

Davis Besse NRC Written Exam (Dec. 2011) Rev. 1

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

Sys #	System	Category	KA Statement
007	Rx Trip Stabilization-Recovery	Knowledge of the reasons for the following as they apply to a reactor trip:	Actions contained in EOP for reactor trip
K/A#	EK3.01	K/A Importance	4.0
Exam Level	RO	Technical References:	DB-OP-02000 TBD, pg 50
References provided to Candidate	None	Level Of Difficulty: (1-5)	2
Question Source:	Bank	10 CFR Part 55 Content:	(CFR 41.5 /41.10 / 45.6 / 45.13)
Question Cognitive Level:	Low - Memory		

Objective:

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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:			

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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; Difference 186 - 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

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:			

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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 HPIs and SA level 3 starts the LPIs. The previous running MU pump will continue to run because a LOOP has not occurred

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:			

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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
Level Of Difficulty: (1-5)	3	Question Cognitive Level:	High Comprehension
10 CFR Part 55 Content:	(CFR 43.5 / 45.13)	Objective:	

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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	Technical References:	DB-OP-02527 pg 80
References provided to Candidate	None	Level Of Difficulty: (1-5)	3
Question Source:	New	10 CFR Part 55 Content:	(CFR: 41.6 / 45.4)
Question Cognitive Level:	Low - Memory		
Objective:			

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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	Technical References:	DB-OP-06262, pg 9, Step 2.2.15
References provided to Candidate	None	Level Of Difficulty: (1-5)	3
Question Source:	New	10 CFR Part 55 Content:	(CFR: 41.10 / 43.2 / 45.12)
Question Cognitive Level:	Low - Memory		
Objective:			

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8. After reaching the post trip window, which of the following sets of post reactor trip responses is RC2, Pressurizer Spray Valve, leaking?
- A. RCS temperature is lowering, RCS pressure is lowering, and Pressurizer level is lowering.
 - B. RCS temperature is rising, RCS pressure is rising, and Pressurizer level is rising
 - C. RCS temperature is stable, RCS pressure is lowering, and Pressurizer level is rising
 - D. RCS temperature is stable, RCS pressure is lowering, and Pressurizer level is stable

Answer: D

Explanation/Justification:

- A. Incorrect. Plausible because the choices are a combinations of the parameters with one parameter moving in the wrong direction
- B. Incorrect. Plausible because the choices are a combinations of the parameters with one parameter moving in the wrong direction
- C. Incorrect. Plausible because the choices are a combinations of the parameters with one parameter moving in the wrong direction
- D. Correct. A leaking PZR spray valve will cause RCS pressure to lower without affecting temperature or level

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
References provided to Candidate		None	Exam Level
Question Source:	Bank	ANO-1 2010 NRC	RO
Question Cognitive Level:	High - Comprehension		Technical References:
Objective:			DB-OP-02513 Symptoms Step 2.2
			Level Of Difficulty: (1-5)
			3.25
			10 CFR Part 55 Content:
			(CFR 41.7 / 45.7)

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9. The plant is operating at 100% power with all systems in normal alignment.
- A valid automatic Reactor trip signal is received
 - The Reactor fails to trip and cannot be manually tripped
 - IAW the guidance provided in DB-OP-02000, RPS, SFAS, SFRCS Trip, OR SG Tube Rupture the crew begins Emergency Boration

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 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:			

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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

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

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	Technical References:	UFSAR pg 15.4-24a, and Fig 15.4.4-5
References provided to Candidate	None	Level Of Difficulty: (1-5)	3
Question Source:	New	10 CFR Part 55 Content:	(CFR 41.8 / 41.10 / 45.3)
Question Cognitive Level:	High - Comprehension		
Objective:			

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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 several 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

Answer: 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	Technical References:	DB-OP-02000 TBD Attach 25 pg 494
References provided to Candidate	None	Level Of Difficulty: (1-5)	4
Question Source:	Bank	79051 Modified by changing Tave to incore TCs in the stem and removing "large" from A and B	
Question Cognitive Level:	High - Comprehension	10 CFR Part 55 Content:	(CFR 41.8 / 41.10 / 45.3)

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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 orifice, Makeup 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	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:			

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13. The following plant conditions exist:

- The plant is operating at 100% power
- Emergency Diesel Generator (EDG)1 Monthly Surveillance Test,DB-SC-03070, is in progress
- EDG 1 is running
- AC101, EDG output breaker, is closed
- AC 110, Bus C1 To Bus C2 Tie Breaker, is closed

The following event occurs:

- All offsite power is lost
- AC110 opens

Based on these conditions, what will be the status of EDG 1 and how will the Operator control frequency?

EDG 1 will be in the _____(1)_____ mode and the operator will control frequency by adjusting the _____(2)_____.

- A. (1) isochronous
(2) EDG 1 SPD CTRL switch
- B. (1) isochronous
(2) DG1 VOLT REGULATOR control switch
- C. (1) droop
(2) EDG 1 SPD CTRL switch
- D. (1) droop
(2) DG1 VOLT REGULATOR control switch

Answer: C

Explanation/Justification:

- A. Incorrect. Plausible because the EDG would shift to isochronous if an SFAS signal was present
- B. Incorrect. Plausible because the EDG would shift to isochronous if an SFAS signal was present
- C. Correct. With AC110 open, electrical bus C2 isolated from electrical bus C1 and no SFAS signal, AC101 will remain closed and therefore the EDG will remain in the Droop mode of operation. The operator adjust **frequency** with the EDG speed switch
- D. Incorrect. Plausible because the EDG would be in the droop mode and the operator adjusts voltage with the voltage regulator control switch

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	Technical References:	DB-SC-03070, pg 19, pg 50 and Attachment 1
References provided to Candidate	None	Level Of Difficulty: (1-5)	4
Question Source:	New	10 CFR Part 55 Content:	(CFR 41.7 / 45.5 / 45.6)
Question Cognitive Level:	High - Comprehension		
Objective:			

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

The following events occur:

- A loss of off-site power occurs.
- #1 Emergency Diesel Generator Failed to start.
- #2 EDG started and reenergized D1 Bus.
- The Command SRO directs performance of DB-OP-02521, Attachment 5, Selective Battery Load Shedding.

Which of the following conditions will exist following performance of Attachment 5, Selective Battery Load Shedding?

- A. YAU will be manually transferred from Inverter YVA to Panel YAR to reduce load on Train 1 DC Batteries. YBU will remain powered via Train 2 DC Batteries.
- B. YAU will remain powered from Inverter YVA. YBU will remain powered from Inverter YVB. Selected loads will be removed from YAU AND YBU by opening procedure directed breakers
- C. Y1 will be manually transferred to alternate supply Rectifier YRF at Inverter YV3. This will allow Y1 to remain energized from #2 EDG. Y2 will automatically be fed from Train 2 DC power via Inverter YV2.
- D. Y1 will automatically be fed from Train 1 DC power via Inverter YV1. Y2 will automatically be fed from F12A bus via Inverter YV2.

Answer: D

Explanation/Justification:

- A. Incorrect. This answer is plausible because if off-site power is available, YAU would be transferred to YAR to reduce load and YBU would remain powered from Train 2 DC Batteries. On a loss of off-site power, YAR will be deenergized. Caution 2.a of Attachment 5 reminds operators not to transfer YAU to dead bus
- B. Incorrect. This answer is plausible because 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 AND YBU.
- C. Incorrect. This answer is plausible because Y1 and Y2 remain energized following a loss of all AC power, however cross connections of essential 120 volt AC power is not an element of the system design
- D. Correct. IAW DB-OP-02521, Attachment 5

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	Technical References:	OS-060 sh 2
References provided to Candidate	None	Level Of Difficulty: (1-5)	4
Question Source:	New	10 CFR Part 55 Content:	(CFR 41.7 / 45.5 / 45.6)
Question Cognitive Level:	High - Analysis		
Objective:			

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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

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	Technical References:	DB-OP-02537, pg 35
References provided to Candidate	None	Level Of Difficulty: (1-5)	3
Question Source:	Bank	10 CFR Part 55 Content:	(CFR 41.7 / 45.5 / 45.6)
Question Cognitive Level:	Low - Memory		
Objective:			

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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 35 °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) SW Train 2
(2) be cooled by Circ Water
- C. (1) SW Train 2
(2) have NO cooling
- D. (1) Backup Service Water Pump
(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 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.
- 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 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.

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	RO
References provided to Candidate		None	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:			

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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

Answer: 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	Technical References:	DB-OP-02528, pg 77
References provided to Candidate	None	Level Of Difficulty: (1-5)	2
Question Source:	Bank	33899	10 CFR Part 55 Content:
Question Cognitive Level:	High - Comprehension		(CFR: 43.5 / 45.13)
Objective:			

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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
References provided to Candidate		None	Exam Level
			RO
			DB-OP-02000 TBD, pg 92
			DB-OP-02000, pg 72
Question Source:	Bank	38987	Level Of Difficulty: (1-5)
Question Cognitive Level:		High - Comprehension	3
			10 CFR Part 55 Content:
			(CFR: 41.5 / 41.10, 45.6, 45.13)
Objective:			

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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	Technical References:	ICS Analog to Digital Drawings
References provided to Candidate	None	Level Of Difficulty: (1-5)	2.5
Question Source:	Bank	46162	10 CFR Part 55 Content:
Question Cognitive Level:	High - Analysis		(CFR 41.8 / 41.10 / 45.3)
Objective:			

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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:			

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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

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

Answer: 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, 15.5, 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:					

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22. A PZR level control malfunction occurred and has been corrected.
The following plant conditions **NOW** exist:

- The Reactor is at 100% rated power and stable
- Pressurizer level is at 170 inches
- MU32 is in AUTOMATIC
- One Makeup Pump is running
- RCS leakage is NORMAL

Which of the following describes the Pressurizer level response and its impact on the Pressurizer support systems if an inadvertent Reactor TRIP occurs?

(Assume No Operator Actions)

- A. PZR level remains on scale and PZR heaters continue to operate normally.
- B. PZR level remains on-scale and PZR heaters cutout on low level.
- C. PZR level goes off-scale low and PZR heaters cutout on low level.
- D. PZR level goes off-scale low and letdown isolates.

Answer: B

Explanation/Justification:

- A. Incorrect. Plausible if candidate incorrectly determines post trip pressurizer level based on normal pressurizer level of 220 inches. Expected normal post trip pressurizer level is approximately 70 inches – on scale and with pressurizer heaters above low level cut off.
- B. Correct. DB-OP-02512, Makeup System Malfunctions, Background Document, page 44 notes “Following a Reactor Trip from a Tave of 582°F to the post trip value of 550°F, is equivalent to approximately 160 inches of Pressurizer level.” A reduction of 160 inches will reduce indicated Pressurizer Level to 10 inches, which is still on scale, but below the pressurizer heater cut out of 40 inches.
- C. Incorrect. Plausible if candidate incorrectly determines the reduction in pressurizer level caused by post trip RCS Tave reduction.
- D. Incorrect. Plausible if candidate incorrectly determines the reduction in pressurizer level caused by post trip RCS Tave reduction.

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-02512, Makeup System Malfunctions Background Document.
Question Source:	Bank	29066	Level Of Difficulty: (1-5) 4
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content: (CFR 41.7 / 45.5 / 45.6)
Objective:			

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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 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	Technical References:	Operations Lesson Plan OPS-SYS-I502
References provided to Candidate	None	Level Of Difficulty: (1-5)	4
Question Source:	New	10 CFR Part 55 Content:	CFR 41.8 / 41.10 / 45.3)
Question Cognitive Level:	Low - Fundamental		
Objective:			

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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.

Answer: 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*
References provided to Candidate		None	Exam Level
Question Source:	Bank	30642	Technical References:
Question Cognitive Level:	Low - Fundamental		Level Of Difficulty: (1-5)
Objective:			10 CFR Part 55 Content:
			RO
			DB-OP-03011 Caution 4.3.15 and 16
			2
			(CFR: 43.5 / 45.13)

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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

IAW the guidance provided in DB-OP-01001, Administrative Control Of Containment Isolation Valves:

What is the status of DR 2012B?

DR 2012B is _____ as long as _____.

- A. Operable
its breaker is open **AND** secured in position under administrative control
- B. Operable
DR 2012A is capable of being closed by SFAS
- C. Inoperable
its breaker is open **AND** secured in position under administrative control
- 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:			

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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
References provided to Candidate		None	Exam Level
Question Source:	Bank	86431	Technical References:
Question Cognitive Level:	High - Comprehension		DB-OP-02532, Loss of NNI/ICS Power Page 30, Loss of NNI-Y AC
Objective:			Level Of Difficulty: (1-5) 3
			10 CFR Part 55 Content: (CFR: 41.5 / 41.10, 45.6, 45.13)

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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

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
References provided to Candidate		None	Exam Level
Question Source:	Bank	86551	Technical References:
Question Cognitive Level:	High - Comprehension		RO
Objective:			DB-OP-02538, pg 15 & 36 DB-OP-02000 Attachment 20, pg 363
			Level Of Difficulty: (1-5)
			3
			10 CFR Part 55 Content:
			(CFR: 41.7 / 45.7)

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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 loop 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:			

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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*
References provided to Candidate		None	Exam Level
Question Source:	Bank	79047	RO
Question Cognitive Level:	High - Comprehension		Technical References: DB-OP-02515 pg 51
Objective:			Level Of Difficulty: (1-5) 3
			10 CFR Part 55 Content: (CFR: 41.7 / 45.6)

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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.

Answer: 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 while 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

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:			

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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

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	Technical References:	DB-OP-02000 TBD pg 271
References provided to Candidate	None	Level Of Difficulty: (1-5)	2
Question Source:	Bank	37346	10 CFR Part 55 Content:
Question Cognitive Level:	Low - Fundamental		(CFR: 41.7)
Objective:			

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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 be accomplished 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 1530/1531 receives signals to go to a throttle position to ensure CS Pumps do not cavitate with the hotter Emergency Sump Water

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*
References provided to Candidate		None	Exam Level
Question Source:	Bank	75956	Level Of Difficulty: (1-5)
Question Cognitive Level:	High - Comprehension		10 CFR Part 55 Content:
Objective:			RO
			DB-OP-02000 pg 301 & 302 Attachment 7, Transferring LPI Suction to the Emergency Sump
			2
			(CFR: 41.2 to 41.9 / 45.7 to 45.8)

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33. The following plant conditions exist:
- A plant startup is in progress
 - Reactor Coolant System pressure is 40 psig
 - Pressurizer temperature is 240 °F
 - Pressurizer bubble is in the process of being drawn
 - Quench Tank pressure is 25 psig

The following event occurs:

- Selected RCS pressure input fails high

How will Quench Tank pressure and temperature respond to these conditions?

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
- 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

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
References provided to Candidate		None	Exam Level
Question Source:	New		RO
Question Cognitive Level:	High - Comprehension		Technical References:
Objective:			SD-039A, pg 70 (2-47)
			Level Of Difficulty: (1-5)
			4
			10 CFR Part 55 Content:
			(CFR: 41.5 / 45.7)

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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

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	Technical References:	DB-OP-06316 pg 79
References provided to Candidate	None	Level Of Difficulty: (1-5)	2.5
Question Source:	New	10 CFR Part 55 Content:	(CFR: 41.7 / 45.5)
Question Cognitive Level:	Low - Fundamental		
Objective:			

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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	Technical References:	SD 3A, 4160v, pg2-6 (30)
References provided to Candidate	None	Level Of Difficulty: (1-5)	4
Question Source:	New	10 CFR Part 55 Content:	(CFR: 41.7)
Question Cognitive Level:	High - Comprehension		
Objective:			

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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
 - Tavg 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 procedure 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	Technical References:	DB-OP-02513 symptoms
References provided to Candidate	None	Level Of Difficulty: (1-5)	2
Question Source:	New	10 CFR Part 55 Content:	(CFR: 41.10 / 43.5 / 45.13)
Question Cognitive Level:	High - Analysis		
Objective:			

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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 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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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) _____, which will initiate a/an _____ (3) _____.

- A. (1) SFRCS actuation
(2) ARTS actuation
(3) Reactor Trip signal
- B. (1) SFRCS actuation
(2) Reactor Trip signal
(3) ARTS actuation
- C. (1) Reactor Trip signal
(2) SFRCS actuation
(3) ARTS actuation
- D. (1) Reactor Trip signal
(2) ARTS actuation
(3) SFRCS actuation

Answer: A

Explanation/Justification:

- A. Correct. Tripping the MFPs will cause an SFRCS actuation on D/P. SFRCS will then initiate an ARTS signal which will send a signal to trip the reactor.
- B. Incorrect. The reactor trip signal will be generated by ARTS, it will not generate the ARTS actuation.
- C. Incorrect. The reactor trip signal is generated by ARTS, which is generated by the SFRCS actuation.
- D. Incorrect. The reactor trip signal is generated by ARTS, which is generated by the SFRCS actuation.

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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	High - Comprehension		RO
Objective:			DB-OP-02508 page 13 SD 010 pg 31 SD 013, pg 44 Fig 2.1-2, pg 47 Fig 2.1-3
			Level Of Difficulty: (1-5) 2.5
			10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13)

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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	Technical References:	DB-OP-06403, Att. 4, pg 57
References provided to Candidate	None	Level Of Difficulty: (1-5)	3
Question Source:	Bank	39026	10 CFR Part 55 Content:
Question Cognitive Level:	Low - Fundamental		(CFR: 41.7)
Objective:			

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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 an 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:	

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41. A steam line break has occurred in Containment.
- Containment temperature is 228 °F
 - Containment pressure is 20 psia.
 - All systems function as designed

Which ONE of the following indicates how Core Flood Tank (CFT) level indication will respond?

Indicated CFT level will be _____ .

- A. higher than actual tank level
- B. lower than actual tank level
- C. the same as actual tank level
- D. zero because it discharged to the RCS

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. The expansion of the reference leg will displace some reference leg inventory back to the core flood tank causing less pressure on the reference side of the differential pressure cell 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 affect reference leg and variable leg equally.
- D. Incorrect. Plausible if candidate believes Main Steam Induce Overcooling is not terminated by SFRCS causing the RCS to cool to approximately 228 °F and cause a full discharge of the Core Flood Tank.

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	Technical References:	DB-OP-02000 Attachment 9 for Pressurizer – impact on Core flood would be similar.
References provided to Candidate	None		
Question Source:	Bank	35585	Level Of Difficulty: (1-5) 2
Question Cognitive Level:	Low - Fundamental	10 CFR Part 55 Content:	(CFR: 41.7 / 45.6)
Objective:			

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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 Component Cooling Water Pump 2 is running
- 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, place Governor Mode Selector Switch to ELECT, allow SFAS to restart EDG in isochronous mode

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
- 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	RO
References provided to Candidate	None	Exam Level	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:			

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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	Technical References:	DB-OP-02000, rev. 25 Att. 7 page 300
References provided to Candidate	None	Level Of Difficulty: (1-5)	3
Question Source:	New	10 CFR Part 55 Content:	(CFR: 41.7)
Question Cognitive Level:	Low - Memory		
Objective:			

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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 1 did 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 1 did 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:					

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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:			

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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
Level Of Difficulty: (1-5)	3	Question Cognitive Level:	High - Comprehension
10 CFR Part 55 Content:	(CFR: 41.7 / 45.5)	Objective:	

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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

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

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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

Answer: B

Explanation/Justification:

- A. Incorrect. Plausible if candidate does not know there is a restricting orifice 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:			

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49. The plant is operating at 100% power in a normal lineup
- A and B, 13.8kv electrical busses, are being supplied from the Aux Transformer
 - All B bus, 13.8 kv electrical bus, LOAD breaker indications have been lost

The following event occurs:

- The reactor trips

Which of the following describes the B, 13.8 kv electrical bus response?

The B bus, will ____ (1) ____ and LOAD breaker indication will ____ (2) ____ .

- A. (1) fast dead transfer to the Startup Transformer 02
(2) be unavailable
- B. (1) fast dead transfer to Startup Transformer 02
(2) indicate breaker status
- C. (1) de-energize
(2) be unavailable
- D. (1) remain powered from Aux Transformer
(2) be unavailable

Answer: A

Explanation/Justification:

- A. Correct – B bus load breakers are supplied from DBP breaker 02. B bus source breakers are supplied from DBN breaker 02. When the reactor trips, power is available to allow transfer from Aux to Startup Transformer 02. The reactor trip and transfer to Startup 02 will not restore power to B bus load breakers.
- B. Incorrect – plausible because B bus load breakers are supplied from DBP breaker 02. B bus source breakers are supplied from DBN breaker 02. When the reactor trips, power is available to allow transfer from Aux to Startup Transformer 02. Plausible if candidate assumes transfer to off-site power will restore load breaker power.
- C. Incorrect – Plausible if candidate assumes fast transfer to Startup Transformer will not occur without power to B bus load breakers.
- D. Incorrect – Plausible if candidate assumes Aux power supply will still be powered following a reactor trip.

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-02538, Loss of D2P and DBP attachment 8 page 3 and DB-OP-02540, Attachment 1
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:			

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

The following event then occurs:

- Loss of Power to the battery chargers supplying DC MCC 1 **AND** DC MCC 2

Following battery load shedding, the batteries will be able to supply power to the essential loads for _____(1)_____.

If the battery's discharge rates are further DECREASED, this will _____(2)_____ the time available for the batteries to supply power to the essential loads.

- A. (1) one hour
(2) decrease
- B. (1) two hours
(2) decrease
- C. (1) one hour
(2) increase
- D. (1) two hours
(2) increase

Answer: C

Explanation/Justification:

- A. Incorrect. Decreasing discharge rates will increase the available time.
- B. Incorrect. Batteries are designed for only 1 hour and decreasing discharge rates will increase the available time.
- C. Correct. IAW DB-OP-02521 Rev 17, page 63 and Electrical Theory.
- D. Incorrect. Batteries are designed for only 1 hour.

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:			

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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 shut down be accomplished?

(3) 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)

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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

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 than 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 than 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	Exam Level
		3.6	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:			

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53. The following plant conditions exist:
- The plant tripped from 100% power
 - SFAS Levels 1, 2 and 3 have actuated due to a loss of coolant accident
 - Service Water Pump 2 failed to start

What will be the impact of this Service Water Pump failure?

- A. The Control Room air temperature may exceed design limits
- B. Emergency Diesel Generator 2 will trip on high temperature
- C. Component Cooling Water Pump 2 will trip on high temperature
- D. Emergency Core Cooling System Room 2 air temperature may exceed design limits

Answer: D

Explanation/Justification:

- A. Incorrect. The Control Room Emergency Ventilation System consists of two 100% capacity trains. Either train can remove heat necessary to ensure Control Room temperatures remain with design limits.
- B. Incorrect. Plausible since the EDG has a high Jacket Water Temperature shutdown, but this shutdown is bypassed on EDG Safety Start.
- C. Incorrect. Plausible since CCW pumps used to get a high temperature trip signal, but this trip signal has been removed.
- D. Correct. ECCS Train 2 Room is cooled by Service Water Train 2. Loss of this cooling could result in the #2 ECCS Room exceeding design limits for room temperature.

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: USAR 9.2.1 pg 9.2-1 and Table 9.4-6 pg 9.4-28
Question Source:	Bank	55581	Level Of Difficulty: (1-5) 3
Question Cognitive Level:	Low - Memory		10 CFR Part 55 Content: (CFR: 41.5 / 45.5)
Objective:			

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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*
References provided to Candidate		None	Exam Level
Question Source:	New		RO
Question Cognitive Level:	High- Comprehension		Technical References: SD 001 pg 2-1 & 2-2
Objective:			Level Of Difficulty: (1-5) 4
			10 CFR Part 55 Content: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

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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 actuate

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*
References provided to Candidate		None	Exam Level
Question Source:	New		RO
Question Cognitive Level:	Low - Memory		Technical References:
Objective:			DB-OP-02000 Table 2 pg 409 -413 DBBP-TRAN-0034, pg 9
			Level Of Difficulty: (1-5)
			10 CFR Part 55 Content:
			3 (CFR: 41.5 / 43.5 / 45.3 / 45.13)

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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 That 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 and increase Reactor power to attempt to return Tave to setpoint

Answer: D

Explanation/Justification:

- A. Incorrect – RCS temperature is part of the BTU calculation, but it only provides an alarm.**
- 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 raising 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
References provided to Candidate		Exam Level	RO
		Technical References:	ICS Analog to Digital Drawing for Feedwater
Question Source:	Bank	39087	Level Of Difficulty: (1-5)
Question Cognitive Level:		High - Analysis	10 CFR Part 55 Content:
Objective:			4 (CFR: 41.5/45.5)

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57. Which of the following conditions requires the Operator to select Incore Thermocouples on the PAMP Tsat Meter when adequate SCM exists?
- A. ONLY 1 RCP running
 - B. ONLY 1 RCP running per loop
 - C. MU/HPI/PORV Cooling
 - D. Natural Circulation

Answer: C

Explanation/Justification:

- A. Incorrect: plausible because there is no force flow in one RCS Loop
- B. Incorrect plausible because some loop flow is reversed through the idled RCP
- C. Correct, there is no flow pass the RCS Th RTD, therefore the operator is required to select the Incore thermocouple,
- D. Incorrect: plausible because no force flow

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	Technical References:	DB-OP-02000 pg 18
References provided to Candidate	None	Level Of Difficulty: (1-5)	3
Question Source:	New	10 CFR Part 55 Content:	(CFR: 41.10 / 45.12)
Question Cognitive Level:	Low - Fundamental		
Objective:			

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58. What is the power supply for Emergency Ventilation Fan 1?

- A. E12A
- B. E23A
- C. F12A
- D. F23A

Answer: A

Explanation/Justification:

- A. Correct
- B. Incorrect. Plausible because the Radwaste Are Exhaust Fan 1 is supplied from E23A.
- C. Incorrect. Plausible because Train 2 of EVS is supplied from F12A
- D. Incorrect. Plausible because the Radwaste Are Exhaust Fan 2 is supplied from F23A.

Sys #	System	Category	KA Statement
027	Containment Iodine Removal System (CIRS)	Knowledge of bus power supplies to the following:	Fans
K/A#	K2.01	K/A Importance	3.1*
Exam Level	RO	References provided to Candidate	None
Technical References:	DB-OP-06504, pg 48	Question Source:	New
Level Of Difficulty: (1-5)	3	Question Cognitive Level:	Low - Memory
10 CFR Part 55 Content:	(CFR: 41.7)	Objective:	

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59. Containment Wide Range Pressure indication (0-200 psia) is located on which ONE of the following control panels?
- A. Post Accident Monitoring Panel
 - B. Engineered Safety Features Panel
 - C. SFAS Channels 1, 2, 3 and 4
 - D. AUX Shutdown Panel

Answer: A

Explanation/Justification:

- A. Correct: CTMT wide range pressure indication is required per TS
- B. Incorrect; plausible because the low range CTMT pressure is located here
- C. Incorrect; plausible because the low range CTMT pressure is located here
- D. Incorrect; plausible because this panel is used to control plant systems when the CTRM is evacuated

Sys #	System	Category	KA Statement
028	Hydrogen Recombiner and Purge Control System (HRPS)	Ability to manually operate and/or monitor in the control room:	Location and interpretation of containment pressure indications
K/A#	A4.02	K/A Importance	3.7*
Exam Level	RO	Technical References:	TS 3.3.17 Table 1 pg 3.3.17-4
References provided to Candidate	None	Level Of Difficulty: (1-5)	2.5
Question Source:	Bank	30707	10 CFR Part 55 Content:
Question Cognitive Level:	Low - Memory		(CFR: 41.7 / 45.5 to 45.8)
Objective:			

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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.
- CTMT Purge is in service on Containment.

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. The Emergency Ventilation System will have to be aligned to the Fuel Handling area in order to continue fuel transfers.
- B. Fuel handling operations in the Spent Fuel Pool area must be suspended until the personnel air lock is closed.
- C. Fuel Handling operations must be stopped until ALL Equipment Hatch bolts are installed.
- D. Spent Fuel Pool level could rise high enough to overflow the Spent Fuel Pool

Answer: D

Explanation/Justification:

- A. Incorrect. Plausible because EVS would take a suction in the Aux Building
- B. Incorrect. Plausible because with the Equipment Hatch on and CTMT Purge Supply running, high air flow rates out the personnel air lock would occur. This area is outside the negative pressure area.
- C. Incorrect. Plausible because install all Containment Hatch bolts is performed for containment integrity and would limit air leakage out of containment in this situation
- D. Correct. With the supply fan still running and the Equipment hatch installed, the pressure in containment will rise. This slight increase in pressure could be enough to force water thru the transfer tube which will cause level in the SFP to rise and possibly overflow. 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 - Comprehensive		10 CFR Part 55 Content:	(CFR: 41.7 / 45.6)
Objective:			

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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

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)	Ability to monitor automatic operation of the Spent Fuel Pool Cooling System including:	Temperature control valves
K/A#	A3.01	K/A Importance	2.5*
References provided to Candidate		None	Exam Level
			RO
			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:	Low - Fundamental	10 CFR Part 55 Content:	(CFR: 41.7 / 45.5)
Objective:			

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62. The plant is operating at 100% power.
- Moving fuel in the Spent Fuel Pool is in progress.

The following event occurs:

- RE 5403, Fuel Handling Area Exhaust Fan Inlet Radiation Monitor, A, B and C fails HIGH.

What will be the plant response, as a result of this HIGH failure condition?

- A. Fuel handling supply and exhaust fans trip,
Station EVS starts automatically,
CV 5025 and CV 5024 EVS dampers from fuel handling close
- B. Fuel handling supply and exhaust fans stay running,
Station EVS starts automatically,
CV 5025 and CV 5024 EVS dampers from fuel handling close.
- C. Fuel handling supply and exhaust fans stay running,
Station EVS stays shutdown,
CV 5025 and CV 5024 EVS dampers from fuel handling remain open
- D. Fuel handling supply and exhaust fans trip,
Station EVS stays shutdown,
CV 5025 and CV 5024 EVS dampers from fuel handling remain closed.

Answer: C

Explanation/Justification:

- A. Incorrect: FH fans will trip and Station EVS will start, but the EVS dampers will remain open
- B. Incorrect: plausible because EVS starts automatically
- C. Correct Answer
- D. Incorrect: plausible because FH fans will trip RE 5403 will trip

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
Level Of Difficulty: (1-5)	3.5	Question Cognitive Level:	Low - Memory
10 CFR Part 55 Content:	(CFR: 41.7 / 45.7)	Objective:	

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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 O₂/H₂ HI

Waste Gas Surge Tank local O₂ and H₂ values are as follows:

- O₂ is 2.5%
- H₂ 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 is _____(1)_____ than the flammability limit 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. H₂ concentrations above 4% are greater than the flammability 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. H₂ concentrations above 4% are greater than the flammability 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. H₂ concentrations above 4% are greater than the flammability 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. H₂ concentrations above 4% are greater than the flammability 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:			

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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:			

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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:			

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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

Answer: 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
References provided to Candidate		DB-PF-06703 CC1.1	Exam Level RO
Question Source:	New		DB-PF-06703 pg 11 & 12, CC1.1 Rev.19
Question Cognitive Level:	High - Application		Level Of Difficulty: (1-5) 3
Objective:			10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.12)

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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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	Low- Fundamental		Level Of Difficulty: (1-5)
Objective:			10 CFR Part 55 Content:
			RO
			NOBP-LP-3008 rev. 06 page 28
			3
			(CFR: 41.10 / 45.12)

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68. NOP-SS-3001, Procedure Review and Approval defines the "Procedure Correction" method of altering a procedure as follows:
Procedure Correction – An alteration type used for editorial changes, corrections of inconsistencies, and similar changes.

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 this 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
References provided to Candidate		None	Exam Level
Question Source:	New		RO
Question Cognitive Level:	Low - Memory		Technical References:
Objective:			NOP-SS-3001 rev. 17 pages 7 and 8
			Level Of Difficulty: (1-5)
			2
			10 CFR Part 55 Content:
			(CFR: 41.10 / 43.3 / 45.13)

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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) $\geq 290,957$ gpm
- B. (1) ≥ 2060.8 psig
(2) ≤ 610 °F
(3) $\geq 389,500$ gpm
- C. (1) ≥ 2006.8 psig
(2) ≤ 525 °F
(3) $\geq 290,957$ gpm
- D. (1) ≥ 2006.8 psig
(2) ≤ 525 °F
(3) $\geq 389,500$ gpm

Answer: A

Explanation/Justification:

- A. Correct. IAW TS 3.4.1
- B. Incorrect. Flow is for all 4 RCPs operating
- C. Incorrect. RCS pressure is 2006.8 instead of 2060.8 and temp is the TS minimum temp for criticality, flow is correct
- D. Incorrect. RCS pressure is 2006.8 instead of 2060.8; temp is the TS minimum temp for criticality; flow is for all 4 RCPs operating

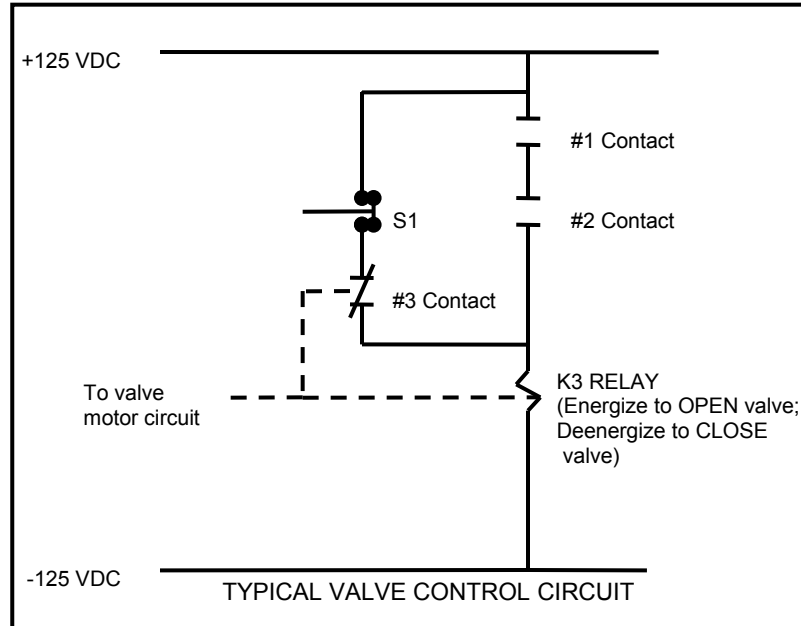
Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of limiting conditions for operations and safety limits.
K/A#	2.2.22	K/A Importance	4.0
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	Low - Memory		Level Of Difficulty: (1-5)
Objective:			10 CFR Part 55 Content:

RO
TS 3.4.1 Amend. 279
4
(CFR: 41.5 / 43.2 / 45.2)

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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:		
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:					

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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

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
References provided to Candidate	None	Exam Level	RO
Question Source:	Bank	BV 2009 NRC Exam	Level Of Difficulty: (1-5)
Question Cognitive Level:	High - Analysis	10 CFR Part 55 Content:	2
Objective:			(CFR: 41.12 / 45.10)

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72. During a Loss of Coolant Accident, the Operator is directed by DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture to locally establish a High Pressure Injection Alternate Minimum Flowpath if BWST Level is being reduced at a rate less than 2 feet per hour.

Which of the following is correct regarding radiological exposure for performance of this task?

- A. The exposure must receive prior approval by the Emergency Director to protect equipment since the expected dose exceeds annual personnel dose and planned special exposure limits
- B. The task has been dose assessed and determined that the worst case exposure is the limits established for a planned special exposure. Radiation Protection approval is required to perform this task since annual limits will be exceed
- C. The task has been dose assessed and determined that the worst case exposure is the limits established for annual personnel dose and planned special exposure. No further approval is required to perform this task
- D. This task is optional for a Loss of Coolant Event and therefore is only performed if the area is determined to be accessible by Radiation Protection

Answer: C

Explanation/Justification:

- A. Incorrect. Plausible if operator assumes establishing Alternate HPI Recirc is accomplished after ECCS Suctions are transferred to the Emergency Sump causing elevated Auxiliary Building Dose Rates.
- B. Incorrect. Plausible if operator assumes establishing Alternate HPI Recirc is accomplished after ECCS Suctions are transferred to the Emergency Sump causing elevated Auxiliary Building Dose Rates.
- C. Correct. DB-OP-02000 Bases and Deviation Document provide the following: The actions in the Auxiliary Building to place the HPI Alternate Recirculation flowpath in service were dose assessed. A worst case dose of 540 mrem was predicted. It should be noted this action will occur prior to transfer of ECCS suction to the Emergency Sump. This action is only required if the BWST depletion rate is less than 2 feet per hour. As a result, at a minimum, 15 hours is available to complete the activity prior to transfer to the Emergency Sump. In addition, a time of 30 minutes in the ECCS Pump Rooms was provided by the calculation for the operator to complete the task. Failure to follow the prescribed route provided in Attachment 14 or attempting the actions after the ECCS suction have been transferred to the Emergency Sump could result in a significantly higher dose. This predicted dose is within the limits established for annual personnel dose and planned special exposure
- D. Incorrect. Plausible because some actions in DB-OP-02000 that are not part of the mitigation strategy are preceded by "IF Accessible". This actions can be omitted or delayed if conditions do not allow access.

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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	Low - Fundamental		RO
Objective:			DB-OP-02000 Attachment 14 and Bases and Deviation Document for DB-OP-02000 for Dose Assessment.
			Level Of Difficulty: (1-5)
			3
			10 CFR Part 55 Content:
			(CFR: 41.12 / 45.9 / 45.10)

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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

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. Lowering 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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	Low - Fundamental		Level Of Difficulty: (1-5)
Objective:			10 CFR Part 55 Content:

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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
References provided to Candidate		None	Exam Level
Question Source:	New		RO
Question Cognitive Level:	Low - Memory		Technical References:
Objective:			RA-EP-02010 Rev. 12 page 7
			Level Of Difficulty: (1-5)
			2
			10 CFR Part 55 Content:
			(CFR: 41.10 / 45.13)

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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

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 $\leq 20^{\circ}\text{F}$. 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

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:					

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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:			

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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

Answer: 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:	

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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 Rupture of _____(1)_____ will be isolated in _____(2)_____.

- 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
- 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 ≥ 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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	High - Comprehension		DB-OP-02000 pg 6
Objective:			USAR Chapter 15 and Table 15.4.2-2
			SRO
			Level Of Difficulty: (1-5)
			3
			10 CFR Part 55 Content:
			CFR: 43(b)(5)

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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 45 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) Within 1 hour, perform Tech Spec SR 3.8.1.1 for OPERABLE Off-Site AC Sources
(2) ~920 megawatts
- B. (1) Within 1 hour, perform Tech Spec SR 3.8.1.1 for OPERABLE Off-Site AC Sources
(2) ~800 megawatts
- C. (1) Declare Off-Site AC Sources Inoperable
(2) ~920 megawatts
- D. (1) Declare Off-Site AC Sources Inoperable
(2) ~800 megawatts

Answer: C

Explanation/Justification:

- A. Incorrect. (1) is Plausible since this would be the required action if 1 of the Off-Site AC Sources were inoperable, however since both are being declared inoperable the required actions are different. (2) correct megawatt limitation for 45 psig of H² gas pressure.
- B. Incorrect. (1) is Plausible since this would be the required action if 1 of the Off-Site AC Sources were inoperable, however since both are being declared inoperable the required actions are different. (2) is Plausible megawatt limitation if candidate uses 30 psig H² gas pressure curve instead of 45 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 H² gas pressure curve instead of 45 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:			

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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 actions is required to be taken **FIRST**?

- A. Go to Section 7, Overcooling, and attempt to reseal 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

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
References provided to Candidate		None	Exam Level
Question Source:	Bank	58982	Level Of Difficulty: (1-5)
Question Cognitive Level:	High - Comprehension		10 CFR Part 55 Content:
Objective:			

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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, specific 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

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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	High - Comprehension		Level Of Difficulty: (1-5)
Objective:			10 CFR Part 55 Content:
			SRO
			DB-OP-02000 TBD pg 7, 448, & 449
			3
			CFR: 43(b)(5)

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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:			

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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 raise 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:			

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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:			

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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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	Low - Memory		SRO
Objective:			Tech Spec bases pages B 3.5.1-3 and B 3.5.1-4
			Level Of Difficulty: (1-5)
			10 CFR Part 55 Content:
			2
			CFR: 43(b)(2)

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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 Tav_g 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
References provided to Candidate		None	Exam Level
Question Source:	New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:	High - Comprehension	10 CFR Part 55 Content:	CFR: 43(b)(5)
Objective:			

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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
References provided to Candidate		None	Exam Level
Question Source:	New		SRO
Question Cognitive Level:	High - Comprehension		Technical References:
Objective:			TS bases page B 3.5.4-1
			Level Of Difficulty: (1-5)
			2
			10 CFR Part 55 Content:
			CFR: 43(b)(2)

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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

Answer: 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
References provided to Candidate		None	Exam Level
Question Source:	New		SRO
Question Cognitive Level:	High - Analysis		Technical References:
Objective:			DB-OP-02000 TBD page 7; DB-OP-02000 page 2 index
			Level Of Difficulty: (1-5)
			3
			10 CFR Part 55 Content:
			CFR: 43(b)(5)

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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:			

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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. The duration of the maintenance activity is expected to be ~ 72 hours.

In order to maintain CONTAINMENT OPERABILITY during this maintenance period:

- (1) What is/are the MINIMUM REQUIRED CTMT area(s) to be inspected for trash and loose debris?
- (2) What is the required frequency of this trash and loose debris inspection?
- A. (1) The 565' Elevation Outside The D-Rings AND outside the Trash Gate Area
(2) At least once daily and again during the final exit after completion of work.
- B. (1) The 565' Elevation Outside The D-Rings AND inside the Trash Gate Area
(2) At least once daily and again during the final exit after completion of work.
- C. (1) The 565' Elevation Outside The D-Rings AND outside the Trash Gate Area
(2) ONLY during the final exit after completion of work.
- D. (1) The 565' Elevation Outside The D-Rings AND inside the Trash Gate Area
(2) ONLY during the final exit after completion of work.

Answer: A

Explanation/Justification:

- A. Correct. IAW DB-OP-03013 and DB-OP-01101 The Shift Manager is required to determine which areas of CTMT are affected by the activity and select those areas to be inspected. TVR 8.5.3.2 requires a daily inspection of containment for all areas of containment affected by an entry, at least once daily while work is ongoing and again during the final exit after completion of work (containment closeout) when CONTAINMENT OPERABILITY is established. SRO only since it requires application of surveillance requirements and the task is unique to the SRO position. The SRO must familiar with the CTMT location of the 6 trisodium phosphate baskets and the surveillance requirements for trash and loose debris inspections.
- B. Incorrect. In Part 1, inside the trash gate area is not required. Part 2 is correct.
- C. Incorrect. Part 1 is correct. Part 2 is not correct since it omits the daily inspection requirement.
- D. Incorrect. In Part 1, the trash gate area is not required. Part 2 is not correct since it omits the daily inspection requirement.

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*
References provided to Candidate		Exam Level	SRO
		Technical References:	DB-OP-03013 pages 4, 6, and pg 18 & 19 Attach. 5
Question Source:	New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:	High - Application	10 CFR Part 55 Content:	CFR: 43(b)(2) & (5)
Objective:			

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91. The following plant conditions exist:

- The reactor has tripped.
- SG 1 Pressure = 780 psig
- SG 1 Level = 124 inches
- RCS Pressure = 1200 psig
- Containment Pressure = 22 psia
- EDG 1 failed to start.
- SG 2 Pressure = 775 psig
- SG 2 Level = 125 inches
- Incore Thermocouples = 572 °F
- C1 and D1 are powered from Off-site power

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
References provided to Candidate		None	Exam Level
Question Source:	Bank	76729	SRO
Question Cognitive Level:	High - Comprehension		Technical References:
Objective:			DB-OP-02000 Specific Rule 3 pg 239, and Table 2. pg 410
			Level Of Difficulty: (1-5)
			4
			10 CFR Part 55 Content:
			CFR: 43(b)(5)

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92. The plant is operating at 100% power at 4 Effective Full Power Days 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 Rod 6-5 reads 0%.
- Relative Position indication for all Group 6 Rods read 100%.

Which of the following DB-OP-02516 CRD Malfunctions sections is required to be implemented in response to this condition?

- A. Dropped Control Rods
- B. Misaligned Control Rods
- C. Control Rod Position Indication Malfunctions
- D. Stuck Control Rods

Answer: C

Explanation/Justification:

- A. Incorrect – Plausible because Rod 6-5 Absolute Position Indication reads 0% as it would if the Rod were actually dropped.
- B. Incorrect – Plausible because Annunciator 5-2-E, CRD ASYMMETRIC ROD is in alarm as it would be if the rod were actually misaligned
- C. Correct. IAW DB-OP-02516 Section 2, Symptoms. SRO only since it requires the SRO to select the appropriate procedure section to address the conditions stated in the stem.
- D. Incorrect – Plausible because at 4 EFPDs, a reactor startup was recently completed. It is possible that Rod 6-5 is stuck on the bottom and did not respond to out commands.

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:	Bank	ANO – 1 2009 SRO question #16	Level Of Difficulty: (1-5) 3
Question Cognitive Level:		High - Analysis	10 CFR Part 55 Content: CFR: 43(b)(5)
Objective:			

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93. The Plant is operating at 100% power with all systems in a normal alignment
- Station 1 Air Compressor (SAC-1) is operating
 - Station 2 Air Compressor (SAC-2) is in standby.

The following alarms are received:

- (9-4-F) INSTR AIR DRYER TRBL
- (9-1-F) INSTR AIR HDR PRESS LO
- (9-3-E) STATION AIR HDR PRESS LO

PI 810, INSTRUMENT AIR HEADER PRESS is 70 psig and slowly lowering
The Outside operator reports excessive air blowdown out of the in-service air dryer.

All other plant equipment operates as designed.

Based on these conditions, as Command SRO, what will be the **FIRST** directed action for the crew to perform?

- A. Trip the Rx and Initiate AFW Flow **AND** Isolation of BOTH SGs.
- B. Manually start the Emergency Instrument Air Compressor (EIAC).
- C. Bypass and isolate the in-service air dryer.
- D. Place the standby Air Dryer in service.

Answer: A

Explanation/Justification:

- A. Correct. IAW DB-OP-02528 supplementary actions (step 4.2.1) trip the reactor and isolate AFW flow when instrument air pressure drops below 75 psig. The SRO must assess the conditions, then select and implement the appropriate supplemental steps to mitigate the condition. Based on the operating characteristics of the air operated valves, the SRO will direct the reactor trip and AFW isolation BEFORE any automatic trip is generated. This is not an immediate action of the procedure, therefore the ROs would not be expected to memorize this setpoint or required action. System knowledge alone cannot be used to answer the question since the valves will not close until sometime later, and only then would a RX trip occur.
- B. Incorrect. Plausible since this action might decrease the instrument air pressure falling trend, however this is not procedurally directed for instrument air dryer malfunctions, and the compressor should have already automatically started.
- C. Incorrect. This would be the first directed action if air pressure was above 75 psig.
- D. Incorrect. This would be the second directed action if air pressure was above 75 psig.

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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	High - Analysis		Level Of Difficulty: (1-5)
Objective:			10 CFR Part 55 Content:

DB-OP-02528 step 4.2.1 page 20

2

CFR: 43(b)(5)

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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
References provided to Candidate		None	Exam Level
Question Source:	New		SRO
Question Cognitive Level:	High - Comprehension		Technical References:
Objective:			DB-OP-06912 pg 13 NOTE and step 4.21
			Level Of Difficulty: (1-5)
			2
			10 CFR Part 55 Content:
			CFR: 43(b)(5)

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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 and operating the hoist with a "dummy" load.
IAW the guidance provided within DB-NE-06308, Main Fuel Handling Bridge Operating Procedure:

Who's permission, (by title) is required to authorize the bypassing of this interlock?

(Assume the suggested troubleshooting evolution is NOT specifically addressed within the procedures.)

- A. Either the Shift Manager OR Fuel Handling Director
- B. Both the Outage Director AND Fuel Handling Director
- C. Both the Outage Director AND Shift Manager
- D. Only the Site Vice President

Answer: 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. Only one of the two is required to authorize this bypassing of the interlock.
- C. Incorrect. These individuals are required to authorize other refueling activities, but bypassing of the interlock must be authorized by either the SM or Fuel handling Director.
- D. Incorrect. This individual is required to authorize emergency exposures beyond 25 REM, but bypassing of the interlock must be authorized by either the SM or Fuel handling Director.

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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References: DB-NE-06308 para 2.1.1 pg 4
Question Cognitive Level:	High - Comprehension		Level Of Difficulty: (1-5) 4
Objective:			10 CFR Part 55 Content: CFR: 43(b)(7)

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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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	High - Comprehension		Level Of Difficulty: (1-5)
Objective:			10 CFR Part 55 Content:
			SRO
			NOP-WM-4300, section 4.7.6 pg 19
			4
			CFR: 43(b)(5)

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97. The following plant conditions exist:
- A large break LOCA has occurred and the Emergency Director has declared a Site Area Emergency.
 - An individual has been injured in an area where the dose rate is 150 Rem/hr.
 - Two members of the emergency response organization need to enter this area to perform life saving activities.

As the Emergency Director, you have been asked to authorize the emergency dose for this life saving activity.

What is the MAXIMUM time that the individuals can remain in the area before exceeding the established emergency dose guidelines of RA-EP-02620, Emergency Dose Control And Potassium Iodide Distribution?

- 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, $25\text{R}/150\text{ R/hr} = .167\text{ hr} \times 60\text{ min./hr} = 10\text{ min}$. 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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	High – Analysis/Application		Level Of Difficulty: (1-5)
Objective:			10 CFR Part 55 Content:
			SRO
			RA-EP-02620 pge 7 step 6.1.3.b.1
			3
			CFR: 43(b)(4)

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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:					

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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
References provided to Candidate		None	Exam Level
Question Source:	New		Technical References:
Question Cognitive Level:	High - Comprehension		Level Of Difficulty: (1-5)
Objective:			10 CFR Part 55 Content:

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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

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
References provided to Candidate		RA-EP-01500, Emergency Classification EAL Tables only	Exam Level
			Technical References:
Question Source:	New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:	High - Application	10 CFR Part 55 Content:	CFR: 43(b)(7)
Objective:			