of actual setpoints and/or logics

B. Incorrect. The reactor should NOT have automatically tripped. Reactor trip is required IAW AOP 1.24.1. The candidate may not be aware of changes to AOP 1.24.1 which now require a reactor trip above 75% power. Previous guidance was less conservative and allowed a power reduction to < 55% within the capacity of one MFP.

C. Correct. Auto reactor trip should NOT have occurred. AOP-1.24.1 requires an immediate action to trip the reactor if one MFP is lost > 70% power.

D. Incorrect. Auto reactor trip should NOT have occurred. Reactor trip is required IAW AOP 1.24.1.

Sys #	System	Category	KA Statement
054	Loss of Main Feedwater	N/A	Ability to interpret control room indications to verify status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

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

References provided to Candidate None **Technical References:** 1OM-1.4.ABT, Rev. 10, pg. 2
1OM-1.4.ABX, Issue 3, Rev. 6, pg. 1
1OM-53C.4.1.24.1, Rev. 7, pg. 2

ɘstion Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.5 / 43.5 / 45.12)

Objective: 1SQS-24.1 4. Describe the control, protection, interlock functions for the field components associated with Main Feedwater, Dedicated Auxiliary Feedwater System, Auxiliary Feedwater System or SGWLC System control, including automatic functions, setpoints, and changes in equipment status as applicable.

ECA - 0.0, "Loss of All Emergency 4KV AC Power" requires personnel be dispatched to locally isolate **RCP seal injection**.

What is the reason for this isolation?

A. To prevent back-flow through seal injection lines.

B. To prevent potential RCP seal damage when a charging pump is restarted.

C. To reduce the possibility of radioactive release to the Auxiliary Building.

D. To prevent steam formation in Component Cooling System due to overheating in the thermal barrier.

Answer: B

Explanation/Justification:

A. Incorrect. Backflow through the seal injection lines will not occur.

B. Correct. According to ECA-0.0 background document, isolating RCP seal injection lines prepare the plant for recovery while protecting the RCPs from seal and shaft damage that may occur when a charging pump is started as part of recovery. With RCP seal injection lines isolated, a charging pump can be started in the normal charging mode without concern for cold seal injection flow thermally shocking the RCPs.

C. Incorrect. This is the reason for isolating seal return versus injection.

Incorrect. This is the reason for isolating CCR return flow.

Sys #	System	Category	KA Statement
055	Station Blackout	Knowledge of the reasons for the following responses as they apply to the Station Blackout:	Actions contained in EOP for loss of offsite and onsite power.

K/A# EK3.02 **K/A Importance** 4.3 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-0.0, Issue 1C, Rev. 8, pg. 7
1OM-53B.4.ECA-0.0, Issue 1C, Rev. 8, pg. 85 & 86

Question Source: Bank - Vision # 17

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 3. State from memory the basis and sequence for the major actions steps of each EOP procedure, IAW BVPS EOP Executive Volume.

. Given the following plant conditions:

- A Loss of ALL AC Power occurred requiring the crew to enter ECA-0.0, "Loss Of All Emergency 4KV AC Power".
- An Emergency Diesel Generator (EDG) was returned to service in Step # 7 **PRIOR** to taking control switches to PULL-TO-LOCK in Step #12, and power was subsequently restored to ONE (1) 4KV Emergency Bus.

Which ONE of the following describes when the EDG will sequence loads onto the bus **AND** the reason for this sequencing?

The last load powered by the EDG completes sequencing ____ (1) ____ after initiating signal.
The reason for sequential loading of the EDG is to _____ (2) _____.

A. (1) 0 - 30 seconds
(2) prevent the emergency bus from becoming inoperable.

B. (1) 0 - 30 seconds
(2) prevent damage to reactor coolant pump seal package.

C. (1) 31- 60 seconds
(2) prevent the emergency bus from becoming inoperable.

D. (1) 31 - 60 seconds
(2) prevent damage to reactor coolant pump seal package.

Answer: C

Explanation/Justification:

A. Incorrect. Improper time. Correct reason.
B. Incorrect. Improper time. Improper reason. Reason is plausible since ECA-0.0 background focuses heavily on protection of RCP seal packages.
C. Correct. According to TS 3.8.1 bases and 1OST-36.3 (4) all EDG loads are sequenced onto the EDG within 60 seconds. The reason for this timing is to recover the unit or maintain it in a safe condition. T.S. 3.8.1 bases furthermore states the reason for EDG load sequencing is to protect the EDG from overload and that improper loading sequence may cause the emergency bus to become inoperable. (ie: EDG overload would result in a loss of the associated emergency bus). No SIS is present so therefore the charging pump output breaker did not shut at time 0.
D. Incorrect. Correct time. Incorrect reason. Reason is plausible since ECA-0.0 background focuses heavily on protection of RCP seal packages.

Sys #	**System**	**Category**	**KA Statement**
056`	Loss of Offsite Power	Knowledge of reasons for the following responses as they apply to a Loss of Offsite Power:	Order and time to initiation of power for the load sequencer.

K/A# AK3.01 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-0.0, Issue 1C, Rev. 8, pg.3, 4, & 6
1OST-36.3 (4), Rev. 28, pg. 73 & 75
TS 3.8.1 Bases, Rev. 0, pg. B3.8.1-1 & 2

Question Source: Bank – Vision # 82012 (2LOT7 NRC – Q#12)

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 3. State from memory the basis and sequence for the major action steps of each EOP procedure, IAW BVPS-EOP Executive Volume.

. The plant is operating at 60% power when the following alarms are received:

- [A9-100], 125VDC BATTERY CHGR 1 FAILURE
- [A9-98], 125VDC BUS 1 VOLTAGE LOW

Several minutes after the alarms are received:

- The plant continues to operate at 60% power.
- 125 VDC BUS 1 Voltage indicates approximately 124 VDC and is slowly DROPPING.
- Station Battery Charger Breaker [BAT-CHG1-1A] has been verified closed and 480V MCC1-E9 is energized.
- No operator actions have yet occurred.

For the given indications, which ONE of the following describes the 125VDC BUS 1 status?

A. Battery Charger 1-1A has failed. Station Battery #1 is supplying 125VDC Bus 1.

B. Station Battery #1 has failed. Battery Charger 1-1A is supplying 125VDC Bus 1.

C. Station Battery #1 and Battery Charger 1-1A have failed. 125 VDC Bus 1 is deenergized.

Station Battery #1 and Battery Charger 1-1A are operating normally. Station Battery is supplying 125 VDC Bus 1.

Answer: A

Explanation/Justification:

A. Correct. When a battery charger is lost, the station battery will automatically supply power to the loads on the effected bus. A9-100 and A9-98 are entry conditions for Loss of DC Power AOP.

B. Incorrect. If battery charger 1 were supplying the normal bus loads, 125VDC Bus 1 voltage would indicate between 127.8V and 135V.

C. Incorrect. Would have resulted in a loss of control power to EDG No. 1 and 4160 V Bus 1AE. Also the plant would no longer be operating at 60% power.

D. Incorrect. If Battery Charger 1 and Station Battery are operating normally, the alarms would not have been received. It is correct that the station battery is supplying the bus.

Sys #	System	Category	KA Statement
058	Loss of DC Power	N/A	Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

K/A# 2.4.47 **K/A Importance** 4.2 **Exam Level** RO

References provided to Candidate None

Technical References: 1OM-39.4.AAI, Rev. 6, pg. 2; 1OM-39.4.AAJ, Issue 3, Rev. 3, pg. 1; 1OM-39.4.A, Rev. 8, Pg. 2-5; 3SQS-39.1 Unit 1 PPT Slide

Question Source: Bank – Vision # 45879

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

Objective: 3SQS-39.1 20. Given a change in plant conditions due to system/component failure, analyze 125VDC distribution system to determine what failure has occurred.

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

- The Unit is operating at 20% power.
- "B" Main Feedwater Pump is running.
- No. 1 Emergency Diesel Generator (EDG) is on clearance.
- A normal plant shutdown is in progress due to inoperability of No. 1 EDG.
- "B" RPRW pump was just taken to Pull-to-Lock due to excessive seal leakage reported in the intake structure cubicle.
- An "A" 4KV Bus electrical fault occurred resulting in the following breakers tripping OPEN:
  - Main Generator Exciter Breaker
  - Both Main Generator Output Breakers
  - 4KV Bus "A" Feeder Breakers from USST and SSST.

Based on these plant conditions, which of the following procedures have **valid** required entry conditions within three (3) minutes after the Bus "A" fault occurred? (assume all systems function as designed)

1. E-0, "Reactor Trip or Safety Injection".
2. AOP 1.26.1, "Turbine and Generator Trip".
3. AOP 1.36.2, "Loss of 4KV Emergency Bus".
4. AOP 1.30.2, "River Water/Normal Intake Structure Loss".

A. 1 **ONLY.**

3 & 4 **ONLY.**

C. 1, 2, 3, & 4.

D. 2, 3, & 4 **ONLY.**

Answer: D

Explanation/Justification:

A. Incorrect. The candidate must recognize that a Loss of the "A" RCP (Bus 1A) will NOT result in a reactor trip since the reactor is < P-8 (30%). Also the stem of the question notes that "B" MFP is running to dispel a challenge that a loss of main feedwater occurred due to a loss of the "B" MFP. This is a plausible distractor and would be a correct choice if the reactor were > 30% power. A reactor trip is NOT required for these conditions since none has occurred and no valid entry conditions have been met.

B. Incorrect. Both choices are correct, however, AOP 1.26.1 will also have valid entry conditions (refer to correct answer). Plausible because a candidate may either not recognize a turbine trip occurred or may think it is not applicable since we are < P-9 (49% reactor power).

C. Incorrect. Incorrect that E-0 is required. (refer to correct answer and A distractor explanations for why other choices are correct).

D. Correct. A Loss of 1AE Bus will occur with No.1 EDG on clearance and a loss of offsite power source available to 4160 Bus A. With no power to the AE Bus entry conditions will be met for AOP 1.36.2. Since a turbine trip will occur due the nature of the electrical fault which opened both main generator breakers and exciter output breakers entry conditions for AOP 1.26.1 are met. Since the "B" RW pump was placed in PTL and a Loss of the AE Bus occurred, there is no power to the "A" Train River Water Pump which results in a Loss of River (Service) Water. Entry conditions for AOP 1.30.2 are met. Entry conditions to procedures are RO level knowledge. Furthermore the majority of the referenced procedures have immediate operator actions which are also required RO knowledge.

Sys #	System	Category	KA Statement
062	Loss of Nuclear Service Water	N/A	Ability to recognize abnormal indications for system operating parameters that are entry conditions for emergency and abnormal operating procedures.

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

References provided to Candidate None **Technical References:** 1OM-53C.4.1.36.2, Rev. 8, pg. 1
1OM-53C.4.1.26.1, Rev. 19, pg. 1
1OM-53C.4.1.30.2, Rev. 8, pg. 1

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.2 / 45.6)

Objective: 1SQS-53C.1 1. State all immediate operator actions associated with the AOPs.
3SQS-53.3 6. Given a set of conditions, locate and apply the proper EOPs IAW BVPS-EOP Executive Volume.

. Given the following plant conditions:

- The plant is operating at 95% power with all systems in NSA.
- Turbine Controls is selected to 1st Stage IN.
- Valve Position Limiter is set 5% above Governor Valve Position.
- DLC System Operations Control Center reports that disturbances have resulted in degraded grid frequency and voltage.
- The Control Room team has entered AOP ½.35.1, "Degraded Grid".

Given these conditions, which ONE of the following describes the relationship between degraded grid frequency/voltage and reactor power?

As grid frequency/voltage continues to drop, reactor power will ____ (1) ____.
An automatic reactor trip will occur if 2/3 Reactor Coolant Pump (RCP) 4KV Bus frequencies drop to ____ (2) ____.

A. (1) increase
(2) 57.5 Hz

B. (1) increase
(2) 58.5 Hz

C. (1) be unaffected
(2) 57.5 Hz

D. (1) be unaffected
(2) 58.5 Hz

Answer: A

Explanation/Justification:

A. Correct. Step 6 of AOP AOP ½.35.1requires reactor power maintained at </= 100% and directs a power reduction is power > 100%. As grid frequency/voltage drops, the power requirement increases which will result in increasing reactor power. An automatic reactor trip will occur if 2/3 RCP 4KV Bus frequencies drop to 57.5 Hz. With the Main Turbine in 1st Stage IN, as grid frequency drops, the increased steam demand will result in a drop in 1st stage pressure which will open governor valves to compensate for the load increase. This in turn will drop RCS Temperature adding positive reactivity which will increase reactor power.
B. Incorrect. Incorrect automatic reactor trip value. Correct reactor power effect on decreasing frequency/voltage.
C. Incorrect. Incorrect that reactor power will be unaffected. Correct automatic reactor trip value.
D. Incorrect. Incorrect that reactor power will be unaffected. Incorrect automatic reactor trip value.

Sys #	System	Category	KA Statement
077	Generator Voltage and Electric Grid Disturbances	Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following:	Reactor power

K/A# AK2.06 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1/2OM-53C.4A.35.1, Rev. 7, pg. 1-4
GOGPF.C5 PPNT Slides, Rev 1, Issue 2

estion Source: New

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

Objective: 1SQS-53C.1 5. Discuss the general flow path of each procedure including the importance of step sequencing, where applicable.

. Given the following plant conditions:

- A Loss of Coolant Accident (LOCA) outside containment occurred.
- The crew is executing procedure steps of ECA-1.2, "LOCA Outside Containment".

What system **AND** parameter will be used in ECA-1.2 to interpret whether break isolation has occurred?

	SYSTEM	**PARAMETER**
A.	Safety Injection	RCS Pressure
B.	Chemical and Volume Control	RCS Pressure
C.	Safety Injection	Spent Fuel Pool Area Radiation Level
D.	Chemical and Volume Control	Spent Fuel Pool Area Radiation Level

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Answer: A

Explanation/Justification:

- Correct. ECA-1.2 checks only the Low Head Safety Injection flowpath for proper valve alignment and also to determine if the source has been isolated. RCS Pressure is used as the determining parameter to ensure the break is isolated. The operating behavior characteristics of the facility is that a potential exists for RWST inventory to be lost to the Aux Building for a LOCA that occurs outside containment in the LHSI system piping.

B. Incorrect. CVCS is a plausible system since it interconnects with the RCS and extends outside containment. RCS pressure is the correct parameter.

C. Incorrect. Correct system. ECA-1.2 does check Aux Bldg and Safeguards Area radiation monitors which makes radiation levels a plausible distractor. Spent Fuel area radiation level is monitored independently of PAB and Safeguards radiation monitors.

D. Incorrect. Incorrect system and parameter. Distractor provides a good balance between other distractors and correct answer.

Sys #	**System**	**Category**	**KA Statement**
W/E04	LOCA Outside Containment	Ability to operate and /or monitor the following as they apply to the (LOCA Outside Containment)	Operating behavior characteristics of the facility.

K/A# EA1.2 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-1.2, Issue 1C, Rev. 1, pg. 3
1OM-53B.4.ECA-1.2, Issue 1C, Rev. 1, pg. 5 & 6

Question Source: Bank – Vision # 82016 (2LOT7 NRC – Q#16)

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR41.7 / 45.5 / 45.6)

Objective: 3SQS-53.5 7. Apply the actions to isolate a loss of coolant outside of containment.

. Given the following plant conditions and sequence of events:

- The Unit is at 100% power with all systems NSA.
- Pressurizer Level AUTO/MAN Controller fails as is.
- The plant is then reduced to 50% power at 3%/min.
- Assume **NO** operator action is taken, related to the failure.

Which ONE of the following describes how Charging Flow **AND** Programmed PRZR Level will indicate at 50% as compared to 100% power?

[FI-1CH-122], Charging Flow will be ___ (1) ___.
[LR-RC459], **PROGRAM** PRZR Level will be ___ (2) ___.

A. (1) lower
(2) lower

B. (1) the same
(2) higher

C. (1) the same
(2) lower

D. (1) the same
(2) the same

Answer: C

Explanation/Justification:

A. Incorrect. These indications are indicative of LT-1RC-459 failed in the as is position.

B. Incorrect. These indications are indicative of plausible candidate misconceptions related to actual vs. programmed PRZR level and distractor balancing.

C. Correct. If the PRZR Level controller fails as is and a load reduction occurred, then the input to FCV-1CH-122 will not change. Therefore FI-1CH-122 indicated flow will remain the same. Since a power reduction results in a Tavg decrease, then Program Level as indicated on LR-RC459 will drop to correlate with the PRZR Level Program. Program level is impacted by power reduction not the failure of LC-459G.

D. Incorrect. Correct that charging flow will remain the same. If the candidate does not understand the implication of power reduction and associated Tavg decrease, it is plausible that actual versus program PRZR would remain the same since charging flow has not changed.

Sys #	**System**	**Category**	**KA Statement**
028	Pressurizer Level Malfunction	Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following:	Sensors and detectors.

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

References provided to Candidate None **Technical References:** 1OM-6.4.IF, Rev. 11, pg. 12
SQS-6.4 PPNT, Rev. 11, Pg 58

Question Source: New

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

Objective:

1SQS-6.4 19. Given a specific plant condition, predict the response of the pressurizer and pressure relief system control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off normal condition: (Process Instrument Failure).

1SQS-7.1 19. Given a specific plant condition, predict the response of the Chemical and Volume Control 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: (Process Instrument Failure)

. Given the following plant conditions:

- The Unit is operating at 70% power when air leakage into the condenser resulted in a rising condenser backpressure.
- The Control Room Team is performing actions of AOP-1.26.2, "Loss of Condenser Vacuum".
- Concurrently, a load reduction is initiated at a rate of 5%/min in accordance with AOP-1.51.1, "Unplanned Power Reduction".
- Five minutes after the load reduction was commenced, condenser backpressure has risen to 5.5 IN HG-ABS.
- Ten minutes after the load reduction was commenced, condenser backpressure has risen to 6 IN HG-ABS and continues to slowly rise.

Which ONE of the following will be the **REQUIRED** action according to AOP-1.26.2, "Loss of Condenser Vacuum"? **(Reference Provided)**

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

B. Trip the reactor if condenser vacuum approaches C-9 setpoint and go to E-0, "Reactor Trip or Safety Injection".

Immediately trip the turbine and go to AOP 1.26.1, "Turbine and Generator Trip".

D. Trip the turbine if condenser vacuum approaches C-9 setpoint and go to AOP 1.26.1, "Turbine and Generator Trip".

Answer: C

Explanation/Justification:

A. Incorrect. This is a correct action if condenser vacuum remains >5.5 IN HG-ABS for more than five minutes and reactor power is > P-9, however, reactor power is < P-9 (49%). After five minutes of load reduction reactor power is 45%.

B. Incorrect. This is a correct action referenced in AOP 1.26.2 if reactor power is > 10%, however, C-9 (10 IN HG-ABS) is not being approached at 6 in hg-abs and other trip criteria have been exceeded.

C. Correct. According to AOP-1.26.2, a turbine trip and entry into AOP 1.26.1 is required if reactor power is < P-9 (49%) and condenser backpressure is > 5.5 IN HG-ABS for > 5 minutes and cannot be restored. Initial reactor power was 70%. After 5 minutes of load reduction at 5%/min, reactor power was lowered 45% which is < P-9 and > 270 MWE (~439 MWE). The reference is provided to comply with the K/A to execute procedures steps and to ensure the question is at the RO level of knowledge. This is not a direct lookup question because it requires application of reactor power and MWE calculations to derive the correct answer. Also it requires the candidate to apply continuous action steps.

D. Incorrect. This is an incorrect action, however, balances out the distracter plausibility.

Sys #	**System**	**Category**	**KA Statement**
051	Loss of Condenser Vacuum	N/A	Ability to interpret and execute procedure steps.

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

References provided to Candidate 1OM-53C.4.1.26.2 (Pg 1-6 ONLY) **Technical References:** 1OM-53C.4.1.26.2, Rev. 3, pg. 2
1OM-26.4.AAC, Rev. 8, pg. 2 & 3

Question Source: Modified Bank (1LOT5 NRC Q#50)

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

ȷective: 3SQS-53.5 27. Apply the actions for Loss of Condenser Vacuum.
1SQS-53C.1 6. Given a set of conditions, apply the correct AOP.

21. Given the following plant conditions and sequence of events occur:

- The Unit is operating at 100% Power with all systems in NSA.
- Chemistry reports High RCS Activity to the Control Room.
- The Control Room Team enters AOP 1.6.6, "High Reactor Coolant System Activity".
- A LOCA occurs and E-0, "Reactor Trip or Safety Injection" actions are in progress.
- The following annunciators are received:
  - A4-71, "RADIATION MONITORING HIGH"
  - A4-72, "RADIATION MONITORING HIGH-HIGH"
- The BOP is requested to determine if Adverse Containment radiation conditions exist.

Which ONE of the following Radiation Monitor indications will be used by the BOP to determine Adverse Containment Radiation conditions?

A. RIS-1RM-201, "Reactor Containment High Range Area Monitor".

B. RIS-1RM-204, "Incore Instrument Transfer Device Area Monitor".

C. RIS-1CH-101A, "Reactor Coolant Letdown High Range Monitor".

RIS-1RM-219A, "Containment High Range Area Monitor".

Answer: D

Explanation/Justification:

A. Incorrect. RM-1RM-201 functions to provide local and control room indication of reactor containment area activity. RM-1RM-201 is an Ion Chamber detector and is more accurate at the mid to high radiation levels. The log scale for this meter is .1 to 1E7 mr/hr. This falls short of the 1E5 r/hr dose necessary to detect adverse containment conditions due to the limitations of its detector. Plausible because it is used in AOP 1.6.6.

B. Incorrect. RM-1RM-204 functions to provide local and control room indication of increased radiation levels during accident conditions, RM-1RM-204 is a Geiger-Mueller detector which functions to detect gamma radiation. The log scale for this meter is .1 to 1E4 mr/hr. This falls short of the 1E5 r/hr dose necessary to detect adverse containment conditions due to the limitations of its detector. Plausible because it is used in AOP 1.6.6.

C. Incorrect. RM-1CH-101A functions to provide local and control room indication of RCS letdown line high activity. RM-1CH-101A is a gamma scintillation detector. The meter indications range from 10 to1E6 CPM. This monitor although plausible because it is used in AOP 1.6.6 is limited to indicating CPM versus containment radiation so therefore will not be used to determine adverse containment radiation levels.

D. Correct. RM-1RM-219A functions to provide area monitoring of the containment for accident monitoring and provides alarms and indications to the control room. The range of its ion chamber detector is 1 to 1E7 R/hr which makes it ideal for monitoring containment radiation conditions.

Sys #	**System**	**Category**	**KA Statement**
061	ARM System Alarms	Knowledge of the operational implications of the following concepts as they apply to Area Radiation Monitoring (ARM) System Alarms:	Detector limitations

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

References provided to Candidate None **Technical References:** 1OM-53C.4.1.6.6, Rev. 4, pg. 1-2
1OM-43.1E, Rev. 6, pg. 4, 10, 11, & 20
1SQS-43.1, Rev. 13, pg 4, 6, 7, 15-17
1OM-43.4.AAB, Issue 4, Rev. 1, Pg 1 & 2
1OM-43.4.AAC, Issue 4, Rev. 1, Pg 1 & 2

Question Source: New

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

'ective: 1SQS-43.1 1. Describe the function of the Radiation Monitoring systems and the associated major components as documented in Chapter 43 of the Unit 1 Operating Manual.

. Given the following plant conditions:

- The Unit is operating at 100% Power with all systems in NSA.
- Annunciator A11-69, "EAST CABLE VAULT FIRE" alarms.
- A serious fire in the East Cable Vault is confirmed.
- Assume all automatic fire suppression systems function as designed.

Based on these plant conditions, which ONE of the following describes the impact on Fire Brigade personnel?

The major concern to Fire Brigade personnel entering the East Cable Vault is ____ (1) ____ due to ____ (2) ____ used to automatically extinguish the fire in this area.

A. (1) asphyxiation from displacement of oxygen
(2) Water and CO2

B. (1) flooding and subsequent electrocution
(2) CO2 and Water

C. (1) asphyxiation from displacement of oxygen
(2) CO2 **ONLY**

D. (1) flooding and subsequent electrocution
(2) Water **ONLY**

Answer: C

Explanation/Justification:

A. Incorrect. Correct that asphyxiation is a major concern. Incorrect that water is used in this space (refer to correct answer).

B. Incorrect. Water is not used in the East Cable Vault as part of any automatic suppression fire fighting systems and therefore flooding is not a major concern.

C. Correct. The candidate must know that CO2 is the fire fighting agent used in the East Cable Vault to automatically distinguish fires. Water is NOT used in this area. The operational implications of a serious fire in the East Cable Vault is that CO2 is a major concern when entering this space due to the safety hazards it may cause (ie: cardiac arrest or nervous system effects).

D. Incorrect. Water is not used in the East Cable Vault as part of any automatic suppression fire fighting systems and therefore flooding is not a major concern. Plausible that water is used as a fire extinguishing agent and flooding would then become a concern.

Sys #	**System**	**Category**	**KA Statement**
067	Plant Fire On-site	Knowledge of the operational implications of the following concepts as they apply to Plant Fire on site:	Fire fighting

K/A# AK1.02 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-33.4.AAP, Rev. 2, pg. 2 & 3
1OM-56B.4.H, Rev. 20, pg 2

Question Source: New

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

Objective: 3SQS-33.1 4. Given a change in plant conditions, describe the response of the fire protection system field indication and control loops, including all automatic functions and changes in equipment status.
11. Given a fire protection system alarm condition and using the ARP, determine the appropriate alarm response, including automatic and operator actions in the control room.

. Given the following plant conditions:

- A major uncontrolled fire in the Control Room has resulted in control room evacuation.
- The Control Room Team is implementing 1OM-56C, "Alternate Safe Shutdown from Outside the Control Room".

What is the reason for maintaining Pressurizer Level during cooldown to Cold Shutdown according to 1OM-56C series procedures?

Pressurizer Level will be maintained 20% - 70% to ___________

A. maintain RCS inventory **ONLY.**

B. maintain RCS inventory **AND** shutdown margin.

C. ensure auto letdown isolation does **not** occur **ONLY.**

D. ensure auto letdown isolation does **not** occur **AND** to maintain shutdown margin.

Answer: B

·planation/Justification:

Incorrect. It is plausible and partially correct that PRZR level is maintained to maintain RCS inventory. S/D margin is also maintained by using a charging pump taking suction from the RWST during these plant conditions IAW 1OM-56C.4.B.

B. Correct. A charging pump is used for both maintaining RCS inventory as well as providing borated water from the RWST to help maintain adequate S/D margin. PRZR heaters are not related to maintaining PRZR level but rather RCS pressure control so therefore are not included as part of answering the K/A. According to 1OM-56C.4.B, PRZR Level is maintained 20% to 70%. This is a higher cognitive question because the intent and methodology of 1OM-56C.4.A/B must be known and is specific to evaluating this set of plant conditions.

C. Incorrect. Plausible that PRZR is maintained > 14% to ensure L/D isolation does not occur, however, the overall purpose of this procedure is perform safe shutdown without the letdown system. Letdown isolation is time critical and manually occurs prior to control room evacuation. Procedurally it is allowable to initiate head vent letdown if required which has no auto letdown isolation features.

D. Incorrect. Incorrect that PRZR level is maintained to ensure auto L/D isolation does not occur. Correct that PRZR level is maintained using a charging pump aligned to the RWST for S/D margin requirements.

Sys #	System	Category	KA Statement
068	Control Room Evacuation	Knowledge of the reasons for the following responses as they apply to the Control Room Evacuation:	Maintenance of PZR level, using charging pumps and heaters.

K/A# AK3.10 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-56C.4.A, Rev. 9, pg. A2-A5; 1OM-56C.4.B, Rev. 42, Pg. 3 & 10-14

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:
3SQS-53.5 13. Describe the actions for control room inaccessibility.
1SQS-56C.1 1. Describe the function of Alternate Safe Shutdown from Outside the Control Room and the associated major components as documented in Operating Manual Chapter 1OM-56C.

. Given the following plant conditions and sequence of events:

- The Plant was operating at 100% power with all systems in NSA.
- A Main Steam Line Break occurred inside containment.
- All systems functioned as designed.
- The Control Room Team is now performing actions of FR-P.1, "Response to Imminent Pressurized Thermal Shock Condition".
- They have determined an RCS Temperature Soak is required.

Which ONE of the following component/system actions is allowed to be performed during the soak period while FR-P.1 is being implemented?

A. Energize Pressurizer Heaters.

B. Place Auxiliary Spray in service.

C. Place RHR in service with flow through MOV-1RH-758.

D. Raise NR S/G water levels to 70% and secure AFW pumps.

Answer: B

Explanation/Justification:

A. Incorrect. Energizing PRZR heaters will raise RCS pressure which is not allowed during soak.

B. Correct. During RCS soak, FR-P.1 directs that cooldown in the RCS will not occur until temperature has been stable for 1 hour. RCS pressure is not to be raised during this period of time. Actions of other procedures may be performed provided that actions do not cooldown or raise pressure until the soak has been completed. Placing Aux Spray in service will lower RCS pressure which is allowable.

C. Incorrect. Placing RHR in service and allowing flow through MOV-1RH-758 will result in an RCS cooldown which is not allowed during soak.

D. Incorrect. Feeding S/Gs will result in RCS cooldown which is not allowed during soak.

Sys #	System	Category	KA Statement
W/E08	RCS Overcooling - PTS	Ability to operate and / or monitor the following as they apply to the (Pressurized Thermal Shock):	Components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

K/A# EA1.1 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-P.1, Issue 1C, Rev. 7, pg. 17; 1OM-53B.4.FR-P.1, Issue 1C, Rev. 7, pg. 50

Question Source: Bank – Vision # 12985

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

Objective: 3SQS-53.3 2. Describe from memory the overall purpose of each procedure, IAW BVPS-EOP Executive Volume.
3. State from memory the basis and sequence for the major actions steps of each EOP, IAW BVPS-EOP Executive Volume.

. Given the following plant conditions:

- The Control Room Team has just transitioned to ES-0.3, “Natural Circulation Cooldown With Steam Void in Vessel (With RVLIS)” due to increased cooldown rate.
- Pressurizer and Steamline Safety Injection (SI) signals are blocked.
- RCS temperature is 500 °F.
- Letdown is in service.
- Due to mis-operation of PRZR heaters, RCS pressure has risen to 2025 psig.

Which ONE of the following is the minimum required action(s) to restore PRZR pressure to 1800 psig?

Use ___ (1) ___ PRZR Spray to reduce PRZR pressure to ___ (2) ___.

A. (1) Normal
(2) 1800 psig

B. (1) Auxiliary
(2) 1800 psig

C. (1) Normal
(2) 1950 psig, block PRZR & SI signals, then reduce pressure to 1800 psig

D. (1) Auxiliary
(2) 1950 psig, block PRZR & SI signals, then reduce pressure to 1800 psig

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Answer: D

Explanation/Justification:

A. Incorrect. Normal spray is unavailable since RCPs are secured in ES-0.3. SI would initiate if PRZR pressure were reduced directly to 1800 psig.
B. Incorrect. Correct that auxiliary spray is used to depressurize. SI would initiate if PRZR pressure were reduced directly to 1800 psig.
C. Incorrect. Normal spray is unavailable since RCPs are secured in ES-0.3. The reblocking sequence is correct.
D. Correct. SI is blocked in ES-0.2 and a cooldown rate of < 25 F/hr is established in ES-0.2. Since the cooldown rate is exceeded, the potential for head void formation exists and transition to ES-0.3 is required. The caution from ES-0.2 still applies upon transition to ES-0.3. The SI system is designed to automatically unblock if PRZR pressure increases above 2000 psig (P-11). A subsequent drop in RCS pressure below the SI setpoint 1845 psig before manually reblocking the SI signal will result in an SI which is undesirable in ES-0.3.

Sys #	System	Category	KA Statement
W/E10	Natural Circulation with Steam Void in Vessel with/without RVLIS	Knowledge of the interrelations between the (Natural Circulation with Steam Void in Vessel with/without RVLIS) 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** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.ES-0.2, Issue 1C, Rev. 11, pg. 8 & 9
1OM-53B.4.ES-0.2, Issue 1C, Rev. 11, pg. 26
1OM-53A.1.ES-0.3, Issue 1C, Rev. 10, pg. 3
1OM-11.1.D, Issue 4, Rev. 1, Pg. 1

…stion Source: Bank – Vision # 8906

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

Objective: 3SQS-53.3 3. State from memory the basis and sequence of major action steps of each EOP, IAW BVPS-EOP Executive Volume.
4. Explain from memory the basis of all cautions and notes, IAW BVPS-EOP Executive Volume.

. Given the following plant conditions:

- Unit 1 has experienced a Loss of Coolant Accident (LOCA).
- Containment pressure initially peaked at 47 psig.
- While monitoring Critical Safety Functions, containment pressure is <u>currently</u> 15 psig.
- NO Quench Spray Pumps are operating.
- Containment Sump Level has been slowly rising and is <u>currently</u> 65 inches.

Based on these conditions, which ONE of the following identifies the status of the Containment Critical Safety Function (CSF) Status Tree?

A RED path condition **CURRENTLY** ____ (1) ____exist for ____ (2) ____.

A. (1) does
(2) containment pressure **<u>ONLY</u>**.

B. (1) does
(2) containment sump level **<u>ONLY</u>**.

C. (1) does
(2) containment pressure **<u>AND</u>** containment sump level.

D. (1) does **<u>NOT</u>**
(2) either containment pressure **<u>OR</u>** containment sump level.

<u>Answer: D</u>

Explanation/Justification:

A. Incorrect. An orange versus red path exist for containment pressure based on no quench spray pumps running and containment pressure > 11 psig. It is plausible that a candidate may select this choice based on exceeding 45 psig which is a red path condition, however, the CSFST asks only if pressure is < 45 psig.
B. Incorrect. Incorrect that a red path exists. Containment sump level would need to be > 81 inches and this would be an orange path condition.
C. Incorrect. No red path condition exists for either a containment high pressure or level condition. Plausible if candidate does not correctly determine plant conditions or know from memory CSFST Red paths which are required RO knowledge at BVPS.
D. Correct. According to Containment CSFST (F-0.5), no red path condition exists for either a containment high presšure or level condition.

Sys #	System	Category	KA Statement
W/E14	High Containment Pressure	Ability to determine and interpret the following as they apply to the (High Containment Pressure):	Facility conditions and selection of appropriate procedures during abnormal and emergency conditions.

K/A# EA2.1 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.F-0.5, Issue 1C, Rev. 2 pg. 1

Question Source: New

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

Objective:
3SQS-53.3 5. Explain from memory the basis for the decision blocks of each Status Tree, IAW BVPS-EOP Executive Volume.
3SQS-53.1 2. Concerning critical safety function restoration, IAW BVPS EOP Executive Volume, state from memory the following, The CFS in order of priority, The priorities of the color-coded end points of the CSFSTs, and the red path summary conditions from EOPs.

Given the following plant conditions:

- A Large Break LOCA occurred.
- An ORANGE path has developed on the Containment CSF Status Tree due to an abnormal rise in containment sump level.
- The Control Room Team transitions to FR-Z.2, "Response to Containment Flooding".

Which ONE of the following describes the source of the abnormally high containment sump level and desired outcome of this procedure?

The source of the abnormally high containment sump level is ___ (1) ___.
The desired operating outcome of this procedure is to ____ (2) ____.

A. (1) Safety Injection
(2) isolate the source of leakage.

B. (1) Safety Injection
(2) verify containment isolation and heat removal.

C. (1) River Water
(2) isolate the source of leakage.

D. (1) River Water
(2) verify containment isolation and heat removal.

Answer: C

Explanation/Justification:

A. Incorrect. High containment sump water level greater than the design flood level provides an indication that water volumes other than those represented by the emergency stored water sources have been introduced into the containment sump (ie: RWST and SI Accumulators) Correct desired outcome (refer to correct answer explanation)

B. Incorrect. Incorrect source. Incorrect desired operating outcome, although plausible since this is the correct outcome of FR-Z.1 versus FR-Z.2.

C. Correct. Fire protection or River Water penetrate containment and could provide large flow rates to components inside the containment and a major break or leak in one of those lines could introduce large quantities of water into the sump. The first action in FR-Z.2 is to try to identify the source of water which is causing containment flooding and isolate it.

D. Incorrect. Correct source. Incorrect desired outcome.

Sys #	System	Category	KA Statement
W/E15	Containment Flooding	Ability to operate and/or monitor the following as they apply to the (Containment Flooding):	Desired operating results during abnormal and emergency situations.

K/A# EA1.3 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53B.4.FR-Z.2, Issue 1C, Rev. 2, pg 1 -3

Question Source: Modified Bank – Vision # 16783

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7 / 45.4 / 45.6)

Objective: 3SQS-53.3 3. State from memory the basis and sequence for the major action steps of each EOP procedure, IAW BVPS EOP Executive Volume.

[illegible]. Given the following plant conditions and sequence of events:

- A Load reduction is in progress at 1% per minute.
- Reactor power is 22% and preparations are being made to take the turbine off-line.
- The Main Feed Regulating Bypass Valves have been transferred to AUTO.
- The 1B Reactor Coolant Pump (RCP) unexpectedly trips.
- No operator actions have been taken and the plant responds as designed.

Which ONE of the following will be the **INITIAL** effects of the RCP shutdown?

1B S/G Steam Flow will ____ (1) ____.
1B S/G Pressure will ____ (2) _____.

A. (1) decrease
(2) decrease

B. (1) increase
(2) decrease

C. (1) decrease
(2) increase

(1) increase
(2) remain the same

Answer: A

Explanation/Justification:

A. Correct. A loss of a single RCP below P-8 will NOT result in a reactor trip. The immediate effects of the tripped RCP in the effected loop is a decrease in steam flow (other two loops pick up flow). S/G pressure will drop since loop Tavg is lower.
B. Incorrect. Correct steam pressure response. Opposite Steam Flow Response. Steam flow will drop in effected loop but will increase in unaffected loops.
C. Incorrect. Correct that steam flow decreases. S/G pressure drops versus increases.
D. Incorrect. All parameter responses are incorrect. Plausible if the candidate does not understand RCP trip effects on these parameters.

Sys #	**System**	**Category**	**KA Statement**
003	Reactor Coolant Pump	Knowledge of the operational implications of the following concepts as they apply to the RCPS:	Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow.

K/A# K5.04 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** GO3ATA 3.2 U1 PPNT Abnormal Transients, Rev. 3

Question Source: New

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

Objective: 1SQS-6.3 19. Given a specific plant condition, predict the response of the RCP and system control room indications and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off normal condition.

. Given the following plant conditions:

- The Unit is operating at Full power with all systems in NSA.
- A Loss of Vital Bus I occurs.

Which ONE of the following identifies how a Loss of Vital Bus I will impact the specified CVCS components from the control room?

[FCV-1CH-122], "Chg Flow to Regen HX Inlet Control Valve" will ____ (1) ____.
The CVCS Blender will ____ (2) ____.

A. (1) CLOSE
(2) be affected

B. (1) CLOSE.
(2) **NOT** be affected

C. (1) remain OPEN
(2) be affected

D. (1) remain OPEN
(2) **NOT** be affected

Answer: C

Explanation/Justification:

A. Incorrect. FCV-1CH-122 is affected by a 125VDC Bus 1 loss not a Loss of Vital Bus I, so therefore will not close. Correct that the blender is affected.

B. Incorrect. FCV-1CH-122 is affected by a 125VDC Bus 1 loss not a Loss of Vital Bus I, so therefore will not close. Incorrect the blender is affected.

C. Correct. FCV-1CH-122 will not be impacted from the MCB, so therefore remains in its open position The Loss of Vital Bus I does impact the control of this valve from the SDP but not the control room. The Blender is rendered OOS by the loss of Vital Bus I.

D. Incorrect. Correct FCV-1CH-122 will remain open. The blender will be OOS and therefore is affected by this control power loss.

Sys #	**System**	**Category**	**KA Statement**
004	Chemical and Volume Control	Knowledge of bus power supplies to the following:	Control instrumentation.

K/A# K2.06 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53.4.1.38.1A, Rev. 3, pg. 12 & 13

Question Source: New

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

Objective: 1SQS-7.1 4. Identify the power supplies for the components identified on the NSA system flow path drawing which are powered from the class 1E electrical distribution system.

Given the following plant conditions:

- The Unit is in Mode 4, cooling down for refueling.
- Residual Heat Removal (RHR) Pump "A" and Heat Exchanger (HX) are in service.
- [MOV-1RH-605], RHR HX BYPASS FCV is in AUTO maintaining 4000 gpm.
- [MOV-1RH-758], RHR HX FCV is throttled from 50 % to 60% OPEN.
- No other operator adjustments are made and all systems function as designed.

Which ONE of the following describes the effect on RHR temperature and RHR flow (3) three minutes after [MOV-1RH-758] is throttled OPEN?

[TR-1RH-604], "RHR LOOP RETURN TEMP (Green Pen)" will be ____ (1) ____.
[FI-1RH-605], "RHR FLOW" will be ____ (2) ____.

A. (1) HIGHER
(2) LOWER

B. (1) LOWER
(2) HIGHER

C. (1) LOWER
(2) THE SAME

D. (1) HIGHER
(2) THE SAME

Answer: C

Explanation/Justification:

A. Incorrect. Incorrect opposite effect on RHR temperature. Plausible that less system flow would cause higher temperatures if the candidate has a misconception of system operation.
B. Incorrect. Correct effect on RHR temperature. Plausible that more system flow would cause lower temperatures if the candidate has a misconception of system operation.
C. Correct. If MOV-1RH-758 is opened, more flow will be directed through the RHR HX. MOV-1RH-605 in automatic will close to maintain set flowrate at 4000 gpm. The effect on RHR temperature is a lower temperature due to more flow through the HX and less flow through the bypass FCV. As a general practice BVPS does not use nitrogen any longer for maintaining a blanket in the PRZR and it is not used in the RHR system.
D. Incorrect. Correct RHR flow effect. Plausible that if the candidate has system misconceptions that they might believe that this adjustment will cause temperature is increase. This is opposite of the correct choice.

Sys #	System	Category	KA Statement
005	Residual Heat Removal	Ability to manually operate and/or monitor in the control room:	RHR Temperature, PZR heaters and flow, and nitrogen.

K/A# A4.03 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-10.1.C, Issue 4, Rev. 0, pg. 3
1OM-10.1.D, Rev. 1, pg. 2
OP Manual Fig. No. 10-1, Rev. 14

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-10.1 18. Given a specific plant condition, predict the response of RHR 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.

Beaver Valley Unit 1 NRC Written Exam (1LOT8)

. Given the following plant conditions and sequence of events:

- The Unit is operating at 100% power with all systems in NSA.
- Low Head Safety Injection Pump [1SI-P-1A] becomes inoperable due to a bearing failure.
- A Large Break LOCA occurs and all other systems function as designed.

Which ONE of the following describes how this failure impacts ECCS performance?

BEFORE transfer to cold leg recirculation there will be ~ __ (1) __ Low Head SI flow.
AFTER transfer to cold leg recirculation there will be Low Head SI flow available to __ (2) __ .

A. (1) 6000 gpm
(2) ONE High Head SI pump **ONLY**.

B. (1) 3000 gpm
(2) ONE High Head SI pump **ONLY**.

C. (1) 6000 gpm
(2) TWO High Head SI pumps.

(1) 3000 gpm
(2) TWO High Head SI pumps.

Answer: D

Explanation/Justification:

A. Incorrect. Incorrect that there is 6000 gpm flow. Plausible because this is the flow for two Low Head SI pumps. Incorrect that only one HHSI pump receives flow, although plausible.
B. Incorrect. Correct flow. Incorrect that only one HHSI pump receives flow, although plausible.
C. Incorrect. Incorrect flow before and after transfer to cold leg recirculation.
D. Correct. A loss of "A" LHSI pump will reduce the capacity of LHSI flow on a LBLOCA by half. Each pump is rated at 3000 gpm, so therefore with only the "B LHSI pump there is 3000 gpm flow. When transfer to cold leg recirculation occurs although there will also be half capacity, each LHSI pump provides flow to each HHSI pump so therefore there will be LHSI flow available to each HHSI pump. The candidate must have knowledge of the system flowpath and be able to understand the impact of a loss of one LHSI pump.

Sys #	System	Category	KA Statement
006	Emergency Core Cooling	Knowledge of the effect of a loss or malfunction on the following will have on the ECCS:	Pumps

K/A# K6.13 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-11.1.C, Rev. 2, pg. 5 & 6
1SQS-11.1 PPT Slide 23 & 27

Question Source: New

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

Objective: 1SQS-11.1 21. Given a change in plant conditions due to a system or component failure, analyze the SI system to determine what failure occurred.

. Given the following plant conditions:

- A reactor trip has occurred from Full Power.
- The crew has transitioned to ES-0.1, "Reactor Trip Response".
- RCS pressure is 1925 psig and slowly LOWERING.
- "A" Charging Pump [1CH-P-1A] is RUNNING.
- "B" Charging Pump [1CH-P-1B] is in STANDBY.
- Charging flow is offscale HIGH.
- Letdown is isolated.
- RCS temperature is 545 °F and STABLE.
- PRZR Level is 10% and LOWERING.

Which ONE of the following actions is procedurally required?

A. Initiate SI and continue in ES-0.1, "Reactor Trip Response".

B. Initiate SI and return to E-0, "Reactor Trip or Safety Injection".

C. Start SI pumps as required to maintain PRZR level and return to E-0.

D. Start SI pumps as required to maintain PRZR level and continue in ES-0.1.

Answer: B

Explanation/Justification:

A. Incorrect. Correct action but incorrect procedure. The candidate must understand that to be in ES-0.1 a prerequisite is that no SI has occurred.

B. Correct. PRZR level is lowering and with charging maximized and letdown isolated the candidate must deduce that PRZR level cannot maintained PRZR level > 4%. Procedurally this requires SI actuation and entry into E-0. The RO candidate must recognize these conditions are abnormal and EOP E-0 entry is required.

C. Incorrect. Would only start HHSI pumps as needed if the crew was in a reduction or SI termination sequence. Since it has not been initiated this action is inappropriate. Correct procedure transition to E-0.

D. Incorrect. Plausible that another charging pump (High Head SI Pump) is started to maintain PRZR level, however, not procedurally correct since letdown is isolated and charging flow is maximized.

Sys #	System	Category	KA Statement
006	Emergency Core Cooling	N/A	Ability to recognize abnormal indications for system operating parameters that are entry level conditions for emergency and abnormal operating procedures.

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

References provided to Candidate None **Technical References:** 1OM-53A.1.ES-0.1, Issue 1C, Rev. 8, pg. 4 (back)
1OM-53B.4.ES-0.1, Issue 1C, Rev. 8, pg. 3

Question Source: Bank – Vision # 45646

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.2 / 45.6)

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

. Given the following plant conditions:

- The Control Room team is establishing a steam bubble in the pressurizer (PRZR) in accordance with 1OM-50.4.L, "Plant Heatup from Mode 6 to Mode 3".
- The plant is in Mode 5 with RCS temperature at 130 °F.

Which ONE of the following identifies the method of forming the PRZR bubble?

While establishing a PRZR bubble, after heating the PRZR to target temperature, RCS pressure is lowered by relieving pressure via ___ (1) ___.
1OM-50.4.L requires a PRZR ___ (2) ___ will be maintained during and after the PRZR steam bubble is formed.

A. (1) the letdown system.
(2) outsurge

B. (1) the letdown system.
(2) insurge

C. (1) a power operated relief valve.
(2) outsurge

D. (1) a power operated relief valve.
(2) insurge

Answer: A

Explanation/Justification:

A. Correct. To form a PRZR bubble, the PRZR is heated to 250 – 275 F. Then PCV-1CH-145 is opened in order to drop RCS pressure. System pressure is relieved to the letdown system via PCV-1CH-145. A PRZR insurge briefing is conducted prior to drawing the PRZR bubble to identify this undesirable condition and therefore an outsurge is to be maintained during and after bubble formation.

B. Incorrect. Correct location for relieving system pressure. Incorrect method of forming PRZR bubble. (refer to correct answer explanation).

C. Incorrect. Incorrect location for relieving system discharge but plausible since the PRZR relieves to the Pressure Relief Tank via PORV's. The candidate if unfamiliar with the method of forming a PRZR bubble may believe that since the PRZR directly discharges to the PRT via PORV's that this is the correct flowpath. Correct method of forming PRZR bubble. (refer to correct answer explanation).

D. Incorrect. All aspects are incorrect, however, plausible and symmetrically balanced. Refer to previous discussions.

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

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

References provided to Candidate Steam Tables **Technical References:** 1OM-50.4.L, Rev. 26, pg. 26 & 27, 118 & 167
1SQS-6.4, Rev. 11, PPNT Slide

Question Source: New Lower – Memory or Fundamental

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.5 / 45.7)

Objective:

. Given the following plant conditions:

- Unit 1 is operating at full power with all systems in NSA.
- [1CC-P-1A], "A" CCR Pump is operating.
- An inadvertent Safety Injection (SI) occurs that results in a plant trip.
- All systems function as designed.

Which ONE of the following describes the status of the following CCR components?

[1CC-P-1A], "A" CCR Pump will be ____ (1) ___.
[TV-1CC-133-2], "Sample Clrs CCR Outlet Isol Vlv" will be____ (2) ____.

A. (1) TRIPPED
(2) CLOSED

B. (1) RUNNING
(2) CLOSED

C. (1) TRIPPED
(2) OPEN

(1) RUNNING
(2) OPEN

Answer: B

Explanation/Justification:

A. Incorrect. Correct that TV-1CC-133-2 is closed. Incorrect status of "A" CCR pump. This would be the condition of the pump if CIB were to occur.
B. Correct. "A" CCR pump will remain running on an SI signal. An inadvertent SI signal will result in a CIA signal. TV-1CC-133-2 will close on a CIA signal.
C. Incorrect. Incorrect status of both "A" CCR pump and TV-1CC-133-2. These are the exact opposite of the correct status.
D. Incorrect. Correct status of "A" CCR pump. Incorrect status of TV-1CC-133-2. Plausible if the candidate does not know SI signal auto actions.

Sys #	**System**	**Category**	**KA Statement**
008	Component Cooling Water	Ability to monitor automatic operation of the CCWS, including:	Automatic actions associated with the CCWS that occur as a result of a safety injection signal.

K/A# A3.08 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** UFSAR Fig. 7.2-1, Sh. 8, Rev. 13
1OM-1.5.B.4, Rev. 15, pg. 2 & 4
1OM-53A.1.1-B, Issue 1C, Rev. 1, pg 7
1SQS-15.1, Rev. 11 PPNT Slides

Question Source: New

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

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

Given the following plant conditions:

- The Unit is at 100% power with all systems in NSA.
- Reactor Coolant System (RCS) Pressure is 2235 psig and STABLE.
- RCS Temperature is 578°F and STABLE.
- PT-1RC-444, "Pressurizer Control Channel", fails **LOW** over a ONE (1) minute period.

With no operator action, which ONE of the following describes the effect on the PRZR pressure control system?

A. TWO PRZR PORVs will be OPEN.

B. ONE PRZR PORV and BOTH PRZR Spray Valves will be OPEN.

C. PRZR B/U heaters will be ON and BOTH PRZR Spray Valves will be CLOSED.

D. PRZR B/U heaters and BOTH PRZR Spray Valves will remain in their NSA positions.

Answer: C

ɿlanation/Justification:

Incorrect. These indications are indicative of PT-1RC-445 failing in the high direction.
B. Incorrect. These indications are indicative of PT-1RC-444 failing in the high direction.
C. Correct. The result of PT-1RC- 444 failing low is BOTH PRZR spray valves will be closed and PRZR heaters will be ON.
D. Incorrect. These indications are indicative of PT-1RC-445 failing in the low direction as opposed to PT-1RC-444.

Sys #	**System**	**Category**	**KA Statement**
010	Pressurizer Pressure Control System (PZR PCS)	Knowledge of the effect of a loss or malfunction of the following will have on PZR PCS:	Pressure detection systems.

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

References provided to Candidate None **Technical References:** 1OM-6.4.IF, Rev. 10, pg. 16-19 & 23

Question Source: Modified Bank – Vision # 82037 (2LOT7 NRC Q#37)

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

Objective: 1SQS-6.4 20. Given a change in plant conditions due to a system or component failure, analyze the Pressurizer and Pressurizer Relief System to determine what failure has occurred.

Given the following plant conditions:

- The plant was operating at full power when a Steam Generator Tube Rupture occurred.
- The Control Room team is implementing E-3, "Steam Generator Tube Rupture".
- The RCS has been cooled to 500 °F in preparation for equalizing RCS pressure with the ruptured S/G pressure.
- The US directs you to depressurize the RCS while maintaining a minimum of 20°F subcooling.

At the current RCS temperature, which ONE of the following is (1) the **LOWEST** RCS pressure can be lowered without violating the 20°F subcooling requirement **AND** (2) potential consequences of reaching 0°F subcooling?

A. (1) ~ 798 psig
(2) unreliable PRZR level indication **ONLY**.

B. (1) ~ 827 psig
(2) unreliable PRZR level indication **ONLY**.

C. (1) ~ 798 psig
(2) unreliable PRZR level indication **AND** delayed SI termination.

D. (1) ~ 827 psig
(2) unreliable PRZR level indication **AND** delayed SI termination.

Answer: C

Explanation/Justification:

A. Incorrect. Correct RCS pressure value (refer to correct answer explanation). Unreliable PRZR level indication is not the only potential operational implication iaw background document if RCS subcooling requirement is violated.
B. Incorrect. Plausible if candidate mistakenly adds 14.7 psi to the saturation pressure. Potential operational implication is not complete.
C. Correct. Saturation pressure for 520 °F is 812.53 psia minus 14.7 psi = 797.83 psig. The PZR PCS is used to lower RCS pressure (ie: spray valve and PORV is opened). The operational implication of not properly using steam tables is potential violation of procedure guidance which could result in unreliable PRZR level indication and delayed SI termination due to RCS voiding which could occur.
D. Incorrect. Incorrect value. Correct operational implications.

Sys #	System	Category	KA Statement
010	Pressurizer Pressure Control System (PZR PCS)	Knowledge of the operational implications of the following concepts as they 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
1OM-53A.1.E-3, Issue 1C, Rev. 14, pg 10, 16-17
1OM-53A.1.6-A, Issue 1C, Rev. 0, pg 1
1OM-53B.4.E-3, Issue 1C, Rev. 14, pg. 89 & 90

Question Source: New

estion Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.5 / 45.7)

jective: 3SQS-53.2 13. State from memory how significant RCS voiding may occur and how this can be mitigated, IAW BVPS EOP Executive Volume.

. Given the following plant conditions:

- The Unit is operating at 25% power.
- Rod Control is in Manual.
- I & C is performing 1MSP-6.13-I, "P-1RC456, Pressurizer Pressure Channel II Test".
- The channel has been removed from service and all applicable bistables have been placed in the tripped position.
- A malfunction of the Channel III OTΔT Bistable occurs causing it to trip.

Based on these plant conditions, which ONE of the following describes the effect on the Reactor Protection System (RPS)?

RPS Bistable Channel II (LOOP 2 O.T. ΔT RX TRIP) white status light will be ____ (1) ____ and a reactor trip ____ (2) ____ occur.

A. (1) ON
(2) will **NOT**

B. (1) OFF
(2) will

(1) ON
(2) will

D. (1) OFF
(2) will **NOT**

Answer: C

Explanation/Justification:

A. Incorrect. Correct Channel II bi-stable lamps status. Incorrect reactor status. Refer to correct answer explanation.
B. Incorrect. Correct reactor trip status. Incorrect Channel II Bi-stable lamp status.
C. Correct. Bi-stable lights will be ON when Channel II bi-stables are tripped in accordance with 1MSP-6.13-I. When performing the COT on PT-1RC-456, this trips the OTΔT trip for Channel II. A subsequent failure of Channel III OTΔT will result in a 2/3 satisfied reactor trip logic and resultant reactor trip generated by RPS. These are design features related to the trip logic of RPS.
D. Incorrect. Incorrect bi-stable light status. Incorrect reactor status.

Sys #	System	Category	KA Statement
012	Reactor Protection	Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following:	Trip logic when one channel OOC or in test.

K/A# K4.01 **K/A Importance** 3.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-6.4.IF, Rev. 11, pg. 21 & 40
1MSP-6.13-I, Issue 4, Rev. 12, pg. 4, 6 & 7
UFSAR Figure 7.2-1, Sh. 5, Rev. B

Question Source: New

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

,ective: 3SQS-1.1 10. Given a specific plant condition, predict or describe the response of the RPS trip logics and ESF actuation signal control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant conditions or for an off-normal condition.

Which ONE of the following is a designed **DIRECT** automatic start signal to [1FW-P-2], Steam Driven Auxiliary Feedwater Pump?

A. Containment Isolation "A" (CIA).

B. Auto trip of the last running Main Feedwater Pump.

C. 1/3 S/G NR level detectors LO-LO on 2/3 Steam Generators.

D. 2/3 S/G NR level detectors LO-LO on 1/3 Steam Generators.

Answer: D

Explanation/Justification:

A. Incorrect. Both electric and steam driven AFW pumps start on an SI signal. SI will actuate CIA however, this is not a direct signal.
B. Incorrect. This is an auto start for the electric AFW pumps.
C. Incorrect. The electric AFW pumps start on 2/3 lo-lo NR levels from 2/3 S/Gs.
D. Correct. According to references 1FW-P-2 starts on a 2/3 lo-lo S/G level signals from a single S/G.

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

K1.07 **K/A Importance** 4.1 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-24.1.D, Rev. 5, pg. 3 & 4
UFSAR Figure 7.2-1, Sh. 7 & 14
1SQS-24.1, Rev. 17, PPNT Slides

Question Source: Bank - Vision # 10281

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-1.1 8. Describe the control, protection and interlock functions of the control room components associated with the RPS trip logics and ESFAS actuation signals, including automatic functions, setpoint and changes in equipment status.

Given the following plant conditions:

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

Which ONE of the following describes the **CURRENT** status of [1VS-F-1A/C] Containment Air Recirculation Fans?

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

Answer: B

Explanation/Justification:

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

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

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

D. Incorrect. Opposite of the correct fan status.

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

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

References provided to Candidate None **Technical References:** 1OM-44C.3.C, Rev. 5, pg. 4 - 6; 1SQS-44C.1 PPNT, Rev. 9, Issue 2 Slides; 1SQS-44C.1, Rev. 9, Pg. 18 -19

Question Source: New

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

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

40. Given the following plant conditions:

- A LOCA has occurred.
- The Control Room Team is performing Step 5 of E-0, "Reactor Trip or Safety Injection".
- RCS pressure is 300 psig and slowly DROPPING.
- Containment pressure is 16 psig.
- RWST level is 44 feet and slowly DROPPING.
- The RO reports NEITHER Quench Spray (QS) pump has started.
- All other systems function as designed.

Which ONE of the following describes the impact of **BOTH** Quench Spray pumps **NOT** running **AND** what action is required?

With no operator action containment pressure will be trending ___ (1) ___.
The required actions are to ___ (2) ___.

A. (1) upward because no containment Quench Spray or Recirculation pumps are running.
(2) obtain SRO approval and then start BOTH QS pumps.

B. (1) downward because Inside Recirculation Spray pumps are running **ONLY**.
(2) obtain SRO approval and then start BOTH QS pumps.

C. (1) upward because no containment Quench Spray or Recirculation pumps are running.
(2) start BOTH QS pumps and then inform SRO of the failures.

D. (1) downward because Inside **AND** Outside Recirculation Spray pumps are running.
(2) start BOTH QS pumps and then inform SRO of the failures.

Answer: C

Explanation/Justification:

A. Incorrect. Part 1 is correct. Part 2 is incorrect. (refer to correct answer explanation).

B. Incorrect. Part 1 incorrect, Plausible if the candidate does not know the auto start setpoint (recently changed at BVPS such that they used to start after a time delay shortly after CIB actuation). Part 2 is incorrect.

C. Correct. With no operator action, containment pressure will continue to rise due to energy from LOCA being added to the containment with no heat sink. Recirc Spray pumps will not start until RWST level drops to 27.5 feet. Management expectations (operator fundamentals) IAW NOP-OP-1002 are that if an ESF component did not auto start as designed a manual attempt to start the component will be made by the operator. This action does not require SRO approval prior to attempting a start of the QS pumps after immediate operator actions are completed at Step 4 E-0.

D. Incorrect. Part 1 incorrect, Plausible if the candidate does not know the auto start setpoints or start logics (recently changed at BVPS such that they used to start after a time delay shortly after CIB actuation). Part 2 is correct.

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

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

References provided to Candidate None **Technical References:** 1OM-53A.1.E-0, Issue 1C, Rev. 11, pg. 7,
NOP-OP-1002, Rev. 5, pg. 43
BVBP-OPS-0024, Rev. 3, Pg. 4
1OM-53A.1.1-E, Issue 1C, Rev. 3, pg. 2, 5 & 6
1OM-53A.1.1-K, Issue 1C, Rev. 4, pg. 5

Question Source: New

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

Objective: 3SQS-53.3 4. Explain from memory, the basis for all cautions and notes, IAW BVPS-EOP Executive Volume.
6. Given a set of conditions, locate and apply the proper EOP IAW BVPS-EOP Executive Volume.

Given the following plant conditions:

- The plant is in Mode 1 @ 100% power with all systems in NSA.

Which ONE of the following conditions or events (considered individually) will require Technical Specification action(s) to be performed within one hour or less?

A. One Quench Spray Pump is declared inoperable.

B. One Containment Pressure Transmitter fails to zero.

C. RWST borated water volume drops to 425,000 gallons.

D. BOTH Train "A" – Phase B (CIB) manual pushbuttons are declared inoperable.

Answer: C

Explanation/Justification:

A. Incorrect. TS 3.6.6 Condition A is a 72 hour action statement for one QS train inoperable.

B. Incorrect. TS 3.3.2 Condition D & E apply. This was a recently changed TS and requires a 72 action statement.

C. Correct. TS 3.5.4 Condition B states that if RWST is inoperable for reasons other than boron concentration or temperature (Condition A), than a 1 hour action statement is applicable. SR 3.5.4.2 requires Unit 1 RWST level to be greater than or equal to 430,500 gallons. If this surveillance is not met than TS LCO actions apply. RO's are required to know ≤ 1hour TS LCO's from memory.

D. Incorrect. TS 3.3.2 Condition B applies. This is a 48 hour action statement.

Sys #	System	Category		KA Statement
026	Containment Spray	N/A		Knowledge of less than or equal to one hour technical specifications for a system.
K/A# 2.2.39	**K/A Importance**	3.9	**Exam Level**	RO

References provided to Candidate None **Technical References:**
BVPS Units 1 & 2 TS 3.5.4, Amend 278/161, pg. 3.5.4-1 & 2
BVPS Units 1 & 2 TS 3.6.6, Amend 278/161, pg. 3.6.6-1
BVPS Units 1 & 2 TS 3.3.2, Amend 282/166, pg. 3.3.2-1 & 10
BVPS Units 1 & 2 TS 3.3.2 Amend 282/166 pg. 3.3.2- 2 & 8-11

Question Source: Modified Bank – Vision # 258

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.7 / 41.10 / 43.2 / 45.13)

Objective: 1SQS-31.1 24. For a given set of plant conditions, from memory determine if the condition meets the criteria for entry into a one hour or less action statement in accordance with technical specifications.

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. Given the following plant conditions:

- The Plant is in Mode 3.
- Main Condenser Steam Dumps are being used to cooldown to Mode 5.
- RCS temperature is 450 °F and slowly DROPPING.
- RCS pressure is 670 psig and STABLE.
- All Systems are in normal alignment for these conditions.

For these conditions, which ONE of the following will cause an automatic Main Steam Line Isolation (MSLI)?

A. Low Steam Line Pressure **ONLY**.

B. High Steam Line Pressure Rate **ONLY**.

C. Low Steam Line Pressure **AND** Containment Hi-Hi Pressure (8 psig).

D. High Steam Line Pressure Rate **AND** Containment Hi-Hi Pressure (8 psig).

Answer: D

_.planation/Justification:

A. Incorrect. This signal is blocked below P-11 and is no longer action based on stated plant conditions.

B. Incorrect. Partially correct. Hi-Hi containment pressure signal is always active.

C. Incorrect. Low steam line pressure is incorrect. Hi-Hi Containment pressure is correct.

D. Correct. A MSLI isolates the reactor building from the containment by closing the MSIV's and other isolation valves. MSLI signals are active as follows: Below P-11 (< 2000 psig) Low Steam Line Pressure is blocked and High Steam Line Pressure Rate is manually inserted. The Hi-Hi containment pressure signal is always active. At BVPS we do not use the terminology reactor building isolation, however, we do use terms such as Containment Isolation, Main Steam Line Isolation. To hit the K/A for BVPS the question focuses on the Main Steam Isolation signal which isolates Main Steam from inside our containment building to outside areas such as the Turbine Building (Main Steam/Aux Steam) and Primary Auxiliary Building (Auxiliary Steam only which is no longer used in the PAB).

Sys #	System	Category	KA Statement
039	Main and Reheat Steam	Knowledge of MRSS design features(s) and/or interlock(s) which provide for the following:	Reactor Building Isolation

K/A# K4.07 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** BVPS UFSAR Figure 7.2-1 SH 6-8
1SQS-21.1 PPNT, Rev. 15

Question Source: Bank – Vision # 81815 (1LOT7 NRC Q#15)

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

Objective: 1SQS-21.1 12. Given a specific plant condition, predict the response of the Main Steam Supply System control room indications and control loops, including all automatic functions and changes in equipment status, for either a change in plant conditions or for an off normal condition.

. Given the following plant conditions and sequence of events:

- The plant is operating at 100% power.
- [1FW-P-3A], Motor Driven Auxiliary Feedwater Pump is OOS.
- A Loss of Offsite power coincident with a turbine trip occurs.
- Bus 1DF has an overcurrent lockout.
- All systems function as designed.

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

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

A. (1) 350
(2) ALL

B. (1) 350
(2) ONLY TWO

C. (1) 700
(2) ALL

D. (1) 700
(2) ONLY TWO

Answer: C

Explanation/Justification:

A. Incorrect. Incorrect capacity. Correct number of S/Gs. Plausible if the candidate does not know the capacities or misunderstands the initial plant conditions.

B. Incorrect. Incorrect capacity. Incorrect number of S/Gs. Plausible if the candidate believes each AFW pump feeds a single S/G or believes the DF Bus loss will impact the AFW throttle valves which are DC powered.

C. Correct. A loss of offsite power coincident with a turbine trip results in a reactor trip and subsequent loss of both MFW pumps. The EDGs are designed to start on a loss of power to AE and DF bus which will power both electric AFW pumps. In the stated conditions, with an overcurrent condition on the DF bus, 1FW-P-3B will not have power. Since 1FW-P-3A is already OOS, only 1FW-P-2 (Turbine Driven AFW pump) will start to provide 700 gpm AFW flow. The AFW system is designed to feed all three S/G based on NSA alignment requirements.

D. Incorrect. Correct capacity. Incorrect number of S/Gs.

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

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

References provided to Candidate None **Technical References:** 1OM-24.1.B, Rev.2, pg. 2
1SQS-24.1, Rev. 17 PPNT slide.

Question Source: New

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

jective: 1SQS-24.1 12. List the nominal values of the control room operating parameters associated with the Main Feedwater, Dedicated Auxiliary Feedwater, Auxiliary Feedwater System and Steam Generator Water Level Control System.

. Given the following plant conditions:

- The Unit is operating at 100% power.
- 1OST-24.2, "Motor Driven Auxiliary Pump Test [1FW-P-3A]" is being performed.
- [FCV-1FW-103A], "A" Motor Driven Auxiliary Pump Recirculation Valve" remote position verification is to be performed as part of this OST.
- [1FW-37], "1FW-P-3A Discharge to "A" Header" Valve" has been isolated.

Which ONE of the following describes [FCV-1FW-103A] position shortly after placing the control switch for [1FW-P-3A] to START?

[FCV-1FW-103A] will be ____ (1) ____ because ____ (2) ____

A. (1) OPEN
(2) discharge flow is below setpoint.

B. (1) CLOSED
(2) discharge flow is above setpoint

C. (1) OPEN
(2) suction flow is below setpoint.

D. (1) CLOSED
(2) suction flow is above setpoint

Answer: C

Explanation/Justification:

A. Incorrect. Correct position. Incorrect system design. (refer to correct answer explanation)

B. Incorrect. Incorrect position. Incorrect system design. (refer to correct answer explanation)

C. Correct. FCV-1FW-103A is designed to open upon AFW pump start to ensure sufficient recirc flow for pump cooling. The recirc valve senses suction flow (<145 gpm) and opens to ensure > 145 gpm is flowing through the pump. In NSA the discharge valve is not shut, The recirc valve will open upon pump start and upon sensing sufficient suction flow will close. In this scenario the OST checks recirc valve operation by positioning FW-37 shut. Therefore there is no flowpath directly to the S/Gs and the recirc valve will open to recirc water back to 1WT-TK-10, ensuring > 145 recirc flow for pump cooling. All distractors are plausible if the candidate does not have knowledge of system design and plant configuration changes. This is higher cognitive because the candidate must analyze the modified NSA plant conditions and apply these condiitons to the design of the AFW system recirculation valves.

D. Incorrect. Incorrect position. Correct system design. (refer to correct answer explanation)

Sys #	**System**	**Category**	**KA Statement**
061	Auxiliary/Emergency Feedwater	Knowledge of AFW design features(s) and/or interlock(s) which provide for the following:	AFW Recirculation.

K/A# K4.08 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-24.1.C, Rev. 5, pg. 9,
1OST-24.2, Rev. 42, pg. 17
OP Manual Fig. 24-2, Rev. 13
1SQS-24.1, Rev. 17 PPNT Slides
1OM-24.4.AAD, Rev. 5, Pg. 2

ɘstion Source: New

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

Objective: 1SQS-24.1 4. Describe the control, protection and interlock functions for the field components associated with the AFW system, including automatic functions, setpoints, and changes in equipment status as applicable.

Given the following plant conditions:

- The Unit is operating at 100% power.
- 1OST-36.1, "Diesel Generator No. 1 Monthly Test" is in progress.
- Emergency Diesel Generator (EDG) No. 1 is paralleled to the grid, carrying about 50% load.
- A grid disturbance causes frequency to drop very slightly.
- Grid Voltage remains constant.

Which ONE of the following describes the response of EDG No. 1 **AND** what is the significance of operating the EDG above 2850 KW for extended periods of time?

The response of EDG No. 1 is that ___ (1) ___ **AND** the potential consequence of operating this EDG > 2850 KW is excessive ____(2)___.

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

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

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

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

---

Answer: A

Explanation/Justification:

A. Correct. If frequency drops, the EDG will attempt to increase speed, which will pick up real load. TS Surveillance 3.8.1.3 bases states that the load band (2340 TO 2600 KW) which is more restrictive than the rated load in 1OST-36.1 (2850 KW) is to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations for DG OPERABILITY.

B. Incorrect. Correct EDG response. Incorrect consequence. The reason for significance of EDG loading is for ensuring loading is maintained >50% for an hour when operating the EDG at low loads for extended periods of time. This limit is plausible in that it is more associated with operating the EDG at low loads and could be confused by the candidate.

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

D. Incorrect. Incorrect EDG response. Reason for load limit is incorrect as explained above.

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

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

References provided to Candidate None

Technical References: GP Electrical Theory, Rev. 2, pg. 114-135
1OST-36.1, Rev. 53, PG. 5 & 6
TS 3.8.1 Amend. 278/161, Pg. 5
TS 3.8.1, Rev. 13, Pg. 17

...stion Source: Modified Bank -Vision # 45778

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

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

. Given the following plant conditions:

- A Loss of ALL AC Power occurred.
- The Control Room Team is performing actions of ECA-0.0, “Loss of All Emergency AC Power”.
- Hydrogen has been vented from the Main Unit and [1LO-M-14], “Air Side Seal Oil Backup Pump”, has been secured.

Which ONE of the following describes the design capacity of the Class 1E batteries, and the effect of stopping [1LO-M-14] while performing ECA-0.0?

The Class 1E battery design capacity is ____ (1) ____.
The effect of stopping [1LO-M-14] while performing ECA-0.0 is ____ (2) ____.

A. (1) 2 hours
(2) a reduction in the battery discharge rate.

B. (1) 2 hours
(2) an extension of battery life up to 6 hours

C. (1) 4 hours
(2) a reduction in the battery discharge rate.

D. (1) 4 hours
(2) an extension of battery life up to 6 hours

Answer: A

Explanation/Justification:

A. Correct. The Class 1E station batteries are designed for two hour operation. The reason for stopping 1LO-M-14 is ECA-0.0 is to help reduce the DC loading on the station batteries. Less load on the battery equates to a lowered discharge rate.
B. Incorrect. Correct time. Incorrect plausible effect.
C. Incorrect. Incorrect time. This is a plausible time since it is the design at other plants. Correct effect.
D. Incorrect. Incorrect time. Incorrect effect.

Sys #	System	Category	KA Statement
063	DC Electrical Distribution	Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including:	Battery capacity as it is effected by discharge rate.

K/A# A1.01 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-39.1.B, Rev.1, pg. 4
3SQS-39.1, Rev. 8, pg. 3
1OM-53B.4.ECA-0.0, Issue 1C, Rev.9, pg 115

Question Source: Modified Bank – (2011 Ginna SRO Retake #14)

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

Objective: 3SQS-39.1 10. Given a 125 VDC Distribution System configuration, and without reference materials describe the 125 VDC Distribution System field response to the following malfunctions, including automatic functions and changes in equipment status: Loss of AC Power, Loss of Station Battery, Loss of DC Power.
16. Describe the battery capacity as it is effected by discharge rate.

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Given the following plant conditions:

- The plant is operating at 100% power with all systems NSA.
- A Loss of 125VDC Bus 2 occurs.
- The Control Room Team has entered AOP 1.39.1B. "Loss of 125VDC Bus 2".
- All systems function as designed and no operator action has occurred.

According to AOP 1.39.1B, which ONE of the following describes the plant status?

	REACTOR TRIP	MAIN STEAMLINE ISOLATION
A.	WILL AUTO OCCUR	**CANNOT** AUTO OCCUR
B.	WILL AUTO OCCUR	CAN AUTO OCCUR
C.	WILL **NOT** AUTO OCCUR	CAN AUTO OCCUR
D.	WILL **NOT** AUTO OCCUR	**CANNOT** AUTO OCCUR

Answer: B

planation/Justification:

Incorrect. Incorrect that a MSLI cannot occur. Plausible because it is partially true that B train MSLI cannot occur since DC Bus 2 power is required however, A Train is still available for MSLI. (refer to correct answer explanation)

B. Correct. Main Feedwater Regulating valves fail closed on a Loss of 125VDC Bus 2, so therefore at 100% power it will not be long before a reactor trip automatically occurs. MSLI valves require DC power to close. Since only the B Train of power was lost, the A Train (DC Bus 1) is still available so therefore MSLI can still auto occur if valid plant conditions warrant.

C. Incorrect. Reactor trip will occur. Plausible if the candidate does not know the status of MFRVs. Correct that MSLI can occur. Also plausible if the candidate does not know the automatic actions of AOP 1.39.1B.

D. Incorrect. Reactor trip will occur. Plausible if the candidate does not know the status of MFRVs. MSLI can occur as previously explained. Also plausible if the candidate does not know the automatic actions of AOP 1.39.1B.

Sys #	System	Category	KA Statement
063	DC Electrical Distribution	N/A	Knowledge of abnormal condition procedures.

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

References provided to Candidate None **Technical References:** 1OM-53C.4.2.39.1B, Rev. 3, pg 1, 2, & 11

Question Source: New

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

Objective: 3SQS-39.1 18. Given a 125 VDC Distribution 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: Loss of DC Bus

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

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

Which ONE of the following describes the impact on EDG 1-1?

EDG 1-1 will _________ cooling water available.

A. trip with

B. trip without

C. continue to run with

D. continue to run without

Answer C

Explanation/Justification:

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

B. Incorrect. EDG 1-1 does not trip but remains running. Plausible that the EDG would trip on an overcurrent condition, however, protection in this scenario is provided by ACB 1E7. Also incorrect that cooling water is not available.

C. Correct. For the given conditions, EDG 1-1 is running paralleled to the grid. An overcurrent condition was caused by the closure of ACB1E7 and results in ACB 1A10 & 1E7 automatically opening. Upon ACB-1E7 opening, the overcurrent condition clears which is indicative of the problem being downstream of ACB 1E7. The EDG will continue to run with cooling. ACB E-9 is unaffected and even if it did open, EDG cooling would be maintained from the other train since RW is cross connected at the CCR HX's. It is not RO knowledge to select procedures so therefore only the first part of the higher cognitive K/A was tested.

D. Incorrect. Correct that EDG 1-1 remains running, incorrect that it is running without cooling. Plausible if the candidate believes the overcurrent trip opens ACB 1E-9 and does not recognize or understand the RW system configuration. (common misconception based on prior plant configuration).

Sys #	**System**	**Category**	**KA Statement**
064	Emergency Diesel Generator	Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Consequences of opening/closing breaker between buses (VARS, out of phase, voltage)

K/A# A2.08 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 3SQS-36.1, Rev. 8 PPNT Slide,
1OM-36.4.Q, Rev. 10, pg. 2 - 4
1OM-36.4.ADB, Rev. 2, pg. 2
1OM-36.4.ADA, Issue 3, Rev. 0, Pg. 1
1OM-36.1.E, Rev. 2, pg. 29 – 31
1OM-30.2, Rev. 16 PPNT Slide

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: 1SQSQ-36.2 13. Given an EDG configuration and without referenced material, describe the EDG control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable: SI or Bus UV.

. Given the following plant conditions:

- The Unit is operating at 100% power with all systems in NSA.
- 1RM-1SV-100, "Condenser Air Ejector Discharge" process detector fails downscale **LOW**.

Which ONE of the following describes how this failure will affect the release in progress?

The air ejector discharge will ________

A. automatically terminate immediately.

B. automatically divert to the containment.

C. automatically terminate after a 30 second time delay.

D. continue to present location unless manually realigned.

Answer: D

Explanation/Justification:

A. Incorrect. Plausible for most upscale detector failures. Opposite of actual effect.

B. Incorrect. Correct if 1RM-1SV-100 failed upscale. An upscale failure would result in SOV-1SV-100A and 100B to energize which causes the air ejector discharge to be diverted to the containment and secures the discharge to gaseous waste system. This design is unique to Unit 1 ONLY, Incorrect. There is no time delay however, plausible since some systems are designed with time delays.

D. Correct. A downscale failure will have no impact on the release in progress and would require a manual alignment for a high radiation condition. According to reference material an alarm would be generated to inform the operators of this condition.

Sys #	System	Category	KA Statement
073	Process Radiation Monitoring	Knowledge of the effect that a loss or malfunction of the PRM system will have on the following:	Radioactive effluent releases.

K/A# K3.01 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None

Technical References: 1OM-43.1.C, Issue 4, Rev. 3, Pg. 10
1OM-43.1.E, Rev. 6, Pg. 25
1OM-43.1.D, Rev. 5, Pg. 2 & 3

Question Source: New

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

Objective: 1SQS-43.1 7. Given a specific plant condition, predict the response of the radiation monitoring system control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off normal condition.

50. Given the following plant conditions:

- The plant is operating at 100% power with all systems in NSA.
- Reactor Plant River Water Pump [1WR-P-1A] is operating.
- Reactor Plant River Water Pump [1WR-P-1B] is in standby.

The following control room indications occur:

- A1-59, "INTAKE STRUCT RIVER WATER PP DISCH LINE A PRESS LOW" is received.
- [PI-1RW-113A], "CCR HEAT EX RIVER WATER PRESS "A" HEADER" lowest observed reading is 26 psig and continues to SLOWLY DROP.
- [PI-1RW-113B], "CCR HEAT EX RIVER WATER PRESS "B" HEADER" lowest observed reading is 38 psig and continues to SLOWLY DROP.
- Assume all systems function as designed and no operator action has yet been taken.

Which ONE of the following describes the **CURRENT** status of [1WR-P-1B] **AND** which Abnormal Operating Procedure (AOP) will be entered to mitigate the consequences of this event?

[1WR-P-1B] will ___ (1) __.
AOP ___ (2) ___ will be entered to mitigate this event.

A. (1) be running
(2) 1.51.1, "Unplanned Power Reduction"

B. (1) be running
(2) 1.30.2, "River Water/Normal Intake Structure Loss".

C. (1) **NOT** be running
(2) 1.51.1, "Unplanned Power Reduction"

D. (1) **NOT** be running
(2) 1.30.2, "River Water/Normal Intake Structure Loss".

Answer: D

Explanation/Justification:

A. Incorrect. Incorrect that 1WR-P-1B is running (refer to correct answer explanation) Incorrect procedure, although plausible that a plant S/D would occur. The ARP directs entry into AOP-1.30.2 which directs a plant trip if River Water and Aux River Water pumps cannot be started.

B. Incorrect. Incorrect that 1WR-P-1B is running (refer to correct answer explanation) Correct procedure.

C. Incorrect. Correct pump status. Incorrect procedure. Prior to attempting a plant shutdown an attempt to restore header pressure should procedurally occur and at the stated RW pressure it is more prudent to restore pressure versus plant S/D.

D. Correct. The standby RW Pump will auto start when PT-1RW-113A senses 20 psig, so therefore is not running based on stated plant conditions. The Annunciator received is related to intake structure low pressure which is indicative of a header leak or loss of the "A" pump. It is more likely based on pressure indications that a leak has occurred and requires entry into AOP 1.30.2 per ARP.

Sys	System	Category	KA Statement
076	Service Water	Ability to (a) predict the impacts of the following malfunctions or operations on the SWS system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of SWS

K/A# A2.01 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None

Technical References: 1OM-30.4.AAC, Rev. 3, pg. 2
1OM-430.4.AAA, Issue 3, Rev. 2, Pg. 1
1OM-30.1.D, Issue 4, Rev. 3, pg. 5,
1OM-30.1.E, Rev. 10, pg.5,

Question Source: New

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

Objective: 1SQS-30.2 5. Given a change in plant conditions, describe the response of the RPRW system field indication and control loops, including all automatic functions and changes in plant equipment.

Given the following plant conditions:

- The Plant was operating at 100% power with all systems in NSA when a Reactor Trip and Safety Injection occurred.
- "A" Steam Generator faulted inside containment causing Containment Pressure to peak at 25 psig and is currently DROPPING.
- All systems have functioned as designed and no operator actions have occurred.

Which ONE of the following describes the position of the River Water System components listed below?

[MOV-1RW-103A-D] = 1A/1B Header RPRW to Recirc Spray HX Isol Valves
[MOV-1RW-114A/B] = CCR HX RW Series Isol Valves

	[MOV-1RW-103A-D]	**[MOV-1RW-114A/B]**
A.	OPEN	OPEN
B.	OPEN	CLOSED
C.	CLOSED	OPEN
D.	CLOSED	CLOSED

Answer: B

Explanation/Justification:

A. Incorrect. Correct position for RSS HX. Incorrect position for CCR HX.
B. Correct. When containment pressure is > 11 psig a CIB signal is generated which isolates the cooling to CCR HX's and lines up cooling to emergency heat loads (RSS HX's).
C. Incorrect. This would be the position if CIB had not occurred.
D. Incorrect. Incorrect position for RSS HX. Correct position for CCR HX.

Sys #	**System**	**Category**	**KA Statement**
076	Service Water	Ability to manually operate and/or monitor in the control room:	Emergency heat loads.

K/A# A4.04 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-53A.1.1-K, Rev. 4, pg. 5
1OM-53A.1.1-E, Issue 1C, Rev. 3, pg. 4 & 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: 1SQS-30.2 15. Given a Reactor Plant River Water System configuration and without referenced material, describe the RPRW System control room response to the following off normal conditions, including automatic functions and changes in equipment status as applicable: Safety Injection, Containment Isolation Signal, Phase B (CIB, EDG start.

. Which ONE of the following describes how [TV-1SA-105], "Station Air Header Trip Valve" performs its system function?

[TV-1SA-105] closes to isolate ____ (1) ____ on lowering air pressure.
Upon restoration of system air pressure above setpoint, [TV-1SA-105] will ____ (2) ____.

A. (1) instrument air from station air
(2) automatically reopen.

B. (1) instrument air from containment air
(2) automatically reopen.

C. (1) instrument air from station air
(2) be closed and must be manually reopened.

D. (1) instrument air from containment air
(2) be closed and must be manually reopened.

Answer: C

Explanation/Justification:

Incorrect. Correct system interface. Incorrect that valve auto reopens (refer to correct answer explanation)

B. Incorrect. Incorrect system interface, TV-1IA-400 is the interface valve between containment and station instrument air. Incorrect that valve auto reopens (refer to correct answer explanation)

C. Correct. TV-1SA-105 will auto close on lowering instrument air pressure as sensed on PS-1SA-105 @ 95 psig. This valve functions to isolate station air from instrument air. The valve is designed to not auto reopen and must be manually reopened when station air pressure is restored to NOP.

D. Incorrect. Incorrect system interface. Correct valve cause-effect relationship.

Sys #	System	Category	KA Statement
078	Instrument Air	Knowledge of the physical connections and/or cause-effect relationships between IAS and the following systems:	Service Air.

K/A# K1.02 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-34.1.D, Issue 4, Rev. 0, pg. 5
1SQS-34.1, Rev. 14, PPNT Slide HO-1

Question Source: New

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

Objective: 1SQS-34.1 3. Describe the control, protection and interlock functions for the field components associated with the Compressed Air System, including automatic functions, setpoints and changes in equipment status as applicable.

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

- The Unit is operating at 100% power with all systems in NSA.
- The following control room annunciators are received:
  - [A6-100], "STA INSTR AIR RCVR TANK PRESS LOW".
  - [A6-109], "STA INSTR AIR RCVR TANK DISCH PRESS LOW".
- [PI-1IA-106], "INSTR AIR HEADER" Pressure Indicator is reading 98 psig and is slowly DROPPING.
- [PI-1IA-106B], "INSTR AIR RCVR" Pressure Indicator is reading 98 psig and is slowly DROPPING.
- All systems function as designed.

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

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

A. (1) OPEN
(2) be RUNNING

(1) CLOSED
(2) be RUNNING

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

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

Answer: C

Explanation/Justification:

A. Incorrect. Correct TV-1SA-105 status. Incorrect 1IA-C-4 status (refer to correct answer explanation).
B. Incorrect. Incorrect TV-1SA-105 status. Incorrect 1IA-C-4 status (refer to correct answer explanation).
C. Correct. TV-1SA-105 closes at 95 psig. Based on control board pressure gauge readings TV-1SA-105 will be open. The Diesel Driven Air Compressor starts at 93 psig and is therefore not yet running based on pressure stated plant conditions.
D. Incorrect. Incorrect pressure indicator. Correct 1IA-C-4 status (refer to correct answer explanation).

Sys #	**System**	**Category**	**KA Statement**
078	Instrument Air	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-53C.4.1.34.1, Rev. 16, pg. 4- 5
1OM-34.4.AAI, Rev. 7 pg. 2 & 5
1OM-34.4.AAD, Rev. 2, pg. 2 & 3
1SQS-34.1, Rev. 14, PPNT Slide HO-1

Question Source: New

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

Objective: 1SQS-34.1 4. Given a change in plant conditions, describe the response of the Compressed Air System field indication and control loops, including all automatic functions and changes in equipment status.

Given the following plant conditions:

- The Plant is in Mode 1 at 100% power.
- Containment Integrity is being analyzed.

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

A. A single Quench Spray Train is inoperable.

B. One containment isolation valve is inoperable.

C. Containment Average Air Temperature is 109 °F.

D. One containment air lock interlock mechanism is inoperable.

Answer: D

Explanation/Justification:

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

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

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

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

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

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

References provided to Candidate None

Technical References:
TS 3.6.5, Amend 280/164, pg. 3.6.5 -1
TS 3.6.3, Amend 278/161, pg 3.6.3 -1 & 2
TS 3.6.2, Amend 278/161, pg. 3.6.2 - 2
TS 3.6.6, Amend 278/161, pg. 3.6.6 - 1
TS 3.6.1 Bases, Rev. 0, pg. B3.6.1 - 1

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41/7 / 45.6)

Objective: 1SQS-47.1 5. From memory and for a given set of plant conditions, determine if the condition meets the criteria for entry into a one hour or less action statement in accordance with technical specifications.

. Which ONE of the following describes the results of the following actions? (Assume all systems function as designed and no other operator action occurs)

Depressing the Train "A" **CIA** manual actuation pushbutton will result in __ (1) __.
Depressing a single Train "B" **CIB** manual actuation pushbutton will result in __ (2) __.

A. (1) ALL CIA valves CLOSING
(2) ALL CIB valves CLOSING

B. (1) ALL CIA valves CLOSING
(2) ALL CIB valves REMAINING AS IS

C. (1) ONLY CIA valves in ONE Train CLOSING.
(2) ALL CIB valves REMAINING AS IS

D. (1) ONLY CIA valves in ONE Train CLOSING.
(2) ALL CIB valves CLOSING

Answer: B

planation/Justification:

Incorrect. Correct CIA response. CIB valves do not automatically reposition. 2/2 pushbuttons are required to initiate CIB train specific.

B. Correct. There are two CIA pushbuttons. (one for each train) Either pushbutton will automatically isolate both trains therefore all CIA valves will close. There are two pushbuttons for each CIB train. Since only one pushbutton was depressed, no CIB valves reposition (2/2 logic required).

C. Incorrect. Incorrect CIA position. Plausible if candidate believes that the pushbuttons are train specific. Correct CIB response.

D. Incorrect. Incorrect CIA position. Plausible if candidate believes that the pushbuttons are train specific. Incorrect CIB response. Plausible if candidate does not know the logic (ie: they believe the logic is ½ versus 2/2)

Sys #	**System**	**Category**	**KA Statement**
103	Containment	Ability to monitor automatic operation of the containment system, including:	Containment isolation.

K/A# A3.01 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** USFAR Fig. 7.2-1, Rev. 22
3SQS-1.1 PPNT Slides 91, 93, 95 & 96

Question Source: New

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

Objective: 3SQS-1.1 8. Describe the control, protection and interlock functions for the control room components associated with the RPS and ESFAS signals, including automatic functions, setpoints and changes in equipment status as applicable.

Given the following plant conditions:

- A Manual Daily Heat Balance was performed during operation at 90% power.
- The feedwater temperature inputs used for the calorimetric were erroneously 10 °F **LOWER** than actual feedwater temperature.
- Power Range Nuclear Instrumentation (NI) was adjusted in accordance with 1OM-54.4.C1-3, "Daily Heat Balance" using Venturi Feedwater Flow Indication.

Which ONE of the following describes the effect of this adjustment?

Indicated power is now ____ (1) ____ than actual power.
Power Range NI was set ____ (2) ____.

A. (1) LESS
(2) conservatively

B. (1) LESS
(2) **NON**- conservatively

C. (1) MORE
(2) conservatively

D. (1) MORE
(2) **NON**- conservatively

Answer: C

Explanation/Justification:

A. Incorrect. Incorrect indicated power. Correct power range setting.
B. Incorrect. Opposite effect. Plausible if candidate has misconceptions of heat balance inputs and calculations.
C. Correct. With a lower than actual feedwater input into Attachment A, the SG Power calculated in step 12 will be lower than actual. This in turn makes step 13 (RCS Output) a higher number which in turn makes step14 (Net Reactor Power) a higher number. Therefore indicated power as calculated by heat balance is higher than actual. When NI adjustments are made they will reduce power so that they are set more conservatively.
D. Incorrect. Correct indicated power. Incorrect power range setting.

Sys #	System	Category	KA Statement
015	Nuclear Instrumentation	Knowledge of the operational implications of the following concepts as they apply to the NIS:	Factors affecting accuracy and reliability of calorimetric calibrations.

K/A# K5.04 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-54.4.C1-3, Rev. 29, pg. 3, 20 & 21; 3SQS-2.1, Rev. 8 PPNT Slides 125 -128

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.5 / 45.7)

Objective: 3SQS-3.1 11. Explain how and why the NIS power range channels are adjusted based on calorimetric data.

. Given the following plant conditions:

- A small fire occurred in the Control Room.
- The Operating Crew is performing actions of AOP 1.33.1A, "Control Room Inaccessibility".
- The RO and SM have manned the Emergency Shutdown Panel (SDP) and are transferring control of equipment.

Which ONE of the following describes parameters that are **DIRECTLY** available to be read at the SDP in order to perform AOP 1.33.1A actions?

1. Wide Range Steam Generator Water Level
2. Charging and Letdown Flow
3. Auxilliary Feedwater Flow
4. Subcooling Margin

A. 1 & 3 **ONLY**.

B. 2 & 4 **ONLY**.

C. 2 & 3 **ONLY**.

D. 1, 3, &, 4 **ONLY**.

Answer: A

Explanation/Justification:

A. Correct. AOP 1.33.1A & BVPS Unit 1 UFSAR reference the parameters (channel values) that can be read at the SDP outside the control room. By design these are NNIS values which are necessary for safe S/D of the plant. From the parameters listed only Wide Range S/G Water Level & AFW Flow can be read at the SDP.

B. Incorrect. Plausible that charging and letdown flow are controlled at the SDP because Charging Pumps and Letdown Valve control is transferred to the SDP. Also plausible that subcooling is monitored, however, there are separate indicators for PRZR Pressure and RCS Temperature. Subcooling Margin can be derived but not directly. Some operators may believe that only NR S/G water level can be monitored at the SDP and rule this choice out as a result.

C. Incorrect. AFW Flow is correct but charging and letdown flow is incorrect.

D. Incorrect. Correct that WR S/G water level and AFW Flow can be monitored, however, as discussed above Subcooling Margin cannot be directly obtained at the SDP.

Sys #	System	Category	KA Statement
016	Non-nuclear Instrumentation	Knowledge of the NNIS design feature(s) and/or interlock(s) which provide for the following:	Reading of NNIS channel values outside control room.

K/A# K4.01 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None

Technical References: 1OM-53C.4.1.33.1A, Rev. 12, Pg. 9-11 & 14
BVPS UFSAR Unit 1, Rev. 21, Table 7.4-1
BVPS UFSAR Unit 1 , Rev.19, Pg 7.8-1

Question Source: New

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

Objective: 3SQS-53.5 13. Describe the actions for control room inaccessibility.

. Which ONE of the following describes the displayed control room indication for a Core Exit Thermocouple (CET) for an open **AND** shorted circuit? (consider each separately)

As compared to a NON-effected CET, a CET detector with an open circuit will read __ (1) __.
As compared to a NON-effected CET, a CET detector with a shorted circuit will read __ (2) __.

A. (1) LOWER
(2) LOWER

B. (1) HIGHER
(2) HIGHER

C. (1) LOWER
(2) THE SAME

D. (1) THE SAME
(2) THE SAME

Answer: A

Explanation/Justification:

Correct. A failed CET sensor due to a shorted circuit or an open circuit will both result in the control room indication failing low.

B. Incorrect. Opposite failure effect. Plausible if the candidate does not know the failure mechanism.

C. Incorrect. Correct for open circuit. Incorrect for shorted circuit. Plausible for adverse containment conditions.

D. Incorrect. This would be the effect if due to adverse containment. The candidate may confuse the purpose of reference junction boxes which are designed to compensate for adverse containment conditions during accidents.

Sys #	System	Category	KA Statement
017	In-Core Temperature Monitor System (ITM)	Knowledge of the effect of a loss or malfunction of the 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. 5. pg. 16 & 17
3SQS-3.1 PPNT Slides

Question Source: New

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

Objective: 3SQS-3.1 10. Describe the response of thermocouple readouts for open and short circuits.

Given the following plant conditions:

- The Plant is in Mode 5 cooling down via Residual Heat Removal (RHR) System.
- Containment Purge is established to the Ventilation Vent.
- A common mode failure results in a complete Loss of RHR and the Control Room team enters AOP 1.10.1, "Loss of RHR Capability".
- Radiation levels are increasing on Ventilation Vent Monitor [RM-VS-109].
- [RM-VS-104A **AND** B] are Out of Service (OOS).

If Radiation Levels continue to increase above the High alarm setpoint for **[RM-VS-109]**, which ONE of the following Containment Purge responses will occur to mitigate the implications of this accident, if any?

1. Containment Evacuation Alarm automatically sounds.
2. CNMT Purge Supply and Exhaust Fans AUTO STOP. [1VS-HV-5 & F-5]
3. CNMT Isolation Purge Supply and Exhaust Dampers AUTO CLOSE. [1VS-D-5-3A & 5A]

A. 1 **AND** 2 ONLY.

B. 2 **AND** 3 ONLY.

1, 2, **AND** 3.

D. No **AUTO** actions will occur.

Answer: D

Explanation/Justification:

A. Incorrect. Correct that #2 occurs, however, #1 will occur from RM-1VS-104B ONLY. Plausible because during fuel movement when fuel is no longer irradiated, the auto closure of these dampers are defeated. Incorrect because RM-1VS-104A is OOS.

B. Incorrect. If RM-1VS-104A was not out of service, 1VS-HV-5, CNMT Purge Supply Fan and 1VS-F-5 CNMT Purge Exhaust Fan both will stop. 1VS-D-5-3A and 1VS-D-5-5A will both close to isolate the CNMT from outside.

C. Incorrect. Containment Evacuation alarm is actuated from RM-1VS-104B ONLY which is OOS.

D. Correct. No automatic actions occur from RM-1VS-109. The procedure requires the operating crew to monitor for increased radiation and consider manual isolation if a high radiation condition exists. This is operationally relevant to the way BVPS is currently operated. These actions serve to help mitigate the effects of a high radiation condition. Mitigation strategy is strictly "manual action". All of the distractors are based on original plant design of RM-1VS-104A and B radiation monitors.

Sys #	System	Category	KA Statement
029	Containment Purge	N/A	Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

K/A# 2.4.9 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None

Technical References: 1OM-44C.4.A, Rev. 21, Pg. 5; 1SQS-44C.1, Rev. 9, PPNT Slide #41; 1OM-43.4.AEG, Rev. 4 pg. 2 & 3; 1OM-43.4.AEH, Rev. 4, pg. 2 & 3

Question Source: New

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

.jective: 1SQS-44C.1 Describe the control, protection and interlock functions for the control room components associated with the containment ventilation system, including automatic functions, setpoints and changes in equipment status as applicable: Hi-Hi Radiation.

Given the following plant conditions:

- Unit 1 is at 55% power with all systems in normal alignment for this power level.
- A Loss of Auto Stop Oil pressure occurs resulting in a Turbine Trip.
- All systems respond as designed.

Which ONE of the following describes the **INITIAL** response of primary plant parameters (RCS temperature and pressure) ten seconds after the turbine trips?

RCS temperature will ____ (1) ____.
RCS pressure will ____ (2) _____.

A. (1) increase
(2) increase

B. (1) increase
(2) decrease

C. (1) decrease
(2) increase

(1) decrease
(2) decrease

Answer: D

Explanation/Justification:

A. Incorrect. Incorrect temperature and pressure. These are indicative of a loss of load without a reactor trip.
B. Incorrect. Incorrect temperature response. Correct pressure response. These are indicative of a loss of load and also a open PORV.
C. Incorrect. Correct temperature response. Incorrect pressure response. These are indicative of a reactor trip with failure of steam dumps.
D. Correct. Since the reactor is > P-9 (49%) a turbine trip results in a reactor trip. A reactor trip will result in an initial drop in RCS temperature since steam dumps will open to maintain RCS temperature at 547 F. The RCS cooldown results in a loss of fluid in the PRZR which in turn drops RCS pressure initially until steam dumps close and PRZR heaters can compensate for the cooldown.

Sys #	System	Category	KA Statement
045	Main Turbine Generator	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with MT/G system controls including:	Expected response of the primary plant parameters (temperature and pressure) following T/G trip.

K/A# A1.05 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** 3SQS-1.1 (Unit 1) Rev. 7 PPNT Slide 121 & 122
Unit 1 Protection Permissives
GO-3ATA-3.2, Rev. 3, PPNT slide # 10, 13, 15

Question Source: New

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

Objective: GO-3ATA-3.2 1. Predict and analyze the plant response (Tavg, reactor power, net reactivity, pressurizer pressure, pressurizer level, steam generator pressure, steam generator level and steam flow to the following transients: Reactor trip from 100% power.

. Given the following plant conditions:

- The plant is operating at 100 % power with all systems in NSA.
- A 10 gpm Steam Generator Tube Leak occurs.
- A Hi-Hi Radiation signal is confirmed on RIS-1SV-100, "Condenser Air Ejector Discharge".
- All systems function as designed.

With no operator action, which ONE of the following describes the plant/component response to this set of plant conditions?

[TV-1SV-100A], Condenser Air Ejector to Containment Trip Valve ____ (1) ____ **AND**
[TV-1SV-100B], Condenser Air Ejector to Gaseous Waste Trip Valve ____ (2) ____.

A. (1) OPENS
(2) CLOSES

B. (1) CLOSES
(2) OPENS

C. (1) REMAINS OPEN
(2) REMAINS CLOSED

D. (1) REMAINS CLOSED
(2) REMAINS OPEN

Answer: A

Explanation/Justification:

A. Correct. If a Hi-Hi radiation level is reached (stated in question stem), then TV-1SV-100A opens and TV-1SV-100B closes to reposition air ejector off gas from the gaseous waste system to the containment. This design in unique to Unit 1.
B. Incorrect. This is opposite of the expected response.
C. Incorrect. This is plausible if the candidate is not familiar with NSA and believes the Hi signal already caused the alignment to occur.
D. Incorrect. This would be the response for a Hi signal versus Hi-Hi signal.

Sys #	**System**	**Category**	**KA Statement**
055	Condenser Air Removal	Knowledge of the physical connections and/or cause-effect relationship between the CARS and the following systems:	PRM system.

K/A# K1.06 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-26.1.B, Rev. 11, pg. 34, 35, & 43
Op Man Fig. 26-6, Rev. 15

Question Source: Modified Bank – Vision # 464

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

Objective: 1SQS-26.1 12. Given a Main Turbine, Main Condenser, Condenser Air Removal system and MSR configuration and without referenced material; describe the Main Turbine, Main Condenser, Condenser Air Removal system and MSR control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable: Air Ejector Air Discharge Hi-Hi Radiation.

Given the following plant conditions:

- The Unit is at 100% Power with all systems in NSA.
- A liquid waste release from [1BR-TK-4B], "B Coolant Recovery Tank" is in progress using [1BR-P-2B], "Evaporator Feed Pump".
- [RM-1LW-104], "Liquid Waste Effluent Monitor has generated a **HIGH-HIGH** radiation signal.

Which ONE of the following describes the resulting position/status of the following Liquid Waste system components?

[1BR-P-2B], "Evaporator Feed Pump"
[TV-1LW-105], "Liquid Waste Effluent Trip Valve"
[FCV-1LW-104-2], "Liquid Waste Effluent High Range Flow Control Valve"

	[1BR-P-2B]	[TV-1LW-105]	[FCV-1LW-104-2]
A.	STOPPED	OPEN	OPEN
B.	RUNNING	OPEN	CLOSED
	STOPPED	CLOSED	OPEN
D.	RUNNING	CLOSED	CLOSED

Answer: D

Explanation/Justification:

A. Incorrect. Evaporator Feed Pump continues to run. The Hi-Hi radiation signal does not trip the pump. Both TV-1LW-105 and FCV-1LW-104-2 receive a close signal.

B. Incorrect. Correct that Evaporator Feed Pump continues to run and FCV-1LW-104-2 closes. TV-1LW-105 also receives a close signal on hi-hi radiation.

C. Incorrect. Evaporator Feed Pump continues to run. The Hi-Hi radiation signal does not trip the pump. TV-1LW-105 does isolate. FCV-1LW-104-2 also isolates.

D. Correct. On a Hi-Hi radiation signal from RM-1LW-104, a signal is generated which isolates liquid waste by closing TV-1LW-105 and FCV-1LW-104-2. The Evaporator Feed Pump continues to run.

Sys #	System	Category	KA Statement
068	Liquid Radwaste	Ability to monitor automatic operation of the Liquid Radwaste system including:	Automatic isolation.

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

References provided to Candidate None **Technical References:** 1OM-17.1.D, Issue 4, Rev. 1, pg. 13 & 14

Question Source: Bank – Vision # 45676

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

Objective: 1SQS-17.1 5. Given a Liquid Waste Disposal System configuration and without reference material, describe the Liquid Waste Disposal System field response to the following off-normal conditions, including automatic functions and changes in equipment status: High-High Radiation.

Given the following plant conditions and sequence of events:

- [1GW-TK-1A], "Gaseous Waste Decay Tank" is being batch discharged to the atmosphere IAW 1OM-19.4.E, "Decay Tank Discharge".
- The National Weather Service informs the Control Room that a Tornado Watch has been issued for Beaver County due to favorable conditions for tornado formation.
- The Control Room Team enters AOP ½.75.1, "Acts of Nature – Tornado or High Wind Conditions".
- The STA reports that the Primary and Redundant Nominal Elev. 150' Wind Speed Meteorological Instrumentation has failed.

Which ONE of the following actions regarding [1GW-TK-1A] discharge is required, if any?

A. No action is required.

B. Suspend the discharge.

C. Increase dilution flowrate.

D. Reduce gas discharge flowrate.

Answer: B

Explanation/Justification:

A. Incorrect. Opposite of the correct answer. Plausible if the candidate does not recognize the impact of meteorological changes/instrumentation.
B. Correct. Discharge must be secured IAW AOP ½.75.1. Also LRM 3.3.3 has an immediate action to secure the discharge due to both primary and backup meteorological instrumentation which is malfunctioning.
C. Incorrect. Plausible action that discharge will continue by increasing dilution, however, not procedurally correct.
D. Incorrect. Plausible action that discharge will continue by reducing the gas discharge flowrate, however, not procedurally correct.

Sys #	System	Category	KA Statement
071	Waste Gas Disposal	Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Meteorological changes.

K/A# A2.08 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None

Technical References:
1SQS-19.1, Rev. 15 PPNT Slide
1OM-19.4.E, Rev. 10, pg. 2
1/2OM-53C.4A.75.1, Rev. 14, pg. 1 & 6
LRM LR-3.3.3, Rev. 71/56, pg 3.3.3-1 & 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-19.1 23. For a given set of plant conditions, determine if the condition meets the criteria for entry into a less than or equal to one hour statement is accordance with the Licensed Requirements Manual and Off Site Dose Manual.

. Given the following plant conditions:

- The plant was operating at Full Power with all systems in NSA.
- A Loss of Offsite Power occurred following a design basis earthquake.
- The Control Room Team is performing actions of E-0, "Reactor Trip or Safety Injection".
- Emergency Diesel Generator 1-1 has failed to start.
- All other systems function as designed.

Which ONE of the following describes the status of power to [1WR-P-9A & 9B], "Auxiliary River Water Pumps"?

[1WR-P-9A] has ____ (1) ____.
[1WR-P-9B] has ____ (2) ____.

A. (1) power
(2) power

B. (1) **no** power
(2) power

(1) power
(2) **no** power

D. (1) **no** power
(2) **no** power

Answer: B

Explanation/Justification:

A. Incorrect. Incorrect that 1WR-P-9A has power (refer to correct answer explanation) Correct 1WR-P-9B status.
B. Correct. 1WR-9A is normally powered from Bus 1AE. Since there is no offsite power and EDG 1-1 failed to start, 1WR-9A has no power. 1WR-P-9B is powered from Bus 1DF. Since all other systems functioned as designed, EDG 1-2 started and is supplying Bus 1DF so therefore 1WR-P-9B does have power.
C. Incorrect. Incorrect that 1WR-P-9A has power (refer to correct answer explanation) Incorrect 1WR-P-9B status. Plausible if the candidate does not understand integrated plant status and power supplies.
D. Incorrect. Correct that 1WR-P-9A has power. Incorrect 1WR-P-9B status

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Sys #	**System**	**Category**	**KA Statement**
075	Circulating Water	Knowledge of bus power supplies to the following:	Emergency/essential SWS pumps

K/A# K2.03 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-30.3.C, Rev. 17, pg. 10 & 11

Question Source: New

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

Objective: 1SQS-30.2 3. Identify the power supplies for the components identified on the NSA system flow-path drawing which are powered from the class 1E electrical distribution system.

. Given the following plant conditions:

- The plant is at full power with all systems in NSA.
- A fire is reported in the Turbine Building.
- Fire Main pressure is 110 psig and slowly LOWERING.
- No action has been taken by the Control Room team and all systems function as designed.

If Fire Main pressure continues to drop which fire pump will start first **AND** what will be the control room indication that it has started?

[1FP-P-1], "Motor Driven Fire Pump" starts ___ (1) ___ [1FP-P-2], "Engine Driven Fire Pump". The control room indication(s) that [1FP-P-1] started is (are) by ____ (2) ____.

A. (1) BEFORE
(2) annunciator ONLY.

B. (1) AFTER
(2) annunciator ONLY.

C. (1) BEFORE
(2) annunciator, RED indicating light, and amp-meter.

D. (1) AFTER
(2) annunciator, RED indicating light, and amp-meter.

Answer: C

Explanation/Justification:

A. Incorrect. Correct sequence. Partial indication (refer to correct answer explanation).
B. Incorrect. Incorrect sequence. Partial indication (refer to correct answer explanation).
C. Correct. Upon lowering fire main water pressure 1FP-P-1 will auto start first @ 105 psig followed by 1FP-P-2 @ 95 psig. Both pumps have an annunciator in the control room which informs the operator that a pump start has occurred. Only 1FP-P-1 has indicating lights and an amp-meter. The candidate may not know the pump start sequence or BSP indications which makes all distractors plausible.
D. Incorrect. Incorrect sequence. Correct indications.

Sys #	System	Category	KA Statement
086	Fire Protection	Ability to manually operate and/or monitor in the control room.	Fire water pumps.

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

References provided to Candidate None **Technical References:** 3SQS-33.1, Rev. 7, PPNT Slide # 64
1OM-33.1.B, Issue 4, Rev. 3, pg. 1
1OM-33.1.D, Issue 4, Rev. 3, pg. 1
1OM-33.4.ACL, Rev. 1, pg. 2

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR:41/7 / 45.5 to 45.8)

Objective: 3SQS-33.1 8. Given a specific plant condition, predict the response of the Fire Protection System control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off normal event.

. Given the following plant conditions:

- The Unit was operating at Full Power with all systems in NSA.
- A LOCA occurred and the Control Room Team transitioned to E-1, "Loss of Reactor or Secondary Coolant".
- The following plant conditions exist:
  - o RCS pressure is 520 psig and slowly DROPPING.
  - o Core Exit Thermocouple Temperatures are 472 °F and slowly DROPPING.
  - o RCS loop cold leg temperatures are 290 °F and slowly DROPPING.
  - o S/G pressures are 800 psig and slowly DROPPING.
  - o RCS ΔT is indicating UPSCALE.
  - o RCP's are **NOT** running.

Based on these conditions, which ONE of the following identifies the non-passive source(s) of SI flow providing core cooling, **AND** what is the status of natural circulation?

A. High Head SI flow **ONLY**; Natural Circulation is occurring.

B. High Head SI flow **ONLY**; Natural Circulation is **NOT** occurring.

High Head **AND** Low Head SI flow; Natural Circulation is occurring.

D. High Head **AND** Low Head SI flow; Natural Circulation is **NOT** occurring.

Answer: B

Explanation/Justification:

A. Incorrect. Correct SI flow but incorrect natural circulation conclusion (refer to correct answer explanation)

B. Correct. With RCS pressure at 520 psig, the High Head SI pumps will be supplying SI flow for core cooling only. The shutoff head for the Low Head SI pumps is about 178 psig so therefore will not be providing flow. Natural circulation is not occurring. Decreasing CETCs, Tcold and upscale ΔT are all indicative of natural circulation making it plausible that natural circulation is occurring. However the candidate must recognize that there is a disconnect between S/G pressures and cold leg temperatures. Saturation temperature for S/G pressures at 800 psig is 520 F. Tcold is well below this temperature implying the cold down is more indicative of SI flow. Break flow is the primary removal mechanism and based on saturation temperature for S/G pressure although some reflux boiling may occur, conditions for natural circulation are not present. Both ES-1.1 & 1.2 reference Attachment 2-G for verification of natural circulation flow in the LOCA series procedures making this question operational relevant

C. Incorrect. Incorrect that Low Head SI flow is occurring and incorrect natural circulation conclusion (refer to correct answer explanation)

D. Incorrect. Incorrect that Low Head SI flow is occurring but correct natural circulation conclusion (refer to correct answer explanation).

Sys #	System	Category	KA Statement
N/A	N/A	Conduct of Operations	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

K/A# 2.1.7 **K/A Importance** 4.4 **Exam Level** RO

References provided to Candidate Steam Tables

Technical References:
1OM-53A.1.2-G, Issue 1C, Rev. 1, Pg. 2
1SQS-11.1, Rev. 13, Pg. 15, 40, & 41
1OM-53A.1.ES-1.2, Issue 1C, Rev. 15, pg. 12
1OM-53A.1.ES-1.1, Issue 1C, Rev. 10, Pg. 13

estion Source: New

.uestion Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.5/ 43.5 / 45.12/45.13)

Objective: 3SQS-53.2 11. State from memory five conditions which indicate that natural circulation is occurring, IAW BVPS EOP Executive Volume.
1SQS-11.1 17. List the nominal value of the control room operating parameters associated with Safety Injection.

Which ONE of the following describes the system or component status for a value displayed in Reverse Video **RED** on the Safety Parameter Display System (SPDS)?

A. Used to denote static or reference material.

B. Used to denote dynamic or important information.

C. Indicates SPDS display values have exceeded their process alarm limits.

D. Indicates a data quality other than good and may need further evaluation.

Answer: C

Explanation/Justification:

A. Incorrect. This condition would be noted by a cyan background color.
B. Incorrect. This condition would be noted by a green background color.
C. Correct. A RED background color indicates SPDS display values have exceeded their limits.
D. Incorrect. This condition would be noted by a yellow background color.

Sys # N/A **System** N/A **Category** Conduct of Operations **KA Statement** 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.D, Issue 4, Rev. 1, pg. 1

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 45.12)

Objective: 1SQS-5C.1 1. Obtain requested information from the SPDS in accordance with the unit specific worksheet.

Given the following plant conditions:

- The Reactor Operator (RO) pages the Primary Auxiliary Building Operator (NLO) and directs him to close [1AC-238], "Chilled Water Return From Unit [1VS-AC-11B]".

Which ONE of the following is a BVPS expectation for this phone communication between the RO and NLO, according to 1/2OM-48, "Conduct of Operations"?

A. A verbatim repeat back must be used.

B. No action will take place until three-part communication is complete.

C. Use of phonetic alphabet is required if three-part communication is **NOT** used.

D. Multiple actions may be part of the communication as long as repeat backs are used.

Answer: B

Explanation/Justification:

A. Incorrect. Repeat backs may be paraphrased.

B. Correct. According to 1/2OM-48, Operational orders SHALL be repeated back to ensure the order is properly understood. Repeat backs are not required for communications which do not direct an action to be performed. In this case the RO gives the field operator an operational order and therefore a repeat back is required. This is a station priority for HIT training and has been a focus OE area for BVPS.

C. Incorrect. Phonetic alphabet should always be used particularly if confusion is possible.

D. Incorrect. Multiple actions should not be contained in communications where control or coordination exists. If it is necessary to give complex or multiple action orders, they should be written down or a copy of the procedure should be obtained.

Sys #	System	Category	KA Statement
N/A	N/A	Conduct of Operations	Knowledge of the stations requirements for verbal communications when implementing procedures.

K/A# 2.1.38 **K/A Importance** 3.7 **Exam Level** RO

References provided to Candidate None **Technical References:** 1/2OM-48.1.D, Rev. 6 pg. 5 - 8

Question Source: Bank – Vision # 1682

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

Objective: 3SQS-48.1 7. From memory, describe the shift rules of practice including: communication.

Given the following plant conditions and sequence of events:

- A Reactor Startup is in progress following a forced outage.
- The RO has just completed taking critical data and is directed to raise power to 4%.
- The RO establishes a positive startup rate and releases the IN-HOLD-OUT switch.
- Rods continue to step outward as indicated on group step counters and IRPI.

Which ONE of the following describes the required immediate action **AND** consequence if **NO** operator action is taken?

The immediate required action is to manually ___ (1) ___.
The consequence of inaction is that reactor power will increase to ___ (2) ___
before being automatically terminated by the Reactor Protection System ____ (3) ____.

A. (1) insert control rods to zero steps.
(2) 20%
(3) Intermediate Range Rod Stop Signal.

B. (1) trip the reactor.
(2) 25%
(3) Power Range Reactor Trip Signal.

(1) insert control rods to zero steps.
(2) 25%
(3) Intermediate Range Reactor Trip Signal.

D. (1) trip the reactor.
(2) 30%
(3) Power Range Reactor Trip Signal.

Answer: B

Explanation/Justification:

A. Incorrect. Inserting control rods is a correct action if criticality is achieved early or outside expected rod height IAW 1OM-50.4.4.D. Since there is an intermediate rod stop at 20%, this is a plausible distractor.

B. Correct. The candidate must interpret indications provided and recognize an uncontrolled rod withdrawal is in progress and then apply immediate action and systems knowledge to demonstrate understanding of operator actions and impact on system conditions if these actions are not taken. Step 1 of AOP 1.1.3 (RCCA Control bank Inappropriate Continuous Movement) checks for load rejection. Since the turbine is not yet in service, the RNO Immediate Action is applicable. Since rods are already in MANUAL and a continuous rod withdrawal is occurring based on symptoms provided, the RO shall trip the reactor. A reactor trip will automatically occur in this scenario when 2/4 power range channels are above 25% because the low power trip was not manually blocked above P-10 (10%) because no operator action is taken.

C. Incorrect. Incorrect action as discussed. It is correct that a reactor trip will occur at 25% based on intermediate current equivalent neutron flux.

D. Incorrect. Correct action. Incorrect low power range setpoint. Plausible if candidate confused with P-8 setpoint (30%).

Sys #	System	Category	KA Statement
N/A	N/A	Equipment Control	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

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

References provided to Candidate None **Technical References:** 1OM-53C.4.1.1.3, Rev. 11, pg. 1 & 2
BVPS Unit 1 & 2 TS 3.3.1, Amend 278/261, Pg. 12
1OM-1.5.B.3, Rev. 12, Pg. 2
1OM-50.4.D, Rev. 53, Pg. 35 & 36

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR 41.5 / 43.5/ 45.12)

Objective: 3SQS-53.3 14. Apply the actions for a rod position malfunction.

. Given the following plant conditions:

- The Shift Manager (SM) has determined that an Annunciator alarm should be disabled.
- The corresponding knife switch for this alarm has been repositioned OPEN.

Which ONE of the following is 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 sticker on the alarm window.
3. Create and hang an Operations information Tag/label or Caution Tag on annunciator window and on the opened slide link/knife switch stating what alarm was removed from service.

A. 1 **ONLY**.

B. 1 **&** 2 **ONLY**.

C. 2 **&** 3 **ONLY**.

D. 1, 2, **&** 3.

Answer: D

Explanation/Justification:

A. Incorrect. Only partially correct.
B. Incorrect. Only partially correct.
C. Incorrect. Only partially correct.
D. Correct. According to NOP-OP-1014 all of these choices are required. All distractors are plausible if candidate does not know procedural required actions.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Equipment Control	Knowledge of the process 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. 1, pg. 24 & 25

Question Source: New

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

According to NOP-OP-4201, "Routine External Exposure Monitoring" which ONE of the following are Federal Legal Dose Limits?

The Federal Occupational Dose Limit for a Declared Pregnant Worker is __ (1) __.
The Federal Occupational Dose Limit for an individual working at a Nuclear Facility is __ (2) __.

A. (1) 100 mr/**term**
(2) 5000 mr/year

B. (1) 500 mr/**term**
(2) 100 mr/year

C. (1) 100 mr/**term**
(2) 100 mr/year

D. (1) 500 mr/**term**
(2) 5000 mr/year

Answer: D

Explanation/Justification:

Incorrect. Incorrect but plausible DPW limit. The DPW is limited to 100 mr/month but may have up to 500 mr for the entire term or gestation period. The occupational limit for an individual working at a nuclear facility is correct.

B. Incorrect. Correct value for DPW. Incorrect value for second limit however, plausible since this is the limit for a member of the general public.

C. Incorrect. Incorrect DPW value but plausible as described above. Incorrect value for second limit however, plausible since this is the limit for a member of the general public.

D. Correct. According to NOP-OP-4201 Attachment B, when a worker declares pregnancy, she will have an administrative level of 500 mr for the term of pregnancy. This ensures the dose to the unborn child is minimized. The federal limit is also 500 mrem for the pregnancy period or term. NOP-OP-4101 also refers the reader to NOP-OP-4202 which defines a declared pregnant worker and specifies the occupational dose limit for the entire period of declared pregnancy is 500 mrem (100 mrem/month) NOP-OP-4101 states the federal limit for occupational exposure is 5000 mr/year. It also states that 100 mrem/year is the limit for a member of the general public which is used as a plausible distractor. These values are consistent with Federal Limits as taught in basic radworker training.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Radiation Control	Knowledge of the radiation exposure limits under normal or emergency conditions.

K/A# 2.3.4 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None

Technical References: NOP-OP-4201, Rev. 1, pg. 3, 21 & 22
NOP-OP-4202, Rev. 0, pg. 3 & 4
FEN-RWT, Chapter 4, Rev. 3, pg. 2

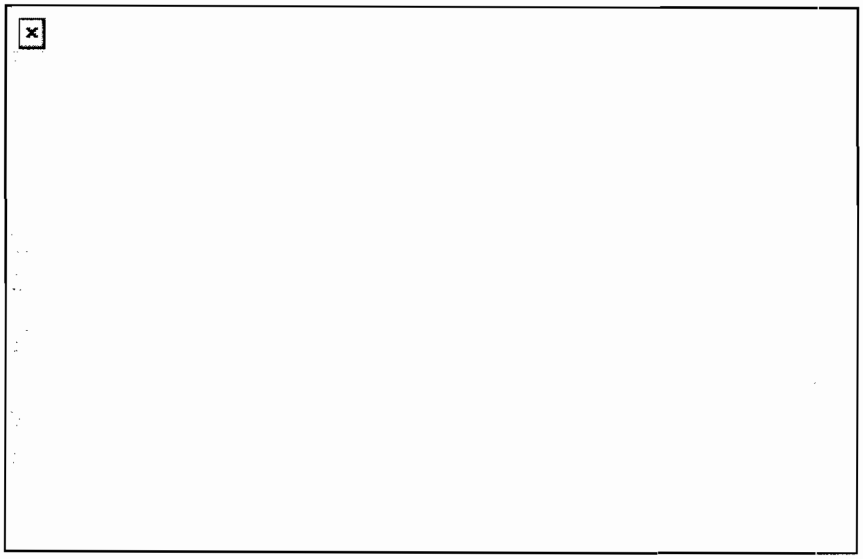
Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental

10 CFR Part 55 Content: (CFR: 41.12 / 43.4 / 45.10)

Objective: FEN-RWT 2. State the federal radiation dose limits for total dose equivalent (TEDE), for skin, extremities, and lens of the eye.

Beaver Valley Unit 1 NRC Written Exam (1LOT8)



Refer to the attached photograph of the LIQUID WASTE EFFLUENT monitor control module. The High-High setpoint will be displayed on the meter when the high-high button is pressed **ONLY** if the rotary switch is in which ONE of the following positions ________?

off

B. cal.

C. h.v.

D. oper.

Answer: B

Explanation/Justification:

A. Incorrect. This position will not cause the setpoint to be displayed. Plausible since it is one of the four positions.

B. Correct. Placing the switch in the cal position will allow the high radiation and high-high radiation alarm setpoints to be displayed when the respective high or high-high pushbutton is pressed.

C. Incorrect. This position will not cause the setpoint to be displayed. Plausible since it is one of the four positions.

D. Incorrect. This position will not cause the setpoint to be displayed. Plausible since it is one of the four positions.

Sys #	System	Category	KA Statement
N/A	N/A	Radiation Control	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personal monitoring equipment, etc.

K/A# 2.3.5 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** 1OM-43.4.C, Rev. 4, pg. 5

Question Source: Bank – Vision # 593

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.11 / 41.12 / 43.4 / 45.9)

ective: 1SQS-43.1 1. Describe the function of the Radiation Monitoring Systems and associated major components as documented in Chapter 43 of the Unit 1 Operating Manual.

. Given the following plant conditions:

- You are directed to post a clearance in the Primary Auxiliary Building (PAB).
- This task requires entry into an Area where General Radiation Levels are 110 mr/hr.

In addition to reviewing your approved RWP and radiological conditions, which of the following item(s) will be required prior to entry into this area IAW NOP-OP-4101 **AND** NOP-OP-4107?
(NOP-OP-4107, "Radiation Work Permit")
(NOP-OP-4101, "Access Controls for Radiologically Controlled Areas")
(Note this is **NOT** an all inclusive list)

1 TLD and a Direct Reading Dosimeter.
2. TLD and an Alarming Direct Reading Dosimeter with appropriate augmentation.
3. Self Briefing documented on NOP-OP-4107-05, "RA Request/Briefing Form".
4. A Trip Ticket that has been initialed by a Radiation Protection Technician.

A. 1 **ONLY**.

B. 1 & 3 **ONLY**.

C. 2 & 4 **ONLY**.

2, 3, & 4 **ONLY**.

Answer: C

Explanation/Justification:

A. Incorrect. Incorrect since this is a HRA. Plausible and correct for RA entry. (refer to correct answer explanation).

B. Incorrect. Incorrect because NOP-OP-4107 does not allow self briefing for entry into a HRA since it is considered higher risk. Plausible if the candidate does not recognize this is a HRA does not know the Access Control & Briefing requirements. (refer to correct answer explanation).

C. Correct. NOP-OP-4101 defines a HRA an accessible area in which radiation levels could result in an individual receiving a deep-dose equivalent in excess of 100 mr/hr at a distance of 30 cm or more from any surface that the radiation penetrates. The candidate must recognize that they are entering a HRA versus RA and then differentiate the briefing and access requirements in order to comply with RWP requirements. NOP-OP-4107, "Radiation Work Permit" requires radiation protection briefings to ensure compliance with RWP requirements when entering HRAs. For entry into radiation areas considered to be low risk, operators are allowed to self brief and do not need formal briefings. According to NOP-OP-4101, For entry into a HRA, a trip ticket and dose alarm augmentation device are required. This is a higher level question because it goes beyond simple recall and requires comprehension of the operationally relevant task to be performed.

D. Incorrect. Incorrect because NOP-OP-4107 allows self briefing for entry into low risk areas ONLY. Since this is a HRA it is considered higher risk. Plausible if the candidate recognizes this is a HRA but does not know the RWP briefing requirements. (refer to correct answer explanation).

Sys #	System	Category	KA Statement
N/A	N/A	Radiation Control	Ability to comply with radiation work permit requirements during normal or abnormal conditions.

K/A# 2.3.7 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-OP-4101, Rev. 5, pg. 3-6, 9 & 17. NOP-OP-4107, Rev. 8, Pg. 14 & 15

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** (CFR: 41.12 / 45.10)

Objective: 3SSG-Admin 16. Describe the controls for maintaining personnel exposures ALARA in accordance with NOP-OP-4107, Radiation Work Permit (RWP).

. Given the following plant conditions:

- A serious fire in the cable spreading room has been reported.
- The Shift Manager determines actions of 1OM-56C, "Alternate Safe Shutdown From Outside the Control Room" are necessary.
- The SM directs the RO to perform actions of 1OM-56.C.4.C, "NCO Procedure".

Which ONE of the following is a Reactor Operator (RO) action performed outside the Control Room?

The RO will ____ (1) ____ Station Air Compressor(s) to ensure ____ (2) ____.

A. (1) start
(2) spurious fire induced operation of AOV's is prevented.

B. (1) start
(2) positive control of AOVs required for Appendix R Safe S/D.

C. (1) stop
(2) spurious fire induced operation of AOV's is prevented.

(1) stop
(2) positive control of AOVs required for Appendix R Safe S/D.

Answer: C

Explanation/Justification:

A. Incorrect. The Station Air Compressors are secured not started. Correct operational effect (refer to correct answer)

B. Incorrect. The Station Air Compressors are secured not started. Plausible reason to maintain air except station air is not relied upon for safe shutdown of the plant.

C. Correct. Station air compressors are secured IAW 1OM-56C.4.C, Attachment 2 NCO Procedure (Licensed Operation for BVPS) The reason for this alignment according to 1OM-56C.4.A which is to prevent spurious fire induced operation of AOVs. (Stopping Air Compressors deactivates valves by failing them shut).

D. Incorrect. Correct that station air compressors are secured. The operational effect is plausible since it does provide positive control, however, these valves are not relied upon for safe shutdown of the plant.

Sys #	System	Category	KA Statement
N/A	N/A	Emergency Procedures / Plan	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-56C.4.A, Rev. 9, pg. A2, A3, & A6
1OM-56C.4.C, Rev. 35, pg.13

Question Source: New

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

Objective: 1SQS-56C.1 1. Describe the function of Alternate Safe Shutdown from Outside the Control Room and the associated major components as documented in Operating Manual Chapter 1OM-56C.

. Given the following plant conditions:

- The Unit is operating at 80% with all systems NSA.
- The Rod Control Selector Switch is in AUTOMATIC.
- Control Bank "D" begins to step in continuously.
- Turbine load is stable.

Which ONE of the following will be the next required immediate operator action?

A. Manually trip the reactor.

B. Place the Control Rod Group Selector Switch in MANUAL.

C. Place the Control Rod Group Selector Switch in Bank "D" position.

D. Place the Control Rod Group Selector Switch in either shutdown bank position.

Answer: B

Explanation/Justification:

Incorrect. The reactor will be tripped as part of the RNO if after placing control rods in manual the rod insertion continues which makes this choice plausible. However, this is not the first IMA which will be performed and would be an incorrect action.

B. Correct. Symptoms or entry conditions have been met for AOP 1.1.3. Step 1 is an IMA required to be performed from memory. Since turbine load is stable as specified in question stem, the RNO required action is to place control rod group selector switch in MANUAL.

C. Incorrect. Incorrect but plausible switch position.

D. Incorrect. Incorrect but plausible switch position.

Sys #	System	Category	KA Statement
N/A	N/A	Emergency Procedures / Plan	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

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

References provided to Candidate None **Technical References:** 1OM-53C.4.1.1.3, Rev. 11, pg. 1 & 2

Question Source: Bank – Vision # 45697

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 43.2 / 45.6)

Objective: 1SQS-53C.1 1. State all Immediate Operator Actions associated with AOPs.

. Given the following plant conditions:

- The Unit is operating at 25% power.
- 1A Reactor Coolant Pump (RCP) breaker OPENS.
- Assume no operator action occurs.

What will be the steady state value of RCS flow in the 1A loop **AND** what is the TS 3.4.4 bases for **NOT** operating in this condition for greater than 6 hours?

RCS flow in the 1A loop will stop and then reverse to a value of ___ (1) ___ of nominal flow. The TS 3.4.4 bases for **NOT** operating with less than 3 RCPs in Mode 1 or 2 is to ___ (2) ___.

(MTC = Moderator Temperature Coefficient)

A. (1) ~ 20 - 30%
(2) preserve assumptions made in the safety analysis.

B. (1) ~ 50 - 60%
(2) preserve assumptions made in the safety analysis.

C. (1) ~ 20 - 30%
(2) meet the safety analysis acceptance criteria for MTC.

D. (1) ~ 50 - 60%
(2) meet the safety analysis acceptance criteria for MTC.

Answer: A

Explanation/Justification:

A. Correct. According to UFSAR analyzed partial loss of forced reactor coolant flow, idle loop flow will be about 30% of nominal flow. This accident assumes 100% power and has a conservative number of S/G tubes plugged which will effect head loss. When run on the simulator from 25% power RCS flow in the idle loop is about 20%. TS 3.4.4 bases states that the safety analyses contains various assumptions for the design bases accident initial conditions which includes RCS flow. All safety analyses performed at rated power assume three RCS loops in operation. The K/A randomly selected is RO in nature. In order to ensure the SRO criteria will be met, a TS bases question is added.

B. Incorrect. Value is incorrect. Correct bases.

C. Incorrect. Correct value. Incorrect bases. This is the bases for TS 3.4.2 (Minimum temperature for criticality) It is also very close to the bases for TS 3.4.4 (RCS Loops – Modes 1 & 2) if DNB were substituted for MTC.

D. Incorrect. Value and bases are incorrect, but balanced for plausibility.

Sys #	System	Category	KA Statement
015/017	RCP Malfunctions	Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow)	Calculation of expected values of flow in the loop with RCP secured.

K/A# AA2.07 **K/A Importance** 2.9 **Exam Level** SRO

References provided to Candidate None **Technical References:** BVPS UFSAR Unit 1, Rev. 24/23/20, pg. 14-1-13-15 & Fig. 14.1-13
BVPS Units 1 & 2 TS 3.4.4, Amend. 278/261, pg. 3.4.4-1
BVPS Units 1 & 2 TS 3.4.4 bases Amend. 278/261, pg 3.4.4-1 & 2

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** CFR: 43 (b)(2)

Objective: 3SQS-RCS-ITS 2. State the purpose of each RCS specification as described in the applicable safety analyses section of the bases.

. Given the following plant conditions and sequence of events that occur on April 20th:

- The Unit is operating at 90% power with all systems in NSA.
- At 0130 – An Inverter is declared INOPERABLE due to maintenance.
- At 1200 – A second Inverter is declared INOPERABLE due to failure.
- At 1230 – The first Inverter is declared OPERABLE.

Which ONE of the following explains whether a completion time extension can be applied to the completion time of required Technical Specification 3.8.7 Action A.1 for the **SECOND** inoperable Inverter **AND** justification? **(Reference Provided)**

The time extension ___ (1) ___ be applied.
The justification for this conclusion is because ___ (2) ___.

A. (1) can
(2) separate entry conditions are **NOT** allowed.

B. (1) can **NOT**
(2) separate entry conditions are allowed.

C. (1) can
(2) SR 3.0.2 can be used to extend the time for inoperable inverters.

D. (1) can **NOT**
(2) SR 3.0.2 can **NOT** be used to extend the time for inoperable inverters.

Answer: A

Explanation/Justification:

A. Correct. TS Use and Application Section 1.3 discuss use of completion times. All of the criteria for use of the completion time extension apply, specifically that the first inverter and second inverter were concurrently inoperable and the second inverter remains inoperable after the first inverter is declared operable. Also there is no exception that allows separate entry conditions for inverters in TS 3.8.7, so therefore the extension time can be applied. The SRO must determine whether completion time extensions are applicable and explain the bases for the decision.

B. Incorrect. Incorrect that the extension time cannot be applied. The logic is opposite of the correct answer.

C. Incorrect. Correct that extension can be applied (refer to correct answer explanation) Incorrect that SR 3.0.2 can be applied to the inoperable inverters since the second inverter failed and was not part of a surveillance. Therefore the 1.25 extension does not apply. Plausible because it does allow an extension of completion times when applicable.

D. Incorrect. Incorrect that the extension cannot be applied. The justification is plausible for the response but nonetheless incorrect.

Sys #	System	Category	KA Statement
057	Loss of Vital AC Inst. Bus	N/A	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

K/A# 2.2.36 **K/A Importance** 4.2 **Exam Level** SRO

References provided to Candidate TS 1.3 pg. 1.3-1 – 11; TS 3.8.7 pg. 3.8.7-1 & 2; TS 3.8.7 Bases pg. B3.8.7-1 -4

Technical References: TS 1.3 pg. 1.3-1 – 11; TS 3.8.7 pg. 3.8.7-1 & 2; TS 3.8.7 Bases pg. B3.8.7-1 -4; TS 3.0 pg. 3.0-4

Question Source: Bank – Vision # 57101

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(2)

Objective: 3SQS-ELEC - ITS 4. Given plant conditions that constitute non-compliance with any Electrical Power Systems LCO or Licensing Requirement determine the applicable conditions(s), required actions(s), and associated completion times.

(SRO ONLY)

Beaver Valley Unit 1 NRC Written Exam (1LOT8)

. Given the following plant conditions:

- The Unit is in Mode 3.
- [TV-1IA-400], "Inst Air to Cnmt Isol" is CLOSED for maintenance.
- [1IA-90], "Bypass Valve for [TV-1IA-400]" is OPEN.
- [PI-1IA-106A], "CNMT Instrument Air (IA) Header Pressure" is 72 psig and DROPPING.
- [PI-1IA-106], "Station Instrument Air Header Pressure" is 92 psig and DROPPING.
- All systems function as designed.

Based on these indications, which ONE of the following is the location of the air leak **AND** what will be the reason for [1IA-90] isolation?

The air leak is located ____ (1) ____ containment.
The Unit Supervisor will direct [1IA-90] CLOSED to comply with ____ (2) ____.

(Technical Specification (TS) 3.6.3, "Containment Isolation Valves"
AOP 1.34.1, Loss of Station Instrument Air" / AOP 1.34.2, "Loss of Containment IA")

A. (1) inside
(2) AOP 1.34.2 **ONLY**.

B. (1) inside
(2) AOP 1.34.2 **AND** TS 3.6.3.

(1) outside
(2) AOP 1.34.1 **ONLY**.

D. (1) outside
(2) AOP 1.34.1 **AND** TS 3.6.3.

Answer: A

Explanation/Justification:

A. Correct. The first question is RO knowledge but necessary to meet the K/A. An RO should be able to differentiate between these indications to determine whether an air leak is inside or outside containment. AOP 1.34.1 step 4 does use this diagnostics to determine procedural flowpath. The second part of the question is SRO knowledge because it requires the candidate to have knowledge of procedure content and bases for the reason the action is being taken beyond assessment of plant conditions. AOP 1.34.2 asks whether Station IA is < 95 psig and directs isolation of 1IA-90 if all available station air compressors are running which they would be if functioning as designed. This is to ensure Containment Air is isolated from Station Air which is being dragged down by the Containment Air sizable air leak. TS 3.6.3 is not the reason for isolation because although the plant is in Mode 3 and CI valves are required to be operable the operator is not closing 1IA-90 to comply with TS 3.6.3 because there is no TS bases accident in progress (ie: LOCA or Rod Ejection Accident). TS 3.6.3 action is applicable but not the reason for directing 1IA-90 closure. The competent SRO should know the bases for TS action and must understand the loss of air accident is not the basis.

B. Incorrect. Correct leak location diagnosis. Partially correct bases. (refer to correct answer explanation)

C. Incorrect. Incorrect leak location diagnosis. Correct bases. (refer to correct answer explanation)

D. Incorrect. Incorrect leak location diagnosis. Incorrect bases. (refer to correct answer explanation)

Sys #	System	Category	KA Statement
065	Loss of Instrument Air	Ability to determine and interpret the following as they apply to the Loss of Instrument Air:	Location and isolation of leaks.

K/A# AA2.03 **K/A Importance** 2.9 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.34.1, Rev. 16, pg. 3
1OM-53C.4.1.34.2, Rev. 9, pg. 2 - 4
BVPS Units 1 & 2 TS Bases, pg. B 3.6.3-1, 2 & 4

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** CFR: 43 (b)(2)

Objective: 1SQS-34.1 13. Given a change in plant conditions due to a system or component failure, analyze the compressed air system to determine what failure occurred.

3SQS-CONT-ITS 2. State the purpose of each containment system specification as described in the applicable safety analyses section of the bases.

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

- The Unit was operating at 100% power with all systems in NSA.
- A reactor trip and safety injection occurred.
- The operating crew is performing E-0, “Reactor Trip or Safety Injection” actions.
- RCS pressure is 25 psig and slowly RISING.
- Steam Generator (S/G) pressures are 650 psig and slowly LOWERING.
- All S/G Narrow Range Levels are < 31%.
- All Auxiliary Feedwater pumps have failed to Auto start and **cannot** be manually started.
- A transition to FR-H.1, “Response to Loss of Secondary Heat Sink” is made.

Which ONE of the following describes the procedural action(s) **AND** bases to mitigate the consequences of these plant conditions?

A. Remain in FR-H.1 because a small break LOCA is in progress **AND** a secondary heat sink is required.

B. Remain in FR-H.1 because a large break LOCA is in progress **AND** a secondary heat sink is required.

Go back to E-0 and transition to E-1 because a small break LOCA is in progress **AND** a secondary heat sink is **NOT** required.

D. Go back to E-0 and transition to E-1 because a large break LOCA is in progress **AND** a secondary heat sink is **NOT** required.

Answer: D

Explanation/Justification:

A. Incorrect. RCS pressure is less than the shutoff of the LHSI pumps (approximately 160 psig). Furthermore, E-1 uses 275 psig to determine the mitigation strategy for a large or small break LOCA. Less than 275 psig, the EOP directs remaining in E-1 until cold leg recirculation criteria is met. Greater than 275 psig directs the crew to transition to ES-1.2 for a post LOCA cooldown and depressurization. Therefore a LB LOCA is in progress and FR-H.1 directs the operator to return to procedure and step in effect.

B. Incorrect. RCS pressure is less than S/G pressure so therefore the S/Gs are not acting as a heat sink but rather a heat source. FR-H.1 directs a transition back to procedure and step in effect. It is correct that a LBLOCA is in progress.

C. Incorrect. Correct procedural transition. Incorrect that a SB LOCA is in progress.

D. Correct. According to the bases of FR-H.1, with RCS pressure less than S/G pressure, the break is of a larger size and SI will provide heat removal. The S/Gs are a heat source as opposed to a heat sink. Therefore a LBLOCA is in progress. FR-H.1 directs a transition back to procedure and step in effect. The impact of no automatic or manual AFW system flow because AFW pumps did not start is that with a LBLOCA, the heat will be removed by break flow alone. The SRO must assess plant conditions and select a procedure to mitigate and proceed based on the facilities approved EOP network procedures.

Sys #	System	Category	KA Statement
W/E05	Inadequate Heat Transfer – Loss of Secondary Heat Sink	Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink)	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A# EA2.2 **K/A Importance** 4.3 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-H.1, Issue 1C, Rev. 13, pg. 2; 1OM-53B.4.FR-H.1, Issue 1C, Rev. 13, pg. 47

ıstion Source: Vision # 45809

uestion Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(5)

Objective: 3SQS-53.3 3. State from memory the basis and sequence for the major action steps of each EOP procedure, IAW BVPS EOP Executive Volume.

(SRO ONLY)
Beaver Valley Unit 1 NRC Written Exam (1LOT8)

/. Given the following plant conditions:

- A Large Break LOCA occurred and the crew has transitioned to ES-1.3, "Transfer to Cold Leg Recirculation".
- They are currently at Step 5, LHSI Pumps – NO SIGNS OF CAVITATION
- The RO reports amps and flow are oscillating on both Low Head SI pumps.
- The STA reports that a RED path exists on Heat Sink.

Which ONE of the following describes the required procedural action given these conditions?

A. Immediately transition to ECA-1.1, "Loss of Emergency Coolant Recirculation"

B. Immediately transition to FR-H.1, "Response to Loss of Secondary Heat Sink".

C. Stop all charging pumps and assess signs of containment sump blockage per ES-1.3.

D. Stop all charging pumps and low head safety injection pumps, then transition to FR-H.1

Answer: C

Explanation/Justification:

Incorrect. Plausible since there are symptoms of a loss of emergency coolant recirculation due to cavitation of LHSI pumps.

B. Incorrect. Plausible since a red path normally would take priority however, FRP's should not be implemented prior to completion of the first five steps of ES-1.3. (refer to correct answer explanation)

C. Correct. The SRO must assess plant conditions and have knowledge of diagnostic steps and decision points in the EOPs which involve transition to event specific sub procedures in order to answer this question. Specifically, they must recognize that oscillating amps and flow of the LHSI pumps is indicative of LHSI pump cavitation. Because cavitation exists, they must have knowledge of the RNO action which is to stop charging pumps. An assessment of containment sump blockage must occur prior to transition. Securing charging pumps prior to transition preserves them from further damage so that they may be available in the long term. The candidate must further understand because they are in step 5 of ES-1.3 that the RED path on heat sink is not to be implemented although contrary to FRP rules of usage. ES-1.3 is an exception to these rules.

D. Incorrect. Correct that all charging pumps are stopped. Transition to FRPs is not allowed by note during Steps 1-5. FR-H.1 will not be effective due to LBLOCA.

Sys #	**System**	**Category**	**KA Statement**
W/E11	Loss of Emergency Coolant Recirc.	N/A	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-1.3, Issue 1C, Rev. 7, pg. 2 & 4
1OM-53B.4.ES-1.3, Issue 1C, Rev. 7, pg. 13
1OM-53G.1.SBCRG-1, Rev. 4, pg. 2

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(5)

Objective: 3SQS-53.3 3. State from memory the basis and sequence for the major action steps of each EOP procedure, IAW BVPS EOP Executive Volume.
3SQS-53.4 6. Given a set of conditions, locate and apply the EOP IAW BVPS-EOP Executive Volume.

Given the following plant conditions and sequence of events:

- A Reactor Trip and Safety Injection from 100% power occurred.
- Main Steam Line Isolation (MSLI) occurred on the "1A" S/G **ONLY**.
- All S/G pressures are 710 psig and DROPPING.
- All S/G NR Levels are off scale LOW.
- RCS cold leg C/D rate is 150 °F/hr.
- AFW flow to EACH S/G has been throttled to 50 gpm per S/G.
- The Control Room Team is performing the actions of ECA-2.1, "Uncontrolled Depressurization of All Steam Generators".

Based on these plant conditions:

(1) What procedure transition is required, if any?
(2) How will AFW flow be addressed?

A. (1) Remain in ECA-2.1.
(2) Continue feeding ALL S/Gs at 50 gpm.

B. (1) Remain in ECA-2.1.
(2) Isolate feed flow to "1B" AND "1C" S/Gs.

. (1) Transition to FR-H.1, "Response to Secondary Heat Sink".
(2) Continue feeding ALL S/Gs at 50 gpm.

D. (1) Transition to FR-H.1, "Response to Secondary Heat Sink".
(2) Isolate feed flow to "1B" AND "1C" S/Gs.

Answer: A

Explanation/Justification:

A. Correct. The SRO must evaluate plant conditions based on instrument/parameter interpretation and then make operational judgment to select the correct procedural course of action. The SRO must have knowledge of ECA-2.1 content. In the set of conditions provided, although only the "A" MSIV went shut, all S/G pressures continue to drop which is indicative of a beyond design basis accident (3 faulted S/Gs). Realistically for this scenario to happen all three S/Gs would need to have partially stuck open safety valves on all three S/Gs. In accordance with ECA-2.1, a minimum of 50 gpm must be maintained to each S/G with a narrow range level < 31%. There is also a note that states FR-H.1 should be implemented only if a total feed flow capability of 370 gpm is not available at any time while in ECA-2.1.

B. Incorrect. Correct procedure. Incorrect procedural action of how AFW should be addressed. Isolating feedwater flow to 1B and 1C would violate ECA-2.1 procedural actions. Candidate might believe that since "A" MSIV closure occurred feedwater isolation to the other two S/Gs is required.

C. Incorrect. Incorrect procedural transition. Correct action in accordance with ECA-2.1 versus FR-H.1.

D. Incorrect. Incorrect procedural transition. Plausible that transition to FR-H.1 should occur due to < 31% NR and less than 370 gpm. Incorrect action, however plausible since there is a preemptive action to isolate feedwater flow to a faulted S/G.

Sys #	System	Category	KA Statement
W/E12	Uncontrolled Depressurization of all Steam Generators	N/A	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

K/A# 2.1.7 **K/A Importance** 4.7 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.ECA-2.1, Issue 1C, Rev. 12, pg. 3
1OM-53A.1.FR-H.1, Issue 1C, Rev.13, pg. 2

Question Source: Vision # 82081 (2LOT7 NRC Q#81)

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(5)

Objective: 3SQS-53.3 4. Explain from memory the basis of all cautions and notes, IAW BVPS-EOP Executive Volume
6. Given a set of conditions, locate and apply the proper EOP IAW BVPS-EOP Executive Volume.

(SRO ONLY)

Beaver Valley Unit 1 NRC Written Exam (1LOT8)

. Given the following plant conditions and sequence of events:

- The Unit is at 75% power with all systems in NSA for this power level.
- The Control Room team notes that control rods are withdrawing at maximum speed.
- The RO places rods in MANUAL and reports control rods have stopped moving.
- The following annunciators are LIT and confirmed VALID:

1. A5-3, NIS 2/3 PWR RANGE NEUTRON FLUX RATE HIGH REACTOR TRIP
2. A5-4, 2/3 LOOPS OVERTEMP ΔT REACTOR TRIP

- The RO reports that Reactor Trip Breakers are CLOSED.
- Reactor Power is 80% and STABLE.

Which ONE of the following identifies (1) the procedural required actions to mitigate this event, **AND** (2) includes the action required if the crew is unable to verify emergency boration flow > 30 gpm in accordance with FR-S.1, "Response to Nuclear Power Generation/ATWS"?

A. (1) Enter FR-S.1 directly, manually trip the turbine, and insert control rods.
(2) Manually align Charging Pump suction to RWST.

B. (1) Attempt to manually trip the reactor in E-0, if unsuccessful, then transition to FR-S.1.
(2) Manually align Charging Pump suction to RWST.

(1) Enter FR-S.1 directly, manually trip the turbine, and insert control rods.
(2) Initiate boration > 30 gpm with the blender in BORATE mode.

D. (1) Attempt to manually trip the reactor in E-0, if unsuccessful, then transition to FR-S.1.
(2) Initiate boration > 30 gpm with the blender in BORATE mode.

Answer: B

Explanation/Justification:

A. Incorrect. Although both E-0 and FR-S.1 both have immediate actions with similar intentions (ie: trip the reactor/turbine), FR-S.1 by WOG rules of usage is not a direct entry procedure such as E-0 and ECA-0.0. Knowledge of administrative procedures that specify hierarchy implementation is SRO level knowledge. The second part of the distractor is correct (explained in correct answer)

B. Correct. The SRO must interpret the annunciators and indications provided and deduce that entry conditions of E-0 are applicable and by WOG rules of usage E-0 must be entered. FR-S.1 is entered only from E-0 or from a CSFST. The SRO must also understand FR-S.1 procedural content of sufficient detail to understand what alternative actions are required to ensure the reactor is safely shutdown as directed by FR-S.1. Step 3.e RNO requires the charging pumps aligned to the RWST if 30 gpm emergency boration cannot be established. This is to ensure adequate negative reactivity insertion.

C. Incorrect. Incorrect procedure as explained above. Incorrect action. This action is plausible since both ES-0.1 and FR-S.1 specify a 30 gpm boration as an action to insert negative reactivity. This action however procedurally specifies opening MOV-1CH-350 and starting a boric acid transfer pump as opposed to using the blender in the borate mode at the same rate.

D. Incorrect. Correct procedure. Incorrect action (refer to previous explanations).

Sys #	System	Category	KA Statement
001	Continuous Rod Withdrawal	N/A	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:** 1/2OM-53B.2, Issue 1C, Rev. 7, Pg. 12 & 13
1OM-53A.1.FR-S.1, Issue 1C, Rev. 5, Pg. 2 & 3
1OM-53B.4.FR-S.1, Issue 1C, Rev. 5, Pg. 60
1OM-2.4.ABC, Rev. 2, Pg. 2
1OM-1.4.ABJ, Issue 4, Rev. 0, Pg. 1

Question Source: Bank – (NAPS 2010 NRC Q#76)

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(5)

Objective: 3SQS-53.3 14. Apply the actions for a rod position malfunction.

Given the following plant conditions:

- The Unit experienced a MANUAL Reactor Trip during a reactor startup due to Hi-Hi S/G Water Level in "C" Steam Generator.
- PRZR pressure dropped to 1910 psig and is RISING.
- Immediate Manual Actions of E-0, "Reactor Trip or Safety Injection" are complete.
- The RO reports TWO control rods are indicating 225 steps.
- All systems function as designed.

What procedure action is required to ensure adequate shutdown margin, if any?

A. No action required. A sufficient amount of boron is being added to the core.

B. No action required. Plant design allows for two control rods to remain stuck out.

C. Normal boration will be performed during performance of ES-0.1, "Reactor Trip Response".

D. Emergency boration will be performed during performance of ES-0.1, "Reactor Trip Response".

Answer: D

Explanation/Justification:

A. Incorrect. Incorrect that no action is required, however, plausible if the SRO makes the decision at Step 4 of E-0 that SI occurred. The design of ECCS is to provide core cooling and boration to maintain a safe reactor shutdown condition. If SI had occurred this would be a correct answer.

B. Incorrect. Incorrect that no action is required and also incorrect since the plant is analyzed for one stuck control rod only, not two. Plausible if the candidate believes SI occurred and the decision to remain in E-0 at step 4 is made and that they have the plant design confused.

C. Incorrect. Correct procedural transition and assessment of plant conditions that no SI has occurred, however, ES-0.1 directs an emergency versus normal boration which is plausible if the SRO does not know procedural content.

D. Correct. The SRO candidate must assess given plant conditions and determine that no SI injection occurred based on stated plant conditions. Therefore the SRO shall make the decision at Step 4 of E-0 to transition to ES-0.1. Step 7 of E-0 verifies all control rods inserted and if directs emergency boration if two or more are not fully inserted. The SRO must know the content of procedures beyond immediate actions.

Sys #	**System**	**Category**	**KA Statement**
005	Inoperable/Stuck Control Rod	Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod:	Required actions if more than one rod is stuck or inoperable.

K/A# AA2.03 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53B.4.E-0, Issue 1C, Rev. 11, pg. 1
1OM-53.A.1.E-0, Issue 1C, Rev. 11, pg. 4
1OM-53A.1.ES-0.1, Issue 1C, Rev. 8, pg. 5
1OM-53B.4.ES-0.1, Issue 1C, Rev. 8, pg. 13

Question Source: Bank – Vision 40313

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(5)

Objective: 3SQS-53.4 2. Apply from memory all of the EOP users guide rules of usage as defined in 1/2 OM-53B.2.
3SQS-53.3 3. Given a set of conditions, locate and apply the proper EOP, IAW BVPS-EOP Executive Volume.
3. State from memory the basis and sequence for the major action steps of each EOP procedure, IAW BVPS-EOP Executive Volume.

,. Given the following plant conditions:

- Unit 1 core off-load is in progress.
- The Control Room receives a report that cable separation has occurred on the upender containing an irradiated fuel assembly from the vertical position.
- The RO reports that [RIS-1VS-103A/B], "Fuel Building Ventilation Exhaust radiation levels are rising and Hi alarms are validated.
- The Control Room has received A4-71, "RADIATION MONITORING HIGH" **ONLY**.
- No other alarms are present.

Have entry conditions been met for the SRO to perform AOP 1.49.1, "Irradiated Fuel Damage" actions **AND** using the Emergency Plan Procedure provided, does an ALERT classification exist for the present plant conditions? (Excluding ED Judgment) **(Reference Provided)**

AOP 1.49.1 entry conditions ____ (1) ____ been met.
An ALERT classification ____ (2) ____ exist for the stated plant conditions.

A. (1) have
(2) does

" (1) have
(2) does **NOT**

C. (1) have **NOT**
(2) does

D. (1) have **NOT**
(2) does **NOT**

Answer: B

Explanation/Justification:

A. Incorrect. Correct AOP application. Incorrect that an ALERT classification exists. The SRO must have good attention to detail to rule out that both 1 and 2 do NOT exist. Although 2 is met, the 1 section needs to have a Hi-Hi rad level on 103A/B before the classification criteria is met. Based on present plant conditions, 103A/B have a Hi versus Hi-Hi alarm.

B. Correct. Symptoms or entry conditions for AOP-1.49.1 have occurred which confirms the occurrence of a fuel handling incident. The control room has received a report of possible irradiated fuel damage and A4-71 is in alarm. This knowledge in itself can be considered RO knowledge but is necessary to meet the K/A. The additional E-Plan application is SRO required knowledge.

C. Incorrect. Incorrect AOP application. Incorrect E-Plan application

D. Incorrect. Incorrect AOP application. Correct E-Plan application

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Sys #	System	Category	KA Statement
036	Fuel Handling Accident	Ability to determine or interpret the following as they apply to the Fuel Handling Incidents:	Occurrence of a fuel handling incident.

K/A# AA2.02 **K/A Importance** 4.1 **Exam Level** SRO

'erences provided to Candidate EPP-I-1A BV1 Entire Classification Section **Technical References:** 1OM-53C.4.1.49.1, Rev. 8, pg. 1; EPP-I-1A, BV1, Rev. 13, pg. 42, 43, 46, 52

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(7)

(SRO ONLY)
Beaver Valley Unit 1 NRC Written Exam (1LOT8)

J. Given the following plant conditions:

- The Unit has experienced a beyond design bases LOCA.
- The Control Room Team is performing FR-C.1, "Response to Inadequate Core Cooling".
- Attempts to establish safety injection are unsuccessful and S/G depressurization was ineffective.
- All Reactor Coolant Pumps (RCPs) are secured & seal injection cannot be established.
- Five Hottest Core Exit Thermocouples are 1225 °F and slowly RISING.
- **ALL** Narrow Range S/G Water Levels are 20% and slowly DROPPING.
- Containment Pressure is 9 psig and slowly DROPPING.

Based on these conditions, which ONE of the following describes whether a RCP will be started according to FR-C.1 **AND** the bases for this action?

A. Start an RCP to restore long term core cooling.

B. Do **NOT** start an RCP because seal injection is required.

C. Do **NOT** start an RCP because S/G tube failure could occur.

Start an RCP to permit circulation of hot gases from the core to the S/Gs.

Answer: C

Explanation/Justification:

A. Incorrect. Incorrect action, however, this is one of the ECCS design criteria. A correct bases is to temporarily restore core cooling.

B. Incorrect. Correct action with incorrect but plausible bases. Normal conditions are desired, however, not required in FR-C.1, therefore seal injection is not required.

C. Correct. FR-C.1 does not allow an RCP start if adequate S/G water level does not exist (ie; 31% (50%)). The bases for this action is high S/G temperatures would occur leading to possible creep failure of the S/G U-Tubes. This is to protect the S/G tubes from creep rupture. This is an SRO only question because the SRO must assess plant conditions and determine the correct course of action for RCP restart. To meet the K/A, the SRO must also know the bases for the action taken (knowledge of procedure content).

D. Incorrect. Incorrect action, however, this is also a correct bases for starting an RCP.

Sys #	System	Category	KA Statement
W/E06	Inadequate Core Cooling	N/A	Knowledge of the specific bases for EOPs.

K/A# 2.4.18 **K/A Importance** 4.0 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.FR-C.1, Issue 1C, Rev. 8, Pg 12-13
1OM-53B.4.FR-C.1, Issue 1C, Rev 8, Pg. 34-36

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(5)

Objective: 3SQS-53.3 3. State from memory the basis and sequence for major action steps of each EOP procedure, IAW BVPS-EOP Executive Volume.
3SQS-53.2 2. State from memory the basis for RCP restart, IAW EOP Executive Volume.

(SRO ONLY)

Beaver Valley Unit 1 NRC Written Exam (1LOT8)

Given the following plant conditions:

- The Unit is in Mode 4 with BOTH "A" and "B" Residual Heat Removal (RHR) Rumps and Heat Exchangers (HX) in service.
- No Reactor Coolant Pumps (RCPs) are operating.
- All systems are in NSA for the current mode of operation.
- The Reactor Operator reports MOV-1RH-700, "RHR Inlet Isolation Valve", has drifted to the FULL CLOSED position and will **NOT** respond.

Which ONE of the following describes the specific impact on RHR, shortly after MOV-1RH-700 fails CLOSED?

RHR flow will ____ (1) ____
RHR Loops will ____ (3) ____ in accordance with T.S. 3.4.6, "RCS Loops - Mode 4".

A. (1) drop
(2) be OPERABLE

B. (1) remain the same
(2) be OPERABLE

C. (1) drop
(2) **NOT** be OPERABLE

D. (1) remain the same
(2) **NOT** be OPERABLE

Answer: C

Explanation/Justification:

A. Incorrect. Correct system response. Incorrect that the system is operable. (refer to correct answer explanation)

B. Incorrect. Incorrect system response. This reflects system response if MOV-1RH-758 failed open. Also reflects an accurate configuration if there were separate RHR loops, similar to Unit 2 design. Incorrect that the system is operable. (refer to correct answer explanation)

C. Correct. If MOV-1RH700 fails closed, system flow as indicated by FT-1RH-605 will drop significantly. Since no suction from the RCS loop to the RHR pump is available, the pump will continue to run and recirculate whatever water is left in the RHR loops. In accordance with the TS 3.4.6 bases an OPERABLE RHR loop comprises an operable RHR pump capable of providing forced flow to an operable RHR HX. RCPs and RHR pumps are operable if they are capable of being powered and are able to provide forced flow if required. Since MOV-1RH-700 is in the flowpath and not capable of being opened, RHR must be declared INOPERABLE. This question requires the SRO to make an operability determination based on system knowledge and TS bases. 1/2OM-48.1.I requires the SRO to make timely operability determinations in order to control and correct the non conforming condition.

D. Incorrect. Incorrect system response. Correct that the system is inoperable. (refer to correct answer explanation)

Sys #	System	Category	KA Statement
005	Residual Heat Removal System (RHRS)	Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	RHR valve malfunction

K/A# A2.04 **K/A Importance** 2.9 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-10.1.C, Issue 4, Rev. 0, Pg. 2 & 3
Op Manual RM-0410-001, Rev. 13
TS 3.4.6 Bases, Rev. 0, Pg. B 3.4.6-2
1/2OM-48.1.I, Rev. 26, Pg. 34

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR 43(b)(2)

Objective: 1SQS-10.1 18. Given a specific plant condition, predict the response of the RHR control room indications and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for an off-normal condition.

(SRO ONLY)
Beaver Valley Unit 1 NRC Written Exam (1LOT8)

. Which ONE of the following will be (1) the number of SI Accumulators required to inject to the core during the blowdown phase of a LOCA to ensure ECCS acceptance criteria of 10 CFR 50.46 are not violated **AND** (2) what is the Technical Specification bases for maintaining **minimum** SI accumulator boron concentration?

A. (1) TWO
(2) To ensure reactor subcriticality during post LOCA.

B. (1) THREE
(2) To ensure reactor subcriticality during post LOCA.

C. (1) TWO
(2) To determine cold leg to hot leg switchover time.

D. (1) THREE
(2) To determine cold leg to hot leg switchover time.

Answer: A

›lanation/Justification:

.. Correct. According to TS 3.5.1 Bases background/LCO, The accumulator size, water volume, and nitrogen cover pressure are selected so that two of the three accumulators are sufficient to partially cover the core before significant clad melting or zirconium water reaction can occur following a LOCA. One of the three accumulators is assumed to be lost via the RCS break during the blowdown phase of a LBLOCA. The LCO establishes minimum conditions required to ensure accumulators are available to accomplish their core cooling safety function following a LOCA. Three accumulators are required to ensure that 100% of the contents of two will reach the core. If less than two accumulators are injected during the blowdown phase of a LOCA, the ECCS acceptance criteria of 10 CFR 50.46 could be violated. The minimum SI accumulator Cb concentration bases is to assure reactor subcriticality in a post LOCA environment. This is an SRO level question because the candidate must have knowledge of TS bases and is beyond above the line knowledge.

B. Incorrect. Incorrect number of accumulators although plausible since the LCO is three accumulators and the candidate may conceive that all three are required to ensure 10 CFR 50.46 criteria is maintained. Correct minimum Cb bases.

C. Incorrect. Correct number of accumulators. Incorrect minimum Cb bases. This is the bases for the maximum Cb allowed.

D. Incorrect. Incorrect number of accumulators. Incorrect minimum Cb bases.

Sys #	**System**	**Category**	**KA Statement**
006	Emergency Core Cooling	N/A	Knowledge of limiting conditions for operations and safety limits.

K/A# 2.2.22 **K/A Importance** 4.7 **Exam Level** SRO

References provided to Candidate None **Technical References:** BVPS Unit 1 & 2 TS 3.5.1, Pgs. B 3.5.1-1-7

Question Source: New

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** CFR: 43 (b)(2)

Objective: 1SQS-11.1 27. Describe the design basis for the Safety Injection System and the associated major components as documented in the USFAR.

Given the following plant conditions and sequence of events:

- The Unit is operating at 100% power.
- The crew is performing 1OM-15.4.H, "Securing a CCR Pump or Placing The Spare CCR Pump In Service", with the following system status:
  - [1CC-P-1B], Component Cooling Pump is running.
  - [1CC-P-1C], Component Cooling Pump is racked onto the "1AE" 4KV Bus in Standby.
  - [1CC-P-1A], Component Cooling Pump is racked in with the control switch in PTL.
- A Loss of Offsite Power coincident with a reactor trip occurs **and** all components function as designed except #2 Emergency Diesel Generator fails to start.

Which ONE of the following describes the status of CCR system pressure **AND** Technical Specification (TS) requirement? (assume **NO** operator action occurred) **(Reference Provided)**

TWO (2) minutes following the Loss of Offsite Power, [A6-35], "PRI COMP COOL PUMP DISCH PRESS LOW" will ___ (1) ___. TS 3.0.3 LCO entry will ___ (2) ___.

A. (1) be LIT
(2) be required

B. (1) **NOT** be LIT
(2) be required

(1) be LIT
(2) **NOT** be required

D. (1) **NOT** be LIT
(2) **NOT** be required

Answer: C

Explanation/Justification:

A. Incorrect. Correct A6-35 status (refer to correct answer explanation). Incorrect that TS 3.0.3 action is required. Plausible because based on stated conditions, there are no CCW pumps running. TS 3.7.7 action for two CCW pumps inoperable is to immediately restore which is applicable in Mode 4 when RHR is required to C/D plant based on Note. The plant is currently in Mode 3 based on reactor trip. This note could be misapplied to mean that since two trains are not available in Mode 3 that TS 3.0.3 is required.

B. Incorrect. A6-35 is LIT based on no CCR pumps running. Plausible if candidate does not understand that EDG auto loading if two CCR pumps are racked onto the emergency bus. Incorrect plausible TS LCO action (explained above).

C. Correct. A6-35 will be LIT. With no offsite power, the only source of power to emergency busses is from the EDG. Since #2 EDG did not start, the B CCR pump is not running. Since there are two pumps racked onto the AE bus, #1 EDG will not auto sequence on "C" CCR pump and is rendered inoperable. This knowledge can be considered RO higher cognitive knowledge but is required to meet the K/A statement. TS application/Bases is SRO knowledge and meets the second part of the K/A. TS bases states that each CCW train is considered operable if it can be started manually. If "A" CCW pump is taken out of PTL the pump will start, therefore LCO 3.0.3 is not required.

D. Incorrect. Incorrect A6-35 status (explained above). Correct LCO action (explained above).

Sys #	System	Category	KA Statement
008	Component Cooling Water	Ability to (a) predict 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:	Loss of CCW Pump

K/A# A2.01 **K/A Importance** 3.6 **Exam Level** SRO

References provided to Candidate BVPS TS 3.03/3.7.7 & Bases

Technical References: 1OM-15.4.H, Rev. 12, Pg. 2
BVPS TS 3.03/3.7.7 & Bases
1SQS-15.1, Rev. 11 PPNT Slides

.estion Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(2)

Objective: 1SQS-15.21 Describe the CCR pump start/stop logic and control room indications that are inputs to the logic.
1SQS-15.22 Describe the design bases for the CCR system.

(SRO ONLY)

Beaver Valley Unit 1 NRC Written Exam (1LOT8)

Given the following plant conditions:

- The Unit is operating at 55% power.
- The RO reports multiple annunciators received related to the Main Feedwater System.
- The following plant indications exist:
  - All STEAM GENERATOR LEVEL DEVIATION annunciators – LIT.
  - All S/G Narrow Range Levels are 50% and SLOWLY DROPPING.
  - "A" Condensate Pump RED Light – NOT LIT and GREEN Light – LIT
  - "B" Condensate Pump RED Light – LIT and GREEN Light – NOT LIT.
  - "A" Main Feedwater Pump – is operating.

What procedure entry and actions are required for the **CURRENT** plant conditions? (AOP -1.24.1, "Loss of Main Feedwater" / E-0, "Reactor Trip or Safety Injection")

A. Enter AOP 1.24.1, Trip the reactor and concurrently perform E-0 actions.

B. Enter AOP 1.24.1, Reduce reactor power to < 50% and continue AOP actions.

C. Trip the reactor, Enter E-0 and do **NOT** perform AOP 1.24.1 actions concurrently.

D. Enter AOP 1.24.1, Reduce power < P-9, then trip the turbine and continue AOP actions.

Answer: B

Explanation/Justification:

A. Incorrect. Correct that AOP 1.24.1 entry conditions are met, however, no reactor trip criteria have been met. The candidate must have detailed AOP content knowledge beyond the immediate actions to deduce this is an incorrect choice.

B. Correct. A trip of an operating condensate pump has occurred. This causes lowering S/G water levels since the capacity of feedwater to the S/Gs is slightly beyond limits for the existing power level. The SRO must have detailed knowledge of ARP and AOP procedural content beyond that of IMA's to correctly deduce E-0 action is not required. The SRO must also have knowledge of administrative procedures that specify implementation of plant normal/emergency procedures as implied by K/A statement. Specific to BVPS, AOP 1.24.1 is not used concurrently once the EOP is entered, rather it is used to pre-empt EOP entry. Therefore the question and distractors are focused on meeting K/A intent and testing SRO knowledge of procedural content , decision making, and BVPS specifics of how our AOP is used in conjunction with E-0.

C. Incorrect. The candidate must assess plant conditions and deduce that no trip criteria has been met. Plausible if the candidate thinks reactor trip criteria is met. There are no concurrent AOP actions once the reactor is tripped.

D. Incorrect. Correct that AOP 1.24.1 is entered. Also correct to reduce turbine load to below 50% which is close to P-9 @ 49%, however, the AOP does not require the turbine tripped which is a plausible action for these plant conditions.

Sys #	**System**	**Category**		**KA Statement**
059	Main Feedwater	N/A		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:**	1/2OM-48.2.C, Rev. 18, pg. 11 1OM-53C.4.1.24.1, Rev. 7, pg 1-5

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(5)

Objective: 3SQS-48.1 9. From memory explain the requirements of adherence to and familiarization with operating procedures.
20. From memory, explain all of the Operations Expectations.

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

- Unit 1 is at Full Power with all systems in NSA.
- RM-1VS-107B, "Reactor Building and SLCRS Vent Release Particulate Gas Monitor" power supply fails.
- The US declares this activity monitor non-functional and consults ½-ODC-3.03, "ODCM: Controls for RETS and REMP Programs" to determine the impact of this failure, if any.
- Assume all other Radiation Monitors are functional.

Which ONE of the following actions according to ½-ODC-3.03 will be taken based on the failed power supply, if any? **(Reference Provided)**

A. Restore non-functional channel within 72 hours.

B. No action is necessary, elevated discharge can continue.

C. Immediately suspend the release of radioactive gaseous effluents.

D. Take grab samples at least every 12 hours and analyze for gross activity.

Answer: B

Explanation/Justification:

A. Incorrect. This is a plausible alternative if the candidate refers to Table 3.3-6 and incorrectly applies action 35.

B. Correct. RM-1VS-107B is referenced several places in the ODCM. Table 3.3-6 1a makes reference to this monitor as a 2nd PMM, so there is no impact on the elevated release because the Primary Instrument is still operable. Table 4.3-3 shows the required surveillance for this instrument. Table 3.3-13 3a requires RM-1VS-107B or RM-1VS-110 ch 5 as the alternate. Since 1 is the minimum channels operable, Action 29 does not need to be performed. Action 30a is N/A since a purge from containment is not in progress. Note that the candidate must understand that in NSA an elevated release from Safeguards is in progress.

C. Incorrect. Plausible if the candidate misunderstands or incorrectly applies the ODCM, but not correct.

D. Incorrect. This is a plausible alternative if there were one less than the minimum required channels. Only one channel is required to be operable per Table 3.3-13, 3a.

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

K/A# A2.01 **K/A Importance** 2.9 **Exam Level** SRO

References provided to Candidate ½-ODC-3.03, Rev. 10 **Technical References:** ½-ODC-3.03, Rev. 10, pg. 20, 24, 37, 39, 43, 46; 1SQS-43.1, Rev. 13, PPNT Slide # 2

Question Source: Modified Bank – Vision # 598

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(2)&(4)

Objective: 1SQS-43.1 12. Using a copy of the Technical Specifications, Licensing Requirements Manual, or Offsite Dose Calculation Manual, analyze a given set of plant conditions for compliance with the licensing requirements, including determination of the equipment operability and applicable action statement.

(SRO ONLY)

Beaver Valley Unit 1 NRC Written Exam (1LOT8)

Given the following plant conditions and sequence of events:

- A Reactor trip coincident with a Loss of Offsite Power occurs.
- Both Trains of RVLIS are **NOT** functioning, but all other systems functioned as designed.
- The Control Room Team is performing ES-0.4, "Natural Circulation Cooldown With Steam Void in Vessel (Without RVLIS)".
- RCS Hot leg temperatures are 450 °F and STABLE.
- RCS Pressure is 1600 psig and STABLE.
- Charging and letdown are placed in MANUAL.
- **During** depressurization to 800 psig, PRZR Level rapidly rises to 90%.
- PRZR Heaters are ALL unavailable.

Which ONE of the following describes what required action will be necessary to mitigate the rising PRZR level condition **AND** impact of PRZR Heaters being unavailable?

The required action in accordance with ES-0.4 to mitigate this condition is to ____ (1) ____.
The impact of PRZR heater unavailability is the **inability** to ____ (2) ____.

A. (1) stop the depressurization
(2) partially or wholly collapse the Rx Vessel Void.

(1) maximize letdown
(2) prevent a water solid RCS and resultant loss of pressure control.

C. (1) maximize letdown
(2) partially or wholly collapse the Rx Vessel Void.

D. (1) stop the depressurization
(2) prevent a water solid RCS and resultant loss of pressure control.

Answer: A

A. Correct. Step 8 of ES-0.4 directs depressurization stopped if PRZR level exceeds 90% or if 800 psig is reached. PRZR heaters are necessary in step 9 to raise RCS pressure 100 psig to collapse the reactor vessel void formation. This is SRO level knowledge based on the need to assess plant conditions and understand how to proceed based on understanding of procedure content. In this situation there is no method to continue in this procedure without PRZR heater control. A procedural do-loop occurs. PRZR heaters are necessary in ES-0.4 for PRZR level control to enhance upper head cooling.

B. Incorrect. This is a required action and bases in ES-0.3 for high PRZR level where the technique employed for RCS cooldown is dramatically different. Under normal circumstances, with the RCS subcooled, this would be correct. However, with a void in the vessel the RCS will not be water solid. Candidate may mis-interpret high PRZR level as an indication of a "water solid" RCS.

C. Incorrect. Incorrect action. Correct impact.

D. Incorrect. Correct action. Incorrect impact although partially correct that a loss of pressure control occurs. (refer to distractor B explanation)

Sys #	System	Category	KA Statement
011	Pressurizer Level Control	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:	Loss of PZR heaters.

K/A# A2.05 **K/A Importance** 3.7 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53A.1.ES-0.4, Issue 1C, Rev. 10, pg. 5 & 6
1OM-53B.4.ES-0.4, Issue 1C, Rev. 10, pg. 18, 20 - 23

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(5)

Objective: 3SQS-53.3 3. State from memory the basis and sequence for major action steps of each EOP procedure IAW BVPS EOP

(SRO ONLY)

Beaver Valley Unit 1 NRC Written Exam (1LOT8)

The Unit is operating at 40% power with all systems in NSA for this power level and the following plant conditions occur:

- A4-76, COMPUTER ALARM ROD DEVIATION/SEQ NIS POWER RANGE TILTS is LIT.
- Individual Rod Position Indication (IRPI) for Control Rod D4 (Bank C) indicates 30 steps.
- Reactor power indicates 34% and is SLOWLY DROPPING.
- RCS Tavg indicates 540°F and is SLOWLY DROPPING.
- The SRO enters AOP 1.1.8, "Rod Inoperability".
- IRPI system functions as designed.

What will be the status of VB-B Control Rod D4 Rod Bottom Light **AND** what action is required?

The VB-B Control Rod D4 Rod Bottom Light will ____ (1) ____.
The required action will be to direct a ____ (2) ____

A. (1) be LIT
(2) turbine load reduction.

B. (1) be LIT
(2) reactor trip.

(1) **NOT** be LIT
(2) turbine load reduction.

D. (1) **NOT** be LIT
(2) reactor trip.

Answer: D

Explanation/Justification:

A. Incorrect. Incorrect RPIS impact. Rod Bottom Light setpoint is < 20 steps. Incorrect procedure use although plausible. A load reduction is warranted and procedurally driven in AOP 1.1.8 to stabilize RCS Tavg within 4 F of Tref.

B. Incorrect. Incorrect RPIS impact. Rod Bottom Light setpoint is < 20 steps. Correct procedural action (refer to correct answer explanation).

C. Incorrect. Correct RPIS impact. Rod Bottom Light setpoint is <20 steps. Incorrect procedural action as discussed above.

D. Correct. The impact of a dropped rod on RPIS is the receipt of a Rod Bottom Light which corresponds to an IRPI position of < 20 steps. Based on the dropped rod and subsequent plant conditions provided (ie: Tavg < 541F) AOP 1.1.8 directs a reactor trip if Tavg is on a declining trend and greater than 10 F below Tref. Tref for 34% is about 557 F which is well outside of the required band. Since these conditions are met, a reactor trip is warranted. The reason for this trip is to comply with TS 3.4.2 which allows 30 minutes to be in Mode 2 with Keff < 1.0. It is required SRO knowledge to understand the content of procedures beyond IMA's. Also at BVPS, there have been some misconceptions in the past that operation below 541F is permitted as long as temperature is restored within 30 minutes which is NOT IAW TS. This relevant BVPS OE.

Sys #	System	Category	KA Statement
014	Rod Position Indication	Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Dropped Rod

K/A# A2.03 **K/A Importance** 4.1 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1SQS-1.4, Rev. 12, PPNT slides
1SQS-1.4 LP, Rev. 12, pg. 4-6, 26-27
1OM-1.4.ABH, Rev. 5, pg. 2
1OM-53C.4.1.1.8, Rev. 3, pg. 1-2
BVPS TS 3.4.2, Amend 278/161

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(5)

Objective: 1SQS-1.4
9. Discuss the IRPI indications following reactor trip, dropped rod, misaligned rod, or loss of electrical power.
13. Given a set of plant conditions, analyze the ARPI system indications to determine specific failure has occurred.

Given the following plant conditions:

- The plant is in Mode 6.
- Fuel Off-Load is in progress.
- A Fuel Assembly inside the manipulator crane mast is being moved away from the reactor vessel toward the Upender.
- A6-30, "REFUELING CAVITY LEVEL LOW" is received.
- A cavity seal ring failure is reported as the cause of this alarm.

As Refueling SRO, which ONE of the following is the safe position for the fuel assembly in transit according to the Alarm Response Procedure?

The fuel assembly in transit will ____________ prior to evacuating containment.

A. be placed in an open area inside the core.

B. remain inside the mast of the manipulator crane.

C. be placed in an upright position in the lifting frame of the upender.

D. be placed in a horizontal position in the lifting frame of the upender.

Answer: A

Explanation/Justification:

A. Correct. Based on the ARP response this is the safest location. There will be more water keeping the assembly covered longer as this is the lowest point in the refueling cavity. It is an SRO function to provide immediate emergency guidance to those situations involving refueling. This is an SRO question based on refueling floor responsibilities.
B. Incorrect. Plausible but incorrect. Suspended from the crane is not the safest location. It is the quickest option to evacuate from containment.
C. Incorrect. Plausible but incorrect. It is safer to place the fuel assembly back into the core leaving it horizontal versus in the upender.
D. Incorrect. Plausible but incorrect. This is the safest location if the assembly is already inside the upender.

Sys #	System	Category	KA Statement
034	Fuel Handling Equipment	N/A	Knowledge of low power/shutdown implications in accidents (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

K/A# 2.4.9 **K/A Importance** 4.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-20.4.AAP, Rev. 16, pg. 2 – 4
1/2RP-1.1, Issue 0, Rev. 22, pg. 8 & 9

Question Source: Bank – Vision # 33264

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** CFR: 43 (b)(7)

Objective: 3SQS-6.13 Given a Fuel Handling System alarm condition and using the ARP(s), determine the appropriate response, including automatic action and operator actions in the control room.

(SRO ONLY)
Beaver Valley Unit 1 NRC Written Exam (1LOT8)

. Given the following plant conditions:

- The Unit is operating at 5% power with all systems in NSA, during a plant shutdown.
- A4-4, PRESSURIZER CONTROL LOW LEVEL is received.
- LT-1RC-459 is reading 16% and is slowly DROPPING.
- LT-1RC-460 is reading 23% and is slowly RISING.
- LT-1RC-461 is reading 23% and is slowly RISING.
- Charging flow is 100 GPM and slowly INCREASING.
- Charging Flow Controller output is DECREASING.

If NO operator action occurs, will a reactor trip occur?
What is the Technical Specification (TS) bases for Pressurizer (PRZR) Level Reactor trip?

A PRZR Level Reactor Trip will _____ (1) _____.
According to TS 3.3.1 Basis, a PRZR Level Reactor Trip is provided as a backup for the PRZR Pressure _____ (2) _____ Reactor Trip.

A. (1) occur
(2) LOW

B. (1) occur
(2) HIGH

. (1) **NOT** occur.
(2) LOW

D. (1) **NOT** occur.
(2) HIGH

Answer: D

Explanation/Justification:

A. Incorrect. Incorrect Reactor trip will NOT occur. Incorrect Bases, although plausible because PRZR does go low before L/D isolates. There is no Low Level PRZR Level Trip, although a common misconception. Lowering level equates to lower pressure. (Refer to correct answer explanation.).

B. Incorrect. Incorrect Reactor trip will NOT occur. Plausible because there is a valid Hi PRZR level reactor trip which would occur if > P-7 (10% power). Correct TS Bases. (Refer to correct answer explanation).

C. Incorrect. Correct Reactor trip will NOT occur. Incorrect that TS bases. (Refer to correct answer explanation).

D. Correct. The SRO candidate must evaluate PRZR level instrumentation and evaluate plant performance to correctly deduce that LT-1RC-459 is failing low. The automatic PRZR level control system is properly responding to this failure by increasing charging flow as the controlling channel drifts lower. This results in increasing PRZR level on the other two properly indicating channels. When the failing PRZR channel LT-1RC-459 reaches 14% PRZR level, automatic letdown isolation will occur. With no operator action the PRZR will continue to fill to the high level setpoint. Because the reactor is at 5% (< P-7), a reactor trip on High PRZR level (92%) will NOT occur. There is no low PRZR Level reactor trip. The basis for the High PRZR Level trip is to provide a backup for the PRZR High Pressure trip. The SRO candidate must evaluate plant conditions and must possess the knowledge of TS bases beyond what is required of an RO. The SRO must apply operator fundamentals per conduct of operations.

Sys #	System	Category	KA Statement
N/A	N/A	Conduct of Operations	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

K/A# 2.1.7 **K/A Importance** 4.7 **Exam Level** SRO

'erences provided to ıdidate None **Technical References:** 1OM-6.4.ABR, Rev. 5, pg. 2, 1OM-6.4.IF, Rev. 11, pg. 12
TS 3.3.1, Amend. 278/161, pg. 13, TS Bases B 3.3.1, Rev. 0, Pg B 3.3.1-20

Question Source: Modified Bank – Vision # 1420/82082 (2LOT7 NRC Q#82)

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR 43 (b)(2)

Objective: 1SQS-6.4 20. Given a change in plant conditions due to system or component failure, analyze to determine what occurred.

,. Given the following plant conditions:

- The plant is operating in Mode 6 with all systems in normal alignment for this Mode.
- Core Off-Loading activities are in progress and core off-load is half complete.
- Source Range Channel N31 fails LOW.
- Source Range Channel N32 remains OPERABLE.

Which ONE of the following activities can be performed **WITHOUT** violating the Technical Specification required actions for Source Range Instrumentation?

A. Install a temporary secondary source into a core location.

B. Move a spent fuel assembly from the core to the Spent Fuel Pool.

C. Move a spent fuel assembly from the upender to the Spent Fuel Pool.

D. Add Hydrogen Peroxide mixed with primary grade water to the refueling cavity for cleanup.

Answer: C

Explanation/Justification:

Incorrect. Plausible that operationally an alternative source could be installed to replace N31, however, it is not allowed by definition.

. Incorrect. Moving a fuel assembly from the core would not be allowable by definition. Removing the assembly would not be considered placing the assembly in a safe location. This is plausible however, the safest location would be back into the core as opposed to removal from the core. This is further plausible because some TS such as TS 3.9.4 LCO preclude core onload but do allow core offload to continue.

C. Correct. Both AOP 1.2.1A for SR Channel Malfunction and TS 3.9.2 direct that core alterations are immediately suspended. Core alterations are defined as movement of any fuel, sources, or reactivity components, within the reactor vessel with the vessel head removed and with fuel in the vessel. The SRO must have knowledge of the administrative requirements associated with refueling activities and have knowledge of TS bases. In order to answer this question the SRO must know the definition of Core Alterations and be able to apply this definition to a set of plant conditions. The movement of a Spent Fuel Assembly from the upender to the SFP is allowable because it is not within the reactor vessel.

D. Incorrect. Plausible that hydrogen peroxide is added to the water for clarity and cleanliness. However, the addition of primary grade water into the RCS would violate the second part of TS 3.9.2 since primary grade water could reduce boron concentration and is not allowed.

Sys #	System	Category	KA Statement
N/A	N/A	Conduct of Operations	Knowledge of procedures and limitations involved in core alterations.

K/A# 2.1.36 **K/A Importance** 4.1 **Exam Level** SRO

References provided to Candidate None **Technical References:** 1OM-53C.4.1.2.1A, Rev. 6, Pg. 1, 2, & 6
BVPS TS Definitions, Amend 278/161, Pg. 1.1-2
BVPS TS 3.9.2, Amend 278/161, Pg. 3.9.2-1
BVPS TS B3.9.2, Rev. 0, Pg. B3.9.2-1 &2
1/2RP-1.1, Issue 0, Rev. 22, Pg. 5

Question Source: Modified Bank – Vision # 81944 (2LOT6 NRC Q#94)

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(6)

Objective: 1SQS-20.1 14. Given a copy of TS or LRM, analyze a given set of plan conditions for compliance with licensing requirements, including the determination of equipment operability and applicable action statements.

(SRO ONLY)

Beaver Valley Unit 1 NRC Written Exam (1LOT8)

/. Given the following plant conditions:

- The Unit is operating at 100%.
- You have just returned from a day off and are reviewing the narrative logs.
- 36 hours ago, a valve was repositioned out of NSA and selected as an OPEN item using the Short Term Configuration Change Process.

Based on the requirements of NOP-OP-1014, "Plant Status Control", does this comply with the Short Term Configuration Change Process?

A. No; a clearance should have been posted 12 hours ago.

B. No; a system status print sheet should have been issued 12 hours ago.

C. Yes; a clearance will only be necessary if restoration does not occur within 12 hours.

D. Yes; a system status print sheet will be necessary if restoration does not occur within 12 hours.

Answer: A

lanation/Justification:

.. Correct. According to NOP-OP-1014, if a component is not restored to its normal configuration within 24 hours, then a clearance is hung to provide a plant status control tracking method and documentation of the deviation from the components normal alignment. A clearance should have been posted 12 hours ago. The SRO is responsible for operating changes and configuration control in the facility.

B. Incorrect. Correct that it does not comply with the short term configuration control process. A System Status Print is required to be filled out at all times reflecting system status conditions, if the system is deemed necessary by the Ops Manager. Either way if not deemed necessary the system status print would not be required. If deemed necessary than it should have been filled out 36 hours ago.

C. Incorrect. Refer to correct answer explanation. The candidate may believe the requirement is 48 hours as opposed to 24 hours.

D. Incorrect. Refer to incorrect choice B explanation. Plausible and balanced distractor.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Equipment Control	Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tagouts, etc.

K/A# 2.2.15 **K/A Importance** 4.3 **Exam Level** SRO

References provided to Candidate None **Technical References:** NOP-OP-1014, Rev. 1, pg. 12-14; 1/2OM-48.3.D, Rev. 6, pg. 8

Question Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(3)

Objective: 3SQS-48.1 20. From memory, explain all of the Operations Expectations.

(SRO ONLY)

Beaver Valley Unit 1 NRC Written Exam (1LOT8)

. Given the following plant conditions:

- The plant is operating at 100% power.
- At 0830 on July 3rd, the (A) Quench Spray Pump (1QS-P-1A) is declared **INOPERABLE**.
- At 2300 on July 5th, the (B) Quench Spray Pump (1QS-P-1B) becomes **INOPERABLE.**
- At 0215 on July 6th, the (A) Quench Spray Pump (1QS-P-1A) is restored to **OPERABLE.**

Including any extensions that are permitted by Technical Specifications and using references provided, which ONE of the following describes the **LATEST** time and date to restore 1QS-P-1B to **OPERABLE** status, without requiring a unit shutdown? **(Reference Provided)**

A. 0830 on July 6th

B. 0830 on July 7th

C. 2300 on July 8th

D. 2300 on July 9th

Answer: B

Explanation/Justification:

A. Incorrect. Refer to correct answer explanation. This answer is plausible if the pumps were associated with the same train in which case the 24 hour extension time would not be applicable.

B. Correct. In accordance with Section 1 of TS (Use and application), when a subsequent train, subsystem, or component expressed in the condition is discovered inoperable or not within limits, the completion time may be extended provided two criteria are met: The subsequent inoperability must exist concurrent with the first inoperability and must remain inoperable or not within limits after the first inoperability is resolved. In this case the more limiting time must be used.

C. Incorrect. This time corresponds with 72 hours from second inoperability which is the less restrictive time and therefore cannot be used.

D. Incorrect. This time corresponds with the 72 hours from the second inoperability plus the 24 hour extension which is another improper application.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Equipment Control	Ability to apply technical specifications for a system.

K/A# 2.2.40 **K/A Importance** 4.7 **Exam Level** SRO

References provided to Candidate TS 1.3, Amend 278/161, pg.1-11)
TS 3.6.6, Amend 278/161 pg 1-2)

Technical References: TS 1.3, Amend 278/161
TS 3.6.6, Amend 278/161

Question Source: Bank – Vision 82088 (2LOT7 NRC Q#88)

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR 43(b)(2)

Objective: 1SQS-13.1 28. Using a copy of TS, analyze a given set of plant conditions for compliance with the licensing requirements; including the determination of equipment operability and applicable actions statements.

. The following radiological conditions exist for an area in the plant:

- General dose rate levels range from 25 – 45 mr/hr.
- A Non-Licensed Operator needs to enter this area to isolate a safety related system during Emergency Operating Procedure Implementation.
- Measurements taken on pipes and valves include:
  - Point 1 is 100 mr/hr at 30 cm.
  - Point 2 is 500 mr/hr at 30 cm.
  - Point 3 is 1100 mr/hr at 30 cm.

Based on these plant conditions, what is the radiological posting required **AND** which entry requirements are applicable according to NOP-OP-4101, "Access Controls for Radiologically Controlled Areas"?

(1) **Radiological posting required**
(2) **NOP-OP-4101 Entry Requirements**

A. (1) Very High Radiation Area.
(2) Shift Manager must grant access.

B. (1) Very High Radiation Area.
(2) Radiation Protection must grant access.

(1) Locked High Radiation Area.
(2) Shift Manager must grant access. Only one key will be required to gain access.

D. (1) Locked High Radiation Area.
(2) Radiation Protection must grant access. Two keys are required to gain access.

Answer: C

Explanation/Justification:

A. Incorrect. A Very High Radiation Area is defined as an accessible area in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 R/hr at a distance of 1 meter or more from a radiation source or any surface that radiation penetrates. The candidate could confuse 500 r/hr with mr/hr requirements. To the extent possible, entry into a VHRA should be forbidden unless there is a sound operational or safety reason for entering. Although entering to isolate a safety related system during EOPs does meet this criteria, it is not a VHRA. Both SM and RP permission is required to enter a VHRA.

B. Incorrect. Refer to discussion above.

C. Correct. A Locked High Radiation Area is defined as an accessible high radiation area in which radiation levels could result in an individual receiving a deep dose in excess of 1000 mr/hr at a distance of 30 centimeters or more from a radiation source or any surface that radiation penetrates. The conditions in the stem of the question meet this criteria. According to both TS 5.7 and NOP-OP-4101, RP permission to gain access to a LHRA during an emergency can be waived. Only one key is required to gain access and can be issued from the CR by the SM.

D. Incorrect. Correct that this is a LHRA. RP can normally provide permission to gain access, however, this is an emergency and their permission is not required, however, it is required that the SM grant access and issue the key of which there is only one versus two required to gain access.

Sys #	System	Category	KA Statement
N/A	N/A	Radiation Control	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:** NOP-OP-4101, Rev. 5, pg. 4 - 5, 10, 11, 14, 15, 20
TS 5.7 Amend. 278/161, pg. 5.7-1

Question Source: Bank – Vision 82097 (2LOT7 NRC Q#97)

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** CFR 43 (b)(4)

Objective: 3SSG-Admin 15. Explain the controls implemented by the Health Physics Program in accordance with: ½-ADM-1601, Radiation Protection Standards and NOP-OP-4102, Radiological Postings, Labeling, and Markings.

(SRO ONLY)
Beaver Valley Unit 1 NRC Written Exam (1LOT8)

Given the following plant conditions:

- The Control Room Team is performing E-3, "Steam Generator Tube Rupture".
- While verifying Station Instrument air available in Step 7, it is determined that Station Instrument Air (IA) has been lost.

Which ONE of the following describes required procedure usage regarding IA restoration?

(AOP-1.34.1 = "Loss of Station Instrument Air")
(EOP Att. 2-S = "Monitoring AFW Pump Performance During Loss of Station Instrument Air")

A. Stop at this step in E-3. Transition to AOP 1.34.1.
When IA is restored, return to Step 7 of E-3 and continue subsequent E-3 steps.

B. Stop at this step in E-3. Transition to AOP 1.34.1.
When IA is restored, return to Step 7 of E-3 and perform EOP Att. 2-S.

C. Continue in E-3 without performing AOP-1.34.1.
Attempt to restore IA using E-3 Step 7.
If IA cannot be restored, perform EOP Att. 2-S.

Continue in E-3 and concurrently perform AOP-1.34.1.
Attempt to restore IA using AOP 1.34.1.
When IA is restored, continue in E-3 and perform EOP Att. 2-S.

Answer: C

Explanation/Justification:

A. Incorrect. EOPs are higher priorities than AOPs. Although plausible that AOP 1.34.1 is used, Step 7 RNO does not refer to usage of AOP 1.34.1.

B. Incorrect. There is no direction to concurrently use AOP 1.34.1 during performance of E-3. EOP Att 2-S is only performed if IA is not restored.

C. Correct. The intent of E-3 is to get S/G pressure equalized with RCS as soon as possible to stop primary to secondary leakage. This can be accomplished without IA which is not required per design bases. Step 7 is a continuous action step and attempts to restore IA should be continued. The SRO needs to prioritize procedures and E-3 has the overriding priority and does not direct concurrent use of AOP 1.34.1. Step 7 RNO does direct performance of Attachment 2-S if IA is not restored because AFW recirculation valves fail closed and this could damage the AFW pump if there is insufficient flow. This is an important concept that the SRO must be aware of, otherwise AFW could be jeopardized. This is an SRO level question because it requires an assessment of Loss of IA impact on E-3 and knowledge of procedure content and diagnostic steps in E-3. The K/A is met because E-3 is a higher priority than AOP 1.34.1. The SRO must know this procedure is not used in conjunction with E-3.

D. Incorrect. Refer to correct answer explanation. There is no direction to concurrently use AOP 1.34.1 during performance of E-3. Step 7 does direct Attachment 2-S usage if IA is not restored, however, IA is restored so therefore is unnecessary to perform. The background document and step bases is more concerned with restoration of air to ensure AFW recirculation. EOP Attachment 2S is referenced if air header pressure cannot be restored.

Sys #	System	Category	KA Statement
N/A	N/A	Emergency Procedures/Plan	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.E-3, Issue 1C, Rev. 14, pg 9
1OM-53B.4.E-3, Issue 1C, Rev. 14, pg 64
1/2OM-53B.2, Issue 1C, Rev. 7, pg 3-7
1/2OM-48.2.C, Rev. 18, pg. 11

estion Source: New

Question Cognitive Level: Higher – Comprehension or Analysis **10 CFR Part 55 Content:** CFR: 43 (b)(5)

Objective: 3SQS-48.1 9. From memory explain the requirements of adherence to and familiarization with operating procedures.
20. From memory, explain all of the Operations Expectations.

(SRO ONLY)
Beaver Valley Unit 1 NRC Written Exam (1LOT8)

,0. Given the following plant conditions:

- The Emergency Director declared a Site Area Emergency at 1215.
- The initial report to state and local government was completed at 1227.
- An upgrade to General Emergency was declared at 1245.

The Protective Action Recommendation (PAR) to the State/County Agencies **must be** given by which ONE of the following times?

A. 1245

B. 1300

C. 1327

D. 1345

Answer: B

Explanation/Justification:

Incorrect. All distractors are plausible but incorrect as they are intervals of the given time in the question stem.
.. Correct. PAR must be declared within 15 minutes of declaration of a GE. This is SRO level knowledge only.
C. Incorrect. All distractors are plausible but incorrect as they are intervals of the given time in the question stem.
D. Incorrect. All distractors are plausible but incorrect as they are intervals of the given time in the question stem.

Sys # N/A | **System** N/A | **Category** Emergency Procedures/Plan | **KA Statement** Knowledge of emergency plan protective action recommendations.

K/A# 2.4.44 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** ½-EPP-IP-4.1, Rev. 28, pg. 10

Question Source: Bank – Vision 17585

Question Cognitive Level: Lower – Memory or Fundamental **10 CFR Part 55 Content:** CFR: 43 (b)(5)

Objective: