1. Plant Conditions:
   - A Reactor Trip has occurred from 100% power
   - DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture immediate actions have been completed

   The following indications are noted:
   - RCS Loop 1 flow: 79 mpph
   - RCS Loop 2 flow: 34 mpph
   - RCP status lights indicate that all 4 RCPs are operating
   - RCP 1-1 current reading = 257 amps
   - RCP 1-2 current reading = 257 amps
   - RCP 2-1 current reading = 161 amps
   - RCP 2-2 current reading = 221 amps

   (1) What plant condition has caused the indications noted?

   AND

   (2) Which of the following actions should be taken in accordance with DB-OP-02515, Reactor Coolant Pump and Motor Abnormal Operating procedure?

A. (1) RCP 2-1 Locked Rotor
(2) Trip RCP 2-1

B. (1) RCP 2-1 Locked Rotor
(2) Trip ALL RCPs

C. (1) RCP 2-1 Shaft Shear
(2) Trip RCP 2-1

D. (1) RCP 2-1 Shaft Shear
(2) Trip ALL RCPs

Answer: C

Explanation/Justification: This question is a KA match because it tests the candidate's ability to operate the Reactor Coolant Pumps following a reactor trip. In the situation provided, a sheared shaft on a reactor coolant pump will cause a reduction in RCS flow in that loop and lower than normal current readings for that RCP. This reduction in flow will cause a Reactor Trip if the plant was initially operating at 100% power. The action directed by the Abnormal Operating Procedure is to stop the affected RCP and therefore the ability of the candidate to monitor (change in RCS Flow) and operate (stop the affected RCP) is tested.

Note: Flow data and RCP amps for a sheared RCP 2-1 shaft is from the Plant Specific DB Simulator.

A. Incorrect
(1) Incorrect because a locked rotor would produce higher than normal motor current indications, not lower. Plausible because the RCS flows provided have Loop 1 at a higher flow which would be expected for a fault in a Loop 2 RCP.
(2) Correct - Tripping RCP 2-1 is the correct action directed by the Abnormal Operating Procedure DB-OP-02515.

B. Incorrect

(1) Incorrect because a locked rotor would produce higher than normal motor current indication not lower. Plausible because the RCS flows provided have Loop 1 at a higher flow which would be expected for a fault in a Loop 2 RCP.

(2) Incorrect because the controlling procedure DB-OP-02515 does not direct tripping all RCPs for this event. Plausible because there are examples of circulating debris in the reactor coolant system causing fuel failure and therefore stopping all RCPs may appear logical if the candidate assumes the RCS now contains RCP debris.

C. CORRECT

(1) Correct because the RCS flows are those indicated by the DB Specific Simulator for a sheared shaft on RCP 2-1.

(2) Correct because the Abnormal Operating Procedure DB-OP-02515 for this event, directs RCP shutdown on current flow of less than 200 amps.

D. Incorrect –

(1) Correct because the current reading provided is indicative of a sheared shaft (lower than normal motor current).

(2) Incorrect because tripping all RCPs is not directed however by DB-OP-02515. Plausible because there are examples of circulating debris in the reactor coolant system causing fuel failure and therefore stopping all RCPs may appear logical if the candidate assumes the RCS now contains RCP debris.

Sys #	System	Category	KA Statement
007	**Reactor Trip Stabilization, Recovery**	Ability to operate and monitor the following as they apply to a reactor trip:	RCP operation and flow rates.

K/A# EA1.04 **K/A Importance** 3.7 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02515 R18, Step 4.6.1 RNO

Question Source: Modified ANO 2014 NRC Exam Q1 **Level Of Difficulty: (1-5)** 2.5

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

Objective: OPS-GOP-115-02K and OPS-GOP-115-03K

Abbreviations: RCS = Reactor Coolant System, RCP = Reactor Coolant Pump.

Tier being Tested: ES401 Tier 1/Group 1 (RO)

2. Plant Conditions:
   - The Reactor is operating at 100% Power
   - Component Cooling Water (CCW) Pump 1 is in service
   - Component Cooling Water flow to all Reactor Coolant Pumps has been lost due to CC1411A, CCW CTMT ISO VLV, failing closed

   Which ONE of the following additional conditions would require immediately Tripping the Reactor AND Stopping ALL Reactor Coolant Pumps when conditions are confirmed?

A. MU38, RCP SEAL RETURN VLV fails closed.

B. Any RCP stator temperature reaching 225°F.

C. MU19, SEAL INJECTION FLOW CONTROL VLV fails closed.

D. Any RCP bearing temperature reaching 150°F.

Answer: C

Explanation/Justification: This is a KA match because the question tests knowledge of conditions that may exists during an RCP malfunction and based on those conditions, when to secure the RCPs during a loss of Seal Cooling or Seal Injection. CCW to all RCPs can be lost by the closure of a CCW Containment Isolation valve CC1411A. Stopping the RCP is required if Seal Injection is also lost. Seal Injection flow can be lost by Seal Injection valve closure or by loss of operating MU Pump. Based on the direction of DB-OP-02515 Step 4.3, the responses for A, B, and D have not reached the threshold to require immediate shutdown of the RCP.

A. Incorrect – Plausible because on the loss of Seal Return Flow by closure of MU38, AOP DB-OP-02515, RCP Malfunctions does require shutdown of the RCP, but 30 minutes is allowed by the procedure to resolve the valve closure. This response is incorrect because immediate tripping of the Reactor Coolant Pumps is not required for conditions provided.

B. Incorrect – Plausible because on the loss of CCW, RCP Stator Temperatures would rise. CCW provide cooling for the RCP Motor. High temperatures can cause motor failure; however, the temperature in this response is below the temperature that requires immediate tripping of the Reactor Coolant pumps and therefore incorrect.

C. CORRECT – DB-OP-02515, RCP Malfunction provides that during a loss of CCW to all RCPs, if Seal Injection flow is lost, immediate shutdown or the RCP is directed. This action is taken to limit hot RCS flow to RCP Seal Package that would occur as the aux impeller circulates hot RCS inventory up to the seal package. Stopping the RCP stops the aux impeller. This is the only response that when coupled with the loss of Component Cooling Water requires immediate tripping of the RCP.

D. Incorrect - Plausible because on the loss of CCW, RCP Bearing Temperatures would rise. CCW provide cooling for the RCP Lube Oil Coolers. High temperatures can cause bearing failure; however, the temperature provided by this response is below the temperature that requires immediate tripping of the Reactor Coolant pumps.

Sys #	System	Category	KA Statement
015	RCP Malfunctions	Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow):	When to secure RCPs on loss of cooling or seal injection.

K/A# AA2.10 **K/A Importance** 3.7 **Exam Level:** RO

References provided to Candidate; None

Technical References: DB-OP-02515 R18, Step 2.4 Section 4.3, Loss of CCW and Section 4.4, Loss of Both CCW and Seal Injection to RCPs

Question Source: DB Exam Bank Q38293

Level Of Difficulty: (1-5) 3

Question Cognitive Level: LO

10 CFR Part 55 Content: (CFR 43.5 / 45.13)

Objective: OPS-GOP-115-05K

Abbreviations: MU = Makeup

Tier being Tested: ES401 Tier 1/Group 1 (RO)

3. Initial plant conditions:
    - Mode 5 with a Reactor Coolant System (RCS) drain in progress
    - Decay Heat Removal (DHR) Loop 2 is in service
    - A steam generator primary side upper manway has been removed
    - RCS Temperature is 120°F and stable
    - Pressurizer level is 200 inches and lowering due to the RCS drain in progress

    The following event occurs:
    - DHR Pump 2 stops and will not restart
    - DHR Pump 1 does not start

    Which of the following actions is required in response to this change in plant conditions?

A. Evacuate Containment to minimize airborne exposure.

B. Start the MDFP and open an AVV to establish natural circ cooling.

C. Start a RCP to establish forced flow.

D. Open the PORV to establish a bleed flowpath.

Answer: A

Explanation/Justification: This question is a KA match. A loss of Decay Heat Removal occurs from a Mode 5 partial drained RCS condition. The question asks about actions following the loss of Decay Heat Removal (RHR) during Shutdown operations and the understanding of the mitigation strategies.

A. CORRECT - Containment is evacuated in order to limit radiological exposure and protect personnel from a hostile environment (steam) due to boiling in the core or from establishing feed and bleed cooling. This action is directed by DB-OP-02527, Loss of Decay Heat Removal step 4.1.3 and the reason described in Attachment 18, Background Document. This response is the only one that provides the procedurally directed action and is therefore the only correct response.

B. Incorrect - Plausible because on a loss of DHR, one method to provide core cooling is to restore SG Heat Transfer. This method could be used in Mode 5 if the RCS was intact. For these plant conditions, this response is incorrect because SG heat transfer is not possible with the RCS breeched by the open SG Primary Side Manway.

C. Incorrect - Plausible because under normal plant conditions, forced circulation is preferred over natural circulation. For these plant conditions, RCPs would have been shut down during RCS cooldown. Since a primary manway is removed, starting an RCS would result in a loss of inventory from the RCS and is not desired and is not directed by the controlling procedure for the plant conditions provided. As a result, this response is incorrect.

D. Incorrect - Plausible because with an upper SG Primary Manway removed, SG heat transfer will not be possible. Normally, when SG heat transfer is not available, feed and bleed cooling is established and opening the PORV would be directed. In this condition, a primary manway is far larger opening than that the PORV. This response is incorrect because opening the PORV is not required or procedurally directed for the plant conditions provided to establish a bleed flowpath.

Sys #	System	Category	KA Statement
EPE 009	Small Break LOCA	Generic	Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

K/A# G2.4.9 **K/A Importance** 3.8 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02527 R20, Loss of Decay Heat Removal step 4.1.3.

Question Source: Bank DB Requal Exam Bank Q37135 **Level Of Difficulty: (1-5)** 3.5

Question Cognitive Level: HI **10 CFR Part 55 Content:** 41.5 / 43.5 / 45.3 / 45.13

Objective: OPS-GOP-127-01K

Abbreviations: MDFP = Motor Drive Feedwater Pump, AVV = Atmospheric Vent Valve, PORV = Power Operated Relief Valve
Tier being Tested: ES401 Tier 1/Group 1 (RO)

4. A Reactor Trip has occurred from 100% power.

The following conditions are noted:
- SFAS Actuation - Levels 1-3
- Subcooling Margin indicates 0°F
- Incore Thermocouple temperatures are lowering and alternating between saturated and superheated conditions (approximately 10°F superheated)
- RCS Temperature vs. Pressure plot is trending parallel to the Saturation Curve

All actions for DB-OP-02000, RPS, SFAS, SFRCS Trip and SG Tube Rupture for these plant conditions have been completed

Which of the following describes the primary mode of RCS cooling for these conditions?

A. Forced Circulation Cooling

B. Single Phase Natural Circulation Cooling

C. Conduction Cooling

D. Boiler Condenser Cooling

Answer: D

Explanation/Justification: This question is a KA match because if tests knowledge of the operational implications of reflux or boiler condenser cooling. At Davis-Besse, the term Boiler Condenser is used vice Reflux. This knowledge is required to determine if the appropriate EOP Section to mitigate the event is Loss of Subcooling Margin or Inadequate Core Cooling. During Boiler Condenser Cooling, Incore thermocouples can trend slightly superheated, but that is not an indication that core cooling is inadequate. Direction to address the situation presented is provided by DB-OP-02000, Section 5, Loss of Subcooling Margin.

The following discussion of Boiler Condenser Cooling is provided by the Technical Bases Document Volume 3 (this is the generic EOP Bases for all B&W plants) "Boiler condenser cooling occurs when RC is boiled in the reactor core forming steam (removing core heat) which then flows through the hot leg piping to the SG where it condenses in the SG tubes. The condensed water then returns to the core by the cold leg piping. For the condensed water to flow back into the reactor core, the RC water level in the SG must be above the elevation of the RCP internal lip. This will provide the driving force to allow the water in the cold leg pipe to flow up and over the RCP discharge into the reactor vessel."

A. Incorrect - Plausible because the question does not provide the status of the RCPs. Forced circulation is the primary means of heat transfer following a reactor trip. Candidate should understand the EOP actions have been completed must know that RCPs are shutdown on a loss of SCM (SCM is less than 20°F subcooled), therefore Forced Circulation is incorrect.

B. Incorrect - Plausible because Single Phase Natural Circulation Cooling is the primary means of heat transfer following a reactor trip when RCPs are not available. Candidate should understand the EOP actions have been completed and must know that RCPs are shutdown on a loss of SCM (SCM is less than 20°F subcooled), therefore single-phase Natural Circulation is incorrect.

C. Incorrect - Plausible because conduction heat transfer from the RCS to Containment is occurring because the temperature of the RCS is higher than the temperature of Containment. Reflective insulation is installed on RCS piping to limit this heat transfer. While conductive heat transfer is occurring, it is not the primary mode of RCS cooling, therefore Conductive Cooling is incorrect.

D. CORRECT – With the RCS Saturated, Boiler condenser cooling occurs when RC is boiled in the reactor core forming steam (removing core heat) which then flows through the hot leg piping to the SG where it condenses in the SG tubes. The condensed water then returns to the core by the cold leg piping. For the conditions provided, boiling in the core is occurring (Subcooling Margin indicates 0°F). Lowering incore temperatures indicate cooling is occurring. For these plant conditions, that cooling would be boiler condenser cooling, therefore this response is the only correct response.

Sys #	System	Category	KA Statement
EPE 011	Large Break LOCA	Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA:	Natural circulation and cooling and reflux boiling

Davis-Besse 1LOT22 NRC Written Exam

K/A#	EK1.01	**K/A Importance**	4.1	**Exam Level:**	RO
References provided to Candidate	None			**Technical References:**	DB-OP-02000 Bases and Deviation Document R22, Step 5.6. & 9.0 B&W Technical Bases Document Volume 3 R12, Page III.B.12 (page 168 of 862 on pdf viewer)
Question Source:	Bank ANO 2014 NRC Exam Q4			**Level Of Difficulty: (1-5)**	3
Question Cognitive Level:	HI			**10 CFR Part 55 Content:**	CFR 41.8 / 41.10 / 45.3
Objective:	OPS-GOP-300-09K				

Abbreviations: SFAS = Safety Features Actuation System

Tier being Tested: ES401 Tier 1/Group 1 (RO)

5. Following the loss of **BOTH** Makeup Pumps from full power operations, as directed by DB-OP-02512, Makeup and Purification System Malfunctions, why is Reactor Coolant System (RCS) pressure reduced to 1700 to 1800 psig after the Reactor is tripped?

Reducing RCS Pressure will_______________.

A. reduce Reactor Coolant Pump seal leak off, preserving RCS Inventory

B. allow the Reactor Protective System to be placed in Shutdown Bypass

C. allow the Safety Features Low RCS Pressure Trip to be blocked

D. allow the High Pressure Injection system to restore RCS Inventory

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Answer: D

Explanation/Justification: This Question is a KA match. Makeup to the RCS is lost when both Makeup Pumps are lost. At normal operating RCS pressure, High Pressure Injection is unable to add inventory to the RCS. When piggybacked, HPI discharge Pressure is approximately 1800 psig. These actions are directed by AOP DB-OP-02512, Loss of Makeup, step 4.1.11 RNO and Attachment 6, RCS Pressure Control after a Reactor Trip. The direction is also provided by EOP DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture, Attachment 1, Primary Inventory Control Step 4.d RNO.

A. Incorrect – Plausible because this would reduce seal leakoff, RCS Inventory is preserved by isolating Letdown for this event. This response is incorrect because the RCS Pressure reduction directed by the controlling procedures allows HPI flow to recover RCS Level and is not performed to reduce RCP seal leakage as directed by DB-OP-02512.

B. Incorrect – Plausible because in a normal shutdown, Shutdown Bypass Operation could be established at this RCS Pressure range. (Set point is 1820 psig) This response is incorrect because establishing Shutdown Bypass operation for RPS is not directed by the controlling procedures.

C. Incorrect – Plausible because in a normal shutdown, RCS Pressure is reduced to slow the transition when blocking the SFAS Low RCS Pressure Trip at 1670 psig prior to SFAS Actuation at 1600 psig. This response is incorrect because the pressure band established is above the RCS pressure that would allow the SFAS LOW RCS Pressure trip to be blocked.

D. CORRECT – The ability to add inventory to the RCS is established by starting High Pressure Injection in piggyback mode which will then provide approximately 1800 psig discharge pressure allowing flow to the RCS. This is the only correct response for the plant conditions provided as directed by the controlling procedures.

Sys #	System	Category	KA Statement
000022	Loss of Reactor Coolant Makeup	Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup:	Actions contained in SOPs and EOPs for RCPs, loss of makeup, loss of charging, and abnormal charging

K/A# AK3.02 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02512 R19, Makeup and Purification System Malfunctions Step 4.1.11 RNO, Attachment 6, and Background Information Attachment 9.

Question Source: Bank NRC Exam DB 2013 Q5 **Level Of Difficulty: (1-5)** 2

Question Cognitive Level: Low - Fundamental **10 CFR Part 55 Content:** (CFR 41.5, 41.10 / 45.6 / 45.13)

Objective: OPS-GOP-300-07K

Tier being Tested: ES401 Tier 1/Group 1 (RO)

6. Initial plant conditions:
   - The Reactor is operating at 100% power
   - Component Cooling Water (CCW) Pump 1 is in STBY
   - Component Cooling Water Pump 2 is RUNNING

   Current plant conditions:
   - The Reactor AND all RCPs have been manually tripped in accordance with DB-OP-02523, Component Cooling Water System Malfunctions
   - DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture immediate actions have been completed
   - Component Cooling Surge Tank Level Side 1 is 33" and steady
   - Component Cooling Surge Tank Level Side 2 is 25" and lowering

   In response to the current plant conditions, the operator is directed to

   (1) ________ CCW Pump 1

   AND

   (2) ________ CCW Pump 2.

A. (1) start
   (2) continue to run

B. (1) start
   (2) stop and lockout

C. (1) lockout
   (2) continue to run

D. (1) lockout
   (2) stop and lockout

Answer: B

Explanation/Justification: This question is a KA match because is tests the ability to operate the CCW System during a loss of CCW inventory.

A leak in the essential portions of the CCW System will cause CCW Surge Tank level to lower until the top of the separator between essential train 1 and 2 is reached. At the point, only the affected train level will continue to lower. This will identify the affected train. CCW flow to Non-Essential Loads will stop when non-essential headers isolate on low CCW Surge Tank Level as designed. The affected essential train would be lost when the affected CCW train inventory is lost. Low Flow in Train 2 would cause an automatic start of CCW train 1, but operator action is directed to start the standby CCW Pump for these conditions rather than rely on the auto start.

A. Incorrect
   (1) Correct: The STBY pump will be started IAW DB-OP-02523 Attachment 2.
   (2) This portion of the response is incorrect: The running pump will be stopped and placed in Lockout IAW DB-OP-02523. Plausible because there are CCW essential loads that are or could be in service that need cooling, however shutdown of these loads will be directed individually directed by DB-OP-026523.

B. CORRECT

(1) Correct: A leak in the essential portions of the CCW System will cause CCW Surge Tank level to lower until the top of the separator between essential train 1 and 2 is reached. At the point, only the affected train level will continue to lower. This will identify the affected train. CCW flow to Non-Essential Loads will stop when non-essential headers automatically isolate as designed. The affected essential train would be lost when the affected CCW train inventory is lost.

(2) Correct because this is the operator action is directed by DB-OP-02523, Attachment 2 which is the controlling procedure for this event.

C. Incorrect –

(1) Incorrect: The STBY pump will be started IAW DB-OP-02523 Attachment 2. Plausible since tripping the reactor and stopping the RCPs are the required actions for a loss of both trains of CCW. (DB-OP-02523 Step 4.3.1 RNO) The applicant may incorrectly determine to lockout the STBY pump so that it does not auto start on the low flow interlock.

(2) Incorrect: The Running pump will be stopped and placed in Lockout. Plausible to keep the affected pump running until the leak location has been determined or until all affected loads are shutdown/removed from service.

D. Incorrect –

(1) Incorrect: The STBY pump will be started IAW DB-OP-02523 Attachment 2. Plausible since tripping the reactor and stopping the RCPs are the required actions for a loss of both trains of CCW. (DB-OP-02523 Step 4.3.1 RNO) The applicant may incorrectly determine to lockout the STBY pump so that it does not auto start on the low flow interlock.

(2) This portion of the response is correct: Since the leak is on the essential header with a Running CCW pump, the Running pump will be stopped and placed in Lockout.

Sys #	**System**	**Category**	**KA Statement**
00026	Loss of CCW	Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water:	The CCWS surge tank, including level control and level alarms, and radiation alarm

K/A# AA1.05 **K/A Importance** 3.1 **Exam Level: RO**

References provided to Candidate None **Technical References:** DB-OP-02523 R13, CCW Malfunctions step 4.1.1 and Attachment 2, Shutdown of a Leaking Essential Header

Question Source: DB Exam Bank ORQ-36766 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** 41.7 / 45.5 / 45.6

Objective: OPS-GOP-123-02K

Abbreviations: CCW = Component Cooling Water

Tier being Tested: ES401 Tier 1/Group 1 (RO)

7. The plant is operating at 100% power.

The selected RCS Pressure Instrument from the Reactor Protective System to Non-Nuclear Instrument System **INSTANTANEOUSLY** fails **HIGH**.

Which of the following describes how the plant will respond to this failure?

(1) The Pressurizer PORV will ________________.

AND

(2) The Pressurizer Spray Valve will _____________.

A. (1) remain closed
(2) remain closed

B. (1) open
(2) open

C. (1) open
(2) remain closed

D. (1) remain closed
(2) open

Answer: B

Explanation/Justification: This question is a match for the KA because it tests knowledges of the relationship between of the impact of a Pressurizer Pressure instrument signal failure impact on the controllers for the Pressurizer PORV, and the Pressurizer Spray Valve. The system knowledge required to correctly answer the question in Reactor Protective System Instrumentation, Non-Nuclear Instrumentation, and the controllers for the Pressurizer Operating System.

A. Incorrect
(1) This portion of the response is incorrect because the input is not SASS protected which means the fault would become the input pressure for controller for the Spray Valve and the PORV. Plausible if the candidate believes the RCS signal is SASS protected like most other NNI signals. Instantaneous failures would normally cause a SASS transfer for SASS protected instrument inputs resulting in no change to the input for the PORV and Spray Valve.
(2) This portion of the response is incorrect because the position provided are incorrect for events provided. Plausible if the candidate believes the RCS signal is SASS protected like most other NNI signals. Instantaneous failures would normally cause a SASS transfer for SASS protected instrument inputs resulting in no change to the input for the PORV and Spray Valve.

B. CORRECT
(1) Correct - The selected RPS Pressure signal is used to control the PORV. A high failure will cause the PORV to open.
(2) Correct - This response provides the correct PZR Spray Valve positions for the plant conditions provided. A high failure will cause the PZR Spray Valve to open.

C. Incorrect
(1) Correct because the selected instrument provides the pressure input for the PORV. The valves open on high pressure. With the failure presented, the valves would open.
(2) This portion of the response is incorrect because the position for the Spray valve provided is incorrect for events provided. Plausible if the candidate believes the RCS signal for the Spray Valve is SASS protected like most other NNI signals. Instantaneous failures would normally cause a SASS transfer for SASS protected instrument inputs resulting in no change to the input for the Spray Valve.

D. Incorrect

(1) This portion of the response is incorrect because the selected instrument provides the pressure input for the PRZ Spray Valve. - Plausible if the candidate thought that the safety grade Reactor Protective System RCS Pressure signal is used to control only the Pressurizer Spray Valve. With this assumption, the PORV would not be affected .

(2) Correct – This portion of the response provides the correct PZR Spray Valve positions for the plant conditions provided.

Sys #	System	Category	KA Statement
027	Pressurizer Pressure Control System (PZR PCS) Malfunction	Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following:	Controllers and positioners

K/A#	AK2.03	**K/A Importance**	2.6	**Exam Level**	RO
References provided to Candidate	None			**Technical References:**	DB-OP-02513 R13, Pressurizer System Abnormal Operation Att. 2 page 54

Question Source:	Bank DB 2013 NRC exam Q8	**Level Of Difficulty: (1-5)**	3
Question Cognitive Level:	High - Comprehension	**10 CFR Part 55 Content:**	(CFR 41.7 / 45.7)

Objective: OPS-GOP-113-05K

Abbreviations: PORV = Power Operated Relief Valve – Note: DB only uses PORVs on the Pressurizer

Tier being Tested: ES401 Tier 1/Group 1 (RO)

8. A Turbine Trip occurs at 100% power.

The following sequence of events occurs:

1. ARTS fails to trip the Reactor
2. Reactor Protective System actuates on High RCS Pressure
3. All Control Rods fail to insert
4. The Reactor Trip Test key is inserted AND rotated clockwise
5. All Control Rods insert
6. Power is observed lowering on the Intermediate Range
7. The EHC-Emergency Trip pushbuttons are depressed
8. Turbine Stop Valve 1 indicates open
9. Turbine Control Valve 1 indicates open

Which ONE of the following actions should be taken NEXT per DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture?

A. Open Generator Output Breakers 34560 and 34561.

B. Stop and lockout both Electro Hydraulic Control (EHC) Pumps.

C. Trip the Primary and Emergency Trip Relays using the local Emergency Trip Pushbuttons.

D. Initiate AFW flow and Isolation of both SGs.

Answer: D

Explanation/Justification: This question is a match for the KA as modified to match Davis-Besse equipment. DB does not have Main Turbine trips switches, but it does have Main Turbine Stop Valve indications as indication lights on a control panel and as it has Main Turbine Stop Valve and Control Valve position indication on the Digital EHC panel. These indications are the equivalent to the Main turbine trip switch position indication in the KA. For this question, knowledge of the indications available and the actions required to respond to an ATWS event where Main Turbine Stop Valve 1 and Control Valve 1 fail to trip close is required to select the correct response.

A. Incorrect because opening the Generator Output Breakers is not the next action directed by DB-OP-02000 for the plant conditions presented – Plausible because following a turbine trip, the generator output breakers are manually opened if they fail to automatically open. If the Turbine has lost motive force (steam flow), continued operations with an output breaker closed would motorize the Main Generator and cause possible generator damage. The candidate may incorrectly select this response based on knowledge that 34560 and 34561 are opened after a turbine trip.

B. Incorrect because stopping and locking out both EHC Pumps is not the next action directed by DB-OP-02000 for the plant conditions presented – Plausible because stopping and locking out the EHC pumps would result in the loss of fluid power holding the Main Turbine stops and control valves open. As a result, the valves would normally close when the EHC accumulator losses pressure causing the valve springs to close the valves. The candidate may incorrectly select this response based on knowledge that stopping the EHC pumps would eventually close the failed open valves, but this would take significantly longer that actuating and isolating SFRCS which closes the MSIVs.

C. Incorrect because tripping the Primary and Emergency Trips relays using the local Emergency Trip Pushbutton is not the next action directed by DB-OP-02000 for the plant conditions presented – Plausible because the Main Turbine is normally tripped by de-energizing the Primary and Emergency Trip Relays. This action results in the loss of fluid power holding the Main Turbine stops and control valves open. The candidate may incorrectly select this response based on knowledge that deenergizing the EHC System Primary and Emergency Trip Relays could eventually close the failed open valves if not mechanically bound.

D. CORRECT – This is the only correct response because this is the action directed by DB-OP-02000 for the plant conditions presented. The main turbine is the largest steam load and can cause rapid overcooling of the RCS if all Main Turbine Stop and Control valves do not close following reactor shutdown. DB-OP-02000 Step 3.5 determines if the expected response (Stop OR Control Valves closed) was obtained or not. If the expected response is not obtained, SFRCS will be manually initiated (initiate and Isolate) to close both Main Steam Isolation Valves which are upstream of the Turbine Stop and Control Valves. Closing both Main Steam Isolation Valves will secure steam flow to the main turbine.

Sys #	**System**	**Category**	**KA Statement**
026	ATWS	Ability to determine or interpret the following as they apply to an ATWS:	Main turbine trip switch position indication

K/A# EA2.06 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate: None **Technical References:** DB-OP-02000, Immediate Actions, Bases and Deviation Document for DB-OP-02000 R22 Step 3.5.

Question Source: Modified Requal Exam Bank 37591 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** 43.5 / 45.13

Objective: OPS-GOP-300-05K

Abbreviations: ARTS = Anticipator Reactor Trip System AFW = Auxiliary Feedwater SG = Steam Generator SFRCS = Steam Feedwater Rupture Control System

Tier being Tested: ES401 Tier 1/Group 1 (RO)

9. The plant has been shut down from 100% power due to a SGTR on #2 Steam Generator.

Current Conditions:

- Reactor Coolant System (RCS) Pressure at 1000 psig
- RCS Temperature at 500°F
- All 4 Reactor Coolant Pumps in service
- Both Main Steam Isolation Valves (MSIVs) are open
- Steam Generator 1 level at 40 inches and steady
- Steam Generator 2 level at 52 inches and slowly rising
- Main Feedwater Pump (MFP) 1 in service

Which of the following items must be completed PRIOR to performing the valve line-up to isolate the faulted Steam Generator as directed by DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture?

A. Defeat the MSIV to TBV interlock by pulling a fuse in the designated ICS Cabinet.

B. Place the Motor Driven Feed Pump in service and shutdown MFP 1.

C. Override the SFRCS HI LVL and LO PRESS TRIPS on the tube ruptured SG.

D. Notify the CSRO that Aux Steam will be lost if NOT transferred to the Aux Boiler.

Answer: A

Explanation/Justification: This question is a KA Match because it tests the candidate's knowledge of what must be completed prior to positioning a valve that is required to be closed to isolate a SG that has a tube rupture. For B&W plants and Davis Besse, during a SG Tube Rupture, as long as SG level can be maintained, steaming of the faulted SG continues until the isolation point of 1000 psig RCS Pressure and 500 F RCS Temperature is reached. At that point, the faulted SG isolated from the Main Steam and Feedwater Systems .

A. CORRECT - The MSIV to TBV interlock must be defeated prior to closing MS100, MAIN STEAM LINE 2 ISOLATION. The interlock will transfer control over to the AVV if the MSIV is closed resulting in a release to the atmosphere. This is the only correct answer because it is the action directed by the procedure prior to isolating the faulted SG.

B. Incorrect because the controlling procedure does not direct placing the MDFP in service prior to isolating the faulted SG– Plausible because the MDFP will eventually be placed in service, however it is not required to be placed in service prior to isolating the faulted SG because MFP 1 is in service.

C. Incorrect because the controlling procedure does not direct over riding the SFRCS Trips – Plausible since it is expected that the Faulted SG will fill up due to the leaking SG tubes. SFRCS is not required to be overridden prior to isolating the faulted SG.

D. Incorrect because the Auxiliary Steam System is supplied steam from SG1, not SG 2 as provided in the stem of the Question. Plausible since this would be required to be completed if the leak was on SG 1.

Sys #	System	Category	KA Statement
038	SG Tube Rupture	Generic	Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

K/A# G2.1.29 **K/A Importance** 4.1 **Exam Level:** RO

References provided to Candidate None **Technical References:** DB-OP-02000 R32 Attachment 17

Question Source: NEW **Level Of Difficulty: (1-5)** 3.5

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.10 / 45.1 / 45.12)

Objective: OPS-GOP-300-14K

Abbreviations: TBV = Turbine Bypass Valves CSRO = Command Senior Reactor Operator, MSIV = Main Steam Isolation Valve
Tier being Tested: ES401 Tier 1/Group 1 (RO)

10. The plant is operating at 100% power.

Which of the following Main Steam Line Ruptures will add the MOST positive reactivity to the core?

Assume all systems function as designed.

(1) _________ of Cycle

AND

(2) Main Steam Line double ended rupture _________ of the Main Steam Isolation Valve.

A. (1) End
(2) upstream

B. (1) End
(2) downstream

C. (1) Beginning
(2) upstream

D. (1) Beginning
(2) downstream

Answer: A

Explanation/Justification: This question is a match for the KA because it requires not only knowledge of the operational impacts caused by a Steam Line Rupture with regards to which failure location would cause the most cooldown but also which time in life would result in the largest positive reactivity impact.

A. CORRECT - The Negative Moderator Temperature and Doppler Coefficients are largest at EOC. This large negative value produces the largest positive reactivity insertion due to the RCS cooldown following the MS Line Rupture. This is the only completely correct response of the 4 responses provided.

B. Incorrect because the break location would not provide the largest possible cooldown - Plausible because RCS temperature would lower causing the addition of positive reactivity. It appears that this break location could be fed by both steam generators causing a larger cooldown, however this break will cause both Main Steam Isolation valves to close, ending the Steam release. As a result, the inventory available to cause cooling is less for this break location than the upstream location which after MSIV's close, cooldown would only occur from a single SG.

C. Incorrect because the Negative Moderator Temperature coefficient is largest at EOC, not Beginning of cycle - Plausible if student doesn't know EOC has the largest negative coefficients

D. Incorrect because the Negative Moderator Temperature coefficient is largest at EOC, not Beginning of cycle and the break location does not provide the largest cooldown- Plausible if student doesn't know EOC has the largest negative coefficients and the break location would be isolated by SFRCS, limiting the magnitude of the overcooling event,

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Sys #	System	Category	KA Statement
040	Excessive Heat Transfer	Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture:	Reactivity effects of cooldown

K/A# AK1.05 **K/A Importance** 4.1 **Exam Level:** RO

References provided to Candidate None **Technical References:** Update Safety Analysis Report R33 Section 15.4.4, Main Steam Line Break Analysis

Question Source: Exam **NRC Exam DB 2011 Q10** **Level Of Difficulty: (1-5)** 3.5

Question Cognitive Level: HI

10 CFR Part 55 Content: (CFR 41.8 / 41.10 / 45.3)

Objective: OPS-GOP-302-02K and OPS-GOP-302-04K

Tier being Tested: ES401 Tier 1/Group 1 (RO)

11. The Reactor was at 100% power.

A Total Loss of all Feedwater Event has occurred.

Actions are in progress in accordance with Section 6, Lack of Heat Transfer, of DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture.

Current Plant Conditions:
- RCS Thot is 580°F and rising
- Two High Pressure Injection Pumps have been started piggybacked from LPI
- Two Makeup Pumps are in service
- Feedwater has been restored from the Emergency Feedwater Pump

Which of the following actions is directed by DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture while attempting to restore heat transfer?

A. Limit Feedwater Flow to 300 gpm
if restoring Steam Generator (SG) level to a single dry SG.

B. Immediately initiate MU/HPI Cooling
if RCS pressure rises above HPI discharge Pressure of 1800 psig.

C. Immediately initiate MU/HPI Cooling
if RC2A PORV opens at RCS pressure setpoint of 2450 psig.

D. Open RC2A PORV until minimum SCM is met
if RCS pressure reaches the PORV setpoint of 2450 psig.

Answer: D

Explanation/Justification: This question is a KA match because it tests candidate knowledge of how the PORV should be operated in response to a loss of all feedwater. The connection to loss of all Feedwater is required because if only MFW is lost, AFW would automatically start negating the need to operate the PORV. Distractors include actions that may be required (establishing Feed and Bleed Cooling) depending on RCS temperature and on the status of MU and HPI Pumps.

A. Incorrect, IAW Specific Rule 4 Step 4.4, there are no Feedwater Flow restrictions when a Lack of Heat Transfer is in progress. Plausible because Feedwater flow is restricted to 300 gpm when 1 RCP is in service in the loop with a dry SG if a Lack of Heat Transfer OR Inadequate Core Cooling does not exist and SG Tube to Shell differential temperature limits are met.

B. Incorrect because this is not the action directed by DB-OP-02000 for the plant conditions provided – Plausible because if Feedwater cannot be restored, the PORV will be opened. RCS Temperature and Pressure will rise due to the lack of heat transfer. Once RCS Pressure rises about approximately 1800 psig, the High Pressure Injection System will not be able to inject into the RCS due to the pumps maximum discharge pressure even piggybacked. Attempting to restore feedwater is permitted as long as 2 MU Pumps are in service and RCS Thot is less than 600°F.

C. Incorrect because this is not the action directed by DB-OP-02000 for the plant conditions provided – Plausible because if Feedwater cannot be restored, the PORV will be opened. It is plausible if the PORV begins to cycle at 2450 psig in response to rising RCS temperatures (lack of feedwater), establishing Feed and Bleed cooling which includes manually opening the PORV would be complete. This is not required because attempting to restore feedwater is permitted as long as 2 MU Pumps are in service and RCS Thot is less than 600°F.

D. CORRECT – DB-OP-02000, Section 6, Lack of Heat Transfer provided direction for operating the PORV in the event all Feedwater is lost including direction for preventing PORV cycling if RCS pressure rises to the PORV auto opening setpoint of 2450 psig. Repeated cycling of the PORV can lead to valve failure. Allowing RCS pressure to cycle in a larger band will reduce the number of cycles while maintaining RCS Pressure in an acceptable region. For the conditions provided, DB-OP-02000 does not required initiation of Feed and Bleed Cooling until RCS Thot reaches 600°F. This is the only correct response because this is the action directed by DB-OP-02000 for the plant conditions provided.

Sys #	System	Category	KA Statement
054	Loss of MFW	Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW):	HPI/PORV cycling upon total feedwater loss

K/A# AK3.05 **K/A Importance** 4.6 **Exam Level:** RO

References provided to Candidate: None

Technical References: DB-OP-02000 R32, RPS, SFAS, SFRCS Trip or SG Tube Rupture Section 6, Lack of Heat Transfer Step 6.10
Caution 6.6, Step 6.6

Question Source: TMI 2010 NRC Exam Q45 Modified for DB

Level Of Difficulty: (1-5) 4

Question Cognitive Level: HI

10 CFR Part 55 Content: (CFR 41.5,41.10 / 45.6 / 45.13)

Objective: OPS-GOP-305-03K

Abbreviations: BOP = Balance of Plant Reactor Operator HPI = High Pressure Injection

Tier being Tested: ES401 Tier 1/Group 1 (RO)

12. Plant Conditions:
    - The reactor was operating at 100% power
    - A Station Blackout event occurs
    - Load Shedding of the Direct Current (DC) Buses has been directed in accordance with DB-OP-02704, Extended Loss of AC Power DC Load Management

    Following completion of Attachment 1, Selective Battery Load Shedding, which of the following best describes SG Level Control?

    SG level will be maintained ___________.

A. in automatic by AFW Level Control Valves

B. using the Emergency Feedwater System

C. in manual by controlling AFPT speed

D. using the Motor Driven Feed Pump

Answer: C

Explanation/Justification: This question is a KA match because it addresses operation and monitoring of the plant following reduction of battery loads following a station blackout.

The AFW Level Control Valves for Train 1 and 2 will fail open when DC loads are shed. In anticipation of the loss of the level control valves, DB-OP-02704 directs the level control valves to be fully opened, and SG level controlled by manually adjusting Auxiliary Feedwater Pump Turbine speed.

A. Incorrect because the level control valves are initially manually opened since they will fail open when they are de-energized during load shed and AFPT Speed is used to control SG Level.
Plausible because during a station blackout, prior to DC Bus Load shed, SG level will be maintained in automatic by the AFW Level Control Valves.

B. Incorrect because AFW will not be lost in this scenario. Therefore, direction to place the EFW Pump in service is not applicable.
Plausible because the EFW system is available during a station blackout.

C. CORRECT because with the normal level control valves initially manually opened and eventually failed open by procedure, AFPT speed will be used to control SG levels.

D. Incorrect because the MDFP will not have a source of power in this scenario.
Plausible because the Station Blackout Diesel Generator supplies power to the MDFP.

Sys #	System	Category	KA Statement
000055	Station Blackout	Ability to operate and monitor the following as they apply to a Station Blackout:	Reduction of loads on the battery

K/A# EA1.04 **K/A Importance** 3.5 **Exam Level:** RO

References provided to Candidate: None

Technical References: Bases and Deviation Document for Station Blackout and Flex Support Guidelines R0 Page 88 discussion on DB-OP-02704 R01, Load Management and DB-OP-02704 R01, DC Load Management Attachment 5.

Question Source: New

Level Of Difficulty: (1-5) 3.5

Question Cognitive Level: HI

10 CFR Part 55 Content: (CFR 41.7 / 45.5 / 45.6**)**

Objective: OPS-FLX-003

Abbreviations: AFPT = Auxiliary Feedwater Pump Turbine

Tier being Tested: ES401 Tier 1/Group 1 (RO)

13. The following plant conditions exist:
    - Plant startup in progress with reactor power at 1%.
    - A loss of offsite power has occurred.
    - All plant systems have responded properly.

    Which of the following describes the expected plant response to this event?

A. AFP 1 will maintain SG 2 at 55 inches, AFP 2 will not be running.

B. Each AFW Train will maintain its respective SG at 49 inches.

C. AFP 1 will maintain SG 1 at 49 inches, AFP 2 will not be running.

D. Each AFW Train will maintain the opposite SG at 55 inches.

Answer: B

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the expected SG Level response (expected SG Level) to a Loss of Off-Site Power. The low power condition raises the complexity because in this condition (at 1% power) the MDFP would be in service supplying Feedwater to both SGs when off-site power is lost.

A. Incorrect – Plausible because AFW Train 1 includes a design that features DC powered components not found in Train 2. The applicant could assume it is the only train to actuate when AC power is initially lost with DC power supplied by Station Batteries. This response is incorrect because AFP 2 would be running and controlling SG 2 level. Offsite power is not required for AFP 2 to operate and control SG 2 level.

B. CORRECT – SFRCS will actuate on the loss of all RCPs when the LOOP occurs, causing both AFW Pumps to start and control level their respective SG at 49". This is the only response that correctly describes the expected response for the plant conditions provided.

C. Incorrect – Plausible because AFW Train 1 includes a design that features DC powered components not found in Train 2. The applicant could assume it is the only train to actuate when AC power is initially lost with DC power supplied by Station Batteries. This response is incorrect because AFP 2 would be running and controlling SG 2 level. Offsite power is not required for AFP 2 to operate and control SG 2 level.

D. Incorrect – Plausible because each AFW train monitors both SG conditions such as level and is able to supply the cross-train SG under some conditions. For example, if a single SG isolated on low SG pressure, both AFW Trains will align to feed the good SG with the cross-train pump attempting the control level at 55 inches while the onside AFW train controls at 49 inches to prevent hunting. This response is incorrect because it does not describe the expected plant response to this event.

Sys #	System	Category	KA Statement
056	Loss of Off-Site Power	Ability to determine and interpret the following as they apply to the Loss of Offsite Power:	S/G level meter scale and pressure gauge

K/A# AA2.81 **K/A Importance** 3.7 **Exam Level:** Reactor Operator

References provided to Candidate None **Technical References:** DB-OP-02000 R32 Specific Rule 4. Bases and Deviation Document for DB-OP-02000 R22 for Specific Rule 4.

Question Source: DB Exam Bank 36854 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 43.5 / 45.13)

Objective: OPS-GOP-301-04K

Abbreviations: MDFP = Motor Driven Feedwater Pump AFP = Auxiliary Feedwater Pump

Tier being Tested: ES401 Tier 1/Group 1 (RO)

14. The normal **AND** alternate DC control power to B Bus **SOURCE** breakers is lost.

Which ONE of the following would be used to open Breaker HX11B if required?

HX11B ____________.

A. control switch (HIS 6208) in the Control Room

B. breaker cubicle control switch ONLY

C. breaker cubicle OPEN plunger

D. breaker cubicle Emergency Control Transfer Switch
AND
breaker cubicle control switch

Answer: C

Explanation/Justification: This question is a KA match because it tests the knowledge of local operator actions following the loss of all DC power to 13.8 kV B Bus Supply breakers. The B Bus supply breakers have a normal and an alternate DC control power to allow remote operation and local operation at the switchgear room using the local control switch. Transfer from normal to alternate supply requires local manual operator action in the High Voltage Switchgear Room. Loss of DC power to both the normal and the alternate power supply would prevent all remote (from Control Room) and local (Switchgear Room) electrical operation at the switchgear. The breaker can still be opened mechanically by a local operator using the open plunger. Operation of this plunger release the breaker mechanical latch allow springs to open the breaker.

A. Incorrect because the loss of control power would prevent remote operation from the Control Room – Plausible if the candidate does not recognize that the loss of DC power prevents remote operation of the breaker, this is the control switch that would be used to open the breaker from the control room.

B. Incorrect because the loss of control power would prevent operation using the breaker cubical control switch – Plausible because if the candidate does recognize that the loss of DC power prevents remote operation of the breaker from the Control Room but does not recognize that the local operation from the breaker cubical is also lost. This is the control switch that would be used to open the breaker from the High Voltage Switchgear Room Breaker HX11B cubicle if DC power was available.

C. CORRECT - Loss of DC power to both the normal and the alternate power supply would prevent all remote (from Control Room) and local electrical operation at the switchgear HX11B cubicle. The breaker can still be opened mechanically using the open plunger which releases the mechanical latch allow springs to open the breaker. This is the only correct response for the plant conditions presented.

D. Incorrect because the loss of control power would prevent operation using the breaker control switch even if the emergency control transfer switch was utilized – Plausible because if DC power to the B Bus 13.8 kV Source breakers was lost due to a Control Room circuit fault; the local control transfer switches can be used to isolate all Control Room Circuits from Breaker HX11B allowing the use of the local control switch at the breaker cubicle. In this condition, even if the local control transfer switch was used to isolate Control Room wiring, no DC power is available which would prevent use of cubicle local control switch.

Sys #	System	Category	KA Statement
058	DC Power	Loss of DC Power	Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

K/A# G2.4.35 **K/A Importance** 3.8 **Exam Level:** Reactor Operator

References provided to Candidate None **Technical References:** DB-OP-01000 R39, Step 4.1

Question Source: DB Exam Bank Q37789 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** 41.10 / 43.5 / 45.13

Objective: OPS-SYS-409-09K

Tier being Tested: ES401 Tier 1/Group 1 (RO)

15. The plant is in Mode 1 at 100% power.
- Service Water Returns aligned to the Cooling Tower.
- Service Water Pump 1 is supplying Primary Loads
- Service Water Pump 2 is supplying Secondary Loads

A Large Break Loss of Coolant Accident (LOCA) occurs.

All equipment responds as designed.

Which of the following Service Water system responses to an SFAS Actuation is specifically designed to ensure adequate cooling of systems required to remove decay heat following a LOCA?

A. SW 1395, SW HDR 2 TO TPCW HX Closes

B. SW 1399, SW HDR 1 TO TPCW HX Closes

C. CT 2955, TPCW HX SUPPLY FROM CIRC WATER Opens

D. SW 2930, SW RETURN ISOLATION – INTAKE FOREBAY Opens

Answer: A

Explanation/Justification: This question is a KA match because it tests the knowledge of alignments within the Service Water (SW) System following an SFAS Actuation that ensure that the required SW flow exists to components cooled by SW when SFAS is actuated. It should be noted that at Davis-Besse, the Service Water System normally supplies safety related loads as well as non-safety related loads. Normally 2 SW pumps are in service with one pump supplying Primary Loads (safety related) and the second pump generally supplying Secondary Loads (non-safety related) such as Turbine Plant Cooling Water. When SFAS Actuates, repositioning of multiple components actuated by SFAS is required to align the SW system to an accident response (LB LOCA) position. As a result, to match the KA, the question tests knowledge of Service Water alignment following an SFAS Actuation.

Misalignment of a component or failure to properly reposition would result in a flow diversion from cooling safety related loads following the SFAS Actuation. A flow diversion would prevent the required SW flow to remove decay heat post LOCA (which causes an SFAS actuation). For example, if Service water was aligned to both safety related loads and non-safety related loads following an SFAS Actuation, insufficient flow to provide cooling to the safety related loads would likely result.

A. CORRECT – In the alignment provided by the question stem, Service Water Pump 2 is aligned to Secondary or generally non-safety related loads like Turbine Plant Cooling Water. In this alignment, SW1395, SW HDR 2 to TPCW HX closes on an SFAS Actuation to ensure all available SW flow from Train 2 is available to provide cooling to safety related loads such as Component Cooling Water and Containment Air Coolers. These loads remove core decay heat following a LB LOCA.

B. Incorrect because for the plant conditions provided, SW1399, SW HDR 1 to TPCW HX is closed at the start of the event. As a result, repositioning of SW1399 is not required to ensure SW Flow from Train 1 is not being diverted to supply cooling to the Non-Safety Related Turbine Plant Cooling Water System. Plausible because SW1399 does get a close signal on an SFAS Actuation, but the valve is already in the closed position at the start of the event. To ensure only one supply to TPCW HX is in service at any given time, there is an interlock that prevents both SW1395 and SW1399 from being open at the same time. As a result, only one SW supply to TPCW is open at the beginning of the event ((SW1395).

C. Incorrect because CT2955 TPCW HX Supply from Circ Water does not reposition to remove decay heat following a LB LOCA event. Plausible because CT2955 does open to supply cooling to Turbine Plant Cooling Water, but TPCW is a non-safety related load which provide cooling for Turbine Plant Loads, not safety related loads required to remove decay heat. In addition, CT2955 is not positioned by SFAS. It opens on low pressure to the common heater that supplies Turbine Plant Cooling Water Heat Exchangers.

D. Incorrect because SFAS does not provide an open signal to SW2930, SW Return Isolation – Intake Forebay. Plausible because the SW System has 4 return flowpaths of which 2 are safety related (Intake Forebay and Intake Structure). The question stem provides that the SW Returns are aligned to the cooling tower Makeup up which is a non-safety related alignment. One of the safety related flowpaths must be open for a seismic event that interrupts norm flow from Lake Erie to the Service Water Intakes. A manual operator action is provided by the EOP DB-OP-02000 (step 10.22) following a LB LOCA to manually align SW returns to either the intake forebay or the intake structure. This is a manual operator actions not an automatic response of the SW System.

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Sys #	System	Category	KA Statement
062	Service Water	Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water:	The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS

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

References provided to Candidate None **Technical References:** DB-OP-02000 R32 step 10.22 and Table 2 SFAS Response

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** 41.4, 41.8 / 45.7

Objective: OPS-SYS-306-11K

Tier being Tested: ES401 Tier 1/Group 1 (RO)

16. Initial Plant Conditions:
    - The reactor is at 100% power
    - Station and Instrument Air are in their normal alignment with SAC 2 in service.

    Current Plant Conditions:
    - A loss of all Off-site power occurred
    - Annunciator 9-1-F INST AIR HDR PRESS LO is in alarm
    - No other operator actions have been completed

    When Bus D2 is reenergized IAW Attachment 6, Reenergization Of Buses D2, F7, and MCC F71, which of the following statements is correct concerning the Station and Instrument Air Compressors?

A. SAC 1 will Auto start.

B. The EIAC will Auto start.

C. SAC 2 will Auto start.

D. No Air Compressors will Auto start.

Answer: B

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the EIAC and SACs, which supply Instrument Air.
A loss of Instrument Air occurs during a loss of offsite power until power is restored to the EIAC. When power is restored, the EIAC will auto start to restore air pressure. SAC 1 and SAC 2 will require in field actions to restore auto start function.

A. Incorrect because SAC 1 is powered from Bus E4 which does not regain power in this scenario.
Plausible since SAC 1 and C1 are both train 1 components. Essential 4160 v bus C1 that supplies power to essential Train 1 loads will be energized in this scenario by the automatic start of EDG 1, however this will not restore power to SAC 1. As a result, SAC 1 will not auto start since power will not be available. Even if power was restored, in field actions would be required for SAC 1 as described in DB-OP-06251, Instrument and Station Air Operating Procedure Section 5.3, Resetting a trip of SAC 1 or SAC 2.

B. CORRECT The Emergency Instrument Air Compressor has provisions to be powered from the SBODG or Emergency Diesel Generators by restoring power to D2. Performance of DB-OP-02000 Attachment 7 will restore power to the EIAC. The setpoint for the low instrument air pressure alarm is 95 psig. The Auto start set point for the EIAC is 95 psig. The EIAC will auto start for these plant conditions once power is restored to the EIAC.

C. Incorrect –SAC 2 will not Auto start in this scenario, in field actions are required for SAC 2 to start. These actions are described in DB-OP-06551, Instrument and Station Air System Operating Procedure, Section 5.3, Resetting a Trip of SAC 1 or SAC 2.
Plausible since its power source is from D2 and power has been restored to D2.

D. Incorrect since the EIAC will Auto start in this scenario.
Plausible since all compressors will be off until Bus D2 is reenergized. Manual actions required to energize Bus D2 are completed by Attachment 6, Reenergization of Buses D2, F7, and MCC F71 as noted in the plant conditions provided by the stem of the question.

Sys #	System	Category	KA Statement
065	Instrument Air	Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air:	Emergency air compressor

K/A# AA1.04 **K/A Importance** 3.5 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02528 R27, Instrument Air System Malfunctions, Attachment 24 Background Discussion DB-OP-06251, Instrument and Station Air Operating Procedure (R51) Section 5.3

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO

10 CFR Part 55 Content: 41.7 / 45.5 / 45.6

Objective: OPS-GOP-128-05K

Tier being Tested: ES401 Tier 1/Group 1 (RO)

17. Plant Conditions:
    - The Reactor is operating at 100% power
    - The Wadsworth Control Center contacts the Control Room to report weather conditions are affecting Grid stability throughout Ohio and Western Pennsylvania.
    - Switchyard voltage is reported to be 336 kV and lowering slowly

    Which of the following actions should be taken to maintain Grid stability, and why?

A. Place the Main Turbine in Manual and Raise Load to increase station MVAR output

B. Raise Main Generator Voltage in Automatic to increase station MVAR output

C. Transfer A and B 13.8 kV Busses to their respective Startup Transformers to increase Station MW output

D. Direct Equipment Operators to start and load both EDGs in parallel with off-site power to increase Station MW output

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Answer: B

Explanation/Justification: This question is a KA match. The KA asks about Generator Voltage and Grid Disturbances related to the definition of several electrical terms.
The question provides a grid voltage that is lower than normal. VARS are needed to support the transmission of electrical power over long distances and maintain adequate Grid voltage. VARS are produced by raising Terminal Voltage of the Main Generator. Since Power is initially at 100%, raising Power with the Main Turbine is not an acceptable alternative. Parallel operation of two EDGs is not allowed and transferring the 13.8 kV busses to off-site power would increase load on the switchyard.

A. Incorrect – Plausible because this action would result in higher switchyard voltage if the load on the Grid remained constant, but is incorrect because power would increase above 100%, which is not permitted.
B. CORRECT– Raising Generator Output Voltage would increase MVAR output which would provide voltage support for the grid.
C. Incorrect – Plausible because transferring house loads to the startup transformers would allow the main transformer to supply an additional 40 MW to the switchyard. Incorrect because the additional output of our main transformer would also be supplying the startup transformers, offering no additional support for the Grid
D. Incorrect – Plausible because the EDGs could supply an additional 5 MW to the switchyard. Incorrect because parallel operation of our EDGs is not allowed.

Sys #	System	Category	KA Statement
077	Generator Voltage and Grid Disturbance	Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances:	Definition of terms: volts, watts, amps, VARs, power factor

K/A# AA1.01 **K/A Importance** 3.3 **Exam Level** RO
References provided to Candidate: None **Technical References:** DB-OP-02546, R07 step 4.4.
Question Source: NEW **Level Of Difficulty: (1-5)** 3.5
Question Cognitive Level: HI **10 CFR Part 55 Content:** 41.4, 41.5, 41.7, 41.10 / 45.8

Objective: OPS-GOP-146-02K
Abbreviations: MVAR = Megavolt Ampere Reactive MW = Megawatts kV = Kilovolts
Tier being Tested: ES401 Tier 1/Group 1 (RO)

18. Following a Reactor Trip from 100% power, the following conditions are noted:
    - ALL Reactor Coolant Pumps are operating
    - BOTH Main Steam Isolation Valves are closed
    - BOTH Auxiliary Feedwater Steam Admission Valves are open
    - AFPT 1 and 2 speed are zero rpm
    - RCS Tave is 570°F and RISING
    - RCS Pressure is 2200 psig and RISING
    - Pressurizer level is RISING

    Which of the following Pump(s) should be started FIRST in accordance with DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture to mitigate this event?

A. Motor Driven Feedwater Pump

B. Emergency Feedwater Pump

C. High Pressure Injection Pumps

D. Low Pressure Injection Pumps

Answer: A

Explanation/Justification: This Question is a KA match because it tests the candidates understanding of the systems available and the priority for use of those systems in response to an Inadequate Heat Transfer event. At DB, DB-OP-02000 provides direction to respond to a lack of heat transfer event. This direction is provided by Specific Rules and by Section 6, LOHT. The Candidate must know that with both MSIVs closed, the Main Feedwater Pumps will not be available. Both AFW Pumps should be running, but as noted by steam admission valves open with zero speed, both AFW Pumps have failed. IF the MDFP also fails, then the EFW Pump would be started. Once the specific rules are completed, Section 6, Lack of Heat Transfer directs starting HPI and LPI to prepare for Feed and Bleed Cooling.

A. CORRECT – DB-OP-02000 Specific Rule 4 which is implemented prior to transition to DB-OP-02000 Section 6, Lack of Heat Transfer directs first attempting to restore feedwater using the MDFP. This is the response that matches the procedure direction to mitigate the events provided in the question stem.

B. Incorrect because the EFW pump is not directed to be started first – Plausible because Specific Rule 4 is implemented prior to transition to DB-OP-02000, Section 6 lack of Heat Transfer. The Specific Rule directs attempting to restore feedwater but directs use of the MDFP prior to placing the EFW Pump in service.

C. Incorrect because the HPI pump is not directed to be started first – Plausible because DB-OP-02000, Section 6 Lack of Heat Transfer directs starting the High Pressure Injection pumps to prepare for Feed and Bleed Cooling. This is incorrect because the MDFP will be started first, negating the need to enter Section 6.

D. Incorrect because the LPI pump is not directed to be started first – Plausible because DB-OP-02000, Section 6 Lack of Heat Transfer directs starting the Low Pressure Injection pumps to prepare for Feed and Bleed Cooling. This is incorrect because the MDFP will be started first, negating the need to enter Section 6.

Sys #	System	Category	KA Statement
E04	Inadequate Heat Transfer Loss of Secondary Heat Sink	Knowledge of the interrelations between the (Inadequate Heat Transfer) 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** 4.2 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02000 R32, Specific Rule 4 and Section 6, Lack of Heat Transfer Step 6.3

Question Source:	NEW	**Level Of Difficulty: (1-5)**	3
Question Cognitive Level:	HI	**10 CFR Part 55 Content:**	41.7, 45.7
Objective:	OPS-GOP-301-04K		

Abbreviations: AFPT = Auxiliary Feedwater Pump Turbine, RCS = Reactor Coolant System
Tier being Tested: ES401 Tier 1/Group 1 (RO)

19. Initial conditions:
- The Reactor is operating at 80% power
- All Integrated Control System Control Stations are in a normal lineup
- Rod Index is 260
- DB-SP-03450, Boron Injection Flowpath Boric Acid Pump 1 Test is in progress

As the result of a lineup error, approximately 10 gallons from Boric Acid Addition Tank 1 was added to the Reactor Coolant System (RCS).

Which of the following describes the expected stable plant condition, due to the change in RCS Boron Concentration, if no operator action is taken?

A. Reactor Power will lower, Tave will lower, Rod Index will remain at approximately 260.

B. Reactor Power will remain at approximately 80%, Tave remain at approximately 582°F, Rod Index will lower.

C. Reactor Power will remain at approximately 80%, Tave will remain at approximately 582°F, Rod Index will rise.

D. Reactor Power will lower, Tave will remain approximately 582°F, Rod Index will remain at approximately 260.

Answer: C

Explanation/Justification: This question is a KA match. Surveillance Testing (DB-SP-03450) of Boric Acid Pump 1 of the Emergency Boration System is in progress. As the result of an improper valve lineup, boric acid from the Emergency Boron System is added to the RCS. The question tests the candidate's knowledge of the impact with the Automatic control systems in response to a boron addition. This tests the knowledge of the operational implications of a boron addition from the Emergency Boration System to an operating reactor.

A. Incorrect because with the Rod Control System in Automatic, Rod Index would rise as the system compensates for the boron addition – Plausible because lowering of reactor power would occur and Rod index would remain the same if the Rod control was in manual.

B. Incorrect because 10 gallons of boric acid would affect core reactivity and cause the automatic control systems to response accordingly – Plausible if the candidate considers the amount of boric acid to be small. As the plant operates, the operators make routine additions of demin water almost daily. This 10 gallon addition is far smaller than a typical demin water addition. As a result, the operator may consider that this small amount of concentrated boric acid will not cause any significant impact on the plant.

C. CORRECT – With ICS in full auto, the Unit Load demand will maintain power approximately 80% and the Tave controller will maintain Tave approximately 582F. Control Rods will automatically withdraw to add positive reactivity equal to the negative reactivity added by the boron reaching the RCS from the Makeup System. As a result, Rod Index will rise as group 7 rods are withdrawn. This is the only completely correct response for the plant conditions provided.

D. Incorrect because with the Rod Control System in automatic, Rod Index would rise as the system compensates for the boron addition – Plausible because lowering of reactor power would occur if the Rod control was in manual. Lowering Reactor Power typically adds a small amount of positive reactivity which could offset the small amount of negative reactivity added by the boric acid.

Sys #	System	Category	KA Statement
025	Emergency Boration	Knowledge of the operational implications of the following concepts as they apply to Emergency Boration:	Relationship between boron addition and reactor power

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

References provided to Candidate: None **Technical References:** DB-SP-03450 R19, Boron Injection Flowpath Test BA Pump 1, Limit and Precaution 2.2.4, Lesson Plan OPS-SYS-I517, Reactor Control, Interactive ICS Trainer

Question Source: NEW **Level Of Difficulty: (1-5)** 3

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Question Cognitive Level: HI

10 CFR Part 55 Content: (CFR 41.8 / 41.10 / 45.3)

Objective: OPS-SYS-512-02K

Tier being Tested: ES401 Tier 1/Group 2 (RO)

20. Plant Conditions:
- A Reactor Startup is in progress in accordance with DB-OP-06912, Approach to Criticality
- All Control Rod Group 1-4 rods have been fully withdrawn and the plant has entered Mode 2
- Withdrawal of Control Rod Group 5 rods is in progress, holding at 50% withdrawn to take data for 1/M plot
- Estimated Critical Rod Position is 70% on Group 6

The following indications are noted:
- Nuclear Instrument NI 3, INTERMEDIATE RANGE LOG N is erratic, bouncing between 10^{-11} and 10^{-7} amps
- Nuclear Instrument NI 4, INTERMEDIATE RANGE LOG N is not on scale
- Nuclear Instrument NI 1, SOURCE RANGE LOG COUNT RATE is stable at 100 cps.
- Nuclear Instrument NI 2, SOURCE RANGE LOG COUNT RATE is stable at 110 cps.
- Annunciator (5-5-E) SUR ROD WITHDRAW INHIBIT alarms and clears multiple times.

Instrumentation & Control Technicians indicate that it will require approximately 90 minutes to resolve this issue.

Which of the following actions and reason for the actions is required by DB-OP-06912, Approach to Criticality in response to these indications?

A. Hold at current conditions to allow I&C to resolve the issue. Gammametric Instruments are available to monitor Reactor Power during the hold.

B. Insert Group 5 Control Rods. Inserting Group 5 rods returns the plant to Mode 3, where only one operable Intermediate Range Nuclear Intermediate Range instrument is required.

C. Insert Groups 2-5 Control Rods. Inserting all Group 2 through 5 Control Rods returns the plant to the conditions that existed prior to the Approach to Criticality to accommodate the delay in the Reactor Startup.

D. Trip the Reactor. Tripping the reactor returns the plant to Mode 3, where only one operable Intermediate Range Nuclear Intermediate Range instrument is required.

Answer: C

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the DB-OP-06912 requirements and the reasons to terminate a reactor startup when any condition physically or administratively delays criticality by more than 1 hour. An erratic Intermediate range nuclear instrument is the condition that is going to delay the Reactor Startup for this question.

A. Incorrect because the controlling procedure directs insertion of all Control Rods except Group 1 rods when an approach to criticality is delay by more than 1 hour – Plausible because multiple methods exist to monitor reactor power and the question does not require a Mode Change to continue the reactor startup since Mode 2 has already been entered prior to the start of Group 5 withdrawal. Plus, the action stops the reactor startup when questionable indication occurs.

B. Incorrect because the controlling procedure directs insertion of all Control Rods except Group 1 rods when an approach to criticality is delay by more than 1 hour – Plausible because inserting Control Rods is directed by DB-OP-06912, Approach to Criticality, however the Control Rods to be inserted for a delay in the startup is incorrect. The procedure directs inserting all Control Rods with the exception of Group 1 which is left withdrawn for trippable reactivity. The candidate may incorrectly assume that return to only Safety Rods with drawn (Group 1-4) would be sufficient to comply with the DB-OP-06912 requirements for a delayed startup.

C. CORRECT - DB-OP-06912 requires terminating a reactor startup when any condition physically or administratively delays criticality by more than 1 hour. Based on the response of NI3 and the estimate time to resolve, this condition will delay startup by more than one hour. The procedure directs inserting groups 2-5 control rods.

D. Incorrect because the controlling procedure directs insertion of all Control Rods except Group 1 rods when an approach to criticality is delay by more than 1 hour – Plausible because the candidate may conclude that tripping the reactor is always safe. For the conditions provided, the issue is only an indication problem, not an actual change in power level and startup rate. If the candidate concludes an actual high startup rate exists, tripping the reactor would rapidly resolve the issue. In addition, tripping the reactor would insert all control rods which is similar to the procedurally directed action to insert all Control Rods with the exception of Group 1.

Sys #	System	Category	KA Statement
033	Nuclear Instruments	Knowledge of the reasons for the following responses as they apply to the Loss of Intermediate Range Nuclear Instrumentation:	Termination of startup following loss of intermediate range instrumentation.

K/A# AK3.01 **K/A Importance** 3.2 **Exam Level:** RO

References provided to Candidate: None

Technical References: TS 3.3.10, DB-OP-02505 R06 Nuclear Instrument Failure Subsection 4.4, and DB-OP-06912, Approach to Criticality R22, step 4.1.23

Question Source: NEW

Level Of Difficulty: (1-5) 3

Question Cognitive Level: HI

10 CFR Part 55 Content: 41.5,41.10 / 45.6 / 45.13

Objective: OPS-GOP-201-05A

Tier being Tested: ES401 Tier 1/Group 2 (RO)

21. The plant is operating at 100% power with all systems in normal alignment.
    - NO planned radioactive liquid releases are in progress
    - RE4686, Storm Sewer Outlet alarms and indicates above its HIGH alarm setpoint

    This alarm indicates possible leakage from which of the following systems?

A. Miscellaneous Liquid Radwaste System

B. Clean Liquid Radwaste System

C. Demineralized Water System

D. Condensate Polishing System

Answer: D

Explanation/Justification: This question is a KA match because it tests the candidate knowledge of an accidental Radioactive Liquid Release. RE4686 alarms which indicates that the storm sewer is potential releasing radioactive material to the environment. Knowledge of the system interrelations will allow the identification of potential sources of the leakage.

A. Incorrect the Miscellaneous Liquid Radwaste System leakage would NOT flow to the Storm Sewer– Plausible because this is a radioactive system, however this system is located in the Auxiliary Building. Leakage from this system would go to a floor drain or sump in the Auxiliary Building and be transported to the Misc Waste Drain Tank, not the storm sewer. There is no connection between the Misc Waste Drain Tank and the Storm Sewer.

B. Incorrect the Clean Liquid Radwaste System leakage would NOT flow to the Storm Sewer – Plausible because this is a radioactive system, however this system is located in the Auxiliary Building. Leakage from this system would go to a floor drain or sump in the Auxiliary Building and be transported to the Misc Waste Drain Tank, not the storm sewer. There is no connection between the Misc Waste Drain Tank and the Storm Sewer.

C. Incorrect because the Demineralized Water System is not radioactive - Plausible because leak from this system can reach the storm sewer, but the system is not radioactive and would not cause an alarm on the Storm Sewer Radiation Monitor.

D. CORRECT – With SG Tube Leak or other events that contaminate the Secondary System, low levels of radioactivity exist in the condensate polishers.. Leakage from this system could reach the storm sewer via the Turbine Building Drains and cause the storm sewer outlet alarm provided in the question stem.

Sys #	System	Category	KA Statement
059	Accidental Liquid Radwaste Release	Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following:	Radioactive-liquid monitors

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

References provided to Candidate None **Technical References:** DB-OP-02531 R24 Attachment 7 page 3 and 4 discussions on contaminated secondary drains

DB-OP-06272 R32 Sect 4.6, ATT 5

Question Source: DB 2013 NRC Exam Q21 **Level Of Difficulty: (1-5)** 3

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

Objective: OPS-GOP-131-11K

Tier being Tested: ES401 Tier 1/Group 2 (RO)

22. The Reactor is operating at 100% power.

The following plant event occurs:

- A radiography source fell out of its container in the Control Room creating dangerous radiation levels in the Control Room
- The Shift Manager directs evacuation of the Control Room in accordance with DB-OP-02508, Control Room Evacuation
- There is no time to complete any Supplemental Actions prior to evacuating the Control Room

In accordance with DB-OP-02508, Control Room Evacuation, which of the following actions will be completed after the Reactor is Tripped?

A. Evacuate the Control Room then
At the Aux Shutdown Panel, take local control of both Main Feedwater flow and RCS inventory.

B. Confirm the Reactor is Tripped then
Evacuate the Control Room, trip both Main Feed Pumps locally and establish local control of Aux Feedwater flow and RCS inventory from the Aux Shutdown Panel.

C. Initiate and Isolate SFRCS and confirm Immediate Actions then
Evacuate the Control Room and establish local control of Aux Feedwater flow and RCS inventory from the Aux Shutdown Panel.

D. Trip all RCPs and confirm Immediate Actions then
Evacuate the Control Room and establish local control of Aux Feedwater flow and RCS inventory from the Aux Shutdown Panel.

Answer: C

Explanation/Justification: This question is a match for the KA because it tests the candidate's ability to operate and/or monitor the operating characteristics of the facility as they apply to a shutdown outside the Control Room. DB has two AOP level procedures that direct operation outside the Control Room, DB-OP-02508, Control Room Evacuation, and the Fire Related DB-OP-02519, Serious Control Room Fire. The question uses a non-fire related event that requires Control Room evacuation and then asks about the mitigation strategy for that event.

Immediate actions are to.
Trip the Reactor.
Initiate and Isolate SFRCS using Manual Actuation Switches.
GO TO DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture for confirmation of Immediate Actions AND return to this procedure.

A. Incorrect because all Immediate Actions have not been completed prior to evacuation and MFW Flow can't be controlled from the Aux SD Panel.
Plausible because the reactor trip is confirmed prior to evacuation and RCS inventory is controlled from the Aux Shutdown Panel.

B. Incorrect because all Immediate Actions have not been completed prior to evacuation and MFP's are NOT tripped locally in this scenario.
Plausible because local control of Aux Feedwater flow and RCS inventory is controlled from the Aux Shutdown Panel and there are steps in Attachment 4 to trip MFPs if IA's have not been completed prior to evacuation.

C. CORRECT – DB-OP-02508, Control Room Evacuation provides direction for operation (shutdown) of the facility from outside the Control Room in response to non-fire events. Response C is a general discussion of the procedure direction provided by DB-OP-02508. This response is the only response provided that matches the procedurally directed actions.

D. Incorrect because RCPs are not tripped IAW DB-OP-02508 in response to the events provided.
Plausible because the actions are similar to those directed by DB-OP-02508 but include actions not directed by DB-OP-02508. All RCPs are tripped in the DB-OP-02519 response that abandons the Control Room.

Sys #	System	Category	KA Statement
068	CR Evac	Ability to operate and / or monitor the following as they apply to the (Shutdown Outside Control Room):	Operating behavior characteristics of the facility.

K/A# AA1.2 **K/A Importance** 3.2 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02508 R18, Control Room Evacuation, Sections 3 and 4 and Attachments 1-5.

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** 41.7 / 45.5 / 45.6

Objective: OPS-GOP-108-03K

Tier being Tested: ES401 Tier 1/Group 2 (RO)

23. The plant is in Mode 4 with a plant heatup in progress.

Maintenance reports that the inner door on the Personnel Hatch Airlock will not seal properly after leaving containment.

In Accordance With Technical Specification 3.6.2, Containment Air Locks, which ONE of the following identifies the action required, if any?

A. No action required, the inner door does not form part of the Containment pressure boundary.

B. Within 1 hour, verify the Personnel Hatch Airlock outer door is closed.

C. Within 24 hours perform DB-SP-03291, Containment Personnel and Emergency Airlock Seal Test.

D. Immediately initiate action to evaluate overall containment leakage rate.

Answer: B

Explanation/Justification: This question is a KA match because it creates a situation where a key element of Containment Integrity is challenged by an inoperable door and then questions the candidate's knowledge of the requirement to confirm the operable door is closed to maintain Containment Integrity.

A. Incorrect because the inner door is part of the Containment Pressure Barrier – Plausible if the Candidate assumes that only the outer door is part of the Containment Pressure Boundary since that door is the last barrier between the airlock and the environment.

B. CORRECT – With the plant in Mode 4, both CTMT Personnel Air Lock doors are required to be operable. The Inner Door cannot be sealed rendering the door inoperable. Technical Specification 3.6.2, Containment Air Locks, Condition A, One or more containment air locks with one containment air lock door inoperable requires that the Operable Door in the affected air lock be closed within one hour. This is the only correct answer IAW TS 3.6.2

C. Incorrect because this is not the action required by TS 3.6.2 for the conditions provided – Plausible because this is the test that will be performed to confirm Containment Airlock operability following repair of the condition that prevents sealing of the Inner Personnel Air Lock Door.

D. Incorrect because this is not the action required by TS 3.6.2 for the conditions provided – Plausible this is a TS requirement (TS 3.6.2 Condition C) if an air lock is inoperable for reasons other than Conditions A or B. Condition A is the condition presented in the Question Stem.

Sys #	System	Category	KA Statement
069	CTMT Integrity	Ability to determine and interpret the following as they apply to the Loss of Containment Integrity:	Loss of containment integrity

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

References provided to Candidate: None **Technical References:** TS 3.6.2

Question Source: 2011 Crystal River NRC Exam Q53 **Level Of Difficulty: (1-5)** 3.5

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 43.5 / 45.13)

Objective: OPS-GOP-420-03K

Tier being Tested: ES401 Tier 1/Group 2 (RO)

24. The plant has tripped from 100% power due to a Loss of Off-Site power.

The crew has routed to DB-OP-02000, Section 9, INADEQUATE CORE COOLING due to multiple equipment failures.

The following conditions are noted:

- A Total Loss of Feedwater has been identified
- Reactor Coolant System Pressure is 600 psig and stable
- Average Incore Thermocouple Temperature is 600°F and slowly rising
- Steam Generator 1 and 2 Levels are 10 inches and stable
- Steam Generator 1 and 2 Pressures are 800 psig and stable

Auxiliary Feedwater Pump 1 has just been restored to service.

Based on these plant conditions, which of the following actions should be taken in accordance with DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture?

Raise SG 1 Level to ___________.

A. 49 inches at full flow

B. 49 inches at a maximum of 100 gpm

C. 124 inches at full flow

D. 124 inches at a maximum of 100 gpm

Answer: C

Explanation/Justification: KA match because the question tests the applicant's knowledge of the mitigation strategy during Inadequate Core Conditions.

The stem states the crew has routed to DB-OP-02000, Section 9, INADEQUATE CORE COOLING. It also gives conditions that indicate both SGs are dry (<16" level AND <960 psig). AFW flow is limited to 100 gpm if no RCPs are in service in the loop with the dry SG except during Lack of Heat Transfer OR Inadequate Core Cooling. DB-OP-02000 R32, Section 9, Inadequate Core Cooling Step 9.6 RNO states IF Feedwater is available, THEN manually increase SG level to 124 inches. During Inadequate Core Cooling, the 100 gpm limit to restore level to a dry SG is not applicable.

A. Incorrect because the level provided is not the level directed by DB-OP-02000- 49 inches is incorrect, plausible because 49 inches is the set point when AFW is initiated for other plant conditions.
- Full flow is correct. IAW Specific Rule 4 SG recovery flow limits do NOT apply during Lack of Heat Transfer OR Inadequate Core Cooling.

B. Incorrect because the level provided is not the level or flowrate directed by DB-OP-02000 - 49 inches is incorrect, plausible because 49 inches is the set point when AFW is initiated for other plant conditions.
- 100 gpm is incorrect, plausible since 100 gpm is the flow limit to a dry SG when no RCPs are in service and Inadequate Core Cooling does not exist.

C. CORRECT – The stem states the crew has routed to DB-OP-02000, Section 9, INADEQUATE CORE COOLING. It also gives conditions that indicate both SGs are dry (<16" level AND <960 psig). IAW Specific Rule 4 SG recovery limits do NOT apply during Lack of Heat Transfer OR Inadequate Core Cooling. DB-OP-02000 R32, Section 9, Inadequate Core Cooling Step 9.6 RNO states IF Feedwater is available, THEN manually increase SG level to 124 inches. Full AFW flow would be used during the restoration of flow to a dry SG. This response provides both the level and flowrate directed by DB-OP-02000.

D. Incorrect because the flowrate for restoration is not the flowrate directed by DB-OP-0200 - 124 inches is correct.
- Incorrect. Flow to a dry SG is NOT limited when ICC Conditions exist. Plausible since flow is limited to 100gpm when no RCPs are not running.

Sys #	System	Category	KA Statement
074	Inadequate Core Cooling	Knowledge of EOP Mitigation Strategies	Knowledge of EOP mitigation strategies.

K/A# G2.4.6 **K/A Importance** 3.7 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02000 R32, Section 9, Inadequate Core Cooling Step 9.6 & SR4

Question Source: New **Level Of Difficulty: (1-5)** 3

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

Objective: OPS-GOP-308-02K

Tier being Tested: ES401 Tier 1/Group 2 (RO)

25. Plant Conditions:
    - The Reactor is operating at 80% power
    - Testing of the ICS Reactor Subsystem is in progress:
      - The Reactor Demand Hand/Auto Station is in MANUAL
      - The Rod Control Panel is in MANUAL
    - All other ICS Control Stations are in their normal lineup

    An event occurs, and the following conditions are noted immediately:
    - Annunciator 14-3-D, ICS MFP LOSS OR LO DEAR RUNBACK is in alarm
    - Annunciator 8-4-B, MFPT 2 TRIP is in alarm

    Which of the following actions is required IAW DB-OP-06401, Integrated Control System Operating Procedure?

A. Use the Rod Control Panel to insert control rods to reduce Reactor Power.

B. Use the Reactor Demand Hand/Auto Station to insert control rods to reduce Reactor Power.

C. Place SG/Rx Demand Station in hand and perform runback.

D. Adjust Unit Load Demand setpoint to reduce Reactor Power.

Answer: A

Explanation/Justification: This question is a KA match because it creates a situation involving a runback caused by a Main Feedwater Pump Trip with an ICS Control Station in Manual. In this condition, ICS will not be able to complete the required power reduction. DB-OP-02014, Alarm Panel 14 Annunciator 14-3-D directs that IF any ICS H\A Station is in HAND, THEN manually perform the runback function on those stations. In this condition, the Reactor Operator is required to perform the power reduction function normally performed by ICS. Reactor Power was specified to be at 80% to avoid any discussion of the time available to initiate the Runback and it is above the directed power level of 55% for a MFP trip. In addition, with both the Reactor Demand Hand/Auto Station and the Diamond Rod Control Panel in Manual, the operator must know to use the downstream station (Rod Control) to complete the power reduction. Action is required because 80% power exceeded the capacity of a single Main Feedwater Pump.

A. CORRECT - the Reactor Demand H/A station and Diamond Rod Control panels are in manual preventing the normal automatic response. That means the ICS Unit Load Demand which would automatically adjust output demand to reduce power to 55% at 20% per minute but this will not cause the required power reduction because the Diamond Control Panel is in manual. Reducing Reactor Power will require insertion of control rods using the IN-HOLD-OUT switch on the Diamond Control Panel.

B. Incorrect because the Diamond Rod Control Panel is in manual and would not respond to the Reactor Demand Hand/Auto Station commands – Plausible because if Rod Control Panel in Automatic, using the using the Reactor Demand Hand/Auto station to reduce demand would cause the Control Rods to insert to reduce Reactor Power.

C. Incorrect because the Reactor/ Steam Generator Master will not be able to cause Control Rods to insert because downstream stations are in manual (Reactor Demand and Rod Control Panel. Plausible because a Main Feedwater Pump Trip has cause and upset in primary to secondary heat transfer. DB-OP-02526, Primary to Secondary Heat Transfer Upset does direct placing the Reactor/Steam Generator in Hand (step 4.2), however DB-OP-02526 states that it is not applicable for a loss of ICS power (Section 1).

D. Incorrect because 20% rate of change is beyond the operator set possible values and it is not the method directed by the controlling procedure – Plausible because if the ICS was in full automatic, and the system failed to detect the MFP trip, this method would cause the power reduction to 55% at 20 percent per minute in a smooth controlled fashion. However, the highest operator set rate of change on the Unit Load Demand is 10% per minute which also makes D incorrect.

Sys #	System	Category	KA Statement

BW A01	Runback	Knowledge of the operational implications of the following concepts as they apply to the (Plant Runback)	Normal, abnormal and emergency operating procedures associated with (Plant Runback).

K/A# AK1.2 **K/A Importance** 3.5 **Exam Level:** RO

References provided to Candidate: None

Technical References: DB-OP-02014, R17 page 30, Ann 14-3-D response, DB-OP-06401 R31, Page 84, step 4.9.2.b
NOP-OP-1002 R16 pages 61 & 62

Question Source: NEW

Level Of Difficulty: (1-5) 3

Question Cognitive Level: HI

10 CFR Part 55 Content: (CFR: 41.8 / 41.10 / 45.3)

Objective: OPS-SYS-512-17K

Tier being Tested: ES401 Tier 1/Group 2 (RO)

26. The following plant conditions exist:
    - The reactor has tripped due to a large RCS Leak
    - SFAS Levels 1-4 have actuated
    - Containment Pressure is 21 psia and slowly lowering
    - RCS Pressure is 300 psig and slowly lowering
    - Average Incore Thermocouple temperature is 420°F and slowly lowering
    - The Borated Water Storage Tank Level is lowering at 1 foot per HOUR, and currently at 9 feet

    Which of the following actions must be completed prior to transferring the Emergency Core Cooling Pump suctions to the Containment Emergency Sump?

A. Verify Piggyback Valves are Open AND place the HPI Alternate Minimum Recirc lines in service.

B. Close RCS High Point Vents AND Shutdown Both AFW Trains.

C. Align the RCS for Long Term Boron Dilution AND verify Both Containment Spray Pump discharge valves are fully open.

D. Stop Both Makeup AND High Pressure Injection Pumps.

Answer: A

Explanation/Justification: This question is a KA match because a loss of Subcooling Margin has occurred due to a small break LOCA (RCS Temp and Pressure provided indicated saturated RCS with BWST level lowering) With these plant conditions, Low Pressure Injection flow into the core does not exist (RCS Pressure greater than LPI discharge Pressure). DB-OP-02000 Section 11, RCS Saturated with the SGs removing heat is the applicable EOP Cooldown Section. Subcooling Margin is less than the 20°F required and is therefore inadequate. Since the HPI termination criteria are not met, HPI flow will be required via piggyback from LPI when transferring to the Emergency Sump. Since rate of BWST usage is less than 2 feet per hour, HPI alternate minimum recirc flowpath will be required. This question tests the proper operations of multiple heat removal systems with inadequate subcooling margin

A. CORRECT - With these plant conditions, Low Pressure Injection flow into the core does not exist (RCS Pressure greater than LPI discharge Pressure). As a result, HPI termination criteria is not met, requiring use of piggyback prior to transferring to the Containment Emergency Sump. Since the rate of BWST usage is less than 2 feet per hour, HPI alternate minimum recirc flowpath will be required as directed by DB-OP-02000, Step 11.16. Of the 4 possible responses, this is the only correct action for the plant conditions provided.

B. Incorrect – Plausible because with a large break LOCA, when SG pressures reach 35 psig, both AFW trains will be shutdown. This scenario is not a large break LOCA and SG cooling is supplementing RCS leak flow to provide some core cooling. This response is incorrect because the Steam Generators (AFW) will remain in service to provided cooling of the SG shell limiting to tube to shell differential temperature and the resultant stress that could lead to a SG Tube failure.

C. Incorrect – Plausible because Long Term boron dilution would be established if proper conditions are met. Conditions in this scenario are not met. The condition that is not met is average Incore Thermocouples temperature is 420°F, it must be < 333°F. CTMT Spray can be shutdown, but only when CTMT pressure has been reduced to 4 psig or 18.7 psia not the 21 psia provided in the question. Containment Spray Pump discharge valves are verified in the throttled position after the transfer is complete.

D. Incorrect – Plausible because both Makeup Pumps will be stopped prior to transfer to the Emergency Sump (OP2000 Att 7 Sect 2) due to MU Pump clearances and the possibility of small debris in the Emergency Sump however, for the given plant conditions HPI must remain in service. HPI also has tighter tolerances, but the HPI Pumps are designed to operate with a suction from the Emergency Sump. This response is incorrect because HPI is the only ECCS pump providing inventory to the RCS. Shutdown of HPI under these conditions would likely cause a loss of core cooling.

Sys #	System	Category	KA Statement

BWE03	Inadequate Subcooling Margin	Knowledge of the interrelations between the (Inadequate Subcooling Margin) 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** 4.3 **Exam Level:** RO

References provided to Candidate None **Technical References:** DB-OP-02000 R32 Steps 11.16 including RNO

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.7 / 45.7)

Objective: OPS-GOP-304-03K

Abbreviations: RCS = Reactor Coolant System, SFAS = Safety Features Actuation System, HPI = High Pressure Injection, AFW = Auxiliary Feedwater

Tier being Tested: ES401 Tier 1/Group 2 (RO)

27. DB-OP-02000, Specific Rule 2 requires immediate tripping of the Reactor Coolant Pumps when adequate subcooling margin (SCM) is lost.

    Which of the following best describes the reason for this action?

    Immediate tripping the Reactor Coolant Pumps is required to ___________.

A. prevent RCP Seal damage

B. shift SG Level Control signal to the 124-inch setpoint

C. minimize the possibility of uncovering the core if flow is lost later

D. actuate the Auxiliary Feedwater Pumps by SFRCS on Loss of 4 RCPs

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Answer: C

Explanation/Justification: As described in EOP TBD Volume 1 Section III.B.1 tripping of all RCPs later (beyond the 2-minute criteria) could cause the core to uncover. This could occur due to conditions beyond the operator's control (i.e., auto RCP trip or loss of bus power). Therefore, to protect the core the RCPs must be tripped within 2 minutes. This action is provided by Specific Rule 2 which matches the EOP Rules required for this KA. The question directly tests knowledge of the rules for manipulating the controls (RCPs) during an Emergency Condition (Loss of SCM) which is also a match for the KA.

A. Incorrect because this is not the reason for tipping the RCPs provided in the DB-OP-02000 Bases and Deviation Document. – Plausible because the RCPs will not have adequate NPSH and running without adequate NPSH could cause vibrations that could lead to RCP Seal Failure.

B. Incorrect because the SG level setpoint will automatically select the high level (124 inches) on the SFAS Actuation that would occur on the loss of SCM either on low RCS pressure or on High Containment Pressure – Plausible because an elevated SG level of 124 inches is required by Specific Rule 4 for loss of SCM.

C. CORRECT – See explanation above. This is the correct reason as provided by the DB-OP-02000 Bases and Deviation Document.

D. Incorrect – Plausible because stopping the 4 reactor coolant pumps will cause an SFRCS Loss of 4 RCPs which will start both AFW trains. AFW is desired to promote SG heat transfer because it sprays high into the SG.

Sys #	System	Category	KA Statement
BW E13	EOP Rules and Enclosures	Knowledge of the reasons for the following responses as they apply to the (EOP Rules):	Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

K/A# EK3.3 **K/A Importance** 3.2 **Exam Level: RO**

References provided to Candidate: None

Technical References: DB-OP-02000 R32, Specific Rule 2, EOP TBD Volume 1 Section III.B.1, and DB-OP-02000 Bases and Deviation Document R22, Page Specific Rule 2 page 443

Question Source: DB1LOT18 Q26

Level Of Difficulty: (1-5) 3

Question Cognitive Level: LO

10 CFR Part 55 Content: (CFR: 41.5 / 41.10, 45.6, 45.13)

Objective: OPS-GOP-301-02K

Tier being Tested: ES401 Tier 1/Group 2 (RO)

28. Initial plant conditions:
    - Reactor Power is at 70%
    - All 4 Reactor Coolant Pumps are running
    - All Integrated Control System (ICS) Hand/Auto Stations are in Auto

    The following event occurs:
    - RCP 2-2 trips
    - Smart Automatic Signal Selector (SASS) functions occur as expected

    (1) Which of the following is the signal the ICS will receive for Tave input?

    AND

    (2) How will the trip of RCP 2-2 impact SG Levels?

A. (1) Loop 2 Tave
(2) SG 1 Level will be higher than SG 2 Level

B. (1) Loop 1 Tave
(2) SG 1 Level will be higher than SG 2 Level

C. (1) Loop 1 Tave
(2) SG 2 Level will be higher than SG 1 Level

D. (1) Loop 2 Tave
(2) SG 2 Level will be higher than SG 1 Level

Answer: B

Explanation/Justification: This Question is a KA match because it tests the candidate's knowledge of the affect that a Reactor Coolant Pump malfunctions (RCP 2-2 trips) has on the Integrated Control System. Normally, ICS uses the Tave Signal from Loop 2. With the trip of RCP 2-2, flow in loop 2 will be less than the flow in Loop 1. The SASS will transfer Tave control from RCS Loop 2 to Loop 1 based on the now higher flow in RCS Loop 1. In addition, the Feedwater Control portion of ICS will re-ratio Feedwater Demands to allow each SG to carry the steam flow loads based on their RCS flow signal and delta Tcold.

A. Incorrect
- (1) Incorrect because this response provides the wrong Tave signal that will be used for control. Plausible since Loop 2 Tave is the normal controlling Tave Loop.
- (2) This portion of the response is correct. Since a Loop 2 RCP trips, Loop 1 will have the highest RCS flow, FW flow and therefore SG level will be higher in SG 1 which is correct.

B. CORRECT
- (1) Correct because the Smart Analog Selector Switch (SASS) for Tave automatically selects the Loop with the Highest RCS Flow when an RCP is stopped. Since a Loop 2 RCS trips, Loop 1 will have the highest flow and Loop 1 Tave will be selected.
- (2) Correct because ICS will ratio FW flow to the Steam Generators based on RCS flow or about 2.4 to 1 with the 2 RCP loop SG receiving the higher Feedwater Flow and will operate at a higher Steam Generator Level.

C. Incorrect
- (1) Correct because the Tave selected by ICS is correct for a trip of RCP 2-2.
- (2) Incorrect because Loop 1 will have the highest flow FW flow and therefore SG level will be higher in SG 1 because ICS re-ratios FW flow to the SG based on RCS flow to that SG. SG1 will have the high level.

D. Incorrect
- (1) Incorrect because this response provides the wrong Tave signal that will be used for control. Plausible because Loop 2 Tave is the normal controlling Tave Loop.
- (2) Incorrect because the SG with the higher level provided is incorrect. Plausible since Loop 2 Tave is the normal controlling Tave Loop. Without the expected SASS transfer, RCS Loop 2 Tave would be the controlling Tave.

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Sys #	System	Category	KA Statement
003	Reactor Coolant Pump System (RCPS)	Knowledge of the effect that a loss or malfunction of the RCPS will have on the following:	ICS

K/A#	K3.05	**K/A Importance**	3.6*	**Exam Level**	RO

References provided to Candidate	None	**Technical References:**	DB-OP-02515 R18 Attachment 1 pg 51
Question Source:	DB 2011 NRC Exam Q 29	**Level Of Difficulty: (1-5)**	3
Question Cognitive Level:	High - Comprehension	**10 CFR Part 55 Content:**	(CFR: 41.7 / 45.6)
Objective:	OPS-GOP-115-03K		
Tier being Tested:	ES401 Tier 2/Group 1 (RO)		

29. The plant is operating at 100% normal full power lineup with no evolutions in progress.

A small RCS Leak has developed.

Which of the following design features or interlocks will prevent the Makeup (MU) Pumps from being operated with insufficient Net Positive Suction Head (NPSH) without operator action?

A. At 25 inches level in the MU Tank, the 3-way Letdown Valve MU11 will switch from the CLN WST position to the MU TK position.

B. At 17 inches level in the MU Tank, the MU Pump Suction valves MU3971 and MU6405 will align to the BWST.

C. At 15 psig MU Tank Pressure, the MU Tank will align to the Nitrogen header to provide required MU Tank Pressure.

D. At 10 psig MU Tank Pressure, Both MU Pumps will trip.

Answer: B

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the design features and interlocks that provide for a minimum level in the MU Tank. Low Makeup Tank level can lead to MU Pump failure if the suction source is not adequate. At Davis-Besse, an interlock exists that causes the MU Pump Suctions to realign from the MU Tank to the BWST at 17 inches in the MU Tank. In addition, but not directly tested, the MU Pumps will trip after 45 seconds if this transfer is not completed within 45 seconds.

A. Incorrect – Plausible because this interlock does exist. In the stem of the question provided, Letdown would not be aligned to the Clean Waste System. Incorrect because, the interlock does not protect the MU Pump from Operating without NPSH. This interlock attempts to correct a potential cause of low MU Tank Level. If Letdown is aligned to the Clean Waste Receiver Tank instead of the MU Tank, at 25 inches, the system will attempt to resolve this issue by realigning Letdown to the MU Tank. Even if Letdown was aligned to Clean Waste, this interlock would not directly protect the MU Pumps.

B. CORRECT - Low Makeup Tank level can lead to MU Pump failure if the suction source is not adequate. An interlock exists that causes the MU Pump Suctions to realign from the MU Tank to the BWST at 17 inches in the MU Tank. Successful transfer ensure the MU Pumps have adequate NPSH provided by the BWST. This interlock protects the MU Pumps and is therefore the only correct response provided.

C. Incorrect – Plausible because NPSH for the MU Pumps is provided by a combination of level and pressure in the MU Tank. Normally, pressure in the MU Tank is provided by Hydrogen Gas. The operators should know that H2 gas is a manual alignment since addition is a routine operation. They will also be aware the N2 gas is available. The operator may incorrectly assume that N2 is provided by an automatic regulator to maintain a minimum pressure which would provide adequate NPSH. This is incorrect.

D. Incorrect – Plausible because NPSH for the MU Pumps is provided by a combination of level and pressure in the MU Tank. Normally, pressure in the MU Tank is provided by Hydrogen Gas. The operators will know that H2 gas is a manual alignment since addition is a routine operation. They will also be aware the N2 gas is available. The operator may assume that N2 is provided by an automatic regulator to maintain a minimum pressure. The candidate may incorrectly assume that given that NPSH is maintained by a combination of level and pressure in the MU Tank, there would be an automatic trip of the Makeup Pumps on low pressure like there is for MU Tank Low Level. This is incorrect.

Sys #	System	Category	KA Statement
004	CVCS	Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following:	Minimum level of VCT (Make-up Tank)

K/A# K4.12 **K/A Importance:** 3.1 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02512, R19, Attachment 9, System Description SD-048 R5 Page 2-26

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content**: 41.7

Tier being Tested: ES401 Tier 2/Group 1 (RO)

30. Plant Conditions:
    - The plant is mode 5 following Reactor Refueling
    - Decay Heat Removal Loop 2 in service
    - All Reactor Coolant Pumps off
    - Preparations are in progress start Reactor Coolant Pumps and to heat up to Mode 2 to perform a Reactor Startup

    An RCS boron reduction of 100 ppm is requested by Reactor Engineering to prepare for the approach to criticality.

    Which of the following is the **minimum** flow conditions to perform the requested RCS Boron reduction?

A. At least one Reactor Coolant Pump is in operation in **each** RCS loop.

B. At least **one** Reactor Coolant Pump is in operation.

C. DHR flow greater than or equal to 2800 gallons per minute.

D. DHR flow greater than or equal to 1350 gallons per minute.

Answer: C

Explanation/Justification: The RCS flowrate through the core shall be ≥2800 gpm whenever a reduction in RCS boron concentration is being made, in all modes. The minimum flowrate of 2800 gpm provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual throughout the RCS and in the core during boron concentration reductions. The reactivity change rate associated with boron concentration reduction will be within the capability for operator recognition and control. This is also the flowrate stipulated in the Technical Specification Bases for having a Decay Heat Removal Loop in operation (SR 3.4.6.1, SR 3.4.7.1, SR 3.4.8.1, SR 3.9.4.1, SR 3.9.5.1)

A. Incorrect because this plant condition would not be the minimum flow required by the Controlling procedure DB-OP-06001, Limit and Precaution 2.1.1 for an RCS boron concentration reduction and DHR related Technical Specifications. – Plausible because 2 RCPs in service would provide good mixing through both RCS Loops however this flow would be significantly higher than the minimum flow asked by the question and single RCP operation in a loop is not permitted.

B. Incorrect because this plant condition would not be the minimum flow required by the Controlling procedure DB-OP-06001, Limit and Precaution 2.1.1 for an RCS boron concentration reduction and DHR related Technical Specifications – Plausible because a single RCP in service would provide good mixing through both RCS Loops via backflow through the idle RCS Loop. However, this flow would be significantly higher than the minimum flow asked by the question and single RCP operation would not be permitted due to NPSH concerns.

C. CORRECT – As noted in explanation above. This is the minimum RCS flow required for a reduction in RCS Boron Concentration.

D. Incorrect because this plant condition would not be the minimum flow required by the Controlling procedure DB-OP-06001, Limit and Precaution 2.1.1 for an RCS boron concentration reduction and DHR related Technical Specifications – Plausible because this flowrate is the minimum LPI Injection flowrate for a number of post LOCA evolutions including isolation of Core Flood tanks, stopping Makeup Pumps, stopping HPI Pumps, and balancing LPI Injection flows. This is number a candidate may select based on use in the LPI system if the candidate is unaware additional flow is required to reduce RCS boron.

Sys #	System	Category	KA Statement
005	RHR	Knowledge of the operational implications of the following concepts as they apply the RHRS:	Dilution and boration considerations

K/A# K5.09 **K/A Importance** 3.2 **Exam Level:** RO

References provided to Candidate None **Technical References:** DB-OP-06001 R24 L&P 2.1.1

Question Source: New **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.5 / 45.7)

Objective: OPS-GOP-434-03K

Abbreviation – ppm = parts per million

Tier being Tested: ES401 Tier 2/Group 1 (RO)

31. The reactor was operating at 100% power.

A Loss of Coolant event has occurred.

Current Plant Conditions:

- Reactor Coolant System pressure is at 100 psig and stable
- Borated Water Storage Tank (BWST) Level is 30 feet and lowering
- Low Pressure Injection (LPI) Pump 1 failed to auto start and will not start manually

Based on current plant conditions, which of the following describes the reason Low Pressure Injection Pumps discharge flowpaths are cross connected as directed by DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture?

Providing flow through both LPI injection lines is established to_______________.

A. increase the likelihood that LPI flow is providing core cooling if one of the injection lines is faulted

B. provide a required suction source for Makeup Pump 1 when the LPI Pump suctions are transferred to the Emergency Sump

C. prevent loss of ALL core injection flow when the LPI Pump suctions are transferred to the Emergency Sump if LOCA location is Core Flood 2 Line break

D. provide Long Term Boron Dilution via the normal (opening DH11 and DH12) flowpath

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Answer: A

Explanation/Justification: KA match because it tests the candidate's knowledge related to operation of the ECCS Pumps (LPI). USAR Section 6.3.2.11, Reliability Considerations, describes cross-connecting LPI injection lines. Providing flow through both LPI injection lines increases the likelihood that the LPI flow is providing Core cooling. Using two injection lines and balancing flow would insure at least 1000 gpm flow to the Core if one of the injection lines is faulted.

A. CORRECT – Providing flow through both LPI injection lines increases the likelihood that the LPI flow is providing Core cooling. Using two injection lines and balancing flow would insure at least 1000 gpm flow to the Core if one of the injection lines is faulted.

B. Incorrect because MU Pumps are stopped prior to transfer of ECCS Pump Suctions to the Containment Emergency Sump – Plausible because cross connecting could provide a suction source for Makeup Pump 1 once the BWST is depleted. At current BWST level, Makeup Pump 1 has a suction source. Incorrect because the MUPs are stopped prior to transfer to the Emergency Sump.

C. Incorrect because ALL Core injection flow would not be lost if the transfer to the Containment Emergency Sump was performed – Plausible because cross connecting would provide additional flow to the core for a train 2 core flood line break. Adequate cooling would still exist via the Train 2 HPI Pump operating in Piggyback from LPI Pump 2 even during a Train 2 Core Flood Line Break .

D. Incorrect because this flowpath for long term boron dilution is not allowed by the Controlling procedure - Plausible because Long Term Boron Dilution normal flowpath would be available if cross connected, however this flow path is not allowed by procedure. (OP2000 Att 12)

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

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

References provided to Candidate: None **Technical References:** Bases and Deviation Document for DB-OP-02000 R22 pg. 502

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Question Source:	NEW	**Level Of Difficulty: (1-5)**	3
Question Cognitive Level:	HI	**10 CFR Part 55 Content**: 41.7	CFR: 41.7 / 45.7
Objective:	OPS-GOP-304-05K		
Tier being Tested:	ES401 Tier 2/Group 1 (RO)		

32. The plant is operating at 100% power.

Which ONE of the following describes the initial Quench Tank (QT) response to venting the Pressurizer to the QT?

Venting the Pressurizer to the Quench Tank will cause Quench Tank __________.

A. level to lower AND pressure to rise

B. BOTH level and pressure to rise

C. ONLY pressure to rise

D. level to rise AND pressure to lower

Answer: B

Explanation/Justification: The question is a KA match because is tests the candidate's ability to predict the changes in Quench Tank Parameters when venting from the Pressurizer to the Quench Tank. Anticipating in Quench Tank parameters will allow the operator to confirm the expected system response is occurring and allow the QT level to be maintained within required limits. The Pressurizer can have a steam bubble (normal operation) or a Nitrogen Bubble (shutdown operations). Venting is performed both at power and during the shutdown condition especially when drawing a steam bubble in the Pressurizer. Venting nitrogen will only cause QT pressure to rise since only gas is carried over. Venting steam will cause temperature and level to rise as the steam is condensed in the QT.

A. Incorrect because the level in the Quench Tank would rise as Steam is quenched – Plausible because steam venting will cause QT pressure and level to rise. The candidate may select this response if they assume that the quenching action of the system causes indicated level to lower.

B. CORRECT- Venting steam will cause temperature and level to rise as the steam is condensed in the Quench Tank. This response provides the expected response to quenching steam from the Pressurizer.

C. Incorrect because level will rise in addition to pressure – Plausible because steam venting will cause QT pressure to rise. It is incorrect because both level and pressure will rise.

D. Incorrect because pressure will rise as steam is quenched – Plausible because steam venting will cause QT level to increase and the candidate may assume as that as stem is quenched, Quench Tank Pressure would lower. This incorrect assessment may be driven by Candidate knowledge that RCS pressure lowers when the Pressurizer spray valve is opened and steam in the Pressurizer is quenched. This is incorrect for the quench tank.

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

K/A# A1.01 **K/A Importance** 2.9 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-06003 R36, Section 3.1 Pressurizer Startup. Note 3.1.21

Question Source: Bank Question 37204 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.5 / 45.5)

Objective: OPS-SYS-104-14K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

33. The Reactor is operating at 100% power.

The following indications are noted:

- Annunciator 4-4-C HOT LEG PRESS LO is in alarm
- Lowering RCS Pressure
- Quench Tank temperature is rising

Based on the indications noted,

(1) The __________ is open,

AND

(2) _________ to mitigate this event.

A. (1) Spray Valve
(2) Close RC10, PZR SPRAY BLOCK Valve

B. (1) Spray Valve
(2) Allow RCS Pressure to lower, until the Spray Valve fully closes

C. (1) PORV
(2) Close RC11, PORV BLOCK Valve

D. (1) PORV
(2) Allow RCS Pressure to lower, until the PORV fully closes

Answer: C

Explanation/Justification: This is a KA Match because if tests the candidate knowledge of the expected indications for an open PORV and then the action to mitigate the event. All 4 events are related to the Pressurizer. Both the PORV and the Spray valve will not cause a change in Containment conditions (until quench tank rupture disc blows) but of those two, only the PORV will cause a rise in QT temperature.

A. Incorrect
(1) Incorrect because an open PZR Spray valve will not cause a change in Quench Tank parameters. Plausible because an open Spray valve will cause RCS Pressure to lower.
(2) Incorrect because closing the Pressurizer Spray Block Valve would not mitigate these events. The Spray valve is not open and closing the block would not mitigate the event.

B. Incorrect
(1) Incorrect because an open PZR Spray valve will not cause a change in Quench Tank parameters. Plausible because an open Spray valve will cause RCS Pressure to lower.
(2) Incorrect because lowing RCS Pressure would not mitigate these events. The Spray valve is not open. Lowering RCS pressure will not mitigate the event.

C. CORRECT –
(1) Correct because with the PORV open, all of the indications provided would the true. The PORV is connected to the Quench Tank. Opening of the PORV would cause RCS Pressure to lower and quench tank temperature to rise.
(2) Correct because the procedurally directed action for an open PORV is provided.

D. Incorrect –

(1) This portion of the response is correct. With the PORV open, all of the indications provided would the true. The PORV is connected to the Quench Tank. Opening of the PORV would cause RCS Pressure to lower and quench tank temperature to rise.

(2) This portion of the response is Incorrect. The correct action for an open PORV is not provided. The candidate may incorrectly select this response based on knowledge that lowering pressure may reseat a leaking safety valve. This is incorrect because this is not action directed by the controlling procedure DB-OP-02513.

Sys #	System	Category	KA Statement
007	Pressurizer	Ability to (a) predict the impacts of the following malfunctions or operations on the P S; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Stuck-open PORV or code safety.

K/A# A2.01 **K/A Importance** 3.9 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02513 R13, PZR Malfunctions Section 2, Symptoms, Section 4.4, Supplemental Actions
DB-OP-02004 R13, Annunciator Panel 4 Alarms (4-4-C)

Question Source: NEW **Level Of Difficulty: (1-5)** 3

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

Objective: OPS-GOP-113-04K

Abbreviations – PORV = Power Operated Relief Valve – At DB, PORVs are only used on the Pressurizer, not the SGs.

Tier being Tested: ES401 Tier 2/Group 1 (RO)

34. Plant Conditions:
    - The plant is operating at 100% power
    - Component Cooling Water Pump 1 in service
    - Component Cooling Water Pump 2 in standby

    Annunciator 11-4-B CCW PMP 1 FLOW LO Alarms.

    Which ONE of the following will occur if Component Cooling Water (CCW) System flow lowers to less than 1000 gpm?

A. The standby CCW pump receives a start signal.

B. CC1328, CRDC Booster pump suction valves, will close.

C. CC1495, Aux Bldg non-essential header isolation valve, will close.

D. SW1424, Service Water to CCW Heat Exchanger 1 Outlet Valve will open.

Answer: A

Explanation/Justification: Davis-Besse does not have CCW flow indicators in the Control Room. There are flow switches that cause a Flow Low alarm and Standby Pump to start under low flow from the operating pump. At DB, normally only one train of CCW is in service and a second train is in standby. The operating train provides cooling for all CCW loads, Essential and Non-Essential. Under Safety conditions, both trains will operate and provide cooling to safety related loads. A low flow alarm will occur at 3000 gpm and at 1000 gpm, the standby CCW Pump will get a start signal.

A. CORRECT – A flow switch will start the standby CCW Pump at a low flow of 1000 gpm.
B. Incorrect because this action does not occur as CCW flow lowers – Plausible because CC 1328 does have an auto closure signal, but it is on a SFAS Actuation signal, not low flow on the operating CCW Train.
C. Incorrect because this action does not occur as CCW flow lowers – Plausible because the non-essential header isolation CC1495 does have an auto closure signal, but it is on a low CCW Surge Tank Level or an SFAS Actuation signal, not low flow on the operating CCW Train.
D. Incorrect because this action does not occur as CCW flow lowers – Plausible because SW 1424 would respond in this scenario. It is expected that it would close, NOT open since the heat load is removed.

Sys #	System	Category	KA Statement
008	CCW	Ability to monitor automatic operation of the CCWS, including:	All flow rate indications and the ability to evaluate the performance of this closed-cycle cooling system

K/A# A3.03 **K/A Importance** 3.0 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02523 R13, CCW Malfunctions Attachment 9 and System Description SD-016, R06, Sections 2.5.2.2 and Section 2.5.2.5

Question Source: BANK 37601 **Level Of Difficulty: (1-5)** 2.5

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5)

Objective: OPS-GOP-123-02K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

35. The plant is at 100% power when the selected RCS narrow range pressure instrument fails low.

Which ONE of the following identifies the expected plant response and operator actions directed by DB-OP-02513, Pressurizer Malfunctions to mitigate this event?

(1) Actual RCS pressure will _________.

AND

(2) Manual control of the Pressurizer ________.

A. (1) rise
(2) heaters are required to prevent a reactor trip on HIGH RCS pressure

B. (1) rise
(2) Spray valve AND the PORV is required to prevent a reactor trip on HIGH RCS pressure

C. (1) lower
(2) heaters are required to prevent a reactor trip on LOW RCS pressure

D. (1) lower
(2) Spray valve AND the PORV is required to prevent a reactor trip on LOW RCS pressure

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Answer: A

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the control scheme for Reactor Coolant Pressure and how a failure of that pressure input would affect Pressurizer Heaters. At DB, an RCS Pressure instrument is selected for pressure control. In response to pressure input fail low malfunction, all pressurizer heaters would energize causing actual RCS pressure to rise. Normally, this would be mitigated by action of the PZR Spray valve, which would overcome the heat input from the pressurizer heaters, but the Spray valve input is failed low and will not open. If the PZR heaters are allowed to remain in automatic, actual RCS Pressure would rise causing a Reactor Trip on High RCS Pressure.

A. CORRECT-
(1) Correct - In response to this malfunction, all pressurizer heaters would energize causing actual RCS pressure to rise. Normally, this would be mitigated by action of the PZR Spray valve, which would overcome the heat input from the pressurizer heaters, but the Spray valve input is failed low and will not open.
(2) Correct - If the PZR heaters are allowed to remain in automatic, actual RCS Pressure would rise causing a Reactor Trip on High RCS Pressure. Manual control of the Pressurizer Heaters can prevent this reactor trip.

B. Incorrect
(1) Correct because actual RCS Pressure will rise.
(2) Incorrect because the PORV does not require manual control to prevent a reactor trip and is NOT directed per DB-OP-02513. Plausible because the input pressure for operating the spray valve and PORV is failed, but not in a direction that would require manual control of either valve to prevent a reactor trip. DB-OP-02513 step 4.1.2 does allow manual operation of the Spray valve to maintain RCS Pressure.

C. Incorrect

(1) Incorrect because the RCS Pressure response is the opposite of the expected response for a low-pressure input – Plausible because indicated RCS Pressure will lower and all pressurizer heaters will energize.

(2) Correct - In response to this malfunction, all pressurizer heaters would energize causing actual RCS pressure to rise. Normally, this would be mitigated by action of the PZR Spray valve, which would overcome the heat input from the pressurizer heaters, but the Spray valve input is failed low and will not open. Manual operating of the Pressurizer Heaters would be required to prevent a Reactor Trip.

D. Incorrect

(1) Incorrect because the RCS Pressure response is the opposite of the expected response for a low pressure input – Plausible because indicated RCS Pressure will lower.

(2) Incorrect because Automatic operation of the PORV and the Spray Valve is lost. Plausible because improper input signal can cause the PORV and Spray valve to open. The failure provided in the question is one that prevent opening, not causing the valves to open. If control of these valves is desired, that control would have to be manual. Either of these choices will reduce actual RCS Pressure however, manual control of the PORV AND Spray valve is not required to prevent a reactor trip.

Sys #	**System**	**Category**	**KA Statement**
010	Pressurizer Pressure Control	Ability to manually operate and/or monitor in the control room:	PZR heaters

K/A# A4.02 **K/A Importance** 3.6 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02513 R13, Symptoms Section 2.1 and Supplemental Actions Section 4.1

Question Source: 2011 CR3 NRC Exam Q16 Modified for DB **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 to 45.8)

Objective: OPS-GOP-113-04K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

36. The Plant is in Mode 1.

In accordance with Technical Specifications, which ONE of the following conditions requires action within **15 minutes** to remain in compliance with Technical Specifications requirements?

A. Pressurizer Level is greater than 228 inches.

B. One Pressurizer Code Safety Valve setpoint is set greater than 2525 psig.

C. No power is available to the Pressurizer Power Operated Relief Valve.

D. The Block Valve for the Pressurizer Power Operated Relief Valve is closed.

Answer: B

Explanation/Justification: This Question is a KA match because it tests the Candidates knowledge of TS Limiting Conditions for Operations related to the Pressurizer including the Code Safety Valves.

A. Incorrect because this is TS 3.4.9 Cond. A which allows 1 hour to restore Pressurizer level to less than 228 inches.– Plausible because when this condition is encountered in the simulator, the candidates take prompt action to restore Pressurizer Level to within limits.

B. CORRECT – Per Technical Specifications Pressurizer Safety Valves to be Operable requires a setting of less than or equal to 2525 psig. A setpoint greater than 2525 renders the valve inoperable. Action is required within 15 minutes per TS 3.4.10 Condition A.

C. Incorrect TS allows 1 hours to complete the required action – Plausible since this condition renders the PORV inoperable and required action within one hour to close the PORV Block valve per TS 3.4.11 Condition B to remove power from the block valve when the PORV inoperable.

D. Incorrect because action is required within one hour for this condition– Plausible since this condition would render the PORV inoperable and requires action per TS 3.4.11 Condition B within one hour to remove power from block valve when the PORV is inoperable.

Sys #	System	Category	KA Statement
010	Pressurizer Pressure Control	Generic	Knowledge of limiting conditions for operations and safety limits.

K/A# G2.2.22 **K/A Importance** 4.0 **Exam Level:** RO

References provided to Candidate None **Technical References:** TS 3.4.10 Condition A

Question Source: DB 2013 NRC Exam Q35 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.5 / 43.2 / 45.2)

Objective: OPS-GOP-434-04K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

37. The plant is operating at 100% power with all ICS Hand/Auto station in a normal lineup.

    Which of the following describes the effect a reactor trip has on the Integrated Control System from this lineup?

    Assume all systems respond to the Reactor Trip as designed.

    No additional malfunctions occur.

A. Normal Turbine Bypass Control setpoint is shifted from 870 psig to 1025 psig.

B. Normal Turbine Bypass Control setpoint is shifted from 880 psig to 1025 psig.

C. Atmospheric Vent Valve Control setpoint is shifted from 920 psig to 995 psig.

D. Atmospheric Vent Valve Control setpoint remains at 1025 psig.

Answer: D

Explanation/Justification: This question is a KA Match because it tests the candidate's knowledge of the Integrated Control System control of the Atmospheric Vent Valves (steam dumps) following a reactor trip signal from the Reactor Protective System. The AVVs have a normal 1025 psig setpoint that they respond to if the Turbine Bypass Valves are available. If the Turbine Bypass valves are not available, then the control logic for the TBVs are used to provide control signals for the AVVs. In this case, no malfunctions are presented that would cause the TBV control logic to be transferred to the AVVs. As a result, the AVV setpoint remains at 1025 psig.

A. Incorrect because the value provided for the controlling pressure is provided to the AVVs, not the TBVs – Plausible because the TBV setpoint does shift with a bias applied when the reactor trips. The pressure provided is the pressure setpoint for the AVVs, not the TBVs. In addition, the normal setpoint for Turbine Header Pressure when at 100% power is 880 psig, not 870 psig. Header Pressure setpoint is adjusted up from the no load 870 setpoint to the normal at power setpoint of 880 psig during the power escalation to 100%.

B. Incorrect because the value provided for the controlling pressure is provided to the AVVs, not the TBVs – Plausible because the TBV setpoint does shift with a bias applied when the reactor trips. The pressure provided is the pressure setpoint for the AVVs, not the TBVs. The turbine header pressures setpoint at 100% power is correct.

C. Incorrect because the setpoint for AVV control is not the pressure used by ICS post trip with TBVs available – Plausible because the normal +50 psig turbine header pressure bias would be applied to the AVVs if the TBVs were not available. In addition, the control press of 995 psig is the normal post trip control pressure if the reactor trip occurred from 100%, but this signal would go to the TBVs, not the AVVs since no information is provided that would make the AVVs unavailable.

D. CORRECT – The AVVs have a normal 1025 psig setpoint that they respond to if the Turbine Bypass Valves are available. If the Turbine Bypass valves are not available, then the control logic for the TBVs is used to provide control signals for the AVVs. In this case, no malfunctions are presented that would cause the TBV control logic to be transferred to the AVVs. As a result, the AVV setpoint remains at 1025 psig.

Sys #	System	Category	KA Statement
012	Reactor Protection	Knowledge of the physical connections and/or cause effect relationship between RPS and the following systems:	SDS-Steam Dump System:

K/A# K1.07 **K/A Importance** 3.2 **Exam Level:** RO

References provided to Candidate None **Technical References:** System Description SD 45 R08, Integrated Control System Page 2-12.

Question Source: NEW **Level Of Difficulty: (1-5)** 3

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

Objective: OPS-SYS-512-06K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

38. The following plant conditions exist:
- The plant is operating at 80% power.
- All plant systems are operating normally.
- A loss of power to 120 volt AC Essential Power Panel Y2 has just occurred.

What is the expected effect on the Control Rod Trip Breakers?

A. Control Rod Drive Trip Breaker A ONLY will open.

B. Control Rod Drive Trip Breaker B ONLY will open.

C. ALL Control Rod Drive Trip Breakers will open.

D. NO Control Rod Drive Trip Breakers will open.

Answer: A

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the power supplies for the Reactor Protective System and the impact that the loss of a single power supply (Y2) for the Reactor Protective System. The candidate must have knowledge of which Essential 120 vac instrument system supplies which RPS Channel 2 and how those channels are inter connected with their respective Control Rod Drive System Trip Breakers.

A. CORRECT – RPS Channel 2 is supplied power from Y2 and will be lost when Y2 power is lost. As a result, the associated CRD Trip Breaker A will open on the loss of power. RPS Channel 2 supplies holding power to CRD Trip Breaker A. This is the only correct response of the 4 provided.

B. Incorrect because the response provides the wrong CRD Trip Breaker that will open – Plausible if the Candidate knows that RPS Channel 2 is supplied from Y2 but then incorrectly assumes that RPS Channel 2 is associated with CRD Trip Breaker B. RPS Channel 2 operates CRD Trip Breaker A, not Trip Breaker B.

C. Incorrect because All CRD Breakers will not open under the given conditions.
Plausible because a deenergized channel will open contacts in the other three breakers. The breakers need an additional input for them to open.

D. Incorrect because a CRD Breaker will open for the conditions provided.
Plausible because a single signal is sent to all 4 RPS channels. The candidate may assume that without a second trip signal, no CRD Trip Breakers will open.

Sys #	System	Category	KA Statement
012	RPS	Knowledge of bus power supplies to the following:	RPS channels, components, and interconnections

K/A# K2.01 **K/A Importance** 3.3 **Exam Level:** RO

References provided to Candidate: None

Technical References: DB-OP-06403 R26, Reactor Protective System and Nuclear Instrument System Operating Procedure ATTACHMENT 4: TYPICAL SIMPLIFIED SCHEMATIC OF A RPS CHANNEL (2).

Question Source: Exam Bank – 295527

Level Of Difficulty: (1-5) 3.5

Question Cognitive Level: LO

10 CFR Part 55 Content: (CFR: 41.7)

Objective: OPS-SYS-504-09K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

39. The plant was operating at 100% power. A large Break Loss of Coolant Accident occurs.

    Both High Pressure Injection (HPI) and both Low Pressure Injection (LPI) pumps failed to start, all other SFAS components actuated as designed.

    What is the potential effect on the Reactor Fuel?

A. Insufficient Injection flow will cause Cladding temperatures to exceed design limits.

B. The operating Makeup Pump with suction from the BWST will maintain Fuel Cladding temperatures within design limits.

C. Core Flood Tanks will maintain Cladding temperature at subcooled conditions.

D. Rising Fuel temperatures will add positive reactivity, causing Fuel Cladding temperatures to exceed design limits.

Answer: A

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the design criteria for the Emergency Core Cooling System that are actuated by SFAS. USAR 3D.1.31 Criterion 35 - Emergency Core Cooling describes the 10 CFR 50.46 design criteria the Emergency Core Cooling Systems. The Low Pressure Injection and High Pressure Injection Systems are actuated by the SFAS System on a LOCA. A beyond design basis failure of the SFAS system could prevent the ECCS Design Criteria from being met.

A. CORRECT - USAR 3D.1.31 Criterion 35 - Emergency Core Cooling describes the 10 CFR 50.46 design criteria for the Emergency Core Cooling Systems. This is one of the fuel related criteria the ECCS Systems are designed to meet. The ECCS System design criteria would not be met if both HPI and LPI Pumps failed to start. As a result, Fuel Cladding Temperatures would exceed design limits for a Large Breaker LOCA.

B. Incorrect because the Makeup Pumps are designed for low volume, high pressure conditions and will not provide the necessary volume to prevent exceeding design limits as the only available injection source during a LB LOCA.
PLAUSIBLE because the Makeup Pumps will provide some core cooling. They are also used during MU/HPI/PORV cooling. In addition, once the BWST is depleted, the MU Pump would not be available to provide RCS inventory and Reactor Fuel Design limits would be exceeded.

C. Incorrect because the Core Flood Tanks have a limited volume and only provide cooling for fuel clad for early portions of the LB LOCA. Core Flood will not maintain subcooled fuel temperatures – Plausible because USAR 3D.1.31 Criterion 35 - Emergency Core Cooling describes the 10 CFR 50.46 design criteria for the Emergency Core Cooling Systems which includes the Core Flood Tanks.

D. Incorrect because the fuel temperature coefficient (doppler) would only add negative reactivity to the core.
PLAUSIBLE because fuel temperature will rise, but not for the reason given.

Sys #	System	Category	KA Statement
013	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:** USAR 3D.1.31 Criterion 35 - Emergency Core Cooling, 10 CFR 50.46
TS Basis 3.3.7

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.7 / 45.6)

Objective: OPS-SYS-303-01K, OPS-SYS-301-02K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

40. The following plant conditions exist:

Annunciators:
- FIRE OR RADIATION TRBL, 9-1-G, due to CTMT Wide Range REs
- SUBCOOL MARGIN LO, 4-1-B
- CTMT PRESS HI, 4-2-A
- CTMT NORM SUMP LVL HI, 4-3-A
- CRD TRIP CONFIRM, 8-1-A

Equipment Status:
- Both High Pressure Injection Pumps are running
- Both Emergency Diesel Generators' output breakers are open

Plant Parameters:
- Pressurizer level is off scale low
- RCS pressure is 1500 psig and lowering
- Containment pressure 24 psia and rising
- BWST Level is 20 feet

Which ONE of the following correctly describes the status of the Containment Air Coolers (CACs) and the Containment Spray (CS) Pumps for the above conditions?

(1) Two CACs running in __________.

AND

(2) Two CS Pumps __________.

A. (1) FAST
(2) RUNNING with their discharge valves THROTTLED

B. (1) FAST
(2) RUNNING with their discharge valves OPEN

C. (1) SLOW
(2) OFF with their discharge valves THROTTLED

D. (1) SLOW
(2) OFF with their discharge valves OPEN

Answer: D

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the effect a Containment Isolation signal from SFAS has on the two systems that provided Containment Cooling following a LOCA event – Containment Air Coolers (CACs) and the Containment Spray System. To predict the response of the system, the candidate must use the Containment and RCS conditions provided to determine the expected SFAS Actuation Level and once the actuation level is determined, the expected response of the two systems. For the conditions provided by the stem, SFAS Level 2 would be actuated (CTMT Spray Discharge Valves and CACs), while SFAS Level 4 (Spray Pumps) would not be actuated. In addition, the position of the CTMT Spray Discharge Valve has a throttle position that is actuated when the ECCS Pump Suctions are transferred from the BWST to the Containment Emergency Sump.

A. Incorrect

(1) Incorrect because Fast is the normal condition for CACs, not the SFAS Actuated Slow position – Plausible because the CAC's would be running, but would be running in slow speed, not fast. Fast would provide more cooling but may overload the CAC motors due to the higher density of the air in CTMT give steam/air mixture.

(2) CTMT Spray would not be spraying CTMT at this CTMT pressure. The CTMT Spray Discharge valves would be open, but the pumps would not be running, the opposite of the conditions provided by this response. The CTMT Spray Discharge valves open on an SFAS Level 2, but the Pumps don't start until an SFAS Level 4.

B. Incorrect

(1) Incorrect because Fast is the normal condition for CACs, not the SFAS Actuated Slow position – Plausible because the CAC's would be running, but would be running in slow speed, not fast. Fast would provide more cooling but may overload the CAC motors due to the higher density of the air in CTMT give steam/air mixture.

(2) Incorrect because CTMT Spray would not be spraying CTMT at the specified CTMT pressure. Plausible the CTMT Spray discharge valves would be full open, but the pumps would not be running. The valves move to the throttle position when the ECCS Pump suctions are transferred to the Emergency Sump.

C. Incorrect

(1) Correct because the Containment Air Coolers would be running in slow speed for the given plant conditions.

(2) Incorrect because the Containment Spray Discharge Valve would be full open for these plant conditions – Plausible because the two CTMT Spray Pumps would be off, however the CTMT Spray Discharge valves would be in the fully open position, not the throttled position. The valves move to the throttle position when the ECCS Pump suctions are transferred to the Emergency Sump.

D. CORRECT –

(1) Correct - For the given conditions, an SFAS Level 2 and 3 trips are expected based on Low RCS Pressure and High Containment Pressure. CACs would be running is slow speed for the plant conditions provided.

(2) Correct - Containment Spray Discharge Valves would be in the open position based on SFAS Level 2 and BWST Level still at 20 feet (transfer to Emergency Sump has not occurred). Containment Spray Pumps will not start until an SFAS Level 4 actuation on High Containment Pressure of approximately 40 psia.

Sys #	System	Category	KA Statement
O22	Containment Cooling System	Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following:	Automatic containment isolation

K/A# K4.03 **K/A Importance** 3.6 **Exam Level: RO**

References provided to Candidate None **Technical References:** DB-OP-02000 R32, Table 2 SFAS L2

Question Source: New **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** 41.7

Objective: OPS-GOP-302-01K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

41. The plant has experienced a Containment Design Basis Loss of Coolant Accident.

Following the transfer of Low Pressure Injection (LPI) Suction to the Emergency Sump, a small rise in Containment pressure is noted.

Which of the following describes the reason for this pressure rise?

Heat removal from Containment is reduced because __________.

A. LPI and Spray discharge temperatures rise significantly when suction is transferred to the sump

B. throttling of the Containment Spray Discharge Valve lowers the heat removal from Spray

C. stopping High Pressure Injection Pump for the transfer lowers core cooling flow

D. establishing Long Term Boron Dilution after the transfer lowers flow through the Decay Heat Cooler

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Answer: A

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the expected response of Containment Pressure following a Design Basis Loss of Coolant Accident related to operation of the Containment Spray System. As described in the UFSAR. Section 6.2.1.3.1, when Containment Spray aligned to the BWST, relatively cool water is sprayed into Containment which causes a reduction in Containment Pressure. When the suction of the Containment Spray Pumps is transferred to the Emergency Sump, relatively warm Emergency Sump water is then sprayed into Containment. This will cause a rise in Containment pressure for approximately 30 minutes.

A. CORRECT- See UFSAR R30 Section 6.2.1.3.2 page 6.2-11 Long-term Containment Analysis. Containment pressure rises for the first 2000 seconds (half hour) after swap to sump.

B. Incorrect because the cause of the rising trend is not due to lower heat removal by CTMT Spray – per UFSAR Section 6.2.1.3.2 page 6.2-11 Long-term Containment Analysis, the majority of heat removal from Containment during recirculation is performed by the CAC and the Decay Heat Removal Cooler, so throttling of spray flow has a minor effect. Plausible because Containment Spray flow is lowered by throttling.

C. Incorrect because HPI flow is not removing heat from Containment – Plausible because HPI is stopped prior to swap to sump. See DB-OP-02000 R28 steps 10.12 and 10.13.

D. Incorrect establishing Long Term Boron Dilution is directed after transferring ECCS Suctions to the Emergency Sump, but this not affect the cooling of containment – Plausible because Long Term Boron Dilution is established following swap to sump. DB-OP-02000 R28 steps 10.13 and 10.17.

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

K/A# A1.01 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** UFSAR R30 Section 6.2.1.3.2 page 6.2-11,

Question Source: DB 2015 Q42 **Level Of Difficulty: (1-5)** 3

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

Objective: OPS-SYS-306-02K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

42. A Reactor Shutdown is in progress due to a Steam Generator Tube Rupture greater than the capacity of available Makeup Pumps.

    Which of the following actions is directed during the Shutdown to minimize offsite releases and align Turbine Building Systems to handle contaminated condensate?

A. Steam Jet Air Ejector Drains are aligned to the Turbine Sump.

B. Moisture Separator Drain Tanks are aligned to Condenser.

C. CST Hotwell Makeup Control Valve is aligned to the Condenser.

D. Vacuum System Vent Filter is removed from service.

Answer: B

Explanation/Justification: In order to test the Main and Reheat Steam System (MRSS) during a LOCA, a SGTR was selected since the SGTR EOP provides actions for the MRSS system while a LOCA from RCS piping does not require specific direction for the Main and Reheat Steam Systems. This is a KA match because the procedure directs realignment of the MSR Drain Tanks from the Deaerator to the Condenser to allow the Condensate Polishers to clean up fission products from the Reactor Coolant System that reach the Steam Generator during a SGTR.

A. Incorrect because the alignment is the opposite of that directed by the controlling procedure – Plausible because realignment of the SJAE drains is directed by DB-OP-02000 using DB-OP-02531, but they are directed from the normal turbine building sump to the condenser to allow the Condensate Polishers to remove fission products.

B. CORRECT – DB-OP-02000 directs performing the Attachments from DB-OP-02531, SGTL to realign the MSR Drain Tanks from the normal Deaerator flowpath to the Condenser will allow the Condensate Polishers to clean up fission products from the Reactor Coolant System that reach the Steam Generator during a SGTR.

C. Incorrect because alignment of the Condensate Storage Tanks (CST) Hotwell Makeup Control Valve to the Hotwell is not directed by procedure.
Plausible because draining the Hotwell to the CSTs as necessary to prevent overfill of the Hotwell is allowed after the CSTs have been drained. It could also be assumed that the CSTs are drained to the hotwell first for dilution prior to draining the hotwell to the East or West Condenser Pit Sumps.

D. Incorrect because the vent filter will be placed in service for a SGTR – Plausible because the Vacuum System vent filter is not normally in service.

Sys #	System	Category	KA Statement
039	Main and Reheat Steam	Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Flow paths of steam during a LOCA

K/A# A2.01 **K/A Importance** 3.1 **Exam Level;** RO

References provided to Candidate: None

Technical References: DB-OP-02000 R32 Step 8.8
DB-OP-02531 R24 SG Tube Leak Attachment 4 CONTROL OF SECONDARY CONTAMINATION AND OFFSITE RELEASES

Question Source: New

Level Of Difficulty: (1-5) 3

Question Cognitive Level: LO

10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Objective: OPS-GOP-131-11K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

43. The plant is operating at 100% power. A steam leak that is a personnel safety hazard develops in the Second Stage Reheat line #1 Moisture Separator.

Which ONE of the following actions is directed by DB-OP-02525, Steam Leaks to mitigate this event?

A. Close MS101, SG1 MAIN STEAM ISOLATION VALVE

B. Close MS100, SG2 MAIN STEAM ISOLATION VALVE

C. Trip the Reactor and Initiate ONLY SFRCS

D. Trip the Reactor and Initiate AND Isolate SFRCS

Answer: D

Explanation/Justification: This is a match for the KA because a steam leak that is a personnel safety hazard must be isolated. This direction is provided by Abnormal Operating Procedure DB-OP-02525, Steam Leaks. The Immediate operator actions for this condition are directed by Step 3.2 which directs Tripping the Reactor and then Initiating and Isolating the Steam Feed Rupture Control System. The SFRCS has two modes that it can be manually actuated from the Control Room front panel, (1) Initiate AND (2) initiate and isolate. The initiate and isolate mode will close both MSIVs and send a trip signal to the Main Turbine these actions will terminate steam flow to the Moisture Separator Reheater.

A. Incorrect because this not the directed action for a steam leak that is a personnel safety hazard – Plausible because the steam leak in on the #1 Moisture Separator. The candidate may incorrectly assume this would isolate steam flow to the leak. It would not because this action alone would not automatically trip the Main Turbine. With the Main Turbine still in operation the steam chest will cross connect both steam lines, Closing the SG1 MSIV would not terminate the event and is not the action directed by DB-OP-02525, Steam Leaks.

B. Incorrect because this not the directed action for a steam leak that is a personnel safety hazard – Plausible because the steam leak in on the 2nd stage of #1 Moisture Separator. The candidate may incorrectly assume this would isolate steam flow to the leak. It would not because this action alone would not automatically trip the Main Turbine. With the Main Turbine still in operation the steam chest will cross connect both steam lines, Closing the SG2 MSIV would not terminate the event and is not the action directed by DB-OP-02525, Steam Leaks.

C. Incorrect because this not the directed action for a steam leak that is a personnel safety hazard – Plausible because a Reactor Trip is required for steam leak that is a personnel safety hazard. The candidate may incorrectly select this event because tripping the reactor would cause the turbine to trip and close the Main Turbine Stop and Control Valves stopping steam flow through the turbine, however the leak is in the reheat steam line, not the steam flowing through the Main Turbine.

D. CORRECT - This direction is provided by Abnormal Operating Procedure DB-OP-02525, Steam Leaks. The Immediate operator actions for this condition are directed by Step 3.2 which directs Tripping the Reactor and then Initiating and Isolating the Steam Feed Rupture Control System. Initiate and Isolate SFRCS will close both MSIVs and send a trip signal to the Main Turbine. These actions will terminate steam flow to the Moisture Separator Reheater as directed by the procedure.

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

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

References provided to Candidate: None **Technical References:** DB-OP-02525 R14, Steam Leaks, Step 3.2

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** 41.5 / 45.5

Objective: OPS-GOP-125-02K

Abbreviations – SFRCS = Steam Feedwater Rupture Control System

Tier being Tested: ES401 Tier 2/Group 1 (RO)

44. Plant conditions:
    - Plant startup is in progress
    - The reactor is at 50% rated power
    - Main Feedwater Pump (MFP) 1 is shutdown
    - Main Feedwater Pump (MFP) 2 is operating in AUTOMATIC
    - All Feedwater Control Valves are in AUTOMATIC
    - Integrated Control System (ICS) is in full AUTOMATIC mode

    Which ONE of the following describes Main Feedwater flow control by ICS if a reactor trip occurs from these plant conditions?

    (1) MFP 2 Speed is initially __________ by ICS.

    AND

    (2) Feedwater Startup Control Valves are initially positioned to __________.

A. (1) maintained at current speed
   (2) a target position until SG Low Levels Limits are reached OR a 2.5 minute timer expires

B. (1) maintained at current speed
   (2) closed until SG Levels are at Low Level Limits

C. (1) set to a target speed of 4600 RPM
   (2) a target position until SG Low Levels Limits are reached OR a 2.5 minute timer expires

D. (1) set to a target speed of 4600 RPM
   (2) closed until SG Levels are at Low Level Limits

Answer: C

Explanation/Justification: This question is a KA match because the plant is increasing power (plant startup) and Main Feedwater is not in a normal 100% power lineup (1 MFP in service). In addition, the response is different than it would be at a lower power level when Steam Generators are on low level limit control (less that approximately 28% power). Knowing the post trip response of the Main Feedwater System is important for the ability of the operator to determine the potential for an overheating or overcooling event after a reactor trip. Improper Feedwater response could lead to either event.

A. Incorrect
   (1) Incorrect because a Reactor Trip will alter the control scheme for Main Feedwater sending the MFP to target speed vice maintaining current speed – Plausible because this is the normal ICS control of Main Feedwater when the plant is operating at 50% power.
   (2) Correct - The response of the Startup Control Valves is correct for the plant conditions provided.

B. Incorrect

(1) Incorrect because a reactor trip will alter the control scheme for Main Feedwater sending the MFP to target speed vice maintaining current speed. Plausible because this is the normal ICS control of Main Feedwater when the plant is operating at 50% power.

(2) Incorrect - The Startup Valve response is also incorrect. While significantly less feedwater flow will be required following a reactor trip, the SUFW valves do not close. Plausible because the position of the valve changes based on a reactor trip. In addition, any trip with an SFRCS Isolation Actuation would cause the valves to close.

C. CORRECT-

(1) Correct - ICS will transfer MFP speed to a target value necessary to overcome elevated SG pressures following a Reactor Trip.

(2) Correct – The FW Start Up valves are initially positioned to a target position and then will transition to level error control that occurs after a 2.5 minute timer has expired OR Steam Generator levels reach low level limits (approximately 40 inches).

D. Incorrect

(1) Correct because because ICS will transfer MFP speed to a target value of 4600 RPM.

(2) Incorrect because the Startup Valve response is incorrect. While significantly less feedwater flow will be required following a reactor trip, the SUFW valves do not close. Plausible because the position of the valve changes based on a reactor trip. In addition, any trip with an SFRCS Isolation Actuation would cause the valves to close but remain closed when low level limits is reached.

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

K/A# A4.12 **K/A Importance** 2.9 **Exam Level: RO**

References provided to Candidate: None

Technical References: Bases and Deviation Document for DB-OP-02000 R22 pg. 465
SD-045 2.1.2.3.10

Question Source: Modified Exam Bank Q38076

Level Of Difficulty: (1-5) 3

Question Cognitive Level: HI

10 CFR Part 55 Content: (CFR: 41.7 / 45.5 to 45.8)

Objective: OPS-SYS-207-04K

Abbreviations – RPM = Revolutions per minute

Tier being Tested: ES401 Tier 2/Group 1 (RO)

45. The following plant conditions exist:
- A large break Loss of Coolant Accident (LOCA) has occurred
- High Pressure Injection (HPI)flow is 900 gpm/pump
- Low Pressure Injection (LPI) flow is 2000 gpm/line
- Auxiliary Feedwater (AFW) System is in operation
- Steam Generator (SG) pressures are 100 psig in each SG

Which ONE of the following statements is correct concerning the operation of the AFW System?

A. AFW flow must be increased to place the SG in wet layup.

B. AFW flow must be increased to match the increase in decay heat.

C. AFW must be kept running to provide SG Shell cooling.

D. AFW must be kept running to ensure adequate core cooling.

Answer: C

Explanation/Justification: KA requires knowledge of the operational implications of the relationship between AFW and RCS heat transfer. This question asks about controlling AFW (operational implications) and is related to RCS Heat Transfer. For the situation provided, ECCS injection flow is removing all core heat (based on 2 LPI injection points and flowrate and 4 HPI Injection points and flowrate). SG heat transfer for core cooling is not required but AFW flow will provide SG cooling minimizing the SG Tube to Shell Differential Temperatures protecting post LOCA containment integrity via SG tubes to Main Steam lines. Emergency Operating Procedure LOCA response directs maintaining AFW in service until SG pressure is 35 psig or AFW turbine speed lowers to 600 rpm. The AFW Pump Turbines are a load on the SG providing steam flow and cooling the SGs.

A. Incorrect because using AFW to place the SG in wet layup would cause carryover to the steam supply for the AFW pumps potentially causing turbine damage – Plausible because SG's are normally placed in Wet Layup when the Plant is shutdown and SG Feedwater is no longer required. In this situation, SG cooling is no longer required to cool the core, however filling to wet layup in with these plant conditions would cause carryover and water to the AFW Pump Turbines.

B. Incorrect because the provided HPI and LPI flows exceed flow required to remove core heat. SG cooling is not required for the plant conditions provided – Plausible because once required SG levels are reach, AFW is automatically throttled by the level control system to match decay heat levels. For a given AFW flow, if heat removal exceeds decay heat, less steam flow is required, and SG levels would rise, and the system would throttle to maintain SG level. If heat removal is less than decay heat, more steam flow is required, and SG levels would lower and the AFW system would throttle to maintain SG level. In addition, for a station blackout event when all SG level could be lost, direction is provided to match AFW flow to expected core decay heat production based on time after shutdown curves.

C. CORRECT - Following a Large Break Loss of Coolant Accident, Core cooling via the SGs is not required. AFPTs are left running until SG pressure decreases to 35 psig for SG heat removal to minimize SG tube to shell differential temperature limiting SG Tube stresses as directed by the controlling procedure, DB-OP-02000.

D. Incorrect because the provided HPI and LPI flows exceed flow required to remove core heat. SG cooling is not required for the plant conditions provided – Plausible because the candidate may not recognize that all core cooling is being provided by ECCS Pump Injection and flow out the large break. For small break LOCA, break flow alone may not provide adequate core cooling, requiring the use of AFW to provide core cooling as a portion of the core cooling.

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

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

References provided to Candidate: None **Technical References:** DB-OP-02000 R32 step 10.25 and related Bases and Deviation Document for that step.

Question Source: Exam bank 36938

Question Cognitive Level: HI

Level Of Difficulty: (1-5) 3 4

10 CFR Part 55 Content: (CFR: 41.5 / 45.7)

Objective: OPS-GOP-304-05K

Abbreviations: gpm = gallons per minute

Tier being Tested: ES401 Tier 2/Group 1 (RO)

46. Plant Conditions:
    - The reactor is operating at 100% power
    - DB-SC-03071, EDG 2 Monthly test is in progress
    - Emergency Diesel Generator (EDG) 2 is ready to be synchronized to Bus D1

    When you close breaker AD101 to parallel EDG 2 with Bus D1, which ONE of the following actions and reason for that action is required **FIRST** by DB-SC-03071, EDG 2 Monthly Test?

A. Turn DG 2 VOLT REGULATOR switch handle clockwise to raise EDG 2 voltage, to prevent an undervoltage condition on Bus D1.

B. Turn DG 2 SPD CTRL switch to LOWER, to prevent an overspeed trip of EDG 2.

C. Turn DG 2 SPD CTRL switch to RAISE, to prevent a reverse power trip of EDG 2.

D. Turn DG 2 VOLT REGULATOR switch handle counterclockwise to lower EDG 2 voltage, to prevent an overvoltage condition on Bus D1.

Answer: C

Explanation/Justification: This question is a KA Match because it tests the knowledge of the cause and effects when paralleling and Emergency Diesel Generator to an essential 4160 volt bus. As describe by DB-SC-03071, Caution 4.2.7 and 4.2.8 "When paralleling an Emergency Diesel Generator to the grid the kW meter will indicate power whether the machine is supplying power or is in a reverse power condition. The following precautions apply: As soon as the output breaker is closed, immediately go to RAISE on the Governor Speed Switch, DG 2 SPD CTRL, and establish a load of 300 kW on the machine If the kW meter responds in a decreasing direction, this indicates a reverse power situation. Continue to hold the Governor Speed Switch, DG 2 SPD CTRL, in RAISE to establish an initial load of 300 kW"

Failure to establish this initial load on the EDG may result in a reverse power trip of the EDG output breaker when EDG 2 is paralleled to the energized essential 4160 volt D1 bus.

A. Incorrect because adjustment of the voltage regulator is not intended to prevent an undervoltage condition on Bus D1 – Plausible because the voltage regulator may be adjusted after the EDG is synchronized to D1 bus by step 4.2.9, but this adjustment is after the initial load is applied to the EDG by step 4.2.8 and the voltage adjustment if for power factor, not to directly prevent an undervoltage condition on D1 bus. The candidate may select this incorrect response based on knowledge that a voltage adjustment may be required.

B. Incorrect because the direction of the speed control adjustment is incorrect – Plausible because the speed control switch will be adjusted after the EDG is synchronized to D1 bus by step 4.2.8, but this adjustment is to increase speed to establish 300 kW on the EDG, not to lower EDG speed. This action would likely result in a reverse power trip of the output breaker. The candidate may select this incorrect response based on knowledge that a speed control adjustment will be required.

C. CORRECT – As directed by DB-SC-03071, Caution 4.2.7 and 4.2.8 "When paralleling an Emergency Diesel Generator to the grid the kW meter will indicate power whether the machine is supplying power or is in a reverse power condition. The following precautions apply: As soon as the output breaker is closed, immediately go to RAISE on the Governor Speed Switch, DG 2 SPD CTRL, and establish a load of 300 kW on the machine If the kW meter responds in a decreasing direction, this indicates a reverse power situation. Continue to hold the Governor Speed Switch, DG 2 SPD CTRL, in RAISE to establish an initial load of 300 kW" .

D. Incorrect because this action is not the first directed after the EDG is paralleled with bus D1 – Plausible because the voltage regulator may be adjusted after the EDG is synchronized to D1 bus by step 4.2.9, but this adjustment is after the initial load is applied to the EDG by step 4.2.8 and the voltage adjustment if for power factor, not to directly prevent an undervoltage condition on D1 bus. The candidate may select this incorrect response based on knowledge that a voltage adjustment may be required.

Sys #	System	Category	KA Statement
062	AC Electrical Distribution System	K1. Knowledge of the physical connections and/or cause/effect relationships between the ac distribution system and the following systems:	ED/G

K/A#	K1.02	**K/A Importance**	4.1	**Exam Level**	RO
References provided to Candidate	None			**Technical References:**	DB-SC-03071 R40, EDG 2 Monthly Steps and Cautions for 4.2.7 and 4.2.8 on page 45.
Question Source:	Exam Bank Q37818			**Level Of Difficulty: (1-5)**	3
Question Cognitive Level:	HI			**10 CFR Part 55 Content:**	(CFR: 41.2 to 41.9)
Objective:	OPS-SYS-406-09K				
Tier being Tested:	ES401 Tier 2/Group 1 (RO)				

47. Which of the following are 250VDC loads supplied from DCMCC1?

1. High Pressure Injection Pump 1 DC Lube Oil Pump
2. Inverter YVA
3. Main Feedwater Pump 1 Turbine Emergency Bearing Oil Pump
4. Essential Distribution Panel D1P

A. 1 and 2

B. 2 and 3

C. 3 and 4

D. 1 and 4

Answer: B

Explanation/Justification: KA match because the question tests the Candidate's knowledge of major DC loads and the ability to distinguish between 250 volt loads and 125 volt loads.

A. Incorrect because the HPI Pump is a 125 volt DC load – Plausible because 2 is correct, Inverter YVA is powered from a 250 volt DC bus, but the HPI LO pumps are powered from essential 125 Volt DC Bus. The candidate may select this response if they don't know the HPI DC Lube oil pumps are 125 volt loads.

B. CORRECT – Both of these loads are powered from the 250 VDC bus - reference DB- OP-02537 page 34 step A.9

C. Incorrect because DIP is a 125 volt DC Load – Plausible because the MFW Pump EBOP is a 250 volt DC load, but essential Distribution panel D1P is a 125 volt DC load powered from DCMCC1. The candidate may select this response if they don't know that D1P is a 125 volt load.

D. Incorrect because both loads are 125 volt DC loads – Plausible because the HPI Lube Oil Pumps and distribution panel D1P are DC loads. The candidate may select this response if they don't know the HPI DC Lube oil pumps and D1P are 125 volt loads, not 250 volts loads.

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

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

References provided to Candidate None **Technical References:** E-1040A R11 DCMCC1 Loads

Question Source: Modified DB 2016 NRC Exam Q49 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: Low **10 CFR Part 55 Content:** 41.7

Objective: OPS-SYS-409-14K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

48. Initial conditions:
- Plant is in mode 1 at 100% power.
- Component Cooling Water (CCW) Pump 2 in service

A seismic event occurs resulting in:
- Loss of DC Bus D2P
- 1 minute later a Loss of off-site power occurs immediately followed by a Large Break LOCA in containment
- SFAS Levels 1-4 have actuated

All immediate actions for the reactor trip have been completed.

No other operator actions have been performed.

What is the correct status of pumps supplied by D1 Bus, 1 minute after the LBLOCA in containment?

A. All Required pumps for SFAS Levels 1-4 are running.

B. ONLY CCW Pump 2 is running.

C. ONLY CCW Pump 2 and Service Water (SW) Pump 2 are running.

D. No pumps are running.

Answer: D

Explanation/Justification: This question is a match for the KA based on it tests the candidate's knowledge of the effect a loss of DC control power will have on D1 Bus components that use DC Control power from Bus D2P.

A. Incorrect because EDG 2 will not start or load following the loss of D2P - Plausible if the student does not know D2P normally supplies the control power needed for all breakers on D1 Bus and the starting circuit for EDG 2. Without DC control power D1 load breakers fail as is and EDG 2 will not start.

B. Incorrect because EDG 2 will not start or load following the loss of D2P - Plausible if the student believes D2P only supplies control power for the load breakers on D1 (Similar to 13.8KV Bus B) and has no impact on EDG 2. If the student determines EDG 2 will start and energize D1 bus, but cannot operate the D1 load breakers, only the previously running pumps would be energized. Answer B could be chosen instead of answer C if the student believes the undervoltage load shedding capability of the SW pump would allow the breaker to open but not reclose.

C. Incorrect because EDG 2 will not start or load following the loss of D2P - Plausible if the student believes D2P only supplies control power for the load breakers on D1 and has no impact on EDG 2. If the student determines EDG 2 will start and energize D1 bus, but cannot operate the D1 load breakers, only the previously running pumps CCW and SW would be running.

D. CORRECT: Correct because D2P is the normally selected control power needed to start EDG 2. With the loss of offsite power and the EDG failing to start due to loss of Control Power, Essential 4160 vac Bus D1 will be de-energized in this scenario preventing any D1 pumps from running.

Sys #	System	Category	KA Statement
063	DC Electrical Distribution	K3.02 - 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:** DB-OP-02537 R20, Page 31 A.1

Question Source: New **Level Of Difficulty: (1-5)** 3.5

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.7 / 45.6)

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Objective: OPS-GOP-137-05K
Abbreviations: SFAS = Safety Features Actuation System, LBLOCA = Large Break Loss of Coolant Accident
Tier being Tested: ES401 Tier 2/Group 1 (RO)

49. The plant is operating at 100% power.

Removing which ONE of the following components from service requires declaring EDG 1 Inoperable?

A. EDG Air Compressor 1

B. EDG 1 Fuel Oil Transfer Pump

C. EDG 1 Air Start Valve DA30

D. EDG 1 DC Turbo Oil Pump

Answer: B

Explanation/Justification: This question is a match for the KA because the candidate must have knowledge of the effect of a malfunction of the Fuel Oil Storage Tank. The candidate must recognize that the Fuel Oil Transfer Pump must be available to maintain the EDG Week Tank system functional.

A. Incorrect because removing a single air compressor from service affects only a single air start side for each EDG. Each EDG has two air start sides and only 1 air start side is required for the EDG to remain operable.
Plausible because normally each EDG has two air start sides available and removing one may lead the examinee to believe functionality of the EDG is affected.

B. CORRECT because Attachment 13 of DB-OP-06316 Diesel Generator Operating Procedure requires the Fuel Oil Transfer Pump to be available for EDG operability. The EDG 1 Fuel Oil Transfer Pump automatically transfers fuel from the EDG 1 Week Tank to the EDG 1 Day tank.

C. Incorrect because closing DA 30 removes only a single air start side from service. Refer to Explanation A above.
Plausible because closing DA 30 will remove one air start side from service. Refer to Explanation A above

D. Incorrect because the EDG has two Turbo Oil Pumps, an AC and a DC oil pump. Either pump being available will maintain EDG operability.
Plausible because normally a DC pump is installed as a required backup to an AC powered pump

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Sys #	System	Category	KA Statement
064	EDG	Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system.	Fuel oil storage tanks

K/A# K6.08 **K/A Importance** 3.2 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-06316 Attachment 13

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** 41.7 / 45.7)

Objective: OPS-GOP-435-01K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

50. The Reactor is operating at 50% power.

If the Reactor Operator selects the GROSS mode on RE609, Main Steam Line 1 Radiation Monitor,

Which of the following describes how displayed count rate will be affected?

Displayed count rate will _____________.

A. rise due to the extended band of isotopes the RE would detect

B. rise due to the detector saturating in the elevated radiation field

C. lower due to lower detector sensitivity to N-16 gammas

D. lower since GROSS mode only indicates when the reactor is shutdown

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Answer: A

Explanation/Justification: This question is a match for the KA because is tests the candidate's ability to operate the Main Steam Line Radiation Monitors. These radiation monitors are unique in that they have the ability to be operate in two different modes. This question tests the ability to recognize difference in operation of radiation monitor between using in Gross mode (used when shutdown) versus using in Analyze mode (used at power)

A. CORRECT– in the analyze mode the detector is calibrated for N16 gammas. The Gross mode allows a greater band of isotopes to be detected providing a high output.

B. Incorrect because the cause of the increase is wrong – plausible if the detector is a GM detector, it could saturate and remain in saturation, but this would cause the count rate to lower.

C. Incorrect because the direction of the trend is in wrong – plausible since N16 gammas are the specific energy level the analyze mode is calibrated for

D. Incorrect the direction of the trend is wrong – plausible since this is when the detector is procedurally placed in the GROSS mode

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Sys #	System	Category	KA Statement
073	Process Radiation Monitor	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including:	Radiation levels

K/A# A1.01 **K/A Importance** 3.2 **Exam Level:** RO

References provided to Candidate None **Technical References:** Bases and Deviation Document for DB-OP-02000 R22 page 36 of 518

Question Source: NRC Exam DB 2016 Q71 **Level Of Difficulty: (1-5)** 3.5

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.5 / 45.7)

Objective: OPS-GOP-303-02K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

51. Which of the following describes a design feature that provides for train separation (mechanical or electrical) between Service Water (SW) Train 1 and SW Train 2?

A. Electrical Interlock between SW2929 SW Returns to Intake Structure AND SW2930, SW Returns to Intake Forebay that prevent the valves from being open at the same time.

B. Mechanical Interlock between AC109, XFER CD9 SW PUMP 1-3 AND AD109, XFER CD9 SW PUMP 1-3 that prevents both breakers from being closed at the same time.

C. Electrical Interlock between SW1395, SW Train 2 to TPCW Heat Exchangers AND SW1399, SW Train 1 to TPCW Heat Exchangers that prevents both valves from being open at the same time.

D. Mechanical Interlock between ACD4, TIE TO FDR BKR AC109 AND ACD5, TIE TO FDR BKR AD109 that prevents both breakers from being closed at the same time.

Answer: D

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the interlocks preventing components from being positioned in such a manner that would cross connect SW Train 1 and Train 2 violating Train Separation.

A. Incorrect because the interlock described in this response does not exist – Plausible because these valves are in two different flow paths for Service Water Returns. The candidate may assume that two train separated flowpaths are required. In addition, one of the Service Return flowpaths is always de-energized to ensure a flowpath is available. This may lead the candidate to assume these valves have an electrical interlock.

B. Incorrect because this is an electrical interlock, not a mechanical interlock – plausible since this is a method that could have been used to prevent both breakers from being closed at the same time. Having both breakers closed at the same time creates the potential for cross connecting the electrical supply to SW Pumps.

C. Incorrect because having both valves open is not desired, but there is not an interlock to prevent this operation – Plausible because with both these valves open, Service Water Train 1 and 2 would be cross connected. These valves auto close on an SFAS Actuation Level 2 and close on low SW pressure 50# to ensure train 1 and 2 are not cross connected when the SW System is performing its safety related function.

D. CORRECT – Kirk Key devices (mechanical interlock) are required to be installed on ACD4/5 when placing SWP 3 in service. This ensures that the SW Trains are not electrically cross connected when the swing Service Water pump is placed in service.

Sys #	System	Category	KA Statement
076	Service Water	Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following:	Service water train separation

K/A# K4.06 **K/A Importance** 2.8 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-06261, step 2.2.16 and Attachment 18, step 8.b and OS 20 R2 Note 13.

Question Source: DB2020 Q53 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: Low **10 CFR Part 55 Content:** (CFR: 41.7)

Objective: SYS-305-02K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

52. The plant is operating at 100% power with Service Water Returns aligned to the Cooling Tower.

A seismic event occurs that has caused a break in the Service Water return to the Cooling Tower Makeup Line.

Which of the following procedurally directed actions are required to respond to this event?

A. Align Circulating Water to supply Service Water Essential Header.

B. Align Circulating Water to supply Service Water Secondary Loads.

C. Align Service Water Returns to the Collection Box.

D. Align Service Water Returns to the Intake Forebay.

Answer: D

Explanation/Justification: This question is a KA match because it tests the Candidates knowledge of potential malfunctions or events that could cause a loss of service water function and then based on that knowledge, determine a procedurally directed action that would mitigate the impact of that malfunction. The service water system returns can be aligned to 4 locations depending on reactor power level and environmental conditions. A commonly used return flowpath is to Cooling Tower Makeup portions of which are non-seismic piping. If a seismic event occurs when the Service Water System returns are aligned to the Cooling Tower Makeup, inventory from the Ultimate Heat Sync could be lost out the break and not replenished from Lake Erie.

This impact can be mitigated by completing a Time Critical Operator action to align the Service Water Returns to the Intake Forebay (all seismic piping) within 2.2 hours following the Service Water return line break in the Cooling Tower Makeup flowpath. Based on the time critical nature of this action, the candidates need a good understanding of the need to complete this action.

A. Incorrect because Circ Water is not used to provide essential load cooling for this malfunction – Plausible because the essential loads are those that must be cooled to maintain safety functions. Procedure direction is provided for using the Circ Water System to cool essential SW for a beyond design bases event, but not for this design bases break in the cooling tower return line.

B. Incorrect because Circ Water could be aligned to provide cooling for secondary loads, but secondary loads would not be lost for these plant conditions and is not the action directed by the control procedure – Plausible because are important loads such as the MDFP the candidate would prefer to have available to respond to plant events.

C. Incorrect because this is not the action directed by the procedure. The collection box is not a seismic return path– Plausible because the Cooling Tower Line is not seismic. If the line collapses , a loss of flowpath could exist. This alignment could restore a flowpath.

D. CORRECT – The initial conditions have SW returns aligned to Cooling Tower Makeup. Following a seismic event, this alignment could lead to depletion of the Ultimate Heat sink via non-seismic piping. Action is required within 2.2 hours to protect the ultimate heat sink inventory by aligning the SW Returns to a Seismic qualified return flowpath providing return inventory to the intake forebay.

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

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

References provided to Candidate None **Technical References:** DB-OP-02511 R21, Loss of Service Water Pumps/System, Subsection 4.4 and Attachments 16 & 22

Question Source: New **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: High

10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45/3 / 45/13)

Objective: OPS-SYS-305-10K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

53. The plant is operating at 100% power.

All Control Room Instrument Air (IA) and Station Air (SA) System pressure indicators begin to lower.

At what Air Pressure will the Main Feedwater Control Valves SP6A and SP6B lock at their current position if SFRCS is not Actuated?

A. 105 psig

B. 90 psig

C. 75 psig

D. 60 psig

Answer: D

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the expected response to lowering air pressure in the Instrument Air System. The Main Feedwater Control Valves are air operated valves. To prevent unintended operation as air pressure lowers, they have a design feature to lock in position in the event Instrument Air header pressure at the valves lowers to 60 psig.

A. Incorrect because this is not the pressure that causes the MFW Control Valves to lockup. See DB-OP-02528 IA Malfunctions page 125. Plausible because this is the instrument air pressure that starts the Standby Instrument Air Compressor.
B. Incorrect because this is not the pressure that causes the MFW Control Valves to lockup See DB-OP-02528 IA Malfunctions page 125. Plausible because this is the instrument air pressure that that causes the system throttle valves to begin throttling to maintain Instrument Air over Station Air. System Throttle or back pressure regulator valves include SA2008, SA HDR BACK PRESS REGULATOR VLV, IA2043, TURBINE BUILDING BACK PRESSURE CTRL and IA2044, AUX BLDG INSTRUMENT AIR CTRL VLV.
C. Incorrect because this is not the pressure that causes the MFW Control Valves to lockup See DB-OP-02528 IA Malfunctions page 125. Plausible because this is the pressure that requires the operator to manually trip the reactor and to initiate and isolate SFRCS.
D. CORRECT – This is the pressure that causes the Main Feedwater Control Valves to lock in position if SFRCS is not actuated on an isolation trip. Actuating SFRCS ports all the air off and allow the Main Feedwater Control Valves to close.

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:** DB-OP-02528 R27 IA Malfunctions page 125

Question Source: New **Level Of Difficulty: (1-5)** 3

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

Objective: OPS-SYS-602-08K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

54. An inadvertent Safety Features Actuation System (SFAS) Level 2 actuation has occurred during testing.

All Level 2 components actuated as designed.

The operating crew has restored RCS Letdown for control of Pressurizer Level.

What is the expected status of the SAM light associated with MU3, LETDOWN ISOLATION?

A. DIM

B. OFF

C. BRIGHT

D. FLASHING

Answer: D

Explanation/Justification: DB does not have Phase A and Phase B resets. The Davis Besse Safety Features Actuation System SFAS has 5 Actuation Levels. Plant conditions are specified by the Emergency Operating Procedure DB-OP-02000 to block and reset equipment that has been actuated by SFAS. This KA is related to restoring equipment following an SFAS Actuation. This question is a KA match for Davis Besse because it tests the Candidate's knowledge for monitoring conditions following an SFAS Actuation and the ability to operate SFAS actuated equipment following actuation. The condition of the SAM light following use of a block which is similar to a reset is tested. The SAM lights or SFAS Status Lights have 4 positions that depend on the status of SFAS (Actuated or not AND blocked or not) and the position of the equipment (in SFAS required position or not). These combinations allow the operator to rapidly determine the status of equipment following an SFAS Actuation. All 4 possible conditions (Off, Dim, Bright, and Flashing) are presented.

A. Incorrect – Plausible because Dim is a possible condition of a SAM Light. Incorrect because DIM is indication of an SFAS trip with the component in the expected position.

B. Incorrect - Plausible because Off is a possible condition of a SAM Light. Incorrect because Off is indication of no SFAS trip for that component.

C. Incorrect – Plausible because Bright is a possible condition of a SAM Light. Incorrect because Bright is an indication of an SFAS trip and the Block pushbutton depressed with the component in it's SFAS position.

D. CORRECT –Flashing is the SAM light status following an SFAS Trip when the trip signal has been blocked and component has been repositioned out of the SFAS alignment.

Sys #	System	Category	KA Statement
103	CTMT	Ability to manually operate and/or monitor in the control room:	Phase A and phase B (Containment Isolation) resets

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

References provided to Candidate None **Technical References:** DB-OP-06405 R15, SFAS Attachment 2, Page 77

Question Source: Exam Bank ORQ-39728 Modified **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 to 45.8)

Objective: OPS-SYS-506-03K

Abbreviations: SFAS = Safety Features Actuation System SAM Lights = Status Actuation Monitoring Light

Tier being Tested: ES401 Tier 2/Group 1 (RO)

55. Which of the following Containment Isolation Valves are in the required position following a SFAS Level 2 actuation?

A. Component Cooling Water System Containment Header Supply Valves CC 1411A and CC 1411B have closed.

B. Reactor Coolant System Letdown Cooler Valve MU2A has closed.

C. Reactor Coolant Pump Seal Return Valve MU38 has closed.

D. Reactor Coolant Pump Seal Injection Valves MU66A-D have closed.

Answer: B

Explanation/Justification: This question is a KA match because it tests the Candidates knowledge of the required position of various Containment Isolation valves following an SFAS Level 2 actuation. SFAS Level 2 would occur on low RCS Pressure of 1600 psig. Each valve presented is a Containment Isolation Valve. The ability to locate Control Room switches will be tested during the Operating Exam. This question focuses on the ability to determine if the indications correctly reflect the desired plant lineup.

A. Incorrect because CC1411A and B close on an SFAS Level 4 Signal – Plausible because these valves receive an SFAS Isolation signal, but on an SFAS level 4, not Level 2.
B. CORRECT – MU2A receives a close signal on an SFAS Level 2 actuation.
C. Incorrect because MU38 closes on an SFAS Level 3 signal – Plausible because these valves receive an SFAS Isolation signal, but on an SFAS level 3, not Level 2.
D. Incorrect because MU66A-D close on an SFAS Level 3 signal – Plausible because these valves receive an SFAS Isolation signal, but on an SFAS level 3, not Level 2.

Sys #	System	Category	KA Statement
103	CTMT	Generic	Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

K/A# G2.1.31 **K/A Importance** 4.6 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02000 R32, Table 2.

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.10 / 45.12)

Objective: OPS-GOP-302-03K

Tier being Tested: ES401 Tier 2/Group 1 (RO)

56. The Reactor is operating at 100% power in a normal system lineup.

The following indications are noted:

- RCS Pressure is slowly rising at 2190 psig
- Annunciator (4-1-E) PZR LO LVL HTR TRIP alarms
- Annunciator (4-2-E) PZR LVL LO alarms
- Annunciator (2-4-C) MU FLOW HI TRN 2 alarms
- A step change in the selected Pressurizer Level recorder from 220 inches to 0 inches is noted.

With these current plant conditions, which of the following describes an action required of the Reactor Operator by DB-OP-02513, Pressurizer Malfunctions and the reason for that action?

A. Place MU32, Pressurizer Level Controller, in Manual and control pressurizer level using an alternate level instrument.
The selected Pressurizer Level instrument has failed.

B. Place MU32, Pressurizer Level Controller, in Manual and control pressurizer level using an uncompensated level instrument.
The selected Pressurizer Temperature for compensation element has failed.

C. Trip the reactor and GO TO DB-OP-02000, RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE.
Pressurizer Level is less than 100 inches.

D. Open the Pressurizer Spray Valve when RCS Pressure reaches 2205 to reduce RCS Pressure.
The Pressurizer Spray Valve will not operate in automatic with low Pressurizer Level.

Answer: A

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the Pressurizer Level Control System when a malfunction occurs and the procedurally directed actions to be taken in response to that malfunction. The event is the selected Pressurizer level d/p instrument failing high. A high transmitter d/p indicates low level. This will cause the level control valve to open resulting in a rise in RCS Pressure. Based on the failure to 0 inches, this is not a temperature compensation failure which if failed low, would only cause an approximately 70 inches decrease in indicated Pressurizer level. DB-OP-02513, Pressurizer Malfunctions Section 2.6 provides the indications and 4.6 provides the expected actions.

A. CORRECT – The event is the selected Pressurizer level d/p transmitter failing Hi. This will cause the level control valve to open resulting in a rise in RCS Pressure. The procedurally directed action is to take MU32 to hand to manually control Pressurizer Level using an alternate instrument.

B. Incorrect because specified action is correct, but the reason for the malfunction does not match the indications provided – Plausible because the action is correct, but the indications don't match a compensation temperature low failure. Based on the failure to 0 inches, this is not a temperature compensation failure which if failed low, would only cause an approximately 70 inches decrease in indicated Pressurizer level.

C. Incorrect – Plausible because actual RCS pressure is rising. Incorrect because the Pressurizer Level instrument failure does not affect the Spray Valve. It will operate normally to reduce RCS Pressure to the desired range. The desire to protect the Pressurizer Heaters is required on actual low Pressurizer level by turning the Pressurizer Heaters to off, not limiting Spray flow. In addition, actual Pressurizer level is rising, and the heaters are covered.

D. Incorrect – Plausible because actual RCS pressure is rising. Incorrect because the Pressurizer Level instrument failure does not affect the Spray Valve. It will operate normally in automatic to reduce actual RCS Pressure to the desired range.

Sys #	**System**	**Category**	**KA Statement**
011	PZR Level Control	Generic	Ability to interpret and execute procedure steps.

K/A# G2.1.20 **K/A Importance** 4.6 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02513 R13, Pressurizer Malfunctions, Subsection 4.6, Failure of Selected Pressurizer Level Instrument.

Question Source: New **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.12)

Objective: OPS-GOP-113-03K

Tier being Tested: ES401 Tier 2/Group 2 (RO)

57. The plant is at 80% power in a normal automatic Integrated Control System (ICS) lineup when Power Range NI-7 upper chamber **ONLY** fails high.

Which ONE of the following ICS cross-limit conditions should occur?

A. Cross limits will decrease FW flow.

B. Cross limits will increase FW flow.

C. Cross limits will increase Rx power.

D. Cross limits will have no effect.

Answer: B

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the interconnection between the Nuclear Instrumentation system and the Integrated Control System. A failure in the Nuclear Instrument occurs. This malfunction causes a false signal to be sent to the Integrated Control System. A high failure was selected to get the signal past the auctioneer high selected NI signal to the ICS. The value was selected to cause a large differential between reactor power demanded and reactor power actual (NI signal). This large difference triggers a cross limit that will cause an increase in FW flow. A cross-limit occurs when reactor power demand is different from the reactor power by at least plus 10% or minus 5%, or that the feedwater flow is less than actual feedwater flow demand by at least 5%. If either of these conditions exist, the ICS will go into cross-limits mode of operation.

A. Incorrect because the cross limit will cause Feedwater flow to rise – Plausible because the candidate could reverse the affect the cross limit causes. This malfunction exceeds the threshold to cause a cross limit response. If the candidate does not understand which direction that FW flow would be altered, they may select this incorrect response.

B. CORRECT – This malfunction would cause an increase of power from NI's from approximately 80% to approximately 102.5%. This exceeds the threshold for actuating a cross limit. Because indicated reactor power (as provided by the auctioneer high select NI) goes up by more than 10%, the cross limit would cause an increase in FW Demand as the system attempts to balance power being produced with Feedwater available to remove the heat.

C. Incorrect because a cross limit will not cause a demanded increase in reactor power, only a decrease – Plausible because the indicated reactor power is higher than the demanded reactor power. The typical ICS response to this condition is to raise reactor power by raising feedwater flow. Because of the magnitude of the error, the cross-limit circuit does not provide a signal to cause an increase in reactor power, only a decrease. The candidate could select this response if they are unaware of this feature of ICS Cross limits.

D. Incorrect because the malfunction will cause a cross limit and it will affect feedwater demand – Plausible because the NI power signal for total power is the power in the top chamber added to the power in the bottom chamber. Since the upper limit on the power range NI signal is 62.5% from each chamber, this malfunction at 100% power would only cause an increase in indicated power of approximately 12.5% however, the reactor demand circuit has an upper limit of approximately 103%. The candidate may incorrectly assume this upper limit is applied to the cross-limit circuit as well. As a result, the candidate may assume this malfunction is not enough to trigger the cross limit and therefore, at 100% power, therefore, response D would be the correct response based on faulty assumptions.

Sys #	System	Category	KA Statement
015	Nuclear Instruments	Knowledge of the physical connections and/or cause/effect relationships between the NIS and the following systems:	ICS

K/A# K1.05 **K/A Importance** 3.9 **Exam Level:** RO

References provided to Candidate: None

Technical References: SD045 R08, 2.1.2.1.3 Track, 2.1.2.3.2, Cross limit from Reactor, and 2.1.2.4.2 Neutron Error, 2.1.2.4.5, Cross limits General, 2.1.2.4.6 Reactor to Feedwater Cross Limit, 2.1.2.4.7 Feedwater to Reactor Cross Limit

Question Source: Exam Bank Question 36948 **Level Of Difficulty: (1-5)** 3.5

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

Objective: OPS-SYS-502-03K

Tier being Tested: ES401 Tier 2/Group 2 (RO)

58. Engineering has determined the non-qualified Incores are not available for indication in the Control Room.

With this plant condition natural circulation can still be verified by verifying the qualified Incore indications at the __(1)__ are tracking with __(2)__.

A. (1) Post-Accident Monitoring Panel
(2) Tcold

B. (1) Post-Accident Monitoring Panel
(2) Thot

C. (1) Auxiliary Shutdown Panel
(2) Tcold

D. (1) Auxiliary Shutdown Panel
(2) Thot

Answer: B

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the natural circulation indications that are available for verifying RCS Natural Circulation in the event that the Incore Temperature Indicators are not available.

A. Incorrect
(1) Correct the Post Accident Monitoring Panel has qualified incore temperature indications.
(2) Incorrect because Thot, not Tcold tracking in conjunction with incore temperature is used to confirm natural circulation – plausible because Tcold does represent RCS temperature and is used in determining natural circulation in conjunction with SG Tsat, not incore temperature.

B. CORRECT –
(1) Correct - There are 2 channels of 8 qualified Incores each that can be monitored from the Post Accident Monitoring panel.
(2) Correct - One of the verifications of natural circulation indications is Incores temperatures tracking with RCS Thot indications.

C. Incorrect
(1) Incorrect - The location of the available instrument is incorrect. There are no incore temperature indications available at the Post Accident Monitoring Panel. Plausible because the auxiliary shutdown panel is used to monitor the plant when indications are unavailable in the control room
(2) Incorrect because Thot, not Tcold tracking in conjunction with incore temperature is used to confirm natural circulation – Plausible because Tcold does represent RCS temperature and is used in determining natural circulation in conjunction with SG Tsat, not incore temperature..

D. Incorrect
(1) Incorrect – The location of the available instrument is incorrect. There are no incore temperature indications available at the Post Accident Monitoring Panel. Plausible because the auxiliary shutdown panel is used to monitor the plant when indications are unavailable in the control room
(2) Correct - One of the verifications of natural circulation indications is Incores temperatures tracking with RCS Thot indications.

Sys #	System	Category	KA Statement
017	In-Core Temperature Monitor	K3 - Knowledge of the effect that a loss or malfunction of the **ITM** system will have on the following:	Natural circulation indications

K/A# K3.01 **K/A Importance** 3.5* **Exam Level** RO

References provided to Candidate None **Technical References:** SD-043 step 2.9, DB-OP-06903 R56, step 6.3

Question Source: DB 2020 NRC Exam Q58 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: Low **10 CFR Part 55 Content:** 41.7 / 45.6

Objective: SYS-503-03K

Tier being Tested: ES401 Tier 2/Group 2 (RO)

59. Which of the following design or normal operating features of the Containment Purge System ensures a negative pressure in Containment when the System is operating on Containment without an SFAS Actuation?

A. The capacity of the Exhaust Fan is higher than the capacity of the Supply Fan.

B. ONLY the Exhaust Fan is operated if Containment is Closed.

C. Recirculation Damper modulates to maintain -0.75 inches wc in Containment compared to the Annulus.

D. Supply Fan Damper modulates to maintain -0.75 inches wc in Containment compared to the Annulus.

Answer: A

Explanation/Justification: This question is a match for the KA because it tests the candidate's knowledge of the design features of the Containment Purge System that ensure a negative pressure is maintained in Containment when the system is in operation.

A. CORRECT – As noted in the System Description for the Containment Purge System, the capacity of the Exhaust Fan is sized to be greater than the capacity of the Supply Fan, so a negative pressure is maintained when the system is in operation.

B. Incorrect because this is not a normally accepted method of operating the Containment Purge System. There is a timed interlock that requires operation of both fans. Either Fan will trip if the other fan is not in operating within the interlocked time.
Plausible because under special conditions such as creating a large opening in the Containment barrier during SG replacement the supply fan interlocks may be disabled to allow operating the exhaust fan alone. This would only be acceptable when a large access point is created through the containment vessel and shield building.

C. Incorrect because the Containment Purge system does not have any type of recirculation damper for pressure control.
Plausible because this type of pressure control is used by the station Emergency Ventilation System to provide a negative pressure in the annulus and penetration rooms

D. Incorrect because the Containment Purge system dampers do not modulate
Plausible because some damper modulation is used by the Station EVS system to control negative pressure. Refer to Explanation C above.

Sys #	System	Category	KA Statement
029	Containment Purge	Knowledge of design feature(s) and/or interlock(s) which provide for the following:	Negative pressure in containment

K/A# K4.02 **K/A Importance** 2.9 **Exam Level:** RO

References provided to Candidate None **Technical References:** SD022D R4, Containment Purge Section 1.2.1.4

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** 41.7

Objective: OPS-SYS-108-02K

Tier being Tested: ES401 Tier 2/Group 2 (RO)

60. Initial plant conditions:
    - The plant is at 100% power
    - Fuel handling operations are in progress in the Spent Fuel Pool (SFP) using the Spent Fuel Pool Bridge
    - A spent fuel assembly is in the UP position on the North Monorail Hoist

    An event occurs. The following conditions are noted in the Control Room:
    - Annunciator Alarm (3-1-B) SFP LVL
    - Annunciator Alarm (9-1-G), FIRE OR RADIATION TRBL
    - Lowering level on LI1600, SFP Level Indication
    - Rising Radiation Levels on SFP Area Radiation Monitors
    - Auxiliary Building Sump Pumps running

    A large leak is reported from the Spent Fuel Pool Cooling Heat Exchanger.

    The SFP Bridge Operator has been unable to place the Spent Fuel Assembly in a safe location.

    (1) At what level in the SFP, is the SFP bridge operator required to abandon the bridge?

    AND

    (2) When is the operating SFP Cooling Pump required to be stopped?

A. (1) 10-foot level
   (2) immediately

B. (1) 10-foot level
   (2) 19-foot level

C. (1) 19-foot level
   (2) immediately

D. (1) 19-foot level
   (2) 19-foot level

Answer: C

Explanation/Justification: This question tests the candidate's knowledge of the action required actions for lower SFP level which will cause radiation levels to rise in the SFP area. As directed by DB-OP-02547, SFP Cooling Malfunctions in the event of a loss of inventory from the SFP Cooling System, the running SFP Cooling Pumps are directed to be stopped immediately to limit the inventory loss from the SFP. The SFP pumps will lose suction from the SFP at approximately 19 feet SFP Level, however stopping prior to loss of suction is directed to limit inventory loss.

DB-NE-06303, Fuel Handling in the Auxiliary Building provides the following" Step 5.6 Falling Spent Fuel Pool Level: a. IF bridge operators have been unable to lower loaded masts before SFP level falls to 19 feet (597' el), THEN they should abandon bridges and evacuate the area to avoid high personnel radiation exposure. In addition, DB-OP-02547 provides the following: "During fuel transfer, a minimum level of 9.5 feet is maintained above the assembly being moved. A minimum of 9.5 feet of borated water above the fuel will ensure adequate biological shielding."

A. Incorrect
- (1) Incorrect because the SFP Level to abandon the bridge is incorrect. Plausible because a minimum of 9.5 feet of borated water above the fuel will ensure adequate biological Shielding, but with a fuel assembly up in the bridge and lowering level the bridge is abandoned at 19 feet SFP Level.
- (2) Correct – As directed by DB-OP-02547, the SFP pumps are stopped as soon as there is indication of a SFP Cooling System leak causing the SFP to lower

B. Incorrect
- (1) Incorrect because the SFP Level to abandon the bridge is incorrect. Plausible because a minimum of 9.5 feet of borated water above the fuel will ensure adequate biological Shielding, but with a fuel assembly up in the bridge and lowering level the bridge is abandoned at 19 feet SFP Level.
- (2) Incorrect because when to stop the SFP Pumps is incorrect – Plausible because SFP pumps will lose suction from the SFP at approximately 19 feet SFP Level.

C. CORRECT –
- (1) Correct - As provided by DB-NE-06303, with an assembly up in the mast, the SFP Bridge must be abandoned at 19 feet in the SFP.
- (2) Correct – As direct by DB-OP-02547, the SPF pumps are stopped if there is indication of a SFP Cooling System leak causing the SFP level to lower. The report from the SFP Heat Exchanger Room confirms the SFP Cooling System leak if the reason SFP level is lowering.

D. Incorrect
- (1) Correct - As provided by DB-NE-06303, with an assembly up in the mast, the SFP Bridge must be abandoned at 19 feet in the SFP.
- (2) Incorrect because the SFP level to stop the SFP pump is incorrect. DB-OP-02547 directs stopping the SFP Pumps as soon as possible on lowering SFP Level. Plausible because SFP pumps will lose suction from the SFP at approximately 19 feet SFP Level.

Sys #	System	Category	KA Statement
033	Spent Fuel Pool Cooling	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling System operating the controls including:	Radiation monitoring systems

K/A# A1.02 **K/A Importance** 2.8 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02547 R8, Spent Fuel Pool Cooling Malfunctions Sect 4.2 and DB-NE-06303 R20, Step 5.6.

Question Source: New **Level Of Difficulty: (1-5):** 3.5

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.5 / 45.5)

Objective: OPS-GOP-130-03K

Tier being Tested: ES401 Tier 2/Group 2 (RO)

61. The plant was operating at 100% power. The reactor is manually tripped due to high vibration on the Main Generator.

The following events occur:
- SP13B1, Steam Line 1 Turbine Bypass Valve sticks full open
- All other Turbine Bypass Valves respond to Steam Generator Pressure as expected

All other equipment functions as designed.

- How will the plant respond to this failure, assuming no operator actions?

AND

- What, if any, operator actions will be **required** to stabilize the plant without relying on Main Steam Safety Valve operation?

A. (1) The unaffected Turbine Bypass Valves will modulate closed to control both SG pressures at the normal post trip setpoint of approximately 995 psig. This condition will not result in an SFRCS actuation.
(2) No Operator Action will be required to stabilize the plant.

B. (1) SFRCS will actuate on low SG1 Level, closing the Main Steam Isolation Valves, and starting Auxiliary Feedwater to restore SG1 Level to 49 inches.
(2) No Operator Action will be required to stabilize the plant.

C. (1) SFRCS will actuate on low SG Pressure on SG1, closing both Main Steam Isolation Valves.
(2) The Operators will block the SFRCS signal to the Atmospheric Vent Valves (AVVs) and use the AVVs in manual to control RCS Tave constant or slightly lowering.

D. (1) SFRCS will actuate on Steam to Feed Differential Pressure on SG1, isolating all Main and Auxiliary Feedwater to SG1.
(2) The Operators will block the SFRCS signal to the Atmospheric Vent Valves (AVVs) and open the Atmospheric Vent Valve on #1 SG to blowdown the affected SG.

Answer: C

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the malfunction of the Turbine Bypass Valve being stuck open impact on the plant, the ability to predict that SG pressure will reduce causing an SFRCS Actuation on low SG pressure. In addition, the expected SFRCS impact on the Stuck open TBV that will close the Main Steam Isolation Valves prevent further stream release and the fact that manual control will be required since the SFRCS Trip will prevent automatic control by the AVVs.

A. Incorrect

(1) Incorrect because a single TBV failed open (5% steam flow) will rapidly exceed the decay heat available from the core after the reactor trip. SG Pressure will not be controlled at 995 psig. – Plausible if the candidate concludes the steam flow rate due to one open TBV is less than the core decay heat rate post trip. This event will exceed the core decay heat rate available even if all other TBVs are closed. If the steam flow was less than core decay heat, then this response would be accurate.

(2) Incorrect – Operator Action will be required because a single TBV failed open (5% steam flow) will rapidly exceed the decay heat available from the core after the reactor trip causing SG pressure to lower – Plausible if the candidate concludes the steam flow rate due to one open TBV is less than the core decay heat rate post trip. This event will exceed the core decay heat rate available even if all other TBVs are closed. If the steam flow was less than core decay heat, then this response would be accurate.

B. Incorrect

(1) Incorrect because the Main Feedwater System has sufficient capacity to maintain SG level for the conditions provided – Plausible because the Steam Generator Level would be lowering with an open TBV, however the Main Feedwater System and AFW, if actuated, can maintain SG level at setpoint even with a full open TBV. The MSIVs would not close on low SG Level. A candidate may select this response it they conclude the Steam Flow rate would exceed Feed flow rate causing a low SG level.

(2) Incorrect – Operator Action will be required because a single TBV failed open (5% steam flow) will rapidly exceed the decay heat available from the core after the reactor trip causing SG pressure to lower – Plausible if the candidate concludes the steam flow rate due to one open TBV is less than the core decay heat rate post trip. This event will exceed the core decay heat rate available even if all other TBVs are closed. If the steam flow was less than core decay heat, then this response would be accurate

C. CORRECT

(1) Correct - Without Operator Action, SG pressure in #1 SG would lower and cause an SFRCS Low SG Pressure on #1 SG at 630 psig. Once the MSIVs close, steam is isolated to the failed open TBV by MSIV closure.

(2) Correct - Operator action to control SG Pressure would be necessary to prevent Main Steam Safety Valves from opening.

D. Incorrect

(1) Incorrect – This is the action to blowdown the affected SG is for steam leaks that present a personnel safety hazard. A TBV blowing to the condenser is not personnel safety hazard – Plausible because SFRCS may actuate on Steam to Feed Differential Pressure once the MSIVs are closed in response to the low SG Pressure.

(2) Incorrect - The actions to blowdown the affected SG are actions taken in response to a Steam Line Break in accordance with DB-OP-02525, Steam Leaks, section 4.2, not an action taken in response to a TBV malfunction. Plausible because a Candidate may select this response if they incorrectly conclude that SG blowdown is required like it is for a Steam break inside CTMT.

Sys #	System	Category		KA Statement
041	Steam Dump System (SDS) and Turbine Bypass Control	Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations:		Steam valve stuck open
K/A#	A2.02	**K/A Importance** 3.6	**Exam Level**	RO
References provided to Candidate	None		**Technical References:**	USAR 10.4.4.1 page 10.4-6, DB-OP-02000 Table 1 Rev. 32
Question Source:	Bank DB2020 Q43		**Level Of Difficulty: (1-5)**	4
Question Cognitive Level:	High - Comprehension		**10 CFR Part 55 Content:**	(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Objective: SYS-202 OPS-GOP-306-06A

Abbreviations: SFRCS = Steam Feedwater Rupture Control System

Tier being Tested: ES401 Tier 2/Group 2 (RO)

62. Plant Conditions:
    - Reactor Power is at 35%
    - A plant shutdown is in progress
    - The Seal Oil Vacuum Tank is removed from service for repairs

    The following conditions were identified:
    - A failure of the Main Generator Hydrogen purity analyzer was discovered
    - After repairs were completed, Hydrogen purity was verified to be at 88% and slowly lowering

    Which of the following actions are required?

A. Scavenge the Main Generator with CO2 until the concentration of air is reduced to less than 1%.

B. Shutdown the Generator and purge with CO2 prior to hydrogen purity dropping below 75%.

C. Reduce MVARS and maintain Generator Cold Gas temperature less than 113°F.

D. Immediately trip the Turbine and perform an Emergency Lowering of Generator Hydrogen Pressure.

Answer: B

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the actions required to mitigate a potential explosive mixture of Hydrogen and Air in the Main Generator.

A. Incorrect because this action is not directed by any procedure as a response to lowering Hydrogen purity. Plausible because reducing the amount of air in the Hydrogen mixture would reduce the possibility of having an explosive mixture.

B. CORRECT – This action is directed by DB-OP-02016 and DB-OP-06210 if purity cannot be maintained above 75%

C. Incorrect because this action is not directed by any procedure as a response to lowering Hydrogen purity. Plausible because reducing MVARS will reduce the amount of heat generated in the main generator, potentially reducing the possibility of an explosion.

D. Incorrect because this action is not directed by any procedure as a response to lowering Hydrogen purity. Plausible because the low purity alarm comes in at 90%.

Sys #	System	Category	KA Statement
045	Main Turbine Generator	Knowledge of the operational implications of the following concepts as the apply to the MT/G System:	Possible presence of explosive mixture in generator if hydrogen purity deteriorates

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

References provided to Candidate: None **Technical References:** DB-OP-06210 R20 L&P 2.2.7 and DB-OP-06301 R35 L&P 2.2.1.f

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** CFR: 41.5 / 45.7)

Objective: OPS-SYS-215-25K

Tier being Tested: ES401 Tier 2/Group 2 (RO)

63. The plant is operating at 100% power.

The following events have occurred simultaneously:

- Annunciator 9-1-G FIRE OR RADIATION TRBL alarms
- Annunciator 9-2-G FIRE WTR ELEC PMP ON alarms

A SINGLE Fire Suppression Area has alarmed as indicated on the Fire/RMS Computer.

The Electric Fire Water Pump is running.

Which ONE of the following actions related to Fire Brigade Activation is required by DB-OP-02529, Fire Procedure, as a result of these events?

A. Dispatch an Operator to investigate the Fire Alarm, Fire Brigade activation is NOT required until a Fire has been confirmed.

B. Activate the Fire Brigade ONLY if another alarm is received from an area adjacent to the original alarm.

C. Activate the Fire Brigade ONLY if the area in alarm is predesignated to require immediate Fire Brigade activation.

D. The Fire Brigade shall be activated immediately.

Answer: D

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the requirements for activating the fire brigade following actuation of the Fire Protection System. Monitoring of the system is required to determine that an area fire alarm is occurring in conjunction with the start of the electric fire pump.

A. Incorrect because a single fire alarm in conjunction with a fire pump start requires immediate activation of the Fire Brigade – Plausible because IF only a single alarm or indication of a Fire exists, THEN verify an Operator has been dispatched to investigate fire related alarms. (Step 4.1). In this scenario, the Fire Brigade is required to be activated immediately.

B. Incorrect because a single fire alarm in conjunction with a fire pump start requires immediate activation of the Fire Brigade – Plausible because IF Multiple Fire Alarms for adjacent detectors OR Fire Detection Zones are indicated by the Fire Detection System then the fire brigade would be activated. (Step 4.2). In this scenario, the Fire Brigade is required to be activated immediately.

C. Incorrect because a single fire alarm in conjunction with a fire pump start requires immediate activation of the Fire Brigade – Plausible because prior to NFP 805 implementation in Feb 2020, there were predesignated areas that required immediate Fire Brigade activation on a single fire alarm.

D. CORRECT – DB-OP-02529, Fire requires immediate activation of the fire brigade if a single Fire Detection Zone OR Fire Suppression Area Alarm is received AND a simultaneous start of a Fire Pump occurs.

Sys #	System	Category	KA Statement
086	Fire Protection	Ability to monitor automatic operation of the Fire Protection System including:	Actuation of the FPS.

K/A# A3.02 **K/A Importance** 2.9 **Exam Level**

References provided to Candidate: None **Technical References:** DB-OP-02529 R12 Fire Procedure Step 4.2

Question Source: Exam bank question Q37751 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5)

Objective: OPS-GOP-129-03K

Tier being Tested: ES401 Tier 2/Group 2 (RO)

64. The following plant conditions exist:
    - A liquid radwaste discharge is in progress from the Clean Waste Monitor Tank (CWMT) 1 to the collection box.
    - The CLEAN WASTE SYSTEM OUT RAD HI annunciator is in alarm.
    - The operator determines that Clean Waste System Outlet Radiation Monitor, RE1770A, is above its high trip setpoint.

    Which ONE of the following is the expected automatic response of the Clean Waste System (CLN WST SYS)?

Valve #	Valve Description
WC1771	CLEAN LIQUID RAD WASTE DISCHARGE FLOW CONTROL VALVE
WC1701A	CLEAN LIQUID RAD WASTE LOW FLOW COLLECTION BOX DISCH HEADER FLOW CONTROL VALVE
WC1701B	CLEAN LIQUID RAD WASTE HI FLOW COLLECTION BOX DISCH HEADER FLOW CONTROL VALVE
WC1704	CWMT 1 OUTLET FLOW CONTROL VALVE

A. The operating CLN WST SYS Transfer Pump trips
AND
WC1771 receives a close signal.

B. The operating CLN WST SYS Transfer Pump trips
AND
WC1704 receives a close signal.

C. WC1701A and WC1701B receive a close signal
AND
WC 1704 receives a close signal.

D. WC1701A and WC1701B receive a close signal
AND
WC1771 receives a close signal.

Answer: D

Explanation/Justification: This Question is a KA match because it tests the candidate's knowledge of the automatic actions of the Radiation Monitoring System response to a high radiation event during a release to the environment from the Clean Waste Monitoring Tank. Operator Action to terminate the release would be required to prevent exceeding release limits if the automatic actions fail to occur as designed.

A. Incorrect because this is not the automatic response of the Clean Waste System to a high outlet radiation alarm - Plausible because tripping the Transfer Pump and closing WC-1771 would terminate the release, but this action is not the as designed response to a high radiation alarm on the Clean Waste system when aligned for a release.

B. Incorrect because this is not the automatic response of the Clean Waste System to a high outlet radiation alarm - Plausible because tripping the Transfer Pump and closing WC-1704 would terminate the release, but this action is not the as designed response to a high radiation alarm on the Clean Waste system when aligned for a release.

C. Incorrect because this is not the automatic response of the Clean Waste System to a high outlet radiation alarm - Plausible because closing the isolation valves to the collection box and closing WC1704 would terminate the release, but this action is not the as designed response to a high radiation alarm on the Clean Waste system when aligned for a release.

D. CORRECT – These valves are the end points for the clean waste system for a CWMT release. These valves receive a close signal on a high radiation event on RE1770A.

Sys #	System	Category	KA Statement
068	Liquid Radwaste	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:** DB-OP-03011 R30 Step 4.12.31 OS28A Sheet 1 R21 CL123, CL13 and OS28A Sheet 4 Rev 15

Question Source: **Exam Bank 38338** **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.7 / 45.7)

Objective: OPS-SYS-115-04K

Tier being Tested: ES401 Tier 2/Group 2 (RO)

65. While operating at 100% power, CT861 (Circulating Water Pump 2 Discharge Valve) goes closed due to a spurious control circuit signal.

The operator attempts to reopen the closed valve, but CT861 will not reopen.

Circulating Water Pump 2 fails to trip automatically.

Which ONE of the following operator action(s) is correct as directed by DB-OP-02517, Circulation Water System Malfunctions?

A. Fully close CT856 (Circulating Water Pump 1 Discharge Valve), then verify Circulating Water Pump 1 trips automatically.

B. Immediately trip Circulating Water Pump 2, then verify CT856 (Circulating Water Pump 1 Discharge Valve) closes to the THROT position.

C. Dispatch an operator to Open CT882 (Waterbox Crossover Valve) then trip Circulating Water Pump 2.

D. Maintain all four Circulating Water Pumps in operation while awaiting troubleshooting CT861 closure.

Answer: B

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the impact on a malfunction (Circ Water Pump Discharge Valves spuriously closes) and then based on that knowledge provides the procedure directed actions to respond to the event. Closure of the Circ Water Pump Discharge Valve causes a loss of flow from the affected Circ Water Pump, effectively causing a loss of that Circ Water Pump. The Circ Water Pumps do not have minimum flow protection. With the Discharge Valve closed, the affected pump must be stopped.

A. Incorrect because this is not the procedurally directed actions for the plant conditions provided – Plausible because Circ Pump 1 is affected by the no flow condition from Circ Pump 2. Each Circ Water Loop has two pumps, Circ Pumps 1 and 2 in Loop 1 and Circ Pumps 3 and 4 in Loop 2. Continued operation of Circ Pump 1 requires placing the discharge valve in the Throttle position. Fully closing Circ Pump 1 discharge valve would cause a loss of Circ Water Loop 1 and likely result in a Turbine Trip on High Condenser Pressure.

B. CORRECT – Closure of the Circ Pump Discharge Valve prevent flow through the affected Circ Water Pump. Without minimum flow protection, continued operation of the affected pump could lead to damage and system leakage from the pump seals. The affected pump must be stopped. Pressing closed on the Circ Pump 2 causes the Circ Pump 2 Discharge valve to move to the throttle position which is required for a single pump operating in a Circ Water Loop.

C. Incorrect because this is not the procedurally directed actions for the plant conditions provided – Plausible because opening the Waterbox Crossover Valve would allow more balance flow between Circ Water Loop 1 and 2. This is incorrect DB-OP-02517 will direct the Waterbox Crossover Valve to be closed when a pump trips.

D. Incorrect because this is not the procedurally directed actions for the plant conditions provided – Plausible because the Circ Pump itself continues to operate. If the candidate does not recognize that the pump requires minimum flow to remove heat generate by the pump, the candidate may assume that continued operation is acceptable. There are a number of other pumps in the plant that are capable of being operated without flow into their system such as the LPI pumps when RCS Pressure is greater than 200 psig or the HPI Pumps when RCS Pressure is greater than 1600 psig. These pumps have recirc flowpaths to allow continued operation. The Circ Water System does not have this capability.

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

Davis-Besse 1LOT22 NRC Written Exam

K/A#	A2.02	**K/A Importance**	2.5	**Exam Level:**	RO
References provided to Candidate:	NONE			**Technical References:**	DB-OP-02517 R09, Circ Water System Malfunctions, Attachment 1, Shutdown of a Loop 1 Pump
Question Cognitive Level:	High			**10 CFR Part 55 Content:** (CFR: 41.5 / 43.5 / 45.3 / 45.13)	
Question Source:	NEW			**Level Of Difficulty: (1-5)**	3
Objective:	OPS-GOP-117-05K				
Tier being Tested:	ES401 Tier 2/Group 2 (RO)				

66. As directed by NOP-OP-1002, Conduct of Operations, a plant announcement on the Gaitronics system is required to alert station personnel to stand clear of switchgear if a pump or other load is being started on which of the following switchgear?

A. DC MCC 2

B. D2

C. E2

D. Y2

Answer: B

Explanation/Justification: This question is a match for the KA because it tests the candidate's knowledge of the procedurally directed requirement to announce plant status changes on 4kV and above switchgear. As directed by NOP-OP-1002, Conduct of Operations, step 4.1.8 - Reactor Operator (RO) At The Controls (ATC) and Balance of Plant Operator (BOP) Responsibilities, Step 4.1.8.18 - "These announcements should include standing clear of any switchgear greater than or equal to 4 kV affected by the starting/stopping of equipment".

A. Incorrect because the voltage for this bus is less than the voltage specified for announcements by NOP-OP-1002 – Plausible because there are large loads on DC MCC 2, but the operating voltage is 250 volts which is less than the 4 kV specified by NOP-OP-1002, Conduct of Operations.

B. CORRECT - As directed by NOP-OP-1002, Conduct of Operations, step 4.1.8. - Reactor Operator (RO) At The Controls (ATC) and Balance of Plant Operator (BOP) Responsibilities step 18, These announcements should include standing clear of any switchgear greater than or equal to 4 kV affected by the starting/stopping of equipment.

C. Incorrect because the voltage for this bus is less than the voltage specified for announcements by NOP-OP-1002 – Plausible because there are large loads on E2, but the operating voltage is 480 VAC which is less than the 4 kV specified by NOP-OP-1002, Conduct of Operations.

D. Incorrect because the voltage for this bus is less than the voltage specified for announcements by NOP-OP-1002 – Plausible because there are essential loads on Y2, but the operating voltage is 120 VAC volts which is less than the 4 kV specified by NOP-OP-1002, Conduct of Operations.

Sys #	System	Category	KA Statement
Gen	1	Conduct of Operations	Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc.

K/A# 2.1.14 **K/A Importance** **Exam Level:** RO

References provided to Candidate: None **Technical References:** NOP-OP-1002 R16, Conduct of Operations, Step 4.1.8.18

Question Source: NEW **Level Of Difficulty: (1-5)** 2.5

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.12)

Objective: OPS-GOP-501-03K

Tier being Tested: ES 401 Generic Knowledge and Abilities Outline (Tier 3) RO

67. Which of the following statements is correct concerning Night Orders?

    In accordance with NOP-OP-1002, Conduct of Operations,
    Night Orders _____________.

A. should be reviewed back to your last shift worked

B. are used to provide post-maintenance testing requirements on non-safety related equipment

C. list Potential Limiting Conditions for Operation action statements associated with planned maintenance

D. are utilized to communicate long term issues and should not be in place longer than a fuel cycle

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Answer: A

Explanation/Justification: KA Match: This question matches the KA by requiring knowledge of the administrative requirements for the use of night orders. NOP-OP-1002, Conduct of Operations provides direction for the use of Night Orders including review requirements. This is an administrative requirement for the Night Orders.

A. CORRECT - IAW NOP-OP-1002, Night Orders should be reviewed back to your last shift worked.

B. Incorrect because NOP-OP-1002, Conduct of Operations specifically states, "Night orders shall not be used to provide post-maintenance testing requirements."
Plausible because Night Orders are used to communicate information that provides emphasis on interpretation or direction. This activity could be considered a direction.

C. Incorrect because this use of night orders is not directed by NOP-OP-1002, Conduct of Operations – Plausible potential limiting conditions for operations are provided as part of the packages for planned maintenance, but these are not provided via the night orders. The candidate may select this response if they know that potential limiting conditions for operation are provided and incorrectly assume that they are provided via night orders.

D. Incorrect because this use of night orders is not directed by NOP-OP-1002, Conduct of Operations.
Plausible because this purpose applies to Standing Orders, another administrative process for providing additional information to the operators.

Sys #	System	Category	KA Statement
GEN	1	Generic	Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

K/A# 2.1.15 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-OP-1002 R15, Conduct of Operations, Step 4.15.2 & 4.13.2.7

Question Source: New **Level Of Difficulty: (1-5)** 2

Question Cognitive Level: Low **10 CFR Part 55 Content:** 41.10 / 45.12

Objective: OPS-GOP-528-03K

Tier being Tested: ES 401 Generic Knowledge and Abilities Outline (Tier 3) RO

68. Refueling operations in accordance with DB-OP-00030, Fuel Handling Operations, are in progress.

    Which ONE of the following requires specific permission from BOTH the Shift Manager and the Fuel Handling Director (SRO)?

A. Inserting a control rod assembly into the core.

B. Removing a control rod from fuel assembly in the Spent Fuel Pool up-ender basket.

C. Disengaging the grapple from a fuel assembly in the Refueling Canal.

D. Placing TS-1 FUEL HOIST OVERLOAD BYPASS in BYPASS during fuel movement.

Answer: D

Explanation/Justification: This question is a KA match because it tests the Candidate's knowledge of the procedures (DB-OP-00030, Fuel Handling Operation) and the Limitations (step 6.13.1) for handling fuel. While granting permission is an SRO Activity, the actual bypasses would be operated by fuel handling operator which are or could be performed by a Reactor Operator Qualified individual. The RO must understand the requirements for operating any Equipment Bypass Switches.

A. Incorrect because approval of both the Shift Manager and Fuel Handling Director (SRO) is not required per DB-OP-00030, Fuel Handling Operations.
Plausible since the SM provides oversight function to ensure the refueling activities are conducted in a safe manner in accordance with Technical Specification requirements. The Fuel Handling Director provides direct supervision of all Fuel Handling activities during the performance of DB-NE-06101, Fuel/Control Component Shuffle.

B. Incorrect because approval of both the SHIFT MANAGER and Fuel Handling Director (SRO) is not required per DB-OP-00030, Fuel Handling Operation.
Plausible since the SM provides oversight function to ensure the refueling activities are conducted in a safe manner in accordance with Technical Specification requirements. The Fuel Handling Director provides direct supervision of all Fuel Handling activities during the performance of DB-NE-06101, Fuel/Control Component Shuffle.

C. Incorrect because approval of both the SHIFT MANAGER and Fuel Handling Director (SRO) is not required per DB-OP-00030, Fuel Handling Operation.
Plausible since the SM provides oversight function to ensure the refueling activities are conducted in a safe manner in accordance with Technical Specification requirements. The Fuel Handling Director provides direct supervision of all Fuel Handling activities during the performance of DB-NE-06101, Fuel/Control Component Shuffle.

D. CORRECT - DB-OP-00030, Fuel Handling Operation) and the Limitations (step 6.13.1) for handling fuel requires direct permission of both the Shift Manager and the Fuel Handling Director (SRO) prior to bypassing any install Equipment Bypasses. TS-1 is the Bypass Switch for the Fuel Hoist overload limit.

Sys #	System	Category	KA Statement
Gen	1	Conduct of Operations	Knowledge of procedures and limitations involved in core alterations.

K/A# 2.1.36 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-00030 R15, Fuel Handling Operation), Limitations (step 6.13.1)

Question Source: TMI 2014 NRC Q68 exam modified to match DB terminology and procedures. **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.10 / 43.6 / 45.7)

Objective: OPS-FHT-104-03K

Tier being Tested: ES 401 Generic Knowledge and Abilities Outline (Tier 3) RO

69. In accordance with the guidance provided in NOP-SS-3001, Procedure Review and Approval, which of the procedure changes, listed below, CAN be made by using the "Procedure Correction" method?

Changes to _________ CAN be made by using the "Procedure Correction" method.

A. Maintenance Rule availability

B. equipment location

C. equipment position

D. setpoints

Answer: **B**

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the process for making changes to procedures. A procedure Correction is one of the methods specified by NOP-SS-3001, Procedure Review and Approval.

A. Incorrect because changes to Maintenance Rule Availability requires the procedure change to be processed as a Significant Change as directed by NOP-SS-3001, Procedure Review and Approval Section 4.2.1.1 Significant Change.
Plausible because changing Maintenance Rule availability as directed by a procedure would require a procedure alteration however, this change must be processed as a significant change, not a procedure correction.

B. CORRECT – NOP-SS-3001, Procedure Review and Approval Process Step 4.2.1.3 provides that procedure correction maybe used to correct the location of a system, structure, or component.

C. Incorrect because changes to a procedure specified equipment position requires the procedure change to be processed as a Significant Change as directed by NOP-SS-3001, Procedure Review and Approval Section 4.2.1.1 Significant Change.
Plausible because changing equipment position as directed by a procedure would require a procedure alteration however, this change must be processed as a significant change, not a procedure correction.

D. Incorrect because changes to procedure setpoints requires the procedure change to be processed as a Significant Change as directed by NOP-SS-3001, Procedure Review and Approval Section 4.2.1.1 Significant Change.
Plausible because changing setpoints as directed by a procedure would require a procedure alteration however, this change must be processed as a significant change, not a procedure correction.

Sys #	System	Category	KA Statement
GEN	2	Equipment Control	Knowledge of the process for making changes to procedures.

K/A# 2.2.6 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-SS-3001 R23 page 8 Step 4.2.1.3

Question Source: NEW **Level Of Difficulty: (1-5)** 3.5

Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** (CFR: 41.10 / 43.3 / 45.13)

Objective: OPS-GOP-515-02K

Tier being Tested: ES 401 Generic Knowledge and Abilities Outline (Tier 3) RO

70. A component is being tracked in the Unit Log as out of normal alignment per the Short-Term Configuration process.

If the component will not be restored to its Normal Configuration within

(1) ______ hours,

(2) a(n)_______ shall be hung.

A. (1) 12
(2) Caution Tag

B. (1) 12
(2) Operations Information Tag

C. (1) 24
(2) Caution Tag

D. (1) 24
(2) Operations Information Tag

Answer: C

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of plant status/configuration control requirements as provided by NOP-OP-1002, Conduct of Operation.

A. Incorrect
- (1) Incorrect because the duration of the allow time is 24 hours, not 12 hours – plausible because 12 hours is the length of a shift.
- (2) Correct as directed by NOP-OP-1014.

B. Incorrect
- (1) Incorrect because the duration of the allow time is 24 hours, not 12 hours – plausible because 12 hours is the length of a shift.
- (2) Incorrect because the tag used to identify the out of position component is incorrect – plausible because NOP-OP-1014 directs the use of Operations Information Tags for other situations.

C. CORRECT
- (1) Correct per NOP-OP-1014, Plant Status Control, if a component will not be restored to its Normal Configuration within 24 hours,
- (2) Correct a Caution Tag is hung IAW NOP-OP-1001, Clearance/Tagging Program for components not restored within 24 hours.

D. Incorrect
- (1) Correct per NOP-OP-1014, Plant Status Control, if a component will not be restored to its Normal Configuration within 24 hours,
- (2) Incorrect because the tag used is the caution tag, not an operations information tag – plausible because NOP-OP-1014 directs the use of Operations Information Tags for other situations.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the process for controlling equipment configuration or status

K/A# 2.2.14 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-OP-1014 R07, Note 4.6 on page 12 & Step 4.6.6

Question Source: NRC Exam DB 2016 Q69 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: Low **10 CFR Part 55 Content:** 41.10 /43.3 / 45.13

Objective: OPS-GOP-505-02K

Davis-Besse 1LOT22 NRC Written Exam

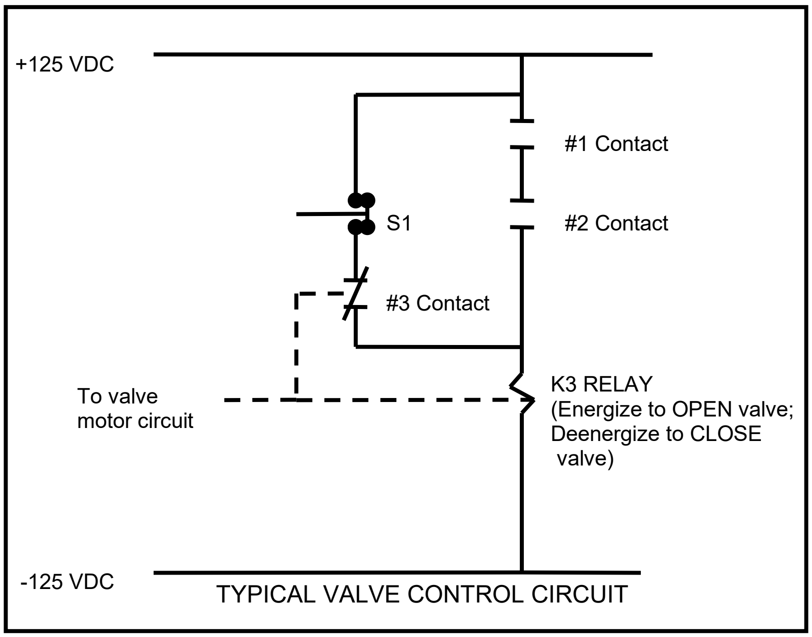
Tier being Tested: ES 401 Generic Knowledge and Abilities Outline (Tier 3) RO

71. Refer to the drawing of a typical valve control circuit for a 480 VAC motor-operated valve shown deenergized (see figure below).

(1) With **NO** initiating condition present (Contact 1 and Contact 2), the valve is currently OPEN.
If the S1 pushbutton is depressed, the valve will ___________

AND

(2) when the S1 pushbutton is subsequently released (spring return) the valve will ____________.



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

B. (1) close
(2) remain closed

C. (1) remain open
(2) close

D. (1) close
(2) open

Answer: B

Explanation/Justification This question is a KA match because it tests the candidate's ability to interpret a typical 480-volt valve control circuit.

When the S1 is depressed, it opens the S1 contact causing the K3 relay to deenergize.
De-energizing the K3 relay causes the valve to close.
Deenergizing the K3 relay also causes the #3 contact to open.
With K3 open when S1 is released and spring returns to the closed position, the open #3 contact prevents the K3 relay from re-energizing and the valve remains closed.
Contacts #1 and #2 are controlled by a different unspecified relay.

A. Incorrect.
- (1) Incorrect because the position is wrong. - Plausible if the applicant believes contacts #1 and #2 will allow the circuit to remain energized.
- (2) Incorrect because the position is wrong. - Plausible if the applicant believes contact #3 remains closed when S1 is depressed due to contacts #1 and #2 changing state.

B. CORRECT
- (1) Correct – When the S1 is depressed, it opens the S1 contact causing the K3 relay to deenergize. De-energizing the K3 relay causes the valve to close.
- (2) Correct - Deenergizing the K3 relay also causes the #3 contact to open. With K3 open when S1 is released and spring returns to the closed position, the open #3 contact prevents the K3 relay from re-energizing and the valve remains closed.

C. Incorrect.
- (1) Incorrect because the position is wrong. - Plausible if the applicant believes contacts #1 and #2 will allow the circuit to remain energized.
- (2) Incorrect because the valve will remain closed. - Plausible since if the candidate believes the contact #3 opens when S1 is depressed and contacts #1 and #2 open when S1 spring returns to its initial position.

D. Incorrect.
- (1) Correct
- (2) Incorrect because the position is wrong. - Plausible if the applicant believes contact #3 will close when S1 spring returns to its closed position.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to obtain and interpret station electrical and mechanical drawings.

K/A# 2.2.41 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** Lesson Plan PWR Generic Fundamentals Components – Breakers, Relays, & Disconnects pg 20

Question Source: DB 2011 NRC Exam Q70 **Level Of Difficulty: (1-5)** 3.5

Question Cognitive Level: High - Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 45.12 / 45.13)

Objective: OPS-SYS-031-02

Tier being Tested: ES 401 Generic Knowledge and Abilities Outline (Tier 3) RO

72. Which of the following are actions an operator is REQUIRED to PERFORM prior to NORMAL USE of a portable radiation survey instrument per DBBP-RP-1007 Meter Source and Response Testing?

(1) Perform an instrument ________

AND

(2) Make an entry in the _________.

A. (1) Calibration
(2) Use/Response Log

B. (1) Calibration
(2) Daily Source Check Log

C. (1) Response Check
(2) Use/Response Log

D. (1) Response Check
(2) Daily Source Check Log

Answer: C

Explanation/Justification: This question is a KA match because it tests the knowledge the candidate needs in order to use portable radiation survey instruments.

A. Incorrect
(1) Incorrect because Calibration is not performed by operator, just checked current by reviewing the calibration sticker. See DBBP-RP-1007 R46 Meter Source and Response Testing step 3.2.1.1. Plausible because the calibration must be current to use the meter.
(2) Correct as required by DBBP-RP-1007 R46 Meter Source and Response Testing.

B. Incorrect
(1) Incorrect because Calibration is not performed by operator, just checked current using the sticker. Plausible because daily source check log entry is required for daily source check.
(2) Incorrect because the Daily Source Check log entry is not performed by the Operator, only used to confirm the DBBP-RP-1007 R46 Meter Source and Response Testing required checks have been performed.

C. CORRECT –
(1) Correct - Response check required per DBBP-RP-1007 R46 Meter Source and Response Testing step 3.2.2.1.
(2) Correct - Use/Response Log entry required per step 3.2.2.1.H

D. Incorrect
(1) Correct - Response check required per DBBP-RP-1007 R46 Meter Source and Response Testing step 3.2.2.1.
(2) Incorrect because Source Check Log entry not made is wrong because operator does not perform the source check. Plausible because response check and source check are required prior to use.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

K/A# 2.3.5 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** DBBP-RP-1007 R46 Meter Source and Response Testing steps 3.2.2. and 3.2.2.1.H

Question Source: DB1LOT15 Q72 **Level Of Difficulty: (1-5)** 3

Davis-Besse 1LOT22 NRC Written Exam

Question Cognitive Level: Low – Memory

10 CFR Part 55 Content: (CFR: 41.11 / 41.12 / 43.4 / 45.9)

Objective: OPS-GOP-511-02K

Tier being Tested: ES 401 Generic Knowledge and Abilities Outline (Tier 3) RO

73. Plant Conditions:
    - The plant is in Mode 6 with Fuel Handling operations in progress.
    - The Containment Purge Ventilation system is in operation on Containment.
    - The Containment Equipment Hatch is ON.

    The Main Fuel Bridge Operator in Containment reports the fuel mast with an irradiated fuel assembly inside has hit an obstruction in the Refueling Canal.

    Annunciator 9-1-G FIRE OR RADIATION TROUBLE alarms.

    A review of the Radiation Monitoring System notes the following:
    - High alarm on RE 5052A, CTMT PURGE EXH FAN IN, PARTICULATE
    - High alarm on RE 5052B, CTMT PURGE EXH FAN IN, I-131
    - High Alarm on RE 5052C, CTMT PURGE EXH FAN IN, XE-133
    - High Alarm on RE4598AA, STATION VENT RADIATION MONITOR
    - High Alarm on RE4598AB, STATION VENT RADIATION MONITOR

    Which of the following describes the Automatic and required Manual actions for the Control Room Area Ventilation Systems for this event in accordance with DB-OP-02530, Fuel Handling Accident?

    (1) The Control Room Normal Ventilation System will be ______________

    AND

    (2) The Control Room Emergency Ventilation System will be ___________.

A. (1) Manually Shutdown
   (2) Manually Started

B. (1) Automatically Shutdown
   (2) Manually Started

C. (1) Manually Shutdown
   (2) Automatically Started

D. (1) Automatically Shutdown
   (2) Automatically Started

Answer: B

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of operator duties in response to a Fuel Handling Accident in accordance with DB-OP-02530, Fuel Handling Accident. The actions for all scenarios follow a general theme. Evacuate the area, isolate the normal ventilation system, start the Emergency Ventilation System to control the radiation release, start Control Room Emergency Ventilation to protect the Control Room crew, and develop a recovery plan. The procedure is divided into three sections. A Fuel Handling Accident in Containment with the Equipment Hatch on, Fuel Handling Accident in Containment with the Equipment Hatch off, and a Fuel Handling Accident in the Spent Fuel Pool Area.

For an accident in containment that causes a high radiation alarm on station vent radiation monitors the Control Room normal Ventilation System will Shutdown automatically and isolate the Control Room. DB-OP-02530 directs confirmation that the shutdown and Isolation occurred correctly using Attachment 2. The procedure then directs manually starting the Control Room EVS system using Attachment 3. This system must be started manually for the events described.

A. Incorrect

1) Incorrect because the Control Room Normal Ventilation System will automatically shutdown and isolate the Control Room on a high radiation event at the station vent. Plausible because the manual start of the Control Room Emergency Ventilation System is directed by DB-OP-02530.
2) Correct as directed by DB-OP-02530. The Control Room Emergency Ventilation System must be manually started for the events described.

B. CORRECT

1) Correct IAW DB-OP-02530, Fuel Handling Accident step 4.1.3 direct confirmation of automatic shutdown of Control Room Normal Ventilation System.
2) Correct IAW DB-OP-02530, Fuel Handling Accident Step 4.1.4 directs manual start of the CR Emergency Ventilation System.

C. Incorrect

1) Incorrect because the Control Room Normal Ventilation System will automatically shutdown and isolate the Control Room on a high radiation event at the station vent. Plausible because the manual start of the Control Room Emergency Ventilation System is directed by DB-OP-02530.
2) Incorrect IAW DB-OP-02530, Fuel Handling Accident Step 4.1.4 directs manual start of the CR Emergency Ventilation System. Plausible because operation of the system is directed in response to a fuel handling accident.

D. Incorrect

1) Correct IAW DB-OP-02530, Fuel Handling Accident step 4.1.3 direct confirmation of automatic shutdown of Control Room Normal Ventilation System.
2) Incorrect IAW DB-OP-02530, Fuel Handling Accident Step 4.1.4 directs manual start of the CR Emergency Ventilation System. Plausible because operation of the system is directed in response to a fuel handling accident.

Sys #	System	Category	KA Statement
3	GEN	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** 3.4 **Exam Level:** RO

References provided to Candidate: None **Technical References:** DB-OP-02530 R13 Fuel Handling Accident steps 4.1.3 and 4.1.4, USAR 9.4.1.3 Safety Evaluation

Question Source: DB Exam Bank 75003 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.12 / 43.4 / 45.9 / 45.10

Objective: OPS-GOP-135-01K

Tier being Tested: ES 401 Generic Knowledge and Abilities Outline (Tier 3) RO

74. Initial Conditions:

- Reactor power is at 100%
- The Command SRO is in the Control Room Cabinet Area
- The Shift Manager and Shift Engineer are not in the CTRM

An event occurs, and the following indications are observed:

- (14-2-D) ICS/NNI 118V AC PWR TRBL
- The Main Feedwater Control BLOCK Valves begin to close

Which of the following are the required Immediate Action(s) of the At The Controls Reactor Operator?

A. Notify the Command SRO that a plant transient is in progress.

B. Trip the Reactor AND Initiate SFRCS using Manual Actuation Switches.

C. Trip the Reactor and Initiate AND Isolate SFRCS using Manual Actuation Switches.

D. Place the Rod Control in MANUAL and BOTH Feedwater Loop Demand Hand/Auto Stations to Hand and stabilize the plant.

Answer: C

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of ATC responsibilities for taking actions during plant transients that require entry into the EOP.

NOP-OP-1002 4.10.2 Step 18: All immediate operator actions in AOPs/ONIs/EOPs shall be committed to memory.

NOP-OP-1002 4.10.3 Step 6: Initiate a manual reactor trip when a situation exists which jeopardizes or threatens to jeopardize public or plant safety, an operating parameter reaches trip criteria, or an automatic reactor trip should have occurred.

NOP-OP-1002 4.1.8 Step 8: Initiate a manual reactor trip when in his/her judgment a situation exists which jeopardizes or threatens to jeopardize public or plant safety, an operating parameter reaches trip criteria, or an automatic reactor trip should have occurred.

DB-OP-01003 6.5.1
a. The Immediate Actions stated in Abnormal Procedures addressing normal Reactor Trips shall be committed to memory by Licensed Operators.
b. When the Abnormal Procedure is entered, the Reactor Operators shall complete the Immediate Actions.

DB-OP-02532 LOSS OF NNI/ICS POWER
3.5 Immediate Actions - Loss of ICS AC Power, DC Power, or Both
3.5.1 Trip the Reactor.
3.5.2 Initiate and Isolate SFRCS using Manual Actuation Switches.
3.5.3 GO TO DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture AND return to this procedure, Subsection 4.5, Supplemental Actions – Loss of ICS AC Power, DC Power, or Both, as conditions permit.
.

A. Incorrect because the conditions provided meet the entry conditions for an Abnormal Operating Procedure with Immediate Operator Actions, the ATC Operator is required to perform the immediate operator actions from Memory, not just notify the Command SRO.
Plausible because notification of the Command SRO of changes in plant status is required by Conduct of Operations, however since the condition provided meet the entry conditions for an Abnormal Operating Procedure, the ATC Operator is required to perform the immediate operator actions from Memory, not just notify the Command SRO.

B. Incorrect because the conditions provided meet the entry conditions for an Abnormal Operating Procedure with Immediate Operator Actions, the ATC Operator is required to perform the immediate operator actions from Memory. The immediate operator action to initiate SFRCS is incorrect. For this event, SFRCS is required to be Initiated and ISOLATED.
Plausible because the Immediate Operator Actions do require manual operations of the Steam Feed Rupture Control System.

C. CORRECT – The indications given for this scenario are for a Loss of ICS Power. The Immediate Actions for this AOP are to Trip the Reactor, Initiate and Isolate SFRCS using Manual Actuation Switches, and GO TO DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture. All immediate operator actions in AOPs and EOPs shall be committed to memory as described in DB-OP-01003, step 6.5.1. The operator shall know the actions (including results not obtained) well enough to complete the intent of each step. When the Emergency Operating Procedure (EOP) or Abnormal Procedure is entered, the Reactor Operators shall complete the Immediate Actions. The Command SRO shall then verify immediate actions have been completed properly by referring to the applicable procedure.

D. Incorrect because the conditions provided meet the entry conditions for an Abnormal Operating Procedure with Immediate Operator Actions, the ATC Operator is required to perform the immediate operator actions from Memory. The immediate operator action is to trip the reactor and initiate and isolate SFRCS, not to attempt to stabilize the plant.
Plausible because these are the required actions for DB-OP-02526, Primary to Secondary Heat Transfer Upset which the RO could incorrectly diagnose with the given conditions. These conditions would cause an upset in Primary to Secondary Heat Transfer.

Sys #	System	Category	KA Statement
4	GEN	Emergency Procedures/Plans	Knowledge of crew roles and responsibilities during EOP usage.

K/A# 2.4.13 **K/A Importance** 4.0 **Exam Level:** RO

References provided to Candidate: None

Technical References: DB-OP-01003 R16, Operations Procedure Use Instructions, Step 6.5.1 a. and b.
NOP-OP-1002 R16 4.10.2 step 18
4.1.8 step 8
DB-OP-02532 R16 Step 2.5.1 & Section 3.5
DBBP-TRAN-0034 R16 page 15

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.10 / 45.12)

Objective: OPS-GOP-302-01K

Tier being Tested: ES 401 Generic Knowledge and Abilities Outline (Tier 3) RO

75. Which ONE of the following describes the offsite notification requirements of the Davis-Besse Emergency Plan when implemented?

State and local counties _____________ of any emergency classification.

A. AND the NRC must be notified within 15 minutes

B. AND the NRC must be notified within 1 hour

C. ONLY, must be notified within 1 hour
WHEREAS, the NRC must be notified within 15 minutes

D. ONLY, must be notified within 15 minutes
WHEREAS, the NRC must be notified within 1 hour

Answer: D

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the techniques for completing Emergency Plan Notification requirements by demonstrating the time allowed to complete the required notifications. Reactor Operators can be tasked with completing the actual notification calls depending on the work load involved in responding to plant events. Some E-Plan entries may not require any specific action by a Reactor Operator to maneuver the plant.

A. Incorrect because the time for notifying the NRC is not correct – Plausible because the notification of the State and Local is the correct 15 minutes and the NRC should be notified as soon as possible, but they must be notified within 1 hour, not 15 minutes

B. Incorrect because the time to notify the State and Counties is incorrect – Plausible because the NRC must be notified within 1 hour. State and Local must be notified within 15 minutes, but 1 hour is plausible since the NRC has one hour to be notified.

C. Incorrect because the times to notify the state and counties and the NRC are wrong – Plausible because the notification of the State and Local and the NRC are reversed.

D. CORRECT – These times are the limits provide by RA-EP-02110, Emergency Notification for entry into the Emergency Plan.
RO Knowledge - OPSGOP603-05K

Sys #	**System**	**Category**	**KA Statement**
4	GEN	Emergency Procedures/Plans	Knowledge of emergency communications systems and techniques.

K/A# 2.4.43 **K/A Importance** 3.2 **Exam Level:** RO

References provided to Candidate: None **Technical References:** RA-EP-02110 R19, Emergency Notifications, Note 6.3.1 and Step 6.5.2
Lesson plan OPSGOPI603

Question Source: Bank DB Requal Q38381 **Level Of Difficulty: (1-5)** 2.5

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.10 / 45.13)

Objective: OPSGOP603-05k

Tier being Tested: ES 401 Generic Knowledge and Abilities Outline (Tier 3) RO

76. Initial plant conditions:
    - The Reactor is at 100% power
    - A Small Break Loss of Coolant Accident has occurred

    Current plant conditions:
    - Letdown has been isolated
    - Standby Makeup Pump has been started
    - Pressurizer Level Control Valve MU 32 is full open
    - Both Trains of HPI and LPI have been started in Piggyback mode
    - Pressurizer Level is 200 inches and lowering at 10 inches per minute

    Based on the current plant conditions, complete the following statements.

    (1) A Reactor Trip will be **REQUIRED** in _________ minutes.

    AND

    (2) The correct Emergency Action Level Classification is ___________.

A. (1) 4 minutes
   (2) Unusual event

B. (1) 4 minutes
   (2) Alert

C. (1) 10 minutes
   (2) Unusual event

D. (1) 10 minutes
   (2) Alert

Answer: D

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the ability to interpret conditions in the RCS with a small break RCS Leak, understand the procedurally directed actions that provide direction for this event, then use that information to determine when a Reactor Trip will be required and then demonstrate understanding of the bases for that trip setpoint. In the case provided, tripping the reactor at 100 inches in the Pressurizer to prevent emptying the Pressurizer Surge Line on the Reactor Trip.

SRO Only: This is an SRO level question because it requires selection of the correct procedure to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction or reason for that action to mitigate the event. For this question, the controlling procedure is not provided in the stem of the procedure. The candidate must know that DB-OP-02000 is the controlling procedure to know the required pressurizer level trip setpoint. Other potential Abnormal Operating Procedures for this event would be DB-OP-02522, Small RCS Leaks and DB-OP-02531, Steam Generator Tube Leak.

This is an also an SRO level question because it tests the duties and responsibilities of the Shift Manager – Classification of Emergency Plan Events. NUREG 1021 Section D.2.d - Questions related to 10 CFR 55.41(b) topics may also be appropriate SRO-level questions if they evaluate K/As at a level that is unique to the SRO job position. This question is unique to the Shift Manager position which can only be performed by an SRO.

A. Incorrect
- (1) Incorrect because the time is wrong – Plausible because in 4 minutes, Pressurizer level will be 160 inches. 160 inches is the level the reactor would be manually tripped on a loss of both MUPs. It is plausible that the ability to maintain RCS pressure via Pressurizer Heaters would be desired to maintain Subcooling Margin during a SGTR that would be causing RCS Pressure to lower.
- (2) Incorrect - While the event meets classification level in the Emergency Plan, the level is incorrect. This event would be an Alert per RA-EP-1500, Emergency Classification. Plausible because the event is classifiable.

B. Incorrect
- (1) Incorrect because the time to the trip is wrong – Plausible because in 4 minutes, Pressurizer level will be 160 inches. 160 inches is the level the reactor would be manually tripped on a loss of both MUPs. The emptying of the Surge Line is correct per the Bases and Deviation Document for DB-OP-02000 Step 8.2.
- (2) Correct - The Classification level of Alert is correct per RA-EP-01500, Emergency Classification.

C. Incorrect
- (1) Correct because in 10 minutes, Pressurizer level will be 100 inches. This is the level directed by DB-OP-02000 to trip the reactor for DB-OP-02522, Small RCS Leaks.
- (2) Incorrect because the classification level is wrong. The Classification level of Unusual Event is not correct per RA-EP-01500, Emergency Classification.

D. CORRECT –
- (1) Correct because in 10 minutes, Pressurizer level will be 100 inches. This is the level directed by DB-OP-02522, Small RCS Leaks. The bases for this setpoint as provided in the Bases and Deviation Document for DB-OP-02000 step 8.2 is to prevent emptying the Pressurizer Surge Line following the Reactor Trip. This procedure action to trip at 100 inches is also used in response to Small RCS Leaks as directed by DB-OP-02522, Small RCS Leaks.
- (2) Correct - The Classification level of Alert is correct per RA-EP-01500, Emergency Classification.

Sys #	System	Category	KA Statement
009	Small Break LOCA	Ability to determine or interpret the following as they apply to a small break LOCA:	Whether PZR water inventory loss is imminent

K/A# EA2.06 **K/A Importance** 4.3 **Exam Level:** SRO

References provided to Candidate: None **Technical References:** DB-OP-02000 R32, Step 8.2 and Base and Deviation Document for DB-OP-02000 for Step 8.2
DB-OP-02522, Small RCS Leaks R19, Step 3.1.
RA-EP-01500 R16, Step 6.1.2.e and D, RA-EP-01500 Classification Wallboard (DBRM-EMER-1500B)

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR 43.5 / 45.13)

Objective: OPS-GOP-131-07K and OPS-GOP-307-05K

Tier being Tested: ES 401 Tier1/Group 1 (SRO)

77. The Reactor is operating at 100% power.

An event occurs which causes the following Annunciators:

- Annunciator 6-2-A, 1-1 SEAL RET TEMP HI
- Annunciator 6-5-C, SEAL INJ FLOW LO
- Annunciator 6-6-C, SEAL INJ TOTAL FLOW

Which of the following identifies the required Section of DB-OP-02515, Reactor Coolant Pump and Motor Abnormal Operations to respond to this event?

A. Section 4.1, RCP Seal Failure

B. Section 4.2, Loss of Seal Injection Water

C. Section 4.3, Loss of Component Cooling Water to All RCPs

D. Section 4.5, Loss of RCP Seal Return Flow

Answer: B

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the symptoms associated with Reactor Coolant Pump Abnormal Operating procedure. This procedure has 6 sections to respond to a variety of potential malfunctions. The question tests the candidate's ability select the appropriate section of the procedure based on Annunciator Alarms only.

SRO Only: This is an SRO level question because it requires selection and coordination of the correct procedures to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction or reason for that action to mitigate the event.

A. Incorrect because the wrong section is specified for the indications provided – Plausible because an RCP Seal Failure will cause multiple Annunciator Alarms including 6-2-A Seal Return Temperature High, but the other alarms are not symptoms of an RCP Seal Failure.

B. CORRECT – These Annunciators Alarms are the specific Annunciator alarms for a loss of Seal Injection Water as provided by DB-OP-02515, Section 2.2, Loss of Seal Injection Water. Additional alarms may occur depending on the cause of loss of seal injection flow. Also, additional seal return flow temperature alarms may occur.

C. Incorrect because the wrong section is specified for the indications provided – Plausible because a loss of Component Cooling Water will cause multiple Annunciator Alarms including 6-2-A Seal Return Temperature High.

D. Incorrect because the wrong section is specified for the indications provided – Plausible because the student may assume that a loss of RCP Seal return flow could lead to Annunciator 6-2-A, 1-1 SEAL RET TEMP HI. A low flow condition could cause temperature to rise.

Sys #	System	Category	KA Statement
015	RCP Malfunctions	Generic	Knowledge of annunciator alarms, indications, or response procedures.

K/A# G2.4.31 **K/A Importance** 4.1 **Exam Level:** SRO

References provided to Candidate: None **Technical References:** DB-OP-02515 R18, Reactor Coolant Pump Abnormal Operation, Section 2 Symptoms

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.10 / 45.3)

Objective: OPS-GOP-115-01K

Tier being Tested: ES 401 Tier1/Group 1 (SRO)

78. Initial Plant Conditions:
    - A plant cooldown is in progress
    - The plant is in Mode 5
    - Reactor Coolant System temperature steady at 180°F
    - Reactor Coolant System Pressure at 200 psig
    - Decay Heat Removal Loop 2 is in service at 3000 gpm
    - Decay Heat Train 1 is aligned for Low Pressure Injection
    - Briefings are in progress to align LPI Train 1 for Standby DHR operation

    An event occurs. The following indications are noted:
    - Pressurizer Level LI RC14 is lowering
    - Annunciator 4-1-E PZR LO LVL HTR TRIP alarms
    - Annunciator 11-1-A CCW RETURN RAD HI alarms
    - Annunciator 11-4-A CCW SURGE TK LVL HI alarms
    - LI 1402, CCW SURGE TANK LEVEL SIDE 1 is rising
    - LI 1403, CCW SURGE TANK LEVEL SIDE 2 is rising

    Which ONE of the following procedures provides direction to mitigate these conditions while maintaining Decay Heat removal?

A. DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture

B. DB-OP-02513, Pressurizer Malfunctions

C. DB-OP-02522, Small RCS Leaks

D. DB-OP-02523, Component Cooling Water Malfunctions

Answer: D

Explanation/Justification: This question is a match for the KA because it tests the candidate's knowledge of an event where the Reactor Coolant system is leaking into the closed loop Component Cooling Water System via the Decay Heat Removal System. There is a leak in the Decay Heat Removal Cooler. Given the pressure in the RCS, this leak will be into the CCW System causing Surge Tank Level to rise. The candidate must demonstrate their ability to select the appropriate procedure to respond to this event. Further complexity is provided because the natural procedure to respond to events while on Decay Heat Removal (DB-OP-02527, Loss of Decay Heat Removal) is not provided as a possible response.

SRO Only: This is an SRO level question because it requires selection and coordination of the correct procedures to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction or reason for that action to mitigate the event.

A. Incorrect because this procedure is not applicable in the Mode presented by the question – Plausible because in Modes 1-4, the EOP DB-OP-02000 does provide direction to address large leaks from the Reactor Coolant System. Once the plant enters Mode 5, this procedure is no longer applicable.

B. Incorrect because this procedure does not provide the necessary direction to mitigate the event – Plausible because at normal operating temperatures and pressures, this procedure provides direction to mitigate the lowering Pressurizer Level and the annunciator for PZR Heater Trip.

C. Incorrect because this procedure does not provide the necessary direction to mitigate the event – Plausible because at normal operating temperatures and pressures, this procedure provides direction to mitigate the lowering Pressurizer Level and the annunciator for PZR Heater Trip.

D. CORRECT – The indications provided are indicative of a leak from the Reactor Coolant System into the Component Cooling Water System. This direction is provided by Section 4.2, Rising or High CCW Surge Tank Level, Step 4.2.5 IF intersystem leakage is suspected, THEN REFER TO Attachment 6, CCW Leak Location Determination Aid Section 1, Rising CCW Surge Tank Level as necessary to locate and isolate the leak. This procedure provides direction via Attachment 6, CCW Leakage Location Determination Aid for a leak from the DHR system into the CCW System. The procedure directs 1. Placing Standby DHR Train in service. 2. Isolating the affected DHR Cooler. REFER TO DB-OP-06012, DH and LPI System Operating Procedure.

Sys #	System	Category	KA Statement
025	Loss of Residual Heat Removal	Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System:	Leakage of reactor coolant from RHR into closed cooling water system or into reactor building atmosphere

K/A# AA2.02 **K/A Importance** 3.8 **Exam Level:** SRO

References provided to Candidate: None **Technical References:** DB-OP-02523 R13, Component Cooling Water Malfunctions, Section 2.2, Rising CCW Surge Tank Level. Step 4.2.5, and Attachment 6, CCW Leak Location Aid

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 43.5 / 45.13)

Objective: OPS-GOP-123-02K

Tier being Tested: ES 401 Tier1/Group 1 (SRO)

79. The Reactor is operating at 100% power following a power increase completed 50 hours ago.

After 50 hours, DB-CH-03000, Primary Coolant System Radiochemistry was completed with the following result:

Dose Equivalent I131	1.2 µCi/gm
RCS Gross Specific Activity	391 µCi/gm

The most recent calculated $\bar{E}$ (E bar) value is 0.1691 MEV.

(1) __________ is out of specification per TS 3.4.16,

AND

(2) the basis of the specification is to maintain dose within the limits in the event of a _________ .

A. (1) Dose Equivalent I131
(2) Steam Generator Tube Rupture

B. (1) Dose Equivalent I131
(2) Large Break Loss of Coolant Accident

C. (1) Reactor Coolant Gross Specific Activity
(2) Steam Generator Tube Rupture

D. (1) Reactor Coolant Gross Specific Activity
(2) Large Break Loss of Coolant Accident

Answer: A

Explanation/Justification: This question is a KA match because is tests the candidate's knowledge of the limitation in the facility license to respond to a Steam Generator Tube Rupture. As noted in the TS 3.4.16 Bases, The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to this analysis is used to assess changes to the facility that could affect RCS specific activity as they relate to the acceptance limits.

SRO Only: The assumed RCS specific activity in the SGTR analysis bounds the LCO limit for RCS specific activity. The LCO for TS 3.4.16 allows Dose Equivalent I131 to be in excess of the 1.0 µCi/gm, but only for 48 hours.

The RCS Gross Specific limit by TS 3.4.16 is 100 divided by E bar (0.1691) which is 591 µCi/gm. As a result, the value provided in the stem (391 µCi/gm) is less than the limit provided by TS 3.4.16.

This is an SRO level question per NURG 1021 Attachment 2 Section B Facility Operating Limitations in the Technical Specifications and Their Bases [10 CFR 55.43(b)(2)] because it tests the candidate's knowledge of the Bases for TS 3.4.16 which provides the following: The specific iodine activity is limited to 1.0 µCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the primary coolant is limited to the number of µCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2-hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose.

The limit on gross specific activity ensures the 2-hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole-body dose. The SGTR accident analysis (Ref. 2) shows that the 2-hour site boundary dose levels are within acceptable limits. Violation of the LCO such that the RCS specific activity is greater than the analysis assumptions, may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

A. CORRECT

(1) Correct – Only the Dose Equivalent Iodine 131 is in excess of the 1.0 µCi/gm allowed by TS3.4.16. Following a power change, DEI 131 may exceed the steady state limit for up to 48 hours. The 48 hours allowed has been used up as provided in the question stem.

(2) Correct - The bases for the limit as noted above is for a Steam Generator Tube Rupture, not a Large Break Loss of Coolant Accident.

B. Incorrect

(1) Correct - Dose Equivalent Iodine 131 is in excess of the 1.0 µCi/gm allowed by TS3.4.16. Following a power change, DEI 131 may exceed the steady state limit for up to 48 hours. The 48 hours allowed has been used up as provided in the question stem.

(2) Incorrect because the bases for the limit is a SGTR, not a Large Break LOCA – Plausible because following a Large Break Loss of Coolant Accident there is always some allowed leakage from Containment. Limiting the RCS Activity prior to the event would reduce the releases from Containment. As a result, the candidate may incorrectly select this response.

C. Incorrect

(1) Incorrect – The RCS Gross Specific Activity is within TS 3.4.16 limits and only the Dose Equivalent Iodine 131 is in excess of the 1.0 µCi/gm allowed by TS3.4.16. A candidate may incorrectly select this response based on incorrect understanding of the RCS Activity Limits

(2) Correct because the Accident of concern is a SGTR, not a LB LOCA as provide in the bases for TS3.4.16.

D. Incorrect

(1) Incorrect – The RCS Gross Specific Activity is within TS 3.4.16 limits and only the Dose Equivalent Iodine 131 is in excess of the 1.0 µCi/gm allowed by TS3.4.16. Plausible because a candidate may incorrectly select this response based on incorrect understanding of the RCS Activity Limits.

(2) Incorrect because the accident of concern for this limit is a SGTR, not a LBLOCA as provided in the TS3.6.16 Bases – Plausible because there is always some leakage from Containment following a LBLOCA. Limiting the RCS Activity prior to the event would reduce the releases from Containment. As a result, the candidate may incorrectly select this response. A candidate may incorrectly select this response based on incorrect understanding of the RCS Activity Limits and the Accident assumptions.

Sys #	System	Category	KA Statement
038	SG Tube Rupture	Generic	Knowledge of conditions and limitations in the facility license.

K/A# G2.2.38 **K/A Importance** **Exam Level:** SRO

References provided to Candidate None **Technical References:** Technical Specification R339 TS3.4.16 and related TS Bases.

Question Source: TMI 2017 NRC Exam Q83 Significantly Modified for DB **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.7 / 41.10 / 43.1 / 45.13)

Davis-Besse 1LOT22 NRC Written Exam

Objective: OPS-GOP-434-05K
Tier being Tested: ES 401 Tier1/Group 1 (SRO)

80. Uninterruptable power supply distribution bus YAU is lost at 100% power.

(1) What actions are required to respond to this event IAW DB-OP-02541, Loss of YAU?

AND

(2) How will DB-OP-02541, Loss of YAU be used in conjunction with DB-OP-02000, RPS, SFAS, SFRCS Trip and SG Tube Rupture?

A. (1) Commence a Plant Shutdown to Low Level Limit Control. Trip the Reactor. Initiate AND Isolate SFRCS.
(2) While working through the supplemental actions of DB-OP-02000, the Command SRO will refer to the Loss of YAU abnormal procedure Attachment 1.

B. (1) Commence a Plant Shutdown to Low Level Limit Control. Trip the Reactor. Initiate AND Isolate SFRCS.
(2) Actions to respond to the loss of YAU are explicitly directed by Supplemental Section 4.0 of DB-OP-02000, Response Not Obtained (RNO) Column.

C. (1) Trip the Reactor.
(2) While working through the supplemental actions of DB-OP-02000, the Command SRO will refer to the Loss of YAU abnormal procedure Attachment 1.

D. (1) Trip the Reactor.
(2) Actions to respond to the loss of YAU are explicitly directed by Supplemental Section 4.0 of DB-OP-02000, Response Not Obtained (RNO) Column.

Answer: C

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the use of an Abnormal Operating Procedure in conjunction with the Emergency Operating Procedure.

SRO Only: This is an SRO level question because it requires selection and coordination of the correct procedures to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction or reason for that action to mitigate the event.

A. Incorrect
(1) Incorrect because a Reactor Trip is required on the loss of ICS power – plausible since a loss of YAU will cause a major plant transient. However, as soon as a loss of YAU is diagnosed the reactor is directed to be tripped.
(2) Correct because the first supplemental step of DB-OP-02541 is to Trip the Reactor. REFER TO Attachment 1, Guidance for DB-OP-02000 Supplemental Actions. GO TO DB-OP-02000, SFAS, SFRCS Trip, or SG Tube Rupture.

B. Incorrect
(1) Incorrect because a Reactor Trip is required on the loss of ICS power – plausible since a loss of YAU will cause a major plant transient. However, as soon as a loss of YAU is diagnosed the reactor is directed to be tripped. DB-OP-02000 supplemental actions specifically direct actions in response to other loss of power events. For example, Step 4.5 directs action in response to a loss of NNI power. Step 4.6 directs actions in response to a loss of ICS power.
(2) Incorrect because there is no specific step directing actions for a loss of YAU power. Plausible because DB-OP-02000 supplemental actions specifically direct actions in response to other loss of power events. For example, Step 4.5 directs action in response to a loss of NNI power. Step 4.6 directs actions in response to a loss of ICS power.

C. CORRECT–
- (1) Correct - A Reactor Trip will not occur due to a loss of YAU.
- (2) Correct - DB-OP-02541, step 4.1 directs: Trip the Reactor REFER TO Attachment 1, Guidance for DB-OP-02000 Supplemental Actions. GO TO DB-OP-02000, SFAS, SFRCS Trip, or SG Tube Rupture. DB-OP-02541, Attachment 1 provides the direction to be used to respond to a reactor trip with a concurrent loss of YAU.

D. Incorrect –
- (1) Correct; A Reactor Trip will not occur due to a loss of YAU. DB-OP-02541, step 4.1 directs: Trip the Reactor REFER TO Attachment 1, Guidance for DB-OP-02000 Supplemental Actions. GO TO DB-OP-02000, SFAS, SFRCS Trip, or SG Tube Rupture.
- (2) Incorrect because there is no specific step directing actions for a loss of YAU power. plausible since DB-OP-02000 supplemental actions specifically direct actions in response to other loss of power events. For example, Step 4.5 directs action in response to a loss of NNI power. Step 4.6 directs actions in response to a loss of ICS power. There is no specific step directing actions for a loss of YAU power.

Sys #	System	Category	KA Statement
057	Loss of Vital AC Inst. Bus	Generic	Knowledge of how abnormal operating procedures are used in conjunction with EOPs

K/A# 2.4.8 **K/A Importance** 3.8 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02541 R17 Step 4.1, Att 1; DB-OP-02000 R32 Section 4, Supplemental Actions

Question Source: Modified DB NRC Exam 2016 Q13 – Significantly altered to make SRO level vice RO level. **Level Of Difficulty: (1-5)** 3.5

Question Cognitive Level: Low **10 CFR Part 55 Content:** 41.10 / 43.5 / 45.13

Objective: GOP-141-01K

Tier being Tested: ES 401 Tier1/Group 1 (SRO)

81. Plant conditions:
    - The reactor is operating at 100 percent
    - A Thunderstorm Warning has been issued
    - A severe Thunderstorm is in progress

    The following conditions are noted:
    - Generator Power = 950 MW
    - Volts Ampere Reactive (Vars) = 500 MVAR LAG
    - Hydrogen Gas Pressure = 60 psig

    Which ONE of the following procedures should be used to mitigate these conditions?

A. DB-OP-01300, Switchyard Management

B. DB-OP-06311, 345 KV Switchyard NO. 1 (Main) Transformer, NO. 11 (Auxiliary) Transformer, and Startup Transformers (01 AND 02)

C. DB-OP-06301, Generator and Exciter Operating Procedure

D. DB-OP-06902, Power Operations

Answer: C

Explanation/Justification: This Question is a KA match because it provides the indications for determining is the Main Generator is operating outside the limits of the capability curve. The operator must use the curve to determine that current conditions are beyond those allowed by the capability cure. The operator must evaluate the conditions provided and then use that knowledge to select the appropriate direction to respond to the event. In this case, an adjustment to Generator Voltage is required to return the generator MVARs to within the limits of the curve.

SRO Only: This is an SRO level question because it requires selection of the correct procedure to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction or reason for that action to mitigate the event.

A. Incorrect because DB-OP-01300 does not provided direction to respond to this specific event – Plausible because the procedure describes the interface between the operators at Davis-Besse and the system dispatchers. Communications between the two groups would be required during this event, but these communications would not mitigate the events in progress.

B. Incorrect because DB-OP-06311 does not provide direction to respond to this specific event – Plausible because DB-OP-06311 contains the Station Voltage Limits and other information related to the Main, Aux, and Startup transformers operation.

C. CORRECT – DB-OP-06301, Section 3.5 provides direction to adjust generator voltage to reduce MVARs out to within the limits of the generator capability curve.

D. Incorrect because DB-OP-06902 does not provide direction to respond to this specific event – Plausible because DB-OP-06902, Power Operations directs MEGAVARS OUT, as indicated on XI 6005, MEGAVARS (C5722). It also informs the crew that the System Dispatcher may request that MEGAVARS be adjusted to meet System Load requirements. REFER TO DB-OP-06301, Generator and Exciter Operating Procedure.

Sys #	System	Category	KA Statement
077	Generator Voltage and Electric Grid Disturbance	Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances:	Generator current outside the capability curve

Davis-Besse 1LOT22 NRC Written Exam

K/A#	AA2.03	**K/A Importance**	3.6	**Exam Level:**	SRO
References provided to Candidate:		CC9.5, ESTIMATED CAPABILITY CURVES - LEAD-LAG		**Technical References:**	DB-PF-06703 R26 Miscellaneous Operations Curves Page 82, Curve CC9.5, DB-OP-06301 R35, GENERATOR AND EXCITER OPERATING PROCEDURE, Section 3.5, steps 3.5.2 & 3.5.4
Question Source:	NRC Exam 2011 ANO U2 Q78 Modified for DB			**Level Of Difficulty: (1-5)**	3.5
Question Cognitive Level:	HI			**10 CFR Part 55 Content:** (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)	
Objective:	OPS-SYS-401-04K				
Tier being Tested:	ES 401 Tier1/Group 1 (SRO)				

82. The following symptoms are observed at 100% power:
- Annunciator (5-1-E) CRD SYSTEM FAULT
- Annunciator (5-2-E) CRD ASYMMETRIC ROD
- Computer Point Q178 CRD ASYMMETRIC ROD ALARM is in alarm
- CRD Mechanism Position Indication Panel - 0 PERCENT light for Rod 6 – 5 is lit
- IN LIMIT on rod control panel is lit for Group 6
- Tave lowers to approximately 579ºF

Which of the following procedures/sections should be used to respond to this event?

A. DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture

B. DB-OP-02516, CRD Malfunction, Section 4.1, Dropped Control Rod

C. DB-OP-02516, CRD Malfunction, Section 4.2, Misaligned Control Rod

D. DB-OP-02516, CRD Malfunction, Section 4.3, Control Rod Position Indication Malfunctions

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Answer: B

Explanation/Justification: This Question is a KA match because it provides the indications for a dropped control rod with the reactor at power. Computer point alarms that indicate a dropped control rod are provided in the question. The operator must evaluate the conditions provided and then use that knowledge to select the appropriate direction to respond to the event. The indications provided are from running the event on the plant specific simulator.

SRO Only: This is an SRO level question because it requires selection of the correct procedure to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction or reason for that action to mitigate the event.

A. Incorrect because only a single rod is dropped and from the plant conditions provided, power would stabilize above 5%. As a result, a reactor trip would not be required and therefore entry into DB-OP-02000 is not warranted – Plausible because if more than one Control Rod is dropped, or power does not stabilize above 5%, a reactor trip is required for these conditions and this would be the appropriate procedure to respond to the Reactor Trip.

B. CORRECT – A single Rod has dropped into the Core. From 100% power, reactor power would stabilize above 5%. This procedure section provides direction to reduce reactor power, determine why the rod dropped, correct condition, and to recover the dropped rod.

C. Incorrect because based on plant conditions provided, a single rod is dropped, not just misaligned – Plausible because the dropped control rod is misaligned with the remaining rods in that group. This section is used to respond to events where control rod API or RPI or both indicate a Control Rod more than 6.5 percent from the group average position. For this condition, Rod 6-5 is at 0% and the remaining rods are at 100%. It is misaligned from the group average but is dropped.

D. Incorrect because this event is an actual dropped rod based multiple indications that the rod is on the bottom – Plausible because the question provides Control Rod position indications that could be faulty. This procedure section provides direction to determine if a position indication is malfunctioning. Also, Incorrect because the change in Tave indicates an actual dropped rod.

Sys #	System	Category	KA Statement
003	Dropped Control Rod	Generic	Ability to use plant computers to evaluate system or component status.

K/A# G2.1.19 **K/A Importance** 3.8 **Exam Level:** SRO

References provided to Candidate: None **Technical References:** DB-OP-02516 R20, CRD Malfunctions, steps 2.1, 4.1.1 and 4.1.2

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.10 / 45.12)

Davis-Besse 1LOT22 NRC Written Exam

Objective: OPS-GOP-116-03K
Tier being Tested: ES 401 Tier1/Group 1 (SRO)

83. The reactor is operating at 100 percent.

An event occurs, the following items are noted:

- No other changes to plant equipment are noted
- No changes to Control Room conditions are noted
- Two Adjacent Fire Alarms are received from the Cable Spread Room

Which procedure should be used to respond to this event?

A. DB-FP-00009, Fire Protection Impairment and Fire Watch

B. DB-OP-02501, Serious Station Fire

C. DB-OP-02519, Serious Control Room Fire

D. DB-OP-02529, Fire

Answer: D

Explanation/Justification: This question is a KA match because it provided the indications for a potential actual fire in the cable spread room and then requires selection of the appropriate procedure to respond to those alarms. In the scenario provided, 2 fire alarms in the Cable Spread Room are in alarm. This requires the use of Abnormal Operating Procedure DB-OP-02529, Fire, to dispatch the Fire Brigade to respond.

SRO Only: This is an SRO level question because it requires selection of the correct procedure to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction or reason for that action to mitigate the event.

A. Incorrect because this is the admirative procedure for responding to out of service alarms, barriers, and other equipment. It is not to be used for actual fire alarms – Plausible because DB-FP-00009 is the correct procedure to use to addressed failed fire detectors. With this fire location, the Cable Spread Room and with no other changes in plant status, the candidate may assume the 2 alarms are spurious and use the Administrative procedure to determine applicable compensatory measures.

B. Incorrect because this procedure is used for fire that affect safe shutdown equipment in locations other than the Control Room or Cable Spread Room – Plausible because a fire in the Cable Spread Room is a serious condition, but this procedure would be required for multiple alarms combined with an impact on safe shutdown equipment. In addition, the response to a cable spread room fire is include in the Control Room response DB-OP-02519, not DB-OP-02501.

C. Incorrect because the fire has not adversely affected safe shutdown equipment as noted by no changes to plant equipment or control room conditions – Plausible because this procedure provides guidance during a serious fire in the Control Room or Cable Spreading Room requiring an evacuation of the Control Room. Since no changes to plant equipment or control room conditions have occurred, this is not the correct procedure to respond to the events provided.

D. CORRECT – With 2 alarms in the cable spread room, the direction to activate the fire brigade is provided in DB-OP-02529. If an actual fire exists AND addition failures that affect safe shutdown equipment occur, the operators would be directed to implement DB-OP-02519, Serious Control Room Fire.

Sys #	System	Category	KA Statement
067	Plant Fire on Site	Generic	Ability to perform specific system and integrated plant procedures during all modes of plant operation.

K/A# G2.1.23 **K/A Importance** 4.4 **Exam Level:** SRO

References provided to Candidate None **Technical References:** DB-OP-02529 R12, Step 4.2 & 4.4

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.2 / 45.6)

Objective: OPS-GOP-129-03K

Davis-Besse 1LOT22 NRC Written Exam

Tier being Tested: ES 401 Tier1/Group 1 (SRO)

84. The Reactor is operating at 100% power.

An event occurs. The following alarms/conditions are noted:

- Annunciator 1-1-B EDG 1 FAULT
- Annunciator 1-4-D BUS C1 VOLTAGE
- Zero volts is noted on buses A, B, C1, C2, D2
- All Immediate Actions have been completed

The Shift Manager directs restoration of power to bus C1. Which of the following procedures would be used FIRST to energize Bus C1?

A. DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture, Attachment 28, Restore Power to C1 OR D1 BUS from the SBODG.

B. DB-OP-02521, Loss of AC Bus Power Sources, Section 4.1, Loss of Essential 4160 Volt AC Bus Power Sources.

C. DB-OP-02700, Station Blackout, Section 3.0, Station Blackout Direction.

D. DB-OP-06316, Emergency Diesel Generator Operating Procedure, Section 5.1, Control Room Operation of EDG 1.

Answer: A

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the plant conditions that would cause an EDG start signal (low voltage on an essential 4160 bus (in this case, C1 Bus) and what procedure will be used to restore power to Bus C1. The plant indications provided indicate that a loss of offsite power has occurred based on 0 volts on A and B bus. This would result in a reactor trip. The Candidate must recognize that a reactor trip has occurred in order to select the correct procedure. In addition, EDG 1 failed to auto start as indicated by the 2 annunciators in alarm. C1 Bus is not locked out because annunciator 1-3-D BUS C1 LOCKOUT is not listed as in alarm. All of the options presented could be used to restore power to Bus C1. DB-OP-02000 Specific Rule 6 (which is encountered early in the Reactor Trip response) directs the use of attachment 28 to restore power from the SBODG following a Reactor Trip. All of the remaining options could be used and provide a number of options but would not be used first following the Reactor trip that would occur on the loss of off-site power.

SRO Only: This is an SRO level question because it requires selection of the correct procedure to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction or reason for that action to mitigate the event.

A. CORRECT – The reactor has tripped due to a loss of Off-Site Power (0 volts on Bus A and B) which causes the loss of all Reactor Coolant Pumps and the resultant Reactor Protective System trip. Once the reactor trips, DB-OP-02000 Specific Rules 6 would direct the use of Attachment 28 to restore power to C1 bus from the SBODG.

B. Incorrect because a Reactor Trip has occurred, and the Emergency Operating Procedure direction takes priority over the Abnormal Operating Procedure direction – Plausible because this procedure could be used to restore power to C1 bus, and it provides a number of options but would not be used first following the Reactor trip that would occur on the loss of off-site power. The Candidate may select this response because it does provide direction to restore power to C1 Bus.

C. Incorrect because a station blackout does not exist. Power is available to D1 bus from EDG 2 – Plausible because zero volts is noted on buses A and B indicating a Loss of Off-Site Power (LOOP). DB-OP-02700 is utilized during a (LOOP) after attempting, without success, to place the Emergency Diesel Generators (EDGs) and Station Blackout Diesel Generator (SBODG) in service to supply power to at least one essential 4160-volt AC bus in accordance with DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture. In this scenario the SBODG, EDG2, and D1 are still available therefore DB-OP-02700 would not be used.

D. Incorrect because a Reactor Trip has occurred. The Emergency Operating Procedure takes priority over this System Operating Procedure – Plausible because this procedure could be used to restore EDG 1 to service, and it provides a number of options but would not be used first following the Reactor trip that would occur on the loss of off-site power. The Candidate may select this response because it does provide direction to place EDG 1 in service which would restore power to C1 Bus.

Sys #	**System**	**Category**	**KA Statement**
BW A05	EDG Actuation	Ability to determine and interpret the following as they apply to the (Emergency Diesel Actuation):	Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

K/A# AA2.1 **K/A Importance** 4.2 **Exam Level:** SRO

References provided to Candidate: None **Technical References:** DB-OP-02000 R32 Attachment 28,

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 43.5 / 45.13)

Objective: OPS-GOP-313-02K

Tier being Tested: ES 401 Tier1/Group 1 (SRO)

85. A reactor trip occurred during a rapid shutdown due to a SGTR.

Current plant conditions:
- SG 1 has a SGTR
- SG 1 pressure is 1000 psig and steady
- SG 2 MSSV is stuck partially open, causing a plant cooldown of 2°F per minute
- SG 2 pressure is 940 psig and lowering
- Immediate Operator Actions are complete
- Supplemental Actions are in progress

Which of the following actions is required to be taken FIRST IAW DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture?

A. Go to Section 7, Overcooling, and attempt to reseat the MSSV.

B. Go to Section 7, Overcooling, and alternately feed SG 2 to control cooldown.

C. Go to Section 8, SGTR, and continue RCS cooldown using SGs 1 and 2.

D. Go to Section 8, SGTR, and continue RCS cooldown using only SG 2.

Answer: A

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the procedure prioritization and selection to address a Design Basis SG Tube Rupture with the additional complication of a failed Main Steam Safety Valve. This Tube Rupture is equivalent to a Small Break LOCA as it assumes without operator action a Reactor Trip will occur on Low RCS pressure. SFAS will actuate with the High Pressure Injection System recovering RCS pressure and preventing a loss of subcooling margin. The question concerns the cooldown and depressurization using the appropriate direction in the Emergency Operating Procedure.

SRO Only:

A SG Tube Rupture was selected to avoid over sampling based on question 88, which used a break in the RCS piping inside Containment to test rapid depressurization following an SFAS Actuation.

This is an SRO level question because it requires selection of the correct procedure section to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction or reason for that action to mitigate the event.

A. CORRECT- DB-OP-02000 Section 4, Supplemental Actions, determines the order of mitigation hierarchy, Section 7 selected before Section 8. Per Section 7, an attempt to stop the overcooling cause should be attempted first.
B. Incorrect cause the strategy described is not directed by the controlling procedure. Plausible because Section 7, Overcooling has a higher priority. The action to feed SG2 is not directed by Section 7, Overcooling.
C. Incorrect because an overcooling event actions take priority over the SGTR actions. Plausible because SG 1 has a tube leak, but Section 7, Overcooling has a higher priority. Section 8 direction will be used after the overcooling is addressed.
D. Incorrect because the overcooling event actions take priority over the SGTR actions. Plausible because SG 1 has a tube leak, but Section 7, Overcooling has a higher priority. Section 8 would be directed after the overcooling is addressed.

Sys #	System	Category	KA Statement
BW/E08	LOCA Cooldown - Depress	Ability to determine and interpret the following as they apply to the (LOCA Cooldown:)	Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

K/A# EA2.2 **K/A Importance** 4.0 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02000 R32, Section 4.0 pages 26 & 28

Question Source: DB NRC Exam 2011 Q80 **Level Of Difficulty: (1-5)** 3

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

Davis-Besse 1LOT22 NRC Written Exam

Objective: OPS-GOP-307-05K
Tier being Tested: ES 401 Tier1/Group 1 (SRO)

86. The reactor is operating at 100% power with all ICS Control Stations in their normal lineup.

The following Annunciator Alarm occurs:

- (6-3-D) 2-2 SEAL RET FLOW HI
- (6-5-A) MONITOR SYSTEM TROUBLE

The following RCP 2-2 conditions are then noted:

- (6-3-D) 2-2 SEAL RET FLOW HI alarm CLEARS
- F855 Seal Return Flow = 0 gpm
- T853 Seal Return Temperature = 137°F
- P853 2nd Seal Cavity Press = 105 psig
- P854 3nd Seal Cavity Press = 62 psig

The crew has implemented DB-OP-02515, Reactor Coolant Pump and Motor Abnormal Operation to mitigate this event.

With these Plant conditions, which of the following procedures and actions is required to be performed NEXT?

A. DB-SP-03357, RCS Water Inventory Balance. Evaluate Reactor Coolant System Leakage.

B. DB-OP-02504, Rapid Shutdown. Reduce power to 72% and then stop RCP 2-2.

C. DB-OP-06401, Integrated Control System Operating Procedure. Stop RCP 2-2 immediately. Verify the plant automatically runs back to less than 72%.

D. DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture. Immediately trip the Reactor and then stop RCP 2-2.

Answer: B

Explanation/Justification: Two of the Three Reactor Coolant Pump 2-2 seals have failed. This is possible if the second stage seal failed and debris from that failure cause the third stage seal to fail. In this condition, the Seal Return High Flow alarm would occur on the second stage failure and then clear on failure of the 3rd stage as the differential pressure that normally drives seal return flow back to the Makeup System is lost. In this condition, seal leakoff flow would increase. DB-OP-02515 requires that 2-2 RCP be removed from service in IAW Attachment 1 which directs reactor power reduction to 72% and then stopping the RCP.

SRO Only: This is an SRO level question because it requires selection of the correct procedure to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction or reason for that action to mitigate the event.

A. Incorrect because higher priority Abnormal Operating Procedure actions (reduce power and stop RCP) are required to be performed next, not the inventory balance – Plausible because based on provided Seal Return temperatures and flows, a large release of seal leakoff is not occurring. Candidate may select this response based on low leakage, less than Technical Specification allowed. This action is also directed per step 4.1.4. Continued operation of RCP 2-2 at 100% power is permitted as long as Reactor Coolant System Leakage remains within Technical Specification Limits.

B. CORRECT - Based on the indications provided, RCP 2-2 Second AND Third Stage Seals have failed. In accordance with DB-OP-O2515, RCP Malfunctions, continued operation with more than one seal stage failed is not permitted, but immediate stop of the RCP is not required or directed. Direction is provided by Attachment 1 to reduce power to 72% then shutdown the RCP.

C. Incorrect because this is not the action directed by the control procedure for this event - Plausible because the only remaining RCP Seal is under almost full RCS pressure. Stopping the RCP is required, however this action from 100% will result in a reactor trip. The ICS RCP runback does not reduce power fast enough to avoid a Reactor Protective System trip on flux/Δflux/flow. The unnecessary trip would further perturbate RCS pressure across the remaining seal.

D. Incorrect because this action is not directed by the controlling procedure - Plausible because the only remaining RCP Seal is under almost full RCS pressure. Stopping the RCP is required; however, this is not done immediately. Tripping the RCP prior to lowing reactor power would result in a flux/Δflux/flow trip. The unnecessary reactor trip would further perturbate RCS pressure across the remaining seal.

Sys #	System	Category	KA Statement
003	RCP	Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Problems with RCP seals, especially rates of seal leak-off

K/A# A2.01 **K/A Importance** 3.9 **Exam Level:** SRO

References provided to Candidate: None

Technical References: DB-OP-02515 R18, Reactor Coolant Pump and Motor Malfunctions Step 4.1.1 and Attachment 1.

Question Source: NEW

Level Of Difficulty: (1-5) 3

Question Cognitive Level: HI

10 CFR Part 55 Content: (CFR: 41.5 / 43.5/ 45.3 / 45/13)

Objective: OPS-GOP-115-02K

Tier being Tested: ES 401 Tier2/Group 1 (SRO)

87. Initial Plant Conditions:
- The Reactor is shutdown in Mode 5
- RCS Cooldown and depressurization are in progress
- DHR Loop 1 aligned for Standby DHR operation
- DHR Loop 2 in service
- Conditions have been established to stop all Reactor Coolant Pumps
- RCS Temperature is 160°F
- RCS Pressure is 210 psig
- Pressurizer Level is 85 inches

When the last RCP is stopped, an event occurs. The following conditions are noted:
- Annunciator (3-1-I), LP INJ 2 FLOW LO
- Incore thermocouple temperatures are rising
- No flow on FYI DH2A, LPI/DH PUMP 2 OUTLET FLOW
- DHR Pump 1 Breaker Status Light is Green
- DHR Pump 2 Breaker Status Light is Red
- DH1A, DH PUMP 2 DISCHARGE TO RCS is closed
- Pressurizer Level is slowly rising

Which ONE of the following procedures/sections of procedures provides the direction necessary to mitigate this event?

A. DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture

B. DB-OP-02527, Loss of Decay Heat Removal, Section 4.1, Loss of DHR Pump

C. DB-OP-02527, Loss of Decay Heat Removal, Section 4.2, Loss of DHR Flowpath

D. DB-OP-02527, Loss of Decay Heat Removal, Section 4.3, Loss of DHR Inventory

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Answer: C

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of system design and expected alarms for a given malfunction and then uses that knowledge to select the proper procedure or section of a procedure to respond to that malfunction.

SRO Only: This is an SRO level question because it requires selection of the correct procedure to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction for that malfunction.

A. Incorrect because this procedure is not applicable for the plant conditions provided – Plausible because a lack of heat transfer event is occurring. DB-OP-02000, Section 6, Lack of Heat Transfer provides direction to respond to an overheating event, however once the plant is on decay heat removal, DB-OP-02527, Loss of Decay Heat Removal becomes the controlling procedure, not DB-OP-02000. A candidate may select this response if they do not know that DB-OP-02527 would be the controlling procedure for this event.

B. Incorrect because the DHR pump has not been lost, only the flowpath disrupted – Plausible because this is the correct procedure to respond to loss of Decay Heat Removal Event, however the #2 DHR pump remains in operation as noted by the breaker status light. The DHR pump has not been lost. This would be the correct procedure section if the DHR Pump was lost. The candidate may select this response if the more likely scenario, loss of DHR pump had occurred.

C. CORRECT – Based on the indications provided, a loss of Decay Heat Removal Flowpath has occurred. This can be determined by the pump still operating in conjunction with the low flow alarm and rising pressure level and RCS temperatures as core begins to heat up due to a loss of cooling.

D. Incorrect because this section of the procedure is used to address the loss of inventory while on DHR, not the loss of flowpath that is provided in the question stem – Plausible because on a loss of inventory from the DHR system, the red status light for the DHR breaker would still be on and the Low Flow Alarm would be possible if the DHR Pump was cavitating or a system break upstream of the flow indicator had occurred. Stopping an RCP does cause a hydraulic transient as system flow coasts down which makes a leak possible. The candidate may select this response if they think a loss of inventory had occurred. Incorrect due to pressurizer level rising.

Sys #	System	Category	KA Statement
005	SP4P RHR	Generic	Ability to verify that the alarms are consistent with the plant conditions.

K/A# G2.4.46 **K/A Importance** 4.2 **Exam Level:** **SRO**

References provided to Candidate: None **Technical References:** DB-OP-02527 R20, Loss of DHR, Section 4.2, Loss of DHR Flowpath

Question Source: NEW **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.3 / 45.12)

Objective: OPS-GOP-127-02K

Tier being Tested: ES 401 Tier2/Group 1 (SRO)

88. The Reactor is operating at 100% power.

A LOCA event occurs.

Current plant conditions:

- All SFAS actuated components have operated as designed
- RCS Pressure is 765 psig and stable
- CTMT Pressure is 27.1 psia and slowly lowering
- Average Incore Thermocouple Temperature is 515°F and slowly lowering
- SG levels are being maintained at 124 inches
- Atmospheric Vent Valves are in Hand controlling RCS temperature
- The BWST is at 9.5 feet and slowly lowering.

Based on Current plant conditions, which Section of DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture, will direct transferring LPI Suctions to the Emergency Sump in accordance with Attachment 7, Transferring LPI Suction to the Emergency Sump?

A. Section 5, Loss of Subcooling Margin

B. Section 10, Large LOCA Cooldown

C. Section 11, RCS Saturated with SGs Removing Heat Cooldown

D. Section 13, RCS Subcooled with SG Removing Heat Cooldown

Answer: C

Explanation/Justification: This question is a match for the KA because it tests the Candidate's knowledge of the Emergency Operating Procedures used to mitigate a rapid depressurization of the Reactor Coolant System that causes an SFAS Actuation. To select the correct section, the candidate must use the plant conditions provided and determine the correct procedure routing.

SRO Only: This is an SRO level question because it requires selection of the correct procedure to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure section that provides the direction to control, or mitigate the consequences of the event. Davis-Besse has a single Emergency Operating Procedure that is divided into multiple sections used to mitigate events based on symptoms that exist. Selecting an EOP section at DB is like selecting an EOP procedure at a facility that has many EOPs that are selected based on symptoms.

A. Incorrect because Section 5 does not provide direction to transfer LPI suctions to the Emergency Sump. Section 5 directs the operators to Section 6, 10, 11 or 13 depending on plant conditions
Plausible because the plant conditions provided indicate a Loss of Subcooling Margin and Section 5 would be the first section entered for this scenario

B. Incorrect because Section 10 entry criteria is not met.
Plausible because the plant conditions indicate a significant LOCA.

C. CORRECT because the current conditions indicate a Saturated RCS (765 psig and 515 Degrees) with SGs removing heat (AVVs controlling Temperature). The correct procedure routing would be entry into Section 5 for a Loss of Subcooling Margin, followed by entry into Section 11.

D. Incorrect because the RCS is not subcooled
Plausible because conditions indicate the SGs are removing heat and the examinee may incorrectly read the provided Steam Tables

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Sys #	System	Category	KA Statement

013	SFAS	Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Rapid depressurization

K/A# A2.03 **K/A Importance** 4.7 **Exam Level:** SRO

References provided to Candidate: Steam Tables **Technical References:** DB-OP-02000 R32, RPS, SFAS, SFRCS Trip or SG Tube Rupture Step 10.6, 5.15

Question Source: NEW **Level Of Difficulty: (1-5)** 3

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

Objective: OPS-GOP-309-04K

Abbreviations: SFAS = Safety Features Actuation System

Tier being Tested: ES 401 Tier2/Group 1 (SRO)

89. Initial plant conditions:
- The Reactor is operating at 100% Reactor Power

An event occurs. After one minute, the following conditions are noted:
- RCS Pressure is 75 psig
- Containment Pressure is 42 psia
- Both Containment Air Coolers are operating in slow speed with Service Water available.
- Both Containment Spray Pumps are off

Complete the following statements:

If NEITHER Containment Spray Pump can be manually started,

(1) the required peak Containment cooling capacity during the post-accident conditions________ be met,

AND

(2) the requirements for removing iodine from the containment atmosphere and maintaining concentrations below those assumed in the safety analysis __________ be met.

A. (1) will
(2) will

B. (1) will
(2) will not

C. (1) will not
(2) will

D. (1) will not
(2) will not

Answer: B

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the impact of the failure of both Containment Spray Pumps during a Design Bases Large Break LOCA Event. The Containment Pressure provided in the stem is higher than the setpoint for SFAS level 4 which provides a start signal for the Containment Spray Pumps.

At Davis-Besse, the Containment Cooling System includes 2 Containment Spray Pumps and 2 in service Containment Air Coolers. One Containment Air Cooler and one Containment Spray Pump operating is sufficient to ensure that Containment Design Pressure will not be exceed during a Design Bases Large Breaker Loss of Coolant Event. As provide in USAR 6.2.2.1 Design Bases "Each containment spray and each containment air cooler is designed for 50 percent of the heat load (75E6 Btu/hr). Two fully redundant heat removal methods composed of one containment spray train and one containment air cooler (CAC) train are provided for post-LOCA heat removal." As noted, the trains are fully redundant meaning that one CAC and one CTMT Spray Pump or two CACs will remove all the required heat from Containment and Design Pressure of CTMT will not be exceeded.

With respect to the second portion on the requirements for iodine removal from Containment Atmosphere,

SRO Only:

Davis Besse has Trisodium Phosphate Dodecahydrate (TSP) Storage Baskets. From the Bases for TS 3.6.7, “The TSP storage baskets are a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA). In the event of an accident such as a loss of coolant accident (LOCA), however, the containment emergency sump will flood to a level above the TSP storage baskets. This level of water will dissolve the TSP in the storage baskets and mix with the containment emergency sump water. Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms”. At Davis Besse, containers of Tri-Sodium Phosphate are located in the low levels of Containment. The Bases for TS 3.6.6, Containment Spray and Air Cooling Systems provides the following: “one containment spray train is required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis”.

This is an SRO level question per NURG 1021 Attachment 2 Section B Facility Operating Limitations in the Technical Specifications and Their Bases [10 CFR 55.43(b)(2)] because it tests the candidate’s knowledge of the Bases for TS 3.6.6 and TS 3.6.7 related to the Safety Functions of the Containment Spray System during a LOCA, both for heat removal and for Containment atmosphere.

A. Incorrect

(1) Correct – See C.1 justification.

(2) Incorrect because the Bases for TS 3.6.6, Containment Spray and Air Cooling Systems provides the following: “one containment spray train is required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis”. If at least one Containment Spray Pump cannot be started, the requirements for maintaining Iodine post LOCA will not be met.
Plausible since the CACs circulate air through CTMT, it may be assumed that the CAC’s also remove the iodine through filtration.

B. Correct

(1) Correct - As provide in USAR 6.2.2.1 Design Bases “Each containment spray and each containment air cooler is designed for 50 percent of the heat load (75E6 Btu/hr). Two fully redundant heat removal methods composed of one containment spray train and one containment air cooler (CAC) train are provided for post-LOCA heat removal.” As noted, the trains are fully redundant meaning that one CAC and one CTMT Spray Pump or two CACs will remove all the required heat from Containment and Design Pressure of CTMT will not be exceeded.

(2) Correct - The Bases for TS 3.6.6, Containment Spray and Air Cooling Systems provides the following: “one containment spray train is required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis”.

C. Incorrect

(1) Incorrect - As provide in USAR 6.2.2.1 Design Bases “Each containment spray and each containment air cooler is designed for 50 percent of the heat load (75E6 Btu/hr). Two fully redundant heat removal methods composed of one containment spray train and one containment air cooler (CAC) train are provided for post-LOCA heat removal.” As noted, the trains are fully redundant meaning that one CAC and one CTMT Spray Pump or two CACs will remove all the required heat from Containment and Design Pressure of CTMT will not be exceeded.
Plausible since both Spray pumps are failed, it may be assumed that at least one is required to remove all the required heat.

(2) Incorrect – See A. 2 justification

D. Incorrect

(1) Incorrect – See C. 1 justification.

(2) Correct – See B. 2 justification.

Sys #	System	Category	KA Statement
026	Containment Spray	Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on 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** 3.9	**Exam Level:**	SRO
References provided to Candidate:	None	**Technical References:**	TS Bases 3.6.6 and 3.6.7 and UFSAR 6.2.2.1
Question Source:	New	**Level Of Difficulty: (1-5)**	3
Question Cognitive Level:	HI	**10 CFR Part 55 Content:** (CFR: 43.5 / 45.13)	

Objective: OPS-GOP-123-02K

Tier being Tested: ES 401 Tier1/Group 1 (SRO)

90. Initial conditions:
    - The reactor was operating at 100% power
    - SBODG is OOS for maintenance
    - A reactor trip occurred due to a Loss of Off-Site Power
    - Immediate Operator Actions of DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture, have been completed

    Current plant conditions:
    - Annunciator 1-3-D, BUS C1 LOCKOUT is in alarm
    - EDG 2 is powering D1 and D2
    - Both AFW Pumps failed to start
    - The Motor Driven Feedwater Pump failed to start
    - The Emergency Feedwater Pump failed to start

    Determine the Emergency Classification for the listed conditions.

    The appropriate Emergency Classification is ___(1)___ based on EAL ____(2)____.

A. (1) Site Area Emergency
(2) SS1.1

B. (1) Alert
(2) SA1.1

C. (1) Alert
(2) FA1

D. (1) Site Area Emergency
(2) FS1

Answer: D

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the impact on a failure of the Auxiliary Feedwater Systems and the Emergency Feedwater System. These plant conditions require activation of the Emergency Plan and with it, the associated notification requirement to the Local and State Agencies and to the NRC.

SRO Only: This is an SRO level question because it tests the duties and responsibilities of the Shift Manager in the Emergency Plan program. NUREG 1021 Section D.2.d - Questions related to 10 CFR 55.41(b) topics may also be appropriate SRO-level questions if they evaluate K/As at a level that is unique to the SRO job position. This question is unique to the Shift Manager position which can only be performed by an SRO.

A. Incorrect
- (1) Incorrect because this is the wrong classification per RA-EP-01500, Emergency Classification
  Plausible because a Site Area Emergency would be declared if EDG 2 was not operating.
- (2) Incorrect because this is the wrong Emergency Action Level per RA-EP-01500, Emergency Classification
  Plausible because SS1.1 would be declared if EDG 2 was not operating.

B. Incorrect
- (1) Incorrect because this is the wrong classification per RA-EP-01500, Emergency Classification
  Plausible because an Alert would be declared if a higher classification did not exist.
- (2) Incorrect because this is the wrong Emergency Action Level per RA-EP-01500, Emergency Classification
  Plausible because with the plant conditions given, SA1.1 would be called if a higher classification did not exist.

C. Incorrect

(1) Incorrect because this is the wrong classification per RA-EP-01500, Emergency Classification – Plausible if the candidate recognizes a potential loss of the Reactor Coolant System Barrier B1 - "Loss of ALL feedwater AND SG cooling is required but does not recognize that this also results in a Fuel Clad Barrier Potential Loss Threshold B.2.

(2) Incorrect because this is the wrong Emergency Action Level per RA-EP-01500, Emergency Classification – Plausible if the candidate recognizes a potential loss of the Reactor Coolant System Barrier B1 - "Loss of ALL feedwater AND SG cooling is required but does not recognize that this also results in a Fuel Clad Barrier Potential Loss Threshold B.2

D. CORRECT –

(1) The criteria for Emergency Plan criteria are provided by RA-EP-01500, Emergency Classification. This procedure provides a table that is used to assess plant conditions. For assessing the barriers to release of fission products, the following is provided for the potential loss of the Reactor Coolant System Barrier - "Loss of ALL feedwater AND SG cooling is required (Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold B.2; both will be met)" As a result, Site Area Emergency is the correct response.

(2) The criteria for Emergency Plan criteria are provided by RA-EP-01500, Emergency Classification. This procedure provides a table that is used to assess plant conditions. For assessing the barriers to release of fission products, the following is provided for the potential loss of the Reactor Coolant System Barrier - "Loss of ALL feedwater AND SG cooling is required (Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold B.2; both will be met)" As a result, Emergency Action Level FS1 is the correct response.

Sys #	**System**	**Category**	**KA Statement**
061	Emergency/ Auxiliary Feedwater	Generic	Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

K/A# G2.4.30 **K/A Importance** 4.1 **Exam Level:** SRO

References provided to Candidate RA-EP-01500 Classification Wallboard (DBRM-EMER-1500B)

Technical References: RA-EP-01500 R16, Step 6.1.2.d and e, RA-EP-01500 R2 Classification Wallboard (DBRM-EMER-1500B)

Question Source: NEW

Level Of Difficulty: (1-5) 3

Question Cognitive Level: HI

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.11)

Objective: OPS-GOP-602-04K

Tier being Tested: ES 401 Tier2/Group 1 (SRO)

91. The reactor was operating at 100% power.

An event occurs. The following indications are noted:

- EI6256, A BUS KILOVOLTS reads 0 volts
- EI6257, B BUS KILOVOLTS reads 0 volts
- Annunciator 1-3-D BUS C1 LOCKOUT is in alarm
- Annunciator 1-3-H BUS D1 LOCKOUT is in alarm

The Station Blackout Diesel Generator will not start.

(1) Which of the following actions should be taken in response to this event?

AND

(2) Which ONE of the following procedures directs this action?

A. (1) Close MU38, RCP Seal Return.
(2) DB-OP-02700, Station Blackout

B. (1) Close MU38, RCP Seal Return.
(2) DB-OP-02521, Loss of AC Bus Power Sources

C. (1) Align Service Water Returns to the Intake Forebay.
(2) DB-OP-02511 Loss of Service Water Pumps / System

D. (1) Align Service Water Returns to the Intake Forebay.
(2) DB-OP-02753, FLEX Replenishment Pump

Answer: A

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the impact on the Reactor Coolant System based on Control Room indications. The indications provided are for a station blackout (loss of Off-Site and On-Site AC power). In this condition, normal inventory addition methods are lost. The candidate must recognize inventory addition methods are lost and action is needed to preserve RCS inventory. Inventory is preserved by isolating RCS Letdown and RCP Seal Return. This is accomplished from the Control Room by closing Air Operated MU3 and MU38.

SRO Only: This is an SRO level question because it requires selection of the correct procedure to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the indications provided and then use that knowledge to determine the appropriate procedure that provides the direction and reason for that action to mitigate the event.

A. CORRECT
(1) Correct - A Station Blackout is in progress; normal RCS inventory addition methods are lost. The candidate must recognize inventory addition methods are lost, and action is needed to preserve RCS inventory. Inventory is preserved by isolating MU38 RCP Seal Return.
(2) Correct - MU38 is closed as directed by DB-OP-02700, Station Blackout.

B. Incorrect
(1) Correct – See A. 1 for explanation.
(2) Incorrect because DB-OP-02521, Loss of AC Bus Power Sources does not direct closing MU38. Plausible since a loss of AC Bus Power Sources is in progress.

C. Incorrect
- (1) Incorrect because this action is not directed by the controlling procedure for the station blackout. Plausible because the FLEX response to this beyond design bases event will eventually need inventory from the ultimate heat sink to provide cooling.
- (2) Incorrect because this action is not directed by the controlling procedure. Plausible since a loss of Service Water will be in progress due to the loss of C1 and D1.

D. Incorrect
- (1) See C. 1 Justification.
- (2) Incorrect because this action is not directed by the controlling procedure. Plausible since FLEX procedures will be utilized to mitigate this event.

Sys #	**System**	**Category**	**KA Statement**
002	Reactor Coolant	Generic	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions

K/A# G2.2.44 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02700 R02 Station Blackout, Step 3.2.1 and 3.2.2.

Question Source: NEW **Level Of Difficulty: (1-5)** 3

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

Objective: OPS-FLX-003

Tier being Tested: ES 401 Tier2/Group 2 (SRO)

92. Plant conditions:
- The plant is operating with ALL ICS stations in their normal line-up
- (14-4-E) ICS INPUT MISMATCH is in alarm
- Main Feedwater flow is at 11.85 MPPH and rising
- Reactor power is at 100.1% and slowly rising

Which ONE of the following procedures should be used to terminate this event?

Assume no Smart Automatic Signal Selector (SASS) operation.

A. DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture

B. DB-OP-02532, Loss of NNI/ICS Power

C. DB-OP-02526, Primary to Secondary Heat Transfer Upset

D. DB-OP-02541, Loss of YAU

Answer: C

Explanation/Justification: This question is a KA match because it tests the candidate knowledge of the effect of a Non-Nuclear Instrumentation System instrument failure. Once the failure effect is determined, the candidate must determine and then use the correct procedure to mitigate this event.

SRO Only: This is an SRO level question because it requires selection of the correct procedure to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the instrument failure and then use that knowledge to determine the appropriate procedure that provides the direction to mitigate the event. Selection of the appropriate procedure is an SRO Level task.

A. Incorrect because this is not the correct procedure to mitigate the events provided – Plausible because this is the RNO response to step 3.1 of DB-OP-02526, Primary to Secondary Heat Transfer Upset if reactor power cannot be reduced below maximum allowed power.

B. Incorrect because this is not the correct procedure to mitigate the events provided – Plausible because the candidate may assume the response information for this event is in the Loss of ICS/NNI Power Procedure since the annunciator in alarm is an ICS INPUT MISMATCH. This procedure does not provided direction to mitigate the effect of an instrument failure.

C. CORRECT –The Command SRO will determine a plant transient is in progress based on given indications. There is not enough information to determine what failed instrument is causing the event. They must make the determination to stop the transient by placing the ICS stations to hand. The correct procedure to respond to this event is DB-OP-02526 Primary to Secondary Heat Transfer Upset.

D. Incorrect because this is not the correct procedure to mitigate the events provided – Plausible because the candidate may assume the response information for this event is in the loss of YAU Procedure. YAU is the normal power supply for X NNI instruments. This procedure does provide operating direction for some events such as the response to a loss of YAU power. This procedure does not provided direction to mitigate the effect of an instrument failure.

Sys #	System	Category	KA Statement
016	Non-Nuclear Instruments	Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Detector Failure

K/A# A2.01 **K/A Importance:** 3.1 **Exam Level:** SRO

References provided to Candidate None **Technical References:** DB-OP-02526, Primary to Secondary Heat Transfer Upset R04, Step 4.3

Question Source: New **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** CFR: 41.5 / 43.5 / 45.3 / 45.5

Objective: OPS-GOP-126-02K

Davis-Besse 1LOT22 NRC Written Exam

Abbreviations: MPPH = Million Pounds Mass Per Hour
Tier being Tested: ES 401 Tier2/Group 2 (SRO)

93. Plant conditions:
- Mode 1 at 100% power
- All Technical Specifications are being met without reliance on provision in the Limiting Conditions for Operability
- Chemistry samples are in progress in response to rising trend on Main Steam Line 1 Radiation monitor.

Chemistry reports that confirmed samples indicate current SG Tube Leakage for SG 1 is 100 gallons per day. No leakage detected on SG 2.

Based on the current Steam Generator Tube Leakage,

(1) SG 1 is __________

AND

(2) _________ is the correct procedure to address this scenario.

A. (1) Operable
(2) DB-OP-01200, RCS Leakage Management

B. (1) Operable
(2) DB-OP-02531, Steam Generator Tube Leak

C. (1) Inoperable
(2) DB-OP-01200, RCS Leakage Management

D. (1) Inoperable
(2) DB-OP-02531, Steam Generator Tube Leak

Answer: A

Explanation/Justification: For these plant conditions, the applicable Technical Specifications are 3.4.13, RCS Leakage and 3.4.17, Steam Generator Tube Integrity. The measured SG 1and 2 Tube leakage is less than the allowed 150 gallons per day total SG Tube Leakage. Entry into TS 3.4.13 is not required. TS 3.4.17 requires that SG Tube Integrity must be maintained. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO. Since the Operations Leakage limits are met, TS entry into TS 3.4.17 is not required.

SRO Only: This is an SRO level question because

1) it requires selection of the correct procedure to mitigate the event as describe in NUREG 1021, ES 401, Attachment 2, E. Assessment of Facility Conditions and Selection of Appropriate Procedures during Normal, Abnormal, and Emergency Situations. The Candidate must determine the impact of the Steam Generator Tube Leakage and then use that knowledge to determine the appropriate procedure that provides the direction to mitigate the event. Selection of the appropriate procedure is an SRO Level task.

2) the decision on the operability of SG 1 requires the use of the TS Bases for TS 3.4.17. The Bases provides the following: There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO. The Bases must be used to determine the TS requirement for SG Tube Integrity and is therefore an SRO responsibility.

A. CORRECT - The measured SG 1and 2 Tube leakage is less than the allowed 150 gallons per day total SG Tube Leakage. Entry into TS 3.4.13 is not required. TS 3.4.17 requires that SG Tube Integrity must be maintained. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO. Since the Operations Leakage limits are met, TS entry into TS 3.4.17 is not required. As a result, SG 1 is operable. For SG Tube Leaks less than the TS limit of 150 gallons per day, DB-OP-01200 RCS Leakage Management provides the controlling direction. DB-OP-02531, Steam Generator Tube Leak purpose provides the following "This procedure addresses Steam Generator Tube leaks greater than TS 3.4.13 limits but less than 50 gallons per minute. Direction for leaks less than the TS Limits is provided by DB-OP-01200, Reactor Coolant System Leakage Management."

B. Incorrect – Incorrect because the leakage is less than the leakage addressed by DB-OP-02531, SG Tube Leak. This procedure addresses Steam Generator Tube leaks greater than TS 3.4.13 limits but less than 50 gallons per minute. Direction for leaks less than the TS Limits is provided by DB-OP-01200, Reactor Coolant System Leakage Management." Plausible because (1) the SG is operable and (2) the SG is leaking, but not at a level that requires entry into DB-OP-02531.

C. Incorrect – Incorrect because SG 1 is operable. The measured SG 1and 2 Tube leakage is less than the allowed 150 gallons per day total SG Tube Leakage. Entry into TS 3.4.13 is not required. TS 3.4.17 requires that SG Tube Integrity must be maintained. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO. Since the Operations Leakage limits are met, TS entry into TS 3.4.17 is not required. As a result, SG 1 is operable. Plausible because a Steam Generator Tube Leak exists and (2) DB-OP-01200, RCS Leakage Management is the correct procedure to address leak rates less that the TS allowed leakage.

D. Incorrect – Incorrect because SG 1 is operable and the procedure to be used is also incorrect. The measured SG 1and 2 Tube leakage is less than the allowed 150 gallons per day total SG Tube Leakage. Entry into TS 3.4.13 is not required. TS 3.4.17 requires that SG Tube Integrity must be maintained. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO. Since the Operations Leakage limits are met, TS entry into TS 3.4.17 is not required. As a result, SG 1 is operable. Plausible because a Steam Generator Tube Leak exists which would normally require declaring a system inoperable and (2) since SG Tube Leakage exists, it is reasonable but incorrect to conclude that DB-OP-02531, SG Tube Leak is the proper procedure to mitigate the event.

Sys #	System	Category	KA Statement
035	Steam Generator	Generic	Ability to determine operability and/or availability of safety related equipment.

K/A# G2.2.37 **K/A Importance** 4.6 **Exam Level:** SRO

References provided to Candidate None **Technical References:** TS 3.4.7 Bases page 3.4.7-3 DB-OP-02531 R24, SG Tube Leak, Section 1, and DB-OP-01200 R15, RCS Leakage Management Attachment 3

Question Source: New **Level Of Difficulty: (1-5)** 4

Question Cognitive Level: HI **10 CFR Part 55 Content**: (CFR: 41.7 / 43.5 / 45.12)

Objective: OPS-GOP-300

Tier being Tested: ES 401 Tier2/Group 2 (SRO)

Davis-Besse 1LOT22 NRC Written Exam

94. In accordance with NOP-OP-1002, Conduct of Operations, “The Command SRO limits the number of personnel in the Control Room as required to maintain a professional environment.”

Which ONE of the following personnel may have their access restricted by this NOP-OP-1002, Conduct of Operations, procedure step?

A. NRC Resident Inspector

B. Quality Assurance Supervisor

C. Fleet Operations Manager

D. Plant Maintenance Manager

Answer: D

Explanation/Justification: This question is a KA match because it tests SRO Level knowledge of the Conduct of Operations requirements to limit the number of personnel in the Control Room, which is a vital area. As required by NOP-OP-1002, Conduct of Operations, Step 4.5.2.23, “The Command SRO limits the number of personnel in the Control Room as required to maintain a professional environment. Control Room access by oversight organizations, NRC, Nuclear Oversight and Fleet Operations, cannot be restricted.”

SRO Only: This is an SRO level question because it tests the duties and responsibilities of the Command SRO as describe in NOP-OP-1002, Conduct of Operations. NUREG 1021 Section D.2.d - Questions related to 10 CFR 55.41(b) topics may also be appropriate SRO-level questions if they evaluate K/As at a level that is unique to the SRO job position. This question is unique to the Command SRO position which can only be performed by an SRO.

A. Incorrect because this individual access cannot be restricted per NOP-OP-1002, Conduct of Operations – Plausible because this individual is not an employee of Energy Harbor. The candidate may select this individual because the remaining three individuals are all Energy Harbor employees.

B. Incorrect because this individual access cannot be restricted per NOP-OP-1002, Conduct of Operations – Plausible because this individual pay grade would likely be the lowest of the individuals presented. The candidate may select this individual because they generally would not be a Manager Level or above individual.

C. Incorrect – because this individual access cannot be restricted per NOP-OP-1002, Conduct of Operations Plausible because this individual is not a member of the Senior Leadership Team for Energy Harbor, but is a Fleet member and not a direct report to anyone at the Davis-Besse site. The candidate may select this individual because they are a part of the Fleet organization and not a direct Davis-Besse employee.

D. Correct – As directed by NOP-OP-1002, this individual is not on the list of those individuals that can **NOT** be restricted access.

Sys #	System	Category	KA Statement
Gen	1	Conduct of Operations	Knowledge of facility requirements for controlling vital/controlled access.

K/A# 2.1.13 **K/A Importance** 3.2 **Exam Level: SRO**

References provided to Candidate: None **Technical References:** NOP-OP-1002 R16, Conduct of Operations Step 4.5.2.23.

Question Source: New **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: LO **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.9 / 45.10)

Objective: OPS-GOP-501-02K

Tier being Tested: ES401 Generic Knowledge and Abilities Outline (Tier 3) (SRO)

95. Plant conditions:

- The plant is in Mode 6 with Fuel Handling in progress in Containment and the Spent Fuel Pool
- All Fuel Handling will be suspended for approximately 30 hours
- All Fuel Handling Surveillances will be maintained current

Which ONE of the following requirements must be observed during the suspension?

A. A qualified individual must be assigned to continuously monitor Refueling Canal Level.

B. A dedicated Reactor Operator must be assigned to monitor the reactivity of the core (neutron count rate).

C. At least one Emergency Ventilation System Fan must be in service on the Spent Fuel Pool.

D. Make an entry in the Unit Log and Fuel Handling Directors Log stating the reason for suspension and indicate whether the Periodic Verifications are to continue.

Answer: D

Explanation/Justification: KA Match: This question matches the KA by requiring knowledge of refueling administrative requirements.

SRO Only: This question is an SRO only question based on NUREG 1021, ES-401, Attachment 2 F. Procedures and Limitations Involved in Initial Core Loading, Alterations in Core Configuration, Control Rod Programming, and Determination of Various Internal and External Effects on Core Reactivity [10 CFR 55.43(b)(6)] Some examples of SRO exam items for this topic include the following: administrative requirements associated with refueling activities.

A. Incorrect because suspending fuel handling activities does not require continuous monitoring of refueling canal level. – Plausible because lowering of Refueling Canal level requires suspension of the Fuel Handling activities.

B. Incorrect because a dedicated individual is NOT required to monitor the reactivity of the core when fuel handling is suspended – Plausible because a dedicated individual is required to be assigned to monitor the reactivity of the core (neutron count rate) during fuel handling activities that add positive reactivity to the reactor core.

C. Incorrect because maintaining one EVS Train is service is not required when suspending fuel handling activities – Plausible because this action would be required if the SFP Ventilation system was not in service. (TS 3.7.13)

D. CORRECT – This is a required action when suspending fuel handling operations as provided by DB-OP-00030, Step 6.3.3.

Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Generic	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:** DB-OP-00030 R15 Step 6.3.3

Question Source: Bank 2013 NRC Exam Q94 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: HI **10 CFR Part 55 Content:** 41.10 / 43.5 / 45.13

Objective: OPS-FHT-100-01K

Tier being Tested: ES401 Generic Knowledge and Abilities Outline (Tier 3) (SRO)

96. An overhead Annunciator in the Control Room is determined to be a nuisance alarm due to equipment conditions. The Shift Manager has determined that the Annunciator Window should be disabled.

(1) What procedure will be utilized to disable the annunciator,

AND

(2) What type of tag will be used when it is disabled?

A. (1) DB-OP-06411, Station Annunciator Operating Procedure
(2) Maintenance Information Tag

B. (1) DB-OP-06411, Station Annunciator Operating Procedure
(2) Red Danger Tag

C. (1) NOP-OP-1002, Conduct of Operations Procedure
(2) Maintenance Information Tag

D. (1) NOP-OP-1002, Conduct of Operations Procedure
(2) Red Danger Tag

Answer: A

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the process for disabling a nuisance annunciator alarm, which can be a distraction to the Control Room Crew.

SRO Only; This question is SRO only because it meets the requirements of the SRO only ES401 Att. 2, Section II .C page 6 third bullet. The SRO is required to know the administrative requirements for disabling annunciators.

A. CORRECT -
(1) Disabling an Annunciator Window is directed by DB-OP-06411, Station Annunciator Procedure Section 4.5
(2) A Maintenance Information Tag is directed to be placed on the alarm card IAW DB-OP-06411

B. Incorrect –
(1) Is correct. Disabling an Annunciator Window is directed by DB-OP-06411, Station Annunciator Procedure Section 4.5
(2) Is incorrect because a Clearance is not necessary nor directed to perform this activity IAW DB-OP-06411. A Red Danger Tag is plausible since they can be used to implement plant modifications when needed for safety requirements. Also, plausible to use OPS Only Clearance for equipment control per NOP-OP-1001 Clearance and Tagging Program Section 4.10. Also, at one time, Danger Tags were required to ensure equipment remained OOS for all engineering changes until the new or modified equipment was ready to be turned over to operations.

C. Incorrect –
(1) Is incorrect because the applicable direction is provided in DB-OP-06411 - Plausible since Conduct of Operations provides guidance for nuisance alarms.
(2) Is correct. A Maintenance Information Tag is directed to be placed on the alarm card IAW DB-OP-06411.

D. Incorrect –
(1) Is incorrect because the applicable direction is provided in DB-OP-06411 - Plausible since Conduct of Operations provides guidance for nuisance alarms.
(2) Is incorrect because a Clearance is not necessary nor directed to perform this activity IAW DB-OP-06411. A Red Danger Tag is plausible since they can be used to implement plant modifications when needed for safety requirements. Also, plausible to use OPS Only Clearance for equipment control per NOP-OP-1001 Clearance and Tagging Program Section 4.10. Also, at one time, Danger Tags were required to ensure equipment remained OOS for all engineering changes until the new or modified equipment was ready to be turned over to operations.

Sys #	System	Category	KA Statement

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N/A	N/A	Generic	Knowledge of the process for controlling temporary design changes

K/A# 2.2.11 **K/A Importance** 3.3 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-06411 Section 4.5 R29

Question Source: Bank DB 2021 SRO Written Exam Q22 **Level Of Difficulty: (1-5)** 3.5

Question Cognitive Level: Low **10 CFR Part 55 Content:** 41.10 / 43.3 / 45.13

Objective: GOP-504-03A

Tier being Tested: ES401 Generic Knowledge and Abilities Outline (Tier 3) (SRO)

97. An Equipment Operator reports one of the close control power fuse holders for Decay Heat Removal Pump 2 is discolored. The fuses are currently installed in the breaker cubicle

The Shift Manager determines this is a Priority 100 Immediate Maintenance condition as defined in NOP-WM-9001, Tool pouch, Minor, Simple, Immediate, and Emergency Maintenance. Troubleshooting is required to determine safety system status.

Per NOP-WM-9001, which of the following is **REQUIRED** to be performed prior to performing this troubleshooting?

A. Perform a 10CFR50.59 screen per NOBP-LP-4003A, 10 CFR 50.59 User Guidelines.

B. Perform a Risk Assessment per NOP-OP-1007, Risk Management.

C. Conduct a briefing per NOBP-OP-0007, Conduct of Infrequently Performed Tests or Evolutions.

D. Duty Maintenance Manager must provide direct oversight of the trouble shooting activities at the switchgear per NOP-ER-3001, Problem Solving and Decision Making.

Answer: B

Explanation/Justification: This question is a KA match because it test's the candidate's knowledge of the process for managing troubleshooting. A discolored fuse holder will require further investigation to determine the status of the fuse inside and the system's ability to perform its intended function. Blown Close Power fuses would prevent automatic start of the LPI Pump on an SFAS Actuation. Trouble shooting will be required to assess operability.

SRO Only: This question is an SRO level question based on NUREG-1021, Revision 10, Section ES-401 Attachment 2, step II.E. SRO based on procedure selection and administrative requirements determined/performed by SRO. The determination of the applicable administrative requirements for trouble shooting is specific to Senior Reactor Operators not Reactor Operators.

A. Incorrect because NOP-WM-9001 does not direct the performance of a 10CFR50.59 Screen – Plausible since this is required whenever the possibility exists that design functions of structures, systems, and components (SSCs) being relied on to support plant operation or achieve and maintain safe shutdown could be adversely impacted by the troubleshooting activity

B. CORRECT – Per NOP-WM-9001, Tool pouch, Minor, Simple, Immediate, and Emergency Maintenance step 3.1.2 a Risk Assessment is required to be performed by the Control Room Unit Supervisor or Shift Engineer prior to the Urgent Maintenance troubleshooting. NOP-OP-1007 is the parent procedure for all risk assessments which includes task risk, and PRA risk

C. Incorrect because NOP-WM-9001 nor NOBP-OP-0007 direct that this type of evolution be treated as an Infrequently Performed Test or Evolution – plausible since this evolution would be infrequent

D. Incorrect because NOP-WM-9001 does not direct that the Duty Maintenance Manager provide direct oversight of trouble shooting activities – plausible since this process is used to ensure that troubleshooting and problem-solving activities for plant issues are conducted consistently and effectively without adverse or unintended consequences on nuclear safety, personnel safety or plant performance. This procedure does not require the Duty Maintenance Manager to provide direct oversight of the trouble shooting activities.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the process for managing troubleshooting activities

K/A# 2.2.20 **K/A Importance** 3.8 **Exam Level** SRO

References provided to Candidate None **Technical References:** NOP-WM-9001 R15, Attachment 1, step 3.1.2

Question Source: Bank DB 2016 NRC Exam Q96 **Level Of Difficulty: (1-5)** 3.5

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

Objective: GOP-504-03A

Tier being Tested: ES401 Generic Knowledge and Abilities Outline (Tier 3) (SRO)

98. The following plant conditions exist:
    - A General Emergency has been declared
    - An operator must be dispatched to the Aux Building to protect valuable property

    In accordance with RA-EP-02620, Emergency Dose Control and Potassium Iodide Distribution,

    (1) what is the **maximum** dose the operator is allowed to receive for this entry,

    AND

    (2) who is responsible for authorizing this dose?

A. (1) 10 rem TEDE
   (2) Emergency Director

B. (1) 10 rem TEDE
   (2) Emergency Radiation Protection (RP) Manager

C. (1) 25 rem TEDE
   (2) Emergency Director

D. (1) 25 rem TEDE
   (2) Emergency Radiation Protection (RP) Manager

Answer: A

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the emergency radiation exposure limits.

SRO Only: This question is an SRO level question based on NUREG-1021, Revision 10, Section ES-401 Attachment 2, step II.D SRO based on knowledge of emergency dose requirements.

A. CORRECT –
   (1) Correct 10 rem is the limit for preventing serious injury and protecting valuable property as provided in RA-EP-02620.
   (2) Correct - Per RA-EP-02620 the Emergency Director is responsible for emergency dose authorizations

B. Incorrect
   (1) Correct 10 rem is the limit for preventing serious injury and protecting valuable property as provided in RA-EP-02620.
   (2) Incorrect because the Emergency Director is responsible for authorizing the dose per RA-EP-02620 – plausible because per RA-EP-02620 the Emergency Radiation Protection (RP) Manager shall be responsible for evaluating, recognizing, and formally recommending in writing to the Emergency Director the need for emergency dose authorization. In addition, the RP manager typically makes all decisions related to radiation protection

C. Incorrect
   (1) Incorrect because the allowable dose for this condition is 10 REM not 25 per RA-EP-02620 – plausible because during declared emergencies emergency workers are allowed to receive up to 25 rem TEDE for the duration of the emergency regardless of normal exposure to date for the year to perform lifesaving actions or to perform actions to protect large populations.
   (2) Correct - Per RA-EP-02620 the Emergency Director is responsible for emergency dose authorizations

D. Incorrect
  (1) Incorrect because the allowable dose for this condition is 10 REM not 25 per RA-EP-02620 – plausible because during declared emergencies emergency workers are allowed to receive up to 25 rem TEDE for the duration of the emergency regardless of normal exposure to date for the year to perform lifesaving actions or to perform actions to protect large populations.
  (2) Incorrect because the Emergency Director is responsible for authorizing the dose per RA-EP-02620 – plausible because per RA-EP-02620 the Emergency Radiation Protection (RP) Manager shall be responsible for evaluating, recognizing, and formally recommending in writing to the Emergency Director the need for emergency dose authorization. In addition, the RP manager typically makes all decisions related to radiation protection

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Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of radiation exposure limits under normal or emergency conditions

K/A# 2.3.4 **K/A Importance** 3.7 **Exam Level** SRO

References provided to Candidate None **Technical References:** RA-EP-02620 R06, step 4.1 and 6.1.3

Question Source: DB NRC 2016 NRC Exam Q98 **Level Of Difficulty: (1-5)** 2

Question Cognitive Level: Low **10 CFR Part 55 Content:** CFR: 41.12 / 43.4 / 45.10

Objective: GOP-601-01K

Abbreviations: TEDE = Total Effective Dose Equivalent

Tier being Tested: ES401 Generic Knowledge and Abilities Outline (Tier 3) (SRO)

99. The Reactor tripped during a rapid shutdown due to a Steam Generator Tube Rupture on SG 1.

Current conditions:
- All Reactor Coolant Pumps are in operation
- A SG 2 steam leak is causing a plant cooldown of 5 °F per minute
- SG 1 level is 145 inches and rising
- SG 1 pressure is 980 psig and lowering
- SG 2 pressure is 700 psig and lowering
- RCS temperature (Tcold) is 500 and lowering
- Immediate Operator Actions (DB-OP-02000 Section 3) are complete
- Supplemental Operator Actions (DB-OP-02000 Section 4) are in progress

Based on Current conditions determine the following.

(1) ____________ of DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture is required to be entered **NEXT**.

AND

(2) Requirements to invoke Specific Rule 5, PTS are ___________.

A. (1) Section 7, Overcooling
(2) MET

B. (1) Section 7, Overcooling
(2) NOT MET

C. (1) Section 8, Steam Generator Tube Rupture
(2) MET

D. (1) Section 8, Steam Generator Tube Rupture
(2) NOT MET

Answer: B

Explanation/Justification: This question is a KA match because it tests the SRO Candidates knowledge of the parameters (RCS Temp, C/D rate, etc.) and the logic (priorities of symptoms for Emergency Operating Procedure DB-OP-02000) used to assess the status of reactor coolant system integrity

SRO Only:

This is an SRO level question based on NUREG 1021 R10, Section ES-401 Attachment 2, step II. E. SRO since must know the priority requirements to determine which malfunction to address first and which routing to select the correct section of the Emergency Operating Procedure. Meets the K/A by assessing the parameters and logic to determine if the requirements to invoke PTS is met and selecting the proper procedure section for mitigation.

A. Incorrect –
(1) Correct – Section 7, Overcooling is a higher priority symptom and is therefore addressed prior to SGTR.
(2) Incorrect because the requirements to invoke PTS have not been met based on the parameters provided. Plausible because cooldown rate is excessive and RCS temperature is low. RCS temperature is 360°F Tcold AND Cooldown greater than 100 °F per hour. Cooldown is greater than 100 °F per hour which meets the PTS criterial, but, Tcold is above 360°F as stated so PTS is not required to be invoked.

B. CORRECT –

(1) Correct - DB-OP-02000 TBD hierarchy states that Section 7 has a higher priority than Section 8.

(2) Correct - Based on the parameters provided, is 360°F Tcold AND Cooldown greater than 100 °F per hour. Cooldown is greater than 100 °F per hour, but, Tcold is above 360°F as stated so PTS is not required to be invoked.

C. Incorrect –

(1) Incorrect because EOP DB-OP-02000, Section 7, Overcooling is a higher priority symptom and is therefore addressed prior to Section 8, SGTR. Plausible because a Steam Generator Tube Rupture was the initiating event and is still in progress.

(2) Incorrect because the requirements to invoke PTS have not been met based on the parameters provided. Plausible because cooldown rate is excessive and RCS temperature is low. RCS temperature is 360°F Tcold AND Cooldown greater than 100 °F per hour. Cooldown is greater than 100 °F per hour which meets the PTS criterial, but, Tcold is above 360°F as stated so PTS is not required to be invoked.

D. Incorrect

(1) Incorrect EOP DB-OP-02000, Section 7, Overcooling is a higher priority symptom and is therefore addressed prior to Section 8, SGTR. Plausible because a Steam Generator Tube Rupture was the initiating event and is still in progress.

(2) Correct - Based on the parameters provided, is 360°F Tcold AND Cooldown greater than 100 °F per hour. Cooldown is greater than 100 °F per hour, but, Tcold is above 360°F as stated so PTS is not required to be invoked.

Sys #	System	Category	KA Statement
04	GEN	Emergency Procedures/Plan	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 system integrity, containment conditions, radioactivity release control, etc.

K/A# 2.4.21 **K/A Importance** 4.6 **Exam Level:** SRO

References provided to Candidate: None

Technical References: DB-OP-01003 R16 Operations Procedure Use Instructions pg 9-10, DB-OP-02000 R32, Section 4 pages 24 – 29, and Specific Rule 5 page 273

Question Source: DB 2016 NRC Exam Q78

Level Of Difficulty: (1-5) 3

Question Cognitive Level: HI

10 CFR Part 55 Content: CFR: 41.7 / 43.5 5.12

Objective: GOP-303, 519,

Tier being Tested: ES401 Generic Knowledge and Abilities Outline (Tier 3) (SRO)

100. The Reactor is operating at 100% power.

The following Annunciators are in alarm due to scheduled maintenance:

- Annunciator 9-2-D, AUX BLR SYS TRBL
- Annunciator 15-5-B, DEHC MINOR ALARM

During maintenance an unexpected event occurs, and the following conditions are noted:

- Computer Point Q007, UNIT ANNUC SYS, alarms
- All Station Annunciator alarm windows that were previously in alarm are now extinguished
- Blue Annunciator 10-6-A has also extinguished
- Reactor Power remains stable at 100%

I&C reports it will take approximately 90 minutes to repair and restore annunciators to their previous condition.

Based on these conditions, complete the following statements.

(1) A loss of__________ is the most likely cause of this event.

AND

(2) __________ is required for this event.

A. (1) a single Annunciator DC Power Supply
(2) No Emergency Plan Classification

B. (1) a single Annunciator DC Power Supply
(2) A declaration of an Unusual Event

C. (1) ALL Annunciator AC Power
(2) No Emergency Plan Classification

D. (1) ALL Annunciator AC Power
(2) A declaration of an Unusual Event

Answer: C

Explanation/Justification: This question is a KA match because it tests the candidate's knowledge of the indications for a loss of all annunciators (as noted by loss of Blue Annunciator 10-6-A) followed by the SRO only actions to classify the event. Classification is an Emergency Plan Procedure. The Station Annunciators has redundant AC power supplies that are protected by an automatic bus transfer (ABT). This scenario would be possible on AC power supply failure and the failure of the ABT to transfer to the remaining power supply.

SRO Only: This is an SRO level question because it tests the duties and responsibilities of the Shift Manager – Classification of Emergency Plan Events. NUREG 1021 Section D.2.d - Questions related to 10 CFR 55.41(b) topics may also be appropriate SRO-level questions if they evaluate K/As at a level that is unique to the SRO job position. This question is unique to the Shift Manager position which can only be performed by an SRO.

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A. Incorrect

(1) Incorrect because the loss of a single DC Annunciator Power supply would not cause all annunciators to extinguish – Plausible because the loss of a single DC power supply will cause some annunciator lights to extinguish, but not all annunciators to extinguish. The alarms in at the start of the event are powered from difference DC Power supplies. In the stem of the question, the annunciators provided are supplied by DC power supplies 9-2-D by DC power Supply 10 and 15-5-B by DC power supply 16. The loss of a single DC power supply will not extinguish all of the provided lights.

(2) Correct because he classification for this event is correct per RA-EP-01500, Emergency Classification. Plausible because there are Emergency Action Levels (EALs) related to annunciator power. The closest potentially applicable EAL is SU3.1 Unusual Event - An UNPLANNED event results in the inability to monitor one or more Table S-2 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. For this case, the computer is still available to monitor the Table S-2 Safety Systems as noted by the computer alarm in the stem. As a result, the correct classification is None.

B. Incorrect

(1) Incorrect because the loss of a single DC Annunciator Power supply would not cause all annunciators to extinguish – Plausible because the loss of a single DC power supply will cause some annunciator lights to extinguish, but not all to extinguish. In the stem of the question, the annunciators provided are supplied by DC power supplies 9-2-D by DC power Supply 10 and 15-5-B by DC power supply 16. The loss of a single DC power supply will not extinguish all of the provided lights.

(2) Incorrect The classification for this even is incorrect per RA-EP-01500, Emergency Classification. Plausible because EAL is SU3.1 Unusual Event - An UNPLANNED event results in the inability to monitor one or more Table S-2 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min is close. For this case, the computer is still available to monitor the Table S-2 Safety Systems as noted by the computer alarm in the stem. As a result, the correct classification is None.

C. CORRECT –

(1) The loss of all AC power supply will cause all annunciator lights to extinguish, In the stem of the question, the annunciators provided are supplied by DC power supplies 9-2-D by DC power Supply 10 and 15-5-B by DC power supply 16. The loss of a single DC power supply will not extinguish all of the provided lights. This would occur for the loss of AC power.

(2) The classification for this event is correct per RA-EP-01500, Emergency Classification. The closest potentially applicable EAL is SU3.1 Unusual Event - An UNPLANNED event results in the inability to monitor one or more Table S-2 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min. For this case, the computer is still available to monitor the Table S-2 Safety Systems as noted by the computer alarm in the stem. As a result, the correct classification is None.

D. Incorrect

(1) Correct because the loss of all AC power supply will cause all annunciator lights to extinguish, In the stem of the question, the annunciators provided are supplied by DC power supplies 9-2-D by DC power Supply 10 and 15-5-B by DC power supply 16. The loss of a single DC power supply will not extinguish all of the provided lights. This would occur for the loss of AC power

(2) Incorrect because the classification for this event is incorrect per RA-EP-01500 Emergency Classification – Plausible because EAL SU3.1 Unusual Event - An UNPLANNED event results in the inability to monitor one or more Table S-2 SAFETY SYSTEM parameters from within the Control Room for ≥ 15 min is close, it is not correct. For this case, the computer is still available to monitor the Table S-2 Safety Systems as noted by the computer alarm in the stem. As a result, the correct classification is None.

Sys #	System	Category	KA Statement
04	GEN	Emergency Procedures/Plan	Knowledge of operator response to loss of all annunciators.

K/A# 2.4.32 **K/A Importance** 4.0 **Exam Level:** SRO

References provided to Candidate: RA-EP-01500 Classification Wallboard (DBRM-EMER-1500B)

Technical References: DB-OP-06411 R29, Station Annunciator Operating Procedure Section 5.1, Loss of Station Annunciators, RA-EP-01500 R16, Emergency Classification, DBRM-EP-1500A R09, Davis-Besse Emergency Action Level Bases Document, Attachment 1, Basis for SU3.1

Question Source: NEW **Level Of Difficulty: (1-5)** 3

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

Objective: OPS-GOP-602-04K

Tier being Tested: ES401 Generic Knowledge and Abilities Outline (Tier 3) (SRO)