1. Which ONE of the following is the basis for TRIPPING all Reactor Coolant Pumps (RCPs) within 2 minutes due to lack of adequate subcooling margin?

A. To prevent two phase flow through the RCPs

B. To reduce the heat input to the RCS from the operating pumps

C. To keep a high void fraction from uncovering the core if pumps were later stopped

D. To increase High Pressure Injection flow by lowering RCS cold leg pressure

---

Answer: **C**

Explanation/Justification:

A. Incorrect. Plausible because two phase flow would occur with a lack of adequate subcooling margin (SCM) and two phase flow would damage the RCP

B. Incorrect. Plausible because stopping the RCP would reduce heat input into the RCS and help to restore SCM

C. Correct. The RCPs are tripped immediately upon loss of adequate SCM to prevent possible core damage if a subsequent trip of the RCPs occurred during certain size small break LOCAs. If the RCS void fraction is greater than about 70 percent when RCPs are tripped, the peak clad temperature can exceed the maximum temperature allowed by 10CFR50.46. A manual trip of the RCPs before the RCS void fraction reaches 70 percent prevents this possibility.

D. Incorrect. Plausible because stopping the RCP would lower the RCP discharge pressure where HPI injects

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Sys #	**System**	**Category**	**KA Statement**
007	Rx Trip Stabilization-Recovery	Knowledge of the reasons for the following as the apply to a reactor trip:	Actions contained in EOP for reactor trip

K/A# EK3.01 **K/A Importance** 4.0 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000 TBD, pg 50

Question Source: Bank 29768 **Level Of Difficulty: (1-5)** 2

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

Objective:

2. The following plant conditions exist:
   - RCS pressure 885 psig
   - Quench Tank pressure 35 psig

   The following event occurs:
   - The PORV starts leaking

   For these conditions, what will be the PORV downstream tailpipe temperature?

A. ≈212 °F

B. ≈280 °F

C. ≈320 °F

D. ≈540 °F

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because this would be the saturation temperature at atmospheric conditions

B. Incorrect. Plausible if the candidate does not realize that this would result in superheat, and thus stops at the saturation line on the mollier diagram.

C. Correct. IAW Steam tables and isenthalpic throttling process.

D. Incorrect. Plausible if the candidate uses the RCS pressure to determine the downstream temperature.

Sys #	System	Category	KA Statement
008	Pressurizer (PZR) Vapor Space Accident	Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following:	Valves

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

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

Question Source: Bank 37991 **Level Of Difficulty: (1-5)** 3

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

Objective:

3. The following plant conditions exist:

- The plant is in Mode 3
- Tave is 534 °F and stable
- Letdown flow is 50 gpm
- MU flow is 150 gpm
- Seal injection flow is 36 gpm with MU19 in HAND
- Seal return flow is 6 gpm
- Pressurizer level is decreasing at 2 inches/minute

Which of the following is the approximate RCS leak rate?

A. 290 gpm

B. 242 gpm

C. 178 gpm

D. 82 gpm

Answer: C

Explanation/Justification:

A. Incorrect. Plausible if candidate assumes all the values should be included into the leak rate

B. Incorrect. Plausible if candidate assume all values, except Pressurizer level drop (48 gpm) should be included in the leak rate

C. Correct. Water loss = 50 gpm + 6 gpm = 56 gpm. Water makeup = 150 gpm + 36 gpm = 186 gpm; Difference186 -56 = 130 gpm loss. Since Pzr level is still lowering at 2 inch/min (@ 24 gal/inch) = 48 gpm,
Total loss is 130 gpm + 48 gpm = 178 gpm leak

D. Incorrect. Plausible if candidate does not recognize that the Pressurizer level drop (48 gpm) should not be subtracted in the leak calculation

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Sys #	System	Category	KA Statement
009	Small Break LOCA	Generic	Knowledge of abnormal condition procedures.

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

References provided to Candidate None **Technical References:** DB-OP-02522 pg 25 & 26

Question Source: Bank 47621 **Level Of Difficulty: (1-5)** 3.5

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

Objective:

4. Initial plant conditions:

- Plant was operating at 100% power.
- #2 Makeup Pump is running.
- All systems were operating normally.

The following events occur and the following conditions exist:

- A large break LOCA
- The reactor has tripped.
- RCS Pressure = 100 psig
- Incore Thermocouples = 325 °F
- Containment Pressure = 28 psia
- EDG 1 did not start.
- All other safety equipment actuated as designed.
- Assume NO Operator actions have been completed

For these conditions, what will be the status of the Makeup, HPI, and LPI pumps?

A. No Makeup Pumps will be running
Both HPI Pumps will be running
Both LPI Pumps will be running

B. Only #2 Makeup Pump will be running
Only #2 HPI Pump will be running
Only #2 LPI Pump will be running

C. No Makeup Pumps will be running
Only #2 HPI Pump will be running
Only #2 LPI Pump will be running

D. Only #2 Makeup Pump will be running
Both HPI Pumps will be running
Both LPI Pumps will be running

Answer: **D**

Explanation/Justification:

A. Incorrect. Plausible if candidate does not recognize no operator actions have been taken and #1 MUP has to be manually started

B. Incorrect. Plausible if candidate does not recognize that D1 is energized from offsite power

C. Incorrect. Plausible if candidate does not recognize that the Makeup Pump only trips if there is a LOOP and LPIP starts

D. Correct. An SFAS (SA) Level 3 has actuated on low RCS pressure (<450 psig) and offsite power is supply power to the essential electrical busses. SA level 2 starts the HPIPs and SA level 3 starts the LPIPs. The previous running MU pump will continue to run because a LOOP has not ocurred

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

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

References provided to Candidate None **Technical References:** DB-OP-02000 Table 2, pg 410 & 412

Question Source: New **Level Of Difficulty: (1-5)** 4

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

Objective:

5. The following plant conditions exist:
   - The plant is operating at 100% power.

   The following occurs:
   - Component Cooling Water (CCW) pressure exiting the Reactor Coolant Pump (RCP) seal package is 145 psig.

   Based on these indications, which of the below conditions exist?

A. Seal injection water is leaking into the CTMT CCW return header.

B. The RCP seal cooler CCW relief valve is lifting.

C. No abnormalities present, CCW system is operating normally.

D. Low CCW pressure on the outlet of the RCP seal cooler caused the RCP CCW outlet valve to close.

Answer: A

Explanation/Justification:

A. Correct. Depending on system load, normal Component Cooling Water pressure is approximately 80 psig. Reactor Coolant Pump Seal Injection Water is supplied from the Makeup Pump. A leak in the RCP Seal cooler would cause a rise in the CCW Header exiting the RCP Seal Package.

B. Incorrect. This is plausible if the candidate incorrectly assumes this portion of the CCW header has a relief valve. The value provided is similar to the value used in other CCW reliefs associated with the Containment header such as CC3953 – setpoint 150 psig. This CCW piping is rated for full reactor coolant system pressure and does not include a relief valve.

C. Incorrect. This is plausible if the candidate does not know normal system operating pressure for the CCW system.

D. Incorrect. This is plausible since the associated valves (CC4100, CC4200, CC4300, and CC4400 close on high CCW System Pressure of 150 psig, not low system pressure.

Sys #	**System**	**Category**	**KA Statement**
015/017	Reactor Coolant Pump (RCP) Malfunctions	Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow):	Cause of RCP failure

K/A# AA2.01 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02523, pg 60 & 70, DB-PF-6704 pg 41 (CCW Curve 14.20a) OS-001B Sh 2, CL-1

Question Source: Bank 35734 **Level Of Difficulty: (1-5)** 3

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

Objective:

6. The following plant conditions exist:
   - Mode 5
   - Decay Heat Pump 2 is cooling the core

   The following events occur:
   - Loss of both Decay Heat pumps occurs
   - High Pressure Injection Train 1 is being used to cool the core.
   - HPI flow is throttled.

   Which of the following instruments will be used to determine RCS temperature per DB-OP-02527, Loss of Decay Heat procedure?

A. Pressurizer water temperature

B. Incore thermocouples temperature

C. Reactor Coolant System Cold Leg temperature

D. Reactor Coolant System Hot Leg temperature

Answer: **B**

Explanation/Justification:

A. Incorrect. Plausible if the candidate assumes Feed and Bleed Cooling out the Pressurizer PORV is in progress. Pressurizer temperature would follow core temperature since in this mode of cooling, RCS inventory travels through the core, then to the pressurizer, and then out the PORV.
B. Correct. Incore Thermocouples are located in the Core immediately adjacent to the Fuel. These indicators provide the best indication of core conditions when forced circulation of the Reactor Coolant System is not in progress.
C. Incorrect. Plausible if the candidate does not recognize that the temperature of the cold leg side of the core, where DH flow enters the core, is not representative of core temperature.
D. Incorrect. Plausible if the candidate assumes that RCS Hot Leg temperature is indicative of Core Temperature as it would be with RCPs in operation.

Sys #	**System**	**Category**	**KA Statement**
025	Loss of Residual Heat Removal System (RHRS)	Generic	Ability to identify post-accident instrumentation.

K/A# 2.4.3 **K/A Importance** 3.7 **Exam Level** RO
References provided to Candidate None **Technical References:** DB-OP-02527 pg 80
Question Source: New **Level Of Difficulty: (1-5)** 3
Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** (CFR: 41.6 / 45.4)
Objective:

7. The plant is at full power.

- The DSA **AND** DSB Emergency Control Transfer Switches on breaker AC 113 for Component Cooling Water (CCW) Pump #1 have been placed in LOCAL.

CCW Pump #1 __(1)___ be started using its Control Room control switch.

CCW Pump #1 __(2)___ start automatically from a Safety Features Actuation Signal.

A. (1) can
(2) will

B. (1) can
(2) will NOT

C. (1) can NOT
(2) will

D. (1) can NOT
(2) will NOT

Answer: **D**

Explanation/Justification:

A. Incorrect. Plausible if the candidate does not know how these switches function

B. Incorrect. Plausible if the candidate does not know how these switches function

C. Incorrect. Plausible if the candidate does not know how these switches function

D. Correct. If the DSA switch is placed in LOCAL, the SFAS signal will be isolated. If the DSB switch is placed in LOCAL, the Control Room signal is blocked.

Sys #	**System**	**Category**	**KA Statement**
026	Loss of Component Cooling Water (CCW)	Generic	Ability to explain and apply system limits and precautions.

K/A# 2.1.32 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-06262, pg 9, Step 2.2.15

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

Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** (CFR: 41.10 / 43.2 / 45.12)

Objective:

8. The plant is operating at 100% power all systems in normal alignment.
   - PRS RC2B Narrow Range Pressure Recorder is selected to RC Pressure Channel 1
   - RC Pressure Channel 1 slowly drifts to 2300 psig
   - All other RPS RCS pressure transmitters are 2155 psig and stable

   The indications of the RCS pressure control system are as follows:
   - The PZR PORV green light is LIT and red light is NOT LIT
   - The PZR spray valve green light is LIT; red and amber lights are NOT LIT
   - SCR heater bank 1 demand is ZERO

   Based on these indications, what is the current status of the RCS pressure control system?

A. PZR PORV, PZR spray valve, and SCR heater bank 1 are all functioning as designed.

B. The PZR spray valve and SCR heater bank 1 are functioning as designed; the PZR PORV is failed closed.

C. PZR PORV and PZR spray valve are functioning as designed; the SCR heater bank 1 is failed at ZERO demand.

D. The PZR PORV and SCR heater bank 1 are functioning as designed; the PZR spray valve is failed closed.

Answer: **D**

Explanation/Justification:

A. Incorrect. The PRZ spray valve should be open for the conditions in the stem. Heaters and PORV are functioning correctly.

B. Incorrect. The PRZ spray valve should be open and the PORV is not failed. Heaters are functioning correctly.

C. Incorrect. The PRZ spray valve should be open for the conditions in the stem. Heaters should be at zero demand. PORV is functioning correctly.

D. Correct. IAW DB-OP-02513 Symptoms Step 2.3, the Spray Valve should open when the selected pressure reaches 2205 psig. Heaters should be off when RCS pressure is greater than 2155, PORV doesn't open until $\geq$2450 psig. Based on current plant conditions (RC Pressure Channel 1 failure at 2300 psig) the Spray Valve should be open, Heaters off, and PORV closed.

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 Symptoms Step 2.3

Question Source: New **Level Of Difficulty: (1-5)** 3.25

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

Objective:

9. An Anticipated Transient Without Scram (ATWS) event has occurred.

Emergency Boration is in progress.

Which of the conditions and reasons listed below will result in achieving Reactor shutdown (Mode 3) conditions the **FASTEST**?

(Consider ONLY the impact of the Emergency Boration and assume the Emergency Boration boron concentration and flowrates are the same for each case).

A. BOL initial conditions due to the more negative boron worth than the EOL initial conditions.

B. BOL initial conditions due to the less negative boron worth than the EOL initial conditions.

C. EOL initial conditions due to the more negative boron worth than the BOL initial conditions.

D. EOL initial conditions due to the less negative boron worth than the BOL initial conditions.

Answer: C

Explanation/Justification:

A. Incorrect. Boron worth is NOT more negative at BOL

B. Incorrect. Boron worth is less negative at BOL, however the impact will be a longer time to reactor shutdown

C. Correct. Boron worth is more negative at EOL, which results in higher neutron absorption rate at EOL and thus a faster time to reactor shutdown

D. Incorrect. Correct initial condition, but the reason is incorrect.

Sys #	System	Category	KA Statement
029	Anticipated Transient Without Scram (ATWS)	Knowledge of the operational implications of the following concepts as they apply to the ATWS:	Effects of boron on reactivity

K/A# EK1.03 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-NE-06201 Fig. 5A; Rx Theory Ch 4

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

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

Objective:

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.

A. End of Cycle
Main Steam Line double ended rupture **UPSTREAM** of the Main Steam Isolation Valve

B. End of Cycle
Main Steam Line double ended rupture **DOWNSTREAM** of the Main Steam Isolation Valve

C. Beginning of Cycle
Main Steam Line double ended rupture **UPSTREAM** of the Main Steam Isolation Valve

D. Beginning of Cycle
Main Steam Line double ended rupture **DOWNSTREAM** of the Main Steam Isolation Valve

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Answer: A

Explanation/Justification:

A. Correct answer - The Negative Moderator Temperature and Doppler Coefficients are largest at EOC. This large negative value produces the largest reactivity insertion due to the RCS cooldown.

B. 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. As a result, the inventory available to cause cooling is less for this break location than the upstream location.

C. Plausible if student doesn't know EOC has the largest negative coefficients

D. Plausible if student doesn't know EOC has the largest negative coefficients

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

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

References provided to Candidate None **Technical References:** UFSAR pg 15.4-24a, and Fig 15.4.4-5

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

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

Objective:

11. Plant conditions:

Time = 0915
- Rx trip occurs from 100% due to loss of offsite power
- All feedwater is lost

Time = 0925
- MU/HPI PORV Cooling in progress
- Pressurizer Level = 320 inches

Time = 0935
- SBODG has been started
- Both SGs indicate 12 inches

Time = 0945
- MDFP is started in the AFW Mode
- 800 gpm feed flow has been established to SG 2

Based on the above conditions, which ONE of the following statements describes the effects the MDFP flowrate will have on the RCS over the next <u>several</u> minutes?

Incore Thermocouples will __________________ with _____________ in RCS pressure.

A. decrease
a decrease

B. increase
an increase

C. increase
no significant change

D. decrease
no significant change

<u>Answer</u>: A

Explanation/Justification:

A. Correct - The rate of addition of MDFP flow to the Steam Generator exceeds the feedwater flow to remove decay heat. As a result, Steam Generator level will rise, and SG Pressure will lower. This will cause primary to secondary differential temperature to rise and result in heat transfer from the primary to the steam generator. As a result, primary temperatures and pressure will lower since the RCS is saturated.

B. Incorrect - Plausible if the candidate assumes addition of water to the SG will cause SG pressure to rise and therefore reduced primary to secondary heat transfer causing an increase in primary temperatures.

C. Incorrect – Plausible if the candidate assumes addition of water to the SG will cause SG pressure to rise but assumes the rise in pressure would be limited by the Main Steam Safety Valves causing no significant change in RCS pressure.

D. Incorrect - Plausible since the rate of addition of MDFP flow to the Steam Generator exceeds the feedwater flow to remove decay heat. As a result, Steam Generator level will rise, and SG Pressure will lower. This will cause primary to secondary differential temperature to rise and result in heat transfer from the primary to the steam generator but if the candidate then assumes the rise in pressure would be limited by the Main Steam Safety Valves causing no significant change in RCS pressure.

Sys #	System	Category	KA Statement
054	Loss of Main Feedwater (MFW)	Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW):	Effects of feedwater introduction on dry S/G

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

References provided to Candidate None **Technical References:** DB-OP-02000 TBD Attach 25 pg 494

Question Source: Bank 79051 Modifed by changing Tave to incore TCs in the stem and removing “large” from A and B **Level Of Difficulty: (1-5)** 4

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

12. The plant was operating at 100% power with all systems in a normal lineup.

    - Letdown flow is 70 gpm.

    The following event occurs:
    - "A" Bus Voltage is zero volts
    - "B" Bus Voltage is zero volts
    - "C1" Bus Voltage is zero volts
    - "D1" Bus Voltage is zero volts
    - Instrument Air Header Pressure lowers rapidly to zero.
    - Assume **NO** Operator actions.

    How will RCS Makeup flow and letdown flow respond to these conditions?

    (1) Actual RCS Makeup flow _______________.

    (2) Actual Letdown flow __________________.

A. (1) rises
(2) remains the same

B. (1) rises
(2) lowers

C. (1) lowers
(2) remains the same

D. (1) lowers
(2) lowers

Answer: **D**

Explanation/Justification:

A. Incorrect. This is plausible if the candidate assumes MU6 fails open on a loss of air and does not understand all Makeup Flow will be lost.

B. Incorrect. This is plausible if the candidate assumes MU6 fails open on a loss of air.

C. Incorrect. This is plausible if the candidate does not understand all Makeup Flow will be lost.

D. Correct. MU6 must be open prior to the event to have 70 gpm letdown flow. MU6 fails closed, so letdown flow lowers. Letdown flow is still through the Letdown orfice, Makuep Flow lowers because power is lost to both Makeup Pumps.

Sys #	System	Category	KA Statement
055	Loss of Offsite and Onsite Power (Station Blackout)	Ability to determine or interpret the following as they apply to a Station Blackout:	Existing valve positioning on a loss of instrument air system

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

References provided to Candidate None **Technical References:** OS 2 Sheet 1 for MU6
DB-OP-06006, pg 140 & 141

Question Source: Bank 84148 **Level Of Difficulty: (1-5)** 3.5

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

Objective:

13. The plant was operating at 100% power with all systems in a normal lineup.

The following event occurs:

- All offsite power is lost
- All systems function as designed

Based on these conditions, what will be the status of the EDGs and how will this affect the operator's ability to control frequency from the control room?

The EDGs will be operating in the _______(1)___________ mode and the operators______(2)______ adjust frequency from the control room.

A. (1) isochronous
(2) can

B. (1) isochronous
(2) can NOT

C. (1) droop
(2) can

D. (1) droop
(2) can NOT

Answer: B

Explanation/Justification:

A. Incorrect – Plausible because the EDG will be operating in Isochronous, however per DB-OP-6316 step 2.2.18, the normal controls for EDG Speed will not function until SYNC TO GRID AFTER LOOP switch is repositioned.

B. Correct per DB-OP-06316, step 2.2.18.

C. Incorrect – Plausible however per DB-OP-6316 step 2.2.18, the normal controls for EDG Speed will not function until SYNC TO GRID AFTER LOOP switch is repositioned.

D. Incorrect – Plausible because the EDG normally operates in Droop mode to allow synchronization.

Sys #	System	Category	KA Statement
056	Loss of Offsite Power	Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power:	Adjustment of speed of ED/G to maintain frequency and voltage levels

K/A# AA1.04 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-6316 step 2.2.18

Question Source: New **Level Of Difficulty: (1-5)** 4

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

Objective:

14. The plant is operating at 100% power in a normal system alignment.

The following events occur:

- C1 Bus Lockout
- C1 Bus Volts are ZERO

IAW Attachment 5 Selective Battery Load Shedding of DB-OP-02521, Loss of AC Bus Power Sources:

What actions are required?

A. Manually transfer YAU from Inverter YVA to Panel YAR.

B. Manually shed selected loads from YAU.

C. Manually transfer Y1 to alternate supply Constant Voltage Transformer XY1 at Inverter YV1.

D. Manually shed selected loads from Y1.

Answer: A

Explanation/Justification:

A. Correct. IAW DB-OP-02521, Attachment 5.

B. Incorrect. This answer is plausible if off-site power is NOT available, and BOTH Emergency Diesel Generators fail to start, load would be removed from both DC Train Batteries by opening load breakers in YAU.

C. Incorrect. This answer is plausible because Y1 remain energized following a loss of all AC power, however the alternate supply is not available when C1 is locked out.

D. Incorrect. This answer is plausible because Y1 remain energized following a loss of all AC power, however the alternate supply is not available when C1 is locked out. Load shedding Y1 would reduce load on batteries but is not directed by the procedure.

Sys #	System	Category	KA Statement
057	Loss of Vital AC Electrical Instrument Bus	Ability to operate and / or monitor the following as they apply to the Loss of Vital AC Instrument Bus:	Manual inverter swapping

K/A# AA1.01 **K/A Importance** 3.7* **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02521, Attachment 5

Question Source: New **Level Of Difficulty: (1-5)** 4

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

Objective:

15. The following plant condition exist:

- Plant is operating at 100% power.
- All systems are in a normal lineup.

A loss of D1P and DAP occurs.

- DB-0P-02537, LOSS OF D1P **AND** DAP has been entered.

Based on these conditions, what will be the status of EDG 1?

EDG 1 will:____ .

A. NOT start automatically and CANNOT be started manually from the Control Room

B. start automatically and run at idle speed (450 rpm) but will NOT accelerate to 900 rpm

C. NOT start automatically but can be started manually from the Control Room

D. start automatically and run at 900 rpm but CANNOT be placed on its associated 4160 VAC Essential Bus

---

Answer: A

Explanation/Justification:

A. Correct. Control power has been lost to EDG 1 and C1. IAW DB-OP-02537, pg 35 this is the result.
B. Incorrect. Plausible because the EDG has an idle start feature and loss of D1P and DAP will affect control logic for EDG start.
C. Incorrect. Plausible because the EDG will not start automatically
D. Incorrect. Plausible because control power is lost and the EDG output breaker, AC101, can not be closed

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Sys #	System	Category	KA Statement
058	Loss of DC Power	Ability to operate and / or monitor the following as they apply to the Loss of DC Power:	Vital and battery bus components

K/A# AA1.03 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02537, pg 35

Question Source: Bank 39158 **Level Of Difficulty: (1-5)** 3

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

Objective:

16. The following plant conditions exist:

- A loss of Offsite power has occurred.
- 4160 V Bus C1 has locked out.
- CCW Train 1 was in operation when the transient occurred.
- Ambient temperature is 85 °F.

Which of the following describes the Service Water and CCW response to these conditions?

(1) ____________ will cool CCW Train 2.

(2) Secondary loads will _______________.

A. (1) Backup Service Water Pump
(2) have NO cooling

B. (1) Backup Service Water Pump
(2) be cooled by Circ Water

C. (1) SW Train 2
(2) have NO cooling

D. (1) SW Train 2
(2) be cooled by Circ Water

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because BUSWP could be align to supply SW 2, but not before SWP 2 restarts when EDG 2 starts

B. Incorrect. Plausible because BUSWP could be align to supply SW 2, but not before SWP 2 started, Circ Water will supply secondary until SWP restarts except that the loss of Off-site power causes all Circ Water Pumps to stop.

C. Correct. D1 losses power, 10 sec late EDG 2 supplies D1, 20 sec SWP 2 starts, SW 1395 will go close on low pressure. SW Train 2 is the normal supply to CCW Train 2. SW Pump 1 will not start (C1 bus not energized), backup cooling from Circ Water System, via CT2955, will not occur because both 13.8 KV electrical busses are deenergized and Circ Water Pumps are off.

D. Incorrect. Plausible because SW 2 does supply CCW, Circ water will supply secondary until SWP restarts except Circ Water is not available during a loss of off-site power.

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

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

References provided to Candidate None **Technical References:** DB-OP-02511 background 4.1
OS-20 sh 1 & sh 2 Control logic statement

Question Source: Bank 36847 **Level Of Difficulty: (1-5)** 3

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

Objective:

17. The plant is at 100% power, all systems are in a normal lineup.

- The air line to MU 32, Makeup Flow Controller, breaks causing a loss of air to MU 32 ONLY.

Determine the effect this failure has on the water levels of the Pressurizer and the Makeup Tank.

Assume no operator actions.

(1) Pressurizer level will _______________.

(2) Makeup Tank level will _______________.

A. (1) rise
(2) rise

B. (1) lower
(2) lower

C. (1) rise
(2) lower

D. (1) lower
(2) rise

<u>Answer</u>: C

Explanation/Justification:

A. Incorrect. Plausible if candidate does not know MU32 fails open on a loss of air

B. Incorrect. Plausible if candidate knows MU32 fails open on loss of air but does not consider source of inventory being added to the Pressurizer.

C. Correct. MU32 fails open on a loss of air. This will add water to the RCS and remove it from the Makeup Tank

D. Incorrect. Plausible if candidate does not know MU32 fails open on a loss of air

---

Sys #	**System**	**Category**	**KA Statement**
065	Loss of Instrument Air	Ability to determine and interpret the following as they apply to the Loss of Instrument Air:	Failure modes of air-operated equipment

K/A# AA2.08 **K/A Importance** 2.9* **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02528, pg 77

Question Source: Bank 33899 **Level Of Difficulty: (1-5)** 2

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

Objective:

18. The following plant conditions exist:

- The plant has experienced a loss of all feedwater event.
- Subcooling margin is 10 °F.
- Makeup, High Pressure Injection, and PORV cooling is providing core cooling.
- Primary to secondary heat transfer does NOT exist.

Auxiliary Feedwater has been regained and both Steam Generators are at the correct levels.

Which of the following actions will be performed **FIRST** to restore primary to secondary heat transfer?

A. Bump start Reactor Coolant Pumps to induce Steam Generator heat transfer.

B. Begin restoration from Makeup, High Pressure Injection, Pilot Operated Relief Valve cooling by closing the Pilot Operated Relief Valve.

C. Raise Steam Generator levels to 200 inches to raise the secondary heat sink.

D. Lower Steam Generator pressures to reduce Steam Generator saturation temperature.

Answer: D

Explanation/Justification:

A. Incorrect - Plausible since bumping Reactor Coolant Pumps is a strategy (CHLA) to induce heat transfer directed by a Severe Accident Mitigation Guidelines. Severe Accident conditions do not exist.

B. Incorrect – Plausible since closing the PORV will prevent further RCS inventory loss that could lead to RCS Hot Leg voids that would interrupt single phase natural circulation heat transfer. Closing the PORV would interrupt the current method of heat transfer.

C. Incorrect – Plausible since raising SG level to 200 inches is used during Inadequate Core Cooling as a strategy to promote boiler condenser cooling. ICC condition (superheated incores) do not exist.

D. Correct. Lowering SG pressure will establish a temperature differential to ensure the SG will act as a heat sink for the RCS. RO only since it involves a mitigation strategy.

Sys #	System	Category	KA Statement
BW/E04	Inadequate Heat Transfer - Loss Of Secondary Heat Sink	Knowledge of the reasons for the following responses as they apply to the (Inadequate Heat Transfer)	Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.

K/A# EK3.3 **K/A Importance** 4.2 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000 TBD, pg 92; DB-OP-02000, pg 72

Question Source: Bank 38987 **Level Of Difficulty: (1-5)** 3

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

Objective:

19. The following initial plant conditions exist;
    - ICS Full Auto
    - Plant Startup in progress
    - The plant is stable at 400 MWE

    The following conditions are observed:
    - ICS Neutron error is +0.25%.
    - The OUT command light is LIT on the Rod Control panel.
    - Reactor power, MWe and Feedwater flow are all rising together.

    Based on these conditions, which of the following failures has occurred?

A. The Reactor Demand station has failed high resulting in a continuous rod withdrawal.

B. A Steam Generator/Reactor Demand station failure has resulted in excessive Feedwater flow and Reactor Demand signals.

C. A Rod Control Panel failure has resulted in a continuous rod withdrawal.

D. A ULD failure has resulted in excessive feedwater flow and Reactor Demand signals.

Answer: C

Explanation/Justification:

A. Incorrect – Plausible because a high failure of the reactor demand station would cause a continuous rod withdraw, however neutron error would be a large negative (rods out command)

B. Incorrect – Plausible because a high failure of the SG/RX Demand station would cause a continuous rod withdraw and FW flow to increase.

C. Correct - The Out command is being generated without appropriate neutron error indication. The ULD, SG/Rx Demand and Rx Demand stations are upstream of the Rod Control Panel and would have to generate a 1% neutron error signal to withdraw rods.

D. Incorrect – Plausible because a high failure of a ULD failure could cause a continuous rod withdraw and FW flow to increase, but neutron error would be a large negative (rods out command)

Sys #	**System**	**Category**	**KA Statement**
001	Continuous Rod Withdrawal	Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal:	Interaction of ICS control stations as well as purpose, function, and modes of operation of ICS

K/A# AK1.14 **K/A Importance** 3.4* **Exam Level** RO

References provided to Candidate None **Technical References:** ICS Analog to Digital Drawings

Question Source: Bank 46162 **Level Of Difficulty: (1-5)** 2.5

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

Objective:

20. The following plant conditions exist:
    - The Reactor Operator is performing the CRD Exercise test for Safety Rod Group 4.
    - All the rods in the group indicate they have moved upon command **EXCEPT** rod 4-1.
    - 100% light for rod 4-1 remains ON, the rest of the Group 4 - 100% lights are OFF.
    - The I&C Technician reports that the CRDM phase lights in the rod control cabinet show that the CRDM phase rotation is operating properly.
    - Relative position indication shows rod 4-1 to be the same as the remainder of the group 4 rods.

    Based on these indications, what is the status of the rod 4-1?

    Rod 4-1 _________________.

A. has dropped

B. is misaligned

C. has a malfunctioning RPI

D. is stuck

Answer: **D**

Explanation/Justification:

A. Incorrect. Plausible if candidate since Relative CRD Position Indication stays aligned with the remainder of the rods in the group even when a control rod is dropped.
B. Incorrect. Plausible if candidate does not know CRD exercise test only inserts Control Rods approximately 3%. A misaligned control rod is one that is greater than 6.5% from group average.
C. Incorrect. Plausible if candidate does not know the 100% lights use magnetic reed switches to determine rod position compare to counting the counting of revolutions used by the Relative Position Indication System.
D. Correct. IAW DB-OP-02516 – Section 2, Symptoms

Sys #	**System**	**Category**	**KA Statement**
005 Rod	Inoperable/Stuck Control Rod	Knowledge of the interrelations between the Inoperable / Stuck Control Rod and the following:	Controllers and positioners

K/A# AK2.01 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02516 – Section 2, Symptoms

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

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

Objective:

21. The following plant conditions exist:

- Emergency Boration is in progress using the Boric Acid Addition Tank Method.
- Boric Acid Pump 2 is out of service.
- All other equipment is operable and in normal alignment.
- Emergency Boration flowrate is 25 gpm.
- Boric Acid Addition Tank T7-1 level is 91 inches
- Boric Acid Addition Tank T7-2 level is 65 inches
- Boric Acid Mixing Tank T6 level is 45 inches
- Acid Storage Tank T59 level is 90 inches

At 25 gpm, how long will it take to reach 20 inches in the tank supplying Boric Acid?

(Reference the attached Tank Curves)

A. ~18 minutes

B. ~160 minutes

C. ~240 minutes

D. ~560 minutes

<u>Answer</u>: C

Explanation/Justification:

A. Incorrect. Plausible since this would be the time associated with the levels in T6. T6 is only used to batch boric acid to either TK7-1 or 2 and is NOT a flowpath for emergency boration. The time does align for the volumes given in the stem.

B. Incorrect. Plausible for using TK7-2, however with Boric acid pump 2 OOS and all other systems aligned normally, this tank would not be aligned to boric acid pump 1. The time does align for the volumes given in the stem.

C. Correct. IAW DB-PF-06705 Figure CC 15.6. Student must realize that the Boric acid addition tank method will utilize TK7-1 since Boric acid pump 2 is OOS. TK7-1 is at 91 in. which is ~7000 gal. 20 in. is ~1000 gal therefore 6000 gal/25 gal/min = 240 min.

D. Incorrect. Plausible since T-59 is a make-up source of boric acid to TK7-1 but it is not used for emergency boration. The time does align for the volumes given in the stem.

Sys #	**System**	**Category**	**KA Statement**
024	Emergency Boration	Generic	Ability to interpret reference materials, such as graphs, curves, tables, etc.

K/A# 2.1.25 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate DB-PF-06705 Figures CC 15.1, 15.6, & 15.8

Technical References: DB-PF-06705 Figure CC 15.6,

Question Source: New

Level Of Difficulty: (1-5) 2

Question Cognitive Level: High – Application/Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.12)

Objective:

22. The plant is at 100% power, all systems are in a normal lineup.

- An I & C Technician inadvertently OPENS the equalizing valve for the selected PZR level instrument.

Which of the following describes the impact, of this action, on PZR level and the makeup system?
(Assume No Operator Actions)

Selected PZR level will___________(1)______________.
Makeup flow will___________(2)______________.

A. (1) rise to offscale high
(2) lower

B. (1) rise to offscale high
(2) remain the same

C. (1) lower to offscale low
(2) rise

D. (1) lower to offscale low
(2) remain the same

Answer: **A**

Explanation/Justification:

A. Correct. Opening the equalizing valve will cause a zero D/P and an offscale high indication. This will cause makeup flow to lower.

B. Incorrect – Plausible since opening the equalizing valve will cause a zero D/P and an offscale high indication. Since the affected instrument is selected for pressurizer level control, makeup flow will not remain the same, it will lower.

C. Incorrect – Plausible since opening the equalizing valve will cause a zero D/P and candidate may assume zero D/P equates to zero level. If the candidate concludes zero level, makeup flow would rise.

D. Incorrect - Plausible since opening the equalizing valve will cause a zero D/P and candidate may assume zero D/P equates to zero level. Since the affected instrument is selected for pressurizer level control, makeup flow will not remain the same.

Sys #	System	Category	KA Statement
028	Pressurizer (PZR) Level Control Malfunction	Ability to operate and / or monitor the following as they apply to the Pressurizer Level Control Malfunctions:	Charging pumps maintenance of PZR level (including manual backup)

K/A# AA1.07 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02513, Step 2.6.6. DB-PF-06703 page 52 curve CC4.1, OS-002 sh 3

Question Source: New **Level Of Difficulty: (1-5)** 4

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

Objective:

23. A Reactor startup is in progress with the following power level:

- Both Source Range NIs are 5E4 cps

What would be the effect of the intermediate range NI3 detector being UNDER compensated at this power level?

NI3 indication would_______________________________________.

A. be higher than actual power

B. be lower than actual power

C. not be affected at this power level

D. would stay at a minimum value

Answer: **A**

Explanation/Justification:

A. Correct. Under compensation of the Intermediate Range Nuclear Instrument would cause some gamma events to be detected by the Nuclear instrument in addition to neutron events. As a result, the output of the instrument would be higher than expected for the given source range level.

B. Incorrect. Plausible if the candidate incorrectly determined the compensation adds to the existing Intermediate Range output level.

C. Incorrect. Plausible if the candidate incorrectly determined the compensation does not affect the intermediate range indication at low powers.

D. Incorrect. Plausible if candidate does not know overlap regions of Source and Intermediate Range Nuclear Instruments.

Sys #	System	Category	KA Statement
033	Loss of Intermediate Range Nuclear Instrumentation	Knowledge of the operational implications of the following concepts as they apply to Loss of Intermediate Range Nuclear Instrumentation:	Effects of voltage changes on performance

K/A# AK1.01 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** Operations Lesson Plan OPS-SYS-I502

Question Source: New **Level Of Difficulty: (1-5)** 4

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

Objective:

24. A release of the Miscellaneous Waste Monitor Tank is to commence shortly.

- The release permit has been approved and given to the assigned operator.
- The Zone 3 Equipment Operator reports he is prepared to release the Miscellaneous Waste Monitor Tank to the collection box.
- The release rate specified on the release permit is 20 gpm.

Which **ONE** of the following actions is required?

A. Perform a single valve lineup, **THEN** perform the release using the 1.5” discharge line.

B. Perform a single valve lineup, **THEN** perform the release using the 3” discharge line.

C. Perform two (2) Independent valve lineups, **THEN** perform the release using the 1.5” discharge line.

D. Stop the release procedure and contact either the Shift Manager OR Command SRO.

<u>Answer</u>: D

Explanation/Justification:

A. Incorrect. Plausible – As noted in DB-OP-03011, Radioactive Batch Liquid Release, in the event that both REs, 1878A and 1878B, are non functional ODCM Table 2-1 requires that two independent verifications of the discharge valve lineup are performed. Since no Radiation Elements are non functional, a single valve lineup is plausible.

B. Incorrect. Plausible – As noted in DB-OP-03011, Radioactive Batch Liquid Release, in the event that both REs, 1878A and 1878B, are non functional ODCM Table 2-1 requires that two independent verifications of the discharge valve lineup are performed. Since no Radiation Elements are non functional, a single valve lineup is plausible.

C. Incorrect. Plausible – As noted in DB-OP-03011, Radioactive Batch Liquid Release, in the event that both REs, 1878A and 1878B, are non functional ODCM Table 2-1 requires that two independent verifications of the discharge valve lineup are performed.

D. Correct. As provided in DB-OP-03011, Radioactive Batch Liquid Release, caution 4.3.15-16, release at a rate of less than 5 gpm or between 15 to 25 gpm shall not be performed due to instrument inaccuracies.

Sys #	System	Category	KA Statement
059	Accidental Liquid Radwaste Release	Ability to determine and interpret the following as they apply to the Accidental Liquid Radwaste Release:	That the flow rate of the liquid being released is less than or equal to that specified on the release permit

K/A# AA2.06 **K/A Importance** 3.5* **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-03011 Caution 4.3.15 and 16

Question Source: Bank 30642 **Level Of Difficulty: (1-5)** 2

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

Objective:

25. The plant is operating at 100% power.

DR 2012B, CTMT Sump Discharge Outside CTMT Isolation valve, has been **CLOSED** to replace the handwheel.

- **NO** motor clutching is required for this evolution
- All Surveillance Requirements for DR 2012B are current
- DR 2012B breaker is OPEN
- DR 2012A, CTMT Sump Discharge inside CTMT Isolation valve is OPEN
- The administrative controls of DB-OP-01001, Administrative Control Of Containment Isolation Valves have been established

What is the status of DR 2012B?

DR 2012B is _________________ as long as ____________________.

A. Operable
its breaker is open **AND** secured in position and administrative controls remain in place

B. Operable
DR 2012A is capable of being closed by SFAS

C. Inoperable
its breaker is open **AND** secured in position and administrative controls remain in place

D. Inoperable
the handwheel is removed

Answer: A

Explanation/Justification:

A. Correct . Valves are OPERABLE when secured in position under administrative control (pg 4) As described in the LCO section of the Bases, the containment isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA. When an automatic containment isolation valve is secured in the closed position, it is performing the required safety function and should be considered OPERABLE. This same logic applies to a power operated remote manual containment isolation valve. Therefore, an automatic or non-automatic power operated containment isolation valve that is closed and deactivated and previously passed all SRs and is NOT known to be degraded is still OPERABLE, provided it is tested within the required frequency. The de-activation should be accomplished using normal design configuration (e.g. open breaker, de-energized solenoid, etc).pg 7. RO knowledge in that ROs need to know the definition of what constitutes an OPERABLE penetration

B. Incorrect. Plausible because the first part is correct. DR 2012 A or B going closed would accomplish the Containment isolation intent, the operability of DR 2012A does not affect the operability of DR 2012B

C. Incorrect. Plausible because the second part is correct

D. Incorrect. Plausible because the second part is correct

Sys #	System	Category	KA Statement
069	Loss of Containment Integrity	Ability to operate and / or monitor the following as they apply to the Loss of Containment Integrity:	Fluid systems penetrating containment

K/A# AA1.03 **K/A Importance** 2.8 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-01001. pg 4 & 7

Question Source: New **Level Of Difficulty: (1-5)** 4

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

Objective:

26. The following plant conditions exist

- The plant is at 100% power
- All systems are in a normal lineup
- MU4, Letdown Block Orifice Isolation, is open

A Loss of NNI-Y AC Power occurs

- **NO** Operator actions have been completed

How will Makeup flow and letdown pressure respond to these conditions?

Makeup flow will ____(1)_____ and Letdown pressure will ____(2)____.

A. (1) rise
(2) lower

B. (1) rise
(2) rise

C. (1) lower
(2) rise

D. (1) lower
(2) lower

Answer: B

Explanation/Justification:

A. Incorrect. Plausible if the candidate assumes the Pressurizer Level Control Valve MU32 fails open and MU 6 fails open or to 50% on a loss of NNI Y AC power. MU32 does fail open on a loss of instrument air. MU6 does fail to 50% open.

B. Correct. MU6 will fail to 50% demand causing excessive flow and pressure in Letdown causing the Letdown relief to lift resulting in a loss of inventory. Makeup flow will rise to maintain Pressurizer level. NOT DJ with Q12 since this Q is loss of power, and MU 6 fails to 50% and then must analyze the impact. Q12 has MU 6 failing closed on loss of air.

C. Incorrect. Plausible if the candidate assumes the Pressurizer Level Control Valve MU32 fails closed on a loss of NNI Y AC power. MU32 Hand Auto Station Lights fail off on a loss of NNI X DC power. Letdown pressure does rise due to MU6 failing to 50%.

D. Incorrect. Plausible if the candidate assumes the Pressurizer Level Control Valve MU32 fails closed on a loss of NNI Y AC power. MU32 Hand Auto Station Lights fail off on a loss of NNI X DC power.

Sys #	System	Category	KA Statement
BW/A03	Loss of NNI-Y	Knowledge of the reasons for the following responses as they apply to the (Loss of NNI-Y)	RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facilities license and amendments are not violated.

K/A# AK3.4 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02532, Loss of NNI/ICS Power Page 30, Loss of NNI-Y AC

Question Source: Bank 86431 **Level Of Difficulty: (1-5)** 3

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

Objective:

27. The following plant conditions exist:

- The plant was at 100% power
- A loss of offsite power has occurred
- The power supply breaker to AF6451, SG 2 AFW Level Control Valve fails OPEN

Which of the following describes the AFW response and the required operator actions to the above events?

A. AFW is lost to SG 2
Align AFP 1 to feed SG 2

B. SG 2 will overfill
Raise the steaming rate on SG 2 to lower level

C. AFW is lost to SG 2
Start the MDFP to feed SG 2

D. SG 2 will overfill
Reduce AFPT 2 speed to lower flow to SG 2

Answer: **D**

Explanation/Justification:

A. Incorrect. Plausible if candidate does not know that AF6451 fails open on loss of power. Aligning AFP 1 to feed SG2 would be the correct action on loss of AFW Train 2 to SG 2

B. Incorrect. Plausible because SG 2 would overfill, raising steaming rate would not prevent the overfill

C. Incorrect. Plausible if candidate does not know that AF6451 fails open on loss of power. Aligning MDFP to feed SG2 would be the correct action on loss of AFW Train 2 to SG 2

D. Correct. AF6451 is a DC powered solenoid valve that fails open on loss of power. This will cause an overfill of SG 2. DB-OP-02000 Attachment 20. page 3 of 3 identifies in the RNO column that if AF6451 can not be controlled, then the next option is to control AFPT speed

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Sys #	System	Category	KA Statement
BW/E14	EOP Enclosures	Knowledge of the interrelations between the (EOP enclosures) and the following	Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

K/A# EK2.2 **K/A Importance** 3.8 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02538, pg 15 & 36
DB-OP-02000 Attachment 20, pg 363

Question Source: Bank 86551 **Level Of Difficulty: (1-5)** 3

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

Objective:

28. The following plant conditions exist:

- 2 Reactor Coolant Pumps (RCPs) are running in Loop "1"
- 1 RCP is running in Loop "2"
- Reactor power is 50% by nuclear instrumentation indication

One of the running RCPs develops high vibration and must be stopped immediately.

- When the RCP is stopped, the plant trips.
- All systems function as designed.

Which Reactor Coolant Pump was stopped and why did the plant trip?

(1) An RCP running in Loop _______ was stopped.

(2) The plant trip was due to__________________.

A. "2"
flux/Δ flux/flow in the RPS

B. "2"
the Power/pump monitors in the RPS

C. "1"
the Power/pump monitors in the RPS

D. "1"
flux/Δ flux/flow in the RPS

Answer: B

Explanation/Justification:

A. Incorrect. Plausible if the candidate incorrectly determines that the RPS flux/ Δ flux/flow trip will actuate when flow is lost in a loop as opposed to total flow.

B. Correct. Power to Pumps monitors will cause a reactor trip if at least one reactor coolant pump in each loop does not have power. Power operations with zero flow in either is not permitted. Turning off a Loop 2 RCP will result in a zero flow in that loop.

C. Incorrect. Plausible since balance flow in RCS is desired to keep both Steam Generators available for heat transfer.

D. Incorrect. Plausible if the candidate incorrectly determines that the RPS flux/ Δ flux/flow trip will actuate when flow is lost in a bop as opposed to total flow.

Sys #	System	Category	KA Statement
003	Reactor Coolant Pump System (RCPS)	Ability to monitor automatic operation of the RCPS, including:	RCS flow

K/A# A3.04 **K/A Importance** 3.6 **Exam Level** RO

References provided to Candidate None **Technical References:** DBBP-TRAN-0034, Davis-Besse Operator Fundamental Memory List Page 6

Question Source: Bank 31657 **Level Of Difficulty: (1-5)** 2

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

Objective:

29. The following plant conditions exist:

- Reactor Power = 70%
- RCS Loop 1 flow = 74 mpph
- RCS Loop 2 flow = 75 mpph

The following event occurs:

- RCP 2-2 trips

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

(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:

A. Incorrect. Plausible since Loop 2 Tave is the normal controlling Tave Loop. Since a Loop 2 RCP trips, Loop 1 will have the highest flow FW flow and therefore SG level will be higher in SG 1 which is correct.

B. Correct. The Smart Analog Selector Switch (SASS) for Tave automatically selects the Loop with the Highest RCS Flow when a RCP is stopped. Since a Loop 2 RCS trips, Loop 1 will have the highest flow and Loop 1 Tave will be selected. ICS will ratio FW flow to the Steam Generators based on RCS flow or about 2.5 to 1 with the 2 RCP loop SG receiving the higher Feedwater Flow and will operate at a higher Steam Generator Level.

C. Incorrect. Plausible Since a Loop 2 RCP trips, Loop 1 will have the highest flow FW flow and therefore SG level will be higher in SG 1.

D. Incorrect. Plausible since Loop 2 Tave is the normal controlling Tave Loop.

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 pg 51

Question Source: Bank 79047 **Level Of Difficulty: (1-5)** 3

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

Objective:

30. Which of the following conditions are required to ENABLE the ICS Feed and Bleed Permissive light? (The Amber light below the batch controller)

A. Control Rod Groups 1, 2, and 3 must be 100% withdrawn and Control Rod Groups 4 and 5 must be at 25% withdrawn.

B. Control Rod Groups 1, 2, 3, and 4 must be 100% withdrawn.

C. Control Rod Groups 1, 2, 3, and 4 must be 100% withdrawn and Control Rod Group 5 must be greater than 25% withdrawn.

D. Control Rod Group 1 is 100% withdrawn.

<u>Answer</u>: C

Explanation/Justification:

A. Incorrect. Plausible because Groups 1, 2, 3, 4, and 5 must be withdrawn, but Group 4 is at the wrong percentage withdrawn
B. Incorrect. Plausible if candidate thinks that only the 4 Safety Groups have to be withdrawn for the interlock to work
C. Correct – Feed and Bleed permissive allows for a change in boron concentration by simultaneously adding inventory to the RCS will removing inventory via letdown. While this method is efficient, the feed and bleed permissive prevents using this method when the reactor is shutdown.
D. Incorrect. Plausible because Group 1 is required to be withdrawn

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Sys #	System	Category	KA Statement
004	Chemical and Volume Control System	Knowledge of the effect of a loss or malfunction on the following CVCS components:	Purpose and function of the boration/dilution batch controller

K/A# K6.13 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-06001, pg 26

Question Source: Bank 31846 **Level Of Difficulty: (1-5)** 3

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

Objective:

31. The following plant conditions exist:

- A large break Loss Of Coolant Accident (LOCA) has occurred from full power.
- Low Pressure Injection Pump 1 failed to start.

Which of the following describes why the LPI pump discharge lines are cross-connected

A. Ensures both High Pressure Injection trains are piggybacked.

B. Ensures thermal stresses are symmetric around the reactor vessel.

C. Ensures Low Pressure Injection flow into the core if the break is on the piping for Low Pressure Injection

D. Ensures sufficient flow through the running Low Pressure Injection pump to prevent dead-heading.

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because the statement is true, but is not the bases for x-connecting

B. Incorrect. Plausible because it would ensure the same temperature of water is going into both sides of the reactor vessel, but is not the bases for x-connecting

C. 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.

D. Incorrect. Plausible because this statement is true, but the LPI pumps have a minimum recirc line to prevent deadheading

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Sys #	System	Category	KA Statement
005	Residual Heat Removal System (RHRS)	Knowledge of RHRS design feature(s) and/or interlock(s) which provide for the following:	Modes of operation

K/A# K4.02 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000 TBD pg 271

Question Source: Bank 37346 **Level Of Difficulty: (1-5)** 2

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

Objective:

32. The following plant conditions exist:

- A large break LOCA initiated an SFAS Level 4 actuation.
- The operating crew is preparing to implement DB-OP-02000, Attachment 7, Transferring LPI Suction to the Emergency Sump.

Which of the following describes operations relative to Containment Spray (CS)?

Valve identification:

CS 1530 - CTMT SPRAY PUMP 1 AUTO CONTROL VALVE
CS 1531 - CTMT SPRAY PUMP 2 AUTO CONTROL VALVE
DH 7A - BWST OUTLET ISOLATION VALVE LINE 2
DH 7B - BWST OUTLET ISOLATION VALVE LINE 1
DH 9A - DH PUMP 2 SUCTION FROM EMERGENCY SUMP
DH 9B - DH PUMP 1 SUCTION FROM EMERGENCY SUMP

A. Stop both CS pumps. Open DH 9A and DH 9B. Close DH 7A and DH 7B. Restart both CS pumps and verify CS 1530 and CS 1531 are fully open

B. Stop both CS pumps. Open DH 9A and DH 9B. Verify DH 7A and DH 7B stroke closed and CS 1530 and CS 1531 go to the throttled position. Restart both CS pumps

C. Open DH 9A and DH 9B. Close DH 7A and DH 7B. Verify CS 1530 and CS 1531 are fully open

D. Open DH 9A and DH 9B. Verify DH 7A and DH 7B stroke closed, and CS 1530 and CS 1531 go to the throttled position.

Answer: **D**

Explanation/Justification:

A. Incorrect. Plausible if the candidate considers the interlock between DH7A/B and DH9A/B that prevents having both valves open at the same time. This interlock is overridden by SFAS Level 5 when BWST level reaches 9 feet.

B. Incorrect. Plausible if candidate does not understand that the transfer to the emergency sump is designed to accomplished with while the affected pumps remain in service.

C. Incorrect. Plausible if the candidate considers the interlock between DH7A/B and DH9A/B that prevents having both valves open at the same time. This interlock is overridden by SFAS Level 5 when BWST level reaches 9 feet.

D. Correct. When DH9A/B leave their CLOSE seats, CS 1530/1530 receives signals to go to a throttle position to ensure CS Pumps do not cavitate with the hotter Emergency Sump Water

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Sys #	System	Category	KA Statement
006	Emergency Core Cooling System (ECCS)	Knowledge of the physical connections and/or cause effect relationships between the ECCS and the following systems:	CSS

K/A# K1.13 **K/A Importance** 3.3* **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000 pg 301 & 302 Attachment 7, Transferring LPI Suction to the Emergency Sump

Question Source: Bank 75956 **Level Of Difficulty: (1-5)** 2

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

Objective:

33. The following plant conditions exist:
    - A plant startup is in progress
    - Reactor Coolant System pressure is 40 psig
    - Pressurizer temperature is 287 °F
    - A Pressurizer bubble is in the process of being drawn
    - Quench Tank pressure is 25 psig

    The following event occurs:
    - Selected RCS pressure input*, in the RPS cabinet,* [1] fails high

    How will Quench Tank pressure and temperature respond to these conditions, *within the first 5 minutes*[2]?

    Quench Tank pressure will ____(1)______ and temperature will _____(2)______ .

A. (1) rise
(2) rise

B. (1) rise
(2) remain the same

C. (1) remain the same
(2) remain the same

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

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because the Pressurizer pressure and temperature are higher than the QT

B. Incorrect. Plausible because the Pressurizer pressure is higher than the QT and at 40 psig N2 could be a source of the PZR bubble in which case temperature would not change.

C. Correct. The PORV will remain close when RCS pressure is less than 150 psig due to a spring in the valve. Therefore no nitrogen will be vented from the Pressurizer to the Quench Tank (QT). This response is based on the PORV that has been installed in the plant prior to outage 17M.

D. Incorrect. Plausible because the Pressurizer temperature is hotter than the QT. Pressure would remain the same if the student believes the QT quench action would automatically maintain pressure.

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

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

References provided to Candidate None **Technical References:** SD-039A, pg 70 (2-47)

Question Source: New **Level Of Difficulty: (1-5)** 4

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

Objective:

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[1] *Clarification added during administration of examination.*

[2] *Clarification added during administration of examination.*

34. The plant is operating at 100% power with the following equipment configuration:

- Component Cooling Water Pump 1 in service
- Component Cooling Water Pump 2 aligned for Standby Operation
- Service Water Pump 1 and Service Water Pump 2 are in service. Service Water Pump 3 is available.
- Control Rod Drive Booster Pump 1 is in service and
- Control Rod Drive Booster Pump 2 is aligned for Standby Operation.
- Both Emergency Diesel Generators are not operating but aligned for Standby Operation.
- Spent Fuel Pool Cooling Water Pump 1 is in service with Spent Fuel Pool Pump 2 in Standby.

Which of the following evolutions would require starting the Standby CCW pump during the performance of the evolution?

A. Placing Service Water Pump 3 in service on Service Water Train 2 per DB-OP-06261, Service Water System Operating Procedure

B. Placing Control Rod Drive Booster Pump 2 in service per DB-OP-06262, Component Cooling Water System Operating Procedure.

C. Starting Emergency Diesel Generator 2 to idle condition per DB-OP-06316, Emergency Diesel Generator Operating Procedure.

D. Starting Spent Fuel Pool Cooling Water Pump 2 to place both Spent Fuel Pool Cooling Trains in service per DB-OP-06021, Spent Fuel Pool Operating Procedure.

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because Component Cooling Water is cooled by Service Water. Transferring from SW Pump 2 to Pump 3 could require additional cooling

B. Incorrect. Plausible since many CCW components are train dependant and required the CCW pump to be in service on that train. In this case, the CRD Booster Pumps are supplied from the common Containment Header which is supplied by either CCW Pump. No CCW Pump Start Required

C. Correct. Of the four evolutions listed, only this evolution requires additional CCW flow or a different CCW train to supply the load to be started.

D. Incorrect. Plausible since many CCW components are train dependant and required the CCW pump to be in service on that train. In this case, the SFP Cooling Pumps supplied from the common Non-Essential Header which is supplied by either CCW Pump. No CCW Pump Start Required

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Sys #	System	Category	KA Statement
008	Component Cooling Water System (CCWS)	Ability to manually operate and/or monitor in the control room:	Conditions that require the operation of two CCW coolers

K/A# A4.10 **K/A Importance** 3.1* **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-06316 pg 79

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

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

Objective:

35. The plant is operating at 100% power with all systems aligned for normal operation.
    - CCW Pump 1 is running
    - CCW pump 3 is being aligned to CCW Loop 1
    - CCW 3 valves have been opened to CCW Loop 1
    - ACD2, CCW 3 tie to C1 bus, is racked in with closed power fuses installed

    The following events occurs:
    - An SFAS Level 2 occurs on Low RCS pressure
    - Loss of all offsite power (LOOP).
    - All systems function as designed.

    What will be the status of CCW Pumps 1 and 3?

A. Both CCW Pump 1 and CCW Pump 3 will start

B. Only CCW Pump 1 will start

C. Only CCW Pump 3 will start

D. Neither CCW pump will start

Answer: B

Explanation/Justification: CCW Pump 3

A. Incorrect. Only CCW pump 1 will start

B. Correct. CCW pumps 1 and 3 both receive backup power from the EDGs, however there is an interlock that prevents CCW pump 3 from receiving the LOOP start signal when CCW pump 1 is racked onto the bus.

C. Incorrect. Only CCW pump 1 will start

D. Incorrect. CCW pump 1 will start

Sys #	System	Category	KA Statement
008	Component Cooling Water System (CCWS)	Knowledge of bus power supplies to the following:	CCW pump, including emergency backup

K/A# K2.02 **K/A Importance** 3.0* **Exam Level** RO

References provided to Candidate None **Technical References:** SD 3A, 4160v, pg2-6 (30)

Question Source: New **Level Of Difficulty: (1-5)** 4

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** (CFR: 41.7)

Objective:

36. The plant is operating at 100% power with all systems aligned for normal operation. The following events then occur:

- Pressurizer Heaters energize
- Annunciator (4-4-C) HOT LEG PRESS LO alarms
- Tave is 582 °F and stable
- PZR Spray valve RC 2 OPEN indicating light is LIT
- PZR Spray valve RC 2 40% OPEN indicating light is NOT LIT
- PZR Spray valve RC 2 CLOSED indicating light is NOT LIT
- RCS pressure is 2150 psig and slowly lowering

Which of the below listed procedures is REQUIRED to be entered to address these conditions?

A. DB-OP-02000, RPS, SFAS, SFRCS Trip, OR SG Tube Rupture

B. DB-OP-02504, Rapid Shutdown

C. DB-OP-02513, Pressurizer System Abnormal Operation

D. DB-OP-02526, Primary to Secondary Heat Transfer Upset

Answer: C

Explanation/Justification:

A. Incorrect. Plausible since RCS pressure is low, however pressure is still above the RX trip setpoint.

B. Incorrect. DB-OP-02504, Rapid Shutdown is used to reduce power for any condition that warrants the need to do so. The conditions listed in the stem do not warrant a rapid shutdown and doing so will not alleviate the situation.

C. Correct. RO ONLY since it involves Abnormal procedure entry. RCS Pressure low with indications of the PZR spray valve open are indications of an inadvertent opening of the spray valve. The annunciator 4-4-C and PZR heaters on are entry conditions for DB-OP-02513.

D. Incorrect. DB-OP-02526 would only be warranted for these conditions IF Tavg was not stable. Since Tavg is stable, the entry conditions are not met and performing this procdure will not alleviate the situation.

Sys #	System	Category	KA Statement
010	Pressurizer Pressure Control System (PZR PCS)	Generic	Knowledge of abnormal condition procedures.

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

References provided to Candidate None **Technical References:** DB-OP-02513 symptoms

Question Source: New **Level Of Difficulty: (1-5)** 2

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

Objective:

37. The following plant condition exist:
    - The plant is in Mode 3 with RCS Cooldown in progress in accordance with DB-OP-06903, Plant Cooldown.
    - RCS Temperature is 275 °F.
    - RCS Pressure is 250 psig.

    The following event occurs:
    - The Selected Reactor Coolant System pressure channel*, in the RPS cabinet,*[1] fails HIGH.

    Assuming **NO** operator action, which of the following is a potential consequence of this failure?

A. DH11, DH Dropline Isolation Valve will automatically close.

B. CF1A, Core Flood Isolation Valve will automatically open.

C. SFAS Low-Low RCS Pressure will actuate.

D. SFAS High Containment Pressure will actuate.

Answer: D

Explanation/Justification:

A. Incorrect. The answer is plausible because DH11 has an automatic close feature when RCS pressure is above 301 psig, but this feature does not use the selected RPS Pressure output to determine RCS pressure. Actual RCS Pressure will lower due to opening the PORV.

B. Incorrect. The answer is plausible because CF1A has an automatic open feature at 770 psig, but this feature does not use the selected RPS Pressure output to determine RCS pressure. Actual RCS Pressure will lower due to opening the PORV.

C. Incorrect. The answer is plausible because failing the Selected RCS Pressure channel high will cause the PORV to open reducing RCS pressure, however the SFAS Low-Low RCS Pressure trip will be blocked in accordance with DB-OP-06903 Plant Cooldown for this plant condition.

D. Correct. The selected RCS pressure channel (either RPS Channel 1 or RPS Channel 2) failing high will cause the PORV to open. This will initiate flow from the pressurizer to the quench tank. The quench tank rupture disc will fail causing CTMT Pressure to rise. At 18.7 psia, SFAS High Containment Pressure will actuate.

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

K/A# K3.03 **K/A Importance** 4.0 **Exam Level** RO

References provided to Candidate None **Technical References:** System Description SD-002 pg 97 (A-1)

Question Source: Bank 29089 **Level Of Difficulty: (1-5)** 2

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

Objective:

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[1] *Clarification added during administration of examination.*

38. The plant is operating at 100% power with all systems aligned for normal operation.
    - DB-OP-02508, Control Room Evacuation has been implemented.
    - The Immediate Actions were **NOT** completed prior to Control Room evacuation

IAW DB-OP-02508, Control Room Evacuation ATTACHMENT 4: Secondary Side Reactor Operator Actions Outside The Control Room you are directed to:

- Trip MAIN FEED PUMP # 1 and #2 using the PULL TO TRIP HANDLE located on #1 & #2 MFPTs.

What will be the consequences of this action?

Tripping the MFPs in this manner will cause a/an_______(1)________, which will initiate a/an________(2)______, eventually a/an________(3)_________ will occur.

A. (1) ARTS actuation
(2) SFRCS actuation
(3) Reactor Trip signal

B. (1) ARTS actuation
(2) Reactor Trip signal
(3) SFRCS actuation

C. (1) Reactor Trip signal
(2) SFRCS actuation
(3) ARTS actuation

D. (1) Reactor Trip signal
(2) ARTS actuation
(3) SFRCS actuation

Answer: B

Explanation/Justification:

A. Incorrect. Plausible if the candidate believes ARTS causes a SFRCS Actuation. SFRCS actuation causes an ARTS actuation, not the other way around.

B. Correct. ARTS will actuate on Low MFPT Hydraulic Oil Pressure when the MFP Turbines are tripped. ARTS will then trip the Reactor. Loss of MFP Discharge Pressure will then cause a SFRCS Actuation on Steam to Feed Differential Pressure.

C. Incorrect. Plausible since the candidates know the Reactor should be tripped if there is no Main Feedwater available. The Reactor Trip is accomplished via ARTS, not prior to the ARTS actuation.

D. Incorrect. Plausible since the candidates know the Reactor should be tripped if there is no Main Feedwater available. The Reactor Trip is accomplished via ARTS, not prior to the ARTS actuation. In addition, SFRCS actuation causes an ARTS actuation, not the other way around.

Sys #	System	Category	KA Statement
012	Reactor Protection System (RPS)	Generic	Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

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

References provided to Candidate None **Technical References:** Memory list Attachment 1 page 3,& 4

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

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

Objective:

39. The following plant conditions exist

- Reactor power is 100%
- RPS channel 1 in manual bypass

The following event occurs:

- RCS pressure exceeds the RPS high pressure trip setpoint
- RPS channels 2 and 4 trip
- RPS channel 3 fails to trip

How will the CRD breakers respond to these conditions?

A. No CRD breakers will open.

B. Only the "A" and "C" breakers will open.

C. Only the "B" and "D" breakers will open.

D. All CRD breakers will open

Answer: D

Explanation/Justification:

A. Incorrect. Plausible if the candidate believes the logic of the RPS is the same as the Steam Feed Rupture Control System where channels 1 and 3 are actuation channel 1 and Channels 2 and 4 are actuation channel 1. If only a single actuation channel trips, both you do not receive a full SFRCS Actuation.

B. Incorrect. Plausible if the candidate does not understand the relationship between RPS Channels and CRD Breakers. RPS Channels 1, 2, 3, 4 supplies CRD Breakers B, A, D, C respectively.

C. Incorrect. Plausible if the candidate does understand the relationship between RPS Channels and CRD Breakers. RPS Channels 1, 2, 3, 4 supplies CRD Breakers B, A, D, C respectively but does not understand the logic of RPS as noted in Distractor 1 above.

D. Correct. IAW DB-OP-06403, Att. 4, pg 57 Relays KB and KD remain energized, and their corresponding contacts in each RPS cabinet remain closed, however, KA and KC de-energize. Corresponding KA and KC contacts open in each cabinet, interrupting power to the associated CRD breakers and causing them to trip.

Sys #	**System**	**Category**	**KA Statement**
012	Reactor Protection System (RPS)	Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following:	Redundancy

K/A# K4.04 **K/A Importance** 3.1 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-06403, Att. 4, pg 57

Question Source: Bank 39026 **Level Of Difficulty: (1-5)** 3

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

Objective:

40. The following plant conditions exist:
- The plant is operating at 100% power.
- All systems are in normal alignment.

The following event occurs:
- A BWST level transmitter fails low in SFAS Channel 1.

What will be the indications on the SA Level 5 Output Module in each SFAS cabinet?

A. A single 1/5 light will be lit on Channel 1 only.

B. A single 1/5 light will be lit on Channels 1 and 3 only.

C. A single 1/5 light will be lit on Channels 1, 2, 3, and 4.

D. Two 1/5 lights will be lit on Channel 1, a single 1/5 light will be lit on Channels 2, 3, and 4.

Answer: C

Explanation/Justification:

A. Incorrect. Plausible if the candidate assumes train separation in SFAS prevents inputs from an SFAS channel being used or sensed by the remaining SFAS channels.
B. Incorrect. Plausible if the candidate believes the logic of the SFAS is similar to the Steam Feed Rupture Control System where channels 1 and 3 are actuation channel 1 and Channels 2 and 4 are actuation channel 1.
C. Correct. Each SFAS Channel receives and input condition for other 3 channels. 2 of 4 logic is then used to confirm a trip condition exists. A single trip will light a 1/5 light in all channels as each channel senses either its own or another channels trip condition.
D. Incorrect. Plausible if the candidate assumes that a 1/5 light is normally lit. Any trip signal would cause an additional 1/5 light to be illuminated.

Sys #	System	Category	KA Statement
013	Engineered Safety Features Actuation System (ESFAS)	Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS:	Sensors and detectors

K/A# K6.01 **K/A Importance** 2.7* **Exam Level** RO
References provided to Candidate None **Technical References:** SD 002, page 2-11 & 2-12 section 2.3.4
Question Source: Bank 37026 **Level Of Difficulty: (1-5)** 3
Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 to 45.8)
Objective:

41. The following plant conditions exist:
- The plant is operating at 100% power.
- All systems are in normal alignment.
- All containment air coolers trip.

As containment air temperature rises:

Which of the following indicates how Core Flood Tank (CFT) level indication will respond to this rising temperature?

Indicated CFT level will ___________________.

A. rise due to heating of the Core Flood Tank level indicator reference leg.

B. lower due to heating of the Core Flood Tank level indicator reference leg

C. remain unaffected due to equal heating of the variable and reference legs

D. remain unaffected due to temperature compensation in the reference leg

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Answer: A

Explanation/Justification:

A. Correct. Elevated containment temperatures will cause the reference leg of the core flood tank level indicator to heat up faster than the volume of water in the tank causing indicated level to be higher than actual level.

B. Incorrect. Plausible if the candidate does not consider the impact on the reference leg and only considers the impact on the Core Flood Tank. Heating of the Core Flood will expand the inventory which could lead one to believe this response is correct.

C. Incorrect. Plausible if the candidate does not consider impact of containment heating or determine CTMT heating affects reference leg and variable leg equally.

D. Incorrect. Plausible if candidate believes CFT level indication is temperature compensated in the same manner as the PZR level indication.

Sys #	System	Category	KA Statement
022	Containment Cooling System (CCS)	Knowledge of the effect that a loss or malfunction of the CCS will have on the following:	Containment instrumentation readings

K/A# K3.02 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000 Attachment 9. Impact on core flood tanks is similar. System Description SD-04, Core Flood Section 2.5.1.1 Core Flooding Tank Level (Instrument String CF3)

Question Source: New **Level Of Difficulty: (1-5)** 2

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

Objective:

42. The following plant conditions exist:
- The plant is operating at 100% power
- EDG 2 has been started IAW DB-OP-06316, Diesel Generator Operating procedure
- Section 4.3 Emergency Diesel Generator (EDG) 2 Operational Checks of the Hydraulic Governor High and Low Speed Stops is in progress
- EDG 2 is at the hydraulic governor high speed stop of 945 rpm

The following event occurs:
- LOCA occurs
- Reactor trips
- SFAS level 2 has actuated

How will EDG 2 respond to these events and what operator actions are required by DB-OP-06316, Diesel Generator Operating procedure?

The EDG speed will ___________(1)____________, and the Operator is required to _________(2)_________.

A. (1) automatically go to 900 rpm
(2) verify EDG Output Breaker is OPEN

B. (1) automatically go to 900 rpm
(2) place the Governor Mode Selector Switch to ELECT

C. (1) remain at 945
(2) lower the hydraulic governor until the EDG speed is 900 rpm and place the Governor Mode Selector Switch to ELECT

D. (1) remain at 945 rpm
(2) Shutdown EDG from Control Room, allow SFAS to restart EDG in isochronous mode

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Answer: C

Explanation/Justification:

A. Incorrect. Plausible because this is the normal speed for an SFAS level 2, however the hydraulic governor would keep speed at 945 rpm. EDG Output breaker would be open for this condition.

B. Incorrect. Plausible because this is the normal speed for an SFAS level 2, the required action is correct.

C. Correct. IAW DB-OP-06316, step 2.2.17, pg 11

D. Incorrect. Plausible because this would work to bring the EDG to 900 rpm but this is not procedurally driven. Also the EDG will not stop unless the "Emergency" shutdown is used.

Sys #	System	Category	KA Statement
064	Emergency Diesel Generator (ED/G) System	Generic	Ability to interpret and execute procedure steps.

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

References provided to Candidate None **Technical References:** DB-OP-06316, step 2.2.17, pg 11, 112, 113, 114

Question Source: New **Level Of Difficulty: (1-5)** 4

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.12)

Objective:

43. Given the following conditions:

- A Loss Of Coolant Accident has occurred.
- All equipment is operating as designed.
- Reactor Coolant System pressure is 100 psia and stable.
- Containment pressure is 16 psia and trending down.
- DB-OP-02000 Attachment 7, Section 1, Actions to close breakers for DH7A, DH7B, DH9A, DH9B, and HP31 are complete

Subsequently, a Safety Features Actuation System Level 5 actuation is received.

Which of the following actions, if any are required?

Valve identification:

DH7A, BWST outlet valve
DH7B, BWST outlet valve
DH9A, CTMT Emergency Sump Isolation valve
DH9B, CTMT Emergency Sump Isolation valve

A. No actions required, DH-9A and DH-9B valves will automatically realign to the CTMT sump.

B. Block the SFAS level 2 signal, THEN manually CLOSE DH-7A and DH-7B

C. Block the SFAS level 2 signal, THEN manually OPEN DH-9A and DH-9B

D. Manually OPEN DH-9A and DH-9B, THEN manually CLOSE DH-7A and DH-7B

Answer: C

Explanation/Justification:

A. Incorrect – Plausible if candidate assumes only restoring power to ECCS Sump Swap valves is required and SFAS Level 5 will automatically initiate transfer.

B. Incorrect – Plausible since DH7A/B is interlocked with DH9A/B such that without an SFAS Level 5 signal, DH7A/B must be closed prior to opening DH9A/B.

C. Correct – With an SFAS Level 5 actuation, the SFAS level 2 must this be blocked to allow manual transfer by opening the DH9A/B valves.

D. Incorrect – Plausible if the candidate assumes the SFAS level 5 completely disables the DH7A/B and DH 9A/B such that manual closure of DH7A/B is required.

Sys #	System	Category	KA Statement
006	Emergency Core Cooling System (ECCS)	Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following:	Parallel redundant systems

K/A# K4.26 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000, rev. 25 Att. 7 page 300

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

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

Objective:

44. The plant is operating at 100% power with all systems aligned for normal operation.

The following events then occur:

- A Large Break LOCA
- Total loss of offsite power occurs
- Containment pressure is 40 psia
- Emergency Diesel Generator 1 fails to start and cannot be manually started
- All other systems function as designed

Based on these conditions, what will be the status of the Containment Spray Pumps?

Containment Spray Pump 1 will be ______(1)_____ and Containment Spray Pump 2 will be ______(2)_____

A. (1) running
(2) running

B. (1) running
(2) stopped

C. (1) stopped
(2) stopped

D. (1) stopped
(2) running

Answer: D

Explanation/Justification:

A. Incorrect. P56-1 will have no power since EDG 1 did not start - Plausible if candidate does not recognize no power available for CS Pump 1

B. Incorrect. P56-2 will have a start signal from high containment pressure, and it will have power from the EDG 2. P56-1 will have no power since EDG 1 did not start – Plausible if candidate does not know power supplies for Containment Spray Pumps.

C. Incorrect. P56-2 will have a start signal from high containment pressure, and it will have power from the EDG 2. P56-1 will have no power since EDG 1did not start – Plausible if candidate does not know SFAS Level 4 setpoint for high containment pressure

D. Correct. P56-2 will have a start signal from high containment pressure, and it will have power from the EDG 2. P56-1 will have no power since EDG 1did not start.

Sys #	System	Category	KA Statement
026	Containment Spray System (CSS)	Knowledge of bus power supplies to the following:	Containment spray pumps

K/A# K2.01 **K/A Importance** 3.4* **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000 pg 413 Table 2 setpoint for high CTMT Pressure is 40 psia per DBBP-TRAN-0034 pg 9

Question Source: New **Level Of Difficulty: (1-5)** 2

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** (CFR: 41.7)

Objective:

45. The following plant conditions exist:

- The plant is operating at 90% power
- Unit Load Demand (ULD) is in MAN
- All systems are in a normal lineup

The following occurs:

- Tave begins to lower
- Containment pressure begins to rise at 0.1 psia/minute
- Containment temperature begins to rise at 0.5 °F/minute
- Containment radiation remains constant
- Generated MWs begin to lower
- Feedwater to SG 1 begins to rises
- No personnel are in Containment

(1) Reactor power will ______(1)________

(2) Which of the following actions are required?

A. (1) lower
(2) Immediately trip the Reactor
Manually actuate SFRCS

B. (1) rise
(2) Immediately trip the Reactor
Manually isolate all feedwater to SG 1

C. (1) lower
(2) Isolate letdown
Start the second Makeup pump

D. (1) rise
(2) Begin a plant shutdown
Monitor Containment conditions

Answer: D

Explanation/Justification:

A. Incorrect. Plausible because tripping the reactor and actuating SFRCS is an action when personnel safety is in jeopardy per immediate action 3.1

B. Incorrect. Plausible because tripping the reactor and actuating SFRCS is an action when personnel safety is in jeopardy per immediate action 3.1

C. Incorrect. Plausible because Reactor Coolant System Leaks will cause containment temperature and pressure to rise. Actions per DB-OP-02522, Small RCS Leaks would direct isolating Letdown and starting a second Makeup Pump based on impact to pressurizer level.

D. Correct. Rising Containment Pressure with lowering Tave is a symptom of a Steam Leak in Containment. DB-OP-02525, Steam Leaks section 4.1 directs a power reduction while monitoring containment conditions. RO only since it involves basic strategy for steam leak in containment.

Sys #	System	Category	KA Statement
039	Main and Reheat Steam System (MRSS)	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:	Increasing steam demand, its relationship to increases in reactor power

K/A# A2.05 **K/A Importance** 3.3 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02525, Section 2.1 pg 5 and 4.1 pg 12.

Question Source: Bank 36971 Modified, determine reactor power change added **Level Of Difficulty: (1-5)** 3

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

Objective:

46. The plant is operating at 100% power with all systems aligned for normal operation.

The following event occurs:

- The selected Feedwater temperature instrument to ICS slowly fails LOW
- SASS does NOT transfer to the good Feedwater temperature instrument

How will the plant respond to these conditions?

A. Tave will decrease

B. Reactor power will increase

C. Feedwater flow demand will decrease

D. OTSG operating range level indication will increase

Answer: C

Explanation/Justification:

A. Incorrect – plausible because Feedwater temperature is used to density compensate feedwater flow measurement. A low failure we causes the Feedwater flow signal to ICS to rise which the candidate may interpret as causing a rise in SG level which would reduce Tave.

B. Incorrect – plausible because Feedwater temperature is used to density compensate feedwater flow measurement. A low failure we causes the Feedwater flow signal to ICS to rise which the candidate may interpret as causing a rise in SG level which would increase reactor power.

C. Correct - Feedwater temperature is used to density compensate feedwater flow measurement. A low failure we causes the Feedwater flow signal to ICS to rise. ICS reduces FW demand to eliminate the flow error.

D. Incorrect – plausible because Feedwater temperature is used to density compensate feedwater flow measurement. A low failure we causes the Feedwater flow signal to ICS to rise which the candidate may interpret as causing a rise in SG level.

Sys #	System	Category	KA Statement
059	Main Feedwater (MFW) System	Ability to monitor automatic operation of the MFW, including:	ICS

K/A# A3.07 **K/A Importance** 3.4* **Exam Level** RO

References provided to Candidate None **Technical References:** Integrate Control System Analog and Digital Drawings.

Question Source: Bank 38874 **Level Of Difficulty: (1-5)** 3

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

Objective:

47. The following plant conditions exist:
   - 100% power
   - ICS Delta T-Cold is in MAN with the meter reading 54% demand

   The ICS Delta T-Cold Hand Auto Station is **THEN** placed in the "AUTO" position.

   How will this impact ICS control of Main Feedwater?

A. The average of feedwater loop 2 and feedwater loop 1 demand will be 54%

B. Feedwater loop 1 demand will be greater than feedwater loop 2 demand

C. Feedwater loop 1 demand will be “boosted” by a 4 °F Delta T-Cold error

D. Feedwater loop 2 demand will be greater than feedwater loop 1 demand

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Answer: **D**

Explanation/Justification:

A. Incorrect. Because the meter does not indicate average demand
B. Incorrect. Because it is the opposite response
C. Incorrect. Because it applies to looking at the MV reading
D. Correct. A reading >50% indicates that loop A demand is > loop B demand, therefore (d) is the correct response.

Sys #	System	Category	KA Statement
059	Main Feedwater (MFW) System	Ability to manually operate and monitor in the control room:	ICS

K/A# A4.10 **K/A Importance** 3.9* **Exam Level** RO
References provided to Candidate None **Technical References:** DB-OP-06401 pg 78 Attach 7
Question Source: New **Level Of Difficulty: (1-5)** 3
Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** (CFR: 41.7 / 45.5 to 45.8)
Objective:

48. The plant is at full power with all systems in normal alignment

- Instrument Air pressure is lost to MS 5889A, #1 Auxiliary Feed Pump Turbine steam admission valve

How will this failure impact Auxiliary Feedwater flow to #1 Steam Generator?

Auxiliary Feedwater flow to #1 Steam Generator will be about __________ as a result of this event.

A. zero

B. 45 gpm

C. 600 gpm

D. 800 gpm

<u>Answer</u>: B

Explanation/Justification:

A. Incorrect. Plausible if candidate does not know there is a restricting oriface around the AFW pump discharge valve

B. Correct. MS5889A fails open on loss of IA , AFPT 1 would start and increase speed to the High Speed stop, the AF6452, discharge valve would CLOSE because SG level is higher than setpoint, but RO416 around AF6452 would allow 45 gpm to pass to the SG. All other valves in the AFW flowpath are normally open. Pump head and flow are synonymous the K/A is being met by addressing the impact on flow.

C. Incorrect. Plausible because this is the design AFW flowrate

D. Incorrect. Plausible because this is max AFP flowrate

Sys #	**System**	**Category**	**KA Statement**
061	Auxiliary / Emergency Feedwater (AFW) System	Knowledge of the operational implications of the following concepts as they apply to the AFW:	Pump head effects when control valve is shut

K/A# K5.03 **K/A Importance** 2.6 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02528, pg 73, SD 015 pg 39 (2-16) & pg 58 (2-35)

Question Source: Bank 86517 **Level Of Difficulty: (1-5)** 3

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

Objective:

49. The plant is operating at 100% power in a normal lineup.

- A loss of DC distribution buses D1N and DAN occurs.
- Prior to any operator actions, a reactor trip occurs.
- All systems respond, to these conditions, as designed.

How will the AC distribution system respond to these events?

"A" Bus ______(1)________ automatically transfer to the Startup Transformer.

"C1" Bus will be powered from _________(2)___________.

A. (1) will NOT
(2) EDG 1

B. (1) will NOT
(2) "C2" bus

C. (1) will
(2) EDG 1

D. (1) will
(2) "C2" bus

Answer: A

Explanation/Justification:

A. Correct. IAW DB-OP-02539, Attachment 1 page 11

B. Incorrect Part 1 is correct. Part 2 is plausible if candidate does not realize that C2 bus is also de-energized.

C. Incorrect. Part 1 is Plausible if candidate does not recognize that the control power for the source breaker is de-energized. Part 2 is correct.

D. Incorrect. Part 1 is Plausible if candidate does not recognize that the control power for the source breaker is de-energized. Part 2 is plausible if candidate does not realize that C2 bus is also de-energized.

Sys #	System	Category	KA Statement
062	AC Electrical Distribution System	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including:	Effect on instrumentation and controls of switching power supplies

K/A# A1.03 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02539, Attachment 1 page 11

Question Source: New **Level Of Difficulty: (1-5)** 2

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

Objective:

50. The plant is operating at 100% power with all systems in normal alignment.

- A Station Blackout occurs
- Battery load shedding was completed in 30 minutes

Following battery load shedding, the batteries will be able to supply power to the essential loads for an additional MINIMUM of ________________.

A. 30 minutes

B. one hour

C. 90 minutes

D. two hours

Answer: A

Explanation/Justification:

A. Correct. Batteries are designed for only 1 hour and it has taken thirty minutes to reduce the discharge rate.
B. Incorrect. Plausible if candidate believes the one hour time begins at the conclusion of load shedding.
C. Incorrect. Plausible if the candidate uses the battery charger allowable outage time of TS 3.8.4.
D. Incorrect. Plausible if the candidate uses the battery charger allowable outage time of TS 3.8.4 and believes the time does not start until the completion of load shedding.

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

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

References provided to Candidate None **Technical References:** USAR pg 3D-9 and 8.3.2.1.2 pg 8.3-41 & 8.3-44

Question Source: New **Level Of Difficulty: (1-5)** 2

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

Objective:

51. A loss of all Off-site power occurs:
- The reactor tripped
- Both EDGs start
- The Immediate and Supplemental Actions of DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture have been completed
- No symptoms required mitigation
- #1 Makeup Pump is in service
- Reactor Coolant System Tave is 552 °F
- RCS Pressure is 1950 psig

The following electrical power conditions then occurred:
- C1 Bus Voltage – 4200 volts
- D1 Bus Voltage – 4200 volts
- C2 Bus Voltage – 0 volts
- D2 Bus Voltage – 0 volts

Subsequently, an operator dispatched to check operating conditions at both Emergency Diesel Generators reports:
- #2 EDG has a significant oil leak and recommends shutdown.

(1) How can #2 EDG shutdown be accomplished?

(2) Following #2 EDG shutdown, will High Pressure Injection Flow Balancing be required?

A. (1) Only from the Local Emergency Shutdown Pushbutton
(2) NO

B. (1) Only from the Local Emergency Shutdown Pushbutton
(2) YES

C. (1) From either the Local Emergency Shutdown Pushbutton or the Control Room Stop Pushbutton
(2) YES

D. (1) From either the Local Emergency Shutdown Pushbutton or the Control Room Stop Pushbutton
(2) NO

Answer: A

Explanation/Justification:

A. Correct. Specific Rule 3.1 requires use of Attachment 11 when only one HPI Train is available, however since Subcooled Margin has not been lost (RCS temperature 552°F and pressure 1950 psig provides approximately 80 °F SCM), HPI Flow Balancing is not required. K/A is being met by requiring knowledge of EDG S/D while loaded and the impact and procedural requirements on flow balancing for the conditions stated in the stem.

B. Incorrect. Plausible – Specific Rule 3.1 requires use of Attachment 11 when only one HPI Train is available. Since Subcooled Margin has not be lost, HPI Flow Balancing is not required.

C. Incorrect. Plausible – Using the Control Room Stop Pushbutton would allow timely shutdown, however following a safety start, the Control Room Stop Pushbutton is bypassed.

D. Incorrect. Plausible – Using the Control Room Stop Pushbutton would allow timely shutdown, however following a safety start, the Control Room Stop Pushbutton is bypassed.

Sys #	System	Category	KA Statement
064	Emergency Diesel Generator (ED/G) System	Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Effects (verification) of stopping ED/G under load on isolated bus

K/A# A2.14 **K/A Importance** 2.7 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000 pg 239 Specific Rule 3 and pg 321 Attach 11, DB-OP-06316 pg 241.

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

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

52. Which of the following actions, if any, will occur in response to a trip of RE1412 or RE1413, Component Cooling Water Radiation Monitors?

A. No Automatic Actions – Radiation Elements RE1412 and RE1413 provide no automatic functions, indication only

B. CC1412, CCW SURGE TANK THREE WAY VENT VALVE will automatically transfer from the atmospheric vent position to the Miscellaneous Waste Drain Tank vent position.

C. CC1411 A, CCW TO CTMT MOTOR OPERATED ISO and CC1411 B, CCW TO CTMT MOTOR OPERATED ISO Valves close to isolate the Containment CCW Header from the Component Cooling Water System.

D. CC1495, CCW TO AUXILIARY BUILDING NON-ESSENTIALS INLET will close to isolate the Auxiliary Building Non-Essential CCW Header from the Component Cooling Water System

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Answer: B

Explanation/Justification:

A. Incorrect. Plausible - Many plant radiation elements provide no automatic functions – indication only.

B. Correct. If the CCW System becomes contaminated, the venting of the CCW Surge Tank that occurs at 5 psig would release radioactive material to the environment. The automatic action aligns the vent to the Miscellaneous Radwaste System for processing

C. Incorrect. Plausible – Rising Radiation Levels in the Component Cooling Water System are indicative of a leak of radioactive material into the CCW System from a system operating at a higher pressure that CCW. An automatic action to isolate CCW from the affected system would stop radioactive material from entering the CCW System while preserving remaining CCW System Functions. Several loads on the Containment Header operate at higher pressures than the CCW system including Letdown Coolers and Reactor Coolant Pump Thermal Barrier Coolers. In addition, CC1411A and CC1411B valves do have an automatic close feature on CCW Surge Tank Level and SFAS Actuation

D. Incorrect. Plausible - Rising Radiation Levels in the Component Cooling Water System are indicative of a leak of radioactive material into the CCW System from a system operating at a higher pressure that CCW. An automatic action to isolate CCW from the affected system would stop radioactive material from entering the CCW System while preserving remaining CCW System Functions. Several loads on the Auxiliary Building Non-essential Header operate at higher pressures than the CCW system including RCP Seal Return Cooler and Spent Fuel Pool Heat Exchangers. In addition, CC1495 has an automatic close feature on CCW Surge Tank Level.

Sys #	System	Category	KA Statement
073	Process Radiation Monitoring (PRM) System	Knowledge of the physical connections and/or cause-effect relationships between the PRM system and the following systems:	Those systems served by PRMs

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

References provided to Candidate None **Technical References:** SD-017A pg 43 (2-24) Section 2.5.6.1

Question Source: New **Level Of Difficulty: (1-5)** 2

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

Objective:

53. The plant is operating at 100% power.

- SW Pump 1 & 2 are in service
- CCW Pump 1 is in service
- CAC 1 is in service
- Service water header pressure control has been established on the primary side via CCW heat exchanger #3, IAW DB-OP-06261, Service Water System Operating Procedure
- SW 1424 CCW Heat Exchanger 1 outlet temperature control valve is in auto and 50% OPEN
- SW 1356 Containment Air Cooler 1 outlet temperature control valve is in auto and 50% OPEN
- SW 37 CCW Heat Exchanger 3 Discharge Iso is throttled OPEN to achieve a SW Pump 1 header pressure of 115 psig.

While in this configuration:

- SW 37 CCW Heat Exchanger 3 Discharge Iso is throttled OPEN to reduce SW Pump 1 header pressure to 95 psig

How will SW 1424 and SW 1356 respond to this throttling of SW 37?

SW 1424 will throttle ____(1)____.

SW 1356 will throttle ____(2)____.

A. (1) open
(2) open

B. (1) open
(2) closed

C. (1) closed
(2) open

D. (1) closed
(2) closed

Answer: A

Explanation/Justification:

A. Correct. Reducing SW pressure will reduce flow thru CCW 1 HX and CAC 1. The TCVs will both throttle open to maintain a constant temperature.
B. Incorrect. Plausible if candidate believes SW 1356 is designed to maintain a constant pressure at the CACs.
C. Incorrect. Plausible if candidate believes SW 1424 is designed to maintain a constant flow thru the CCW HX.
D. Incorrect. Plausible if candidate believes SW 1356 and 1424 are designed to maintain a constant SW pressure.

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

K/A# A1.02 **K/A Importance** 2.6* **Exam Level** RO
References provided to Candidate None **Technical References:** OS-020 sheets 1 & 2
Question Source: New **Level Of Difficulty: (1-5)** 3
Question Cognitive Level: High - Analysis **10 CFR Part 55 Content:** (CFR: 41.5 / 45.5)
Objective:

54. The plant is operating at 100% power in a normal system alignment:

- Station Air Compressor 2 is in lead and running
- Station Air Compressor 1 in lag and standing by
- The Emergency Instrument Air Compressor is available and in standby

The following event occurs:

- A material handling accident has caused a large break in the Service Air Header downstream of SA2008, STATION AIR HEADER BACK PRESSURE REGULATOR.

Which of the following describes the impact of this break

A. When the Service Air header pressure drops to 90 psig SA2008, STATION AIR HEADER BACK PRESSURE REGULATOR will trip closed. Service Air will be lost.

B. SA2008 STATION AIR HEADER BACK PRESSURE REGULATOR will fail open. Service and Instrument Air will be lost.

C. SA6445, IA/SA CROSS TIE SOLENOID VALVE will close to allow the Emergency Instrument Air Compressor to be dedicated to the Instrument Air System.

D. SA6445, IA/SA CROSS TIE SOLENOID VALVE will open to allow the Emergency Instrument Air Compressor to be dedicated to the Instrument Air System.

Answer: C

Explanation/Justification:

A. Incorrect - Plausible because SA2008 will begin to throttle at 90 psig on the instrument air header. SA2008 will throttle as necessary to maintain back pressure between 80 – 90 psig.
B. Incorrect – Plausible because SA2008 will begin to throttle at 90 psig on the instrument air header. It will not result in a loss of Service and Instrument Air
C. Correct Answer IAW System Description 001, Service and Instrument Air
D. Incorrect – does move in response to the failure, however it is moving in the opposite direction.

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

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

References provided to Candidate None **Technical References:** SD 001 pg 2-1 & 2-2

Question Source: New **Level Of Difficulty: (1-5)** 4

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

Objective:

55. The plant is operating at 100% power in a normal system alignment.
- A LOCA occurs inside CTMT
- The reactor automatically trips
- CTMT pressure rises to 42 psia
- RCS pressure is 400 psig
- SFAS Levels 1 **AND** 2 have actuated

What SFAS related actions, if any, are required?

A. No actions are required

B. Manually actuate SFAS level 3 components **ONLY**

C. Manually actuate SFAS level 4 components **ONLY**

D. Manually actuate SFAS level 3 **AND** 4 components

Answer: D

Explanation/Justification:

A. Incorrect. Only SFAS levels 1 and 2 actuated. SFAS levels 1, 2, 3, and 4 should be actuated
B. Incorrect. Plausible since RCS pressure is below the SFAS level 3 setpoint
C. Incorrect. Plausible if candidate does not recognize that RCS pressure is below the SFAS level 3 setpoint and CTMT is above the setpoint
D. Correct. With CTMT pressure greater than 40 psig, SFAS levels 1, 2, 3, and 4 should be actuated.

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

K/A# A2.03 **K/A Importance** 3.5* **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000 Table 2 pg 409 -413
DBBP-TRAN-0034, pg 9

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

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

Objective:

56. The following plant conditions exist:

- The reactor is operating at 100% power
- All ICS stations are in automatic
- Tave is selected to Loop 2

The following event occurs:

- Selected RCS Loop 2 Thot slowly fails LOW such that a SASS transfer does NOT occur

Which of the following describes the effect on ICS?

A. A BTU limit has been exceeded causing Main Feedwater flow to lower

B. Temperature compensation for RCS Loop 2 flow is lost causing a runback on low RCS flow

C. The differential temperature between RCS Loops 1 and 2 increases causing feedwater loop flows to re-ratio

D. Selected Tave lowers causing ICS to pull rods in an attempt to return Tave to setpoint

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Answer: D

Explanation/Justification:

A. Incorrect – RCS temperature is part of the BTU limit calculation, but it only provides an alarm, no adjustment to feedwater.
B. Incorrect – Plausible since temperature compensation is used for RCS Flow indication, however with 4 RCPs in service, actual RCS Flow is not used to determine is a runback due to loss a Reactor Coolant Pump is required. RCP Breaker status is used.
C. Incorrect – Plausible since temperature compensation is used for RCS Flow indication, however with 4 RCPs in service, actual RCS Flow is not used to adjust FW flow to individual Steam Generators.
D. Correct – The selected Tave signal fails lows causing a reduction in indicated Tave. ICS will respond by pulling control rods and therefore rasing reactor power to restore Tave to setpoint.

Sys #	System	Category	KA Statement
001	Control Rod Drive System	Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CRDS controls including:	RCS average temperature indications (T-ave.)

K/A# A1.07 **K/A Importance** 3.7 **Exam Level** RO

References provided to Candidate None **Technical References:** ICS Analog to Digital Drawing for Feedwater

Question Source: Bank 39087 **Level Of Difficulty: (1-5)** 4

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

Objective:

57. When adequate SCM exists:

Which of the following conditions REQUIRES the Operator to select Incore Thermocouples on the PAMP Tsat Meter?

A. ONLY 1 RCP running

B. ONLY 1 RCP running per loop

C. MU/HPI/PORV Cooling

D. Natural Circulation has been verified to exist

Answer: C

Explanation/Justification:

A. Incorrect: plausible because there is no force flow in one RCS Loop, however the idle loop will have back flow from the running RCP.
B. Incorrect plausible because total RCS flow if reduced. Incores are required without forced or natural circulation.
C. Correct, there is no flow past the RCS Th RTD, therefore the operator is required to select the Incore thermocouple,
D. Incorrect: plausible because no forced flow. With confirmed NC, selection of Incores is not required.

Sys #	System	Category	KA Statement
017	In-Core Temperature Monitor (ITM) System	Generic	Ability to use plant computers to evaluate system or component status.

K/A# 2.1.19 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000 pg 18

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

Question Cognitive Level: Low - Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 45.12)

Objective:

58. The plant is operating at 100% power in a normal system alignment.

The following plant conditions are noted:

- Annunciator 14-2-D, ICS/NNI 118V AC PWR TRBL alarms
- Loss of blue light on SASSed instrument's selector switches
- SCR Bank, RC PRESSURE CONTROL, Hand Auto Station Lights are Both ON.
- RCP Seal Injection total flow indication is lost
- A significant plant transient is in progress.

(1) Which NNI Power Supply has been lost?

(2) What actions are required to respond to this condition?

A. (1) NNI X AC Power
(2) Trip the Reactor, Go To DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture

B. (1) NNI X AC Power
(2) Trip the Reactor, Initiate and Isolate SFRCS, Go To DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture

C. (1) NNI Y AC Power
(2) Trip the Reactor, Go To DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture

D. (1) NNI Y AC Power
(2) Trip the Reactor, Initiate and Isolate SFRCS, Go To DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture

---

Answer: B

Explanation/Justification:

A. Incorrect – Plausible because This immediate action for a loss of NNI X AC Power with a significant Transient in Progress, but does not include required action to initiate and isolate SFRCS.
B. Correct – This is the correct immediate action for a loss of NNI X AC Power with a significant Transient in Progress.
C. Incorrect – Plausible because This is immediate action for a loss of NNI X AC Power with a significant Transient in Progress, but does not include required action to initiate and isolate SFRCS. In this case, NNI Y AC is lost, not NNI X AC.
D. Incorrect – Plausible because The immediate action for a loss of NNI X AC Power with a significant Transient in Progress. In this case, NNI Y AC is lost, not NNI X AC

Sys #	System	Category	KA Statement
016	Non-Nuclear Instrumentation System (NNIS)	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:	Loss of power supply

K/A# A2.02 **K/A Importance** 2.9* **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02532, Loss of NNI/ICS Power step 3.1

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

Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Objective:

59. The plant is operating at 100% power in a normal system alignment.
- DB-SP-03320, Hydrogen Dilution System Train 1 Quarterly Test is in progress.
- CV5090 Hydrogen Dilution System 1 Containment Isolation is OPEN.
- Hydrogen Dilution Blower 1 is RUNNING.

An SFAS actuation occurs. All equipment responds as designed.

You are directed to place Hydrogen Dilution System Train 1 in service to reduce Containment Hydrogen.

What are the MINIMUM actions required, if any, to place Hydrogen Dilution System Train 1 in service?

A. No actions are required. SFAS Actuation has properly aligned the system.

B. Restart Hydrogen Dilution Blower 1 ONLY.

C. BLOCK the SFAS signal and OPEN CV5090 Hydrogen Dilution System 1 Containment Isolation ONLY.

D. BLOCK the SFAS signal and OPEN CV5090 Hydrogen Dilution System 1 Containment Isolation AND restart Hydrogen Dilution Blower 1.

Answer: C

Explanation/Justification:

A. Incorrect – Plausible because the H2 Dilution Blow is already running with Containment Isolation valve open, candidate may assume no action is required, however SFAS level 2 will close the Containment Isolation Valve.

B. Incorrect – Plausible since on an SFAS with a Loss of Off-Site Power, the H2 Dilution blower would load shed and require restart.

C. Correct – SFAS Level will close the Containment Isolation. The H2 Dilution Blower would continue to operate.

D. Incorrect - Plausible because the H2 Dilution Blow is already running with Containment Isolation valve open, candidate may assume containment isolation will close and blower will load shed.

Sys #	System	Category	KA Statement
028	Hydrogen Recombiner and Purge Control System (HRPS)	Ability to manually operate and/or monitor in the control room:	HRPS controls

K/A# A4.01 **K/A Importance** 4.0* **Exam Level** RO

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

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

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

Objective:

60. Fuel transfer operations are in progress and all systems are in normal alignment for this condition.

- The Equipment Hatch is installed with FOUR (4) bolts only.
- The personnel airlock is closed.
- CTMT Purge is in service on Containment.
- The PZR code safeties are removed.

The following events occur
- CTMT Purge Exh Fan trips.
- CTMT Purge Supply Fan does NOT trip.

What will be the impact of this ventilation alignment?

A. Incore Tank level will not change since it is vented during fuel handling operations

B. RCS level will not change since the Decay Heat removal system is a closed loop cooling system

C. Pressurizer level could rise since the pressurizer is vented to containment atmosphere

D. Spent Fuel Pool level could rise high enough to overflow to the miscellaneous waste drain tank

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Answer: D

Explanation/Justification:

A. Incorrect. Plausible the incore tank is vented, but will follow level in the refueling canal since the incore guide tubes provide a flowpath between the Refueling Canal and the incore tank.

B. Incorrect. Plausible because the Decay Heat Remove System is closed loop until the RCS is drained and vented. Once the head is removed and refueling canal filled, RCS level be equal to Refueling canal level and change as refueling canal level changes.

C. Incorrect. Plausible because pressurizer level is vented to containment atmosphere, but a pressure rise in containment would not cause Pressurizer level to rise.

D. Correct. With the supply fan still running and the Equipment hatch installed, the pressure in containment will rise. This increase in pressure (up to 6.5 inches water) could be enough to force water thru the transfer tube which will cause level in the SFP to rise and possibly overflow to the MWDT. A caution in DB-OP-06502, L&P 2.2.3. pg 5 alerts the operators to this potential condition.

Sys #	System	Category	KA Statement
029	Containment Purge System (CPS)	Knowledge of the effect that a loss or malfunction of the Containment Purge System will have on the following:	Containment parameters

K/A# K3.01 **K/A Importance** 2.9 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-06503, L&P 2.2.3. pg 5

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

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

Objective:

61. The following plant conditions exist:

- The core was defueled 24 hours ago
- DH Pump 1 is providing cooling for the Spent Fuel Pool (SFP)
- DH 13B, DH Cooler 1 Bypass Valve is 20% open
- DH 14B, DH Cooler 1 Outlet Valve, is 20% open
- SFP Pump 1 is out of service to replace the motor. The liquid side of the SFP pump is intact. The SFP breaker is tagged open.

The following event occurs:

- The SFAS solenoid to DH14B fails to the SA position (de-energized)

What effect, if any, will this have on SFP temperature and why?

SFP temperature will _____(1)_______ because __________(2)_________________.

A. (1) lower
(2) DH 14B fails to the full flow position

B. (1) rise
(2) DH 14B fails closed

C. (1) be unaffected
(2) the SFP cooler valves are manually throttled to control SFP flow and cooling

D. (1) be unaffected
(2) the cooling water system (Component Cooling Water) is not effected

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Answer: A

Explanation/Justification:

A. Correct. DH14B will open due to the solenoid failure and raise flow through the DH cooler and SFP lowering the SFP temperature

B. Incorrect. Plausible because this is the normal SFP cooling alignment and would be the effect if the candidate assumes these coolers are used with the DH system

C. Incorrect. Plausible because it is the opposite of the correct answer and would be true if the candidate thinks that DH 14B fails closed on lose of the solenoid power

D. Incorrect. Plausible because CCW cools both the SFP and DH coolers, but the increase SFP flowrate through the DH Cooler would lower the water temperature returning to the SFP and lower the temperature of the SFP

Sys #	System	Category	KA Statement
033	Spent Fuel Pool Cooling System (SFPCS)	Knowledge of the physical connections and/or cause/effect relationships between the Spent Fuel Pool Cooling System and the following systems:	RHRS

K/A# K1.02 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** OS-004, sheet 2, Control Logic 1 and 9, DB-OP-06405 pg 16 -Caution 3.4.2, and pg 80 - Attachment 5.

Question Source: New **Level Of Difficulty: (1-5)** 4

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

Objective:

62. The plant is operating at 100% power in a normal system alignment.
- Movement of fuel in the Spent Fuel Pool is in progress.

The following event occurs:
- RE 8446, Fuel Handling Area Ventilation System Exhaust Radiation Monitor fails HIGH.

How will the Fuel Handling and Emergency Ventilation Systems respond to these conditions?

A. Fuel handling supply and exhaust fans trip.
Station EVS starts automatically.

B. Fuel handling supply and exhaust fans stay running.
Station EVS starts automatically.

C. Fuel handling supply and exhaust fans stay running.
Station EVS stays shutdown.

D. Fuel handling supply and exhaust fans trip.
Station EVS stays shutdown.

Answer: A

Explanation/Justification:

A. Correct. IAW OS-0033D and OS-034 sh1.
B. Incorrect: Plausible since a trip of related RE 5403, Fuel Area Exhaust Monitor will not cause Fuel Handling Supply and Exhaust to shutdown.
C. Incorrect Plausible since a trip of related RE 5403, Fuel Area Exhaust Monitor will not cause Fuel Handling Supply and Exhaust to shutdown and EVS would stay shutdown.
D. Incorrect: Plausible, in general process radiation monitors will cause realignment of the system and SU of the backup system. EVS is manually started for high radiation in other Aux building ventilation systems.

Sys #	System	Category	KA Statement
034	Fuel Handling Equipment System (FHES)	Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling System :	Radiation monitoring systems

K/A# K6.02 **K/A Importance** 2.6 **Exam Level** RO
References provided to Candidate None **Technical References:** DB-OP-02530, pg 24, 26, 31
Question Source: Bank (Modified) 35232 **Level Of Difficulty: (1-5)** 3.5
Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** (CFR: 41.7 / 45.7)
Objective:

63. The plant is operating at 100% power with all systems in normal alignment.

- Waste Gas Decay Tank 2 is aligned for cover gas
- Waste Gas Decay Tank 3 is tagged out

The following annunciator alarms on the RADWASTE CONTROL ALARM PANEL:

- 50-1-K WGST 0_2/H_2 HI

Waste Gas Surge Tank local 0_2 and H_2 values are as follows:

- 0_2 is 2.5%
- H_2 is 4.5%

Based on these indications, what is the current status of the Gaseous Radwaste System?

The concentration of gases in Waste Gas Surge Tank are _______(1)_________ than the Technical Normal Conditions limits of TRM 8.7.5 Explosive Gas Mixtures and _____________(2)_____________ may also contain this same concentration of gases.

A. (1) greater
(2) Waste Gas Decay Tank 1

B. (1) less
(2) Waste Gas Decay Tank 1

C. (1) greater
(2) Waste Gas Decay Tank 2

D. (1) less
(2) Waste Gas Decay Tank 2

Answer: A

Explanation/Justification:

A. Correct. O2 & H2 concentrations are greater than the TRM limit. IAW Figure OS-0030 sh 1, the WGST is normally aligned to WGDT 1 and WGDT 2 is normally aligned to receive cover gases which are not sampled by the detectors that alarmed and indicate locally.

B. Incorrect. O2 & H2 concentrations are greater than the TRM limit. IAW Figure OS-0030 sh 1, the WGST is normally aligned to WGDT 1 and WGDT 2 is normally aligned to receive cover gases which are not sampled by the detectors that alarmed and indicate locally.

C. Incorrect. O2 & H2 concentrations are greater than the TRM limit. IAW Figure OS-0030 sh 1, the WGST is normally aligned to WGDT 1 and WGDT 2 is normally aligned to receive cover gases which are not sampled by the detectors that alarmed and indicate locally.

D. Incorrect. O2 & H2 concentrations are greater than the TRM limit. IAW Figure OS-0030 sh 1, the WGST is normally aligned to WGDT 1 and WGDT 2 is normally aligned to receive cover gases which are not sampled by the detectors that alarmed and indicate locally.

Sys #	System	Category	KA Statement
071	Waste Gas Disposal System (WGDS)	Knowledge of the operational implication of the following concepts as they apply to the Waste Gas Disposal System:	Relationship of hydrogen/oxygen concentrations to flammability

K/A# K5.04 **K/A Importance** 2.5 **Exam Level** RO

References provided to Candidate None **Technical References:** TRM 8.7.5 rev. 0 bases page B 8.7.5-1; Figure OS-0030 sh 1 rev. 35

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

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

Objective:

64. The plant is in Mode 3 following an inadvertent Reactor Trip.

- All systems in normal alignment for this post-trip condition EXCEPT Circulating Water Pump 4 is on clearance.
- BOTH Circulating water bypass lines are CLOSED

The following event occurs:

- Circulating Water Pump 1 TRIPS

How will the circulating water pump discharge valves respond to this pump tripping?

Circulating Water Pump 1 discharge valve will close in ____(1)______ speed to ~40 degrees open, then fully close in ______(2)______ speed.
Circulating Water Pump 2 discharge valve will ___________________(3)__________________.

A. (1) fast
(2) slow
(3) remain in its current position

B. (1) fast
(2) slow
(3) go to the "Throttle" position

C. (1) slow
(2) fast
(3) remain in its current position

D. (1) slow
(2) fast
(3) go to the "Throttle" position

Answer: B

Explanation/Justification:

A. Incorrect. Circ water pump 2 discharge valve will go to the throttle position.

B. Correct. IAW DB-OP-02517 page 60. The design of the circ water pump valve interlocks is to protect the pumps and allow them to remain available to perform their heat sink function. For the conditions in the stem, if Circ water pump 2 discharge valve does not go to the "throttle" position, the pump could experience run-out conditions, which could lead to a pump trip, which would reduce the heat removal capability of the condenser. In this manner, the circ water pump discharge valve interlocks help provide for a heat sink for the RCS.

C. Incorrect. Circ water pump 2 discharge valve will go to the throttle position, and the speeds for the Circ water pump 1 discharge valve are reversed.

D. Incorrect. the speeds for the Circ water pump 1 discharge valve are reversed.

Sys #	System	Category	KA Statement
075	Circulating Water System	Knowledge of circulating water system design feature(s) and interlock(s) which provide for the following:	Heat sink

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

References provided to Candidate None **Technical References:** DB-OP-02517 Rev. 06 page 60

Question Source: New **Level Of Difficulty: (1-5)** 2

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** (CFR: 41.7)

Objective:

65. Water from the Fire Protection System is capable of providing a “backup” water supply via existing piping to which of the below listed systems?

A. Component Cooling Water System

B. Circulating Water System

C. Auxiliary Feedwater System

D. Service Water System

Answer: C

Explanation/Justification:

A. Incorrect. FPS does not connect to CCW
B. Incorrect. FPS does not connect to Circulating water
C. Correct. IAW plant P&IDs
D. Incorrect. FPS does not connect to Service water

Sys #	**System**	**Category**	**KA Statement**
086	Fire Protection System (FPS)	Knowledge of the physical connections and/or cause-effect relationships between the Fire Protection System and the following systems:	AFW system

K/A# K1.03 **K/A Importance** 3.4 **Exam Level** RO

References provided to Candidate None **Technical References:** Figure OS-017A sh 1 grid A1 and Note 2; Figure OS-047A sh 4 grid F56

Question Source: New **Level Of Difficulty: (1-5)** 2

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

Objective:

66. An RCS cooldown is in progress with the following plant conditions:

- RCS average temperature at 425 °F.
- RCP 1-1 is in operation
- RCP 1-2 is shutdown
- RCP 2-1 is in operation
- RCP 2-2 is shutdown

Using the attached RCS Pressure/Temperature Limit curve:

What is the MINIMUM allowed pressure for the current RCS average temperature?

A. 300 psig

B. 500 psig

C. 650 psig

D. 750 psig

<u>Answer</u>: D

Explanation/Justification:

A. Incorrect. This is the pressure for saturation as given by curve 5

B. Incorrect. This is the minimum subcooling pressure as given by curve 2

C. Incorrect. This is the minimum NPSH pressure for 2 RCP/loop as given by curve 4

D. Correct. This is the minimum NPSH pressure for 1 RCP/loop as given by curve 3

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to interpret reference materials, such as graphs, curves, tables, etc.

K/A# 2.1.25 **K/A Importance** 3.9 **Exam Level** RO

References provided to Candidate DB-PF-06703 CC1.1 **Technical References:** DB-PF-06703 pg 11 & 12,CC1.1 Rev.19

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

Question Cognitive Level: High - Application **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.12)

Objective:

67. IAW the guidance provided within NOBP-LP-3008, FENOC Electrical Arc Practices
What are the MINIMUM PPE requirements to RACK-IN an energized 480V Unit Substation Breaker?

(1) Fire retardant clothing rated at a minimum of 65 calories with blast hood
(2) 100% Cotton/Natural Fiber clothing.
(3) 20 cal flash suit including hood
(4) rubber gloves and leathers
(5) 8 calorie Nomex Coveralls
(6) Arc Face Shield

A. 3, 5, & 6

B. 1, 2, & 6

C. 1, 4, & 5

D. 2, 3, & 4

Answer: D

Explanation/Justification:

A. Incorrect. Items 3, 5, and 6 are required PPE for other electrical tasks.
B. Incorrect. Items 1 and 6 are required PPE for other electrical tasks.
C. Incorrect. Items 1 and 5 are required PPE for other electrical tasks.
D. Correct. IAW Attachment 2 page 28 of NOBP-LP-3001 rev. 06

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).

K/A# 2.1.26 **K/A Importance** 3.4 **Exam Level** RO
References provided to Candidate None **Technical References:** NOBP-LP-3008 rev. 06 page 28
Question Source: New **Level Of Difficulty: (1-5)** 3
Question Cognitive Level: Low- Fundamental **10 CFR Part 55 Content:** (CFR: 41.10 / 45.12)
Objective:

68. IAW the guidance provided in NOP-SS-3001, Procedure Review and Approval, which of the procedure changes, listed below, can **NOT** be made by using the “Procedure Correction” method?

(1) Changes to setpoints
(2) Changing procedure number or title
(3) Changes to equipment position
(4) Correcting Table of Contents
(5) Correcting typographical errors
(6) Changing the purpose of the procedure

A. 1, 2, & 4

B. 2, 4, & 5

C. 1, 3, & 6

D. 3, 5, & 6

Answer: C

Explanation/Justification:

A. Incorrect. Items 2 and 4 are allowed
B. Incorrect. All 3 are allowed.
C. Correct. Items 1 and 3 are specifically called out as not being allowed. Item 6 is a “significant change” as defined by NOP-SS-3001
D. Incorrect. Item 5 is allowed

Sys #	System	Category	KA Statement
N/A	N/A	Generic	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 rev. 17 pages 7 and 8
Question Source: New **Level Of Difficulty: (1-5)** 2
Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** (CFR: 41.10 / 43.3 / 45.13)
Objective:

69. What are the Technical Specification 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits, in Mode 1, with three RCPs operating?

RCS loop pressure shall be _____(1)______

RCS hot leg temperature shall be _____(2)______

RCS total flow rate shall be _____(3)______

A. (1) ≥ 2060.8 psig
(2) ≤ 610 °F
(3) ≥ 290,957 gpm

B. (1) ≥ 2064.8 psig
(2) ≤ 525 °F
(3) ≥ 389,500 gpm

C. (1) ≥ 2064.8 psig
(2) ≤ 610 °F
(3) ≥ 389,500 gpm

D. (1) ≥ 2060.8 psig
(2) ≤ 525 °F
(3) ≥ 290,957 gpm

Answer: A

Explanation/Justification:

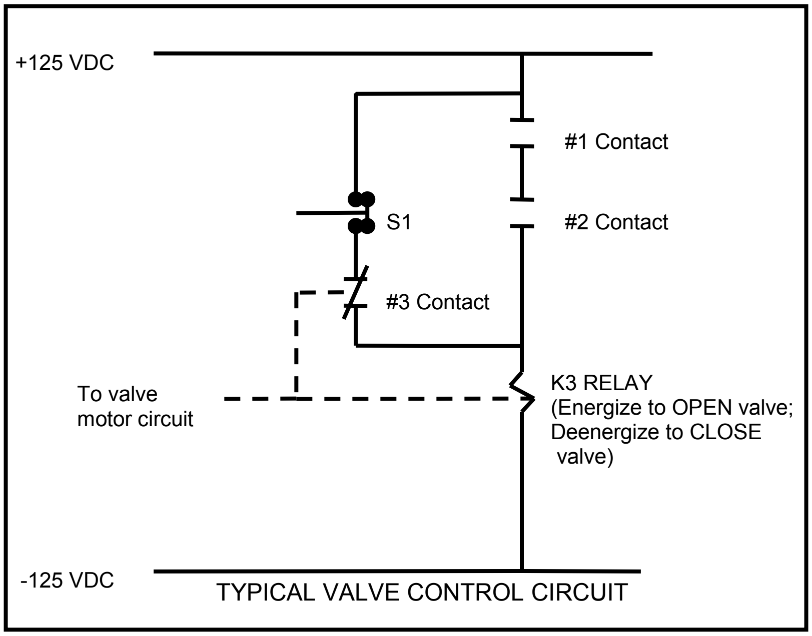
A. Correct. IAW TS 3.4.1
B. Incorrect. Flow is for all 4 RCPs operating and temp is the TS minimum temp for criticality.
C. Incorrect. RCS pressure is 2064.8 instead of 2060.8, Flow is for all 4 RCPs operating
D. Incorrect. temp is the TS minimum temp for criticality;

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of limiting conditions for operations and safety limits.

K/A# 2.2.22 **K/A Importance** 4.0 **Exam Level** RO
References provided to Candidate None **Technical References:** TS 3.4.1 Amend. 279
Question Source: New **Level Of Difficulty: (1-5)** 4
Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** (CFR: 41.5 / 43.2 / 45.2)
Objective:

70. Refer to the drawing of a typical valve control circuit for a 480 VAC motor-operated valve (see figure below).

With **NO** initiating condition present, the valve is currently OPEN. If the S1 pushbutton is depressed, the valve will _______(1)_________ and when the S1 pushbutton is subsequently released (spring return) the valve will _______(2)_________.



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:

A. Incorrect. Wrong initial response; wrong subsequent response.
B. Correct. Right initial response; right subsequent response.
C. Incorrect. Wrong initial response; right subsequent response.
D. Incorrect. Right initial response; wrong subsequent response.

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 FundamentalsComponents – Breakers, Relays, & Disconnects pg 20

Question Source: Bank BV 2009 NRC exam **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:

71. The following conditions exist at a job site:

- The general area radiation levels are 40 mr/hr.
- Radiation levels with shielding are 10 mr/hr.
- Total time for a Radiation Protection individual to install **AND** remove the shielding is fifteen (15) minutes.
- Time to conduct the task with one worker is one (1) hour.
- Time to conduct the task with two workers is twenty (20) minutes

Assumptions:

- If shielding is used, shielding is installed and removed by Radiation Protection personnel.

Which of the following will result in the **LOWEST** whole body dose for the evolution?

Conduct the task with ____________________

A. One (1) worker with shielding.

B. Two (2) workers with shielding.

C. One (1) worker without shielding.

D. Two (2) workers without shielding.

Answer: B

Explanation/Justification:

A. Incorrect. Dose to install shielding = 10 mr + 10mr/hr = 20 mr

B. Correct. Dose to install shielding = 10 mr + (.33)(10) = (3.3)(2) = 6.6 + 10 = 16.6 mr. In order to comply with radiation work permit requirements of maintaining dose as low as reasonably achievable, the lowest dose derived is by using a worker to install shielding and use two workers to perform the job with the shielding in place.

C. Incorrect. Dose with one worker without shielding is 40 mr x 1hr = 40 mr

D. Incorrect. Dose with two workers without shielding is 40 mr x (.33)(40) = (13.2)(2) = 26.4 mr

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Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to comply with radiation work permit requirements during normal or abnormal conditions.

K/A# 2.3.7 **K/A Importance** 3.5 **Exam Level** RO

References provided to Candidate None **Technical References:** NOP-OP-4107 rev. 7 page 6 item 4.1.4.1 and 2 Radiation Work Permit (RWP)

Question Source: Bank BV 2009 NRC Exam **Level Of Difficulty: (1-5)** 2

Question Cognitive Level: High - Analysis **10 CFR Part 55 Content:** (CFR: 41.12 / 45.10)

Objective:

72. During a design bases Loss of Coolant Accident, the operator is directed to restore power to DH7A, DH7B, DH9A, DH9B, and HP31 to prepare for ECCS suction transfer to the Emergency Sump.

Which of the following describes the Radiological consequences of performing this task?

A. Since the task is completed prior to transfer of ECCS Pump suctions to the Emergency Sump, elevated dose rates are not expected in the Auxiliary Building.

B. Elevated dose rates are expected in the ECCS Pump Rooms, but no other high radiation areas will exist in the Auxiliary Building.

C. High dose rates are expected in the Auxiliary Building, but due to short time required to perform this task, prior approval from the Emergency Director is not required.

D. Extremely High does rates in the Auxiliary Building are expected. The Emergency Director must authorize this emergency exposure to protect equipment important to the health and safety of the public.

Answer: C

Explanation/Justification:

A. Incorrect – Plausible because the task is completed prior to sump transfer. Sump transfer will raise dose rates in the auxiliary building.

B. Incorrect – Plausible because the ECCS Pump rooms are low in the Auxiliary Building adjacent to Containment and the Containment isolation for the Emergency Sump. This area will experience elevated dose rates, but other areas in the Auxiliary Building will as well.

C. Correct DB-OP-02000 Attachment 7 and Bases and Deviation Document for DB-OP-02000 for Dose Assessment and Attachment 7. Following the prescribe path will ensure the worst case projected dose remains acceptable

D. Incorrect - Plausible because there will be high radiation levels in the Auxiliary Building. Following the prescribe path will ensure the worst case projected dose remains acceptable. Prior approval the Emergency Director is not required.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

K/A# 2.3.12 **K/A Importance** 3.2 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000 Attachment 7 and Bases and Deviation Document for DB-OP-02000 for Dose Assessment.

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

Question Cognitive Level: Low - Fundamental **10 CFR Part 55 Content:** (CFR: 41.12 / 45.9 / 45.10)

Objective:

73. During Inadequate Core Cooling, the operator is directed by DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture to lower SG Pressure in an attempt to induce Primary to Secondary Heat Transfer.

While reducing Steam Generator Pressure the operators are cautioned **NOT** to allow SG pressure to drop below 35 psig.

What is the reason for this caution?

35 psig in the SGs is the ____.

A. value used in DB-OP-02000 to determine if an Overcooling of the Reactor Coolant System is in progress

B. setpoint for Steam Feed Rupture Control System actuation, which will isolate the atmospheric vent valves

C. the minimum pressure required to efficiently operate the Steam Jet Air Ejectors which are required to maintain Condenser Vacuum

D. the minimum steam pressure required to run an Auxiliary Feedpump Turbine at a speed that will adequately provide bearing lubrication

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Answer: D

Explanation/Justification:

A. Incorrect. Plausible because DB-OP-02000 directs implementation of the Overcooling Section when SG Pressure is less than 960 psig, however Section 9 direction for Inadequate Core Cooling takes precedence over Section 7 Direction for Overcooling in accordance with DB-OP-02000, Bases and Deviation Document.

B. Incorrect. Plausible because if Steam Generator Pressure is reduce below 620 psig, SFRCS will actuate on Low Steam Generator causing both Atmospheric Vent Valves (AVV) to close which would stop the pressure reduction. In this case, manual control of the AVV would be established to further lower as required.

C. Incorrect. Plausible if the candidate assumes the steam is being dumped to the Condenser. Lowing steam generator pressure will cause Auxiliary Steam Pressure to be reduced if being supplied via the Main Steam Reducer. Air Ejectors become inefficient with low input steam

D. Correct. RO question since it involves a mitigation strategy. Lower SG Pressure to less than 35 psig would result in a loss of feedwater to the Steam Generators.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the operational implications of EOP warnings, cautions, and notes.

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

References provided to Candidate None **Technical References:** DB-OP-02000 Step 9.7 Caution

Question Source: New **Level Of Difficulty: (1-5)** 2

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

Objective:

74. A General Emergency has been declared and all Emergency Response facilities have been manned and activated. All appropriate turnovers are complete.

Who, by title, has the authority to make Offsite Protective Action Recommendations (PARs), for these conditions?

A. Emergency Plant Manager

B. Emergency Offsite Manager

C. Dose Assessment Coordinator

D. Emergency Director

Answer: D

Explanation/Justification:

A. Incorrect. Plausible since some items may be delegated to the EPM, but not PARs.

B. Incorrect. Plausible since this individual is responsible for offsite plume tracking and dose assessment.

C. Incorrect. Plausible since this individual is responsible for the dose calculations, but not the associated PARs.

D. Correct. IAW RA-EP-02010 page 7 the ED is responsible for offsite PARs and this is a Non-Delegable Responsibility.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the lines of authority during implementation of the emergency plan.

K/A# 2.4.37 **K/A Importance** 3.0 **Exam Level** RO

References provided to Candidate None **Technical References:** RA-EP-02010 Rev. 12 page 7

Question Source: New **Level Of Difficulty: (1-5)** 2

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

Objective:

75. The following plant conditions exist:

- The reactor was tripped 5 minutes ago.
- A 60 gpm LOCA is in progress.
- RCS pressure is 2155 psig.
- RCS temperature is 555 °F
- Pressurizer level is 100 inches and steady.
- The reactor operators are performing Attachment 1, Primary Inventory Control Actions and Attachment 2, Steam Generator Inventory and Pressure Control Actions.

The following indications occur:

- SUBCOOL MARGIN LO (4-1-B)
- TDI 4950, TSAT meter, reads zero.
- **NO** other initial plant conditions have changed.

The operators are required to _______________.

A. trip all RCPs and continue in DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture

B. trip all RCPs and implement DB-OP-02522, Small RCS Leaks

C. leave the RCPs running and implement DB-OP-02522, Small RCS Leaks

D. leave the RCPs running and continue in DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture

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Answer: D

Explanation/Justification:

A. Incorrect. Plausible because this is the action that should be taken for a Loss of Subcooled Margin (LSCM) per DB-OP-02000

B. Incorrect. Plausible because tripping RCPs is the action for LSCM and a change in RCS leak size is the most likely cause for a rapid change in subcooling margin

C. Incorrect. Plausible because a change in the RCS leak size is the most likely cause for a rapid change in subcooling margin

D. Correct. Annunciator 4-1-B alarms when subcooling margin is $\leq$20°F. Since no initial conditions, saturation temperature for 2155 psig (~2170 psia) is ~669°F. Therefore subcooling is 114°F (669-555). The alarms/.indication is the result of a failed instrument and actions associated with the alarm should not be taken

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Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to verify that the alarms are consistent with the plant conditions.

K/A# 2.4.46 **K/A Importance** 4.2 **Exam Level** RO

References provided to Candidate None **Technical References:** DB-OP-02000 Supplemental Actions and Attachment 1, Primary Inventory Control Actions

Question Source: Bank 74326 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: High - Analysis **10 CFR Part 55 Content:** (CFR: 41.10 / 43.5 / 45.3 / 45.12)

Objective:

76. The Unit is operating at 100% power with all systems in normal alignment.

- A loss of both Makeup Pumps has occurred **AND** neither pump can be started.
- PZR Level is 200 inches and lowering at 1 inch/min.
- RCS Tavg is 582 °F and stable
- **NO** Operator Actions have been taken
- DB-OP-02512, Makeup and Purification System Malfunctions has been entered

If the trends continue, how long will it be before the Reactor is required to be tripped?

A. 20 minutes

B. 40 minutes

C. 100 minutes

D. 160 minutes

Answer: B

Explanation/Justification:

A. Incorrect. Plausible if candidate believes the procedural requirement to trip the Rx is 180 inches or 20 inches below existing level.

B. Correct. IAW DB-OP-02512 page 9 carryover step. SRO must be familiar with the supplemental actions that require a Rx trip if PZR level is less than 160 inches (with Tavg at 582°F). SRO must then be able to calculate how long until PZR level will reach the required level.

C. Incorrect. Plausible because for small RCS Leaks and SG Tube Ruptures, the Reactor is tripped at 100 inches. This would require 100 minutes.

D. Incorrect. Plausible this would be the 40 inches level where the PZR heaters cut-out.

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

K/A# AA2.04 **K/A Importance** 3.8 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02512 page 9 carryover step

Question Source: New **Level Of Difficulty: (1-5)** 2

Question Cognitive Level: High - Analysis **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

77. The following plant conditions exist:
    - The plant is operating at 100% power.
    - CCW Train 1 Ventilation is removed from service due to a motor problem.
    - No loss of Safety Function exists.

    What LCO Required Action(s), if any, must be entered?

    - 3.5.2, ECCS - Operating
    - 3.7.7, Component Cooling Water (CCW) System
    - 3.8.1, AC Sources - Operating

A. No LCO actions are required

B. LCO 3.7.7 only

C. LCO 3.7.7 and LCO 3.5.2 only

D. LCO 3.7.7 and LCO 3.8.1 only

<u>Answer</u>: D

Explanation/Justification:

A. Incorrect. CCW Train 1 Ventilation impacts the Operability of CCW Train 1 since it is a support system necessary for CCW to perform its design function.

B. Incorrect. Plausible if student applies TS 3.0.6 without regard to the Note in TS 3.7.7

C. Incorrect. Plausible if student fails to apply TS 3.0.6 and believes that ECCS is also to be declared inoperable when CCW is inoperable.

D. Correct. IAW DB-OP-06362 page 9 step 2.2.16; SRO only since the candidate is required to apply TS 3.0.6 and its exception regarding the EDG cooling. SRO must make an operability determination for CCW when CCW ventilation is inoperable, then apply the appropriate TS actions.

Sys #	**System**	**Category**	**KA Statement**
026	Loss of Component Cooling Water (CCW)	Generic	Ability to determine operability and/or availability of safety related equipment.

K/A# 2.2.37 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-06262 page 9 step 2.2.16; TS 3.7.7 pg 3.7.7-1

Question Source: Bank 79610 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: High - Application **10 CFR Part 55 Content:** CFR: 43(b)(2)

Objective:

78. The following plant conditions exist:
    - Plant is operating at 100% power
    - Plant is in a normal lineup

    The following event occurs
    - Makeup Tank level lowers at 2 inches/minute
    - Tave is 582 °F and steady
    - Annunciator 12-1-A, MN STM LINE 1 RAD HI, is in alarm
    - Annunciator 9-4-A, VAC SYS DISCH RAD HI, is in alarm

    (1) Which of the following procedures is required to be implemented?

    (2) The Updated Final Safety Analysis Report assumes a Steam Generator Tube Leak/Rupture of ________________ will be isolated in ______________.

A. (1) DB-OP-2531, Steam Generator Tube Leak Abnormal Procedure
(2) 1 gpm, 20 minutes

B. (1) DB-OP-2531, Steam Generator Tube Leak Abnormal Procedure
(2) 435 gpm, 34 minutes

C. (1) DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture
(2) 1 gpm, 20 minutes

D. (1) DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture
(2) 435 gpm, 34 minutes

Answer: **D**

Explanation/Justification:

A. Incorrect. Plausible because this is the procedure implement if the size of the leak is not estimated correctly, 1 gpm is the size of the leak assumed before the accident, and 20 minutes is the UFSAR assumption of when Operators take actions to depressurize and cooldown the RCS

B. Incorrect. Plausible because this is the procedure implement if the size of the leak is not estimated correctly

C. Incorrect. Plausible because this is the correct procedure to enter. 1 gpm is the size of the leak assumed before the accident, and 20 minutes is the UFSAR assumption of when Operators take actions to depressurize and cooldown the RCS r

D. Correct The MUT holds 30 gal/inch. A 2"/minute drop would indicate a 60 gpm leak. DB-OP-02531 identifies that if the SG Tube Leak is $\geq$50 gpm then DB-OP-02000 should be used. **Discussed with Keith we thought it would be an OK SRO only question based on the K/A. Brought UFSAR knowledge into the question in order to reach the SRO level of a K/A that on the surface seems unlikely to get to the SRO level.**

Sys #	**System**	**Category**	**KA Statement**
038	Steam Generator Tube Rupture (SGTR)	Generic	Knowledge of EOP entry conditions and immediate action steps.

K/A# 2.4.1 **K/A Importance** 4.8 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02000 pg 6
USAR Chapter 15 and Table 15.4.2-2

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

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

79. The Unit is operating at 100% power with all systems in normal alignment EXCEPT
- There was a hydrogen gas leak on the main generator.
- The hydrogen leak has been isolated.
- Main Unit hydrogen gas pressure is 40 psig and stable.

System Dispatch informs the Shift Manager that Emergency Operations (degraded grid) conditions exist, **AND** requests the Unit to maintain a power factor of 0.95 lagging **WITH** maximum permissible megawatts.

- The Command SRO has implemented DB-OP-02546 Degraded Grid

(1) IAW the guidance provided within DB-OP-02546 Degraded Grid, what actions are required for the Off-Site AC Sources?

(2) What will be the MAXIMUM permissible megawatt output for the Main Generator?

(Refer to the attached Estimated Capability Curves - Lead-Lag).

A. (1) Declare ONE Off-Site AC Source Inoperable
(2) ~880 megawatts

B. (1) Declare ONE Off-Site AC Source Inoperable
(2) ~800 megawatts

C. (1) Declare ALL Off-Site AC Sources Inoperable
(2) ~880 megawatts

D. (1) Declare ALL Off-Site AC Sources Inoperable
(2) ~800 megawatts

Answer: **C**

Explanation/Justification:

A. Incorrect. (1) Plausible since a degraded grid affects the operability of Off-Site Sources, but causes all sources to be inoperable, not just a single source. (2) correct megawatt limitation for 40 psig of H^2 gas pressure.

B. Incorrect. (1) Plausible since a degraded grid affects the operability of Off-Site Sources, but causes all sources to be inoperable, not just a single source.. (2) is Plausible megawatt limitation if candidate uses 30 psig H^2 gas pressure curve instead of 40 psig curve.

C. Correct. (1) IAW DB-OP-02546 Degraded Grid step 4.7 page 10. SRO must declare both Off-Site AC Sources inoperable and IAW step 4.8 refer to and apply the limitations of the appropriate generator capability curve. Off-Site AC Sources are declared inoperable because the grid voltage may not support TS requirements for operability. This requires the SRO to have knowledge of when and how to implement attachments of the abnormal procedures, and to coordinate these procedure steps.

D. Incorrect. (1) Correct IAW DB-OP-02546 Degraded Grid step 4.7 page 10. (2) incorrect but is Plausible megawatt limitation if candidate uses 30 psig H^2 gas pressure curve instead of 40 psig curve.

Sys #	System	Category	KA Statement
077	Generator Voltage and Electric Grid Disturbances	Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances:	Operating point on the generator capability curve

K/A# AA2.01 **K/A Importance** 3.6 **Exam Level** SRO

References provided to Candidate DB-PF-06703 Miscellaneous Operation Curves Rev. 19 page 82 **Technical References:** DB-PF-0670 pg 82; DB-OP-02546 step 4.7 pg 10

Question Source: New **Level Of Difficulty: (1-5)** 2

Question Cognitive Level: High - Application **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

80. The following plant conditions exist:

- A reactor trip occurs during a rapid shutdown due to an SGTR on SG 1.
- An SG 2 MSSV sticks partially open, causing a plant cooldown of 2 °F per minute.
- SG 1 pressure is 1000 psig and steady
- SG 2 pressure is 940 psig and lowering

Which of the following DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture actions is required to be taken **FIRST**?

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:

A. Correct. DB-OP-02000 TBD hierarchy states that Specific Rules have a higher priority than symptom and attachment 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. SRO only since the SRO must determine the hierarchy of implementation.

B. Incorrect. Plausible because SG 1 has the tube leak

C. Incorrect. Plausible because Section 8 would be used for the SGTR, but Section 7 is a higher priority and should be used first

D. Incorrect. Plausible because n 8 would be used for the SGTR, but Section 7 is a higher priority and should be used first

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Sys #	System	Category	KA Statement
BW/E05	Steam Line Rupture - Excessive Heat Transfer	Ability to determine and interpret the following as they apply to the (Excessive Heat Transfer)	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, Section 4.0 pg 20 & 22

Question Source: Bank 58982 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

81. The following initial plant conditions exist:
    - Reactor at 100 percent power.
    - A and B electrical busses are on Auxiliary Transformer 11.

    An event occurs that results in the following indications:
    - Busses A, B, C2, D2 and D1 indicate zero volts.
    - Bus C1 voltage is 4160 volts with AC 101 (DG1 to Bus C1) closed.
    - EDG 2 failed to start.
    - Tave is 552 °F and stable.
    - RCS pressure is 2155 psig and stable.
    - All other equipment operates as design.
    - All DB-OP-02000 Immediate Actions are complete.

    What procedure/procedure section is required to be implemented **NEXT**?

A. Specific Rule 2

B. Specific Rule 6

C. DB-OP-02521, Loss of AC Bus Power Sources

D. DB-OP-06316, Diesel Generator Operating Procedure

Answer: B

Explanation/Justification:

A. Incorrect. Plausible specific rule 2 is normally implemented before specific rule 6. However, since subcooling exists, specifc rule 2 is not applicable.

B. Correct, Loss of C2 and D2 would cause the reactor to trip, therefore, the correct routing would be to the immediate actions, Specific Rules 6 should be followed next to address the loss of power. Part 1 is RO only knowledge. Part 2 is SRO only since it requires knowledge of hierarchy implementation in the EOP network.

C. Incorrect. Plausible since immediate action of the LOAC procedure addresses loss of power, but routes you to DB-OP-2000 if C1 or D1 is energized. C1 is energized in the stem of the question.

D. Incorrect. Plausible because Immediate Actions should be performed and the EDG procedure provides directions to start, parallel, and load the EDG to supply D1 Essential Bus

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Sys #	System	Category	KA Statement
BW/E10	Post-Trip Stabilization	Generic	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

K/A# 2.2.44 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02000 TBD pg 7, 448, & 449

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

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

82. The following plant conditions exist:

- A serious Control Room fire has resulted in an evacuation of the Control Room
- All Immediate and Supplementary actions of DB-OP-02519, Serious Control Room Fire, were performed in the Control Room prior to evacuation.

Which one of the following is the desired SG level and the method used to control that level?

A. $\leq$ 49"
By manually controlling both SG AFW flow control valves

B. $\leq$ 49"
By using the governor to control AFPT speed

C. $\leq$ 124"
By manually controlling both SG AFW flow control valves

D. $\leq$ 124"
By using the governor to control AFPT speed

Answer: **D**

Explanation/Justification:

A. Incorrect. Plausible because this is the lower SFRCS control level, flow control valves cannot be controlled locally

B. Incorrect. Plausible because this is the lower SFRCS control level and correct controlling method

C. Incorrect. Plausible because this is the correct level, but incorrect controlling method, flow control valves cannot be controlled locally

D. Correct. IAW DB-OP-02519 Attachment 1, Shift Manager Actions Outside the Control Room, page 3 of 7 identifies conditions. SRO only since it requires knowledge of the procedure content beyond IMAs. At Davis Besse the performance of this attachment is for SRO shift managers only. The ROs are independently performing other attachments within the procedure.

Sys #	**System**	**Category**	**KA Statement**
067	Plant Fire On Site	Generic	Knowledge of abnormal condition procedures.

K/A# 2.4.11 **K/A Importance** 4.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02519 ATT. 1 pg 14 & 15

Question Source: Bank 29279 **Level Of Difficulty: (1-5)** 3

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

83. The following plant conditions exist:

- The plant was at 100% power.
- Toxic fumes have entered the control room.
- DB-OP-02508, Control Room Evacuation has been implemented.
- The reactor and turbine have been tripped.
- SFRCS has been actuated.
- Letdown is isolated.
- The standby Makeup Pump has been started.
- Local shutdown control from the Aux Shutdown Panel has been established.
- AFPT 1 and AFPT 2 red HSS lamps are on.
- Steam generator pressures are between 980 and 1000 PSIG and stable
- STEAM GEN 1 level is 49 inches and stable
- STEAM GEN 2 level is 40 inches and slowly lowering

IAW the appropriate attachment of DB-OP-02508 Control Room Evacuation:

What actions are required to control STEAM GEN 2 level?

A. Throttle Open AF 3869, AFW discharge cross-connect valve

B. De-energize modulating solenoid for AF 6451, AFP 2 Level Control valve.

C. De-energize AF 3872, AFPT 2 discharge and THEN Throttle Open

D. Raise AFPT 2 speed using HIS ICS38A, AFPT 2 GOV SPD CONT

Answer: B

Explanation/Justification:

A. Incorrect. Plausible since this is one of the options for raising SG-2 water level in procedure DB-OP-02519 Serious Control Room Fire This will not increase AFW flow to SG-2 since AFPT-2 is running. This action is also not directed to be performed in Attachment 1 of DB-OP-02508 Control Room Evacuation.

B. Correct. The SRO must analyze the conditions in the stem and select Attachment 1 of DB-OP-02508 Control Room Evacuation as the guiding procedural section. This section is unique to the Unit Supervisor SRO position. For conditions given in the stem, De-energizing ZC 6451, AFP 2 Modulating Solenoid Valve will cause the valve to fail open. AFPT 2 will already be at the high speed stop based on the SFRCS actuation, so AFW flow to SG-2 will increase.

C. Incorrect. Plausible since this is one of the options for raising SG-2 water level in procedure DB-OP-02519 Serious Control Room Fire. This will not increase AFW flow to SG-2 since this valve will already be open for the conditions stated in the stem.

D. Incorrect. Plausible since this would raise SG-2 level if AFPT 2 wasn't already at the high speed stop. Lowering AFPT-2 speed will eventually be necessary when level has risen as a result of De-energizing ZC 6451, AFP 2 Modulating Solenoid Valve.

Sys #	System	Category	KA Statement
068	Control Room Evacuation	Ability to determine and interpret the following as they apply to the Control Room Evacuation:	S/G level

K/A# AA2.01 **K/A Importance** 4.3 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02508 Control Room Evacuation Rev. 10 Attachment 1 step 6 page 10

Question Source: New **Level Of Difficulty: (1-5)** 4

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

84. The plant is operating at 100% power with all systems in normal alignment. The following VALID radiation alarm is received:

- RI 1998, Failed Fuel and Letdown Rad WARN

(1) What does this alarm indicate?

(2) The Tech Spec LCO limits on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed a small fraction of what guideline **AND** analyzed accident?

A. (1) Indicates RCS activity has increased to twice the background count rate.
(2) 10 CFR 100 dose guideline limits following an SGTR accident.

B. (1) Indicates RCS activity has increased to twice the background count rate.
(2) 10 CFR 100 dose guideline limits following a Main Steam Line Break accident.

C. (1) Indicates a fuel failure equivalent to 0.007% of all fuel elements.
(2) 10 CFR 20 dose guideline limits following an SGTR accident.

D. (1) Indicates a fuel failure equivalent to 0.007% of all fuel elements.
(2) 10 CFR 20 dose guideline limits following a Main Steam Line Break accident.

Answer: A

Explanation/Justification:

A. Correct. IAW DB-OP-02535 Rev. 08 page 11 the setpoint for the letdown Rad monitor WARN alarm is based on 2 times background and the setpoint for the HIGH alarm is based on the Davis-Besse historical highest failed fuel level of 0.007% failed fuel. IAW Tech Spec bases page B 3.4.16-1 The LCO limits on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following an SGTR accident. SRO only since the candidate will need knowledge of the TS bases to differentiate the analyzed accidents and the required 10CFR guidelines.

B. Incorrect. Wrong analyzed accident.

C. Incorrect. Wrong alarm bases and 10CFR guideline.

D. Incorrect. Wrong alarm bases, analyzed accident and 10CFR guideline.

Sys #	System	Category	KA Statement
076	High Reactor Coolant Activity	Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity:	RCS radioactivity level meter

K/A# AA2.03 **K/A Importance** 3.0 **Exam Level** SRO

References provided to Candidate None **Technical References:** Tech Spec bases page B 3.4.16-1, DB-OP-02535 Rev. 08 page 11

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

Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** CFR: 43(b)(2)

Objective:

85. What is the Core Flood Tank (CFT) Technical Specification Bases for

(1) The maximum nitrogen cover pressure limit?

(2) The maximum volume limit?

A. (1) Ensures that the amount of CFT inventory that is discharged through the break will not be larger than that predicted by the safety analysis.
(2) Ensures that the reactor will remain subcritical during the reflood stage of a large break LOCA.

B. (1) Ensures that the amount of CFT inventory that is discharged through the break will not be larger than that predicted by the safety analysis.
(2) Ensures the proper gas volume exists to ensure injection and the ability of the CFTs to fully discharge.

C. (1) Ensures that the sump pH will be maintained between 7.0 and 11.0 following a LOCA.
(2) Ensures that the reactor will remain subcritical during the reflood stage of a large break LOCA.

D. (1) Ensures that the sump pH will be maintained between 7.0 and 11.0 following a LOCA.
(2) Ensures the proper gas volume exists to ensure injection and the ability of the CFTs to fully discharge.

Answer: **B**

Explanation/Justification:

A. Incorrect. Part (1) is correct. Part (2) however is the TS bases for minimum boron concentration in the CFT.
B. Correct. IAW Tech Spec bases pages B 3.5.1-3 and B 3.5.1-4. SRO only in that it requires the candidate to differentiate terminologies used in the TS bases. All of the distractors are Tech Spec bases for the CFTs the SRO candidate must sort out the correct terminology to answer the question.
C. Incorrect. Part (1) is the CFT TS bases for maximum allowable boron concentration. Part (2) is the TS bases for minimum boron concentration in the CFT.
D. Incorrect. Part (1) is the CFT TS bases for maximum allowable boron concentration. Part (2) is correct.

Sys #	System	Category	KA Statement
BW/E08	LOCA Cooldown	Generic	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

K/A# 2.2.25 **K/A Importance** 4.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** Tech Spec bases pages B 3.5.1-3 and B 3.5.1-4

Question Source: New **Level Of Difficulty: (1-5)** 2

Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** CFR: 43(b)(2)

Objective:

86. The following plant conditions exist:
    - The plant is at 100% power.
    - Makeup Pump 1 is out of service.
    - All other systems are in normal alignment.

    The following alarms are then received:
    - (6-5-C) SEAL INJ FLOW LO
    - (6-6-C) SEAL INJ TOTAL FLOW
    - (4-2-E) PZR LVL LO

    The following conditions now exist:
    - Makeup Pump 2 discharge pressure is 0 psig
    - MU32, PZR LEVEL CONTROL, is at 100% demand
    - MU19, RCP SEAL INJ FLOW CONTROL, is at 100% demand
    - PZR level is 205 inches and slowly lowering

    Based on these conditions, which of the following actions are required?

A. Lineup and start piggyback operations and commence a reactor shutdown to Low Level Limits

B. Trip the Reactor GO TO DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture.

C. Trip the Reactor, Stop ALL RCPs GO TO DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture.

D. Place MU32 in HAND and adjust demand to obtain desired pressurizer Level.

Answer: A

Explanation/Justification:

A. Correct. IAW DB-OP-02512 supplementary actions (step 4.1.11). The SRO must assess the conditions, then select and implement the appropriate supplemental steps to mitigate the conditions posed in the stem of the question. PRZ level is being lost and the MU System should be responding to recover. However, MU pump 2 is not working so the SRO must decide what actions will be directed.

B. Incorrect. Plausible if PZR level is below 160 inches with Tavg at 582°F or PZR level 20 inches below normal. However neither of these conditions are present in the stem of the question.

C. Incorrect. Plausible since this would be required under these conditions if CCW flow was also lost the RCP seals.

D. Incorrect. Plausible since these would be the required actions for a failed PRZ level instrument that is inputting to the MU32 controller.

Sys #	System	Category	KA Statement
004	Chemical and Volume Control System	Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of PZR level (failure mode)

K/A# A2.02 **K/A Importance** 4.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02512 Rev.11 Makeup and Purification System Malfunctions step 4.1.11 RNO page 12

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

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

87. IAW with the Technical Specification bases for the Borated Water Storage Tank (BWST), what is a potential consequence of maintaining an improper boron concentration within the tank?

Improper boron concentrations could result in:

A. Excessive peak containment pressure during a large break LOCA.

B. A positive moderator temperature coefficient following a large break LOCA.

C. Excessive caustic stress corrosion of mechanical components and systems inside containment.

D. Insufficient cooling capacity of the ECCS when the transfer to the recirculation mode occurs.

Answer: C

Explanation/Justification:

A. Incorrect. Plausible since it is physically possible to plug the containment spray nozzles thereby leading to inadequate heat transfer inside the containment. However, this is not discussed in the TS bases as a potential consequence of improper boron concentration.

B. Incorrect. Plausible since excessively high boron concentrations could lead to a positive MTC. However, in the TS bases, a positive MTC is not discussed. The boron concentration's impact on SDM is addressed but not the impact on MTC.

C. Correct. IAW TS bases page B 3.5.4-1. SRO only since the candidate will need knowledge of the TS bases and to analyze the impacts of high or low boron concentration on CNMT sump PH. It is this impact that leads to the excessive caustic stress corrosion of mechanical components.

D. Incorrect. Plausible since this is the TS bases for the minimum volume requirement of the BWST.

Sys #	System	Category	KA Statement
006	Emergency Core Cooling System (ECCS)	Generic	Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

K/A# 2.2.25 **K/A Importance** 4.2 **Exam Level** SRO

References provided to Candidate None **Technical References:** TS bases page B 3.5.4-1

Question Source: New **Level Of Difficulty: (1-5)** 2

Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** CFR: 43(b)(2)

Objective:

88. The plant is operating at 100% power with all systems in normal alignment.
    - A Pressurizer Code Safety valve PARTIALLY OPENS and cannot be closed.
    - All systems respond as designed.
    - The control room crew has entered DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture.

    The following plant conditions currently exist:
    - RCS pressure is 1500 psig and stable
    - (9-4-A) VAC SYS DISCH RAD HI is in alarm
    - (12-1-A) MN STM LINE 1 RAD HI is in alarm
    - RCS subcooling is 15 °F and stable
    - CTMT pressure is 16 psia and slowly rising

    Based on the event(s) in progress and the current plant conditions:

    (1) Which SFAS signals will have actuated?
    (2) As Command SRO, which section of DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture are you required to address **FIRST**?

A. (1) Level 1 and 2 ONLY.
(2) Lack Of Adequate Subcooling Margin

B. (1) Level 1 and 2 ONLY.
(2) Steam Generator Tube Rupture

C. (1) Level 1, 2 and 3.
(2) Lack Of Adequate Subcooling Margin

D. (1) Level 1, 2 and 3.
(2) Steam Generator Tube Rupture

<u>Answer</u>: A

Explanation/Justification:

A. Correct. IAW Bases and Deviation Document for DB-OP-02000 Rev. 18 page 18; DB-OP-02000 Rev. 25. Part 1 is RO knowledge. Part 2 is SRO only since it involves knowledge of diagnostics that involve transitions to the appropriate section of the EOPs. The SRO must recognize that Both the SGTR and lack of subcooling sections are applicable and decide which section will be addressed FIRST.

B. Incorrect. Part 1 is correct. Part 2 is applicable but not the first required section to be addressed.

C. Incorrect. Part 1 is not correct since level 3 should not have actuated. Part 2 is correct.

D. Incorrect. Part 1 is not correct since level 3 should not have actuated. Part 2 is applicable but not the first required section to be addressed.

Sys #	System	Category	KA Statement
013	Engineered Safety Features Actuation System (ESFAS)	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 None **Technical References:** DB-OP-02000 TBD page 7; DB-OP-02000 page 2 index

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

Question Cognitive Level: High - Analysis **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

89. The plant is operating at 35% power with all systems in normal alignment for this power level.

The following alarms are simultaneously received:

- (11-1-C) SW PMP 1 STRNR DISCH PRESS LO
- (11-2-C) SW PMP 2 STRNR DISCH PRESS LO
- (11-1-B) CCW HX 1 OUTLET TEMP HI
- (11-3-F) TPCW HX OUTLET TEMP HI
- (11-1-D) FORE BAY LVL <562 FT
- (11-4-D) TRAVELING SCREEN TRBL

The following Control Room indications exist:

- All Control Room Service Water pump status lights are green
- All CRD stator temperatures are 185 °F and slowly rising
- All RCP Seal Out temperatures are 185 °F and slowly rising
- All RCP Motor Stator temperatures are 275 °F and slowly rising
- All Main Generator Cold Gas temperatures are 115 °F and slowly rising

Based on these conditions, which of the following actions are required?

A. Trip the Turbine **THEN** GO TO DB-OP-02500, Turbine Trip.

B. Trip the Turbine **THEN** Trip the Reactor THEN GO TO DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture.

C. Trip the Reactor **THEN** GO TO DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture.

D. Trip the Reactor **THEN** Trip all RCPs THEN GO TO DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture.

Answer: D

Explanation/Justification:

A. Incorrect. Plausible since this would be the required action if the CRD motors were not above the threshold value of 180°F

B. Incorrect. Plausible if candidate desires to remove secondary heat load to protect primary heat loads.

C. Incorrect. Plausible since none of the RCP parameters are above their setpoints for tripping RCPs.

D. Correct. IAW DB-OP-02511 step 4.3.10 the RX and RCPs are required to be tripped since CRD stator temps are >180°F. The candidate must prioritize and interpret the significance of all of the service water alarms, which is indicative of a total loss of all SWS (ROs would be expected to recognize this AB entry as well). SRO only since it requires the candidate to know the content of the appropriate procedure section and select the required actions and procedure for recovery. SRO must have supplemental action knowledge of the trip criterion for each of the parameters listed.

Sys #	System	Category	KA Statement
076	Service Water System (SWS)	Generic	Ability to prioritize and interpret the significance of each annunciator or alarm.

K/A# 2.4.45 **K/A Importance** 4.3 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02511 page 46 step 4.3.10

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

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

90. The plant is operating at 100% Power with all systems in normal alignment.

- Maintenance is planning to make multiple CTMT entries to replace the cables that tie the 6 trisodium phosphate baskets together.

Which of the following is a requirement in order to maintain CONTAINMENT AIRLOCK OPERABILITY during this maintenance period?

A. Complete DB-PF-03290, CONTAINMENT PERSONNEL AND EMERGENCY AIR LOCK DOORS INTERLOCK TEST on the affected airlock following each entry.

B. Complete DB-PF-03290, CONTAINMENT PERSONNEL AND EMERGENCY AIR LOCK DOORS INTERLOCK TEST on the affected airlock at the conclusion of the Maintenance Activity.

C. Complete DB-PF-03291, CONTAINMENT PERSONNEL AND EMERGENCY AIRLOCKS SEAL LEAKAGE TEST within 7 days of opening the air lock.

D. Complete DB-PF-03291, CONTAINMENT PERSONNEL AND EMERGENCY AIRLOCKS SEAL LEAKAGE TEST on the affected airlock following each entry.

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because the airlock would be operated and the interlock prevents both doors from being open at the same time.

B. Incorrect. because the airlock would be operated and the interlock prevents both doors from being open at the same time.

C. Correct. DB-PF-03291 and TS 3.6.2

D. Incorrect. Seal leakage test is only required once every seven days IAW with DB-PF-03291

Sys #	System	Category	KA Statement
103	Containment System	Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Necessary plant conditions for work in containment

K/A# A2.02 **K/A Importance** 3.2* **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-PF-03291 and TS 3.6.2

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

Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** CFR: 43(b)(2)

Objective:

91. The following plant conditions exist:

- The reactor has tripped.
- SG 1 Pressure = 780 psig SG 2 Pressure = 775 psig
- SG 1 Level = 124 inches SG 2 Level = 125 inches
- RCS Pressure = 1200 psig Incore Thermocouples = 572 °F
- Containment Pressure = 22 psia C1 and D1 are powered from Off-site power
- EDG 1 failed to start.

Which of the following provides the correct directions for operating, **Makeup, High Pressure Injection**, and **Low Pressure Injection** Systems for this event?

A. **Makeup** - Maintain full injection flow. Throttling MU flow is not required.
High Pressure Injection - Checking HPI flow balance is not required. Maintain full injection flow.
Low Pressure Injection - Open Piggyback Valve to supply MU and HPI.

B. **Makeup** - Maintain full injection flow. Throttling MU flow is not required.
High Pressure Injection - Check HPI Train 1 flow balance. Throttle HPI as necessary to flow balance HPI injection flow.
Low Pressure Injection - Close the Piggyback Valve. Supplying MU to HPI is not permitted under these plant conditions.

C. **Makeup** - Throttle #1 Makeup Pump to limit flow to 275 gallons per minute.
High Pressure Injection - Check HPI Train 1 flow balance. Throttle HPI as necessary to flow balance HPI injection flow.
Low Pressure Injection - Open Piggyback Valve to supply MU and HPI.

D. **Makeup** - Throttle #1 Makeup Pump to limit total flow to 275 gallons per minute.
High Pressure Injection - Checking HPI flow balance is not required. Maintain full injection flow.
Low Pressure Injection - Close Piggyback Valve. Supplying MU and HPI is not permitted under these plant conditions.

Answer: A

Explanation/Justification:

A. Correct. Based on RCS temperature and pressure, Subcooling Margin is less than 20 °F. As a result, a Loss of Subcooling Margin (SCM) exists. Specific Rule 3 governs the Makeup, High Pressure Injection and Low Pressure Injection Systems during a Loss of SCM. Makeup system is not throttled when a loss of SCM exist. HPI and LPI are operated a full capacity. Since offsite power remains available, both HPI trains are in service. As a result, HPI flow Balancing is not required. SRO since it requires specific knowledge of procedure content.

B. Incorrect. Plausible if the candidate does not recognize that with offsite power, both HPI trains are available and HPI flow balancing is not required.

C. Incorrect. Plausible is the candidate does not recognize that Specific Rule 3 limits for Makeup flow are not required when Subcooling Margin is lost. Makeup flow should be maximized.

D. Incorrect. Plausible is the candidate does not recognize that Specific Rule 3 limits for Makeup flow are not required when Subcooling Margin is lost. Makeup flow should be maximized.

Sys #	System	Category	KA Statement
002	Reactor Coolant System (RCS)	Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of coolant inventory

K/A# A2.01 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02000 Specific Rule 3 pg 239, and Table 2. pg 410

Question Source: Bank 76729 **Level Of Difficulty: (1-5)** 4

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

92. The plant is operating at 100% power when the following annunciators are received:

- (5-1-E) CRD LCO
- (5-2-E) CRD ASYMMETRIC ROD

The following plant conditions are noted:

- Reactor Power is stable at 100% power.
- All Control Rod Group 6 100% lights are lit.
- Absolute Position indication for Rods 6-1 & 6-5 read 0%.
- Relative Position indication for all Group 6 Rods read 100%.

Based on these conditions, what actions are required?

A. Contact I & C to investigate the rod position malfunction for Rods 6-1 and 6-5. Reactor Power may remain at 100% power.

B. Contact I & C to investigate the misaligned Rods 6-1 and 6-5. Reduce Reactor Power to 60%.

C. Contact I & C to investigate the dropped Rods 6-1 and 6-5. Reduce Reactor Power to 50%.

D. Manually trip the reactor due to multiple dropped rods.

Answer: A

Explanation/Justification:

A. Correct. DB-OP-02516, CRD Malfunction Section 2.4, Symptoms pg 8
B. Incorrect. Plausible for a misaligned control rods with 4 RCPs in service per TS 3.1.4.
C. Incorrect. Plausible because DB-OP-02516, CRD Malfunctions, power is reduced to 50% for a dropped Rod.
D. Incorrect. Plausible for multiple dropped rods.

Sys #	System	Category	KA Statement
014	Rod Position Indication System (RPIS)	Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of LVDT

K/A# A2.06 **K/A Importance** 3.0* **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02516, CRD Malfunction Section 2.4, Symptoms pg 8

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

Question Cognitive Level: High - Analysis **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

93. The Plant was operating at 100% power with all systems in a normal alignment
   - A total loss of instrument air occurred
   - DB-OP-02528, Loss of Instrument Air has been entered
   - The reactor has been tripped and SFRCS has been actuated
   - RCP seal return flow has been restored IAW DB-OP-02528, Attachment 5 RCP Seal Return Restoration.

   How will this restoration of RCP seal return flow impact Technical Specification 3.6.3, Containment Isolation Valves for the RCP seal return penetration?

   The RCP seal return penetration is_______________________________________.

A. INOPERABLE; an SFAS level 3 signal will close the INSIDE isolation valves but the OUTSIDE isolation valve will remain open

B. INOPERABLE; an SFAS level 3 signal will NOT close the INSIDE or OUTSIDE isolation valves

C. OPERABLE; an SFAS level 3 signal will close the INSIDE and OUTSIDE isolation valves

D. OPERABLE; an SFAS level 3 signal will close the INSIDE isolation valves and the dedicated operator will close the OUTSIDE isolation valve

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

Explanation/Justification:

A. Correct. IAW DB-OP-02528 Attachment 5; TS 3.6.3
B. Incorrect. The inside valve is an MOV and will respond to the SFAS level 3 signal.
C. Incorrect. Plausible since the inside valve will respond and the candidate does not recognize that hand jacking will defeat the SFAS actuation.
D. Incorrect. Plausible if candidate believes that stationing an operator at the valve is a substitute for automatic actuation of the valve.

Sys #	**System**	**Category**	**KA Statement**
079	Station Air System (SAS)	Generic	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

K/A# 2.1.7 **K/A Importance** 4.4 **Exam Level** SRO
References provided to Candidate None **Technical References:** DB-OP-02528 Attachment 5; TS 3.6.3
Question Source: New **Level Of Difficulty: (1-5)** 2
Question Cognitive Level: High - Application **10 CFR Part 55 Content:** CFR: 43(b)(2) & (5)
Objective:

94. An approach to criticality is being performed in accordance with DB-OP-06912, APPROACH TO CRITICALITY. Current plant status is as follows:

- The reactor is subcritical
- CRD Safety Group 1 is 100 percent withdrawn and at the OUT LIMIT
- Control Rod Safety Groups 2 through 4 are withdrawn to the OUT LIMIT.
- Control Rod Regulating Groups 5 through 7 are fully inserted.
- While attempting to withdraw Control Rod Regulating Group 5 a problem is encountered with the Control rod drive system. Control Rod Regulating Group 5 will NOT withdraw.

Maintenance has been contacted, and reports that the problem is limited to Control Rod Regulating Group 5 AND it will be several hours before they can repair the problem.

As Shift Manager and IAW the guidance provided in DB-OP-06912, APPROACH TO CRITICALITY what direction are you required to give the crew regarding the continued approach to criticality?

A. Suspend all rod motion.

B. Manually trip the reactor.

C. Manually Insert ONLY Control Rod Groups 4 through 2.

D. Manually Insert Control Rod Groups 4 through 1.

Answer: C

Explanation/Justification:

A. Incorrect. Plausible since this would be a normal response to rod control problems. However, given the conditions in the stem where an approach to criticality is being made, DB-OP-06912 provides direction to insert groups 2-4. Also, the rod control problem is limited to group 5.
B. Incorrect. Plausible since this would be an ultra conservative action. However, this is not what DB-OP-06912 directs.
C. Correct. IAW DB-OP-06912, Approach To Criticality page 13 NOTE 4.21. SRO only since the candidate will need to assess the conditions in the stem and select the appropriate procedural section that provides the required recovery actions. The NOTE preceding step authorizes the Shift manager ONLY as the individual responsible for determining what constitutes a “delay”. Since the delay will be > 1 hour, the SM is then required to direct the crew to manually insert CR groups 2-4.
D. Incorrect. Inserting CRD Safety group 1 is not required.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of procedures, guidelines, or limitations associated with reactivity management.

K/A# 2.1.37 **K/A Importance** 4.6 **Exam Level** SRO
References provided to Candidate None **Technical References:** DB-OP-06912 pg 13 NOTE and step 4.21
Question Source: New **Level Of Difficulty: (1-5)** 2
Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** CFR: 43(b)(5)
Objective:

95. During refueling operations, the MAIN BRIDGE OPERATOR (MBO) is experiencing difficulties with the Rod Hoist. In an effort to troubleshoot the problem, he suggests activating the TS-6 ROD HOIST OVERLOAD BY-PASS interlock and operating the hoist with a "dummy" load.

IAW the guidance provided within DB-NE-06308, Main Fuel Handling Bridge Operating Procedure:

Who (by title) is permitted to authorize the bypassing of the TS-6 ROD HOIST OVERLOAD BY-PASS interlock?

(Assume the suggested troubleshooting evolution is NOT specifically addressed within the procedures.)

A. Fuel Handling Director

B. Field Supervisor

C. Unit Supervisor

D. Outage Director

<u>Answer</u>: A

Explanation/Justification:

A. Correct. IAW DB-NE-06308. SRO only since it is the responsibility of the fuel handling director (refuel floor SRO) or the SRO shift manager to authorize this evolution.

B. Incorrect. Plausible since the Field Supervisor is an SRO and available to respond to the Fuel handling bridge.

C. Incorrect. Plausible since the Unit Supervisor is the command SRO, and as such gives permission for most evolutions but NOT the bypassing of fuel handling bridge interlocks. ONLY the SM and Fuel handling director are authorized to permit this evolution

D. Incorrect. Plausible since this is the senior FENOC individual on shift during an outage.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of new and spent fuel movement procedures.

K/A# 2.1.42 **K/A Importance** 3.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-NE-06308 para 2.1.1 pg 4

Question Source: New **Level Of Difficulty: (1-5)** 4

Question Cognitive Level: Low - Memory **10 CFR Part 55 Content:** CFR: 43(b)(7)

Objective:

96. Which of the following describes the requirement for work start authorization of Work Orders for an order that has **NOT** been pre-authorized for Lead Work Group Supervisor authorization in accordance with NOP-WM-4300, Order Execute Process?

A. Lead Work Group Supervisor shall process order to work start authorization and sign/date hard copy order.

B. Lead Work Group Supervisor shall process order to work start authorization and an Operations SRO will sign/date hard copy order.

C. Operations SRO will review order and process order to work start authorization status and sign/date hard copy order.

D. Operations SRO will review order and process order to work start authorization status. Lead Work Group Supervisor will sign/date hard copy order.

Answer: C

Explanation/Justification:

A. Incorrect. This is the process for pre-authorized work orders.
B. Incorrect. This is the process for pre-authorized work orders, however the operations SRO does not sign/date the hardcopy.
C. Correct. IAW NOP-WM-4300, section 4.7.6. SRO only since it requires the SRO to know which procedure governs this activity and to have additional knowledge of the procedure's requirements for work start authorization.
D. Incorrect. The operations SRO must sign/date the hardcopy.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of maintenance work order requirements.

K/A# 2.2.19 **K/A Importance** 3.4 **Exam Level** SRO
References provided to Candidate None **Technical References:** NOP-WM-4300, section 4.7.6 pg 19
Question Source: New **Level Of Difficulty: (1-5)** 4
Question Cognitive Level: Low - Fundamental **10 CFR Part 55 Content:** CFR: 43(b)(5)
Objective:

97. You are the Shift Manager.

The following plant conditions exist:

- A large break LOCA has occurred and you have just declared a Site Area Emergency
- An operator contacts the Control Room and reports he is in the Auxiliary Building, and is experiencing severe chest pains; he then stops communicating.
- The dose rate at the Operator's location is estimated to be 150 Rem/hr.
- Two individuals are needed to respond to the operator.

What is the MAXIMUM time that the two individuals can remain in the area?

A. 4 minutes

B. 6 minutes

C. 10 minutes

D. 30 minutes

Answer: C

Explanation/Justification:

A. Incorrect. Plausible since this would be the limit for protecting valuable property.
B. Incorrect. Plausible if the student makes an error in calculating the time.
C. Correct. IAW RA-EP-02620 the established dose is 25 R. Therefore, 25R/150 R/hr = .167 hr X 60 min./hr = 10 min. SRO only since this task is for the emergency director to authorize this exposure. The SRO must analyze and interpret the data given in the stem and then select and apply the appropriate emergency dose guideline, from the Emergency Dose Control And Potassium Iodide Distribution procedure.
D. Incorrect. Plausible since this would be the correct time for the "old" guideline of 75 R.

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 pge 7 step 6.1.3.b.1
Question Source: New **Level Of Difficulty: (1-5)** 3
Question Cognitive Level: High – Analysis/Application **10 CFR Part 55 Content:** CFR: 43(b)(4)
Objective:

98. The Plant is operating at 100% power with all systems in normal alignment. Movement of Spent Fuel within the Spent Fuel Pool is in progress, to support Dry Fuel Storage.

A 0.01 g earthquake occurs resulting in the following plant conditions:

- All Control Room Annunciators are Lost
- **NO** RPS, SFAS or SFRCS Trip actuations occur
- The Plant remains at 100% power and is stable
- Computer systems remain **AVAILABLE**
- Damage to a Spent Fuel Assembly in the Spent Fuel Pool occurs
- Spent Fuel Area Radiation Monitor RE 8426 is reading 1500 mr/hr
- Spent Fuel Area Radiation Monitor RE 8427 is reading 950 mr/hr
- Fuel Handling Area Radiation Monitor RE 8417 is reading 900 mr/hr
- Fuel Handling Area Radiation Monitor RE 8418 is reading 950 mr/hr

If all of these conditions continue until T = 18 minutes, what is the highest Emergency Plan Classification **REQUIRED**, if any, at T = 18 minutes?

(Reference RA-EP-01500, Emergency Classification EAL Tables)

A. No E-Plan classification required

B. Unusual Event

C. Alert

D. Site Area Emergency

Answer C

Explanation/Justification:

A. Incorrect. Plausible if the fails to realize that the listed radiation monitors are used for EAL classifications.

B. Incorrect. Plausible if the candidate ONLY considers the radiation readings as unplanned (RU2) or only considers the earthquake indications (HU3)

C. Correct. IAW RA-EP-01500 RA2 Alert indicator #2 page 28. In order to obtain the correct answer the candidate must have knowledge of the fixed radiation monitors that provide indications for emergency action levels in the E-plan. Not all fixed radiation monitors are used for indications of EAL entry. This knowledge and application of the knowledge is an SRO function. SROs are expected to know which radiation monitors are used in the EALs and apply their readings to the appropriate classification. The SRO must recognize that the radiation monitor readings posed in the stem of the question are VALID indicators of required EAL entry.

D. Incorrect. Plausible with the loss of all annunciators for >15 minutes this would be a SAE if a significant transient were in progress and the computers were also unavailable.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

K/A# 2.3.15 **K/A Importance** 3.1 **Exam Level** SRO

References provided to Candidate RA-EP-01500, Emergency Classification EAL Tables only

Technical References: RA-EP-01500, RA2 #2 pg 28

Question Source: New

Level Of Difficulty: (1-5) 3

Question Cognitive Level: High - Application

10 CFR Part 55 Content: CFR: 43(b)(7)

Objective:

99. The Reactor has tripped. The following plant conditions are noted:

SG 1 Pressure = 780 psig
SG 2 Pressure = 775 psig
SG 1 Level = 124 inches
SG 2 Level = 125 inches
RCS Pressure = 1000 psig
Incore Thermocouples = 600 °F
Containment Pressure = 22 psia

Which section of DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture provides the appropriate mitigation strategy for the plant conditions noted above?

A. Section 5, Loss of Subcooling Margin

B. Section 6, Lack of Heat Transfer

C. Section 7, Overcooling

D. Section 9, Inadequate Core Cooling

Answer: **D**

Explanation/Justification:

A. Incorrect. Plausible because a loss of Subcooled Margin exists, however Inadequate Core Cooling (superheated incores) takes priority over Loss of Subcooling Margin.

B. Incorrect – Plausible because a lack of heat transfer exists, however Inadequate Core Cooling (superheated incores) takes priority over lack of heat transfer.

C. Incorrect. Plausible because Steam Generator Pressure is low which is an indication of overcooling. An overcooling does not exist based on core temperature. Inadequate Core Cooling (superheated incores) takes priority over Overcooling.

D. Correct – Superheating is indicated based on RCS Pressure vs Temperature. DB-OP-02000 Figure 2 indicated region 2 of Inadequate Core Cooling. Indication of superheated incore thermocouple temperature requires entry into Inadequate Core Cooling. SRO must have knowledge of the priority and the bases of the priorities in order to select the appropriate actions.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.

K/A# 2.4.23 **K/A Importance** 4.4 **Exam Level** SRO

References provided to Candidate None **Technical References:** DB-OP-02000 TBD pg 7

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

Question Cognitive Level: High - Comprehension **10 CFR Part 55 Content:** CFR: 43(b)(5)

Objective:

100. The plant tripped 40 minutes ago due to a 45 gpm tube leak in Steam Generator 2.
- An Unusual event was declared per EAL SU7 and notifications to the State of Ohio and counties were completed.
- DB-OP-02504, Rapid Shutdown was used to shutdown the plant.
- Reactor Coolant System (RCS) temperature is 350 °F.
- RCS pressure is 980 psig.

The tube leak has risen to 300 gpm in Steam Generator 2.

As the Emergency Director, re-evaluate the events and determine what, if any, change to the Emergency Classification is required.

(Reference RA-EP-01500, Emergency Classification EAL Tables)

A. No change required

B. Upgrade to FU1, Unusual Event

C. Upgrade to FA1, Alert

D. Upgrade to FS1, Site Area Emergency

Answer: C

Explanation/Justification:

A. Incorrect. Plausible because this EAL still applies, but the Alert is made because of the higher classification
B. Incorrect. Plausible if the candidate considers the tube rupture a loss of Containment
C. Correct. IAW EAL FA1
D. Incorrect. Plausible if the candidate considers the tube leak as a loss of RCS and Containment

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Sys #	**System**	**Category**	**KA Statement**
N/A	N/A	Generic	Knowledge of the emergency action level thresholds and classifications.

K/A# 2.4.41 **K/A Importance** 4.6 **Exam Level** SRO

References provided to Candidate RA-EP-01500, Emergency Classification EAL Tables only

Technical References: RA-EP-01500, pg 25, FA1 Potential Loss of RCS – RCS Leakrate

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

Question Cognitive Level: High - Application **10 CFR Part 55 Content:** CFR: 43(b)(7)

Objective: