# Beaver Valley Unit 2 NRC Written Exam (2LOT7)

1. Given the following plant conditions:

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

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

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

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

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

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

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**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. It is true that a transition to ES-0.1 is made after completion of IMA's, however, emergency as opposed to normal boration is procedurally required to ensure adequate shutdown margin is maintained.

**B.** Correct. Plant conditions do not support safety injection initiation criteria so therefore a transition to ES-0.1 is warranted. ES-0.1 requires emergency boration for more than one stuck control rod to ensure adequate shutdown margin is maintained.

**C.** Incorrect. The additional boron from safety Injection is not required for these plant conditions to maintain adequate S/D Margin. Safety Injection not required unless pressurizer pressure is < 1860 psig nor is it required for two stuck rods. 2000 psig and dropping is a normal plant response following reactor trip.

**D.** Incorrect. The additional boron from safety Injection is not required for these plant conditions to maintain adequate S/D Margin. Safety Injection is not required based on PRZR Level nor is it required for the two stuck rods. PRZR level trending to 20% is a normal plant response on a reactor trip.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 007 | Reactor Trip | Knowledge of the operational implications of the following concepts as they apply to a reactor trip: | Shutdown Margin |

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

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

**Question Source:** New **Level Of Difficulty: (1-5)**

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

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

2. 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 (2RCS*PCV455C) momentarily lifts and does **NOT** fully reseat.
- A4-1E, "PRESSURIZER CONTROL PRESS DEVIATION HIGH/LOW", annunciates.
- PRZR Pressure is 2170 psig and slowly lowering.

How will the PRZR Spray Valves (2RCS*PCV455A and 2RCS*PCV455B), PRZR PORV Block Valves (2RCS*MOV535, 536, 537) and the PRZR PRESS MASTER CONTROLLER (2RCS-PK444A) automatically respond to these plant conditions?

PRZR Spray Valves ____ (1) ____.
PRZR PORV Block Valves ____ (2) ____.
PRZR PRESS MASTER CONTROLLER output control signal___ (3) ___.

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

B. (1) close
(2) remain open
(3) decreases

C. (1) remain open
(2) close
(3) increases

D. (1) close
(2) close
(3) decreases

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**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Spray Valves close on lowering PRZR pressure. Block Valves will remain open, since NSA is the OPEN position. Master pressure controller response would decrease as opposed to increase.

**B.** Correct. Deviation alarm annunciates when PRZR pressure drops to 2185 psig. This lowering demand signal closes spray valves and energizes PRZR heaters to restore PRZR pressure. The PRZR PORV Block valves will remain open due to control switches NSA OPEN. A lowering PRZR pressure due to the vapor space leak caused by PORV leak-by results in a lowering demand signal on the PRZR Master Pressure controller.

**C.** Incorrect. PRZR Spray Valves close on lowering PRZR pressure. PRZR Block Valves will automatically close when 2/3 PRZR pressures channels are @ 2185 psig if the control switch was in AUTO versus OPEN. Incorrect master pressure controller response.

**D.** Incorrect. Correct Spray Valve Position. PRZR Block Valves would automatically close when 2/3 PRZR channels are @ 2185 psig if the control switch was in AUTO versus OPEN. Correct master pressure controller response.

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| 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: | controllers and positioners. |

**K/A#** AK2.03 **K/A Importance** 2.5 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-6.4.IF, Rev 12; 2OM-6.3.C, Rev. 12

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-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)

3. Given the following plant conditions:

- Unit 2 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 473°F and stable.
- The hottest CET is 483°F and stable.
- Total Feed Flow is 600 gpm and stable.
- Pressurizer Level is 20% 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".

Due to a concern with the indication of the Subcooling Margin Monitor, 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?

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

(1) (2)

A. 35°F, shall.

B. 45°F, shall.

C. 35°F, shall NOT.

D. 45°F, shall NOT.

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**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 45°F. Incorrect that transition criteria is met. (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 criteria > 41°F (59°F) is NOT met. With containment pressure @ 5.5 psig, adverse numbers must be used. Since subcooling is NOT met, Attachment A-5.1 is to be used which still does not meet the required 47°F subcooling margin criteria. Additionally, the required 38% PRZR level is not met. Based on these numbers, transition criteria is NOT met and a transition to ES-1.1 shall NOT occur. Attachment A-5.1 is provided as it is not reasonable that a candidate should memorize adverse criteria numbers. It is reasonable however that the candidate should know SI termination criteria from memory, so therefore this criteria is NOT provided.

**D.** Incorrect. If the candidate uses Thot as opposed to CET to calculate subcooling, it will come out to 45 °F. If the candidate recognizes adverse containment criteria, this distractor is plausible since 45°F is not above the required minimum in accordance with Attachment A-5.1 which is 47°F and therefore a transition shall NOT be made. PRZR level criteria is also NOT met which is another transition criteria NOT met.

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| 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), 2OM-53A.1.A-5.1, Rev. 1

**Technical References:** 2OM-53A.1.E-1, Rev. 12; 2OM-53A.1.A-5.1, Rev. 1; 2OM-53B.4.E-1, Rev. 12; 2OM-53B.5.GI-11, Rev. 0

**Question Source:** New Modified 2008 Salem NRC Exam **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.7 / 45.7)

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

4. Given the following plant conditions:

- The Unit is operating at 25% power with all systems in NSA.
- Annunciator A2-5E, "REACTOR COOLANT LOOP FLOW LOW" is received.
- Reactor Coolant System (RCS) Loop "A" Flow indication channels I, II, & III indicate 80% and slowly lowering.
- "A" Reactor Coolant Pump (RCP) motor amps indicate 0 amps.
- No operator actions have been taken.

Based on these plant conditions, which ONE of the following alarms, if any, confirms a reactor trip?

A. No reactor trip alarm will be present.

B. A5-2G, "1/3 RCP LOOP FLOW LOW REACTOR TRIP".

C. A5-2H, "2/3 RCP LOOP FLOW LOW REACTOR TRIP".

D. A5-3F, "REACTOR COOLANT PUMP AUTO – STOP".

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. The candidate must analyze the indications provided and deduce that since reactor power is < 30% (P-8), and with only one RCP lost, that the reactor will NOT trip. Therefore no reactor trip alarm will be present.

**B.** Incorrect. This would be true if reactor power were above 30%.

**C.** Incorrect. This would be true if 2 of 3 low flow indicators were present on two loops with reactor power > 10% (P-7)

**D.** Incorrect. This annunciator is present based on no amperage indicated for the "A" RCP, however, it is NOT a reactor trip alarm nor will a reactor trip have occurred. It is plausible since it is on the first out annunciator panel.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 015/017 | Reactor Coolant Pump (RCP) Malfunctions | Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow) | Reactor Trip Alarms, switches, and indicators |

**K/A#** AA1.03 **K/A Importance** 3.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** USFAR Fig. 7.3-10, Rev. 9; 2OM-6.4.IF, Rev. 12

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-6.3 17. Given a RCP and support system configuration and without referenced material, describe the RCP and support system response to the following off-normal conditions, including automatic functions and changes in the equipment status as applicable: Reactor Trip

5. Given the following plant conditions:

The Unit is at 100% power with all systems in NSA.

**ONLY** the following annunciators are received in the Control Room:

- A2-3E, "CHARGING FLOW PATH TROUBLE"
- A2-3F, "LETDOWN FLOW PATH TROUBLE"
- No other Control Room Annunciators are in alarm.
- Five (5) minutes has elapsed.

Which ONE of the following describes the event in progress?

A. Charging flow control valve failure.

B. Charging pump trip on overcurrent.

C. Charging line leak inside containment.

D. Charging line leak outside containment.

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. These annunciators coupled with PRZR Level dropping and VCT Level rising correspond with 2CHS*FCV122 (Charging Flow Control Valve) failed closed and are supported by stated references. This failure results in a loss of reactor coolant makeup from CVCS.

**B.** Incorrect. A charging pump trip would cause these alarms and indications (VCT level rising and PRZR Level dropping), however, A2-3D, "Charging Pump Auto-Start/Auto-Stop annunciator would also be in alarm as well as seal injection alarms.

**C.** Incorrect. For a charging system leak inside containment, VCT level would drop as charging flow control valve opens to maintain PRZR level on programmed band. Additionally, there potentially would be other alarms associated with this condition location and size dependent such as containment radiation, sump levels, humidity etc. PRZR level would trend downward during this event. Charging Flow Path Trouble is received on a high charging flow rate and would be plausible for a charging system leak if the leak were downstream of the VCT. Letdown Flow Path Trouble is plausible for a high flowrate in the letdown system indicative of a leak upstream of the VCT.

**D.** Incorrect. For a charging system leak outside containment, VCT level would drop as charging flow control valve opens to maintain PRZR level on programmed band for a leak downstream of the VCT. For an upstream leak, VCT level would drop to about 20% and automatic makeup to the VCT would occur. PRZR level would trend downward during this event. Charging Flow Path Trouble is received on a high charging flow rate and would be plausible for a charging system leak if the leak were downstream of the VCT. Letdown Flow Path Trouble is plausible for a high flowrate in the letdown system indicative of a leak upstream of the VCT.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 022 | Loss of Reactor Coolant Makeup | Ability to operate and / or monitor the following as they apply to the Loss of Coolant Makeup: | Whether charging line leak exists. |

**K/A#** AA2.01 **K/A Importance** 3.2 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.7.1, Rev. 4; OM Fig. 7-1A, Rev. 19

**Question Source:** Bank Vision - 46561 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-7.1 19. Given a CVCS configuration and without reference material, describe the CVCS control room response to the following off normal conditions, including all automatic functions and changes in equipment status as applicable

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

- OPPS is in service.
- Residual Heat Removal System (RHS) has been placed in service in accordance with 2OM-10.4.A, "RHS Startup".
- The plant has just entered Mode 5 using "A" RHS Train.
- A Loss of Residual Heat Removal (RHR) and resultant RCS heat-up and pressurization occurs.

Under these plant conditions, why are the AUTO closure interlocks "defeated" for RHS Train "A" Supply Isolation (2RHS*MOV701A & 2RHS*MOV702A) and RHS Train "A" Return Isolation (2RHS*MOV720A)?

A. To prevent inadvertent closure due to spurious actuation.

B. The PRZR Safety Valves are capable of protecting the RHS piping.

C. To prevent "AUTO CLOSURE" during EDG Load Sequencer Testing.

D. The PRZR Liquid Temperature sensor used in the interlock will no longer provide a valid signal.

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. Although these valves do have an auto closure feature on high RCS pressure, this feature is defeated shortly after placing RHS in service IAW 2OM-10.4.A. The auto close feature is designed to auto isolate RHS during RCS heat-up and pressurization in case the valves are left inadvertently open, not to protect the RHS (low pressure piping) from over-pressurization during RHS operation which is the stated plant condition in the question stem.

**B.** Incorrect. PRZR safety valves are not relied on for protection during these plant conditions, but rather the PORVs are used for RCS overpressure protection (OPPS is in service). During normal plant conditions, the PRZR Safeties are for RCS versus RHR piping protection.

**C.** Incorrect. The signal generated during a test is not a spurious signal which is the bases of defeating the interlock to these valves.

**D.** Incorrect. BVPS Unit 1 has an temperature input to this interlock to allow opening versus to prevent closure of the valves. This interlock is not applicable to BVPS 2.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 025 | Loss of Residual Heat Removal System (RHRS) | Knowledge of the reasons for the following as they apply to the Loss of Residual Heat Removal System: | Isolation of RHR low-pressure piping prior to pressure increase above specified level. |

**K/A#** AK3.02 **K/A Importance** 3.3 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-10.4.A, Rev. 37<br>2SQS-10.1, Rev.17

**Question Source:** Modified Bank Vision - 873 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.5 / 41.10 / 45.13)

**Objective:** 2SQS-10.1 18. Given a specific plant condition, predict the response of RHS 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.

7. Given the following plant conditions:

- The Unit is operating at 90% power, with all systems in NSA.
- The pressure transmitter that inputs to the Master PRZR Pressure Controller begins to **slowly** drift upward.

Assuming the transmitter continues to fail in the upward direction, which one of the following describes the effect on both the Pressurizer Pressure Control System (PRZR PCS), and Reactor Coolant System (RCS), and what operator actions will mitigate the effects of this event, according to 2OM-6.4.IF, "Instrument Failure Procedure"?

The PRZR PCS will ____ (1) ____ causing actual RCS pressure to ____ (2) ____.
This transient will be mitigated by ____ (3) ____.

A. (1) increase PRZR heater output
(2) increase
(3) manually controlling the PRZR spray valve controller(s).

B. (1) decrease PRZR heater output
(2) decrease
(3) manually controlling the PRZR spray valve controller(s).

C. (1) close PRZR spray valves
(2) increase
(3) manually controlling the Master PRZR pressure controller.

D. (1) open PRZR spray valves and one PRZR PORV
(2) decrease
(3) closing the effected PRZR PORV AND by manually controlling the PRZR spray valve controller(s).

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**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. This would be the correct plant response if 2RCS*PT444 was failing in the opposite direction.

**B.** Incorrect. It is correct that the heater output is decreasing and RCS pressure would drop. Incorrect that the only action is to take manual control of the each PRZR spray valve controller. 2RCS-PCV455C PORV would still receive an increasing pressure signal from 2RCS*PT444. If this PORV is not closed the plant will trip on low RCS pressure.

**C.** Incorrect. It is plausible that if the failure was in the opposite direction spray valves will close. Therefore it is logical that RCS pressure will increase as a result and controlling the master PRZR controller in manual is also plausible although not procedurally supported.

**D.** Correct. The candidate must know that 2RCS*PT444 provides the input to the Master Pressurizer controller. If this transmitter fails in the upward direction, the effect is for the PRZR PCS to open both spray valves and to also open 2RCS*PCV455C (PORV). This results in a pressure drop in the RCS. Correct operator action according to 2OM-6.4.IF is to take manual control of the spray valves and PORV to mitigate the pressure drop.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 027 | Pressurizer Pressure Control System (PZR PCS) Malfunction | 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** 4.2 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-6.4.IF, Rev. 12; 2SQS-6.4, Rev. 12

**Question Source:** Modified Bank Vision - 46716 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-6.4 19. Given a specific plant condition, predict response of the PRZR and PRZR Relief system control room indications and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or off-normal condition: Process Instrument Failure.

8. 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 operators are performing the actions of FR-S.1, "Response to Nuclear Power Generation - ATWS".

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.

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**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.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 029 | Anticipated Transient Without Scram (ATWS) | Knowledge of the operational implications of the following concepts as they apply to the ATWS: | Definition of negative temperature coefficient as applied to large PWR coolant systems. |

**K/A#** EK1.05 **K/A Importance** 2.8 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.FR-S.1, Rev. 4<br>2OM-53B.4.FR-S.1, Rev. 4<br>GO-GPF.R4, Rev. 1

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-53.3 1. State from memory the basis and sequence of major action steps of each EOP IAW BVPS-EOP Executive Volume.

GO-GPF.R4 2. Define the term MTC of reactivity

9. Given the following plant conditions:

- The Unit has been operating at 100% power with all systems in NSA for 300 days.
- A double ended steam line break occurs upstream of 21A S/G Main Steam Line Isolation Valve inside containment.
- Low Steamline Pressure Safety Injection fails to **AUTO** occur.
- Auxiliary Feedwater has been isolated to the faulted S/G.

Based on these plant conditions, how will the plant respond?

RCS cooldown and depressurization will ________

A. **be terminated** when Hi-Hi containment pressure MSLI auto occurs.

B. **be terminated** by manually actuating low steamline pressure safety injection.

C. **continue** after low PRZR pressure safety injection occurs until 21A S/G blows dry.

D. **continue** until high steamline pressure rate MSLI occurs and **ALL** S/G's blow dry.

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**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. High containment pressure SI will occur at 5 psig containment pressure. A design basis SLB inside containment will result in containment pressure reaching well beyond 5 psig. At 7 psig Hi-Hi Containment Pressure MSLI occurs. However, since the break is upstream of the MSIV, the RCS cooldown and depressurization will not be terminated, despite MSLI and SI actuation, until the faulted S/G blows dry.

**B.** Incorrect. Manually actuating low steam line pressure SI is plausible since it failed to actuate, however, MSLI has already occurred on Hi-Hi containment pressure (based on a design bases fault of a S/G inside containment and resultant containment pressure reaching > 30 psig. Low steamline pressure safety injection will not terminate a SLB inside containment. It will however, terminate a SLB outside containment, downstream of the MSIV's.

**C.** Correct. 2OM-21.1.B (Main Steam Summary Description) states that MSIV closure prevents B/D of all three S/Gs in case of a SLB. It also details the signals which cause MSLI to occur. (Low Steam Line Pressure – when not blocked, High Steam Line Pressure Rate – when < P-11, Intermediate Hi containment pressure, or manual isolation) The stem of the question is silent on the status of MSLI. The candidate must conclude that although low steam line pressure SI failed to occur, that Hi-HI containment pressure MSLI has occurred. Since MSLI did occur, it will prevent the blowdown of the non-faulted S/Gs. According to E-2 background document, a double ended break from full power will result in a drop of RCS temperature and depressurization until low pressure SI occurs. After decay heat is removed, the core will begin restoring RCS temperature and pressure (ie: S/G blows dry).

**D.** Incorrect. High Steam line pressure rate will only occur when <P-11 (2000 psig) and MSLI signal is manually blocked. Based on SLB location inside versus outside containment, even if the isolation were to occur, it will NOT stop the RCS cooldown and depressurization. It is not necessary for all three S/Gs to blow dry for the cooldown and depressurization to be reversed. The candidate must understand that the purpose of MSLI when a break occurs between the S/G and MSIVs is to prevent the blowdown of the non-faulted S/Gs, so therefore, must be able to conclude that only one versus three S/Gs will blow dry.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 040 | Steam Line Rupture | Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture: | RCS shrink and consequent depressurization |

**K/A#** AK1.03 **K/A Importance** 3.8 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-21.1.B, Issue 4, Rev 0; 2OM-53B.4.E-2, Rev. 7; TS 3.7.2 Bases, Rev. 0

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comprehension **10 CFR Part 55 Content:** (CFR 41.8 / 41/10 / 45.3)

**Objective:** 2SQS-21.1 18. Describe the basis for the Main Steam Supply System and associated major components as documented in the UFSAR.

# Beaver Valley Unit 2 NRC Written Exam (2LOT7)

10. Given the following sequence of events:

- The Unit was initially operating at 70% power with all systems in NSA.
- One Main Feedwater Pump has tripped.
- 21A NR S/G levels are ALL indicating 22% and LOWERING.
- 21B NR S/G levels are ALL indicating 19% and LOWERING.
- 21C NR S/G levels are ALL indicating 18% and LOWERING.
- Pressurizer pressure is indicating 2310 psig on all protection pressure channels and rising.
- Plant Operation at power continues and no operator action has been taken.

Based on the **current** conditions, which of the following reactor trip first-out annunciators will be present?

1. Annunciator A5-2C – SG 21A LEVEL LOW-LOW REACTOR TRIP
2. Annunciator A5-3C – SG 21B LEVEL LOW-LOW REACTOR TRIP
3. Annunciator A5-4C – SG 21C LEVEL LOW-LOW REACTOR TRIP
4. Annunciator A5-3H – PRESSURIZER PRESS HIGH REACTOR TRIP

A. 2 AND 3 **ONLY**.

B. 3 AND 4 **ONLY**.

C. 1, 2, 3, AND 4.

D. 1, 2, AND 3 **ONLY**.

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. With no operator action and reactor power at 70%, One MFP will be insufficient to maintain S/G Water Levels. S/G water levels will drop and RCS pressure will rise in response to the stated plant conditions. An automatic reactor trip should have occurred based on 2/3 indicators on both "B" and "C" S/G < 20.5%. SG21B and SG21C Level Low-Low Level Reactor Trip Annunciators are in alarm. The RO candidate must be able to recognize plant conditions that warrant prompt action, especially in situations where the reactor protection system did not function as designed.

**B.** Incorrect. PRZR pressure alarm does not annunciate until 2/3 PRZR Pressure Protection channels are > 2375 psig.

**C.** Incorrect. SG 21A Level LOW-LOW reactor trip annunciator is not present until S/G level is < 20.5% on 2/3 channels. PRZR pressure alarm does not annunciate until 2/3 PRZR Pressure Protection channels are > 2375 psig.

**D.** Incorrect. SG 21A Level LOW-LOW reactor trip annunciator is not present until S/G level is < 20.5% on 2/3 channels.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 054 | Loss of Main Feedwater (MFW) | Ability to determine and interpret the following as they apply to the Loss of Feedwater (MFW): | Reactor Trip first-out panel indicator |

**K/A#** AA2.07 **K/A Importance** 3.4 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-1.4.ABK, Rev. 5<br>2OM-1.4.ABP, Rev. 5<br>2OM-1.4.ABO, Issue 4, Rev. 1

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-24.1 14. Given a Main Feedwater, Startup Feedwater, AFW or SGWLC configuration and without referenced material, describe the associated system's control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable: reactor trip, SG low-low level.

11. The Unit is operating at 100% power when a Station Blackout causes a reactor trip. Fifteen (15) minutes after the trip, the following conditions exist:

- "A" Steam Generator (S/G) pressure is 1000 psig and STABLE.
- "B" S/G pressure is 1005 psig and STABLE.
- "C" S/G pressure is 995 psig and STABLE.
- All Reactor Coolant Pumps (RCPs) are "OFF".
- Reactor Coolant System (RCS) pressure is 2230 psig and slowly RISING.
- T-hot is 575°F in all three (3) loops and STABLE.
- Core exit thermocouples indicate 580°F.
- T-cold is 555°F in all three (3) loops and STABLE.
- All Systems function as designed.

Based on these conditions, what is the condition of RCS natural circulation **AND** RCS heat removal?

Natural Circulation does ___ (1) ___.
RCS Heat Removal ___ (2) ___.

A. (1) exist
(2) is being maintained by Condenser Steam Dumps

B. (1) exist
(2) is being maintained by S/G Atmospheric Steam Dumps

C. (1) **NOT** exist
(2) may be established by opening the Condenser Steam Dumps

D. (1) **NOT** exist
(2) may be established by opening the S/G Atmospheric Steam Dumps

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Natural circulation conditions do not exist IAW Attachment A-1.7. Condenser Steam dumps are unavailable.

**B.** Incorrect. Natural circulation conditions do not exist IAW Attachment A-1.7. Atmospheric steam dumps are not maintaining heat removal.

**C.** Incorrect. Correct that natural circulation does not exist, however, condenser steam dumps are unavailable.

**D.** Correct. Tcold is too hot for existing steam pressure. Steam temperature and Tcold should be about the same if natural circulation is present. Without power to condenser cooling tower pumps, the condenser is unavailable and therefore atmospheric steam dumps must be used to increase steaming rate and thus establish natural circulation of the RCS through S/G cooling.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 055 | Loss of Offsite and Onsite Power (Station Blackout) | Ability to determine or interpret the following as they apply to a Station Blackout. | RCS core cooling through natural circulation cooling to S/G cooling. |

**K/A#** EA2.02 **K/A Importance** 4.4 **Exam Level** RO

**References provided to Candidate** Steam Tables (Red) **Technical References:** 2OM-53A.1.ECA-0.1, Rev.7
2OM-53A.1.A-1.7, Rev. 1
2OM-53A.1.A-5.1, Rev 1

**Question Source:** Bank Vision - 17261 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 43.5 / 45.13)

**Objective:** 3SQS-53.2 12. State from memory the five conditions which indicate natural circulation is occurring iaw BVPS EOP Executive Volume.

12. Given the following plant conditions:

- A Loss of ALL AC Power occurred requiring the crew to enter ECA-0.0, "Loss Of All 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 how the EDG will sequence loads onto the bus **AND** the reason for this sequencing?

All loads powered by the EDG complete sequencing between ____ (1) ____ after initiating signal. The reason for sequential loading of the EDG is to _____ (2) _____.

A. (1) .5 - 30 seconds
(2) prevent the emergency bus from becoming inoperable.

B. (1) .5 - 30 seconds
(2) prevent damage to reactor coolant pump seal package.

C. (1) .5 - 60 seconds
(2) prevent the emergency bus from becoming inoperable.

D. (1) .5 - 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 2OM-36.1.C all EDG loads are sequenced onto the EDG between .5 to 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).

**D.** Incorrect. Correct time. Incorrect reason. Reason is plausible since ECA-0.0 background focuses heavily on protection of RCP seal packages.

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| 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:** 2OM-53A.1.ECA-0.0, Rev 9
2OM-36.1.C, Rev. 4
TS 3.8.1 Bases, Rev. 0

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **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.

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

- The Unit is operating at 100% power with all systems in NSA.
- The control room receives A1-4H, "SERVICE WATER SYSTEM TROUBLE" followed shortly after by A1-4G, "SERVICE WATER HEADER PRESSURE LOW".
- "A" SW Header Pressure indicates 30 psig and slowly DROPPING.
- "B" SW Header Pressure indicates 78 psig and STABLE.
- One (1) minute has elapsed.

Based on these plant conditions, what will be the status of Secondary Component Cooling System (CCS) Water Heat Exchanger (HX) Service Water Supply Header "A"("B") Isolation Valve(s) **AND** what is the reason for this plant response?

CCS Water HX Service Water Supply Header **"A"** Isolation Valves (2SWS*MOV107A/B)
CCS Water HX Service Water Supply Header **"B"** Isolation Valves (2SWS*MOV107C/D)

A. BOTH **"A"** Header Valves isolate to allow more water to go to the CCP HX's.

B. ONLY ONE **"A"** Header Valve isolates to allow more water to go to the CCP HX's.

C. BOTH **"A"** AND **"B"** Header Valves isolate to restore SW Header pressure to normal.

D. ONLY ONE **"A"** Header and ONE **"B"** Header Valve isolate to restore SW Header pressure to normal.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Only one "A" Header Valve will isolate. See explanation for correct answer.

**B.** Correct. A low discharge pressure (<34 psig) for 45 seconds causes the header isolation for the affected header ONLY to close. For the "A" Header 2SWS*107A closes. (For the "B" Header 2SWS*107D will close) The stated conditions are indicative of an "A" header rupture only so therefore only one "A" header valve will close. The reason stated in 2OM-30.1.B for auto closure of these valves is to allow more water to flow to the CCP HX's. The K/A is met because a loss of SW causes a loss of SCC which in turn has an effect on the SW header discharge flow to the CCP HX's.

**C.** Incorrect. Plausible reason if the rupture were in the section of piping which cools CCS, however, based on indications provided, only the "A" header is affected.

**D.** Incorrect. Again a plausible reason and correct plant response if both "A" and "B" headers had low system pressure, however, only the "A" header is affected.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 062 | Loss of Nuclear Service Water | Knowledge of the reasons for the following responses as they apply to Loss of Nuclear Service Water: | Effect on the nuclear service water discharge flow header of a loss of CCW |

**K/A#** AK3.04 **K/A Importance** 3.5 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.30.1, Rev. 8
2OM-30.4.AAC, Rev. 15
2OM-30.4.AAB, Rev. 3
2OM-30.1.B, Rev. Issue 4, Rev. 5
2SQS-30.1 Lesson Plan Simplified Diagrams.

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comprehension **10 CFR Part 55 Content:** (CFR 41.4 / 41.8 / 45.7)

**Objective:** 2SQS-30.1 16. Given a specific plant condition, predict the response of the Service Water System control room indication and control loops, including all automatic functions and changes in equipment status for either a change in plant condition or for an off-normal condition.

14. Given the following plant conditions:

- The Unit is at 100% power with all systems in NSA with the exception of 2IAC-MOV131 which is OPEN.
- A6-3C, "STATION INSTRUMENT AIR RECEIVER TANK TROUBLE" annunciates.
- 2IAS-P106, "Station Instrument Air Header" dropped to 82 psig but is currently reading 92 psig and rising.

Given these plant conditions and assuming no operator action, what will be the status of 2SAS-AOV105, SAS Main Header to Service Air Header AOV **AND** 2IAC-MOV131, Containment Instrument Air Backup Supply Valve?

| | 2SAS-AOV105 Status | 2IAC-MOV131 Status |
|---|---|---|
| A. | OPEN | CLOSED |
| B. | CLOSED | OPEN |
| C. | OPEN | OPEN |
| D. | CLOSED | CLOSED |

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. 2SAS-AOV105 closes and 2IAC-MOV131 remains open. It is plausible that 2IAC-MOV131 is closed since it is currently listed NSA. When Containment Instrument Air Compressors are officially retired, OM-34.3.B.4 will be updated to reflect the actual NSA open position which reflects current plant operations.

**B.** Correct. 2SAS-AOV105 automatically closed on lowering instrument air header pressure (<90 psig by alarm response and < 86 psig by AOP). This closure separates station air header from the instrument air header to preserve the more vital instrument air if the leak is in the Station Air header. This valve does not auto reopen on rising air header pressure which is the stated case. 2IAC-MOV131 is currently maintained open to supply instrument air to the containment. (NSA position is closed until containment air compressors are retired). This valve does not auto close on low air header pressure, so therefore remains open for the stated plant conditions. All other combinations are plausible based on candidates understanding of system operation with regard to how these valves operate to minimize the drain on the instrument air system.

**C.** Incorrect. 2SAS-AOV105 closed. Correct that 2IAC-MOV131 remains open.

**D.** Incorrect. Correct that 2SAS-MOV105 closed. Incorrect that 2IAC-MOV131 closes.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 065 | Loss of Instrument Air | Ability to operate and/or monitor the following as they apply to the Loss of Instrument Air: | Components served by instrument air to minimize drain on system |

**K/A#** AA1.02 **K/A Importance** 2.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-34.4.AAA, Rev. 10
2OM-53C.4.2.34.1, Rev. 14
2OM-34.1.D, Rev. 3
2OM-34.3.B.4, Rev. 10
2SQS-34 Powerpoint Simplified Print

**Question Source:** Bank Vision - 68005 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-34.1 25. Given a specific plant condition, predict the response of the compressed air system control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant conditions or for an off-normal condition.

15. Given the following plant conditions:

- The Unit is at 100% power with all systems in NSA.
- The DLC System Operations Control Center informs the control room of possible grid instability.
- The control room team enters AOP 1/2 .35.1, "Degraded Grid".

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

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

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

A. (1) will AUTO occur
(2) RAISE

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

C. (1) will AUTO occur
(2) LOWER

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

---

**Answer:** **C**

**Explanation/Justification:**

**A.** Incorrect. Over-excitation adjustment made shall be in the lower direction only

**B.** Incorrect. Auto transfer will occur. Over-excitation adjustment made shall be in the lower direction only

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

**D.** Incorrect. Auto transfer will occur.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 077 | Generator Voltage and Electric Grid DIsturbances | Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: | Voltage Regulator Controls |

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

**References provided to Candidate** None **Technical References:** 1/2OM-53C.4A.35.1, Rev. 7
2SQS-35.3 Powerpoint Simplified Prints

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.5 / 45.5, 45.7, and 45.8)

**Objective:** 2SQS-35.3 5. Given a set of plant conditions and the appropriate procedures, apply the operational sequence, parameter limits, precaution and limitations, and cautions and notes applicable to the completion of the task activities in the control room.

16. 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 are 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 |

---

**Answer: A**

**Explanation/Justification:**

**A.** 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.

**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.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| W/E04 | LOCA Outside Containment | 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** None **Technical References:** 2OM-53A.1.ECA-1.2, Rev. 1; 2OM-53B.4.ECA-1.2, Rev. 1,

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR41.10 / 43.5 / 45.12)

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

17. Given the following plant conditions:

- The Unit was operating at 100% with all systems in NSA.
- A tornado passed through the site causing a Loss of Off-Site Power.
- All equipment functioned as designed.
- The PPDWST (2FWE*TK210) was ruptured by the tornado resulting in total unavailability.
- All Narrow Range (NR) Steam Generator (S/G) Water Levels are off-scale low.
- All S/G pressures are 400 psig and slowly rising.

Which ONE of the following, if any, are currently capable of providing feed flow to the S/Gs **without performing** any field operations?

1. Motor Driven Auxiliary Feedwater Pumps
2. Turbine Driven Auxiliary Feedwater Pump
3. Condensate Pumps
4. Startup Feedwater Pump

A. 4 **ONLY**.

B. 3 AND 4 **ONLY**.

C. 1 AND 2 **ONLY**

D. No feed flow capability currently exists.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Although the S/U Feedwater pump would have power from the ERF EDG in this situation, it would have no suction source since the condensate and heater drain pumps are not powered from the EDG's, therefore suction pressure start permissives would not be satisfied.

**B.** Incorrect. Condensate pumps are powered from non-emergency busses and therefore are not available.

**C.** Incorrect. Both Motor Driven and Turbine Driven AFW pumps take suction from PDWST. An alternate suction source from Service Water is available, however, this requires manual action from outside the control room.

**D.** Correct. No feedwater sources are readily available which results in loss of heat sink entry conditions.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| W/E05 | Loss of Secondary Heat Sink | Knowledge of the interrelations between the Loss of Secondary Heat Sink and the following: | Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. |

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

**References provided to Candidate** None **Technical References:** 2OM-22A.1.D, Rev. 4; 2OM-24.1.D, Rev. 6; OP Manual Fig. 24-1, Rev. 11; OP Manual Fig. 24-3, Rev. 12

**Question Source:** Bank Vision - 9166 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-24.1 16. Given a specific plant condition, predict the response of the MF System, Startup Feedwater, AFW or SGWLC systems control room indication and control loops, including automatic functions and changes in equipment status, for either a change in plant condition or for an off-normal condition.

18. Which ONE of the following describes the mitigation strategy while performing ECA-1.1, "Loss Of Emergency Coolant Recirculation" following a **small break** LOCA?

RCS cooldown and depressurization ____ (1) ____ be initiated.
Safety Injection flow will be reduced to minimum using ____ (2) ____.

A. (1) will **NOT**
(2) **ONE** HHSI/Charging and **ONE** LHSI pump for RCS decay heat removal.

B. (1) will **NOT**
(2) **ONE** HHSI/Charging pump for RCS decay heat removal with **BOTH** LHSI pumps secured.

C. (1) will
(2) **ONE** HHSI/Charging pump for RCS decay heat removal with **BOTH** LHSI pumps secured.

D. (1) will
(2) **ONE** LHSI pump for RCS decay heat removal and **ONE** HHSI/Charging pump aligned to the normal Charging flowpath.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. LHSI is secured and cooldown and depressurization is initiated.

**B.** Incorrect. It is correct that SI flow is reduced to one HHSI pump and both LHSI pumps are secured, however, a cooldown and depressurization is initiated.

**C.** Correct. The mitigating strategy used in ECA-1.1 is to conserve RWST inventory by minimizing SI flow and cooling down and depressurizing to reduce break flow thus extending the time to recover emergency core cooling recirculation capability. SI flow is minimized and LHSI pumps are secured in ECA-1.1.

**D.** Incorrect. It is correct that a cooldown is initiated; however, LHSI pumps are secured. If the break is made small enough to establish sufficient subcooling, then SI is terminated and only one charging pump remains running.

---

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| W/E11 | Loss of Emergency Coolant Recirculation | N/A | Knowledge of EOP mitigation strategies. |

**K/A#** 2.4.6 **K/A Importance** 3.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53B.4.ECA-1.1, Rev. 9; 2OM-53A.1.ECA-1.1, Rev. 9

**Question Source:** Bank 2LOT4 NRC Exam **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-53.3 Describe from memory the overall purpose of each procedure IAW BVPS-EOP Executive Volume.

19. Given the following plant conditions:

- The Unit has been operating at 100% power, beginning of core life (BOL).
- The control room operator receives A4-9F, "ROD AT BOTTOM" annunciator.
- ONE (1) Rod Bottom Light is LIT on VB-B.
- The control room team enters AOP 2.1.8, "Rod Inoperability" based on numerous other confirmatory indications, however NO actions have been taken yet.
- Dropped rod worth is approximately 1000 pcm.

What is the definition of Power Defect, **AND** what will be the impact of this same event (equivalent rodworth) at End of Core Life (EOL)?

Power defect is defined as the ___ (1) ___.
The temperature change as a result of the dropped rod will be ___ (2) ___.

A. (1) total amount of reactivity added due to a given change in power.
(2) less at BOL than EOL.

B. (1) change in reactivity due to per % change in power.
(2) less at BOL than EOL.

C. (1) total amount of reactivity added due to a given change in power.
(2) greater at BOL than EOL.

D. (1) change in reactivity due to per % change in power.
(2) greater at BOL than EOL.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Correct definition. Temperature change would be greater versus less.
**B.** Incorrect. The definition is for a coefficient as opposed to defect. Incorrect effect.
**C.** Correct. Power defect is defined as the total amount of reactivity added due to a given change in power. The temperature change as a result of the dropped rod will be greater at BOL than EOL because the value of power defect is much more negative at EOL.
**D.** Incorrect. The definition is for a coefficient as opposed to defect. Correct effect.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 003 | Dropped Control Rod | Knowledge of the operational implications of the following concepts as they apply to Dropped Control Rods: | Definition and application of power defect |

**K/A#** AK1.15 **K/A Importance** 2.8 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-1.4.AAA, Rev. 5
GO-GPF.R4, Rev. 1
BVPS II Curve Book CB-21, Issue 15, Rev. 0

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** Go-GPF.R4
8. Describe the components of power coefficient.
9. Explain the differences between reactivity coefficients and reactivity defects.
10. Explain and describe the effect of power defect and Doppler defect on reactivity.

20. The Unit is in Mode 6. A fuel assembly is being lowered into the core.

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

What manipulator crane interlock provides this protection?

A. Overload

B. Underload.

C. Tube Down.

D. Bridge-Trolley Hoist.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Overload will stop UPWARD motion if an assembly is binding while moving upward.
**B.** Correct. In accordance with LP 3SQS-6.13 slide 99 and 2RP-3.3.
**C.** Incorrect. Tube down interlock will stop hoist downward motion when the hoist is all the way down.
**D.** Incorrect. Bridge-Trolley Hoist interlock will only allow motion/movement in one direction at a time.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 036 | Fuel Handling Incidents | Knowledge of the reasons for the following responses as they apply to the Fuel Handling Incidents: | Interlocks associated with fuel handling equipment |

**K/A#** AK3.02 **K/A Importance** 2.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** LP 3SQS-6.13 Slide 99, Rev. 3
2RP-3.3, Rev. 3

**Question Source:** Bank 2LOT6 NRC Exam **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.5 / 41.10 / 45.6 / 45.13)

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

21. Unit 2 is operating at 100% power when the following alarms are received:

- A1-5B, "GASEOUS WASTE TANK RUPTURE DISC TROUBLE".
- A1-5A, "GASEOUS WASTE SYSTEM TROUBLE".
- Computer Alarm "GWS-TK-21 RLF VLV GWS-PS126 RUPTUR".

Additionally the following information is received:

- Health Physics reports increased radiation on Unit 1 RM-1GW-108A Radiation Monitor.
- The PAB Auxiliary Operator reports 2GWS-PCV116, "Surge Tank Pressure Control Valve" is CLOSED.

Using the system prints and plant conditions provided, which ONE of the following is the source of the gaseous waste system release?

A. TK-21 Rupture Disk **ONLY**.

B. TK-21 Rupture Disk AND Relief Valve lifted.

C. TK-22A Rupture Disk AND Relief Valve lifted.

D. GWS Header Rupture Disk AND Relief Valve lifted.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. TK-21 rupture disk has ruptured. However, the relief valve would also need to lift in order to complete the flow path to Unit 1 radiation monitor. It is plausible since the PRZR PRT rupture disk causes a high radiation level when its rupture disk blows.

**B.** Correct. In accordance with OP Manual Fig 19-1 (Unit 2)/ Fig 19-1/3 (Unit 1) and ARP 2OM-19.4AAB, there are three inputs to this alarm. TK-21, TK-22s and the GWS Header. The candidate must apply the system print to differentiate that PS-126 monitors TK-21 rupture disk and at 5 psig alarms indicating that there is pressure downstream of the rupture disk. In order to receive the radiation alarm in Unit 1 which is also common to all three rupture disk discharge paths, the relief valve would also have to lift @ 100 psig. 2OM-19.4AAA is also common to both TK-21 and the GWS Headers.

**C.** Incorrect. Refer to correct answer explanation.

**D.** Incorrect. Refer to correct answer explanation

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 060 | Accidental Gaseous-Waste Release | Ability to determine and interpret the following as they apply to the Accidental Gaseous Radwaste: | The possible location of a radioactive-gas leak, with the assistance of PEO, health physics and chemistry personnel. |

**K/A#** AA2.02 **K/A Importance** 3.1 **Exam Level** RO

**References provided to Candidate**
OP Manual Fig. Unit 2 Fig 19-1, Rev. 8
OP Manual Fig. Unit 2 Fig. 19-3, Rev. 4
OP Manual Fig. Unit 1 Fig. 19-1, Rev. 16

**Technical References:**
2OM-19.4AAB, Rev. 4
2OM-19.4.AAA, Rev. 8
2SQS-19.1 PPT Figures
OP Manual Fig. Unit 2 Fig 19-1, Rev. 8
OP Manual Fig. Unit 2 Fig. 19-3, Rev. 4
OP Manual Fig. Unit 1 Fig. 19-1, Rev. 16

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-19.1 12. Given a change in plant conditions due to system or component failure, analyze the Gaseous Waste Disposal System to determine what failure has occurred.

22. Unit 2 Control Room has been evacuated during a major uncontrolled fire in the control room.

Which ONE of the following equipment/control systems are available at the Alternate Shutdown Panel during these plant conditions?

1. Secondary Component Cooling Water Pump [2CCS*P21A]
2. Primary Component Cooling Water Pump [2CCP*P21A]
3. Service Water Pump [2SWS*P21A]
4. Reactor Coolant Pump [2RCS*P21A]
5. SG Atmospheric Steam Dump Valve [2SVS*PCV101A]

A. 2, 3, 4 ONLY.

B. 1, 4, 5 ONLY.

C. 2, 3, 5 ONLY.

D. 1, 2, 3, 5 ONLY.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Reactor Coolant Pumps are not located on the ASP.

**B.** Incorrect. Secondary Component Cooling Water Pumps and Reactor Coolant Pumps are not located on the ASP.

**C.** Correct. According to 2OM-53C.4.2.33.1A for major uncontrolled fires in the control room that 2OM-56C will be implemented. 2OM-56C .4.F-1 lists the vital equipment which is on the Alternate Shutdown Panel. Primary Component Cooling Water Pumps, Service Water Pumps, and SG Atmospheric Dump Valve for "A" and "B" S/Gs are located on the ASP.

**D.** Incorrect. Secondary Component Cooling Water Pumps are not located on the ASP.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 067 | Plant Fire On Site | Ability to determine and interpret the following as they apply to the Plant Fire on Site: | Vital equipment and control systems to be maintained and operated during a fire |

**K/A#** AA2.16 **K/A Importance** 3.3 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.33.1A, Rev. 12<br>2OM-56C.4.F-1, Rev. 12<br>2OM-56C.4.A, Rev. 11

**Question Source:** New **Level Of Difficulty: (1-5)**

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

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

23. The Unit is operating at 100% with all systems NSA when the following sequence of events occur:

- All Main Steam Isolation Valves (MSIVs) close.
- A reactor trip occurs.
- "A" Reactor Trip Breaker (RTB) fails to open.
- Several Main Steam Safety Valves fail open resulting in an automatic safety injection.
- The crew is progressing through the Emergency Operating Procedures (EOP) set and has just transitioned to ES-1.1, "SI Termination".

Using the reference provided, and based on these plant conditions, which ONE of the following describes the status of resetting Safety Injection in order to secure the "A" High Head Safety Injection (HHSI) pump?

The "A" Train of Safety Injection (SI) ____________________

A. can be reset 75 seconds **AFTER** the SI actuation regardless of reactor coolant system pressure.

B. can be reset 75 seconds **AFTER** the SI actuation only if reactor coolant system pressure is > P-11.

C. cannot be reset until "A" RTB is opened **IF** an automatic SI signal is still present.

D. cannot be reset until "A" RTB is opened **AND** all automatic SI signals are cleared.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. SI cannot be reset until the "A" RTB is opened. 75 seconds is plausible since SI cannot be reset until after 75 seconds has elapsed.

**B.** Incorrect. SI cannot be reset until the "A" RTB is opened. 75 seconds is plausible since SI cannot be reset until after 75 seconds has elapsed.

**C.** Correct. According to the reference provided, the RTB (P-4) must be opened AND the SI reset button must be depressed in order to reset SI.

**D.** Incorrect. SI cannot be reset with S/G pressure < 500 psig without opening the "A" RTB. If S/G pressure is > 500 psig, then SI can be reset after a time delay by depressing the SI reset button.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| W/E02 | SI Termination | Ability to operate and / or monitor the following as they apply to the (SI Termination): | 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** 4.0 **Exam Level** RO

**References provided to Candidate** None **Technical References:** UFSAR Functional Diagram 7.3-13, Rev. K; 2SQS-11.1, Rev. 15

**Question Source:** Bank Vision - 1519 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-11.1 15. Describe the control, protection and interlock functions for the control room components associated with SI system, including automatic functions, setpoints and changes in equipment status as applicable: SI actuation and reset.

# Beaver Valley Unit 2 NRC Written Exam (2LOT7)

23. The Unit is operating at 100% with all systems NSA when the following sequence of events occur:

- All Main Steam Isolation Valves (MSIVs) close.
- A reactor trip occurs.
- "A" Reactor Trip Breaker (RTB) fails to open.
- Several Main Steam Safety Valves fail open resulting in an automatic safety injection.
- The crew is progressing through the Emergency Operating Procedures (EOP) set and has just transitioned to ES-1.1, "SI Termination".

~~Using the reference provided, and~~ based on these plant conditions, which ONE of the following describes the status of resetting Safety Injection in order to secure the "A" High Head Safety Injection (HHSI) pump?

The "A" Train of Safety Injection (SI) ____________________

A. can be reset 75 seconds **AFTER** the SI actuation regardless of reactor coolant system pressure.

B. can be reset 75 seconds **AFTER** the SI actuation only if reactor coolant system pressure is > P-11.

C. cannot be reset until "A" RTB is opened **IF** an automatic SI signal is still present.

D. cannot be reset until "A" RTB is opened **AND** all automatic SI signals are cleared.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. SI cannot be reset until the "A" RTB is opened. 75 seconds is plausible since SI cannot be reset until after 75 seconds has elapsed.

**B.** Incorrect. SI cannot be reset until the "A" RTB is opened. 75 seconds is plausible since SI cannot be reset until after 75 seconds has elapsed.

**C.** Correct. According to the reference provided, the RTB (P-4) must be opened AND the SI reset button must be depressed in order to reset SI.

**D.** Incorrect. SI cannot be reset with S/G pressure < 500 psig without opening the "A" RTB. If S/G pressure is > 500 psig, then SI can be reset after a time delay by depressing the SI reset button.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| W/E02 | SI Termination | Ability to operate and / or monitor the following as they apply to the (SI Termination): | 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** 4.0 **Exam Level** RO

**References provided to Candidate** None **Technical References:** UFSAR Functional Diagram 7.3-13, Rev. K; 2SQS-11.1, Rev. 15

**Question Source:** Bank Vision - 1519 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-11.1 15. Describe the control, protection and interlock functions for the control room components associated with SI system, including automatic functions, setpoints and changes in equipment status as applicable: SI actuation and reset.

24. Given the following plant conditions:

- A reactor trip and safety injection occurred.
- The crew is responding to a Small Break Loss of Coolant Accident (SBLOCA).
- All Reactor Coolant Pumps (RCP) have been tripped.
- The crew is depressurizing the Reactor Coolant System (RCS) in accordance with Step 16 of ES-1.2, "Post LOCA Cooldown and Depressurization".
- A PORV is being used to depressurize the RCS.
- During depressurization, Pressurizer (PRZR) level **rapidly** rises to 50%.

Based on these plant conditions, which ONE of the following is the required action **AND** bases for this action?

A. Continue the depressurization to **minimize** break flow.

B. Immediately stop the depressurization to **prevent** upper head void formation.

C. Immediately stop the depressurization to **prevent** PRZR water solid conditions.

D. Stop the depressurization to **allow** restoration of reactor coolant system subcooling.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Although the note prior to step 16 refers to upper head voiding and the plant indications support this is occurring, the correct action is to stop the depressurization as opposed to continue the depressurization. Continued depressurization will decrease break flow which makes this distractor plausible. The background document does allow for continued depressurization if subcooling is lost further improving plausibility.

**B.** Incorrect. Although correct that the depressurization should be immediately stopped, it is to late to prevent upper head formation as a rapidly increasing PRZR level is the indication that head voiding is occurring.

**C.** Correct. The note prior to commencing depressurization in ES-1.2 to refill the PRZR states that a head void may occur during depressurization if RCPs are not running. This will result in rapidly rising PRZR level as hotter water in the reactor head flashes forming an upper head void. This void displaces water into the PRZR. It is important that the candidate understands the bases of this step during this evolution so that depressurization can be stopped quickly to avoid a water solid PRZR. This is in accordance with the bases of the Note prior to step 16.

**D.** Incorrect. Although it is correct that the depressurization will be stopped due to rapidly increasing PRZR level, it is incorrect that it is to restore RCS subcooling. The bases of step 16 states is subcooling is lost during depressurization, it will be re-established after stopping the depressurization based on PZRZ level, which makes this distractor plausible.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| W/E03 | LOCA Cooldown and Depressurization | N/A | Knowledge of specific bases for EOPs. |

**K/A#** 2.4.18 **K/A Importance** 3.3 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.ES-1.2, Rev. 9; 2OM-53B.4.ES-1.2, Rev. 9

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comprehension **10 CFR Part 55 Content:** (CFR 41.10 / 43.1 / 45.13)

**Objective:** 3SQS-53.3 4. State from memory the basis for ALL cautions and notes iaw BVPS-EOP Executive Volume.

25. Given the following plant conditions:

- The Unit has experienced a Steam Generator Tube Rupture coincident with a small break LOCA inside Containment.
- Initially, all equipment functions as designed.
- The crew has transitioned to ECA-3.2, "SGTR with Loss of Reactor Coolant – Saturated Recovery Desired".

While performing the actions of ECA-3.2, "SGTR with Loss of Reactor Coolant – Saturated Recovery Desired", the STA completes another pass through the status trees and reports the following with his/her recommendations:

- NO orange or red path exists.
- A Yellow path on core cooling exists.
- A Yellow path on containment radiation exists.

The STA recommends re-establishing RCS subcooling by transitioning into FR-C.3, "Response to Saturated Core Cooling" and completing all steps necessary to re-establish RCS subcooling.

As the reactor operator at the controls (ATC), do you AGREE or DISAGREE with the recommended procedure transition **AND** WHY?

A. Agree. The Yellow path procedure for core cooling is a higher priority than the ECA procedures.

B. Agree. Restoring RCS subcooling is the highest priority for current plant conditions.

C. Disagree. The Yellow path procedure for CNMT radiation is a higher priority than the ECA procedures.

D. Disagree. Minimizing subcooling is the highest priority for current plant conditions.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Only red or orange path procedures have a higher priority than the ORPs.

**B.** Incorrect. Restoring RCS subcooling is a priority in most EOPs. However, in this case with a SGTR and LOCA, the mitigation strategy is to minimize subcooling to minimize primary to secondary leakage. Therefore there is note at the beginning of FR-C.3 that directs the crew to return to ECA-3.2 because of conflicting priorities.

**C.** Incorrect. Only red or orange path procedures have a higher priority than the ORPs.

**D.** Correct. Restoring RCS subcooling is a priority in most EOPs. However, in this case with a SGTR and LOCA, the mitigation strategy is to minimize subcooling to minimize primary to secondary leakage. Therefore there is note at the beginning of FR-C.3 that directs the crew to return to ECA-3.2 because of conflicting priorities. The reason for this note is that FR-C.3 directs reestablishment of RCS subcooling via SI flow. This action is inconsistent with the action in ECA-3.2 which is to minimize primary to secondary leakage. Therefore the mitigation strategy of ECA-3.2 takes priority over FR-C.3.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| W/E07 | Saturated Core Cooling | Knowledge of the reasons for the following responses as they apply to the Saturated Core Cooling: | RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitation in the facilities license and amendments are not violated. |

**K/A#** EK3.4 **K/A Importance** 3.3 **Exam Level** RO

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

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comprehension **10 CFR Part 55 Content:** (CFR 41.5 / 41.10 / 45.6 / 45.13)

**Objective:** 3SQS-53.3 4. Explain from memory the basis for all cautions and notes, IAW BVPS-EOP Executive Volume.

26. While trying to establish Reactor Coolant System (RCS) flow during a Loss of ALL Normal 4KV Power, which ONE of the following would cause natural circulation flow to **RISE**?

A. **LOWERING** RCS pressure using auxiliary spray.

B. **RAISING** the demand on the Residual Heat Release Valve.

C. **LOWERING** the setpoint on the Condenser Steam Dump Valve Controller.

D. **RAISING** the setpoint on the Steam Generator Atmospheric Relief Valves.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. While it is plausible to use aux spray without forced RCS cooling flow, lowering RCS pressure will reduce subcooling which does not enhance natural circulation cooling.

**B.** Correct. Increasing the steam rate will help establish the required Delta Temperature and this ensures natural circulation cooling of the RCS.

**C.** Incorrect. Condenser steam dumps will be unavailable due to loss of ALL normal power and therefore no condenser availability due to no cooling tower pumps. Lowering the setpoint would increase the cooldown rate and is plausible.

**D.** Incorrect. Although a plausible available method to increase steaming rate, the operator would need to lower the setpoint of the atmospheric dump valve versus raise the setpoint

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| W/E09 | Natural Circulation Operations | Knowledge of the interrelations between the (Natural Circulation Operations) and the following: | Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility. |

**K/A#** EK2.2 **K/A Importance** 3.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.ES-0.2, Rev. 9
2OM-53B.4.ES-0.2, Rev. 9
2OM-53B.5.GI-4, Rev. 0

**Question Source:** Bank Vision - 9342 **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-53.2 11. List from memory the conditions needed to cause/allow natural circulation to occur, IAW BVPS EOP Executive Volume.

27. Given the following plant conditions:

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

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

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

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

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

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

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Containment Radiation levels exceeded 1E + 5 R/hr, so therefore adverse parameters are required.

**B.** Incorrect. Although it is true that the limit of 1E + 5 R/hr was exceeded and also true that the radiation levels are now below this limit, 2OM-53B.5.GI-2 requires that integrated dose remained less than 1E +6 R/hr. This value is not known in the stated plant conditions and until it is known, the operator must continue to use adverse parameters.

**C.** Correct. IAW 2OM-53B.5.GI-2, and in conjunction with justifications above.

**D.** Incorrect. There is no way of determining if integrated dose was exceeded based on stated plant conditions. Additionally, it is not true that whenever dose rate is exceeded that the integrated dose is exceeded.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| W/E016 | High Containment Radiation | Knowledge of the interrelations between the High Containment Radiation and the following: | Facilities heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and the relations between the proper operation of these systems to the operation of the facility. |

**K/A#** EK2.2 **K/A Importance** 2.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53B.5.GI-2, Rev. 0

**Question Source:** Bank Vision - 46225 **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-53.2 15. Define from memory adverse containment conditions IAW BVPS EOP Executive Volume.

28. Given the following plant conditions:

- The plant is at 90% power with all systems in NSA.
- A simultaneous over-current trip and lockout of the following breakers due to **BUS** electrical fault occurs:
  - 2D US SERV TFMR TO 4KV BUS 2C ACB 242D
  - 2D US SERV TFMR TO 4KV BUS 2D ACB 342D

Which ONE of the following describes the Reactor Coolant Pumps (RCPs) that will be running ten (10) seconds after this event takes place?

A. RCP 21A and 21B.

B. RCP 21 A and 21C.

C. RCP 21B and 21C.

D. All RCPs are running.

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. 21A & 21B RCPS are powered from 4KV Bus 2A and 2B which are unaffected by the stated plant conditions.

**B.** Incorrect. "D" USST lockout will prevent "B"SSST from energizing 4KV Bus 2C, therefore 21C will not have power.

**C.** Incorrect. "D" USST lockout will prevent "B"SSST from energizing 4KV Bus 2C, therefore 21C will not have power.

**D.** Incorrect. USST's are placed in service > 65% power. It is plausible that all RCPs would be running if SSST's are in service supplying the 4KV busses.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 003 | Reactor Coolant Pump System (RCPs) | Knowledge of bus power supplies to the following: | RCPs |

**K/A#** K2.01 **K/A Importance** 3.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-36.1.C, Rev. 4

**Question Source:** Bank Vision - 45775 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.7)

**Objective:** 2SQS-6.3 4. 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,

29. Given the following plant conditions:

- The plant is operating at 15% power.
- Preparations are being made to synchronize to the grid.
- A2-4D, "REACTOR COOLANT SEAL TROUBLE" annunciator is received.
- The Reactor Operator (RO) determines that 21B Reactor Coolant Pump (RCP) seal leakoff flow has risen to six (6) GPM.
- The crew has entered AOP-2.6.8, "Abnormal RCP Operation".

Based on these plant conditions, which ONE of the following describes the problem with 21B RCP **AND** REQUIRED procedural action?

A. #1 seal has failed. Trip the reactor, enter E-0, "Reactor Trip or Safety Injection", and trip 21B RCP.

B. #2 seal has failed. Trip the reactor, enter E-0, "Reactor Trip or Safety Injection", and trip 21B RCP.

C. #1 seal is degraded. Monitor seal injection and RCP bearing temperatures to determine additional action.

D. #2 seal is degraded. Monitor seal injection and RCP bearing temperatures to determine additional action.

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. According to A2-4D Annunciator Response, a high seal leakoff @ 5.8 gpm brings in the alarm and is indicative of a #1 seal failure. In accordance with AOP-2.6.8, any leakage > 6 gpm requires a reactor trip per E-0 and a trip of the affected RCP. The upper range of indication in the control room is 6 gpm.

**B.** Incorrect. Correct action, incorrect RCP seal. #2 seal failure would be indicated by a low seal leakoff flow.

**C.** Incorrect. Correct seal, although > 6gpm is indicative of a failure as opposed to degradation. Even if one could argue this finer point, the required action is that of which is required for a #2 seal failure of a minor nature.

**D.** Incorrect. Incorrect seal and incorrect action which is correct for a #2 seal degradation.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 003 | Reactor Coolant Pump System (RCPs) | N/A | Knowledge of annunciator alarms, indication, or response procedures. |

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

**References provided to Candidate** None **Technical References:** 2OM-7.4.AAH, Rev. 22; 2OM-53C.4.2.6.8, Rev. 6

**Question Source:** Bank Vision - 46073 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Analysis **10 CFR Part 55 Content:** (CFR 41.10 / 45.3)

**Objective:**

2SQS-6.3 20. Given a change in plant conditions due to system or component failure, analyze the reactor coolant pump and support system to determine what failure has occurred.

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

30. Given the following plant conditions:

- An ATWS has occurred.
- Both boric acid transfer pumps are tripped and CANNOT be started.

Which boration flowpath is <u>UNAVAILABLE</u> due to the loss of the boric acid transfer pumps?

The flowpath through ____________________

A. 2CHS*MOV350, "Emergency Borate Valve".

B. 2CHS*MOV115C, "Charging Pump Suction from VCT".

C. 2CHS*MOV115B, "Charging Pump Suction from RWST".

D. 2CHS*127, "Boric Acid Hold Tank to Boric Acid TK21A Isolation".

---

**<u>Answer:</u> A**

**Explanation/Justification:**

**A.** Correct. According to Op Manual Figure 7-1A & 7-2, the 2CHS*MOV350 flowpath requires the Boric Acid Transfer Pumps, so therefore is unavailable.

**B.** Incorrect. This plausible flowpath is from the VCT to the suction of the charging pumps. This flowpath would still be available.

**C.** Incorrect. This plausible flowpath is from the RWST to the suction of the charging pumps. This flowpath would still be available.

**D.** Incorrect. This is a fill method from the boric acid holding tanks from Unit 1 to the Boric Acid Tank which is the closest BVPS has to an interconnection with a BWST. Since this is a supply to TK21A, it is a plausible manual source of water which is available.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 004 | Chemical and Volume Control System | Knowledge of the physical connections and/or cause-effect relationships between the CVCS and the following: | BWST |

**K/A#** K1.22 **K/A Importance** 3.4 **Exam Level** RO

**References provided to Candidate** None **Technical References:** OM Fig. 7-1A, Rev. 19; OM Fig. 7-2, Rev. 18

**Question Source:** Modified Bank Vision - 16762 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-7.1 15. Draw and Label the CVCS NSA system flowpath as it applies to a licensed operator and as illustrated on simplified one-line diagrams for RCP Seal Injection, Excess Letdown and CVCS Blender.

31. Given the following plant conditions:

- The plant is in Mode 4 at 300°F cooling down at 50°F/hr.
- "A" Residual Heat Removal System (RHS) is in service.
- "B" RHS is in standby.
- A complete Loss of Containment Instrument Air (IA) occurs.

Assuming no operator action, what operational impact will the Loss of Containment IA have on RHS **AND** Nil Ductility Temperature (NDT)?

| | RHS Operational Impact | NDT Operational Impact |
|---|---|---|
| A. | Both Trains Unaffected | No impact on NDT |
| B. | Maximum Flow through "A" RHS HX | Closer to NDT |
| C. | Minimum Flow through "A" RHS HX | Further from NDT |
| D. | Maximum Flow through "A" & "B" RHS HX | Closer to NDT |

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. RHS is affected by the loss of containment air. Plausible if the candidate does not know the impact on this system and therefore there would be no impact on NDT.

**B.** Correct. A Loss of Containment IA will cause 2RHS*HCV758A, "RHR HX Flow Control Valve" to fail open and 2RHS* FCV605A, "RHR HX Bypass Valve to fail closed. This results in approximately 4400 gpm flow and maximum RHR cooling and thus maximum RCS cooling. NDT for BVPS Unit 2 is currently around 140 F for 1/4T and 129 F for 3/4T (limiting ART values @ 22EFPY). The stated plant condition when the loss of containment IA is 300 F. A maximum cooldown will result in moving closer to the plant current NDT which is below 300 F.

**C.** Incorrect. Opposite effect however plausible if the candidate does not know fail positions of the system.

**D.** Incorrect. Only the "A" Train is effected since the "B" Train is in standby and therefore the "B" RHR Pump is not running and there are no auto starts to place this system in service without operator action. Plausible that if the "A" & "B" train were in service that a cooldown would result in moving closer to current plant NDT.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 005 | Residual Heat Removal System (RHRS) | Knowledge of the operational implications of the following concepts as they apply to RHRS: | Nil Ductility transition temperature (brittle fracture) |

**K/A#** K5.01 **K/A Importance** 2.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.34.2, Rev. 5; Op Manual Figure 10-1, Rev. 16; LRM Fig. 5.2-1, Rev. 62

**Question Source:** Modified Bank Vision - 68032 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-10.1 18. Given a specific plant condition, predict the response of the RHS 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.

32. Given the following plant conditions:

- Outside air temperature is 35 °F and STABLE. Safeguards air temperature is 65 °F and STABLE
- A8-6B, "HEAT TRACING SYSTEM TROUBLE" alarm is received.
- The PCS indicates the trouble is associated with Heat Trace SFGDS PNL A1 (2HTS*PNLA1SG).
- Investigation reveals that the alarm is associated with the Low Head Safety Injection recirculation piping designated 2SIS-008-346-2.
- It is determined that the RTD input for circuit 2HTS*JB-40A & B accurately indicates (45 °F) and the controller setpoint is NSA @ (60 °F).
- No flow through the system has recently occurred and power is available.

Based on these plant conditions, which ONE of the following describes the required action?

Using references provided, the corrective action is to _________.

A. energize both trains of heat trace circuits.

B. raise the heater cycle (on/off) setpoint to clear the alarm.

C. immediately verify system flow through the effected pathway.

D. de-energize the primary and energize the redundant heat trace circuit.

---

Answer: **D**

**Explanation/Justification:**

**A.** Incorrect. Energizing both trains of heat trace is incorrect based on 2OM-45D.3.C Note 2. This note states that Train A & B shall not be energized simultaneously for this circuit.

**B.** Incorrect. Since the RTD is providing an actual reading of 45 F and the controller setpoint is NSA @ (60 F), then the problem is associated with the heat tracing circuit in service not functioning properly. There is no procedural allowance for field adjustment of setpoints.

**C.** Incorrect. With outside temperature @ 35F and local piping temperature reading 45 F, the operability of the LHSI system is not immediately challenged. It is not procedurally driven to take this action based on any of the references provided. It is plausible since this section of heat trace does impact the recirculation of the LHSI which could impact operability if the line were to freeze. This is based on industry OE@ Point Beach.

**D.** Correct. According to 2OM-45.D.4.AAA, troubleshoot to determine whether this is an individual heat trace circuit temperature high/low issue or an individual circuit controller failure. Based on stated conditions, this issue is more associated with a controller failure and therefore, the ARP directs a verification of the redundant circuit. This particular circuit has only one power supply energized at a time, so therefore since Train A has failed the redundant circuit will be energized in accordance with 2OM-45D.3.D Note 2. The operator should have system knowledge of heat trace operations. If the RTD input signal (45 F) is less than the under-temperature setpoint (ie: 60 F based on 2OM-45D.5.B.2), the control annunciator is received.

---

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

**K/A#** A2.07 **K/A Importance** 2.8 **Exam Level** RO

| **References provided to Candidate** | | **Technical References** | |
|---|---|---|---|
| | 2OM-45D.4.AAA, Rev. 7<br>2OM-45D.5.B.2, Rev. 12,<br>2OM-45D.3.C, Rev. 12,<br>2OM-45.5.B.1, Rev. 3<br>OM Figs. 45D 7 & 8, Rev. 30<br>OM Fig. 11-1, Rev.28 | | 2OM-45D.4.AAA, Rev. 7<br>2OM-45D.5.B.2, Rev. 12<br>2OM-45D.3.C, Rev.12<br>2OM-45.5.B.1, Rev. 3<br>OM Fig. 45D 7 & 8, Rev. 30<br>OM Fig. 11-1, Rev. 28<br>2OM-45D.1.D, Issue 4, Rev. 0<br>13-2, 2SQS-45D.1, Rev. 6 |

**Question Source:** Modified Bank Vision - 67505 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.5 / 45.5)

**Objective:** 2SQS-11.1 2SQS-45D.1 Given a change in plant conditions due to a system or component failure, analyze the Safety Injection System to determine what failure has occurred. Given a Heat Tracing alarm condition and using the ARP, determine the appropriate alarm response, including automatic and operator actions in the control room.

33. Given the following plant conditions:

- A Load Rejection has occurred.
- Pressurizer (PRZR) PORV operation has resulted in high pressure and temperature in the PRZR Relief Tank.
- Annunciator A4-3H, "PRESSURIZER RELIEF TANK TROUBLE" is received due to high tank temperature above 125°F.
- The PORV has reseated.

Which ONE of the following describes the operational design features which provide PRZR Relief Tank Cooling?

2RCS-MOV516, "PRZR Relief Tank Spray Valve"____ (1) ____.
2RCS-AOV519, "PRZR Relief Tank Primary Grade Makeup Water Inlet Valve"____ (2) ____.

A. (1) automatically opens
(2) automatically opens

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

C. (1) must be manually opened
(2) automatically opens

D. (1) must be manually opened
(2) must be manually opened

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Neither valve is designed to automatically open.
**B.** Incorrect. 2RCS-MOV516 must be manually opened.
**C.** Incorrect. 2RCS-AOV519 must be manually opened.
**D.** Correct. Neither valve is designed to automatically open. In accordance with 2OM-6.4AAY, both valves are opened to reduce tank temperature.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 007 | Pressurizer Relief Tank/Quench Tank System (PRT) | Knowledge of PRTS design feature(s) and/or interlocks which provide for the following: | Quench Tank Cooling |

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

**References provided to Candidate** None **Technical References:** 2OM-6.4AAY, Rev. 8

**Question Source:** Modified Bank Vision - 45761 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.7)

**Objective:** 2SQS-6.4 6. Given a change in plant conditions, describe the response of the PRZR and PRZR Relief System field indications and control loops, including all automatic functions and changes in equipment status.

34. How do the following valves function to control Turbine Plant Component Cooling Water (TPCCW) temperature?

    2CCS-TCV215, "TPCCW Heat Exchanger (HX) Bypass Temperature Control Valve"
    2CCS-DCV215, "TPCCW Heat Exchanger (HX) Differential Pressure Control Valve"

    As ___ (1) ___ modulates closed, the differential pressure (DP) across the HX ___ (2) ___ which functions to control ____ (3) ____ flow through the TPCCW HX.

A. (1) 2CCS-<u>TCV</u>215
(2) increases
(3) Secondary Component cooling

B. (1) 2CCS-<u>TCV</u>215
(2) decreases
(3) Service Water cooling

C. (1) 2CCS-<u>DCV</u>215
(2) increases
(3) Service Water cooling

D. (1) 2CCS-<u>DCV</u>215
(2) decreases
(3) Secondary Component cooling

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. In accordance with the 2OM-28.1.D, as 2CCS-TCV215 modulates closed, the DP across the CCS HX increases. This in turn controls the amount of CCS flow through the CCS HX which is the medium for controlling system temperature.

**B.** Incorrect. Correct Valve. Incorrect DP response. Incorrect cooling water medium. Service Water is the actual cooling medium for CCS, however, it is not modulated.

**C.** Incorrect. Incorrect valve and cooling water medium as explained above (CCS versus SWS), however DP response is correct.

**D.** Incorrect. Incorrect valve and incorrect DP response, however, correct cooling medium.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 008 | Component Cooling Water System (CCWS) | N/A | Knowledge of the purpose and function of major system components and controls. |

**K/A#** 2.1.28 **K/A Importance** 4.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-28.1.D, Issue 4, Rev. 1; Op Manual Fig. 28-1

**Question Source:** Modified Bank Vision - 1649 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.7)

**Objective:** 2SQS-28.1 1. Describe the function of the Turbine Plant Component Cooling Water system and the associated major components as documented in Chapter 28 of the Operating Manual.

35. The Unit is at 80% power with all systems in NSA when the following alarms are received:

- A6-1H, "PRI COMPONENT COOLING WATER SYSTEM TROUBLE"
- A4-5C, "RADIATION MONITORING LEVEL HIGH"

Primary Component Cooling (CCP) Surge Tank level is 70% and slowly rising with the CCP Surge Tank Level Control Valves closed.

(1) Which ONE of the following is the cause of these plant conditions?

**AND**

(2) What procedure entry will be required?

A. (1) System expansion due to temperature increase.
(2) Reference 2OM-15.4.AAC, "Alarm Response for CCP System Trouble" to check system parameters.

B. (1) In-service CCP Heat Exchanger Tube Leak.
(2) Reference AOP 2.15.1, "Loss of Primary Component Cooling Water" to validate and isolate the leak.

C. (1) CVCS Non-Regenerative Heat Exchanger Tube Leak.
(2) Reference AOP 2.15.1, "Loss of Primary Component Cooling Water" to validate and isolate the leak.

D. (1) RCP Thermal Barrier Heat Exchanger Tube Leak.
(2) Reference E-0, "Reactor Trip or Safety Injection" to trip the reactor and then trip the affected RCP.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Rising system temperature will result in increasing surge tank level and 2OM-15.4.AAC does provide mitigating actions for this condition, however, the candidate must rule this out based on radiation level.

**B.** Incorrect. Correct procedure, however, a leak in the CCP HX would cause a dropping level in CCP Surge Tank with no radiation alarm.

**C.** Correct. A CVCS NRHX tube leak would cause the higher pressure from this system to increase **CCP** surge tank level and would also cause a radiation monitor high level. AOP 2-15-1 is the correct procedural guidance to determine if this is the correct leak location and for other actions.

**D.** Incorrect. Could certainly be an RCP Thermal Barrier HX tube leak, this would result in increasing surge tank level and a radiation monitoring alarm. Incorrect procedural guidance since AOP 2-15-1 would direct verification that auto isolation has occurred or a manual isolation if <58 gpm. The reactor would not need to be tripped because the RCP can still be operated without CCP cooling as long as seal injection is not lost.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 008 | Component Cooling Water System (CCWS) | Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: | PRMS alarm. |

**K/A#** A2.04 **K/A Importance** 3.3 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-15.4.AAC, Rev. 6; 2OM-53C.4.2.15.1, Rev. 3

**Question Source:** Modified Bank Vision - 45762 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-15.1 Given in-leakage or out-leakage to/from the CCP system, describe all the means by which the in-leakage can be detected, the consequences of the leakage and the actions taken to correct the leakage.

36. The Unit is operating at 100% power.

The Pressurizer (PRZR) heater control switches are positioned as follows:

- Group "A": AUTO after STOP.
- Group "B": AUTO after STOP.
- Group "C": ON.
- Group "D": AUTO after START.
- Group "E": AUTO after STOP.

The control room crew has just completed a down-power to 80% in accordance with 2OM-52.4.B, "Load Following", for scheduled condenser water box cleaning.
A loss of 480VAC Emergency Bus 2N occurs.

What PRZR heater manipulation(s), if any, will be required to equalize Reactor Coolant System and PRZR boron concentration within 50 ppm?

A. Place Group "A" or "B" in AUTO after START.

B. Place Group "B" or "E" in AUTO after START.

C. Re-energize Group "C" by placing the control switch back to the ON position.

D. No actions are required because Group "C" and Group "D" remain energized.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Group A heaters are powered from Bus N which has no power.

**B.** Correct. PRZR heaters are required to ensure RCS boron concentration remains equalized based on P & L's of Load Following procedure (2OM-52.4.B) Both "B" and "E" PRZR heaters are powered from the opposite emergency bus P, so therefore either can be energized to satisfy the requirement.

**C.** Incorrect. Bank C heaters would remain energized during this transient since this bank is powered from non emergency 480VAC Bus C.

**D.** Incorrect. Group "D" heaters are powered from Bus N so therefore would not be energized PRZR. It is true that Group "C" heaters are energized.

---

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 010 | Pressurizer Pressure Control System (PZR PCS) | Knowledge of the bus power supplies to the following: | PZR Heaters |

| | | | | | |
|---|---|---|---|---|---|
| **K/A#** | K2.01 | **K/A Importance** | 3.0 | **Exam Level** | RO |
| **References provided to Candidate** | | | None | **Technical References:** | 2OM-52.4.B, Rev. 51<br>2OM-6.3C, Rev. 12 |
| **Question Source:** | Bank | Vision - 16943 | | **Level Of Difficulty: (1-5)** | |
| **Question Cognitive Level:** | | Lower - Memory | | **10 CFR Part 55 Content:** | (CFR 41.7) |

**Objective:** 2SQS-6.4 4. Identify the power supplies for the components identified on the NLO NSA flow-path drawing which are powered from the class 1E electrical distribution system.

37. Given the following plant conditions:

- The Unit is at 100% power with all systems NSA.
- Reactor Coolant System (RCS) Pressure is 2235 psig and STABLE.
- RCS Temperature is 547°F and STABLE.
- 2RCS*PT444, "Pressurizer (PRZR) Control Channel", fails **LOW** over a ONE (1) minute period.

With no operator action, actual RCS pressure **initially** will be ____ (1) ____ and ____ (2) ____.

A. (1) slowly RISING
(2) PRZR heaters will be ON and BOTH PRZR Spray Valves will be CLOSED.

B. (1) STABLE
(2) PRZR heaters and BOTH PRZR Spray Valves will remain in their NSA positions.

C. (1) slowly LOWERING
(2) ONE (1) PRZR PORV and BOTH PRZR Spray Valves will be OPEN.

D. (1) rapidly LOWERING
(2) TWO (2) PRZR PORVs will be OPEN.

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. The result of 2RCS*PT444 failing low is BOTH PRZR spray valves will be closed and PRZR heaters will be ON. This will cause RCS pressure to rise slowly. The immediate pressure rise was validated on the simulator.

**B.** Incorrect. These indications are indicative of 2RCS*PT445 failing in the low direction as opposed to 2RCS*PT444.

**C.** Incorrect. These indications are indicative of 2RCS*PT444 failing in the high direction. Plausible because with 2RCS*PT444 failed low, heaters will energize over a period of time and eventually two PORVs will open which results in lowering RCS pressure. The other PORV and spray valves which are operated off the Master Pressure Controller as a result of 2RCS*PT444 failing low will not operate.

**D.** Incorrect. These indications are indicative of 2RCS*PT445 failing in the high direction. Plausible because with 2RCS*PT444 failed low, heaters will energize and over a period of time RCS pressure will rise until eventually two PORVs will open which results in lowering RCS pressure..

---

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

**K/A#** K3.01 **K/A Importance** 3.8 **Exam Level** RO

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

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.7 / 45.6)

**Objective:** 2SQS-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.

38. Given the following plant conditions:

- The Unit is operating at 25% power.
- Rod Control is in Manual.
- I & C is performing a Channel Operating Test on 2RCS-P456, "Pressurizer (PRZR) Pressure Loop Protection Channel II".
- The channel has been removed from service and all applicable bistables have been placed in the tripped position in accordance with 2MSP-6.13-I.
- 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 (RCS LOOP B OT ΔT REACTOR TRIP) status light will be ____ (1) ____ and a reactor trip ____ (2) ____ occur.

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

B. (1) OFF
(2) will

C. (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 2MSP-6.13-I. When performing the COT on 2RCS-PT456, this also 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.

**D.** Incorrect. Incorrect bi-stable light status. Incorrect reactor status.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 012 | Reactor Protection System (RPS) | Knowledge of the effect of a loss or malfunction of the following will have on the RPS: | Bistables and bistable test equipment |

**K/A#** K6.01 **K/A Importance** 2.8 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-6.4.IF, Rev. 12
2MSP-6.13-I, Issue 4, Rev. 10
2OM-6 Figure 6-62, Issue 1, Rev. 6
UFSAR Figure 7.3-10, Rev. 9

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.5 / 45.7)

**Objective:** 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.

39. Given the following plant conditions:

- A Large Cold Leg Break Loss of Coolant Accident (LOCA) occurred just prior to a refueling outage following 400 days at full power.
- Six (6) hours into the accident, the crew transitions to ES-1.4, "Transfer to Hot Leg Recirculation" but is unable to reset Safety Injection (SI) Recirc Mode due to a malfunction of the Engineered Safety Features Actuation System (ESFAS).

Based on these plant conditions AND according to ES-1.4 background document, what is the adverse effect of **NOT** being able to establish the Hot Leg Injection Flowpath for a prolonged period?

A. Fouling of the core heat transfer surfaces due to dilution of boric acid.

B. Reflux cooling could be lost due to boron precipitation in the hot leg nozzles.

C. Debris from the in-core sump could block coolant flow by blocking the injection suction flow path.

D. Blocked coolant flow channels that could lead to inadequate core cooling due to boron precipitation.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Fouling of the core heat transfer surfaces is a result of boron precipitation not dilution.

**B.** Incorrect. Boron precipitation is a concern in the core not the hot legs.

**C.** Incorrect. In-core sump blockage is a common industry concern, however, not the reason for adverse effects if hot leg recirculation is not established.

**D.** Correct. According to ES-1.4 Background document, the purpose of hot leg injection is to terminate boiling in the core and to prevent boron precipitation in the core for a large cold leg break LOCA. If the boron concentration reaches the solubility limit, boron will begin to precipitate out of solution, forming a solid that could block the coolant flow channels in the core. This could lead to Inadequate core cooling.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 013 | Engineered Safety Features Actuation System (ESFAS). | Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: | Fuel |

**K/A#** K3.01 **K/A Importance** 4.4 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53B.4.ES-1.4, Rev. 5

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-1.1 9. Given a RPS Logics or ESF configuration and without referenced materials, describe the RPS & ESF control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable to a LOCA.

40. Given the following plant conditions:

- The Unit was operating at 100% power.
- The reactor tripped on low Pressurizer (PRZR) pressure.
- Tavg is 543 °F and STABLE.
- PRZR pressure stabilized at 1700 psig.
- Containment pressure is 2 psig and slowly RISING.
- The LOWEST Steam Generator pressure is 930 psig.
- All Steam Generator NR levels are 60% and STABLE.
- No operator actions have been taken.

Which of the following Engineered Safety Features Actuation (ESFAS) systems will be actuated for these conditions?

A. Safety Injection <u>AND</u> CIB.

B. Safety Injection, CIA, <u>AND</u> Full FWI.

C. Main Steam Line Isolation <u>AND</u> CIA.

D. Safety Injection, CIA, CIB, <u>AND</u> Full FWI.

---

**<u>Answer:</u> B**

**Explanation/Justification:**

**A.** Incorrect. Although SI would be actuated, CIB will not actuate unless containment pressure is > 11.1 psig.

**B.** Correct. When PRZR pressure drops below 1845 psig, Safety Injection would actuate, which would result in a CIA and FWIS.

**C.** Incorrect. CIA would be actuated, however, MSLI does not actuate unless containment pressure is > 7psig or S/G pressure < 500 psig.

**D.** Incorrect. SI, CIA and FWI would have actuated, however, containment pressure is < 11.1 psig so therefore CIB should not have actuated.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 013 | Engineered Safety Features Actuation System (ESFAS) | N/A | Ability to evaluate plant performance and make operational judgements 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** None **Technical References:** 3SQS-1.1, Rx Trip/ECCS Setpoints

**Question Source:** Modified Bank Vision - 45765 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Analysis **10 CFR Part 55 Content:** (CFR 41.5 / 43.5 / 45.12 / 45.13)

**Objective:** 3SQS-1.1 9. Given a RPS trip logic and ESFAS configuration and without referenced material, describe the RPS & ESFAS control room response to the following actuation signals, including automatic functions and changes in plant equipment status as applicable: LOCA

41. Which ONE of the following signals will automatically close 2SWS*MOV152-1/2SWS*MOV152-2, CNMT Air Recirc Clg Coils Outside/Inside CNMT Serv Wtr Inlet Valves **AND** 2SWS*MOV155-1/2SWS*MOV-155-2, CNMT Air Recirc Clg Coils Outside/Inside CNMT Serv Wtr Outlet Valves ?

A. SI **ONLY**.

B. CIA **ONLY**.

C. CIB **ONLY**.

D. MSLI **ONLY**.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Cooling remains aligned.

**B.** Incorrect. Cooling remains aligned.

**C.** Correct. According to 2OM-53A.1.A.0.5 and OP Manual Fig. 29-4, 2SWS*MOV152-1/2 and 2SWS*MOV155-1/2 close upon a receipt of a CIB signal. This isolates chilled water cooling to the containment air recirculation heat exchangers which provide cooling to containment.

**D.** Incorrect. Cooling remains aligned.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 022 | Containment Cooling System (CCS) | Knowledge of the physical connections and/or cause-effect relationships between CCS and the following systems: | Chilled Water |

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

**References provided to Candidate** None **Technical References:** 2OM-53A.1.A-0.5, Rev. 2
OP Manual Fig. 29-4, Rev. 13

**Question Source:** Bank Vision - 46455 **Level Of Difficulty: (1-5)**

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

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

42. Given the following plant conditions:

- A Loss of Coolant Accident (LOCA) has occurred.
- Reactor Coolant System (RCS) pressure is 425 psig and stable.
- Containment pressure is 11.5 psig and slowly dropping.
- All equipment is operating as designed and no operator action has been taken.

Based on these plant conditions, what will be the status of the following Service Water Cooling Valves?

2SWS*MOV103A/B, "Recirculation Spray Heat Exchanger Service Water Supply Header A/B Isolation Valves" ___ (1) ___.

2SWS*MOV104A/B/C/D, "Recirculation Spray Heat Exchanger Cooling Water Supply Valves" __ (2) __.

A. (1) reposition OPEN
(2) reposition OPEN

B. (1) will remain OPEN
(2) reposition OPEN

C. (1) reposition OPEN
(2) will remain OPEN

D. (1) will remain CLOSED
(2) will remain OPEN

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. 2SWS*MOV104A/B/C/D are NSA open and receive no auto signal. It is plausible that if the candidate does not know system cause-effect relationships and believe that these valves auto open. It is correct that 2SWS*MOV103A/B auto open.

**B.** Incorrect. Opposite of actual system NSA and operations.

**C.** Correct. NSA position for 2SWS*MOV103A/B is closed. These valves auto open on CIB which occurs @ 11.1 psig containment pressure. 2SWS*MOV104A/B/C/D are NSA open and receive no auto signal.

**D.** Incorrect. It is plausible that the candidate does not know when CIB occurs, so therefore this is the NSA configuration of the plant.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 026 | Containment Spray System (CSS) | Knowledge of the physical connections and/or cause-effect relationships between CSS and the following system: | Cooling Water |

**K/A#** K1.02 **K/A Importance** 4.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-30.1.D, Rev. 7
2OM-30.3.B.1, Rev. 39
2OM-53A.1.A-0.5, Rev. 2
OM Fig. 30-1, Rev. 32
OM Fig. 30-3, Rev. 22

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-30.1 14. Given a service water configuration and without referenced materials, describe the SW control room response to the following actuation signals, including automatic functions and changes in plant equipment status as applicable: SI, CIA, and CIB.

43. What is the **MAXIMUM** Reactor Coolant System (RCS) Technical Specification (T.S.) cooldown (C/D) rate that can be established **AND** what is the bases of this cooldown rate according to Beaver Valley Power Station Technical Specfication.?

The maximum T.S. RCS C/D rate is ____ (1) ____ **AND** the TS bases for this C/D rate is to provide a margin to brittle fracture of the ____ (2) ____.

A. (1) 90°F/hr
(2) RCS and Pressurizer

B. (1) 90°F/hr
(2) RCS and Reactor Vessel

C. (1) 100°F/hr
(2) RCS and Pressurizer

D. (1) 100°F/hr
(2) RCS and Reactor Vessel

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. 90 F/hr is the maximum allowed administrative cool-down rate in accordance with 2OM-52.4.R.1F. TS bases specifically states that this limit does not apply to the PRZR.

**B.** Incorrect. 90 F/hr is the maximum allowed administrative cool-down rate in accordance with 2OM-52.4.R.1F Correct bases for TS C/D rate of 100 F/hr limit.

**C.** Incorrect. 100 F/hr is the TS maximum allowed C/D rate for the RCS. Bases does not include the PRZR.

**D.** Correct. According to TS 3.4.3, 100 F/hr is the maximum allowed RCS cooldown rate. The bases of this C/D rate is so that the RCS is not operated under conditions that can result in brittle fracture of the RCPB. Violating LCO limits places the reactor vessel outside the bounds of the stress analyses. The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the Reactor Coolant Pressure Boundary.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 039 | Main and Reheat Steam System (MRSS) | Knowledge of the operational implications of the following concepts as they apply to the MRSS: | Bases for RCS cooldown limits |

**K/A#** K5.05 **K/A Importance** 2.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** BVPS TS 3.4.3, Amend 278/161
LRM 5.2.1.1, Rev. 62
BVPS TS 3.4.3 Bases, Rev. 0

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-ITS.007 2. State the purpose of each TS 3.4 specification as described in the Applicable Safety Analysis section of the TS Bases.

44. Given the following plant conditions:

- The Unit is operating at 100% with all systems in NSA.
- A Feed Flow instrument failure on the 21B Steam Generator (S/G) causes 2FWS*FCV488, "Main Feed Regulating Valve" to go full open.
- A turbine/reactor trip occurs from the resultant 21B Hi-Hi S/G water level.
- Reactor Coolant System (RCS) temperature is 547°F and stable.
- All equipment operates as designed and no operator action has occurred.

Based on these conditions, what will be the status of the following feedwater components, **30 seconds after the trip**?

| | 2FWS*FCV488 (Main FRV) | 2FWS*FCV489 (Main FRV Bypass) | Main Feedwater Pumps |
|---|---|---|---|
| A. | OPEN | CLOSED | RUNNING |
| B. | CLOSED | CLOSED | TRIPPED |
| C. | CLOSED | CLOSED | RUNNING |
| D. | CLOSED | OPEN | TRIPPED |

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. 2FWS*FCV488 closes on a full FWI signal. It is plausible that this valve would reopen once the Hi-HI S/G water level signal clears as it taps off above the set/reset device. It is possible that on a reactor trip, the S/G water level will shrink down far enough to clear the Hi-HI signal and thus any feedwater/steamflow mismatch would reopen 2FWS*FCV488. The MFP will trip due to Full FWI and will not restart without operator action.

**B.** Correct. In accordance with 2OM-24.4.N, Feedwater Isolation occurs on a Hi-HI S/G level. This results in MFRV's and MFRBV's closing and the MFP's tripping.

**C.** Incorrect. This would be the correct response for a partial FWI signal. On a Low Tavg (< 554 F) and a reactor trip, a partial feedwater isolation will occur. As a result, 2FWS*FCV488 will close and not reopen. MFRBV's NSA position is closed in manual, so therefore 2FWS*FCV489 will be closed. The MFP will remain running on a partial FWI.

**D.** Incorrect. Incorrect 2FWS*FCV489 position. Correct MFP status. Correct 2FWS*FCV488 status. Plausible if the candidate does not understand the feedwater isolation logic.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 059 | Main Feedwater (MFW) System | Ability to manually operate and monitor in the control room: | Initiation of automatic feedwater isolation |

**K/A#** A4.12 **K/A Importance** 3.4 **Exam Level** RO

**References provided to Candidate** None **Technical References:** UFSAR Figure 7.3-18, Rev. 9

**Question Source:** Modified Bank Vision - 68086 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comphrension **10 CFR Part 55 Content:** (CFR 41.7 / 45.5 – 45.8)

**Objective:** 2SQS-24.1 16. Given a specific plant condition, predict the response of the MFW, S/U FW, AFW, or SGWLC systems 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.

45. Given the following plant conditions:

- The Unit is at 100% power with all systems in NSA.
- 2RCS*P21A, "A Reactor Coolant Pump", trips on overcurrent.
- A reactor trip occurs.
- Tavg is 547 °F and STABLE.
- The crew is performing Immediate Operator Actions of E-0, "Reactor Trip or Safety Injection", and **no operator actions** have been taken.
- All equipment functions as designed.

Which ONE of the following describes the Feedwater System response for these plant conditions several minutes after the reactor trip occurs?

Feedwater flow to the Steam Generators (S/Gs) will be provided by _______________.

A. Auxiliary Feedwater **ONLY**.

B. Main Feedwater via Main Feedwater Bypass Valves **ONLY**.

C. Auxiliary Feedwater **AND** Main Feedwater via Main Feedwater Bypass Valves.

D. Auxiliary Feedwater **AND** Main Feedwater via Main Feedwater Regulating Valves.

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. Motor Driven and Turbine Driven AFW pumps auto start on several signals by design. A reactor trip from 100% power will result in low-low S/G water levels in any 2/3 S/Gs which is one of the auto starts. Main Feedwater will be available as long as no safety injection or Hi-HI water level exists. With Tavg 547 F and stable, it is clear that no SI has occurred. A trip of the RCP results in a reactor trip which will cause S/G shrink in the unaffected 21B and 21C S/Gs. Water level in 21A S/G will be higher due to the RCP trip however will NOT approach 90% (Hi-HI level). The MFRBVs are available, however, NSA @ 100% MFRBV controllers are in MANUAL and closed, therefore will NOT be providing fill to the S/Gs without operator action. Therefore ONLY AFW is available to fill S/Gs given these plant conditions.

**B.** Incorrect. It is correct that MFW and bypass valves are available with manual operation to feed the S/Gs, however, AFW has also auto started based on design features as discussed in the correct answer explanation.

**C.** Incorrect. Correct that AFW and MFW are available, however, the MFRVBVs controllers are NSA Closed and in MANUAL at 100% power and without operator action will NOT be filing S/Gs.

**D.** Incorrect. It is correct that AFW is providing feedwater to the S/Gs, however, a partial feedwater isolation has occurred. With reactor trip breakers open (P-4) and Tavg < 554 F, the Main Feedwater Regulating Valves are interlocked closed and cannot be opened.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 059 | Main Feedwater (MFW) System | Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: | Feedwater fill for S/Gs upon loss of RCP(s) |

**K/A#** K4.13 **K/A Importance** 2.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2SQS-24.1, Rev. 22; 2OM-52.4.A, Rev. 69

**Question Source:** Modified Bank Vision - 45756 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Analysis **10 CFR Part 55 Content:** (CFR 41.7)

**Objective:** 2SQS-24.1 5. Given a change in plant conditions, describe the response of the MFW, S/U FW, AFW, or SGWLC systems field indication and control loops, including all automatic functions and changes in equipment status.

46. The plant is operating at 100% full power all systems in NSA EXCEPT Turbine Driven AFW Pump [2FWE*P22] is on clearance.

- A 750 gpm Steam Generator Tube Rupture occurs in the "C" Steam Generator, causing a reactor trip and safety injection.
- All systems respond as designed.
- **BOTH** AFW Throttle Valves [2FWE*HCV100A(B)] for the "C" Steam Generator have been throttled to 50% OPEN.
- All SG NR levels are greater than 12%.

The crew has entered E-3, Steam Generator Tube Rupture and has progressed to the step to isolate AFW flow to the ruptured Steam Generator.

- AFW Throttle Valve [2FWE*HCV100A] is manually closed from the benchboard.
- 2MCC*E14 de-energizes and power is lost to AFW Throttle Valve [2FWE*HCV100B].
- SI has **NOT** been Reset

(1) How will AFW Throttle Valve [2FWE*HCV100B] respond to this loss of power?
(2) IAW E-3, "Steam Generator Tube Rupture" what procedural action, if any, will be **REQUIRED** to isolate AFW to the Ruptured Steam Generator?

A. (1) AFW Throttle Valve [2FWE*HCV100B] will fail to 100% OPEN.
(2) Dispatch an operator to locally CLOSE AFW Throttle Valve [2FWE*HCV100B].

B. (1) AFW Throttle Valve [2FWE*HCV100B] will fail CLOSED.
(2) NO additional actions are necessary.

C. (1) AFW Throttle Valve [2FWE*HCV100B] will fail AS-IS.
(2) Do NOT reset SI, and Place AFW Pump 2FWE*P23B in pull-to-lock.

D. (1) AFW Throttle Valve [2FWE*HCV100B] will fail AS-IS.
(2) Reset SI, and Place AFW Pump 2FWE*P23B in pull-to-lock.

Answer: **D**

**Explanation/Justification:**

**A.** Incorrect. AFW throttle valves will fail as is upon loss of power. Incorrect action IAW E-3 step 5.b RNO. This has been a recent change to the EOP to expedite the time to isolate AFW to a ruptured SG. The action to locally close the valve will work to isolate the valve, but ONLY after first attempting to S/D the motor driven AFW pump.

**B.** Incorrect. AFW throttle valves will fail as is upon loss of power. Correct action for failed closed.

**C.** Incorrect. Correct failure mode for AFW throttle valves. Incorrect action IAW E-3 step 5.b RNO. This has been a recent change to the EOP to expedite the time to isolate AFW to a ruptured SG. The step requires SI reset before placing the pump in PTL.

**D.** Correct failure mode for AFW throttle valves. Correct action IAW E-3 step 5.b RNO. This has been a recent change to the EOP to expedite the time to isolate AFW to a ruptured SG. The step requires SI reset before placing the pump in PTL. The candidate must predict the impact of the electrical failure to the hydraulic motors within the actuator. Then the candidate must be familiar with (from memory) the procedural guidance to isolate AFW flow to a ruptured SG when it cannot be accomplished from the bench board control switch.

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 061 | Auxiliary/Emergency Feedwater (AFW) System | Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on these predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: | Air or MOV failure |

**K/A#** A2.07 **K/A Importance** 3.4 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.E-3, Rev. 152OM-24.3.C, Rev.14 , 2SQS-24.1 PPT, ,Rev. 22

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.5 / 43.5 / 45.3 / 45.13)

**Objective:** 2SQS-24.1 15. Given a MFW, S/U FW, AFW, or SGWLC system configuration and without referenced material, describe the associated system's control room response to the following off-normal conditions, including auto functions and changes in equipment status as applicable: Loss of IA, Loss of electrical power.

47. Given the following plant conditions:

- The Unit is operating at 100% power.
- 2OST-36.2, "Emergency Diesel Generator (2EGS*EG2-2) Monthly Test" is in progress.
- 2-2 EDG 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 2-2 EDG **AND** what is the significance of operating the EDG above 4535 KW for extended periods of time?

The response of 2-2 EDG is that ___ (1)___ **AND** the significance of operating this EDG > 4535 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 LOWERS 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 LOWERS.
(2) accumulation of combustion and lubricating products in the exhaust system

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**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 (3814 to 4238 KW) which is more restrictive than the rated load in 2OST-36.2 (4535 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. KW output will rise when the EDG attempts to raise grid frequency. 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. 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. KW will rise. Reason for load limit is incorrect as explained above.

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| 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
2OST-36.2, Rev. 59
TS 3.8.1 & Bases, Amend. 278/161 Rev. 0

**Question Source:** Modified Bank Vision - 45778 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comprehension **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.

48. The plant is operating at 60% power when the following alarm is received:

- A8-9A, "125V DC BUS 2-1 TROUBLE".

The following control room indications are present:

- 125 VDC Bus 2-1 voltage indicates 124 VDC and slowly dropping.
- Battery Charger Breaker BAT*CHG 2-1 Green light is LIT and Red light is NOT LIT.
- Battery Breaker 2BAT*BKR 2-1 Green light is NOT LIT and Red light is LIT.

Based on these indications, which ONE of the following describes the 125VDC BUS 2-1 status?

A. Station Battery 2-1 has failed. Battery Charger 2-1 is supplying the 125VDC Bus 2-1.

B. Battery Charger 2-1 has failed. Station Battery 2-1 is supplying the 125VDC Bus 2-1.

C. Battery Charger 2-1 is supplying the 125VDC Bus 2-1 with lower than normal voltage.

D. Station Battery 2-1 <u>and</u> Battery Charger 2-1 have failed. 125VDC BUS 2-1 is lower than normal voltage.

---

**<u>Answer:</u> B**

**Explanation/Justification:**

**A.** Incorrect. If the station battery failed, the bus voltage would be higher since the battery charger supplies the bus at about 136 VDC. Both sets of indicating lights for the battery BKR and charger would be RED.

**B.** Correct. These indications are consistent with a failed battery charger and the 125VDC Bus being supplied by its associated battery. Normally the battery chargers and rectifiers supply voltage to the battery bus while simultaneously supplying a float charge to the battery to maintain the battery in a fully charged state. In this configuration normal bus voltage is about 136 VDC and both sets of indicating lights for the battery BKR and charger would be RED. Beaver Valley Unit 2 does not have control room indication of battery voltage. However, by disconnecting the battery charger and allowing the battery to carry the DC bus, we have indication of battery voltage thru the DC bus voltage indicator.

**C.** Incorrect. The battery charger is not supplying the bus since its respective indicating light is GREEN.

**D.** Incorrect. There would not be the voltage indicated in the control room if both the battery and battery charger failed.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 063 | DC Electrical Distribution System | Ability to manually operate and / or monitor in the control room: | Battery voltage indicator |

**K/A#** A4.02 **K/A Importance** 2.8 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-39.1.B, Issue 4, Rev. 0; 3SQS-39, Rev. 7

**Question Source:** Bank Vision - 45979 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Analysis **10 CFR Part 55 Content:** (CFR 41.7 / 45.5 / 45.8)

**Objective:** 3SQS-39.1 18. Given a 125VDC Distribution System configuration, and without reference material, describe the 125VDC Distribution System control room response to the following malfunctions, including automatic functions and changes in equipment status. Loss of Station Battery, Loss of AC Power, Loss of DC Bus

49. Given the following plant conditions:

- The plant is operating at 100% power with all systems NSA.
- A Loss of 125VDC Bus 2-2 occurs.

How are the following components affected by the Loss of 125VDC Bus 2-2?

(1) SG Main Feed Regulating Valves
(2) Main Steam Isolation Valves

A. (1) Fail CLOSED
(2) Fail AS IS

B. (1) Fail CLOSED
(2) Fail CLOSED

C. (1) Fail AS IS
(2) Fail CLOSED

D. (1) Fail OPEN
(2) Fail AS IS

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Correct fail position for MFRVs. Incorrect fail position for MSIVs. Plausible if candidate does not know power supply. Also plausible in that some plant are designed to fail as is.

**B.** Correct. According to the loss of DC Bus 2-2 AOP (2OM-53C.4.2.39.1B) 125DC Bus 2-2 provides DC control power to the MFRVs and MSIVs. The effect of a loss of power to these valves is that they fail to the CLOSE position.

**C.** Incorrect. Correct fail position for MSIVs. Incorrect fail position for MFRVs. Plausible in that other power supply failures cause MFRVs to lock in their as is position.

**D.** Incorrect. MFRVs fail closed as opposed to open. It is plausible that the system is designed to provide flow to the S/Gs upon failure. MSIVs fail closed. Refer to above plausibility statement.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 063 | DC Electrical Distribution | Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: | Components using DC control power |

**K/A#** K3.02 **K/A Importance** 3.5 **Exam Level** RO

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

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**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

50. What is the electrical power supply for the 2-1 Emergency Diesel Generator (EDG) Fuel Oil Transfer Pump (2EGF*P21A)?

A. MCC*2-E08

B. PNL-DC2-07

C. MCC*2-E07

D. PNL*VITBS2-1D

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**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. This is the power supply to 2EGF*P21C and 2EGF*P21D, opposite EDG fuel oil transfer pumps.
**B.** Incorrect. EDG Fuel Oil Transfer Pump Strainer High D/P alarm and other EDG alarms come from this power supply.
**C.** Correct. According to 2OM-36.1.C, 2EGF*P21A is powered from MCC*2-E-07
**D.** Incorrect. This Vital Bus supplies vital instrumentation and is associated with Bus 2-1 which is indirectly associated with EDG 2-1.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 064 | Emergency Diesel Generators (ED/G) | Knowledge of bus power supplies to the following: | Fuel oil pumps |

**K/A#** K2.02 **K/A Importance** 2.8 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-36.1.C, Rev. 4

**Question Source:** Bank Vision - 9011 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.7)

**Objective:** 3SQS-36.1 No K/A related directly to power supplies.

51. The Unit is operating at 100% power with all systems NSA.

Given each of the following plant conditions separately, what will be the status of the 2-2 Emergency Diesel Generator (EDG)?

(1) 2/3 Steam Generator (S/Gs) Pressure transmitters on 1/3 S/Gs are indicating 475 psig.
(2) 2/2 Undervoltage relays sensing the tie line between 2D10 and 2F7 @ 90% for 90 seconds.
(3) 2-2 EDG is running for Monthly OST (Test Run Mode) and A8-5E, "LOCAL PANEL TROUBLE" alarm is received. The NLO reports Jacket Coolant Temperature High is the cause.

A. (1) Auto Started
(2) Auto Started
(3) Auto Stopped

B. (1) Auto Started
(2) Standby
(3) Auto Stopped

C. (1) Standby
(2) Auto Started
(3) Running

D. (1) Auto Started
(2) Auto Started
(3) Running

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. 2/3 S/G pressure in 1/3 S/Gs causes an SI which auto starts the EDG. ½ UV relays sensing tie line between 2D10 and 2F7 <93.4% for 90 seconds results in degraded DF voltage condition causing an auto start of the EDG. A high jacket coolant temp when in the test mode will cause an auto stop of the EDG.
**B.** Incorrect. UV conditions met for EDG auto start.
**C.** Incorrect. Low S/G pressure condition satisfies SI logic which causes EDG auto start. In test mode high jacket cooling temp will trip EDG.
**D.** Incorrect. In test mode high jacket cooling temp will trip EDG.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 064 | Emergency Diesel Generators (ED/G) | Ability to monitor automatic operation of the ED/G system, including: | Start and stop |

**K/A#** A3.06 **K/A Importance** 3.3 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-36.1.C, Rev. 4
3SQS-36.1, Rev. 6
2OM-36.4.AED, Rev. 12
2OM-36.1.D, Issue 4, Rev. 3

**Question Source:** New **Level Of Difficulty: (1-5)**

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

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

52. Given the following plant conditions:

- The Unit is operating at 100% power with all systems NSA.
- Waste Gas Storage Vault Radiation Monitor [2RMQ-RQ303A or B] reaches its **HIGH** setpoint.

Which ONE of the following will be the plant response?

1. 2HVQ-FN214A, "Decontamination Building Filtered Exhaust Fan" STOPS.
2. 2HVQ-FN214B, "Decontamination Building Normal Exhaust Fan" STOPS.
3. 2HVQ-FN214A, "Decontamination Building Filtered Exhaust Fan" STARTS.
4. 2HVQ-MOD23, "Filter Assembly Bypass Damper" CLOSES.
5. 2HVQ-MOD24 & 26, "Filter Assembly Inlet and Outlet Dampers" CLOSE.

A. 1 AND 5.

B. 2 AND 4.

C. 2 AND 5.

D. 2, 3 AND 4.

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**Answer: A.**

**Explanation/Justification:**

**A.** Correct. According to 2OM-43.5.B.3 when 2RMQ-RQ-303 A or B sense a High Radiation signal, 2HVC-FN214A stops and 2HVQ-MOD24 and 26 will close.

**B.** Incorrect. Plausible that a fan will stop and damper will close if the candidate does not know the system interlocks and design features that terminate the radiation release. (opposite logic)

**C.** Incorrect. Plausible that a fan will stop and damper will close if the candidate does not know the system interlocks and design features that terminate the radiation release. (opposite logic)

**D.** Incorrect. This is the correct response when 2RMQ-RQ-303A or B sense an ALERT radiation level which is below the high setpoint.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 073 | Process Radiation Monitoring (PRM) System | Knowledge of the PRM system design feature(s) and/or interlock(s) which provide for the following: | Release termination when radiation exceeds setpoint |

**K/A#** K4.01 **K/A Importance** 4.0 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-43.5.B.3, Rev. 2

**Question Source:** Modified Bank Vision - 15666 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.7)

**Objective:** 2SQS-43.1 6. Describe the control, protection and interlock functions for the control room components associated with the RM system, including automatic functions, and changes in equipment status as applicable.

53. Given the following plant conditions:

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

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

CCP HX's = Primary Component Cooling Water Heat Exchangers
CCS HX's = Secondary Component Cooling Water Heat Exchangers
EDG's = Emergency Diesel Generators
RSS HX's = Recirculation Spray Heat Exchangers

| | CCP HX's | CCS HX's | EDG's | RSS HX's |
|---|---|---|---|---|
| A. | YES | YES | YES | YES |
| B. | YES | YES | YES | NO |
| C. | NO | NO | NO | NO |
| D. | YES | NO | YES | NO |

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. CCS HX will isolate on SI/CIA. RSS HX's will be isolated until CIB actuates at 11.1 psig containment pressure.

**B.** Incorrect. CCS HX will isolate on SI/CIA.

**C.** Incorrect. CCP HX's will not isolate until CIB at 11.1 psig containment pressure so therefore will be providing flow and temperature control. EDG will have cooling even though they will be running unloaded in this plant configuration.

**D.** Correct. At > 5 psig containment pressure, SI and CIA have actuated. 2SWS*MOV107A-D close isolating CCS HX's, therefore there will be no cooling or temperature control to the CCS HX's. The SI signal will start EDGs and open 2SWS*MOV113A&D, therefore providing cooling to EDG's. CIB does not actuate until 11.1 psig, so therefore 2SWS*MOV106A&B will remain open providing cooling and therefore temperature control to the CCP HX's. 2SWS*MOV103A&B remain shut and do not open until containment pressure reaches 11.1 psig (CIB).

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

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

**References provided to Candidate** None **Technical References:** 2OM-30.1.D, Rev. 7; 2SQS-30.1 PPT, Rev. 20

**Question Source:** Modified Bank Vision - 18041 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-30.1 14. Given a service water system configuration and without referenced material, describe the SW system control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable: SI, CIA, CIB

54. Given the following plant conditions:

- The Unit is at 100% power with all systems in NSA.
- Station Air Compressor 2SAS-C21A is running and 2SAS-C21B is in Standby.
- Unit 2 Instrument Air Pressure is **CURRENTLY** 98 psig and RISING, recovering from a failure that caused Instrument Air System Pressure to drop to 85 psig before recovering.
- No operator actions have been taken

IF all systems respond as designed, what will be the **CURRENT** status of the following air compressors?

1. Diesel Driven Instrument Air Compressor [2IAS-C21]
2. Condensate Polishing Air Compressor [2SAS-C22]
3. Station Air Compressor [2SAS-C21B]

A. 1. Running
2. Running
3. NOT Running

B. 1. Running
2. NOT Running
3. NOT Running

C. 1. NOT Running
2. NOT Running
3. Running

D. 1. NOT Running
2. Running
3. Running

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. 2IAS-C21 is NOT Running and 2SAS-C21B is running.
**B.** Incorrect. 2IAS-C21 is NOT Running and 2SAS-C22 & 2SAS-C21B are running.
**C.** Incorrect. 2SAS-C22 is running.
**D.** Correct. According to 3SQS-34.1, 2SAS-C21B auto starts @ 89 psig, 2SAS-C22 auto starts @ 90 psig, 2IAS-C21 auto starts @ 82 psig. Since instrument air pressure dropped to 85 psig, then 2SAS-C22 AND 2SAS-C21B are running and 2IAS-C21 is NOT running.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 078 | Instrument Air System (IAS) | Ability to monitor automatic operation of the IAS, including: | Air Pressure |

**K/A#** A3.01 **K/A Importance** 3.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2SQS-34.1, Rev. 16; 2SQS-34.1 PPT, Rev. 16

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-34.1 14. Given a Unit 2 compressed air configuration and without referenced material, describe the compressed air system control room response to the following off-normal conditions, including automatic functions and changes in plant equipment status as applicable: Loss of Instrument Air, Loss of Electrical Power.

55. Given the following plant conditions:

- An Emergency Plant Shutdown is in progress in accordance with AOP 2.51.1, "Emergency Shutdown" due to a Reactor Coolant System leak into containment.
- The plant is currently at 50% power, shutting down at 2%/minute.

Which ONE of the following describes ALL of the containment parameters that will be trending upward, as monitored in the control room?

A. Containment Pressure **ONLY.**

B. Containment Temperature **ONLY.**

C. Containment Pressure and Humidity **ONLY.**

D. Containment Humidity, Pressure, and Temperature.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Containment humidity and temperature will also be rising. Both of these parameters are also monitored from the control room.

**B.** Incorrect. Containment humidity and pressure will also be rising. Both of these parameters are also monitored from the control room.

**C.** Incorrect. Containment temperature will also be rising. This parameter is also monitored from the control room.

**D.** Correct. According to 2OM-53C.4.2.6.7, Rising containment pressure, temperature and humidity are all symptoms of an RCS leak into containment. All three of these parameters are monitored from the control room as part of the Containment Vacuum and Leakage Monitoring System. There are other symptoms related to an RCS leak such as radiation levels and sump levels, however, these are not required by the K/A. A competent RO candidate must be able to predict and monitor changes in parameters associated with an RCS leak and must know which containment parameters are monitored in the control room. Note that humidity readings on the control board are labeled moisture versus humidity.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 103 | Containment System | Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: | Containment pressure, temperature, and humidity |

**K/A#** A1.01 **K/A Importance** 3.7 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.6.7, Rev. 3

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-12.1

9. List the nominal value of the control room operating parameters associated with the containment vacuum and leakage monitoring system.

17. In the control room, locate all of the control functions and instrumentation associated with the containment vacuum and leakage monitoring system.

56. Given the following plant conditions:

- The Unit is operating at 100% power with all systems in NSA.
- Rod Control is in Automatic.
- Control Bank "D" Rods are at 226 steps.
- The controlling T-ref signal fails to a value of 565°F.

Which ONE of the following describes the automatic operation of the control rod drive system?

Control rod speed will **initially** indicate a speed of ___(1)___ steps per minute, AND control rod motion will ___ (2) ___.

A. (1) 48
(2) be inward

B. (1) 48
(2) **NOT** change

C. (1) 72
(2) be inward

D. (1) 72
(2) **NOT** change

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Incorrect initial rod speed. Any Tavg –Tref mismatch greater than 5 F causes maximum inward rod speeds. Correct that rods would step inward.

**B.** Incorrect. Incorrect initial rod speed. If the candidate thinks the failure would cause outward rod motion, they would also understand that we have disabled the auto outward rod motion feature which makes it plausible to have indicated speed but no change in rod height.

**C.** Correct. 100% power Tavg is around 578 F. If the controlling channel of Tref fails to 565 F, this creates a 13 F mismatch causing control rods to move inward initially at the maximum speed of 72 spm.

**D.** Incorrect. Correct speed. If the candidate thinks the failure would cause outward rod motion, they would also understand that we have disabled the auto outward rod motion feature which makes it plausible to have indicated speed but no change in rod height.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 001 | Control Rod Drive System (CRDS) | Ability to monitor automatic operation of the CRDS, including: | Rod height |

| | | | |
|---|---|---|---|
| **K/A#** A3.02 | **K/A Importance** 3.7 | **Exam Level** | RO |
| **References provided to Candidate** | None | **Technical References:** | 2OM-1.1.B, Rev. 5<br>3SQS-1.3 PPT, Rev.5<br>OM Fig. 1-48 (2.1.5), Issue 1, Rev. 1 |
| **Question Source:** Bank | Vision - 12884 | **Level Of Difficulty: (1-5)** | |
| **Question Cognitive Level:** | Higher - Comprehension | **10 CFR Part 55 Content:** | (CFR 41.7 / 45.13) |

**Objective:** 3SQS-1.3 19. Determine how automatic rod control is affected when any of the process control input signals fail.

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

- A Large Break LOCA occurs from 100% power.
- The control room team is progressing through the EOP set, currently in E-1, "Loss of Reactor or Secondary Coolant".
- Containment Pressure peaked at 42 psig and is now 4 psig and lowering.
- Eight hours into the event, A1-2B, "HYDROGEN LEVEL HIGH/HIGH –HIGH" alarm is received.
- Containment Hydrogen Concentration on both 2HCS-HI100A and 2HCS-HI100B indicate 4.5% and slowly RISING.

Based on these plant conditions, which ONE of the following predicts the impact **AND** what action should the control room team take to mitigate the consequences of this event?

Hydrogen at this concentration is ___(1)___.
This condition will be mitigated by starting the ____(2)____.

A. (1) lower than the flammability limit which poses no danger to equipment located in containment.
(2) hydrogen recombiner.

B. (1) greater than the flammability limit which can lead to equipment damage in containment.
(2) containment atmosphere purge blower.

C. (1) greater than the flammability limit which can lead to equipment damage in containment.
(2) hydrogen recombiner.

D. (1) lower than the flammability limit which poses no danger to equipment located in containment.
(2) containment atmosphere purge blower.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. 4.5% is > the flammability limit. Although BVPS still has not retired the Hydrogen recombiners in place, they are no longer procedurally used.

**B.** Correct. 4% is the flammability limit (Lower Explosive Limit). Above this limit poses danger to containment equipment due to explosive hazard (reference TMI). According to ARP 2OM-46.4.ABD, the TSC should be notified and 2OM-46.4.D will be implemented to place the Containment Atmosphere Blower into operation (8 hours into accident) to reduce H2 concentration.

**C.** Incorrect. Correct impact. Although BVPS still has not retired the Hydrogen recombiners in place, they are no longer procedurally used.

**D.** Incorrect. Incorrect impact. Correct mitigation component as described above.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 028 | Hydrogen Recombiner and Purge Control System (HRPS) | Ability to (a) predict the impacts of the following malfunctions or operations on the HRPS: and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: | The hydrogen air concentration in excess of limit flame propagation or detonation with resulting equipment damage in containment. |

**K/A#** A2.03 **K/A Importance** 3.4 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-46.4.ABD, Rev.3
2OM-46.4.D, Rev. 3
LRM 3.7.6 & Bases , Rev. 52

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-46.1 17. Given a specific plant condition, predict the response of the Post DBA Hydrogen 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.

58. Given the following plant conditions:

- The Unit is in Mode 6.
- Fuel movement is in progress.
- The Containment Hatch is closed.
- The Refueling SRO reports a fuel assembly has been dropped inside containment.
- 2HVR*RQ104A and B, "Containment Purge Exhaust Radiation Monitors alarmed and have been verified to be at the **ALERT** level.
- No operator actions have occurred.

Based on these plant conditions, how will current radiation levels impact the operations of the Containment Purge System?

2HVS-FN263B, "Leak Collection Normal Exhaust Fan" will be ___ (1) ____.
2HVR*MOD23A(B), "Containment Purge Exhaust Isolation Dampers" will be ____(2)____.

A. (1) tripped
(2) closed

B. (1) tripped
(2) open

C. (1) running
(2) closed

D. (1) running
(2) open

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Plausible since some of our radiation monitors will cause fans to trip and valves to close, however, incorrect system response for stated plant conditions.

**B.** Incorrect. Plausible since some of our radiation monitors will cause fans to trip and valves will remain open at the alert level, however, incorrect response for stated plant conditions.

**C.** Incorrect. This would be the system response if the High Radiation Level was reached.

**D.** Correct. Normally if within allowable limits, containment purge is directed to the ventilation vent exhaust. If airborne is present (ie: Alert level), the operators are directed to manually align the system through the SLCRS Main Filter Bank. Since no operator action has occurred, the leak exhaust fan will not have been secured and the dampers will remain open as part of this lineup. The high radiation alarm closes the dampers, not the alert alarm.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 029 | Containment Purge System (CPS) | Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment purge systems controls including: | Radiation levels |

**K/A#** A1.02 **K/A Importance** 3.4 **Exam Level** RO

**References provided to Candidate** None **Technical References:**
2OM-43.1.C, Rev. 4
2OM-43.4.AAN, Rev. 3
2OM-53C.4.2.49.1, Rev. 9
2SQS-43.1 PPT, Rev.10

**Question Source:** Modified Bank Vision - 46328 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.5 / 45.5)

**Objective:** 2SQS-43.1 7. Given a specific plant condition, predict the response of the RM 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.

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

- The Unit is operating at 15% power.
- 2FWE*P23B, "B" Motor Driven Auxiliary Feedwater (MDAFW) Pump has been on clearance for 24 hours.
- TS 3.7.5, "Auxiliary Feedwater (AFW) System" compensatory actions to realign OPERABLE AFW pumps to separate train supply headers was completed.
- An inadvertent reactor trip has occurred.
- All equipment functions as designed and no Safety Injection has occurred.

Which ONE of the following describes the operation of 2FWE*P22, "Turbine Driven AFW Pump Two (2) minutes after the reactor trip?

2FWE*P22 is aligned to discharge to Auxiliary Feedwater (AFW) Header ____________.

A. "A" and is feeding all steam generators.

B. "B" and is feeding all steam generators.

C. "A" and is <u>NOT</u> providing AFW flow to any steam generators.

D. "B" and is <u>NOT</u> providing AFW flow to any steam generators.

---

**<u>Answer:</u> D**

**Explanation/Justification:**

**A.** Incorrect. Normally 2FWE*P22 is aligned to the "A" AFW header, however, the "B" MDAFW Pump is inoperable and in accordance with TS 3.7.5, the 2FWE*P22 was realigned to the "B" AFW Header within two hours. This ensures and AFW is aligned to each header from separate trains and would be NSA for these plant conditions. None of the AFW pumps would be running since the trip was from low power and none of the automatic start signals will start the AFW pumps. Main Feedwater will still be available via the MFRV Bypass Valves to feed S/G's.

**B.** Incorrect. Correct train. No AFW pumps running.

**C.** Incorrect. Incorrect train (see above). Correct that no AFW flow to any S/Gs.

**D.** Correct. 2FWE*P22 was realigned to the "B" AFW Header in accordance with TS 3.7.5 due to inoperability of 2FWE*P23B. No auto start signals are present for 2FWE*P22 since this is a low power trip. MFW is supplying S/Gs via the MFRBV's in NSA lineup. Candidates must know the AUTO start signals to 2FWE*P22 in order to rule out that the pump has not auto started. AMSAC is N/A < 25%. Lo-Lo S/G water levels will not occur on 1/3 S/G's from a low power trip. SI has not occurred and no RCP undervoltage has occurred.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 035 | Steam Generator System (S/GS) | Knowledge of the physical connections and/or cause-effect relationship between the S/GS and the following systems: | MFW/AFW systems |

**K/A#** K1.01 **K/A Importance** 4.2 **Exam Level** RO

**References provided to Candidate** None **Technical References:**
TS 3.7.5, Amend. 278/161
2OM-24.1.D, Rev. 6
OM Fig. 24-2A, Rev. 14
OM Fig. 24-3, Rev. 12

**Question Source:** Bank Vision - 46327 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-24.1 14. Given a Main Feedwater, S/U FW, AFW, or SGWLC system configuration and without reference material, describe the associated system's control room response to the following actuation signals, including automatic functions and changes in equipment status as applicable: Reactor Trip

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

- The Unit is operating at 85% power at end of core life.
- Rod Control is being maintained in MANUAL.
- Tavg is on programmed band at 573°F.
- A load increase to 87% power is performed, initiated by the turbine.

Based on these plant conditions, which ONE of the following describes the relationship between moderator temperature coefficient (MTC) and boron concentration?

____ (1) ___ reactivity will be added due to MTC and a ___ (2) ___ will be required to maintain Tavg on programmed band.

A. (1) Positive
(2) boration

B. (1) Negative
(2) boration

C. (1) Positive
(2) dilution

D. (1) Negative
(2) dilution

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Correct reactivity effect. Incorrect boron effect. Refer to correct answer explanation.

**B.** Incorrect. Incorrect reactivity effect. Incorrect boron effect. Refer to correct answer explanation.

**C.** Correct. At end of life core conditions, MTC is negative meaning a decrease in RCS temperature adds positive reactivity. An increase in steam demand with no operator action or rod control outward movement will cause a drop in RCS temperature and positive reactivity addition to occur as a result. Reactor power will be higher and RCS temperature will be lower. To restore RCS temperature a dilution will need to occur which will add positive reactivity to counteract the positive reactivity added by MTC and allow RCS temperature to be increased to restore program band.

**D.** Incorrect. Positive reactivity versus negative is added by MTC. Correct boron change.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 045 | Main Turbine Generator (MT/G) System | Knowledge of the operational implications of the following concepts as they apply to the MT/G system: | Relationship between moderator temperature coefficient and boron concentration in RCS as T/G load increases. |

**K/A#** K5.17 **K/A Importance** 2.5 **Exam Level** RO

**References provided to Candidate** None **Technical References:** General Physics Reactor Theory: GO-GPF.R4/R8, Rev. 1

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:**

GO-GPF.R8 18. Describe the means by which reactor power will be increased to rated power.

21. Explain the relationship between steam flow and reactor power given specific conditions.

GO-GPF.R4 3 Describe the effect on the magnitude of the moderator temperature coefficient of reactivity from the changes in moderator temperature, boron concentration, and core age.

61. Given the following plant conditions:

- The Unit is operating at 100% power with all systems in NSA.
- A liquid waste discharge is in progress to the Unit 2 Cooling Tower Blowdown.
- 2SGC-RQ100, "Liquid Waste Process Effluent Detector" fails **HIGH**.

Which ONE of the following describes the impact on the Liquid Radwaste System?

The discharge will ________________.

A. automatically terminate immediately.

B. continue unless manually terminated.

C. automatically terminate after a short time delay. (ie: < 10 seconds)

D. automatically terminate after a long time delay. (ie: > 30 seconds)

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. A high failure of this radiation detector will close 2SGC*HCV100, thus immediately terminating the release of liquid discharge.
**B.** Incorrect. A high failure will cause an automatic termination.
**C.** Incorrect. There is no time delay, however, this is plausible since some systems are designed with shorter time delays.
**D.** Incorrect. There is no time delay, however, this is plausible since some systems are designed with longer time delays.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 068 | Liquid Radwaste System (LRS) | Knowledge of the effect of a loss or malfunction on the following will have on the Liquid Radwaste System: | Radiation Monitors |

**K/A#** K6.10 **K/A Importance** 2.5 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-43.1.E, Rev. 4

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-43.1 7. Given a specific plant condition, predict the response of the RM 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.

62. Which ONE of the following describes the interlock feature associated with the Gaseous Waste Storage Tank Inlet **AND** Outlet Header Isolation Valves?

2GWS-AOV104 - Gaseous Waste Storage Tank **INLET** Header Isolation Valve
2GWS-AOV105 - Gaseous Waste Storage Tank **OUTLET** Header Isolation Valve

A. 2GWS-AOV104 must be OPEN in order to OPEN 2GWS-AOV105.

B. 2GWS-AOV105 must be CLOSED in order to OPEN 2GWS-AOV104.

C. If 2GWS-AOV105 leaves its OPEN seat, 2GWS-AOV104 will automatically CLOSE.

D. If 2GWS-AOV104 leaves its OPEN seat, 2GWS-AOV105 will automatically CLOSE.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Both valves cannot be opened at the same time.
**B.** Correct. The interlock between these two valves is such that both cannot be opened at the same time.
**C.** Incorrect. This is a plausible interlock but not applicable to these valves.
**D.** Incorrect. This is a plausible interlock but not applicable to these valves.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 071 | Waste Gas Disposal System (WGDS) | Knowledge of design feature(s) and/or interlock(s) which provide for the following: | Isolation of waste gas release tanks |

**K/A#** K4.04 **K/A Importance** 2.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-19.1.D, Issue 4, Rev. 0; 2SQS-19.1, Rev. 15; OM Fig. 19-3, Rev. 4.

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.7)

**Objective:** 2SQS-19.1 2. Describe the control, protection and interlock functions for the field components associated with the GWDS, including automatic functions, setpoints, and changes in equipment status as applicable.

63. 2RMC*RQ201, "Control Room Area Radiation Monitor" indication on the RM-11 console grid display is backlit **WHITE**.

Which ONE of the following describes the status of 2RMC*RQ201?

A. Monitor has FAILED.

B. Monitor is OFF-LINE.

C. HIGH alarm setpoint has been reached.

D. ALERT alarm setpoint has been reached.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Purple or blue indicates monitor failure.
**B.** Correct. White indicates monitor is off-line.
**C.** Incorrect. Red indicates high alarm.
**D.** Incorrect. Yellow indicates alert setpoint has been reached.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 072 | Area Radiation Monitoring (ARM) System | N/A | Knowledge of annunciator alarms, indications, or response procedures. |

**K/A#** 2.4.31 **K/A Importance** 2.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-43.1.D, Issue 4, Rev. 0

**Question Source:** Bank Vision - 45814 **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-431 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 conditions or for an off-normal condition.

64. Given the following plant conditions:

- The plant is operating at 100% power with all systems NSA.
- A Loss of Power to 4160 VAC Bus 2DF occurs.
- All automatic plant functions occur as designed except Train "B" Service Water Pump(s) fail to restart.
- No subsequent operator action has yet occurred.

Which ONE (1) of the following describes the status of "B" Standby Service Water Pump (2SWE-P21B)?

2SWE-P21B will be __________

A. RUNNING due to loss of 4160 VAC Bus 2DF.

B. RUNNING due to low service water system header pressure.

C. NOT RUNNING because it is unaffected by the Loss of 4160 VAC Bus 2DF.

D. NOT RUNNING because operator action is required to start the pump based on plant conditions.

---

**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. 2SWE-P21B is NOT running. Plausible because EDG 2-2 auto started and will have sequenced on the normal SWS pump. This loss of power will normally result in a restart of the associated SWS pump not the SWE pump.

**B.** Incorrect. 2SWE-P21B is NOT running. Plausible because EDG 2-2 auto started and will have sequenced on the normal SWS pump. The stem of the question states that this did not occur, resulting in a low SWS header pressure. A low SWS header pressure normally results in an auto start of the associated SWE pump. Because of the unique logic associated with the SWE pumps, this low pressure condition and associated bus power supply loss will NOT result in an automatic start of the SWE pump without specific operator action.

**C.** Incorrect. Correct that 2SWE-P21B is NOT running, however, it is affected by the loss of 2DF Bus. Refer to correct answer and 2SQS-30.1 Logic PPT diagram.

**D.** Correct. According to 2OM-30.3.C 2SWE-P21B is powered from 4160 VAC Bus DF. Based on 2SQS-30.1 PPT logic, if 2DF Bus UV occurs 2SWE-P-21B is locked out until the associated EDG sequencer has timed out and then must be manually started by cycling the control switch to STOP and then START.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 075 | Circulating Water System | 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:** 2OM-30.3.C, Rev. 15
2SQS-30.1, Rev. 20

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.7)

**Objective:** 2SQS-30.1 3. Identify the power supplies for the components identified on the NSA system flowpath drawing which are powered from class 1E electrical distribution system (For 4160 V system include the power train and bus designation. For 480 V system include only the power train)
15. Given a SWS configuration and without referenced material, describe the SWS control room response to the following off-normal conditions, including automatic functions and changes in equipment status as applicable (b) Loss of electrical power.

65. A fire occurs in Emergency Diesel Building 2-2.

Which ONE of the following describes the operation of the Fire Protection System?

Diesel Generator Building 2-2 ____________________

A. $CO_2$ System will discharge immediately.

B. Sprinkler System will discharge immediately.

C. $CO_2$ System will discharge after a time delay.

D. Sprinkler System will discharge after a time delay.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Unit 2 has a 60 second pre-discharge time delay.

**B.** Incorrect. Although there are water hose reels in the EDG rooms and other areas of the plant are equipped with automatic sprinkler water, the EDG room is automatically protected by CO2 discharge, either manually or automatically.

**C.** Correct. 3SQS-33.1 states that CO2 discharge occurs by an actuation or by local pushbutton stations. In either case the system is interlocked to discharge CO2 into the area after the discharge timer times out (60 seconds).

**D.** Incorrect. As described above.

---

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 086 | Fire Protection System (FPS) | Knowledge of design feature(s) and/or interlock(s) which provide for the following: | CO2 |

**K/A#** K4.06 **K/A Importance** 3.0 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 3SQS-33.1, Rev. 6

**Question Source:** Bank Vision - 46537 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.10 / 43.5 /45.3 /45.12)

**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 off-normal condition.

66. Given the following plant condition:

- The Reactor is operating at 100% power with all systems NSA.
- The BOP announces ALL Main Turbine Throttle and Governor Valves are CLOSED.
- The Reactor Operator at the Controls (RO ATC) notes the Reactor did **NOT** auto trip as expected.

Which ONE of the following describes the conservative decision making requirements according to NOP-OP-1002, "Conduct of Operations" for the stated plant condition?

The RO ATC ______________________________

A. will trip the reactor **ONLY** after obtaining a peer check to concur with this action.

B. will trip the reactor **WITHOUT** prior approval from the Unit SRO.

C. will trip the reactor **ONLY** after the Unit SRO has been notified and concurs with the action.

D. must immediately obtain approval from the Unit SRO prior to tripping the reactor.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Peer checks are not required during transient situations.

**B.** Correct. In accordance with NOP-OP1002, The RO ATC has the responsibility to initiate a manual reactor trip when in his/her judgement a situation exists which jeopardizes public or plant safety, an operating parameter reaches a trip criteria, or an automatic reactor trip should have occurred.

**C.** Incorrect. Action may be taken before notifying another team member.

**D.** Incorrect. Action may be taken without permission in this situation.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Conduct of Operations | Knowledge of conservative decision making practices. |

**K/A#** 2.1.39 **K/A Importance** 3.6 **Exam Level** RO

**References provided to Candidate** None **Technical References:** NOP-OP-1002, Rev. 5

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.10 / 43.5 / 45.12)

**Objective:** 3SQS-48 1. From memory, explain the duties and responsibilities of Operations personnel.

21. From memory, explain all operations expectations.

67. Given the following plant conditions:

- A reactor startup is in progress in accordance with 2OM-50.4.D2, "Reactor Startup from Mode 3 to Mode 2".
- The estimated critical position (ECP) is Control Bank "D" at 100 steps.
- Reactor Coolant System Temperature is 547°F and stable.
- The +/- 500 pcm positions from the ECP are 35 steps and 185 steps on Control Bank "D".
- Control Bank "B" withdrawal is in progress.
- N31 1/M plot indicates that criticality will be achieved on Control Bank "C" at approximately 100 steps.
- N32 1/M plot indicates that criticality will be achieved on Control Bank "C" at approximately 80 steps.

Based on these plant conditions, which ONE of the following actions is required according to 2OM-50.4.D2?

A. Insert all control rods to zero steps, verify shutdown margin, recalculate ECP.

B. Continue the reactor startup to obtain additional 1/M data to validate the accuracy of the plot.

C. Immediately emergency borate, manually trip reactor and go to E-0, "Reactor Trip or Safety Injection".

D. Stop the reactor startup and verify that the source range NI pulse height discrimination voltage is properly set.

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. According to Attachment 3 (Continuous Reactor Operator Actions) of 2OM-50.4.D2, if criticality is predicted below RIL, (outside the -500 pcm ECP), then action 2 is applicable which is to insert all control rods to zero steps, verify RCS boron, perform a S/D margin calculation and recalculate the ECP.

**B.** Incorrect. Would not proceed in the face of uncertainty when two data points indicate criticality below the RIL is predicted.

**C.** Incorrect. Correct action if reactor is critical below RIL.

**D.** Incorrect. With N31 and N32 1/M plots indicating different values, this strengthens the plausibility of the candidate thinking that this may be a pulse height discrimination problem. This action is not procedurally approved based on these plant conditions during a reactor startup.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Conduct of Operations | Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant temperature, secondary plant, fuel depletion etc. |

**K/A#** 2.1.43 **K/A Importance** 4.1 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-50.4.D2, Rev. 0

**Question Source:** Bank 2LOT4 NRC Exam **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comprehension **10 CFR Part 55 Content:** (CFR 41.10 / 43.6 / 45.6)

**Objective:** 3LOT-M4D1 1. Explain precaution and limitations, reactor theory and kinetics applicable to the startup IAW OM-50.4.D, Reactor Startup from Mode 3 to Mode 2 and BVPS Reactor Theory Manual.

68. Which ONE of the following is the bases for Technical Specification 2.1.2, "Reactor Coolant System (RCS) Pressure Safety Limit"?

To protect the integrity of the ___ (1) ___ which prevents the ___ (2) ___.

A. (1) reactor fuel,
(2) release of radionuclides contained in the fuel to the atmosphere.

B. (1) RCS piping and components,
(2) release of radionuclides contained in the fuel to the atmosphere.

C. (1) reactor fuel,
(2) main steam safety valves and reactor protection system actuation from occurring.

D. (1) RCS piping and components,
(2) main steam safety valves and reactor protection system actuation from occurring.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. TS 2.1.1 bases is to protect the fuel. TS 2.1.2 focuses on the RCS barrier.

**B.** Correct. According to TS 2.1.2 bases, the safety limit on RCS pressure protects the integrity of the RCS against overpressurization. In the event of a fission product failure, fission products are released in the RCS. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on the RCS, the continued integrity is ensured.

**C.** Incorrect. Fuel incorrect. MSSV's and RPS actuation will occur to prevent actual RCS pressure from exceeding 2735 psig.

**D.** Incorrect. RCS piping correct. MSSV and RPS actuation portion incorrect as explained above.

---

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| N/A | N/A | Equipment Control | Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. |

**K/A#** 2.2.25 **K/A Importance** 3.2 **Exam Level** RO

**References provided to Candidate** None

**Technical References:** TS 2.1.2, Amend. 278/161
TS 2.1.2 Bases, Rev. 0

**Question Source:** New

**Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory

**10 CFR Part 55 Content:** (CFR 41.5 / 41.7 / 45.2)

**Objective:** 3SQS-ITS.003 1. State the purpose of each ITS Safety Limit as described in the applicable safety analyses section of the ITS bases.

69. Given the following plant conditions:

- The plant is operating at 100% power.
- A single control rod drops to the bottom of the core.
- The Reactor Operator notes Pressurizer pressure dropped to 2150 psig before recovering to normal operating pressure.
- The plant remains at power and no automatic protective or operator actions have occurred.
- Tavg drops to 570°F and then stabilizes.

What impact will this event have on the Departure from Nucleate Boiling (DNB) Technical Specification (T.S.) **AND** Shutdown Margin as it is defined in Technical Specifications?

During this event, the DNB Technical Specification limit ____ (1) ____ exceeded.

After Tavg stabilizes at 570°F, the Technical Specification defined Shutdown Margin is ____ (2) ____ the Technical Specification defined Shutdown Margin that existed **BEFORE** the rod dropped.

A. (1) was
(2) less than

B. (1) was
(2) the same as

C. (1) was <u>NOT</u>
(2) less than

D. (1) was <u>NOT</u>
(2) the same as

---

**<u>Answer:</u> B**

**Explanation/Justification:**

**A.** Incorrect. Correct that DNB TS limit was exceeded. Incorrect that SDM is less. Refer to correct answer explanation.

**B.** Correct. According to TS 3.4.1 and COLR 5.1.10 values, DNB is exceeded when Pressurizer pressure drops to < 2214 psia. Shutdown Margin is unchanged based on a dropped rod. SDM is defined as a reactivity balance that shows how much a reactor is or can be made subcritical. For a critical reactor, margin to criticality is simply rod worth minus power defect. The negative reactivity from the dropped rod worth is compensated for by the drop in Tavg and resultant positive reactivity added. Two minutes after the rod drop is stipulated so that Xenon changes do not come into play.

**C.** Incorrect. Incorrect that DNB TS limit was NOT exceeded. Incorrect that SDM has decreased. Refer to correct answer explanation.

**D.** Incorrect. Incorrect that DNB TS limit was NOT exceeded. Correct that SDM has remained the same.

---

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| N/A | N/A | Equipment Control | Knowledge of the conditions and limitations in the facility license. |

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

**References provided to Candidate** None **Technical References:** TS 3.4.1, Amend. 278/161
COLR Cycle 15,Rev. 66
GO-RT-9 LP, Rev. 6

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Analysis **10 CFR Part 55 Content:** (CFR 41.7 / 41.10 / 43.1 / 45.13)

**Objective:** 3SQS-ITS.005 3. Given plant conditions, determine the criteria necessary to ensure compliance with each ITS 3.2 LCO IAW Bases

70. Using Operations Manual (OM) Figure 21-4 as a reference, which ONE of the following describes information that can be obtained from this mechanical print?

    1. Piping size
    2. Valves that are **NOT** part of the Main Steam System (MSS)
    3. Valves which can be operated from the Main Control Board
    4. Steam Dump Logic Details
    5. System parameter monitoring instrumentation

A. 1, 4, & 5 ONLY.

B. 1, 2, 3, & 4 ONLY.

C. 1, 2, 3, & 5 ONLY.

D. 2, 3, 4, & 5 ONLY.

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Can determine which valves are operated from MCB by the CS B designator. The detail of Steam Dump Logic is not provided on this mechanical print. The Electrical and Logic prints would need referencing. Valves that are NOT part of the MSS are not included.

**B.** Incorrect. System parameter monitoring instrumentation is provided, ie: pressure, temperature, flow transmitters. Steam Dump logic is NOT included.

**C.** Correct. All of the items can be obtained with the exception of steam dump logic detail which can not be determined without the use of associated electrical/logic prints.

**D.** Incorrect. Piping size can be obtained by the numbers and arrows followed by the inch sign. Steam Dump logic detail is not provided..

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Equipment Control | Ability to obtain and interpret station electrical and mechanical drawings |

**K/A#** 2.2.41 **K/A Importance** 3.5 **Exam Level** RO

**References provided to Candidate** OP Manual Fig. 21-4, Rev. 7

**Technical References:** OP Manual Fig. 21-4, Rev. 7
Vond/Flow Diagram Symbology Sh 1, Rev. 1
Vond/Flow Diagram Symbology Sh 2, Rev. 6

**Question Source:** New

**Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application

**10 CFR Part 55 Content:** (CFR 41.10 / 45.12 / 45.13)

**Objective:** 2SQS-20.1 14. Given a set of plant conditions and appropriate procedure(s), apply the operational sequence, parameter limits, precautions and limitations, and cautions & notes applicable to the completion of the task activities in the control room.

71. The following conditions exist at a job site:

- The general area radiation levels are 40 mr/hr.
- Radiation level with shielding is 10 mr/hr.
- Time for one worker to install **AND** remove shielding is fifteen (15) minutes.
- Time to conduct the task of one worker is one (1) hour.
- Time to conduct the task with two workers is twenty (20) minutes

Assumptions:

- Shielding is installed and removed by one (1) worker.

In order to comply with radiation work permit requirements, which ONE of the following will result in the **LOWEST** whole body dose?

Conduct the task with ___________

A. One (1) worker with shielding.

B. Two (2) workers with shielding.

C. One (1) worker without shielding.

D. Two (2) workers without shielding.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Dose to install shielding = 10 mr + 10mr/hr = 20 mr

**B.** Correct. Dose to install shielding = 10 mr + (.33)(10) = (3.3)(2) = 6.6 + 10 = 16.6 mr. In order to comply with radiation work permit requirements of maintaining dose as low as reasonably achievable, the lowest dose derived is by using a worker to install shielding and use two workers to perform the job with the shielding in place.

**C.** Incorrect. Dose with one worker without shielding is 40 mr x 1hr = 40 mr

**D.** Incorrect. Dose with two workers without shielding is 40 mr x (.33)(40) = (13.2)(2) = 26.4 mr

---

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| N/A | N/A | Radiation Control | Ability to comply with radiation work permit requirements during normal and abnormal conditions. |

**K/A#** 2.3.7 **K/A Importance** 3.5 **Exam Level** RO

**References provided to Candidate** None **Technical References:** NOP-OP-4107, Rev. 4<br>NOP-OP-4102, Rev. 1

**Question Source:** Bank Vision - 46780 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Analysis **10 CFR Part 55 Content:** (CFR 41.12 / 45.10)

**Objective:** 3SSG-Admin 16. Describe the controls for maintaining personnel exposures ALARA IAW NOP-OP-4107.

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

- The Unit is operating at 100% power with all systems in NSA.
- A very small fuel element failure has just been confirmed.
- A Large Break LOCA inside containment occurs.
- All systems function as designed.
- Fifteen (15) minutes has elapsed.

Which of the following radiation monitors will <u>accurately indicate radiation levels</u> post trip?

1. Containment Gas Monitor [2RMR*RQ303A & B]
2. Reactor Coolant Letdown [2CHS-RQ101A & B]
3. Reactor Containment Area Low Range [2RMR*RQ201]
4. Recirc Spray Heat Exchanger Discharge [2SWS*RQ100A-D]

A. 1 and 2 <u>**ONLY**</u>.

B. 2 and 3 <u>**ONLY**</u>.

C. 3 and 4 <u>**ONLY**</u>.

D. 1 and 4 <u>**ONLY**</u>.

---

<u>**Answer:**</u> **C**

**Explanation/Justification:**

**A.** Incorrect. Initially, the containment atmosphere process radiation monitors will show increased particulate and gaseous activity. The containment gas monitor is isolated upon a CIA signal and therefore will not accurately indicate post trip plant conditions. The trends prior to CIA are useful for diagnosis. The reactor coolant radiation monitors predominantly monitor for failed fuel and RCS crud burst. These monitors are also isolated upon a CIA signal and therefore although plausible are not available to accurately monitor post trip radiation levels unless charging is un-isolated which is not the case 15 minutes post trip.

**B.** Incorrect. The containment area low range will accurately indicate. The letdown radiation monitor will not accurately indicate because it is isolated on a CIA (SI signal).

**C.** Correct. On a CIB signal, 2SWS-103A & B open. This signal also starts a ten minute timer in the radiation monitor circuit that will start the sample pumps after the time delay. The reactor containment low range monitor is available and will accurately monitor post trip radiation levels.

**D.** Incorrect. Containment Gas monitor is incorrect for the reasons described above. The recirc spray discharge monitor is correct.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Radiation Control | Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. |

**K/A#** 2.3.15 **K/A Importance** 2.9 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2SQS-43.1, Rev. 10; 2OM-43.4.4.AAF, Rev. 8; 2OM-43.4.ABT, Rev. 6

**Question Source:** Modified Bank Vision - 1246 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comprehension **10 CFR Part 55 Content:** (CFR 41.12 / 43.4 / 45.9)

**Objective:** 2SQS-43.1 7. Given s specific plant condition, predict the response for the RM system control room indication and control loops, including all automatic functions and changes in equipment status, for either a change in plant conditions or off-normal condition.

73. Which ONE of the following describes the bases for why Functional Restoration Procedures (FRPs) are <u>NOT</u> implemented until specifically directed in ECA-0.0, "Loss of All AC Power" series?

A. All FRPs are written on the premise that at least one 4KV emergency bus is energized.

B. ECA-0.0 actions must be performed in sequence. Implementing FRPs interrupts the sequence and timing of steps.

C. ECA-0.0 includes all the key actions of RED path FRPs. Performing FRPs would be redundant and prolong the time until RCS depressurization was performed.

D. Certain diagnostic steps must be performed to minimize RCS leakage through the RCP seals. These steps are specific to ECA-0.0 and are not performed in any FRP.

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. ECA-0.0 B/G document states that FRPs are written based on the premise that at least one AC emergency bus is energized.

**B.** Incorrect. ECA-0.0 is written to explicitly monitor and maintain CSF's.

**C.** Incorrect. ECA-0.0 is written to explicitly monitor and maintain CSF's.

**D.** Incorrect. Steps in ECA-0.0 are performance not diagnostic. The statement itself is correct with this exception, however, not what is asked for in the stem of the question.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Emergency Procedures/Plan | Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations. |

**K/A#** 2.4.23 **K/A Importance** 3.4 **Exam Level** RO

**References provided to Candidate** None **Technical References:** 2OM-53B.4.ECA-0.0, Rev. 9

**Question Source:** Bank Salem 2008 NRC Exam **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-53.3 4. Explain from memory the basis for all cautions and notes IAW BVPS-EOP Executive Volume.

74. Given the following plant conditions:

- The Control Room has been evacuated due to a major fire in the control room.
- The crew is implementing 2OM-56C, "Alternate Safe Shutdown From Outside the Control Room", which will allow cold shutdown to be achieved within 72 hours.

Which ONE of the following describes why the Auxiliary Feedwater Pump suction source is shifted to the back-up service water supply during the course of this procedure implementation?

A. To minimize the impact of fire induced spurious signals due to hot shorts.

B. To minimize the number of powered components for hot shutdown and subsequent cooldown.

C. Normal suction sources may be insufficient to complete cooldown to Mode 5 in the required time.

D. Alternate suction sources maximize the use of localized manual operations for control of parameters.

---

**Answer** C

**Explanation/Justification:**

**A.** Incorrect. Incorrect reason, plausible because it is one of the objectives of the procedure.
**B.** Incorrect. Incorrect reason, plausible because it is one of the objectives of the procedure.
**C.** Correct. According to 2OM-56C.2.A, precaution and limitation #10, 2FWE*TK210 and 2WTD-TK23 (normal suction sources) may not be sufficient to complete a cooldown to Mode 5.
**D.** Incorrect. Incorrect reason, however plausible because it is one of the objectives of the procedure.

---

| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Emergency Procedures/Plan | Knowledge of fire protection procedures |

**K/A#** 2.4.25 **K/A Importance** 3.3 **Exam Level** RO

**References provided to Candidate** None

**Technical References:** 2OM-56C.1.A, Issue 4, Rev. 1
2OM-56C.2.A, Rev. 3
2OM-56C.4.A, Rev. 12
2OM-56C.4.B, Rev. 30

**Question Source:** New **Level Of Difficulty: (1-5)**

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

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

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

- A Plant Startup in accordance with 2OM-52.4.A, Raising Power from 5% to Full Load Operation".
- Reactor power is 5% and slowly RISING.
- A4-9F, "ROD AT BOTTOM" is received.
- The Reactor Operator verifies only ONE Rod Bottom light is LIT.
- Before the crew takes any actions, additional alarms are received in the Control Room:

- A4-2F, "PRESSURE RELIEF BLOCK".
- A12-3F, "2/3 LO-LO Tavg (P-12)"
- A4-8A, "ROD CONTROL SYSTEM URGENT ALARM".

Which ONE of these plant conditions **REQUIRES** a manual reactor trip?

A. A4-9F, "ROD AT BOTTOM".

B. A12-3F, "2/3 LO-LO Tavg (P-12)".

C. A4-2F, "PRESSURE RELIEF BLOCK".

D. A4-8A, "ROD CONTROL SYSTEM URGENT ALARM".

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. According to ARP for A4-9F, a reactor trip is only required if two or more rod bottom indications are received. The ARP refers the operators to AOP 2.1.8.

**B.** Correct. According to Attachment 1 of 2OM-52.4.A when Tavg drops to 541 F, the reactor shall be tripped. 2OM-1.4.ACK Alarm Response Procedure for 2/3 Lo-Lo Tavg (P-12) states that this alarm is received upon 2/3 loop Tavg inputs sensing < 541 F. AOP 2.1.8 furthermore requires a reactor trip if RCS TAvg < 541 F.

**C.** Incorrect. According to 2OM-6.4.AAV, this alarm is received at 2185 psig. This is an expected alarm on a dropped rod due to decreasing RCS pressure. The low pressure reactor trip does not occur until 1945 psig and the ARP does not require a reactor trip.

**D.** Incorrect. This alarm could potentially have resulted in the dropped rod. The candidate could mistake this alarm for a General Warning Alarm. The candidate needs to understand that this alarm will block auto and manual rod motion but does not require a reactor trip.

---

| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| N/A | N/A | Emergency Procedures/Plan | Ability to prioritize and interpret the significance of each annunciator or alarm. |

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

**References provided to Candidate** None **Technical References:** 2OM-1.4.AAA , Rev. 5<br>2OM-1.4.ACK, Rev. 4<br>2OM-52.4.A, Rev. 69<br>2OM-53C.4.2.1.8, Rev. 3<br>2OM-6.4.AAV, Issue 4, Rev. 0<br>2OM-1.4.AAB, Rev. 6

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comprehension **10 CFR Part 55 Content:** (CFR 41.10 / 43.5 /45.3 /45.12)

**Objective:** 3SQS-53.5 14. Apply the actions for a rod position malfunction.

76. Given the following plant conditions:

- A Loss of Coolant Accident (LOCA) occurred from 100% power.
- The crew is performing the actions of E-0, "Reactor Trip or Safety Injection" currently at Step 13, "Check if RCPs Should be Stopped".
- Tcold is 270°F and trending DOWN.
- RCS pressure is 75 psig and trending DOWN.
- Containment pressure has peaked at 10 psig and is STABLE.
- All Engineered Safeguards Equipment functions as designed with the exception of High Head Safety Injection Pumps which cannot be started.
- Reactor Coolant Pumps (RCPs) are running.

What actions are **required** regarding the RCPs **AND** what procedure transition, if any, is **required**?

A. Trip RCPs AND continue in E-0, "Reactor Trip or Safety Injection".

B. Trip RCPs AND transition to FR-P-2, "Response to Anticipated Pressurized Thermal Shock Condition".

C. Keep RCPs running AND continue in E-0, "Reactor Trip or Safety Injection".

D. Keep RCPs running AND transition to FR-P-2, "Response to Anticipated Pressurized Thermal Shock Condition".

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. RCPs are not required to be tripped based on plant conditions. Plausible that they are tripped because during large break LOCAs, the operation of the RCPs has little if any effect during mitigation and recovery. Remaining in E-0 is the correct procedural application.

**B.** Incorrect. RCPs are not tripped based on plant conditions It is plausible that a transition to FR-P.2 is made since Tcold is < 275 F, however, EOP rules of usage specifies that CSFST monitoring should not be monitored until directed by E-0 or a transition is made to another procedure. Neither of these has occurred based on stated plant conditions. Additional the SRO candidate should have the knowledge that this is a yellow path CSF, which requires transition based on his judgment only.

**C.** Correct. RCPs are required to be running during these plant conditions. Although D/P between RCS pressure and the highest S/G is less than 220 PSID which meets trip criteria, since there is no HHSI flow, RCPs are NOT secured. CIB has not actuated because containment pressure did not exceed 11.1 psig which would require RCPs to be secured. The SRO must be able to assess plant conditions and prescribe procedures to proceed based on these conditions. Remaining in E-0 is the correct course of action until a transition to E-1 is directed.

**D.** Incorrect. Correct that RCPS must be kept running. FR-P.2 transition criteria is met, however, not required based on rules of usage.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 011 | Large Break LOCA | Ability to determine or interpret the following as they apply to a Large Break LOCA: | Actions to be taken based on temperature and pressure – saturated and superheated. |

**K/A#** EA2.01 **K/A Importance** 4.7 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.E-0, Rev. 8
2OM-53A.1.F-0.4, Rev. 0
1/2OM-53B.2, Rev. 7
2OM-53B.5.GI-6, Rev. 1
1/2OM-53B.2, Rev. 7

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 43.5 / 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.

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

- The Unit is at 80% power.
- Charging Pump/High Head Safety Injection Pump, [2CHS*P21A] is in service.
- A Loss of Emergency Bus 2AE occurs.
- The crew entered AOP 2.36.2, "Loss of 4KV Emergency Bus".
- No HHSI pump can be started.
- Primary Component Cooling (CCP) Water flow to the 21A Reactor Coolant Pump (RCP) thermal barrier indicates ZERO (0) gpm.

Based on these plant conditions and in accordance with applicable procedure, what sequence of actions is required for the RCPs?

A. After ONE (1) minute, Trip Reactor, complete IMA of E-0, then trip 21A RCP, Close PRZR Spray Valves for affected RCP.

B. After ONE (1) minute, Trip Reactor, complete IMA of E-0, then trip 21A RCP, Close RCP Seal Leakoff Valve for affected RCP.

C. Within THREE (3) – FIVE (5) minutes, Trip Reactor, Trip 21A RCP, Complete IMA of E-0, Close PRZR Spray Valves for affected RCP.

D. Within THREE (3) – FIVE (5) minutes, Trip Reactor, Trip 21A RCP, Complete IMA of E-0, Close RCP Seal Leakoff Valve for affected RCP.

---

**Answer: A**

**Explanation/Justification:**

- **A.** Correct. The SRO candidate must recognize that a loss of charging pumps results in a loss of seal injection and also recognize that a loss of thermal barrier cooling to the 21A RCP has occurred. AOP 2.36.2, requires that the RCP can be run under these conditions for only one minute before RCP damage can result and therefore directs a reactor trip and trip of the affected RCP, After E-0 IMAs are complete and then close PRZR Spray valves for affected RCP(s). This requires deep SRO procedural knowledge.
- **B.** Incorrect. Correct except Seal Leakoff Valve is NOT closed. This valve would be closed in AOP 2.6.8 for seal failure actions when leakoff flow is > 6 gpm.
- **C.** Incorrect. Correct actions, incorrect time. This time is associated with AOP 2.6.8 for seal failure actions when leakoff flow is > 6 gpm.
- **D.** Incorrect. Time and actions are associated with seal failure actions when leakoff flow is > 6 gpm.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 026 | Loss of Component Cooling Water (CCW) | Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: | The length of time after the loss of CCW flow to a component before that component may be damaged. |

**K/A#** AA2.06 **K/A Importance** 3.1 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.36.2, Rev. 10

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-53.2 26. Describe the actions for Loss of Component Cooling Water.

78. Given the following plant conditions:

- The plant has been operating at 95% power with a small Fuel Element Failure.
- Chemistry reports that the latest sample has shown a sharp increase in dose equivalent I-131 currently reading 40 µCi/gm.

According to technical specification (TS) 3.4.16, which ONE of the following is the required action **AND** bases for this action?

The required TS 3.4.16 action is to ___ (1) ___ **AND** the bases for this action is to ensure TEDE at the site boundary and in the control room will **NOT** exceed ____ (2) ____.

A. (1) be in Mode 3 with Tavg < 500°F within 6 hours
(2) 10 CFR 100 dose guideline limits during a Loss of Coolant Accident.

B. (1) be in Mode 3 with Tavg < 500°F within 6 hours
(2) 10 CFR 50.67 dose guideline limits during a Steam Line Break or Steam Generator Tube Rupture.

C. (1) restore Dose Equivalent I-131 to within limit within 48 hours
(2) 10 CFR 100 dose guideline limits during a Loss of Coolant Accident.

D. (1) restore Dose Equivalent I-131 to within limit within 48 hours
(2) 10 CFR 50.67 dose guideline limits during a Steam Line Break or Steam Generator Tube Rupture.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Correct action statement. Incorrect bases, refer to correct answer explanation.
**B.** Correct. At 40 microcuries/gm, I-131 is well above TS limits in the unacceptable region of Figure 3.4.16-1. TS 3.4.16 LCO is to be in Mode 3 and cool down to 500 F within 6 hours. The bases is based on a SGTR or SLB as opposed to LOCA. After BVPS power up-rate, the source term is now based on 10CFR50.67 as opposed to 10CFR100 dose guideline limits.
**C.** Incorrect. Incorrect action statement. Incorrect bases.
**D.** Incorrect. Incorrect action statement. Correct bases.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 038 | N/A | 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** TS 3.4.16, Amend. 278/161 (BASES NOT PROVIDED) **Technical References:** TS 3.4.16, Amend. 278/161; TS 3.4.16 Bases, Rev. 0

**Question Source:** Modified Bank Vision - 45836 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.5 / 43.2 / 45.2)

**Objective:** 3SQS-ITS.007

2. State the purpose of each ITS 3.4 specification as describes in the applicable safety analyses section of the ITS bases.

3. Given plant conditions, determine the criteria necessary to ensure compliance with each TS 3.4 LCO in accordance with the bases, surveillance requirements and applicability.

79. Given the following plant conditions:

- The plant is currently at 45% power with all systems in NSA.
- VCT Level is 22% and slowly dropping.
- Multiple indications and alarms are received as follows:
    - A1-1C, "VITAL BUS INVERTER OPERATION/TROUBLE".
    - A5-2A, "REACTOR PROTECTION SYSTEM TRAIN A TROUBLE"
    - NIS Channel 1 Instrumentation Rack is de-energized.
    - Zero voltage indicated on Battery 2-1 output voltmeter (Local)
- The appropriate AOP is entered.

What **ADDITIONAL** condition will require the Unit Supervisor to direct a MANUAL Reactor Trip and entry into E-0, "Reactor Trip or Safety Injection"?

A. Condenser vacuum drops to 22 inches HG-vac.

B. PRZR Level drops to 20% and RCS makeup is required.

C. "B" and "C" Thermal Barrier Outlet Isol. Vlvs [2CCP-107B & 107C], fail closed.

D. A loss of power to Power Range nuclear instrumentation channel N41 occurs.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. AOP 2.38.1A, Attachment 1 references a lowering condenser vacuum and directs a reactor trip if the condenser unavailable setpoint is approaching. The candidate must recognize that at 22 in Hg-Vac that the condenser is still available. At 24 in Hg-Vac a turbine trip will occur, however, because reactor power is < P-9 (49%), a reactor trip will not occur.

**B.** Correct. The SRO candidate must recognize the symptoms as a Loss of Vital Bus 1 and enter into AOP 2.38.1A. This loss results in a loss of letdown, coupled with a loss of the ability to makeup to the VCT. With no letdown and no auto makeup to the VCT, VCT level will continue to drop (auto makeup occurs at 20%). No normal boration capability exists and VCT swapover to the RWST will occur at 5%, even though the AOP. directs reducing charging flowrate to minimum. Attachment 1 of AOP 2.38.1A directs a reactor trip and entry into E-0. This question requires the SRO candidate to use deep procedural section actions.

**C.** Incorrect. 2CCP-AOV107A closes on a loss of Vital Bus 1. 2CCP-107B and 107C do NOT close. A closure of these valves does NOT require entry into E-0 unless seal injection is also lost which is not the case with the stated plant conditions

**D.** Incorrect. Loss of power to N41 does occur as a result of Loss to this vital bus, however, this condition does not require a reactor trip.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 057 | Loss of Vital AC Electrical Instrument Bus | N/A | Knowledge of EOP entry conditions and immediate action steps. |

**K/A#** 2.4.1 **K/A Importance** 4.8 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-53C.4.2.38.1A, Rev. 4

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-38.1 14. Given a change in plant conditions due to a system/component failure, analyze 120 VAC to determine what failure has occurred.

80. Given the following plant conditions:

- The Unit is at 100% power with all systems in NSA, when the following annunciators are received:
  - A8-9B, "125VDC BUS 2-2 TROUBLE".
  - A1-1A, "DC DISTRIBUTION PANEL LOSS OF CONTROL DC".
  - 125 Volt DC Bus 2-2 voltage indicates ZERO (0) volts.

Based on these plant conditions:

(1) What procedure or procedures will be entered to address these plant conditions?
(2) Which of the compensatory actions listed will be required by the procedure(s) in effect?

A. (1) AOP 2.39.1B, "Loss of 125VDC Bus 2-2" **ONLY**.
(2) Establish a continuous fire watch in EDG 2-1 room.

B. (1) AOP 2.39.1B, "Loss of 125VDC Bus 2-2" **ONLY**.
(2) Perform 2OST-36.7, "Offsite to Onsite Power Distribution System Breaker Alignment Verification".

C. (1) E-0, "Reactor Trip or Safety Injection" **AND** AOP 2.39.1B, "Loss of 125VDC Bus 2-2".
(2) Establish a continuous fire watch in EDG 2-1 room.

D. (1) E-0, "Reactor Trip or Safety Injection" **AND** AOP 2.39.1B, "Loss of 125VDC Bus 2-2".
(2) Perform 2OST-36.7, "Offsite to Onsite Power Distribution System Breaker Alignment Verification".

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**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. E-0 must also be entered as a result of the Rx trip associated with MSIV closure. Incorrect compensatory action. These are the compensatory actions for loss of 125VDC Bus 2-5.
**B.** Incorrect. E-0 must also be entered as a result of the Rx trip associated with MSIV closure. Correct compensatory action.
**C.** Incorrect. Correct procedural entries. Incorrect compensatory action. These are the compensatory actions for loss of 125VDC Bus 2-5.
**D.** Correct. According to 2OM-53C.4.2.39.1B, MSIVs fail closed. From 100% power, this will result in a reactor trip and E-0 will be entered while concurrently performing actions in this AOP. EDG 2-2 starting circuitry is also lost which renders EDG inoperable. The AOP has specific guidance in Attachment 1, Step 12 that directs the performance of 2OST-36.7 as a result of EDG 2-2 starting circuitry power loss. The SRO must recognize these loads are powered from 125VDC Bus 2-2, select the appropriate procedures to address the conditions, and recognize the impact by selecting the appropriate compensatory action.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 058 | Loss of DC Power | Ability to determine and interpret the following as they apply to the Loss of DC Power | DC Loads lost; impact on ability to operate and monitor plant systems |

**K/A#** AA2.03 **K/A Importance** 3.9 **Exam Level** SRO

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

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-39.1 25. Given a set of plant conditions, recommend corrective actions for the SM that mitigates the condition including basis.

81. Given the following plant conditions:

- A Reactor Trip and Safety Injection from 100% power occurred.
- Main Steam Line Isolation (MSLI) has occurred on the "21A" S/G ONLY.
- All S/G pressures are 710 psig and continue to DROP.
- All S/G NR Levels are off scale LOW.
- RCS cold leg C/D rate is 175 °F/hr.
- AFW flow to EACH S/G has been throttled to 50 gpm per S/G.
- Crew 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 21B AND 21C S/Gs.

C. (1) Transition to FR-H.1, "Response to Secondary Heat Sink".
(2) Isolate feed flow to 21B AND 21C S/Gs.

D. (1) Transition to FR-H.1, "Response to Secondary Heat Sink".
(2) Continue feeding all S/Gs at 50 gpm.

---

**Answer: A**

**Explanation/Justification:**

**A.** Correct. In accordance with ECA-2.1, a minimum of 50 gpm must be maintained to each S/G with a narrow range level < 12%. There is also a note that states FR-H.1 should be implemented only if a total feed flow capability of 340 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 21B and 21C would violate ECA-2.1 procedural actions. Candidate must recognize that all S/Gs are faulted.

**C.** Incorrect. Incorrect procedural transition. Incorrect action, however, would be the preemptive action to isolate feedwater flow to a faulted S/G

**D.** Incorrect. Incorrect procedural transition. Correct action in accordance with ECA-2.1 versus FR-H.1.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| W/E12 | Uncontrolled Depressurization of all Steam Generators | N/A | Ability to determine operability and/or availability of safety related equipment. |

**K/A#** 2.2.37 **K/A Importance** 4.6 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.ECA-2.1, Rev. 10; 2OM-53A.1.FR-H.1, Rev. 9

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.7 / 43.5 / 45.12)

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

82. Given the following plant conditions:

- The Unit is operating at 5% power with all systems in NSA, during a plant shutdown.
- A4-1C, PRESSURIZER CONTROL LEVEL HIGH/LOW is received.
- 2RCS*LT459 is reading 16% and is slowly DROPPING.
- 2RCS*LT460 is reading 23% and is slowly RISING.
- 2RCS*LT461 is reading 23% and is slowly RISING.
- Charging flow is 100 gpm and slowly INCREASING.
- Charging Flow Controller output is DECREASING.

Based on these indications and with NO operator action, which ONE of the following describes the operational impact, if any?

A reactor trip on Pressurizer Level _____ (1) _____.
Technical Specification entry into TS 3.3.1, "Reactor Trip System Instrumentation" is_____ (2) _____.

A. (1) WILL occur.
(2) NOT REQUIRED.

B. (1) WILL occur.
(2) REQUIRED.

C. (1) WILL **NOT** occur.
(2) REQUIRED.

D. (1) WILL **NOT** occur.
(2) is **NOT** REQUIRED.

---

**Answer:** D

**Explanation/Justification:**

**A.** Incorrect. Incorrect Reactor trip will NOT occur. Correct that TS entry is NOT required. Refer to correct answer explanation.

**B.** Incorrect. Incorrect Reactor trip will NOT occur. Incorrect that TS entry is NOT required. Refer to correct answer explanation.

**C.** Incorrect. Correct Reactor trip will NOT occur. Incorrect that TS entry is required. Refer to correct answer explanation.

**D.** Correct. The SRO candidate must evaluate PRZR level instrumentation and evaluate plant performance to correctly deduce that 2RCS*LT459 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. At 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. TS 3.3.1 entry in NOT required when < P-7. The SRO candidate must evaluate plant conditions and must possess the knowledge of TS applicability beyond that of what is required for RO knowledge.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 028 | Pressurizer Level Malfunction | 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:** 2OM-6.4.AAL, Rev. 7
2OM-6.4.IF, Rev. 12
TS 3.3.1, Amend. 282/166
TS Table 3.3.3-1, Amend. 278/161

**Question Source:** Modified Bank Vision - 1420 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Analysis **10 CFR Part 55 Content:** (CFR 43.2)

**Objective:** 2SQS-6.4 20. Given a change in plant conditions due to system or component failure, analyze to determine what occurred.

83. Given the following plant conditions:

- A Reactor Startup is in progress.
- IR Channel N-35 indicates $4 \times 10^{-11}$ amps.
- IR Channel N-36 indicates $7 \times 10^{-11}$ amps.
- SR Channel N-31 pulse height discrimination circuit fails causing N-31 to indicate off-scale HIGH.

Based on these plant conditions, which ONE of the following describes the impact on plant operations?

Rod bottom lights ___ (1) ___ be LIT.
The Unit Supervisor will ____ (2) ____.

A. (1) will
(2) use gammametrics in lieu of N-31 to comply with Technical Specification 3.3.1 and to confirm the reactor trip.

B. (1) will
(2) NOT use gammametrics in lieu of N-31 to comply with Technical Specification 3.3.1 but can use gammametrics to confirm the reactor trip.

C. (1) will NOT
(2) use gammametrics in lieu of N-31 to comply with Technical Specification 3.3.1 and to confirm proper overlap.

D. (1) will NOT
(2) NOT use gammametrics in lieu of N-31 to comply with Technical Specification 3.3.1 but can use gammametrics to confirm proper overlap.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Correct that a reactor trip has occurred and rod bottom lights would be LIT, however, in accordance with AOP 2.2.1A, gammametrics cannot be used to comply with TS 3.3.1.

**B.** Correct. The candidate must recognize that the plant is currently just below P-6 and just on scale in the IR. If discrimination voltage fails high this failure will cause N-31 to indicate $1 \times 10^6$ cps. 1 of 2 SR indications > $1 \times 10^5$ cps results in a reactor trip as evidenced by rod bottom lights LIT. AOP 2.2.1A states that gamametrics can provide indication but is not to be used to meet TS 3.3.1 requirements.

**C.** Incorrect. Reactor trip has occurred. Rod bottom lights will be LIT. Gammametrics in this condition cannot be used to comply with TS 3.3.1.

**D.** Incorrect. Reactor trip has occurred. Rod bottom lights will be LIT. The use of gammametrics is correct and plausible for a reactor trip had one occurred.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 032 | Loss of Source Range Nuclear Instrumentation | Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: | Confirmation of reactor trip |

**K/A#** AA2.06 **K/A Importance** 4.1 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-2.1.B, Rev. 3; 2OM-53C.4.2.2.1A, Rev. 8; 3SQS-2.1 LP PPT, Rev. 6

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-2.1 14. Analyze a given set of conditions to determine what NIS failure has occurred.
15. Given an NIS failure, predict NIS and intersystem related response for the given failure.

84. Given the following plant conditions:

- While operating at 100% power, a drop in Pressurizer (PRZR) pressure results in a reactor trip and safety injection.
- Containment pressure is 46 psig and slowly RISING.
- Reactor Coolant Pumps have been secured.
- RVLIS Full Range is indicating 20%.
- Three Max Core Exit Thermocouples are indicating 745°F and slowly RISING.
- RCS pressure is 47 psig and STABLE.
- The crew is currently performing actions in E-1, "Loss of Reactor or Secondary Coolant".

Based on these plant conditions, which ONE of the following is the required action for the Unit Supervisor to take?

A. Continue actions of E-1 to restore/monitor ESF equipment.

B. Transition to FR-C.2, "Response to Degraded Core Cooling".

C. Transition to FR-C.1, "Response to Inadequate Core Cooling".

D. Transition to FR- Z.1, "Response to High Containment Pressure".

---

**Answer:** C

**Explanation/Justification:**

**A.** Incorrect. Although it is correct that ESF equipment should be restored/monitored, procedural rules of usage require a transition to the higher priority CSF's as a result of beyond design basis accident conditions.
**B.** Incorrect. FR-C.2 is a lower priority orange path procedure. The candidate must recognize that the parameters provided warrant a red versus orange path entry.
**C.** Correct. According to CSFST's and EOP rules of usage, the US shall transition to the highest red path condition which is an inadequate core cooling condition. Adequate subcooling does not exist, No RCPs with three max TC > 729 and < 40% RVLIS entry into FR-C.1 is required.
**D.** Incorrect. FR-Z.1 is a lower priority red path condition. 45 psig is an entry condition for this FRP.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 074 | Inadequate Core Cooling | N/A | Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant integrity, containment conditions, radioactivity release control, etc. |

**K/A#** 2.4.21 **K/A Importance** 4.6 **Exam Level** SRO

**References provided to Candidate** Steam Tables **Technical References:** 2OM-53A.1.F.02, Rev. 1; 1/2OM-53B.2, Rev. 7

**Question Source:** Modified Bank BVPS Unit 1 2007 Audit Exam **Level Of Difficulty: (1-5)**

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

**Objective:** 3SQS-53.1

1. Apply from memory all of the EOP user guide rules of usage as defined in 1/2OM53B.2.
2. Concerning CSF restoration, IAW BVPS EOP Executive Volume, state from memory the following: The CSF in order of priority, the priorities of the color coded end points of the CSF status trees, the red path summary conditions from the EOPs.

85. Given the following plant conditions:

- A Large Break Loss of Coolant Accident has occurred.
- The Unit Supervisor has completed ES-1.3, "Transfer to Cold Leg Recirculation" and transitions back to E-1, "Loss of Reactor or Secondary Coolant", procedure step in effect.
- Containment Pressure is 10 psig and slowly DROPPING.
- No containment Quench Spray Pumps are running.
- Containment Sump Level is 190 inches and slowly RISING.

Based on these plant conditions, what Functional Restoration Procedure (FRP) transition is required <u>AND</u> what actions will be taken by the Unit Supervisor (US)?

An orange path exists, transition to ___ (1) ___.
The US will ___ (2) ___.

A. (1) FR-Z.2, "Response to Containment Flooding".
(2) remain in FR-Z.2 until the orange condition has cleared.

B. (1) FR-Z.1, "Response to High Containment Pressure".
(2) remain in FR-Z.1 until the orange condition has cleared.

C. (1) FR-Z.2, "Response to Containment Flooding".
(2) after completing FR-Z.2 procedural steps, transition back to E-1.

D. (1) FR-Z.1, "Response to High Containment Pressure".
(2) after completing FR-Z.1 procedural steps, transition back to E-1.

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**<u>Answer:</u> C**

**Explanation/Justification:**

**A.** Incorrect. It is correct that an orange path FR-Z.2 condition does exist. However, the SRO should know the rules of usage and background information which states that once all actions of this procedure are completed, the operator shall return to procedure and step in effect.

**B.** Incorrect. Although plausible, an orange condition exists only if containment pressure is > 11 psig with no quench spray pumps. In the stated conditions, containment pressure is 10 psig and dropping. Again the procedural usage portion is incorrect.

**C.** Correct. According to 2OM-53A.1.F-0.5, containment sump level > 187 inches is an orange path condition and warrants entry into FR-Z.2. 2OM-53B.4.FR-Z.2 background document states that once all actions of this procedure are completed that the containment status tree function may not be restored to a green condition. In this case, the appropriate FRP does not need to be implemented again since all actions have already been performed in order to adhere to facility license and amendments.

**D.** Incorrect. Incorrect procedure but correct procedural usage for the actions of this procedure as explained above.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| W/E15 | Containment Flooding | Ability to determine and interpret the following as they apply to the (containment flooding) | Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments. |

**K/A#** EA2.2 **K/A Importance** 3.3 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.F-0.5, Rev. 3
2OM-53B.4.FR-Z.2, Rev. 2

**Question Source:** New **Level Of Difficulty: (1-5)**

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

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

86. Given the following plant conditions:

- The Unit is in Mode 4 with "A" Train of Residual Heat Removal (RHS) in service.
- The "B" Train of RHR is available.
- No Reactor Coolant Pumps (RCPs) are operating.
- All systems are in NSA for the current mode of operation.
- A1-5H, "RESIDUAL HEAT REMOVAL SYSTEM TROUBLE" alarm is received.
- The Reactor Operator reports 2RHS*HCV758A, "RHS Train "A" HX Outlet Flow Control Valve", has drifted to the FULL CLOSED position and will **NOT** respond.

Which ONE of the following describes the specific impact on RHS, TWO (2) minutes after 2RHS*HCV758A fails CLOSED?

Train A RHS flow ____ (1) ____ AND RHS HX "A" Outlet Temperature will ____ (2) ____.
Train "A" RHS Loop ____ (3) ____ in accordance with T.S. 3.4.6, "RCS Loops – Mode 4" bases.

A. (1) rises
(2) drop
(3) is OPERABLE

B. (1) remains the same
(2) rise
(3) is OPERABLE

C. (1) drops
(2) rise
(3) is **NOT** OPERABLE

D. (1) remains the same
(2) rise
(3) is **NOT** OPERABLE

---

**Answer:** D

**Explanation/Justification:**

**A.** Incorrect. Incorrect system response. This would be the system response if 2RHS*HCV758A failed open. Incorrect that the system is operable, however, if the candidate thinks that the system flow increases, it is plausible that the system will still be operable by definition.

**B.** Incorrect. Correct system response. Incorrect that the system is operable.

**C.** Incorrect. Incorrect system response. Correct that the system is inoperable.

**D.** Correct. If 2RHS*HCV758A fails closed, 2RHS*FCV605A which is NSA automatic will open to maintain a set system flow. Although system flow in Train A remains the same, there is now more flow bypassing Train A RHS HX, so the result is an increase in Train A RHS temperature as well as RCS temperature. 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 2RHS*HCV758A is in the flowpath and not capable of being opened, the RHS loop 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.

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| 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:** OM Fig. 10-1, Rev 16, 2OM-10.1.D, TS 3.4.6 Bases, Rev 0, 1/2OM-48.1.I, Rev. 24

**Question Source:** Modified Bank Vision - 9121 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comprehension **10 CFR Part 55 Content:** (CFR 43.2 / 43.5 / 45.3 / 45.13)

**Objective:** 2SQS-10.1 18. Given a specific plan condition, predict the response of the RHS control room indications and control loops, including all automatic functions and changes in equipment status, for either a change in plant condition or for off-nor.

87. Given the following plant conditions:

- The Unit is operating at 100% power with all systems in NSA.
- 2OST-13.2, "Quench Spray Pump [2QSS*P21B] Test" is being performed.
- During performance of this surveillance, the following Containment parameters are noted:
  - Containment temperature is 110 °F.
  - Containment pressure is 13.8 psia.

Which **ONE** of the following Technical Specification(s) LCO(s) are **NOT** met **AND** what is the bases?

A. TS 3.6.4, "Containment Pressure". To ensure DBA LOCA analysis initial assumptions are preserved **ONLY**.

B. TS 3.6.5, "Containment Air Temperature". To ensure CIB will be actuated by containment pressure **ONLY**.

C. TS 3.6.6, "Quench Spray System". To provide the heat removal capability during DBA LOCA **AND** TS 3.6.4, "Containment Pressure". To ensure DBA LOCA analysis initial assumptions are preserved.

D. TS 3.6.6, "Quench Spray System". To provide the heat removal capability during DBA LOCA **AND** TS 3.6.5, "Containment Air Temperature". To ensure assumptions for LOCA/SLB analysis preserved.

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**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. Containment pressure is within required band of 12.8 - 14.2 psia, therefore no entry is required. Correct TS bases.

**B.** Incorrect. Containment air temperature is NOT within required band of 70 to 108 F, therefore entry is required. Incorrect bases for lower limit.

**C.** Incorrect. Containment Quench Spray pumps are all required to be operable in Mode 1. 2QSS*P21B is made inoperable in accordance with 2OST-13.2. Correct 3.6.6 bases. TS 3.6.4 does NOT apply, however bases is correct.

**D.** Correct. . Containment Quench Spray pumps are all required to be operable in Mode 1. 2QSS*P21B is made inoperable in accordance with 2OST-13.2. Correct 3.6.6 bases. Containment air temperature is NOT within required band of 70 to 108 F therefore entry into TS 3.6.5 is required. Correct bases for 3.6.5. SRO level due to knowledge of TS bases that is required to analyze TS required actions as well as application.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 022 | Containment Cooling | N/A | Ability to recognize system parameters that are entry-level conditions for Technical Specifications. |

**K/A#** 2.2.42 **K/A Importance** 4.6 **Exam Level** SRO

**References provided to Candidate** None

**Technical References:** 2OST-13.2, Rev. 30; TS 3.6.4 and Bases, Rev. 6; TS 3.6.5 and Bases, Rev. 0; TS 3.6.6 and Bases, Rev. 12

**Question Source:** New

**Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory

**10 CFR Part 55 Content:** (CFR 41.7 / 41.10 / 43.2 / 45.3)

**Objective:** 2SQS-13.1 Given a set of plant conditions, recommend action(s) for the SM that mitigates the condition, including the basis for the recommendations.

88. Given the following plant conditions:

- The plant is operating at 100% power.
- At 0830 on July 3rd, the (A) Quench Spray Pump (2QSS*P21A) is declared **INOPERABLE**.
- At 2300 on July 5th, the (B) Quench Spray Pump (2QSS*P21B) becomes **INOPERABLE.**
- At 0215 on July 6th, the (A) Quench Spray Pump (2QSS*P21A) 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 2QSS*P21B to **OPERABLE** status, without requiring a unit shutdown?

A. 0830 on July 6th

B. 0830 on July 7th

C. 2300 on July 8th

D. 2300 on July 9th

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**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.

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

**K/A#** A2.04 **K/A Importance** 4.2 **Exam Level** SRO

**References provided to Candidate** TS 1.3, Amend 278/161; TS 3.6.6, Amend 278/161

**Technical References:** TS 1.3, Amend 278/161; TS 3.6.6, Amend. 278/161

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.5 / 43.5 / 45.3 / 45.13)

**Objective:** 2SQS-13.1 20. Using a copy of TS and/or LRM, analyze a given set of plant conditions for compliance with the licensing requirements; including the determination of equipment operability and applicable actions statements.

89. 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 < 12%.
- All Auxiliary Feedwater pumps have failed to automatically start and **cannot** be manually started.
- A transition to FR-H.1, "Response to Loss of Secondary Heat Sink" is made by the operating crew.

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.

C. 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.

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**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. RCS pressure is less than the shutoff of the LHSI pumps (approximately 160 psig). Furthermore, E-1 uses 225 psig to determine the mitigation strategy for a large or small break LOCA. Less than 225 psig, the EOP directs remaining in E-1 until cold leg recirculation criteria is met. Greater than 225 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.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 061 | Auxiliary/Emergency Feedwater (AFW) System | Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: | Automatic control malfunction |

**K/A#** A2.05 **K/A Importance** 3.4 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-53A.1.FR-H.1, Rev. 9
2OM-53B.4.FR-H.1, Rev. 9

**Question Source:** Bank Vision - 45809 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comprehension **10 CFR Part 55 Content:** (CFR 41.5 / 43.5 / 45.3 / 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.

90. Given the following plant conditions:

- The Unit is at 75% power with all systems in NSA.
- Train "B" is the Protected Train.
- Simultaneous annunciators are received.

In accordance with the guidance in 1/2OM-48.2.C, "Adherence and Familiarization To Operating Procedures", which ONE of these <u>**VALID**</u> annunciators (simultaneously received), will be addressed first?

A. **A7-4G**, "GENERATOR PT BLOWN FUSE" (Yellow Border)

B. **A5-8C**, "US SERV TRMR 2C OVERCURRENT GEN TRIP" (No Border)

C. **A8-3B**, "4160V EMER BUS 2DF ACB2F7 OVERCURRENT TRIP" (No Border)

D. **A7-2H**, "TURBINE E-H FLUID TROUBLE/LOSS OF DC CONTROL" (Yellow Border)

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**<u>Answer:</u> B**

**Explanation/Justification:**

**A.** Incorrect. This is a plausible AC electrical alarm that has a lower priority then the A5 alarm.
**B.** Correct. In accordance with ½ OM-48.2.C, First Out Annunciators (A5) take priority over red or yellow border annunciators. These alarms do not have special marking since they already have a red/white feature to identify their significance. It is the SRO responsibility to determine priority.
**C.** Incorrect. This is a plausible AC electrical alarm that would be of lesser priority then the other alarms.
**D.** Incorrect. This is a plausible alarm that would be of lesser priority than the A5 alarm.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 062 | AC Electrical Distribution | N/A | Ability to prioritize and interpret the significance of each annunicator or alarm. |

**K/A#** 2.4.45 **K/A Importance** 4.3 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 1/2OM-48.2.C, Rev. 16; 2OM-36.4.AAC, Issue 1, Rev. 16

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.10 / 43.5 / 45.3 / 45.12)

**Objective:** 3SQS-36.1 Given a 4KV alarm condition and using alarm response procedure(s), determine the appropriate alarm response, including automatic and operator actions in the control room.

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

- The Unit is operating at 100% power with all systems in NSA.
- Pressurizer (PRZR) Level Channel 2RCS*LT459 fails LOW.
- A reactor trip occurs as a result of no operator action.
- The Balance of Plant Operator (BOP) requests to perform pre-emptive actions during the course of E-0, "Reactor Trip or Safety Injection" immediate actions.
- Pre-emptive actions were NOT pre-briefed prior to reactor trip.

What caused the reactor trip AND how will the SRO prioritize response procedures?

The reactor trip was caused by _____ (1) ______.
The SRO _____ (2) _____ allow the BOP to carry out pre-emptive actions at this time.

A. (1) HIGH PRZR Level
(2) shall NOT

B. (1) LOW PRZR Level
(2) shall

C. (1) HIGH PRZR Pressure
(2) shall

D. (1) LOW PRZR Pressure
(2) shall NOT

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. A low failure of 2RCS*LT459 during NSA results in a letdown isolation (<14% PZR level). With no operator action charging will continue to fill the PRZR until it reaches 92% on 2/3 PRZR channels resulting in a reactor trip. BVPS-OPS-0024, "Transient Response Guidelines" states that pre-emptive actions will be performed only after IOAs are performed and with SRO concurrence. IOAs will be considered performed only after they are read and verified. The procedure allows relaxation of this requirement if they were pre-briefed prior to tripping the reactor. It is an SRO responsibility to have detailed knowledge of how and when to implement procedures including how to coordinate procedure steps.

**B.** Incorrect. There is no trip on low PRZR level however this is a common misconception. SRO shall NOT allow BOP to carry out pre-emptive actions until IOAs are complete (refer to correct answer explanation)

**C.** Incorrect. RCS pressure will increase due to increasing PRZR level which makes this a plausible reactor trip. SRO shall NOT allow BOP to carry out pre-emptive actions until IOAs are complete. (refer to correct answer explanation)

**D.** Incorrect. This is a valid reactor trip however PRZR will increase as opposed to decrease during these plant conditions. Correct that the SRO shall not allow pre-emptive actions to occur.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| 011 | Pressurizer Level Control System (PZR LCS) | N/A | Knowledge of annunciator alarms, indications, or response procedures |

**K/A#** 2.4.31 **K/A Importance** 4.1 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 2OM-6.4.IF, Rev. 12; BVBP-OPS-0024, Rev. 3

**Question Source:** New **Level Of Difficulty: (1-5)**

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

**Objective:** 2SQS-6.4 21. Given a set of plant conditions and appropriate procedures, apply the operational sequences, parameter limits, P&Ls, cautions and notes applicable to the completion of the task activities in the control room: 2OM-6.4.IF

92. Given the following plant conditions:

- The Unit is operating at 100% power with all systems in NSA.
- Intermediate Range (IR) Channel N35 compensating voltage fails low.

Based on these plant conditions, what procedure will be used to mitigate the consequences of this event **AND** what impact would this failure have if a reactor trip were to occur prior to procedural implementation?

The Unit Supervisor will implement ___ (1) ___ actions in response to this failure.
If a reactor trip were to occur prior to implementation of these actions, N35 will indicate ____ (2) ____.

A. (1) AOP 2.2.1B, "Intermediate Range Channel Malfunction"
(2) low and the source range will be energized as soon as N36 is < P-6.

B. (1) AOP 2.2.1B, "Intermediate Range Channel Malfunction"
(2) high and the source range will need to be manually re-energized as soon as N36 is < P-6.

C. (1) 2OM-1.4.IF, "Reactor Control and Protection Operating Instrument Failure Procedure"
(2) low and the source range will be energized as soon as N36 is < P-6.

D. (1) 2OM-1.4.IF, "Reactor Control and Protection Operating Instrument Failure Procedure"
(2) high and the source range will need to be manually re-energized as soon as N36 is < P-6.

---

**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. AOP 2.2.1B will be entered in response to a N35 drifting indication. If compensating voltage drifts low, the result will be a higher than normal indication. It is plausible if N35 indicated low that SR would energize as soon as N36 is < P-6.

**B.** Correct. Correct procedural reference. N35 will fail high based on a loss of compensating voltage. If N35 fails high than 2/2 logic will not occur when N36 drops below P-6 which requires manual re-energization when N36 is < P-6. SRO must select correct procedures and have detailed knowledge of what compensatory actions are addressed in the procedure.

**C.** Incorrect. 2OM-1.4.IF although plausible based on its title is not applicable to an NI failure. NI failures are the only instrument failures at BVPS that have separate AOP's and are not addressed by a specific instrument failure procedure. If compensating voltage drifts low, the result will be a higher than normal indication. It is plausible if N35 indicated low that SR would energize as soon as N36 is < P-6.

**D.** Incorrect: 2OM-1.4.IF although plausible based on its title is not applicable to an NI failure. NI failures are the only instrument failures at BVPS that have separate AOP's and are not addressed by a specific instrument failure procedure. N35 indication and source range response are correct.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 015 | Nuclear Instrumentation System (NIS) | Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and based on these predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: | Faulty or erratic operation of detectors or compensating voltage |

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

**References provided to Candidate** None **Technical References:** 2OM-53C.2.2.1B, Rev. 3; 1/2OM-2.1.B, Rev. 3

**Question Source:** Modified Bank Vision - 1615 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Comprehension **10 CFR Part 55 Content:** (CFR 41.5 / 43.5 / 45.3 / 45.5)

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

93. Given the following plant conditions:

- Unit 2 is currently in Mode 6.
- Fuel movement is in progress.
- Spent Fuel Pool (SFP) boron concentration (Cb) sample results have significantly dropped since last sample and are currently at the Technical Specification limit of 2000 PPM.

If SFP Cb continues to drop what is the impact <u>AND</u> what will be the Technical Specification required action?

A 5% (Keff < .95) shutdown margin will be ____ (1) ____.
The Technical Specification required action will be that fuel movement in the SFP ___ (2) ___.

A. (1) maintained regardless of SFP Cb.
(2) can continue as long as fuel is moved into proper SFP regions.

B. (1) **<u>no longer</u>** maintained regardless of SFP Cb.
(2) must be immediately suspended and boron concentration restored within limits.

C. (1) maintained as long as no credible boron dilution event reduces SFP Cb < 450 PPM.
(2) can continue as long as fuel is moved into proper SFP regions.

D. (1) maintained as long as no credible boron dilution event reduces SFP Cb < 450 PPM.
(2) must be immediately suspended and boron concentration restored within limits.

---

**<u>Answer:</u> D**

**Explanation/Justification:**

**A.** Incorrect. Unit 1 does not require soluble boron in their SFP to maintain Keff < .95 provided that storage verification has been completed. For Unit 2, Keff cannot be maintained < .95 for a credible dilution event. (refer to correct answer explanation)

**B.** Incorrect. Unit 1 does not require soluble boron in their SFP to maintain Keff < .95 provided that storage verification has been completed. It is not correct that Keff < .95 is maintained regardless of SFP Cb in Unit 2. The second part of the statement is correct.

**C.** Incorrect. First part is correct (refer to correct answer explanation). Second part is incorrect action statement for Unit 2 but plausible for Unit 1.

**D.** Correct. According to TS 3.7.16 and its associated bases, the >2000 PPM limit conservatively assures Keff is maintained within the limit (Keff <.95) for the worst case misplaced fuel assembly accident. In addition, this limit ensures no credible boron dilution event will reduce Cb < 450 ppm required during non-accident conditions to maintain Keff < .95. TS 3.7.16 required action is to immediately restore SFP Cb to > 2000 ppm and suspend fuel movement within the SFP.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| 033 | Spent Fuel Pool Cooling System (SFPCS) | Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System; and (b) based on these predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: | Inadequate SDM |

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

**References provided to Candidate** None **Technical References:** TS 3.7.16, Amend. 278/161
TS 3.7.16 Bases, Rev. 5

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.5 / 43.5 / 45.3 / 45.13)

**Objective:** 2SQS-20.1 13. Describe the design bases for the fuel pool cooling and purification system and the associated major components as documented in the UFSAR.

14. Using a copy of the TS or LRM, analyze a given set of plant conditions for compliance with the licensing requirements, including the determination of equipment operability and applicable action statements.

94. Which of the following refueling activities must be performed by the **Refueling** Senior Reactor Operator (SRO) in accordance with 1/2OM-48.1.A, "Duties and Responsibilities of the Operations Group"?

1. Ensure Refueling Technical Specifications are met prior to beginning the shift.
2. Ensure overall adherence to Refueling Technical Specifications.
3. Conduct a Pre-Job Brief prior to initial fuel movement to ensure contingency actions are understood.
4. Authorize temporary changes to the refueling procedures.

A. 1 and 3 **ONLY**.

B. 1 and 2 **ONLY**.

C. 2 and 3 **ONLY**.

D. 3 and 4 **ONLY**.

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**Answer: A**

**Explanation/Justification:**

**A.** Correct. According to 1/2OM-48.1.A, the refueling SRO is responsible for refueling TS prior to beginning the shift and conducting a pre-job brief prior to assuming the shift for initial fuel movement.
**B.** Incorrect. The SM is responsible according to 1/2OM-48.1 for overall adherence to all refueling technical specifications.
**C.** Incorrect. The SM is responsible according to 1/2OM-48.1 for overall adherence to all refueling technical specifications.
**D.** Incorrect. Temporary changes to refueling procedures are authorized by the control room SRO.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Conduct of Operations | Knowledge of the fuel handling responsibilities of SROs |

**K/A#** 2.1.35 **K/A Importance** 3.9 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 1/2OM-48.1.A, Rev. 2

**Question Source:** Bank Vision - 46675 **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Fundamental **10 CFR Part 55 Content:** (CFR 41.10 / 43.7)

**Objective:**
3SQS-48.1 1. From memory, explain the duties and responsibilities of Operations Personnel.
3SQS-6.14 2. 2. State the refueling responsibilities of the following BVPS personnel: Refueling SRO.

95. Given the following plant conditions:

- The Unit is in Mode 6.
- Core reload is in progress in accordance with 1/2RP-3.24, "Core Reload".

Under which of the following circumstances is the Refueling SRO authorized to BYPASS Refueling Interlocks, according to 1/2RP-3.24?

1. When moving underwater lights in the core.
2. When positioning Fuel Assembly Loading Guides (shoehorns).
3. During emergency conditions.
4. During non-emergency conditions when an interlock malfunctions.

A. 1 AND 3 **ONLY.**

B. 1, 2 AND 4 **ONLY.**

C. 2, 3 AND 4 **ONLY.**

D. 1, 2, AND 3 **ONLY.**

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**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. May bypass interlocks when positioning fuel assembly loading guides.
**B.** Incorrect. May bypass during emergency conditions when authorized by the refueling SRO. May not bypass when an interlock malfunctions under non-emergency conditions.
**C.** Incorrect. May bypass interlocks when moving underwater lights in the core. May not bypass when an interlock malfunctions under non-emergency conditions.
**D.** Correct. According to 1/2RP-3.24, the SRO may authorize bypassing refueling interlocks when moving underwater lights and positioning the fuel assembly loading guides. Interlocks shall not be bypassed unless authorized by written procedure unless directions are given in an emergency condition by the Refueling SRO.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| N/A | N/A | Conduct of Operations | Knowledge of refueling administrative requirements. |

**K/A#** 2.1.40 **K/A Importance** 3.9 **Exam Level** SRO
**References provided to Candidate** None **Technical References:** 1/2RP-3.24, Issue 0, Rev. 7
**Question Source:** Modified Bank Vision - 45854 **Level Of Difficulty: (1-5)**
**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (41.10 / 43.5 / 45.13)
**Objective:** 3SQS-6.14 3. State when refueling interlocks can be bypassed.

96. Given the following plant conditions:

- The Unit is operating at 100% power with all systems in NSA.
- Scaffolding is scheduled to be built around an operating Heater Drain Pump in preparation for upcoming outage.
- No grid activities are scheduled to be performed.
- You are asked as part of your Senior Reactor Operator (SRO) responsibilities to perform a risk assessment in accordance with NOP-OP-1007, "Risk Management".

Using the reference provided, which ONE of the following describes the **HIGHEST** classification of risk associated with the stated plant conditions?

A. Green

B. Yellow

C. Orange

D. Red

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**Answer: B**

**Explanation/Justification:**

**A.** Incorrect. Refer to correct answer explanation.

**B.** Correct. According to NOP-OP-1007, "Risk Management", section 2.3 specifies that exempt activities are listed in Attachment 1. Exempt activities shall be conducted as Green Risk activities. Attachment 1, bullet 4 states that for the work to be exempt is must be a maintenance activity where work is not within the power block which is not the case (Heater Drain Pump is within power block), so therefore this is NOT a green risk exempt activity. Section 2.7 states that for plant conditions a SRO shall perform risk assessment using Attachment 2 & 3. Attachment 3, bullet 4 states that any physical activity performed near protected train equipment or trip sensitive equipment that would cause a plant transient is a yellow risk activity.

**C.** Incorrect. Refer to correct answer explanation.

**D.** Incorrect. Refer to correct answer explanation.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Equipment Control | Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator. |

**K/A#** 2.2.17 **K/A Importance** 3.8 **Exam Level** SRO

**References provided to Candidate** NOP-OP-1007, Rev. 7 (Procedure Body and Attachment 1,2, 3 ONLY) **Technical References:** NOP-OP-1007, Rev. 7

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.10 / 43.5 / 45.13)

**Objective:** 9. Explain the process for planning and scheduling maintenance and PRA integration into the 12 week schedule in accordance with NOP-OP-1007.

97. 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 <u>**AND**</u> 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.

C. (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 to individuals in which radiation levels could result in a dose >/= 500 R/hr at a distance of 1 meter from a radiation source. 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 area to individuals in which radiation levels could result in dose rates >/= 1000 mr/hr. 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 control room 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. VHRA's require the use of two keys.

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| 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. 1 / TS 5.7, Amend. 278/161

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Lower - Memory **10 CFR Part 55 Content:** (CFR 41.12 / 43.4 / 45.9 / 45.10)

**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.

98. Given the following plant conditions:

- The plant is operating at 100% power with all system in NSA.
- Gaseous Waste Storage Tanks (2GWS-TK25A-G) pressures are 10 psig and STABLE.
- Gaseous Waste Surge Tank (2GWS-TK21) pressure is 62 psig and STABLE.
- The Waste Gas Storage Tank Radiation Monitor (2GWS-RQ104) is out of service (OOS).
- Oxygen Analyzer (2GWS-OA100A) is also OOS.
- Reactor Coolant System activity is 25 μCi/ml.

It is desired to fill the Gaseous Waste Storage Tanks in accordance with 2OM-19.4.G, "Filling Unit 2 Gaseous Waste StorageTanks from Unit 2 Surge Tank".

Given these plant conditions, while filling the Gaseous Waste Storage Tanks, what LRM/ODCM compensatory actions are **REQUIRED**?

At least once per ___ (1) ___ hours, take grab samples and analyze for ____ (2) ____.

A. (1) FOUR
(2) Oxygen concentration **ONLY**.

B. (1) FOUR
(2) Oxygen concentration **AND** once per 24 hours for radioactive content.

C. (1) TWENTY FOUR
(2) Oxygen concentration **ONLY**.

D. (1) TWENTY FOUR
(2) BOTH Oxygen concentration **AND** radioactive content.

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**Answer: D**

**Explanation/Justification:**

**A.** Incorrect. This oxygen sample time is the time limit if BOTH oxygen analyzers were OOS. If the candidate does NOT correctly apply the ODCM surveillance, then this distractor would appear plausible.
**B.** Incorrect. This Oxygen sample time is the time limit if BOTH oxygen analyzers were OOS. Correct actions for radioactive content.
**C.** Incorrect. At Unit 2 both Oxygen and radioactive content must be sampled and analyzed. If the candidate does NOT correctly apply the ODCM surveillance, then this distractor would be plausible.
**D.** Correct. In accordance with LRM 3.3.12 condition B.1 and ODCM attachment O surveillance 4.11.2.5.1.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Radiation Control | Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions. |

**K/A#** 2.3.14 **K/A Importance** 3.8 **Exam Level** SRO

**References provided to Candidate** LRM 3.3.12, Rev. 52; TS 5.5.8, Amend 278/170; ½ ODCM Section 3.0.3; Rev. 8

**Technical References:** LRM 3.3.12, Rev. 52; TS 5.5.8, Amend 278/170; ½ ODCM Section 3.0.3, Rev. 8

**Question Source:** Bank 2LOT6 NRC Exam **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.12 / 43.4 / 45.10)

**Objective:**

99. Given the following plant conditions:

- The STA informs you of the following Critical Safety Function (CSF) Status Tree information:
  - All Narrow Range S/G Water levels are 5% and LOWERING.
  - Total available Feedwater Flow is Zero (0) GPM.
  - All Core Exit Thermocouples are 750 °F and slowly RISING.
  - RVLIS Full Range is 55% and slowly DROPPING.
  - No RCPs are currently running.

Which procedure transition is immediately required **AND** why?

A. FR-C.1, "Response to Inadequate Core Cooling" – Extreme Challenge to Clad/Matrix Barrier.

B. FR-C.1, "Response to Inadequate Core Cooling" – Severe Challenge to Vessel/Containment Barrier.

C. FR-H.1, "Response to Loss of Secondary Heat Sink" – Extreme Challenge to Clad/Matrix Barrier.

D. FR-H.1, "Response to Loss of Secondary Heat Sink" – Severe Challenge to Vessel/Containment Barrier.

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**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Although core cooling is a higher priority in terms of sequence, the red path will always trump an orange path condition according to EOP users guide. Incorrect reason why due to severe versus extreme and incorrect barriers challenged. Red path conditions are not met until RVLIS < 40%.

**B.** Incorrect. Incorrect in that an orange path is a lower priority than a red path. Based on stated plant conditions an orange core cooling path is met only. Correct reason why.

**C.** Correct. In accordance with 1/2OM-53B.2, even though Core Cooling is a higher priority than Heat Sink, the first red path encountered must be entered. FR-H.1 Bases states that a red path on heat sink is an extreme challenge to clad/matrix barrier and immediate operator attention is warranted. With NR S/G water levels < 12% and available total feedwater flow @ 0 GPM a Red Heat Sink Path exists, SRO is responsible for prioritizing and selecting appropriate procedure.

**D.** Incorrect. Correct procedure. The challenge to FR-H.1 is extreme versus severe since a red versus orange path exists. Also the challenge is to the clad versus vessel/containment.

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| **Sys #** | **System** | **Category** | **KA Statement** |
|---|---|---|---|
| N/A | N/A | Emergency Procedure/Plan | Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. |

**K/A#** 2.4.22 **K/A Importance** 4.4 **Exam Level** SRO

**References provided to Candidate** None **Technical References:** 1/2OM-53B.2, Rev. 7
2OM-53B.4.F-0.2, Rev. 1
2OM-53B.4.F.0.3, Rev. 2
2OM-53A.1.F-0.2, Rev. 1
2OM-53A.1.F.0.3, Rev. 2

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.7 / 41.10 / 43.5 / 45.12)

**Objective:** 3SQS-53.1 3SQS-53.3 2. Concerning CSF restoration, IAW BVPS EOP Executive Volume, state from memory the following: The CFS in the order of priority, the priorities of the color coded end points of the CSFSTs, The red path summary conditions for EOPs.
8. For a given event, apply the CSFS form to advise the operating crew of CSF priorities.

100. Given the following plant conditions:

- Unit 2 is in Mode 4 following a refueling outage.
- An explosive device detonates in the Unit 2 Spent Fuel Pool Area.
- The Outside Tour Operator reports visible damage and a small fire around the spent fuel pool outside wall.
- Security confirms armed intruders have arrived from the Ohio River and are being contained at the Administrative Building by site security force.
- Radiation levels are all normal.

Based on these plant conditions and using EPP-I-1b, "Recognition and Classification of Emergency Conditions", determine the highest classification for this event?

A. Alert

B. Unusual Event

C. Site Area Emergency

D. General Emergency

---

**Answer: C**

**Explanation/Justification:**

**A.** Incorrect. Plausible based on Tab 4.6 (Security), Tab 4.1 (Fire), Tab 4.2 (Explosion), or Tab 4.7 (ED Judgement). However, this is not the highest area of classification. These classifications are valid.

**B.** Incorrect. There are many Tabs which could be classified as an Unusual Event, such as Tab 4.1 (Fire), Tab 4.2 (Explosion), Tab 4.6 (Security), or Tab 4.7 (ED Judgement), however, this is not the highest area of classification. These classifications are valid.

**C.** Correct. Tab 4.6 of EPP-I-1b states that a site area emergency is applicable if a notification from site security force that an armed attack, explosive attack, or other hostile action is occurring or has occurred in the protected area. If the intruders have reached the Administrative Building, they have breached the protected area fence.

**D.** Incorrect. There are no conditions which would result in a GE classification. For escalation in Tab 4.6 (security), Hostile force would have to take control of plant equipment required to maintain safety functions. For escalation in Tab 4.7 (ED Judgement), security event would have to result in an actual loss of physical control of the facility or releases that can be reasonably expected to exceed EPA protective action guidelines. The stem of the question states that radiation levels are normal and that site security is containing the intruders at the Administrative Building. This means that the intruders are inside the protected area but have not reached the actual plant.

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| Sys # | System | Category | KA Statement |
|---|---|---|---|
| N/A | N/A | Emergency Procedures/Plan | Knowledge of the procedures related to a security event (non-safeguards information) |

**K/A#** 2.4.28 **K/A Importance** 4.1 **Exam Level** SRO

**References provided to Candidate** EPP-I-1b (Pg 14-53)/Divider Card, Rev. 14 **Technical References:** EPP-I-1b (Pg 14-53)/Divider Card, Rev.14

**Question Source:** New **Level Of Difficulty: (1-5)**

**Question Cognitive Level:** Higher - Application **10 CFR Part 55 Content:** (CFR 41.10 /43.5 /45.11)

**Objective:** EPP-9281 11. Given specific plant conditions, classify the condition in accordance with EPP I-1A & B.

# 2LOT7 NRC Exam ANSWER KEY

NAME:____________________________

SSN:_______________________________

| | | | | |
|---|---|---|---|---|
| 1. | A | | C | D |
| 2. | A | | C | D |
| 3. | A | B | | D |
| 4. | | B | C | D |
| 5. | | B | C | D |
| 6. | | B | C | D |
| 7. | A | B | C | |
| 8. | | B | C | D |
| 9. | A | B | | D |
| 10. | | B | C | D |
| 11. | A | B | C | |
| 12. | A | B | | D |
| 13. | A | | C | D |
| 14. | A | | C | D |
| 15. | A | B | | D |
| 16. | | B | C | D |
| 17. | A | B | C | |
| 18. | A | B | | D |
| 19. | A | B | | D |
| 20. | A | | C | D |
| 21. | A | | C | D |
| 22. | A | B | | D |
| 23. | A | B | | D |
| 24. | A | B | | D |
| 25. | A | B | C | |
| 26. | A | | C | D |
| 27. | A | B | | D |
| 28. | | B | C | D |
| 29. | | B | C | D |
| 30. | | B | C | D |
| 31. | A | | C | D |
| 32. | A | B | C | |
| 33. | A | B | C | |
| 34. | | B | C | D |
| 35. | A | B | | D |
| 36. | A | | C | D |
| 37. | | B | C | D |
| 38. | A | B | | D |
| 39. | A | B | C | |
| 40. | A | | C | D |
| 41. | A | B | | D |
| 42. | A | B | | D |
| 43. | A | B | C | |
| 44. | A | | C | D |
| 45. | | B | C | D |
| 46. | A | B | C | |
| 47. | | B | C | D |
| 48. | A | | C | D |
| 49. | A | | C | D |
| 50. | A | B | | D |
| 51. | | B | C | D |
| 52. | | B | C | D |
| 53. | A | B | C | |
| 54. | A | B | C | |
| 55. | A | B | C | |
| 56. | A | B | | D |
| 57. | A | | C | D |
| 58. | A | B | C | |
| 59. | A | B | C | |
| 60. | A | B | | D |
| 61. | | B | C | D |
| 62. | A | | C | D |
| 63. | A | | C | D |
| 64. | A | B | C | |
| 65. | A | B | | D |
| 66. | A | | C | D |
| 67. | | B | C | D |
| 68. | A | | C | D |
| 69. | A | | C | D |
| 70. | A | B | | D |
| 71. | A | | C | D |
| 72. | A | B | | D |
| 73. | | B | C | D |
| 74. | A | B | | D |
| 75. | A | | C | D |
| 76. | A | B | | D |
| 77. | | B | C | D |
| 78. | A | | C | D |
| 79. | A | | C | D |
| 80. | A | B | C | |
| 81. | | B | C | D |
| 82. | A | B | C | |
| 83. | A | | C | D |
| 84. | A | B | | D |
| 85. | A | B | | D |
| 86. | A | B | C | |
| 87. | A | B | C | |
| 88. | A | | C | D |
| 89. | A | B | C | |
| ~~90.~~ | ~~A~~ | | ~~C~~ | ~~D~~ |
| 91. | | B | C | D |
| 92. | A | | C | D |
| 93. | A | B | C | |
| 94. | | B | C | D |
| 95. | A | B | C | |
| 96. | A | | C | D |
| 97. | A | B | | D |
| 98. | A | B | C | |
| 99. | A | B | | D |
| 100. | A | B | | D |