

1. The following plant conditions exist:

- RCS pressure is 250 psig
- Pressurizer temperature is 406 °F
- Quench Tank pressure is 80 psig
- Containment pressure is 14.7 psia
- The crew has just finished drawing a Pressurizer steam bubble.

The following event occurs:

- The Pressurizer Safety fails open and the Quench Tank rupture disc ruptures.

What will be the Pressurizer Safety Valve downstream temperature, for these conditions?

- A. ~212 °F
- B. ~325 °F
- C. ~345 °F
- D. ~406 °F

Answer: B

Explanation/Justification:

- A. Incorrect. Plausible because this is the saturation temperature for 14.7 psia which the candidate could select if isenthalpic throttling is not considered.
- B. Correct answer IAW Steam tables and isenthalpic throttling process. When the Pressurizer Safety Valve fails open, the rupture disc on the safety valve will blow releasing pressurizer steam to the CTMT atmosphere. Therefore the downstream conditions will be the CTMT conditions.
- C. Incorrect. Plausible because this is the saturation temperature for 80 psig (Quench Tank pressure)
- D. Incorrect. Plausible because this the temperature at which the event started

Sys #	System	Category	KA Statement
000008	Pressurizer (PZR) Vapor Space Accident	AK1. Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident:	Thermodynamics and flow characteristics of open or leaking valves
K/A#	AK1.01	K/A Importance 3.2	Exam Level RO
References provided to Candidate	Steam Tables	Technical References:	Steam Tables
Question Source:	New	Level Of Difficulty: (1-5) 3	3
Question Cognitive Level:	High - Application	10 CFR Part 55 Content:	(CFR 41.8 / 41.10 / 45.3)
Objective:			

2. A small break loss of coolant accident has occurred.

Which of the following describes the function of the Steam Generator **required** to mitigate this event?

- A. Steam Generators are not required to mitigate any loss of coolant accidents.
- B. For certain small break LOCAs, heat removal by the SGs is required to satisfy the acceptance criteria of 10CFR50.46, Acceptance Criteria for Emergency Core Cooling Systems.
- C. Only the isolation of Containment provide by the Main Steam Isolation Valves is required to mitigate a loss of coolant accident.
- D. Boiler-Condenser Cooling provided by the Steam Generators is required to ensure condensed steam is returned to the Reactor Vessel to provide adequate RCS inventory for loss of coolant accidents.

Answer: B

Explanation/Justification:

- A. Incorrect - Maintaining SG's available as a heat removal capability is required to ensure that Core cooling is provided if flow out the break is not sufficient.
- B. Correct per DB-OP-02000 Bases and Deviation Document Step 5.6 and 5.7. Maintaining SG's available as a heat removal capability will ensure that Core cooling is provided if flow out the break is not sufficient.
- C. Incorrect – Although the MSIVs will isolate Containment, without a break in the Steam Generator or Main Steam Line, Containment Integrity is not affected by the position of the MSIV.
- D. Incorrect – Although the condensed steam is returned to the reactor vessel, adequate RCS inventory requires HPI or LPI operation.

Sys #	System	Category	KA Statement	
000009	Small Break LOCA	EK2. Knowledge of the interrelations between the small break LOCA and the following:	S/Gs	
K/A#	EK2.03	K/A Importance	3.0	Exam Level
References provided to Candidate		None	Technical References:	RO DB-OP-02000 R19 Bases and Deviation Document Steps 5.6 and 5.7 for SBLOCA requirement to maintain SG available. 10CFR50.46
Question Source:		New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content:	(CFR 41.7 / 45.7)
Objective:				

3. A Large Break Loss of Coolant Accident occurred.

Which of the following sets of Plant Conditions indicates inadequate core cooling exists?

Incore Thermocouple temperature average _____(1)_____.

WITH

Reactor Coolant System Pressure _____(2)_____.

- A. (1) 400 °F
(2) 400 PSIG
- B. (1) 600 °F
(2) 1540 psig
- C. (1) 500 °F
(2) 680 psig
- D. (1) 550 °F
(2) 600 psig

Answer: D

Explanation/Justification: ICC is normally determined using Figure 2 of DB-OP-02000 which labels the Temperature/Pressure relationships that represent ICC conditions. For this question, steam tables are provide. Candidate must demonstrate understanding of plant condition using the saturation curve.

- A. Incorrect –ICC is indicated by Superheated conditions. Values provided indicate subcooled conditions exist.
- B. Incorrect –. ICC is indicated by Superheated conditions. Values provided indicate saturated conditions exist. Candidate that assumes highest temperature is indicative of ICC would pick this combination.
- C. Incorrect – ICC is determined using Figure 2 of DB-OP-02000. ICC is indicated by Superheated conditions. Values provided indicate saturated conditions exist. .
- D. Correct – ICC is determined using Figure 2 of DB-OP-02000. ICC is indicated by Superheated conditions. Values provided indicate superheated conditions exist, ICC exists.

Sys #	System	Category	KA Statement
000011	Large Break LOCA	EA2. Ability to determine or interpret the following as they apply to a Large Break LOCA:	Verification of adequate core cooling
K/A#	EA2.10	K/A Importance 4.5	Exam Level RO
References provided to Candidate		Steam Tables	Technical References: DB-OP-02000 R26 Figure 2 Bases and Deviation Document for DB-OP-02000 R19 step 5.13
Question Source:		New	Level Of Difficulty: (1-5) 3
Question Cognitive Level:		High - Application	10 CFR Part 55 Content: (CFR 43.5 / 45.13)
Objective:			

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4. The plant is operating at 100% power with all systems in normal alignment for this power level.

Which of the following abnormal conditions requires an **IMMEDIATE** power reduction and stopping the affected Reactor Coolant Pump?

- A. MU59A, RCP 2-1 Seal Return Isolation Valves fails closed.
- B. Computer Point L828, 2-1 Motor Lower Bearing Low Oil Level Alarm with stable bearing temperatures.
- C. Computer Point T828, 2-1 Motor Stator Temperature Alarm with indicated temperature 350 °F.
- D. Computer Points for 2-1 Seal Cavity Pressure P833 (second stage) reads 1100 psig, and P834 (third stage) reads 50 psig

Answer: C

Explanation/Justification:

- A. Incorrect – Shutdown is required within 30 minutes, not immediately.
- B. Incorrect – Shutdown is required if bearing temperatures are rising with low oil level, not immediately.
- C. Correct – Power reduction and Shutdown is immediately required per DB-OP-02515 Step 4.6.1 RNO.
- D. Incorrect – Values provided indicated a single RCP Seal Stage is failed. Immediate Shutdown is not required for single stage failure per DB-OP-02515, Step 4.1.1

Sys #	System	Category	KA Statement	
000015/ 000017	Reactor Coolant Pump (RCP) Malfunctions	Generic	Knowledge of abnormal condition procedures.	
K/A#	2.4.11	K/A Importance	4.0	Exam Level
References provided to Candidate		None	Technical References:	RO DB-OP-02515 R11, RC Pump and Motor Malfunctions Step 4.6.1 RNO
Question Source:		New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:		Low - Memory	10 CFR Part 55 Content:	(CFR: 41.10 / 43.5 / 45.13)
Objective:				

5. Following the loss of **BOTH** Makeup Pumps from full power operations, why is RCS pressure reduced to 1700 to 1800 psig?

Reducing RCS Pressure will_____.

- A. reduce Reactor Coolant Pump seal leak off, preserving RCS Inventory.
- B. allow the Reactor Protective System to be placed in Shutdown Bypass.
- C. allow the Safety Features Low RCS Pressure Trip to be blocked.
- D. allow the High Pressure Injection system to restore RCS Inventory.

Answer: D

Explanation/Justification:

- A. Incorrect – Plausible because this would reduce seal leakoff, RCS Inventory is preserved by isolating Letdown for this event.
- B. Incorrect – Plausible because in a normal shutdown, Shutdown Bypass Operation can be established at this RCS Pressure range.
- C. Incorrect – Plausible because in a normal shutdown, RCS Pressure is reduced to slow the transition when blocking the SFAS Low RCS Pressure Trip at 1670 psig prior to SFAS Actuation at 1600 psig..
- D. Correct – The ability to add inventory to the RCS is established by starting High Pressure Injection in piggyback mode which will then provide approximately 1800 psig discharge pressure allowing flow to the RCS.

Sys #	System	Category	KA Statement
000022	Loss of Reactor Coolant Makeup	AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup:	Actions contained in SOPs and EOPs for RCPs, loss of makeup, loss of charging, and abnormal charging
K/A#	AK3.02	K/A Importance	3.5
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	DB-OP-02512 R14, Makeup and Purification System Malfunctions Attachment 6.
Question Cognitive Level:	Low - Fundamental	Level Of Difficulty: (1-5)	2
Objective:		10 CFR Part 55 Content:	(CFR 41.5, 41.10 / 45.6 / 45.13)

6. The following plant conditions exist:

- A plant cooldown is in progress for refueling.
- Reactor Coolant Pumps 2-1 and 2-2 are in service.
- DH Train 2 is in service
- DH Train 1 is out of service being transferred from LPI to DHR Mode
- RCS temperature is 180 °F
- Pressurizer level is 80 inches
- RCS pressure is 220 psig

The following event occurs:

- A loss of Off-Site Power occurs
- EDG 2 fails to start

Based on these conditions:

In accordance with DB-OP-02527, Loss of Decay Heat Removal, what is the **PRIORITY** for how core heat removal will be established?

- A. Maintain current RCS temperature Conditions using Turbine Bypass Valves and Natural Circulation.
- B. Allow RCS to heatup to Mode 4, then use Atmospheric Vent Valves and Natural Circulation to control RCS temperature.
- C. Allow RCS to heatup to Mode 4, then use Makeup, High Pressure Injection, and the High Point Vents to establish Feed and Bleed Cooling.
- D. Maintain current RCS temperature Conditions using Makeup, High Pressure Injection, and the PORV to establish Feed and Bleed Cooling.

Answer: B

Explanation/Justification:

- A. Incorrect – This outcome would be desired to avoid transition back into Mode 4, but the loss of offsite power has caused a loss of Circ Water Pumps and therefore the main condenser. TBVs will close once Condenser Pressure rises to 17 inch HgA.
- B. Correct - Step by step priority as listed in DB-OP-02527 R15, Loss of Decay Heat Removal step 4.1.7 RNO
- C. Incorrect – At low RCS Pressures, the flow out the High Point Vents will be insufficient to remove core decay heat. The RCS would heatup beyond Mode 4 (greater than 280 F). Candidate may assume PORV is not available for this scenario. The PORV is DC Powered
- D. Incorrect – Although Feed and Bleed cooling would be successful in removing decay heat, it is likely the PORV flow at low RCS pressures would not be sufficient to allow continued cooldown. In addition, SG heat transfer is prioritized above Feed and Bleed Cooling in DB-OP-02527, Loss of Decay Heat Removal.

Sys #	System	Category	KA Statement		
000025	Loss of Residual Heat Removal System (RHRS)	AA1. Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System:	RCS/RHRS cooldown rate		
K/A#	AA1.01	K/A Importance	3.6	Exam Level	RO
References provided to Candidate		None	Technical References:	DB-OP-02527 R15, Loss of Decay Heat Removal step 4.1.7 RNO.	
Question Source:	New	Level Of Difficulty: (1-5)		3	
Question Cognitive Level:	High - Comprehension	10 CFR Part 55 Content:		(CFR 41.7 / 45.5 / 45.6)	

7. The following plant conditions exist:

- The plant is operating at 100% power with all systems in normal alignment for this power level.

The following indications occur:

- 11-4-B, CCW PMP 1 FLOW LO, annunciator is in alarm with a flowrate of 2400 gpm.
- 2-3-A, LETDOWN TEMP HI, annunciator is in alarm with a temperature of 144 °F.
- 11-1-B, CCW HX 1 OUTLET TEMP HI, annunciator is in alarm with a temperature of 122°F.

Which one of the following actions will automatically occur?

- A. CCW Pump 1 will trip.
- B. The standby CCW pump will start.
- C. CCW Non-Essential Header will isolate.
- D. Letdown cooler inlet isolation valve, MU 2B, will close.

Answer: D

Explanation/Justification:

- A. Incorrect – Plausible because the CCW Pump is operating at a low flow condition. It would be logical to have the pump trip to protect the pump.
- B. Incorrect – Plausible because the CCW Pump is operating at a low flow condition, but above the Flowrate to cause and automatic start of the Standby Pump
- C. Incorrect – Plausible because the CCW system is operating abnormally. Closing the non-essential header isolation could protect the essential functions provided by CCW.
- D. Correct – CCW temperatures associated with the letdown cooler will rise with reduced flow that would lead to a high temperature isolation of letdown flow.

Sys #	System	Category	KA Statement
000026	Loss of Component Cooling Water (CCW)	AA1. Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water:	Flow rates to the components and systems that are serviced by the CCWS; interactions among the components
K/A#	AA1. 07	K/A Importance	2.9
References provided to Candidate	None	Exam Level	RO
Question Source:	BANK 37623	Technical References:	DB-OP-02002 R08 Annunciator 2-3-A Letdown/Makeup Alarm Panel 2 Annunciators
Question Cognitive Level:	High - Comprehension	Level Of Difficulty: (1-5)	3
Objective:		10 CFR Part 55 Content:	(CFR 41.7 / 45.5 / 45.6)

8. The plant is operating at 100% power with all systems in normal alignment for this power level.

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

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

The Pressurizer PORV will _____(1)_____.
 The Pressurizer Spray Valve will _____(2)_____.
 The Pressurizer Heaters will _____(3)_____.

- A. (1) remain closed
 (2) remain closed
 (3) remain energized
- B. (1) open
 (2) open
 (3) de-energize
- C. (1) open
 (2) remain closed
 (3) de-energize
- D. (1) remain closed
 (2) open
 (3) remain energized

Answer: B

Explanation/Justification:

- A. Incorrect – Plausible if the candidate believes the RCS signal is SASS protected like most other NNI signals. Instantaneous failures would normally cause a SASS transfer for SASS protected instrument inputs resulting in no change to the input for the PORV, Spray Valve, or Pressurizer Htrs.
- B. Correct – The selected RPS Pressure signal is used to control the PORV, the PZR Spray Valve, and the Pressurizer Heaters. A high failure will cause the PORV to Open, the Pressurizer Spray Valve to Open, and the Pressurizer Heaters to turn off.
- C. Incorrect – Plausible if the candidate thought that the safety grade Reactor Protective System RCS Pressure signal is used to control the PORV. Since PORV has the most impact on the plant, this conclusion is logical.
- D. Incorrect – Plausible if the candidate thought that the safety grade Reactor Protective System RCS Pressure signal is used to control only the Pressurizer Spray Valve. The remaining positions would be correct if supply with a different pressure signal.

Sys #	System	Category	KA Statement
000027	Pressurizer Pressure Control System (PZR PCS) Malfunction	AK2.03 Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following:	Controllers and positioners
K/A#	AK2.03	K/A Importance	Exam Level
		2.6	RO
References provided to Candidate	None	Technical References:	DB-OP-02513, Pressurizer System Abnormal Operation Att. 2 page 54

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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9. The plant is operating at 100% power with all systems in normal alignment for this power level.

The Reactor Protective System (RPS) generates a valid reactor trip signal, but the Control Drive Trip Breakers fail to open.

In accordance with DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture, the Reactor Operator in the Control Room momentarily deenergizes 480 volt Unit Substations E2 AND F2.

Following restoration of power to E2 and F2, which of the following previously running loads will return to operation without operator action?

- A. Radwaste Exhaust Fan.
- B. Main Station Exhaust Fan.
- C. Clean Waste Monitor Tank Transfer Pump.
- D. Spent Fuel Pool Pump.

Answer: D

Explanation/Justification:

- A. Incorrect – There is no seal in feature for this fan. The Radwaste Ventilation System would be lost until the fan is restarted
- B. Incorrect – There is no seal in feature for this fan. The Main Station Exhaust System would be lost until the fan is restarted
- C. Incorrect – There is no seal in feature for this pump. This RCS inventory addition source would be lost until the pump is restarted.
- D. Correct – The controller for the SFP Pumps have a seal in feature that would restart the pump following restoration of power.

Sys #	System	Category	KA Statement
000029	Anticipated Transient Without Scram (ATWS)	EK2. Knowledge of the interrelations between ATWS and the following:	Breakers, relays, and disconnects
K/A#	EK2.06	K/A Importance	2.9*
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	Eng Change Package 10-0654
Question Cognitive Level:	High - Comprehension	Level Of Difficulty: (1-5)	2.5
Objective:		10 CFR Part 55 Content:	(CFR 41.7 / 45.7)

10. Plant conditions are as follows:

- A SG tube has ruptured on SG 1.
- The reactor is tripped.
- RCS pressure is 1990 psig.
- RCS Tave is 548 °F.
- BOTH SGs are being steamed through the Turbine Bypass Valves.

Which one of the following will occur if SG 1 exceeds 250 inches?

- A. The MSIV on SG 1 closes so that ONLY SG 2 may be steamed to the condenser.
- B. SFRCS will realign Aux Feedwater to ONLY feed SG 2.
- C. The MSIVs on BOTH SGs close and prevent steaming of BOTH SGs to the condenser.
- D. BOTH the AFW supply and Main Steam isolation close for SG 1.

Answer: C

Explanation/Justification: SFRCS will actuate on high SG level. The setpoint is 250 inches.

- A. Incorrect – Plausible since only the #1 SG MSIV closes since that is the only SG operating at a high level.
- B. Incorrect – Plausible since #1 SG is at a high level, we should stop feeding it by aligning both AFW pumps to feed #2 SG.
- C. Correct – High level in either SG will close both MSIVs.
- D. Incorrect – Plausible because elevated level in #2 SG will promote heat transfer that may be needed if #1 SG is removed from service.

Sys #	System	Category	KA Statement
000038	Steam Generator Tube Rupture (SGTR)	EA2. Ability to determine or interpret the following as they apply to a SGTR:	Status of MSIV activating system
K/A#	EA2.12	K/A Importance	Exam Level
		3.9*	RO
References provided to Candidate		None	Technical References:
BANK 36449			DB-OP-02000 R26 Table 1 SFRCS Response
Question Cognitive Level:		High - Comprehension	Level Of Difficulty: (1-5)
Objective:			2.5 - 3
			10 CFR Part 55 Content:
			(CFR 43.5 / 45.13)

11. INITIAL CONDITIONS:

- RCS temperature 500 °F
- RCS pressure 1000 psig
- RCS cooldown in progress
- A Main Steam Line Break on #2 SG in Containment occurs.

CURRENT CONDITIONS:

- RCS temperature 425 °F
- RCS pressure 750 psig

Assuming no change in Main Steam Line break size, from initial to current condition, subcooling margin has ____ (1) ____ and steam flow out the break has ____ (2) ____.

- A. (1) risen
(2) lowered
- B. (1) risen
(2) remained the same
- C. (1) lowered
(2) lowered
- D. (1) lowered
(2) remained the same

Answer: A

Explanation/Justification: Note: During an RCS Cooldown, the SFRCS Low SG Pressure Trip would be blocked at the RCS temperature provided. SFRCS would not actuate on Low SG Pressure for this scenario.

- A. Correct – Subcooled margin for the initial conditions would be approximately 45 degrees while SCM for current conditions would be approximately 85 degrees. Steam Flow would be reduced as SG Pressure Lowers.
- B. Incorrect – While SCM will have risen as noted in A above, steam line break flow will be dependant on SG pressure. As the RCS cools, SG pressure will lower and therefore break flow will lower.
- C. Incorrect – Candidate may select this response assuming lower RCS temperatures produces lower SCM.. While steam line break flow will be dependant on SG pressure. As the RCS cools, SG pressure will lower and therefore break flow will lower.
- D. Incorrect – Candidate may select this response assuming lower RCS temperatures produces lower SCM.. While steam line break flow will be dependant on SG pressure. As the RCS cools, SG pressure will lower and therefore break flow will lower.

Sys #	System	Category	KA Statement	
000040	Steam Line Rupture – Excessive Heat Transfer	Generic	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	
K/A#	2.1.7	K/A Importance	4.4	Exam Level
References provided to Candidate		Steam Tables	Technical References:	RO Fundamental Theory - Steam Table and General Physics HTFF Chapter 4 page 3
Question Source:		New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content:	(CFR: 41.5 / 43.5 / 45.12 / 45.13)
Objective:				

12. The plant is operating at 55%.

The following event occurs:

- Both Main Feedwater Pumps trip

Without other induced changes in plant conditions,

The Control Rod Trip breakers open **DIRECTLY** due to an _____(1)_____

The Turbine trips **DIRECTLY** due to _____(2)_____

- A. (1) RPS trip
(2) CRD Trip Confirm
- B. (1) RPS trip
(2) SFRCS trip
- C. (1) ARTS trip
(2) SFRCS trip
- D. (1) ARTS trip
(2) CRD Trip Confirm

Answer: D

Explanation/Justification:

- A. Incorrect – The RPS trips would not actuate until plant condition such as RCS Pressure changed. As a result, RPS would not directly trip the reactor for this scenario. This is the basis for installing the ARTS System.
- B. Incorrect – The RPS trips would not actuate until plant condition such as RCS Pressure changed. As a result, RPS would not directly trip the reactor for this scenario. This is the basis for installing the ARTS System. SFRCS does directly generate a Turbine Trip Signal.
- C. Incorrect – ARTS would trip the reactor, but tripping both MFP will not directly trip SFRCS. An SFRCS Trip will directly trip the Main Turbine.
- D. Correct – ARTS senses MFP Turbine status and causes a reactor trip if both MFP Turbine Trip. CRD Trip Confirm will cause EHC to trip the Main Turbine.

Sys #	System	Category	KA Statement
000054	Loss of Main Feedwater (MFW)	AA2. Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW):	Occurrence of reactor and/or turbine trip
K/A#	AA2.01	K/A Importance 4.3	Exam Level RO
References provided to Candidate		None	Technical References: DB-OP-06202 pg72, DBBP-TRAN-0034 pg 6&7
Question Source:		New	Level Of Difficulty: (1-5) 3
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content: (CFR: 43.5 / 45.13)
Objective:			

13. The plant is operating at 100% power with all systems in normal alignment for this power level.

A Tornado hits the Switchyard damaging all three offsite lines causing a loss of offsite power.

Approximately 1 minute after the Reactor Trip, the following conditions are noted:

- A Bus = zero volts
- B Bus = zero volts
- C1 Bus = zero volts
- D1 Bus = zero volts
- 1-3-H, D1 Bus Lockout
- Breaker AD213, SBODG to D2 BUS TIE BREAKER tripped open due to a D2 Lockout.

Which of the following strategies must be implemented to restore power to an essential 4160 volt bus?

- A. Start EDG1 to restore power to Bus C1.
- B. Start the SBODG to restore power to Bus C1.
- C. Start the SBODG to restore power to Bus D1.
- D. Start EDG 2 to restore power to Bus D1.

Answer: A

Explanation/Justification:

- A. Correct – The SBODG is not available due to lockout on Bus D2 which causes AD213 being open. EDG2 is not available due to D1 being locked out.
- B. Incorrect – The SBODG is not available due to lockout on Bus D2 indicated by breaker AD213 being open.
- C. Incorrect – The SBODG is not available due to lockout on Bus D2 indicated by breaker AD213 being open.
- D. Incorrect - EDG2 is not available due to D1 being locked out.

Sys #	System	Category	KA Statement
000055	Loss of Offsite and Onsite Power (Station Blackout)	EA1. Ability to operate and monitor the following as they apply to a Station Blackout:	Restoration of power with one ED/G
K/A#	EA1.06	K/A Importance	4.1
Exam Level	RO	Technical References:	DB-OP-02000 R26 Specific Rule 6 Step 6.1 RNO
References provided to Candidate	None	Level Of Difficulty: (1-5)	3
Question Source:	New	10 CFR Part 55 Content:	(CFR 41.7 / 45.5 / 45.6)
Question Cognitive Level:	High - Comprehension	Objective:	

14. The plant had been operating at 100% power

The following event occurs:

- Loss of off-site power
- All systems work as designed
- Natural circulation flow has been confirmed in accordance with DB-OP-06903, Plant Cooldown.

Which one of the following actions will raise the heat transfer rate from the Reactor Coolant System to the Steam Generators?

- A. Lowering Steam Generator steaming rates
- B. Lowering Steam Generator water levels
- C. Raising Steam Generator pressures
- D. Raising Steam Generator Auxiliary Feedwater flow rates

Answer: D

Explanation/Justification:

- A. Incorrect – Lowering SG Steaming rate will cause a rise in SG pressure and a lowering of differential temperature between the RCS and the SG, reducing the heat transfer rate.
- B. Incorrect – Lowering SG level will reduce the heat transfer surface area of the SG, reducing the overall heat transfer coefficient, reducing the heat transfer rate.
- C. Incorrect – A rise in SG pressure will lower the differential temperature between the RCS and the SG, reducing the heat transfer rate.
- D. Correct – Raising AFW Flow rates will provide additional cooling flow and level in the SG providing a larger heat sink inducing a higher heat transfer rate.

Sys #	System	Category	KA Statement
000056	Loss of Offsite Power	AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power:	Principle of cooling by natural convection
K/A#	AK1.01	K/A Importance 3.7	Exam Level RO
References provided to Candidate		None	Technical References: Lesson Plan OPS-SYS-I103.09 pages 23 & 24
Question Source:		BANK 36546	Level Of Difficulty: (1-5) 3
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content: (CFR 41.8 / 41.10 / 45.3)
Objective:			

15. The following conditions exist:

- The plant is operating at 100% power during the Winter.
- SW Pump 1 (Loop) is supplying Primary loads
- SW Pump 2 (Loop) is supplying Secondary loads
- SW Pump 3 breaker is racked out.

A rupture downstream of SW1399, SW HDR 1 TO TPCW HX occurs.

All automatic actions occur as designed.

Which ONE of the following describes the automatic response if any of Service Water and Circulating Water Systems?

- A. No Impact –SW1 continues to carry Primary Loads, SW 2 continues to carry Secondary Loads.
- B. SW 1 continues to carry Primary Loads, SW 1395 SW HDR 2 TO TPCW HX closes to isolate the break, and CT2955, TPCW HX SUPPLY FROM CIRC WTR opens to allow Circ Water to carry TPCW load.
- C. SW1 carries Train 1 Essential Loads. SW Train 2 carries Train 2 Essential Loads only. CT2955, TPCW HX SUPPLY FROM CIRC WTR opens to allow Circ Water to carry TPCW load.
- D. SW1 carries Train 1 Essential Loads. SW Train 2 carries Train 2 Essential Loads only. SW1395 close to isolate Secondary Header. CT2955, TPCW HX SUPPLY FROM CIRC WTR initially opens, but then closes to isolate the leak. TPCW Cooling is lost.

Answer: D

Explanation/Justification:

- A. Incorrect – The piping downstream of SW1399 and SW1395 is common, The breaks prevents either SW Line from supplying these loads. In addition, the break will cause the CT2955 Check Valves to sense low pressure causing a loss of Circ Water Supply as well. Plausible if candidate assumes the supplies are independent and a break on the out of service supply will not affect the in service supply.
- B. Incorrect – Plausible because Circ Water provide backup cooling for TPCW loads when SW supply is lost, but not when lost due to line break.
- C. Incorrect – Plausible because Circ Water provide backup cooling for TPCW loads when SW supply is lost, but not when lost due to line break.
- D. Correct –.The piping downstream of SW1399 and SW1395 is common, The breaks prevents either SW Line from supplying these loads. In addition, the break will cause the CT2955 Check Valves to sense low pressure causing a loss of Circ Water Supply to TPCW as well

Sys #	System	Category	KA Statement
000062	Loss of Nuclear Service Water	AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water:	The conditions that will initiate the automatic opening and closing of the SWS isolation valves to the nuclear service water coolers
K/A#	AK3.01	K/A Importance	3.2*
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	OS020 SH2 R45 Service Water CL6 & CL9
Question Cognitive Level:	High - Comprehension	Level Of Difficulty: (1-5)	3
Objective:		10 CFR Part 55 Content:	(CFR 41.4, 41.8 / 45.7)

16. Which ONE of the following describes the purpose of the Main Generator Under-excited Reactive Ampere Limiter (URAL)?
- A. Establishes a MINIMUM megawatt output (loading) for the main generator to prevent a reverse power condition.
 - B. Establishes a MINIMUM LAGGING Power Factor to maintain grid stability.
 - C. Prevents voltage Regulator output from RISING to a level which would cause excessive armature heating.
 - D. Prevents the Voltage Regulator output from LOWERING to a level which could cause the Main Generator to drop out of synchronization (slip poles) with the grid.

Answer: D

Explanation/Justification:

- A. Incorrect – Plausible if the Candidate believe excitation levels are related to minimum loading level to prevent a reverse power condition.
- B. Incorrect – Plausible if the Candidate believe excitation levels are related to Power Factor to ensure the limiting power factors for Generator Operation are observed.
- C. Incorrect – Plausible if the Candidates assume the limiter acts to reduce current flow and therefore heat.
- D. Correct - The under excited reactive ampere limit circuit acts to limit the amount of under excitation permitted on the generator. This limit is for the purpose of allowing the generator to be safely operated, continuously in an under excited condition, with sufficient margin between the excitation limit and the stability limit of the generator.

Sys #	System	Category	KA Statement
000077	Generator Voltage and Electric Grid Disturbances	AK1. Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances:	Under-excitation

K/A#	AK1.03	K/A Importance	3.3	Exam Level	RO
References provided to Candidate	None	Technical References:			System Description SD005 R4, Main Generator page 2-15
Question Source:	BANK 32205	Level Of Difficulty: (1-5)	2.5 - 3		
Question Cognitive Level:	Low - Fundamental	10 CFR Part 55 Content:	(CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)		
Objective:					

17. A loss of ALL feedwater has occurred. Both MU pumps are running.

Attempts are being made to restore feedwater to both SGs in accordance with DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture.

The following plant conditions exist:

- RCS pressure is 2200 psig and lowering
- The PORV (RC2A) is open.
- T-hot is 615 °F and rising in Loop 1 and 610 °F and rising in Loop 2.

Based on these conditions, what will be the status of PORV Block and PORV control switches?

The PORV block valve (RC 11) control switch will be (1) ; the PORV (RC2A) control switch will be (2) .

- A. (1) OPEN
(2) AUTO
- B. (1) OPEN
(2) LOCK OPEN
- C. (1) CLOSED
(2) AUTO
- D. (1) CLOSED
(2) LOCK OPEN

Answer: B

- Explanation/Justification:** At Davis-Besse, the beyond design bases Loss of all Feedwater event is mitigated via MU/HPI PORV Cooling. This question is related to control of the PORV and Flowpath through the PORV Block which is a fundamental knowledge.
- A. Incorrect – The position of the PORV Block is correct, but having the PORV in Auto will cause the valve to close when RCS Pressure reaches 2155 psig stopping the MU/HPI PORV Cooling Flowpath.
 - B. Correct – These positions are the DB-OP-02000 Attachment 4 position of the valves during MU/HPI PORV Cooling.
 - C. Incorrect – The position of the PORV Block is incorrect stopping flow and having the PORV in Auto will cause the valve to close when RCS Pressure reaches 2155 psig also stopping the MU/HPI PORV Cooling Flowpath.
 - D. Incorrect – The position of the PORV Block is incorrect, stopping the MU/HPI PORV Cooling Flowpath.

Sys #	System	Category	KA Statement
BW/E04	Inadequate Heat Transfer - Loss Of Secondary Heat Sink	EA1. Ability to operate and / or monitor the following as they apply to the (Inadequate Heat Transfer)	Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
K/A#	EA1.1	K/A Importance 4.4	Exam Level RO
References provided to Candidate		None	Technical References: DB-OP-02000 R26 Attachment 4 page 271
Question Source:		BANK 37388	Level Of Difficulty: (1-5) 3
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content: (CFR: 41.7 / 45.5 / 45.6)
Objective:			

18. Following a normal Reactor Trip, from full power operation, DB-OP-02000, RPS, SFAS, SFRCS, Trip or SG Tube Rupture, directs the operator to use the plant computer to record the following computer point voltage values:

- J213, J215, and J217 for J Bus Voltage

AND

- J221, J223, and J225 for K Bus Voltage

What is the reason for recording these voltages?

To complete the Surveillance required to verify compliance with _____

- A. TS 3.8.1, AC Sources Operating. The normal opening of ACB 34560 and 34561 following a reactor trip will impact operability of off site power sources.
- B. TS 3.8.2, AC Sources Shutdown. The normal transfer of A and B Buses to the Startup Transformers following a Reactor Trip, may have rendered the Shutdown AC Sources inoperable.
- C. TS 3.8.9 Distribution Systems Operating. The normal opening of ACB 34560 and 34561 following a reactor trip will impact operability of the Distribution system - Operating.
- D. TS 3.8.10, Distribution Systems Shutdown. The normal transfer of A and B Buses to the Startup Transformers following a Reactor Trip, may have rendered the Shutdown Distribution Systems inoperable.

Answer: A

Explanation/Justification:

- A. Correct - The opening of the Generator Output Breakers disrupts the normal ring bus configuration which may impact Off-Site Sources. As a result, this surveillance verifies the Off-Site Sources remain operable. Comprehension because candidate must know connect knowledge of ring bus voltage impact on Technical Specifications.
- B. Incorrect –TS 3.8.2 is only applicable in Modes 5 and 6. The candidate may select this TS since the Main Generator is shutdown.
- C. Incorrect - The opening of the Generator Output Breakers disrupts the normal ring bus and does not affect the Distribution Systems Operating which are the in plant electrical distribution.
- D. Incorrect –TS 3.8.10 is only applicable in Modes 5 and 6. The candidate may select this TS since the Main Generator is shutdown and power is being supplied from the Startup Transformers vice the Auxiliary Transformers.

Sys #	System	Category			KA Statement
BW/E10	Post-Trip Stabilization	Generic			Ability to use plant computers to evaluate system or component status.
K/A#	2.1.19	K/A Importance	3.9	Exam Level	RO
References provided to Candidate		None	Technical References:		DB-OP-02000 R26 Attachment 26 Page 400 Bases and Deviation Document for DB-OP-02000 R19 Attachment 26 page 501
Question Source:		New	Level Of Difficulty: (1-5)		2
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:		(CFR: 41.10 / 45.12)
Objective:					

19. The plant is in Mode 6.

All of the following conditions ensure adequate Shutdown Margin is maintained for Fuel Handling Operations except for _____?

- A. Minimum RCS Fill water temperature is 70 °F based on moderator temperature coefficient.
- B. Nuclear Instrumentation should be monitored closely during a fill with fuel in the core. If an unexplained rise in neutron count rate occurs, filling shall immediately stop and the cause determined.
- C. DB-OP-06904, Attachment 2, Isolation of Water Sources to the RCS is performed to Caution Tag required valves closed prior to Fuel Handling Operations.
- D. Fuel Assembly movements are to be performed in the prescribed order and to the locations specified by the Fuel Movement Sequence Sheets.

Answer: A

Explanation/Justification:

- A. Correct - 70°F is the minimum temperature for the Reactor Vessel for performing Hydrostatic Testing in accordance with DB-OP-06000, RCS Fill and Vent. It is not related to Shutdown Margin.
- B. Incorrect – Monitoring Nuclear Instrumentation for unexplained rise in neutron count rate could be indicative of insufficient shutdown margin.
- C. Incorrect – Potential RCS Dilution flowpaths are tagged to prevent inadvertent dilution of the Reactor Coolant System which would reduce shutdown margin.
- D. Incorrect – The sequence provided in the Fuel Movement Sequence sheets ensures shutdown margin is maintained as positive reactivity is added to the core. Improper placement of a number of assemblies could result in inadequate shutdown margin in that portion of the core.

Sys #	System	Category	KA Statement
000036	Fuel Handling Incidents	AK1. Knowledge of the operational implications of the following concepts as they apply to Fuel Handling Incidents :	SDM
K/A#	AK1.02	K/A Importance 3.4	Exam Level RO
References provided to Candidate		None	Technical References: DB-OP-06000 R26, RCS Fill and Vent Step 2.2.4, 2.2.6, DB-OP-06904 R42, Shutdown Operations Step 7.2 DB-OP-00030 R12, Fuel Handling Operations Step 4.3
Question Source: New		Level Of Difficulty: (1-5)	4
Question Cognitive Level: Low - Fundamental		10 CFR Part 55 Content:	(CFR 41.8 / 41.10 / 45.3)
Objective:			

20. The plant is operating at 100% power with all systems in normal alignment for this power level.

Rising Condenser Pressure is noted.

In accordance with DB-OP-02518, High Condenser Pressure, SFRCS is actuated using the Initiate and Isolate push buttons at _____ (1) _____ inches HGA in order to _____ (2) _____.

- A. (1)10
(2) ensure a source of feedwater remains available for the Steam Generators.
- B. (1)10
(2) protect the Condenser from Turbine Bypass Steam Flow
- C. (1)17
(2) ensure a source of feedwater remains available for the Steam Generators.
- D. (1)17
(2) protect the Condenser from Turbine Bypass Steam Flow

Answer: A

Explanation/Justification:

- A. Correct – The main Feedwater Pumps trip at 12.5 in HGA, but top of scale for Control Room indicators is 10 inches HGA. SFRCS is actuated at 10 inches to ensure a source of FW is available.
- B. Incorrect – The Turbine Bypass valves will auto close at 17 in HGA to protect the condenser.
- C. Incorrect – The reason is correct, but the setpoint is not correct. The main Feedwater Pumps trip at 12.5 in HGA.
- D. Incorrect - The Turbine Bypass valves will auto close at 17 in HGA to protect the condenser. SFRCS actuation is not required for this feature.

Sys #	System	Category	KA Statement
000051	Loss of Condenser Vacuum	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	4.2
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	DB-OP-02518 R06, High Condenser Pressure step 4.5 and Attachment 2.
Question Cognitive Level:	Low - Fundamental	Level Of Difficulty: (1-5)	2.5
Objective:		10 CFR Part 55 Content:	(CFR: 41.5 / 43.5 / 45.12)

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

- NO planned radioactive liquid releases are in progress.
- RE4686, Storm Sewer Outlet alarms and indicates above its HIGH alarm setpoint.

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

- A. Miscellaneous Liquid Radwaste System.
- B. Clean Liquid Radwaste System
- C. Demineralized Water System.
- D. Condensate Polishing System.

Answer: D

Explanation/Justification: Later

- A. Incorrect – Plausible because this is a radioactive system, however this system is located in the Auxiliary Building. Leakage from this system would go to a floor drain or sump and be transported to the Misc Waste Drain Tank, not the storm sewer. There is no connection between the Misc Waste Drain Tank and the Storm Sewer.
- B. Incorrect – Plausible because this is a radioactive system, however this system is located in the Auxiliary Building. Leakage from this system would go to a floor drain or sump and be transported to the Misc Waste Drain Tank, not the storm sewer. There is no connection between the Misc Waste Drain Tank and the Storm Sewer.
- C. Incorrect - Plausible because leak from this system can reach the storm sewer, but the system is not radioactive and would not cause an alarm on the Storm Sewer Radiation Monitor.
- D. Correct – With SG Tube Leak, activity levels in the condensate polishers will rise. Leakage from this system could reach the storm sewer via the Turbine Building Drains.

Sys #	System	Category	KA Statement
000059	Accidental Liquid Radwaste Release	AK2. Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following:	Radioactive-liquid monitors
K/A#	AK2.01	K/A Importance 2.7	Exam Level RO
References provided to Candidate		None	Technical References: DB-OP-02531 R19 Attachment 7 page 3 of 4
Question Source:		New	Level Of Difficulty: (1-5) 3
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content: (CFR 41.7 / 45.7)
Objective:			

22. Waste Gas Decay Tank 1 is being discharged to the station vent IAW DB-OP-03012, Radioactive Gaseous Batch Release. WG1821, Waste Gas To Station Vent Flow Control is being utilized for this batch release.

The following valid alarms and indications are received:

- RE1822A Waste Gas System Radiation Monitor alarms WARN & HIGH
- RE1822A Waste Gas System Radiation Monitor indicates offscale high

No automatic actions have occurred.

Based on these conditions, which of the following valves FAILED to automatically CLOSE?

1. WG1819, Waste Gas To Station Vent Isolation
2. WG1820, Waste Gas To Station Vent Isolation
3. WG1821, Waste Gas To Station Vent Flow Control
4. WG1836, Waste Gas Decay Tank 1 To Station Vent Control

- A. 1 & 2 only
- B. 1 & 4 only
- C. 2 & 3 only
- D. 3 & 4 only

Answer: A

Explanation/Justification:

- A. Correct – RE1822A trip should have caused both the Waste Gas to Station Vent Isolations to Close.
- B. Incorrect – Plausible because RE1822A trip should have caused WG1819 to close and since WG1836 a control valve is in the release flowpath, it is plausible that the controller should have closed as it well.
- C. Incorrect – Plausible because RE1822A trip should have caused WG1820 to close and since WG1821 a control valve is in the release flowpath, it is plausible that the controller should have closed it as well.
- D. Incorrect – Plausible because if the RE1822A trip would use a controller to provide isolation, it is logical that WG1821 and WG1836 would close to provide isolation.

Sys #	System	Category	KA Statement
000060	Accidental Gaseous-Waste Release	AA2. Ability to operate and / or monitor the following as they apply to the Accidental Gaseous Radwaste:	Valve lineup for release of radioactive gases
K/A#	AA2.06	K/A Importance	3.6*
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	OS-030 Sheet 1 (B-16) and Sheet 2 CL-1
Question Cognitive Level:	Low - Memory	Level Of Difficulty: (1-5)	3
Objective:		10 CFR Part 55 Content:	(CFR 41.7 / 45.5 / 45.6)

23. DB-OP-02012, STM GEN/SFRCS ALARM PANEL 12 ANNUNCIATOR procedure directs Radiation Protection to be notified to take local surveys of the Main Steam Line area when annunciator 12-1-A, MN STM LINE 1 RAD HI comes into alarm.

Which of the following is the reason for this direction?

- A. To evaluate for initiating conditions into RA-EP-02861, Radiological Incidents
- B. To obtain data to support leak rate calculation for DB-OP-02522, Small RCS Leaks
- C. To project off site doses from the Station Vent in accordance with RA-EP-02240, Offsite Dose Assessment.
- D. To verify affected SG diagnosis in accordance with DB-OP-02531, Steam Generator Tube Leak.

Answer: D

Explanation/Justification:

- A. Incorrect – Plausible because high radiation levels would be an initiating condition for the radiological incidents off normal procedure but this alarm is to support indications of a steam generator tube leak
- B. Incorrect – Plausible because there is a leak rate calculation that uses RE indications in the calculation but it uses steam jet air ejector discharge RE1003A & B.
- C. Incorrect – Plausible Steam Generator Tube Leaks will cause a release of radioactive material, but checking radiation levels in the Main Steam Line area will not allow determination of dose from the station vent.
- D. Correct – DB-OP-02012 directs checking symptoms in accordance with DB-OP-02531 along with alarm verification to access entry conditions into the steam generator tube leak abnormal procedure.

Sys #	System	Category	KA Statement
000061	ARM System Alarms	AK3. Knowledge of the reasons for the following responses as they apply to the Area Radiation Monitoring (ARM) System Alarms:	Guidance contained in alarm response for ARM system
K/A#	AK3.02	K/A Importance 3.4	Exam Level RO
References provided to Candidate		None	Technical References: DB-OP-02012 R10 Page 4 Step 3.4
Question Source:		New	Level Of Difficulty: (1-5) 3
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content: (CFR 41.5,41.10 / 45.6 / 45.13)
Objective:			

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- 24.
- Plant is in mode 6
 - The Refueling Canal is greater than 23 feet
 - Fuel Handling is in progress in Containment

Identify the ONE situation below that represents a condition that would require the handling of irradiated fuel in CTMT to be stopped:

- A. Equipment Hatch is removed. A Maintenance team is assigned to install the hatch but is not present.
- B. Maintenance removes SP 17B6, SG1 Main Steam Safety Valve in the Main Steam Line room AND SG 1 Secondary Manway is open for inspection.
- C. An operator is signed into the Containment Closure Control log for MU66D, Reactor Coolant Pump 1-2 Seal Injection Flow Isolation and is draining its piping for a Local Leakrate Test
- D. BOTH air lock doors of the CTMT personnel hatch are opened. An Operator is assigned to be responsible for closing ONE door.

Answer: B

Explanation/Justification:

- A. Incorrect – Plausible if the candidate does not know the equipment hatch can be open during fuel handling since it was previously required closed. They may also assume the team must be staged which is not correct.
- B. Correct– This creates a path from Containment to atmosphere. This is RO knowledge- to apply changes in plant conditions to immediately stop fuel handling operations.
- C. Incorrect – Plausible since this will create a path between CTMT and atmosphere but is administratively controlled by the CTMT Closure Control log
- D. Incorrect – Plausible if the Candidate knows this is outside of the EVS boundary but does not know this is allowed by procedure

Sys #	System	Category	KA Statement
000069	Loss of Containment Integrity	AA1. 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	Exam Level
		2.8	RO
References provided to Candidate		None	Technical References:
Question Source:		New	Level Of Difficulty: (1-5)
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:
Objective:			

DB-OP-06904 R42 Note 11.1, OS-008 SH 1 R35

3

(CFR 41.7 / 45.5 / 45.6)

25. The following plant conditions exist:

The plant is at 90% power

ICS is in a normal lineup

The following alarms occurs:

- 8-4-A, MFPT 1 TRIP alarms
- 10-1-A, MFP 1 DISCH HI PRESS TRIP alarms
- 13-4-C, DEAR STRG TK LVL
- 14-3-D, ICS MFP LOSS OR LO DEAR RUNBACK alarms

Main Generator load is lowering and stabilizes at approximately 700 MWe with #1 Deaerator level at 9 feet.

Based on these plant conditions, what procedures and associate actions are **required**?

- A. Trip the reactor and enter DB-OP-02000 in accordance with DB-OP-02014, MSR/ICS Alarm Panel 14 Annunciators
- B. Stabilize the plant at the current power level in accordance with DB-OP-06401, ICS Procedure, section for plant stabilization following a runback
- C. Place SG/RX Demand Station in HAND and perform runback to 55% power in accordance with DB-OP-02010, Feedwater Alarm Panel 10 Annunciators
- D. Place Feedwater Loop Demands and the Rod Control Panel in MANUAL and stabilize Reactor power and Tave in accordance with DB-OP-02526, Primary to Secondary Heat Transfer Upset

Answer: C

Explanation/Justification:

- A. Incorrect – Plausible because DB-OP-02014 directs tripping reactor if deaerator level approaches low off scale
- B. Incorrect – Plausible because DB-OP-06401 provides direction for stabilization following a runback, however reactor power was not reduced below runback setpoint for loss of a MFP.
- C. Correct – DB-OP-02010 for MFPT trip provides this direction for a MFPT Trip
- D. Incorrect – Plausible because DB-OP-02526, Primary to Secondary Heat Transfer Upset provides these directions for plant stabilization upon a plant upset

Sys #	System	Category	KA Statement
BW/A01	Plant Runback	AA2. Ability to determine and interpret the following as they apply to the (Plant Runback)	Facility conditions and selection of appropriate procedures during abnormal and emergency operations.
K/A#	AA2.1	K/A Importance 3.0	Exam Level RO
References provided to Candidate		None	Technical References: DB-OP-02010 R17 Page 4
Question Source:		BANK 75948	Level Of Difficulty: (1-5) 4
Question Cognitive Level:		High - Analysis	10 CFR Part 55 Content: (CFR: 43.5 / 45.13)
Objective:			

26. The Plant is at 50% power with a shutdown in progress. Entry into Mode 5 to perform maintenance is planned
The following annunciator alarms come in:

- (14-2-D) ICS/NNI 118VAC PWR TRBL
- (14-4-E) ICS INPUT MISMATCH
- (14-4-F) ICS INPUT TRANSFER

Other indications:

- Loss of blue light on SASS instrument's selector switches.
- SCR Bank, RC PRESSURE CONTROL, Hand/Auto Station Lights Both ON

Assuming the condition can not restored to normal which of the following actions must be taken?

- A. Operation of DHR Train 1 will be required vice the normal DHR Train 2 for cooldown.
- B. Control Atmospheric Vent Valves, ICS11A and ICS11B in manual for cooldown.
- C. Transfer the EHC Control Panel to manual for turbine control.
- D. Close MU 85, Letdown Flow Control Inlet Isolation to MU 6 to isolate Letdown.

Answer: A

Explanation/Justification:

- A. Correct – must diagnose loss of NNI X AC and identify appropriate response. Although the Decay Heat Cooler SFAS Valves, DH13A and DH14A, solenoids are DC powered their controls along with various DH Train 2 indications and alarms will be out of service since NNI X AC lost.
- B. Incorrect – Plausible because this is the response for loss of ICS power.
- C. Incorrect – Plausible because this is a response for loss of NNI-X DC power.
- D. Incorrect – Plausible because this is a response for loss of NNI-Y AC power.

Sys #	System	Category	KA Statement
BW/A02	Loss of NNI-X	AK2. Knowledge of the interrelations between the (Loss of NNI-X) 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#	AK2.2	K/A Importance	3.8
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	DB-OP-02532 R10 Step 4.1.13
Question Cognitive Level:	High - Comprehension	Level Of Difficulty: (1-5)	4
Objective:		10 CFR Part 55 Content:	(CFR: 41.7 / 45.7)

27. Following a Reactor Trip, a severe overcooling has caused a loss of Adequate Subcooling Margin.

Once the Reactor is confirmed as shutdown using the Immediate Operator Actions, based on the priorities provided by DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture, which the following is implemented **FIRST** to mitigate this event

- A. Section 5, Loss of Subcooling Margin
- B. Section 7, Overcooling
- C. Specific Rule 2, Loss of Subcooling Margin
- D. Specific Rule 4, Steam Generator Control

Answer: C

Explanation/Justification:

- A. Incorrect – As provided in the Bases and Deviation Document for DB-OP-02000, The hierarchy in DB-OP-02000 between the various sections is as follows: 1. Immediate Actions 2. Specific Rules 3. Procedure Sections 4. Attachments. Rules are implemented prior to sections.
- B. Incorrect – As provided in the Bases and Deviation Document for DB-OP-02000, The hierarchy in DB-OP-02000 between the various sections is as follows: 1. Immediate Actions 2. Specific Rules 3. Procedure Sections 4. Attachments. Rules are implemented prior to sections.
- C. Correct – As provided in the Bases and Deviation Document for DB-OP-02000, The hierarchy in DB-OP-02000 between the various sections is as follows: 1. Immediate Actions 2. Specific Rules 3. Procedure Sections 4. Attachments. Rules are implemented prior to sections. Rules are implemented in numerical order
- D. Incorrect - As provided in the Bases and Deviation Document for DB-OP-02000, The hierarchy in DB-OP-02000 between the various sections is as follows: 1. Immediate Actions 2. Specific Rules 3. Procedure Sections 4. Attachments. Rules are implemented prior to sections. Rules are implemented in numerical order

Sys #	System	Category	KA Statement	
BW/E13	EOP Rules	EA1. Ability to operate and / or monitor the following as they apply to the (EOP Rules)	Desired operating results during abnormal and emergency situations.	
K/A#	EA1.3	K/A Importance	3.4	Exam Level
References provided to Candidate		None	Technical References:	RO Bases and Deviation Document for DB-OP-02000 R19, Prioritization of DB-OP-02000 Sections.
Question Source:		New	Level Of Difficulty: (1-5)	2
Question Cognitive Level:		High - Analysis	10 CFR Part 55 Content:	(CFR: 41.7 / 45.5 / 45.6)
Objective:				

28. The plant is operating at 70% power with all systems in normal alignment for this power level. All four Reactor Coolant Pumps are in service.

Motor current for the 1-1 RCP is noted to be 290 amps.

(1) Which of the following describes the current status of the RCP motor current reading?

(2) What action is **required**, if any, for this condition?

- A. (1) This motor current reading is lower than normal.
(2) RCP 1-1 shutdown is required.
- B. (1) This motor current reading is lower than normal.
(2) RCP 1-1 shutdown is **NOT** required.
- C. (1) This motor current reading is higher than normal.
(2) RCP 1-1 shutdown is **NOT** required.
- D. (1) This motor current reading is higher than normal.
(2) RCP 1-1 shutdown is required.

Answer: C

Explanation/Justification:

- A. Incorrect – At normal operating RCS temperatures and pressures, normal RCP Motor Current is approximately 260 amps. The Operating Limits requiring shutdown are less than 200 amps or greater than 370 amps per DB-OP-02515, Reactor Coolant Pump and Motor Abnormal Operations.
- B. Incorrect – At normal operating RCS temperatures and pressures, normal RCP Motor Current is approximately 260 amps. The Operating Limits requiring shutdown are less than 200 amps or greater than 370 amps per DB-OP-02515, Reactor Coolant Pump and Motor Abnormal Operations
- C. Correct – At normal operating RCS temperatures and pressures, normal RCP Motor Current is approximately 260 amps. The Operating Limits requiring shutdown are less than 200 amps or greater than 370 amps per DB-OP-02515, Reactor Coolant Pump and Motor Abnormal Operations
- D. Incorrect – At normal operating RCS temperatures and pressures, normal RCP Motor Current is approximately 260 amps. The Operating Limits requiring shutdown are less than 200 amps or greater than 370 amps per DB-OP-02515, Reactor Coolant Pump and Motor Abnormal Operations

Sys #	System	Category	KA Statement
003	Reactor Coolant Pump System (RCPS)	A3. Ability to monitor automatic operation of the RCPS, including:	Motor current
K/A#	A3.02	K/A Importance	2.6
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	DB-OP-02515 R11, Reactor Coolant Pump and Motor Abnormal Operations step 4.6.1
Question Cognitive Level:	High - Comprehension	Level Of Difficulty: (1-5)	3
Objective:		10 CFR Part 55 Content:	(CFR: 41.7 / 45.5)

29. The following plant conditions exist:

- Mode 1 at 15% power

The following event occurs:

- RCP 1-1 is shutdown by the crew due to excessive vibrations.
- No other operator actions are taken.

Which one of the following represents the condition of the plant, once stabilized?

- A. Tave will be selected to Loop 1.
- B. Loop 1 FW flow will be 2.4 times greater than Loop 2 FW flow
- C. Loop 2 FW flow will be 2.4 times greater than Loop 1 FW flow.
- D. Tave will be selected to Loop 2.

Answer: D

Explanation/Justification:

- A. Incorrect – In accordance with DB-OP-02515 R11, RCP and Motor Abnormal Attachment 1 for Stopping a RCP Step 5. SASS will align Tave to the loop with 2 RCPS in service.
- B. Incorrect – Plausible because the normal response at 72% power when an RCP would be shutdown is for FW Flow to Loop with the highest RCS flow to be 2.4 time greater than the remaining loop. A trip from 15% with SG on Low Level limits negates flow control. The SG Will be on Level Control.
- C. Incorrect – Plausible because the normal response at 72% power when an RCP would be shutdown is for FW Flow to Loop 2 to be 2.4 time greater. A trip from 15% with SG on Low Level limits negates flow control. The SG Will be on Level Control
- D. Correct – In accordance with DB-OP-02515 R11, RCP and Motor Abnormal Attachment 1 for Stopping a RCP Step 5. SASS will align Tave to the loop with 2 RCPS in service.

Sys #	System	Category	KA Statement
003	Reactor Coolant Pump System (RCPS)	K3. Knowledge of the effect that a loss or malfunction of the RCPS will have on the following:	RCS
K/A#	K3.01	K/A Importance	3.7
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	DB-OP-02515 R11, RCP and Motor Abnormal Attachment 1 for Stopping a RCP
Question Cognitive Level:	High - Comprehension	Level Of Difficulty: (1-5)	3
Objective:		10 CFR Part 55 Content:	(CFR: 41.7 / 45.6)

30. The plant is operating at 100% power with all systems in normal alignment for this power level.
Makeup Pump 2 is in service.

Which of the following conditions would cause MU3971 Makeup Pump 2 Suction Valve to transfer from the Makeup Tank to the BWST assuming lock is **NOT** depressed for the valve?

- A. SFAS Level 2
- B. Makeup Tank Level less than 10 inches
- C. Loss of NNI X AC Power
- D. Loss of D2P and DBP

Answer: B

Explanation/Justification: KA Statement is for design features and/or interlocks on the letdown system for the letdown tank bypass valve. The closed valve for Davis Besse would be the MU Pump Suction Valves. These valves can be aligned to take a suction on the Makeup Tank or on the BWST. In the BWST position, the Makeup Tank is effectively bypassed.

- A. Incorrect – No automatic feature exists, however plausible because this action would protect BWST inventory for use by ECCS Systems.
- B. Correct – Low Makeup Tank Level of 10 inches will cause an auto transfer from the MU Tank to the BWST..
- C. Incorrect – Plausible because the MU3971 Auto Transfer from the BWST to the MU Tank is lost when NNI X AC power is lost.
- D. Incorrect – Plausible because the MU3971 Auto Transfer from the BWST to the MU Tank is lost when D2P and DBP power is lost

Sys #	System	Category	KA Statement
004	Chemical and Volume Control System	K4. Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following:	Control interlocks on letdown system (letdown tank bypass valve)
K/A#	K4.14	K/A Importance	2.8*
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	DB-OP-02002 R08 page 16 note 3.5
Question Cognitive Level:	Low - Memory	Level Of Difficulty: (1-5)	2
Objective:		10 CFR Part 55 Content:	(CFR: 41.7)

31. The following plant conditions exist:

- The plant is shutdown for maintenance in MODE 5.
- The RCS is vented to Containment Atmosphere.
- Shutdown cooling is provided by DHR Train 2.
- The RCS is 30 inches above the centerline of the RCS hotlegs.
- The plant has been shutdown for 10 days and RCS temperature is 100 °F.
- DH14A, Decay Heat Cooler 2 Outlet Valve is full open
- DH13A, Decay Heat Cooler,2 Bypass Valve is full closed

The following event occurs:

- DH14A, Decay Heat Cooler 2 Outlet Valve fully closes.

Based on these conditions, what is the time to RCS boil?

(Reference Attached)

- A. 24 minutes
- B. 31 minutes
- C. 35 minutes
- D. 155 minutes

Answer: C

Explanation/Justification:

- A. Incorrect – Plausible if the Candidate misreads the curve. 24 minutes would be the time to boil if the initial temperature was 140°F.
- B. Incorrect – Plausible if the Candidate misreads the curve. 31 minutes would be the time to boil if the initial temperature was 100°F but the candidate used the Low RCS level of 6 inches above Hot Leg Center Line. 30 inches is a low RCS level, but not for this curve.
- C. Correct – From DB-PF-06703 R20, Page 57 CC6.3c, the correct time to boil is 35 minutes.
- D. Incorrect – Plausible if the Candidate uses CC6.3d, time to boil to top of core which will be provided.

Sys #	System	Category	KA Statement
005	Residual Heat Removal System (RHRS)	K6. Knowledge of the effect of a loss or malfunction on the following will have on the RHRS:	RHR heat exchanger
K/A#	K6.03	K/A Importance 2.5	Exam Level RO
References provided to Candidate		DB-PF-06703 Rev 20 CC6.3.c and CC6.3.d	Technical References: DB-PF-06703 Rev 20 CC6.3.c and CC6.3.d
Question Source:		BANK 29427	Level Of Difficulty: (1-5) 3
Question Cognitive Level:		High - Application	10 CFR Part 55 Content: (CFR: 41.7 / 45.7)
Objective:			

32. Which ONE of the following is the reason a break in the 14 inch line between the reactor vessel and CF 30, CFT 1-2 TO REACTOR CHECK VALVE, will not result in exceeding the peak allowable cladding temperature of 2200 °F?
- A. Leak is at an elevation that will not uncover the core.
 - B. The Core Flood line flow restrictor at the RV limits the size of the leak from the RCS.
 - C. 14 inches is less than the size required to cause a large break LOCA.
 - D. One train of Core Flood meets all postulated loss of coolant accidents.

Answer: B

Explanation/Justification:

- A. Incorrect – Plausible if the candidate assumes the injection lines enter the vessel above the top of the core so the core won't uncover
- B. Correct – CFTs are not redundant therefore the flow restrictor limits leak size to allow one CFT to limit peak clad temperature
- C. Incorrect – Plausible if Candidate does not know what break size is classified as a large break LOCA
- D. Incorrect – Plausible because most safety systems have 2 fully redundant trains, only one of which is required to meet ECCS Criteria. Both Core Flood Tanks are required to meet ECCS Criteria.

Sys #	System	Category	KA Statement
006	Emergency Core Cooling System (ECCS)	K6. Knowledge of the effect of a loss or malfunction on the following will have on the ECCS:	Core flood tanks (accumulators)
K/A#	K6.02	K/A Importance	3.4
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	SD-040 R4 page 1-4 step 1.2.3.2
Question Cognitive Level:	Low - Fundamental	Level Of Difficulty: (1-5)	3
Objective:		10 CFR Part 55 Content:	(CFR: 41.7 / 45.7)

33. Given the following:

- The PORV is leaking
- The Quench Tank Circulating Pump is isolated for maintenance
- Reactor Coolant System pressure is 2155 psig

The Quench Tank Relief Valve failing open will cause level to rise in the:

- A. Waste Gas Surge Tank
- B. Reactor Coolant Drain Tank
- C. Containment Normal Sump
- D. Clean Waste Receiver Tank

Answer: C

Explanation/Justification:

- A. Incorrect – Plausible since the Quench Tank can be lined up to vent to the Waste Gas Header
- B. Incorrect – Plausible since a majority of the RCS relief valves relieve to the RCDT
- C. Correct – because the Quench Tank relieves to the Normal Sump
- D. Incorrect – Plausible since RCS discharge would be considered clean waste

Sys #	System	Category	KA Statement
007	Pressurizer Relief Tank/Quench Tank System (PRTS)	K3. Knowledge of the effect that a loss or malfunction of the PRTS will have on the following:	Containment
K/A#	K3.01	K/A Importance	3.3
References provided to Candidate		None	Exam Level
Question Source:		New	RO
Question Cognitive Level:		Low - Fundamental	Ops Schematic OS-001A Sheet 3
Objective:			Level Of Difficulty: (1-5)
			10 CFR Part 55 Content:
			3
			(CFR: 41.7 / 45.6)

34. Reactor Power is 75% and stable.

- Component Cooling Water (CCW) Pump 1 is running
- Component Cooling Water (CCW) Pump 2 is in standby
- Component Cooling Water (CCW) Pump 3 is aligned to side 1 as spare

The following occurs:

- CCW Pump 1 trips
- CCW Pump 2 does not start

The Reactor Operator attempts to start CCW Pump 1 and 2 from the control room and neither pump starts

Based on these conditions, identify the ONE statement below that identifies the **required** action(s) to be implemented

- A. Reduce Reactor Power to 72% in preparation for shutdown of an RCP.
- B. Trip the Reactor and trip all RCPs.
- C. Commence a Rapid Shutdown and monitor the Reactor Coolant Pumps.
- D. Monitor the Reactor Coolant Pumps and place the spare Component Cooling Water Pump 3 in service.

Answer: B

Explanation/Justification:

- A. Incorrect – Plausible since this would be the actions for loss of CCW to one RCP when reaching the required RCP trip parameters
- B. Correct – CCW abnormal procedure directs tripping the Reactor and all 4 RCPs in the event of the running and standby pumps being unable to be started
- C. Incorrect – Plausible because reducing power would reduce heat loading and the candidate may assume it is not required to trip the RCP or the Reactor until required trip parameters are reached.
- D. Incorrect – Plausible if the candidate assumes the spare CCW pump may be able to be placed in service prior to reaching required RCP trip parameters

Sys #	System	Category	KA Statement
008	Component Cooling Water System (CCWS)	A2. Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of CCW pump
K/A#	A2.01	K/A Importance 3.3	Exam Level RO
References provided to Candidate		None	Technical References: DB-OP-02523 R09 step 4.3.1 page 28
Question Source:		New	Level Of Difficulty: (1-5) 2.5
Question Cognitive Level:		High - Analysis	10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.13)
Objective:			

35. The Plant is in Mode 1

In accordance with Technical Specifications, which one of the following conditions requires action to be completed in **less than 30 minutes** to remain in compliance with Technical Specifications requirements?

- A. Pressurizer Level is greater than 228 inches.
- B. One Pressurizer Code Safety Valve setpoint is set greater than 2525 psig.
- C. No power is available to the Pressurizer Power Operated Relief Valve.
- D. The Block Valve for the Pressurizer Power Operated Relief Valve is closed.

Answer: B

Explanation/Justification:

- A. Incorrect – Plausible because when this condition is encountered in the simulator, the candidates take prompt action to restore Pressurizer Level to within limits.
- B. Correct – Per Technical Specifications Pressurizer Safety Valves to be Operable requires a setting of less than or equal to 2525 psig. A setpoint greater than 2525 renders the valve inoperable. Action is required within 15 minutes per TS 3.4.10 Condition A.
- C. Incorrect – Plausible since this condition renders the PORV inoperable and required action within one hour to close the PORV Block valve per TS 3.4.11 Condition B.
- D. Incorrect – Plausible since this condition would render the PORV inoperable and requires action per TS 3.4.11 Condition B.

Sys #	System	Category	KA Statement		
010	Pressurizer Pressure Control System (PZR PCS)	Generic	Knowledge of less than or equal to one hour Technical Specification action statements for systems.		
K/A#	2.2.39	K/A Importance	3.9	Exam Level	RO
References provided to Candidate		None	Technical References:	TS 3.4.10 Condition A (Amendment 279)	
Question Source:		New	Level Of Difficulty: (1-5)		3.5 - 4
Question Cognitive Level:		Low - Memory	10 CFR Part 55 Content:		(CFR: 41.7 / 41.10 / 43.2 / 45.13)
Objective:					

36. Essential Bank 2 Pressurizer heater bank control switch is in the ON position. If RCS pressure is stable at the normal operating point, and Pressurizer level decreases to 37", which ONE of the following explains the status of the Essential Bank 2 heaters?

The heater bank is _____

- A. energized because manual control overrides the Pressurizer low-low level heater cutoff.
- B. de-energized because the Pressurizer low-low level heater cutoff overrides manual control.
- C. energized because Pressurizer level is above the low-low level heater cutoff setpoint
- D. de-energized because normal RCS pressure is above the heater bank cycle setpoint.

Answer: A

Explanation/Justification:

- A. Correct – In automatic, the design of the Pressurizer Heaters removes power on LOW LOW pressurizer Level (40 inches). Operating the Pressurizer in manual (ON) overrides this design feature.
- B. Incorrect – Plausible if the candidate does not understand that the "ON" position for the heaters overrides the Low level cutoff.
- C. Incorrect – Plausible if the candidate does not know the setpoint for low low pressurizer level and thinks it is less than 40 inches. A pressurizer level of 37 inches is still above the top of all Pressurizer heaters
- D. Incorrect – Plausible if the candidate does not understand the interlock but knows this bank of heaters is off at normal RCS Pressure.

Sys #	System	Category	KA Statement
010	Pressurizer Pressure Control System (PZR PCS)	K4. Knowledge of PZR PCS design feature(s) and/or interlock(s) which provide for the following:	Prevention of uncovering PZR heaters
K/A#	K4.02	K/A Importance	3.0
References provided to Candidate	None	Exam Level	RO
Question Source:	BANK 37164	Technical References:	DB-OP-06003 R29 Attachment 7 PZR Heater Control Panel Placard DB-OP-02513 R11 4.6.4 RNO
Question Cognitive Level:	High - Comprehension	Level Of Difficulty: (1-5)	3
Objective:		10 CFR Part 55 Content:	(CFR: 41.7)

37. The following plant conditions exist:

The plant is operating at 100% power.
RPS Channel 1 is in Manual Bypass.

The following event occurs:

- RCS Pressure exceeds the RPS High RCS Pressure Trip setpoint
- RPS Channels 2 and 4 Trip
- RPS Channel 3 fails to trip

How will the CRD Breakers respond to these conditions **AND** what operator actions will be required?

- A. No CRD Trip Breakers will open, manually trip the reactor and enter DB-OP-02000, RPS, SFAS, SFRCS TRIP, or SG Tube Rupture.
- B. Only the “A” and “C” breakers will open, place Rod control to MANUAL.
- C. Only the “B” and “D” breakers will open, place Rod control to MANUAL.
- D. All CRD Trip Breakers will open, enter DB-OP-02000, RPS, SFAS, SFRCS TRIP, or SG Tube Rupture.

Answer: D

Explanation/Justification: Note: Actuation of an RPS Channel trips the respective CRD Breaker by removing DC control power from the breaker. This DC Control Power is internally generated in the Reactor Protective System, not supplied from an external source..

- A. Incorrect – Plausible if the candidate believes the logic of 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 2. If only a single actuation channel trips, a full SFRCS actuation does not occur. If candidate believes the breakers should have opened, manually tripping would be the correct action.
- B. Incorrect – Plausible if the Candidate does not understand the relationship between RPS Channels and CRD Breakers. RPS Channels 1, 2, 3, 4, supply CRD Breaker B, A, D, C respectively. With any CRD breaker, rod control is placed in manual to prevent rod motion.
- C. Incorrect – Plausible if the Candidate does understand the relationship between RPS Channels and CRD Breakers. RPS Channels 1, 2, 3, 4, supply CRD Breaker B, A, D, C respectively but does not understand the logic of RPS as noted in distractor 1 above. With any CRD breaker, rod control is placed in manual to prevent rod motion.
- D. Correct – IAW DB-OP-06403, Attachment 4, Page 59, Relays KB and KD remain energized and their correstonding contacts in each RPS cabinet remain closed, however the KA and KC relays de-energize. The corresponding KA and KC contacts open in each cabinet interrupting DC Control Power to the associated CRD Breaker and causing the breakers to trip. Entry conditions have been met for entering DB-OP-02000.

Sys #	System	Category	KA Statement
012	Reactor Protection System (RPS)	A2. Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of dc control power
K/A#	A2.07	K/A Importance	3.2*
References provided to Candidate		None	Exam Level
Question Source:		New	RO
Question Cognitive Level:		High - Analysis	DB-OP-06403 R19, Attachment 4, Page 59
Objective:			Level Of Difficulty: (1-5) 2.5
			10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.5)

38. The plant was operating at 75% power with all systems in normal alignment for this power level.

The following plant conditions **NOW** exist:

- SG 1 pressure is 880 psig.
- SG 2 pressure is 150 psig.
- Reactor Coolant System pressure is 1700 psig and steady.
- Reactor Coolant System temperature is 530 °F and steady.
- Containment pressure is 19 psia and lowering.
- All systems function as designed.

With NO operator action, what will be the control level setpoint for SG 1?

- A. 49 inches
- B. 55 inches
- C. 124 inches
- D. 130 inches

Answer: D

Explanation/Justification:

- A. Incorrect – Plausible if the Candidate does not diagnose an SFAS level 2 trip or SG 2 isolation on low pressure since this is normal level for a SFRCS actuation without SG low pressure trip
- B. Incorrect – Plausible if the Candidate knows level is controlled at 55" by the opposite side pump on SG low pressure SFRCS trip but doesn't diagnoses the SFAS level 2 trip or know setpoint is raised to high on a SA2
- C. Incorrect – Plausible since this is the normal level for SFRCS actuation on an SFAS 2 with no SG isolation. Diagnoses SA2 but not SFRCS low pressure.
- D. Correct – SFAS level 2 on CTMT pressure (18.7psia) will raise the setpoint to high and AFP 2 will control level at 130" with AFP 1 setpoint at 124" due to SG 2 SFRCS low pressure trip (630psig)

Sys #	System	Category	KA Statement
013	Engineered Safety Features Actuation System (ESFAS)	K1. Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems:	AFW System
K/A#	K1.07	K/A Importance 4.1	Exam Level RO
References provided to Candidate	None	Technical References:	OS-17A SH1 R26 CD-1, DBBP-TRAN-0034 R06 page 8&9
Question Source:	New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:	High - Comprehension	10 CFR Part 55 Content:	(CFR: 41.2 to 41.9 / 45.7 to 45.8)
Objective:			

39. The plant is in Mode 1 at 100% power with Service Water Returns aligned to the Cooling Tower.

A Large Break Loss of Coolant Accident occurs.

All equipment responds as designed.

Which of the Service Water System conditions below would result in inadequate service water flow to the Containment Air Cooler to remove the heat from Containment for this design bases event?

- A. A loss of air to the in service CAC Outlet Temperature Control Valves.
- B. The Service Water non-seismic header ruptures.
- C. Train 1 SW flow is inadvertently aligned to CAC 1 and CAC3.
- D. SW 2931, CLNG TOWER MAKEUP is inadvertently closed.

Answer: C

Explanation/Justification:

- A. Incorrect – Plausible because the loss of air to the CAC Outlet Temperature Control Valves will cause the valves to fail open and allow full flow, but this is the expected condition for the LOCA event when SFAS Actuates.
- B. Incorrect – Plausible because a rupture of the non-seismic header would divert Service Water flow from essential component, however in this condition, SW1395 and SW1399 would isolate the non-essential header.
- C. Correct – In modes 1,2 and 3 service water must be isolated to the spare CAC to ensure flow through the two in service CACs is adequate to support post LOCA cooling requirements
- D. Incorrect – Plausible since this is the inservice SW return flowpath, however the SW return flowpaths for the Intake Structure and the Forebay would open on high return pressure of 50 psig to provide a safety grade flowpath.

Sys #	System	Category	KA Statement
022	Containment Cooling System (CCS)	A1. Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including:	Cooling water flow
K/A#	A1.04	K/A Importance 3.2	Exam Level RO
References provided to Candidate	None	Technical References:	DB-OP-06016 R29 Step 2.2.4 page 4
Question Source:	New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:	High - Analysis	10 CFR Part 55 Content:	(CFR: 41.5 / 45.5)
Objective:			

40. The Plant is at 50% Power with #1 Makeup Pump out of service.

The following occurs:

ANNUNCIATOR ALARMS:

- SEAL INJ FLOW LO, 6-5-C
- SEAL INJ TOTAL FLOW, 6-6-C
- PZR LVL LO, 4-2-E

CTRM INDICATIONS:

- #2 Makeup Pump discharge pressure reads 0 psig
- MU32, PZR LEVEL CONTROL, indicates 100% demand
- MU19, RCP SEAL INJ FLOW CONTROL, indicates 100% demand
- PZR level is 155 inches

The crew has entered the appropriate Abnormal Operating Procedure.

What actions are **required** based on plant conditions?

- A. Trip the Reactor. GO TO DB-OP-02000, RPS, SFAS, SFRCS TRIP, or SG Tube Rupture.
- B. Commence a plant shutdown. GO TO DB-02504 Rapid Shutdown.
- C. Trip Reactor Coolant Pumps 1-2 and 2-2. GO TO DB-OP-02515, Reactor Coolant Pump and Motor Abnormal Operation.
- D. Place MU32 in hand. GO TO DB-OP-02513 Pressurizer Abnormal Operation.

Answer: A

Explanation/Justification:

- A. Correct – Minimum level for Tave 582°F is 160 inches below which requires tripping the Reactor. This is the mitigating strategy for a loss of all Makeup Pumps. Tripping at 160 inches will ensure a minimum inventory is maintained in the Pressurizer and then depressurize to allow use of HPI to recover Pressurizer level.
- B. Incorrect – Plausible if Candidate knows a shutdown is required but does not recognize PZR level less than 160 inches requires a reactor trip.
- C. Incorrect – Plausible because MU Pump 2 is lost and reactor power is less than the 55% setpoint when 1 RCP is running in each loop.
- D. Incorrect – Plausible because MU32 is 100% open and a pressurizer control failure may be diagnosed

Sys #	System	Category	KA Statement		
004	Chemical and Volume Control System	Generic	Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.		
K/A#	2.4.4	K/A Importance	4.5	Exam Level	RO
References provided to Candidate		None	Technical References:		DB-OP-02512 R14 step 4.1.3 page 8
Question Source:		New	Level Of Difficulty: (1-5)		3
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:		(CFR: 41.10 / 43.2 / 45.6)
Objective:					

41. The plant is operating at 100% power

- EDG 2 has been started in accordance with DB-OP-06316, Emergency Diesel Generator Operating Procedure using the IDLE Start pushbutton and is running at 450 rpm.

The following event occurs:

- All Undervoltage Relays on D1 are actuated.
- All proper automatic actions occur.

Which of the following automatic and/or manual actions will be **required** to re-energize D1 bus?

- A. The EDG field will flash automatically.
The EDG will accelerate to 900 RPM,
then AD101 EDG 2 Output Breaker will auto close.
- B. The EDG will accelerate to 900 RPM.
The Idle Release Pushbutton must be depressed to flash the EDG field,
then AD101, EDG 2 Output Breaker must be manually closed.
- C. The operator must depress the Idle Release Pushbutton before EDG 2 will accelerate to 900 RPM.
The EDG field will automatically flash,
then AD101, EDG 2 Output Breaker will auto close.
- D. The operator must manually raise EDG 2 speed to 900 rpm.
The EDG field will flash automatically and the EDG output breaker,
AD101 EDG 2 Output Breaker must be manually closed.

Answer: A

Explanation/Justification:

- A. Correct - The Idle Start/Stop Circuitry inhibits the voltage regulator by applying field shorting. An automatic start signal will release the Idle Start relay, accelerate the EDG, and enable the voltage regulator. The EDG output breaker would then auto close to restore power to D1 Bus.
- B. Incorrect - The Idle Start/Stop Circuitry inhibits the voltage regulator by applying field shorting. An automatic start signal will release the Idle Start relay, accelerate the EDG, and enable the voltage regulator. Depressing the Idle release will not be necessary to flash the field.
- C. Incorrect - The Idle Start/Stop Circuitry inhibits the voltage regulator by applying field shorting. An automatic start signal will release the Idle Start relay, accelerate the EDG, and enable the voltage regulator. Depressing the Idle release will not be necessary to accelerate the EDG.
- D. Incorrect - The Idle Start/Stop Circuitry inhibits the voltage regulator by applying field shorting. An automatic start signal will release the Idle Start relay, accelerate the EDG, and enable the voltage regulator. Operator action to raise EDG speed will not be required. In addition, Operator action will not be necessary to close the EDG Output Breaker. The EDG output breaker would auto close to restore power to D1 Bus.

Sys #	System	Category	KA Statement
062	AC Electrical Distribution System	K1. Knowledge of the physical connections and/or cause/effect relationships between the ac distribution system and the following systems:	ED/G
K/A#	K1.02	K/A Importance	Exam Level
		4.1	RO
References provided to Candidate		None	Technical References:
			DB-OP-06316 R54, EDG Operating Procedure Step 2.2.12.

Question Source:	New	Level Of Difficulty: (1-5)	4
Question Cognitive Level:	High - Comprehension	10 CFR Part 55 Content:	(CFR: 41.2 to 41.9)
Objective:			

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42. The Plant has experienced a Loss Of Coolant Accident with SFAS Levels 1 and 2 initiating. All equipment responded as designed.

Subsequently, a Loss of Off-Site Power occurs and Bus F1 is lost when the #2 EDG starts and restores power to Bus D1.

The Loss Of Coolant Accident continues to degrade with SFAS Levels 3, and 4 initiating

Without Operator action, what is the current status of the Containment Spray Train 2?

Containment Spray Pump 2 is _____ (1) _____.

CS1531, Containment Spray 2 Discharge Valve is _____ (2) _____.

- A. (1) Off
(2) Closed
- B. (1) Running
(2) Closed
- C. (1) Off
(2) Open
- D. (1) Running
(2) Open

Answer: C

Explanation/Justification:

- A. Incorrect – Plausible if the candidate believes CS1531 opens on the SFAS Level 4 actuation, but does realize the Containment Spray Pumps is a 480 volt load and is lost when F1 is lost.
- B. Incorrect – Plausible if the candidate believes CS1531 opens on the SFAS Level 4 actuation, but fails to realized the Containment Spray Pump is a 480 load as unlike the other SFAS Actuated Pumps that are supplied from 4160 essential power.
- C. Correct CS1531 opens on the SFAS Level 2 actuation and is therefore unaffected when F1 loses power. Since the Containment Spray pump is supplied from F1, it will be off when F1 is lost.
- D. Incorrect – Plausible is the candidate understands CS1531 opens on the SFAS Level 2 actuation, but fails to realized the Containment Spray Pump is a 480 load as unlike the other SFAS Actuated Pumps that are supplied from 4160 essential power.

Sys #	System	Category	KA Statement
026	Containment Spray System (CSS)	K2. Knowledge of bus power supplies to the following:	MOVs
K/A#	K2.02	K/A Importance	Exam Level
		2.7*	RO
References provided to Candidate		None	Technical References: OS-005 R12
Question Source:		New	Level Of Difficulty: (1-5)
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:
Objective:			3 (CFR: 41.7)

43. A Plant Startup is in progress. The Main Turbine has been synchronized to the grid.

- Reactor power is 20% and stable
- Power range NI8 calibration is in progress
- Reactor Demand is in Manual
- Diamond Rod Control Panel is in Auto

A short time later, the following indications are present:

- Reactor Power is rising
- Megawatts are lowering
- Feedwater is rising
- RCS pressure is lowering

Which of the following explains why reactor power is increasing?

- A. Positive reactivity is being added due to lowering Tave
- B. An Undesired Rod withdrawal is in progress
- C. ICS is raising power in response to lowering megawatts
- D. I&C has placed NI8 in Test Operate while it was the highest indicating NI

Answer: A

Explanation/Justification:

- A. Correct – A steam leak is in progress per DB-OP-02525, Steam Leaks. A lowering Tave will add positive reactivity with a negative moderator coefficient.
- B. Incorrect – Plausible because an undesired rod withdraw will raise power but would not include the listed symptoms
- C. Incorrect – Plausible because raising power would normally increase megawatts but ULD output tracks megawatts in manual
- D. Incorrect – Plausible because ICS power selects the highest auctioneered power and power would increase if the ULD was in auto

Sys #	System	Category	KA Statement
039	Main and Reheat Steam System (MRSS)	K5. Knowledge of the operational implications of the following concepts as they apply to the MRSS:	Effect of steam removal on reactivity
K/A#	K5.08	K/A Importance 3.6	Exam Level RO
References provided to Candidate	None	Technical References:	DB-OP-02525 R10 Page 5
Question Source:	New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:	Low - Fundamental	10 CFR Part 55 Content:	(CFR: 441.5 / 45.7)
Objective:			

44. The following plant conditions exist:

- The reactor is operating at 50% rated power.
- One main feedwater pump (MFP) is operating in AUTOMATIC.
- All Feedwater Control Valves are in AUTOMATIC.
- ICS is in full AUTOMATIC mode.

Which one of the following describes feedwater flow control by ICS following a manual reactor trip?

- A. Places the MFP at a constant target speed and immediately controls the Feedwater Control Valves position based on feedwater flow error.
- B. Places the MFP at a constant target speed and immediately controls the Feedwater Control Valves position based on SG level error.
- C. Runs the MFP to a target speed which is then modified by SG feedwater flow error and positions Feedwater Control Valves to a target position until a 2.5 minute timer expires.
- D. Runs the MFP to a target speed which is then modified by SG level error and positions Feedwater Control Valves to a target position until SGs are at low level limits or a 2.5 minute timer expires.

Answer: D

Explanation/Justification:

- A. Incorrect – Rapid Feedwater Reduction will actuate. Feedwater Control valves will control on SG level error, not Feedwater flow error.
- B. Incorrect – Rapid Feedwater Reduction will actuate. Feedwater Control valves will control on SG level error, but a timer operates to allow SG level to lower to low level limits
- C. Incorrect – Rapid Feedwater Reduction will actuate. Feedwater Control valves will control on SG level error, not Feedwater flow error.
- D. Correct – With full automatic ICS operation and SG not initially on low level limit control, a reactor trip will caused the MFP to go to target speed, the SU SG Level controls to target position until SG on Low Level Limit or 2.5 minute timer times out.

Sys #	System	Category	KA Statement
059	Main Feedwater (MFW) System	A1. Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including:	Feed Pump speed, including normal control speed for ICS
K/A#	A1.07	K/A Importance	2.5*
Exam Level	RO	References provided to Candidate	None
Technical References:	Lesson Plan OPS-SYS-I512 R06 page 13 & 14	Level Of Difficulty: (1-5)	3.5
Question Source:	BANK 38076	10 CFR Part 55 Content:	(CFR: 41.5 / 45.5)
Question Cognitive Level:	Low - Memory	Objective:	

45. The plant is at 70% Power.

- Annunciator 10-1-C, MFPT 1 Lube Oil Press Lo alarms
- PI1206, Header pressure indicates 3.6 Psig
- Neither the Preferred or Standby #1 MFPT Main Oil Pump are running.
- Both MFP Turbines are operating at approximately 4400 rpm.

The plant remains stable at 70% power.

Which of the following actions are **required**?

- A. Trip #1 MFPT only if PI1206, #1 MFPT Lube Oil Header pressure lowers to 3 psig.
- B. Start MFPT 1 Emergency Bearing Oil Pump and then Trip #1 MFPT.
- C. Start the Motor Driven Feedwater Pump and Trip #1 MFPT.
- D. Reduce Reactor power to 60% in preparation for loss of #1 MFPT.

Answer: B

Explanation/Justification:

- A. Incorrect – Plausible if the MFPT emergency bearing oil pump auto started and 3 psig was the MFPT trip setpoint
- B. Correct – DB-OP-02010 directs starting the MFPT emergency bearing oil pump and tripping MFPT 1 if bearing header goes below 4.0 psig which is the auto trip setpoint
- C. Incorrect – Plausible since MFPT 1 should be tripped starting the MDFP will provide additional inventory to the SG which may facilitate maintaining SG Level.
- D. Incorrect – Plausible since 60% is the high discharge pressure of MFPT runback target.

Sys #	System	Category	KA Statement		
059	Main Feedwater (MFW) System	A4. Ability to manually operate and monitor in the control room:	MFW turbine trip indication		
K/A#	A4.01	K/A Importance	3.1*	Exam Level	RO
References provided to Candidate		None	Technical References:	DB-OP-02010 R17 pages 8 & 9	
Question Source:		New	Level Of Difficulty: (1-5)		3
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:		(CFR: 41.7 / 45.5 to 45.8)
Objective:					

46. The Plant is at 100% power.
- The zone 3 Equipment Operator reports the piping at AF608, Auxiliary Feedwater to Steam Generator Line 1 Stop, is hot to the touch and Auxiliary Feedwater (AFW) Train 1 is Steam Bound

Which of the following statements is the correct impact of this condition on Emergency Feedwater and IAW DB-OP-06233, Auxiliary Feedwater System, what action is **required**?

- A. AFW Train 1 is Inoperable. Immediately vent and refill AFW Train 1. AFW Train 2 and the Motor Driven Feed Pump remain operable.
- B. AFW Train 1 and the Motor Driven Feed Pump are Inoperable. Initiate actions to commence a Reactor shutdown within one hour.
- C. AFW Train 1 and AFW Train 2 are Inoperable. Start the Motor Driven Feed Pump in the Auxiliary Feedwater mode to condense the steam bubble at AF608.
- D. AFW Train 1 is Inoperable and AFW Train 2 and the Motor Driven Feed Pump will be rendered Inoperable due to closing AF608. Take action immediately to restore Operability

Answer: D

- Explanation/Justification:** Note: At DB, the Auxiliary Feedwater and Main Feedwater System do not share physical connections since they feed the Steam Generators via separate headers. In order to use the KA, a back leakage from the Steam Generator question was used into the Auxiliary Feedwater System.
- A. Incorrect – Correct that AFW train 1 is inoperable. Correct action to fill and vent AFW train 1. Incorrect that AFW Train 2 and MDFP must remain operable while filling and venting AFW Train 1. Closing AF608 is required to perform the fill and vent, and this will render AFW Trains 1 and 2 and the MDFP inoperable.
 - B. Incorrect –Plausible because a low SG Pressure condition will align AFW Train 1 and 2 to Feed SG 1 via AF608. Also, the MDFP could be used to provide cool water from the Condensate Storage Tank to mitigate this condition.
 - C. Incorrect – Plausible because without a low SG Pressure condition, only AFW Train 1 and the MDFP would use the AF608 flowpath
 - D. Correct – Because a low SG Pressure condition, AFW Train 1 and 2 and the MDFP could use the AF608 flowpath. As a result, all three would be inoperable with AFW Train 1 steam bound.

Sys #	System	Category	KA Statement
061	Auxiliary / Emergency Feedwater (AFW) System	A2. Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Back leakage of MFW
K/A#	A2.06	K/A Importance 2.7	Exam Level RO
References provided to Candidate	None	Technical References:	DB-OP-06233 R35 Steps 2.1.6 & 4.9.5.a.3 and TS 3.7.5 Condition E
Question Source:	New	Level Of Difficulty: (1-5)	4
Question Cognitive Level:	High - Comprehension	10 CFR Part 55 Content:	(CFR: 41.5 / 43.5 / 45.3 / 45.13)
Objective:			

47. (1) What is the power supply to Inverter YVA when the static transfer switch is in Normal
- (2) What is the power supply if the static transfer switch transfers to Alternate?
- A. (1) Non-essential 480 VAC
(2) Non-essential 120 VAC
- B. (1) Non-essential 480 VAC
(2) Essential 120 VAC
- C. (1) 250 VDC
(2) Non-essential 120 VAC
- D. (1) 250 VDC
(2) Essential 120 VAC

Answer: C

Explanation/Justification: Inverter YVA supplies the uninterruptable 120 vdc bus YAU.

- A. Incorrect – Plausible because YVA does not use essential power. Both choices use non-essential power.
- B. Incorrect – Plausible because YAU is an important plant power supply for fire protection, communications, ICS, NNI etc. It is logical this power would be essential when transferred to alternate.
- C. Correct – This is the configuration for Inverter YVA as provide in the System Operating Procedure for normal lineup
- D. Incorrect – Plausible because YAU is an important plant power supply for fire protection, communications, ICS, NNI etc. It is logical this power would be essential when transferred to alternate. Also 250 VDC is feed from the Safety Related Station Batteries 1P and 1N.

Sys #	System	Category	KA Statement
062	AC Electrical Distribution System	A3. Ability to monitor automatic operation of the ac distribution system, including:	Operation of inverter (e.g., precharging synchronizing light, static transfer)
K/A#	A3.04	K/A Importance	Exam Level
		2.7	RO
References provided to Candidate		None	Technical References: DB-OP-06319 R25, page 2 & 192
Question Source:		BANK 32192	Level Of Difficulty: (1-5)
			4
Question Cognitive Level:		Low - Memory	10 CFR Part 55 Content:
			(CFR: 41.7 / 45.5)
Objective:			

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48. The Plant has experienced a complete loss of AC Power. Performance of DB-OP-02521, Loss of AC Bus Power Sources, is in progress.

At 1 hour following the beginning of the event AC power is still lost with required actions of DB-OP-02521 to reduce battery discharge rate completed.

(1) What is the current DC alignment

AND

(2) How long will it be before DC power is no longer available?

- A. (1) Batteries 1P, 1N, 2P and 2N are in service.
(2) less than 2 hours
- B. (1) Batteries 1P and 1N are in service. Batteries 2P and 2N are in standby.
(2) approximately 8 hours
- C. (1) Battery 1N is in service. Batteries 1P, 2P and 2N are in standby.
(2) approximately 16 hours.
- D. (1) Battery 1P is in service. Batteries 1N, 2P and 2N are in standby.
(2) greater than 24 hours

Answer: D

Explanation/Justification: DC Bus Load shedding is performed to reduce Discharge Rate and therefore extend battery life.

- A. Incorrect – Plausible because the batteries are designated as having a 1500 amp-hour rating based on an 8 hour discharge rate.
- B. Incorrect – Plausible if it is assumed there are 250V loads required to remain energized following load shedding
- C. Incorrect – Plausible since one battery (1P) will remain in service and 32 hours is a multiple of 8.hours
- D. Correct – DB-OP-02521 will direct reducing 1P to minimum required loading and completely unloading the remaining three batteries to be used in series to extend battery capacity. This is a new configuration that extends the time to meet minimum DC Bus Loads.

Sys #	System	Category	KA Statement
063	DC Electrical Distribution System	A1. 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:	DB-OP-02521 R20 Attachment 17 page 128	Question Source:	New
Level Of Difficulty: (1-5)	4	Question Cognitive Level:	High - Comprehension
10 CFR Part 55 Content:	(CFR: 41.5 / 45.5)	Objective:	

49. Reactor Power is 100% with all systems in a normal alignment.

The following events have occurred:

All AC power has been lost.

Following DC Bus load shed per DB-OP-02521, Loss of AC Bus Power Sources, AFW Pump 1 is in service supplying SG 1.

With no operator action, as Battery voltage lowers toward zero, what will be the effect on SG 1 Level?

SG 1 level will:

- A. Lower due to AFW Pump Discharge Target Rock valve failing closed
- B. Lower due to AFW Pump speed going to the low speed stop
- C. Rise due to AFW Pump Discharge Target Rock valve failing open
- D. Rise due to AFW Pump speed going to the high speed stop

Answer: C

Explanation/Justification: Loss of all AC Power

- A. Incorrect – Plausible because SG level will lower if the target rock valve were to fail closed
- B. Incorrect – Plausible because SG level would lower if the turbine went to its low speed stop
- C. Correct – Target rock fails open on low voltage and SG will have full flow.
- D. Incorrect – Plausible because SG level will rise if the turbine went to its high speed stop

Sys #	System	Category	KA Statement		
063	DC Electrical Distribution System	A4. Ability to manually operate and/or monitor in the control room:	Battery voltage indicator		
K/A#	A4.02	K/A Importance	2.8*	Exam Level	RO
References provided to Candidate		None	Technical References:	DB-OP-02521 R20, Att 17 page 130	
Question Source:		BANK 79886	Level Of Difficulty: (1-5)		3
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:		(CFR: 41.7 / 45.5 to 45.8)
Objective:					

50. The following plant conditions exist:

- Emergency Diesel Generator 1 monthly surveillance test is in progress
- Emergency Diesel Generator 1 is paralleled with the C1 bus

The following event occurs:

- A Safety Features Actuation System Level 2 signal occurs

Assuming no operator action is taken and the EDG output breaker opens as designed, how will the #1 Emergency Diesel Generator respond to this SFAS Level 2 signal?

The Emergency Diesel Generator Governor will _____

- A. transfer to the isochronous mode and all engine trips will be active
- B. remain in the droop mode and all engine trips will remain in active
- C. remain in the droop mode and non-vital engine trips will be bypassed
- D. transfer to the isochronous mode and non-vital engine trips will be bypassed

Answer: D

Explanation/Justification:

- A. Incorrect – Although the Governor will transfer to Isochronous mode, non-vital engine trips will still be active. This is plausible since the EDG was already running at the time to of the SFAS start signal.
- B. Incorrect – This is plausible since the EDG was already running at the time to of the SFAS start signal. In normal parallel operation, the EDG operates in Droop with all engine trips active.
- C. Incorrect – This is plausible since the EDG was already running at the time to of the SFAS start signal. In normal parallel operation, the EDG operates in Droop with all engine trips active. Since an SFAS occurred, it is reasonable to assume some normal trips are bypassed.
- D. Correct – The Governor will transfer to the Isochronous mode, and non-vital engine trips are bypassed.

Sys #	System	Category	KA Statement
064	Emergency Diesel Generator (ED/G) System	K4. Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following:	Governor valve operation
K/A#	K4.03	K/A Importance	Exam Level
		2.5	RO
References provided to Candidate		None	Technical References:
Question Source:		BANK 32132	Level Of Difficulty: (1-5)
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:
Objective:			3 (CFR: 41.7)

51. Which of the following process radiation monitors will cause an automatic action to occur upon reaching its high alarm setpoint?

1. RE1412, Component Cooling Return Line 1
2. RE4598AA, Station Vent Discharge
3. RE5052A, Containment Purge Exhaust
4. RE5403A, Fuel Handling Area Exhaust System
5. RE5405A, Radwaste Area Exhaust
6. RE8432, Service Water System Outlet Header

- A. 1, 2, 3, & 5
- B. 1, 2, 5, & 6
- C. 1, 3, 4, & 6
- D. 2, 3, 4, & 5

Answer: A

Explanation/Justification:

- A. Correct – RE1412 will isolate the Component Cooling Water (CCW) surge tank 3 way valve from atmosphere. RE4598, RE5052 and RE5405 will trip their respective ventilation fans and close their respective isolation dampers
- B. Incorrect – RE8432 has no automatic function but is plausible since CCW does have an automatic function
- C. Incorrect – Fuel Handling Ventilation is not tripped by RE5403 but is plausible since it is tripped by RE8446/RE8447 and the other ventilation systems also have automatic actions associated with their radiation monitors, RE8432 has no automatic function but is plausible since CCW does have an automatic function
- D. Incorrect –Fuel Handling Ventilation is not tripped by RE5403 but is plausible since it is tripped by RE8446/RE8447 and the other three ventilation systems all have automatic actions associated with their radiation monitors

Sys #	System	Category	KA Statement
073	Process Radiation Monitoring (PRM) System	K1. 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 2-24, 2-25, 2-26 and 2-27
Question Source:	New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:	Low - Fundamental	10 CFR Part 55 Content:	(CFR: 41.7 / 45.8)
Objective:			

52. The plant is at 100% power with all systems in normal alignment **EXCEPT** Containment Air Cooler 2 is running in SLOW speed for testing

The following event occurs:

- Loss of off-site power.
- EDG 2 fails to start.
- All other systems function as designed.

5 minutes after the loss of off-site power occurred, assuming no Operator action, what will be the position of the following valves?

- (1) SW1366 - CTMT Air Cooler 1 Inlet Iso
(2) SW1367 - CTMT Air Cooler 2 Inlet Iso

- A. (1) Open
(2) Open
- B. (1) Open
(2) Closed
- C. (1) Closed
(2) Open
- D. (1) Closed
(2) Closed

Answer: C

Explanation/Justification:

- A. Incorrect – plausible since refill logic will close SW1366 but would then reopen if CAC 1 was in slow if an SFAS level 2 existed.
- B. Incorrect – plausible if Candidate does not know SW1367 is powered from F12A via EDG2 since SW1367 would close and remain closed by refill logic due to CAC 2 was in fast and SW1366 would reopen since CAC 1 was in slow
- C. Correct – Refill logic will close SW1366 which must be manually opened since no SFAS signal is present. SW1367 will remain open since power is lost
- D. Incorrect – plausible if Candidate does not know SW1367 is powered from F12A via EDG 2 since SW1367 would close and remain closed by refill logic due to CAC 2 was in fast

Sys #	System	Category	KA Statement
076	Service Water System (SWS)	K2. Knowledge of bus power supplies to the following:	Reactor building closed cooling water
K/A#	K2.04	K/A Importance	2.5*
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	OS-020 Sheet 2, R45 CL11
Question Cognitive Level:	High - Comprehension	Level Of Difficulty: (1-5)	4
Objective:		10 CFR Part 55 Content:	(CFR: 41.7)

53. The plant is in Mode 3 normal operating temperature and pressure with both steam line isolation valves OPEN.

Instrument air is lost to MS101, Main Steam Line 1 Isolation Valve

(1) How will MS101, Main Steam Line 1 Isolation Valve respond to this loss of instrument air?

(2) How will this loss of instrument air affect MS101, Main Steam Line 1 Isolation Valve Tech Spec required stroke time?

- A. (1) fail closed
(2) WILL still meet its Tech Spec required stroke time
- B. (1) fail closed
(2) WILL NOT meet its Tech Spec required stroke time
- C. (1) remain open
(2) WILL still meet its Tech Spec required minimum stroke time
- D. (1) remain open
(2) WILL NOT meet its Tech Spec required stroke time

Answer: C

Explanation/Justification:

- A. Incorrect - Plausible if candidate knows an accumulator exists but determines it is only for assisting closure to meet minimum stroke requirements
- B. Incorrect - Plausible if candidate knows an accumulator exists but determines it is only for ensuring closure without meeting stroke requirements
- C. Correct – Accumulator will both hold open MSIV and pneumatic via N2 assist closing springs to meet design minimum required closing requirement
- D. Incorrect - Plausible if Candidate knows an accumulator exists but determines it is only for temporarily maintaining valve open and not also close assist

Sys #	System	Category	KA Statement
078	Instrument Air System (IAS)	K1. Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems:	MSIV air
K/A#	K1.05	K/A Importance	Exam Level
		3.4*	RO
References provided to Candidate		None	Technical References:
Question Source:		New	Level Of Difficulty: (1-5)
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:
Objective:			

54. Which of the following systems, interlocks, or controls ensure the Containment Vessel remains above the **MINIMUM** internal design pressure?
- A. BWST maximum Temperature limits.
 - B. Containment Spray nozzle size and location
 - C. Containment Spray Discharge Valve Throttle position
 - D. Containment Vacuum Relief Valves.

Answer: D

Explanation/Justification:

- A. Incorrect – Plausible since a temperature decrease in the BWST would result in a lower Containment Pressure during an inadvertent spray event. BWST maximum temperatures ensure post LOCA injection removes the heat assumed in the accident analysis.
- B. Incorrect – Plausible since the spray patterns and location would affect the low pressure created during an inadvertent spray event.
- C. Incorrect – Plausible since throttling spray flow would affect pressure reduction but this interlock actuates post LOCA to prevent runout of the CTMT Spray Pumps.
- D. Correct – The CTMT Vacuum Relief capacity is designed to protect containment against an inadvertent actuation of CTMT Spray causing significant reduction of Containment Pressure (absolute scale).

Sys #	System	Category	KA Statement	
103	Containment System	Generic	Knowledge of the purpose and function of major system components and controls.	
K/A#	2.1.28	K/A Importance	4.1	Exam Level
References provided to Candidate	None	Technical References:	RO	SD-022F R01 Step 1.1.2.1
Question Source:	New	Level Of Difficulty: (1-5)	3	
Question Cognitive Level:	Low - Fundamental	10 CFR Part 55 Content:	(CFR: 41.7)	
Objective:				

55. The Plant is in Mode 1 at 100% power with all systems in a normal alignment.

Containment Operability is being evaluated.

Which one of the following containment conditions and/or malfunctions will require Technical Specification action within one hour or less.

- A. Emergency Air Lock Inner and Outer Doors have failed seal leakage tests.
- B. DR2012A, CTMT Sump Pumps Discharge Inside CTMT Isolation has failed its SFAS stroke time.
- C. Containment Pressure is + 20 inches water gauge.
- D. Containment average air temperature is 115 °F.

Answer: A

Explanation/Justification:

- A. Correct – This would render the CTMT Airlock Inoperable under CTMT Systems for TS 3.6.2, Containment Air Locks. CTMT Integrity/Operability under TS 3.6.1 will be affected if total air lock leakage makes overall CTMT leakage exceed allowable amount. TS 3.6.2 Condition C Action C.1 requires action to evaluate overall leakage to be initiated immediately.
- B. Incorrect – Plausible because this would render a CTMT Isolation valve Inoperable under CTMT Systems for TS 3.6.3, Containment Isolation Valves, this is a 4 hour action
- C. Incorrect – Plausible because this would be an inoperable condition under CTMT Systems for TS 3.6.4, Containment Pressure. If >25 inches this is a one hour action statement.
- D. Incorrect – Plausible because this would be an inoperable condition under CTMT Systems for TS 3.6.5 Containment Air Temperature if greater than 120F.

Sys #	System	Category	KA Statement
103	Containment System	K3. Knowledge of the effect that a loss or malfunction of the containment system will have on the following:	Loss of containment integrity under normal operations
K/A#	K3.02	K/A Importance	3.8
Exam Level	RO		
References provided to Candidate	None	Technical References:	TS 3.6.2 Condition C Required Action C.1
Question Source:	New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:	Low - Fundamental	10 CFR Part 55 Content:	(CFR: 41.7 / 45.6)
Objective:			

56. RCS Temperature is 180 °F.

RCS Pressure is 180 psig.

Which of the following pumps can be started **IMMEDIATELY** from the Control Room to add boric acid from the Borated Water Storage Tank (BWST) to the Reactor Coolant System, if necessary?

1. Boric Acid Addition Pumps
2. High Pressure Injection Pumps
3. Low Pressure Injection Pumps
4. Makeup Pumps

- A. 1 & 2 only
- B. 1 & 4 only
- C. 2 & 3 only
- D. 3 & 4 only

Answer: D

Explanation/Justification:

- A. Incorrect – RCS Pressure is too high for BAAT Pumps, and HPI is disabled when RCS temperature is less than 280°F by racking out the breakers.
- B. Incorrect – RCS Pressure is too high for BAAT Pumps but the MU Pumps would be a viable source
- C. Incorrect - HPI is disabled when RCS temperature is less than 280°F by racking out the breakers, but the LPI pumps would be a viable source
- D. Correct – Both the Makeup Pumps and the LPI pumps can be started immediately in these plant conditions.

Sys #	System	Category	KA Statement
002	Reactor Coolant System (RCS)	K1. Knowledge of the physical connections and/or cause-effect relationships between the RCS and the following systems:	Borated water storage tank
K/A#	K1.03	K/A Importance 3.8	Exam Level RO
References provided to Candidate		None	Technical References: DB-OP-06903, R42 Step 4.47 for MU disabled, Step 4.21 for HPI disabled and note 4.46 for BAAT Transfer Pump Discharge Pressure.
Question Source:		New	Level Of Difficulty: (1-5) 3
Question Cognitive Level:		High - Analysis	10 CFR Part 55 Content: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
Objective:			

57. The Plant is at 100% Power when control rod 7-3 drops to the bottom of the core. The following alarms are in:

- 5-1-E CRD LCO
- 5-2-E CRD ASYMETRIC ROD

The Reactor Operator places the Rod Control Panel in **MANUAL**, and moves the CRD T Handle to insert Control Rods.

How will Group 7 respond to this **IN** command?

Group 7 will _____ (1) _____ because _____ (2) _____.

- A. (1) insert
(2) an Asymetric Fault exists
- B. (1) insert
(2) the dropped rod is unattached from its leadscrew
- C. (1) Not move
(2) an In Limit exists
- D. (1) Not move
(2) a Sequence Fault exists.

Answer: C

Explanation/Justification:

- A. Incorrect – Plausible because asymmetry fault bypasses the in limit when rods in auto
- B. Incorrect – Plausible because the dropped rod will not move if unattached however, the alarms provided come from lead screw position, not physical rod position. As a result, they would not be affected by a detached rod.
- C. Correct – the in limit interlock prevents rod insertion when any rod in a group has a rod bottom light lit unless the in limit bypass button is depressed
- D. Incorrect – Plausible because based on initial Group 7 position, a sequence fault could occur, but group 7 rod at its in limit will prevent insertion

Sys #	System	Category	KA Statement
014	Rod Position Indication System	K4. Knowledge of RPIS design feature(s) and/or interlock(s) which provide for the following:	Rod bottom lights
K/A#	K4.03	K/A Importance	Exam Level
		3.2	RO
References provided to Candidate		None	Technical References: DB-OP-06402 R23 page 152 Att 2 (#16)
Question Source:		New	Level Of Difficulty: (1-5)
			4
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:
			(CFR: 41.5 / 45.7)
Objective:			

58. MI-05254, Nuclear Instrumentation NI05 (RPS CH 2) Power Range Adjustment is in Progress.

- The Load Control Panel is in Manual
- The Rod Control Panel and Reactor Demand are in Auto
- NI6 Indicates 99.8%
- NI7 Indicates 99.6%
- NI8 Indicates 99.4%

I&C has informed the Shift Manager they have completed calibration and are returning the Power Range Test Module rotary switch to the OPERATE position.

- Due to an error, NI5 gain is set incorrectly and NI5 currently reads 105%

When I&C returns the Power Range Test Module to OPERATE position, how will the regulating control rods respond?

- A. No effect
- B. Insert
- C. Withdraw
- D. Trip

Answer: B

Explanation/Justification:

- A. Incorrect – Plausible if candidate does not know the controlling NI is high auctioneered
- B. Correct – Correct answer – Highest NI will control rods and greater than or equal to 1% neutron error will insert rods
- C. Incorrect – Plausible if candidate assumes power must be raised to match indication (also opposite of correct answer)
- D. Incorrect – Plausible since 105% is greater than the high power trip setpoint of 104.7% however only a single channel is affected and the reactor will not trip.

Sys #	System	Category	KA Statement		
015	Nuclear Instrumentation System	A4. Ability to manually operate and/or monitor in the control room:		Selection of controlling NIS channel	
K/A#	A4.01	K/A Importance	3.6*	Exam Level	RO
References provided to Candidate		None	Technical References:	M-533-180-1 ICS Reactor Control Digital Logic	
Question Source:		New	Level Of Difficulty: (1-5)		3
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:		(CFR: 41.7 / 45.5 to 45.8)
Objective:					

59. The plant was operating at 100% power. The reactor is manually tripped due to high vibration on the Main Generator.

The following events occur:

- All Turbine Bypass Valves open to control Steam Generator Pressure.
- SP13B1, Steam Line 1 Turbine Bypass Valve sticks full open.

All other equipment functions as designed.

- (1) How will the plant respond to this failure, assuming no operator actions?
- (2) What, if any, operator actions will be **required** to stabilize the plant without relying on the Main Steam Safety Valve operation?

- A. (1) The unaffected Turbine Bypass Valves will modulate closed to control both SG pressures at the normal post trip setpoint of approximately 995 psig. This condition will not result in an SFRCS actuation.
(2) No Operator Action will be required to stabilize the plant.
- B. (1) SFRCS will actuate on low SG1 Level, closing the Main Steam Isolation Valves, and starting Auxiliary Feedwater to restoring SG1 Level to 49 inches.
(2) No Operator Action will be required to stabilize the plant.
- C. (1) SFRCS will actuate on low SG Pressure on SG1, closing both Main Steam Isolation Valves.
(2) The Operators will use the Atmospheric Vent Valves in manual to control RCS Tave constant or slightly lowering.
- D. (1) SFRCS will actuate on Steam to Feed Differential Pressure on SG1, isolating all Main and Auxiliary Feedwater to SG1.
(2) The Operators will open the Atmospheric Vent Valves on #1 SG to blowdown the affected SG.

Answer: C

Explanation/Justification:

- A. Incorrect – Plausible if the candidate concludes the steam flow rate due to one open TBV is less than the core decay heat rate post trip. This event will exceed the core decay heat rate even if all other TBVs are closed. If the steam flow was less than core decay heat, then this response would be accurate.
- B. Incorrect – Plausible because the Steam Generator Level would be lowering with an open TBV, however the Main Feedwater System and AFW, if actuated, can maintain SG level at setpoint even with an open TBV. The MSIVs would not close on low SG Level.
- C. Correct – Without Operator Action, SG pressure in #1 SG would lower and cause an SFRCS Low SG Pressure on #1 SG at 630 psig. Once the MSIVs close, SG Pressure will rise causing the low pressure trip to reset allowing AFW flow to #1 SG. Operator action to control SG Pressure would be necessary to prevent Main Steam Safety Valves from opening.
- D. Incorrect – Plausible because SFRCS will eventually actuate on Steam to Feed Differential Pressure once the MSIVs are closed in response to the low SG Pressure. The actions to blowdown the affected SG are actions taken in response to a Steam Line Break in accordance with DB-OP-02525, Steam Leaks, section 4.2, not an action taken in response to a TBV malfunction.

Sys #	System	Category	KA Statement
041	Steam Dump System (SDS) and Turbine Bypass Control	Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations:	Steam valve stuck open
K/A#	A2.02	K/A Importance 3.6	Exam Level RO
References provided to Candidate		None	Technical References:
Question Source: New		Level Of Difficulty: (1-5)	4
Question Cognitive Level: High - Comprehension		10 CFR Part 55 Content:	(CFR: 41.5 / 43.5 / 45.3 / 45.13)

60. The Plant is in Mode 3. All systems are in a normal alignment.

Main Condenser pressure is 0.6 IN HgA

If the only operating steam jet air ejector suction valve is closed, isolating the SJAE from the Main Condenser, how will this affect Condenser pressure?

Main Condenser Pressure will:

- A. remain constant.
- B. control at approximately 3.0 IN HgA.
- C. control at approximately 10 IN HgA.
- D. eventually rise to Atmospheric Pressure.

Answer: B

Explanation/Justification:

- A. Incorrect – Plausible if Candidate determines loss of suction alignment will have no effect due to vacuum being maintained by condensate depression as the Turbine Bypass valves dump steam to the Main Condenser.
- B. Correct – The Mechanical Hogger starts at 4.5 IN HgA and reduces pressure to 3.0 IN HgA where its pressure control valve will open to atmosphere to maintain pressure at approximately 3 IN HgA.
- C. Incorrect – Plausible the Steam Hogger used during startup will lower condenser pressure to approximately 10 inches HgA, but would not be in service if condenser vacuum has already been established..
- D. Incorrect – Plausible because none condensable gases would eventually cause condenser pressure to reach atmospheric pressure.

Sys #	System	Category	KA Statement	
055	Condenser Air Removal System	K3. Knowledge of the effect that a loss or malfunction of the CARS will have on the following:	Main condenser	
K/A#	K3.01	K/A Importance	2.5	Exam Level
References provided to Candidate	None	Technical References:	RO	OS-015 CL-6
Question Source:	New	Level Of Difficulty: (1-5)	3.5	
Question Cognitive Level:	High - Comprehension	10 CFR Part 55 Content:	(CFR: 41.7 / 45.6)	
Objective:				

61. Chemistry has sampled and analyzed the Miscellaneous Waste Monitor Tank and recommended it for release.

The following conditions exist:

- The MINIMUM Dilution Flow has been established
- The calculated desired recirculation time is 180 minutes
- Miscellaneous Waste System Outlet Radiation Elements RE1878A and RE1878B have been lined up and confirmed operable.
- Chemistry drew the tank sample after 120 minutes of recirculation
- The tank has NOW been recirculating for 200 minutes
- The release valve lineup was been completed satisfactorily.

Based on these conditions, what is the status of the prepared release?

The release _____.

- A. Can proceed, RE1878A and RE1878B will automatically stop the release, if necessary
- B. Can proceed, the valve lineup has been verified correct
- C. Can NOT proceed, until dilution flow has been increased
- D. Can NOT proceed, since the sample may not be representative of the tanks content

Answer: D

Explanation/Justification:

- A. Incorrect. Plausible since it is true that the RMs will auto isolate the release if high activity is detected. However, IAW DB-OP-03011 the permit should be voided.
- B. Incorrect. Plausible since the required flowpath is available. However, IAW DB-OP-03011 the permit should be voided.
- C. Incorrect. Plausible since this is a required action for certain release flowrates.
- D. Correct. IAW DB-OP-03011 Revision 21 pages 11 & 12 minimum required recirculation time has NOT been met to obtain two volume turnover. Therefore, the sample taken by chemistry may not be representative of the tanks content. The permit cannot be approved and should be voided. Candidate must recognize that the recirc time must be met before the sample is drawn NOT before the tank can be discharged and know the reason for the required recirculation time.

Sys #	System	Category	KA Statement
068	Liquid Radwaste System	A2. Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Lack of tank recirculation prior to release
K/A#	A2.02	K/A Importance 2.7*	Exam Level RO
References provided to Candidate		None	Technical References: DB-OP-03011 Revision 21 pages 11 & 12
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)
Objective:			

62. The following plant conditions exist:

A small break Loss of Coolant Accident has just occurred.

All systems function as designed.

Reactor Coolant System pressure is 1550 psig.

Containment pressure is 16.5 psia.

Without operator action, which one of the following radiation detectors will provide indication of actual Containment radiation levels?

1. RE 4596A, CONTAINMENT HIGH RANGE RADIATION ELEMENT
2. RE 4597 AA, CONTAINMENT NORMAL RANGE RADIATION MONITOR
3. RE 4597 AB, CONTAINMENT ACCIDENT RANGE RADIATION MONITOR

- A. Only 1
- B. Only 2
- C. 2 and 3
- D. 1 and 3

Answer: A

Explanation/Justification:

- A. Correct – At 1550 psig in the RCS, an SFAS Level 1&2 will have actuated. This causes the isolation valves to RE4597 AA and AB to close leaving only RE4596 available.
- B. Incorrect – Plausible because although at 1550 psig in the RCS, an SFAS Level 2 will have actuated, these RCS condition are indicative of a small break LOCA event. As a result, candidate may assume only the normal range is available and accident range monitors are not yet in service.
- C. Incorrect – Plausible because at 1550 psig in the RCS, an SFAS Level 2 will have actuated. Since RE4596 cabling can be affected by high temperatures condition in Containment it is plausible this detector is not used in this scenario..
- D. Incorrect – Plausible because at 1550 psig in the RCS, an SFAS Level 2 will have actuated. It is logical the normal range monitor will have isolated leaving the accident range RE4597 and high range RE4596 available.

Sys #	System	Category	KA Statement
072	Area Radiation Monitoring System	A1. Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including:	Radiation levels
K/A#	A1.01	K/A Importance 3.4	Exam Level RO
References provided to Candidate		None	Technical References: DB-OP-02000 R26 Table 2 SFAS Valves expected response level 2
Question Source:		New	Level Of Difficulty: (1-5) 2.5
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content: (CFR: 41.5 / 45.5)
Objective:			

63. The following plant conditions exist:

- Plant is operating at 100% power.
- Service Water Loop 1 is supplying Secondary Loads
- Service Water Loop 2 is supplying Primary Loads

The following event occurs:

- Bus C1 Locks Out

Without operator action, how will Turbine Plant Cooling Water loads be cooled?

- A. Fire Protection System
- B. Service Water Train 1
- C. Service Water Train 2
- D. Circulating Water

Answer: D

Explanation/Justification:

- A. Incorrect – Plausible because Service Water Pump 1 will de-energize on a C1 bus lockout and fittings are available to supply cooling water to heat exchangers from fire protection system.
- B. Incorrect – Plausible if Candidate does not know Service Water Pump 1 is powered from C1 or assumes C1 will be restored by an EDG Start.
- C. Incorrect – Plausible if the Candidate assumes secondary loads will auto transfer to Service Water train 2, however SW1395 does not have an auto open feature.
- D. Correct – Service Water Pump 1 will de-energize and SW1399, SW 1 Isolation to Secondary Loads will close when SW Pressure drops below 50 psig. CT2955 will open when SW pressure drops below 30 psig allowing Circulating Water to cool secondary loads

Sys #	System	Category	KA Statement	
075	Circulating Water System	K2. Knowledge of bus power supplies to the following:	Emergency/essential SWS pumps	
K/A#	K2.03	K/A Importance	2.6*	Exam Level
References provided to Candidate		None	Technical References:	RO OS-020 SH1 and SH2
Question Source:		New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content:	(CFR: 41.7)
Objective:				

64. The following plant conditions exist:

Plant is operating at 100% power.

Annunciator 9-1-F, INSTR AIR HDR PRESS LO alarms.

The Reactor Operator reports that Instrument Air pressure (using PI810) reads 72 psig and the secondary plant appears stable.

Which one of the following sets of actions is **required** to be performed?

- A. Manually trip the reactor and initiate AFW flow and isolation of both SG's.
- B. Start the standby Station Air Compressor and the Emergency Instrument Air Compressor, and perform a rapid shutdown per DB-OP-02504, Rapid Shutdown.
- C. Dispatch operators to locate the cause of excessive air demand and maintain reactor power at the present level.
- D. Rapidly lower power per DB-OP-02504, Rapid Shutdown, until Instrument Air pressure rises to approximately 90 psig.

Answer: A

Explanation/Justification:

- A. Correct – This Instrument Air Header Pressure (even with stable plant) requires tripping the reactor and initiating and isolating SFRCS which is an entry condition to the Emergency Operating Procedure DB-OP-02000.
- B. Incorrect – Plausible because starting the Standby and EIAC could improve condition in the instrument air system and the plant is stable, however this pressure is below minimum for continued power operation.
- C. Incorrect – This is Plausible because these actions are consistent with operator response to stable low air pressure of a dryer switching failure.
- D. Incorrect – Plausible because continued operation is permitted with a stable low air pressure of 90 psig.

Sys #	System	Category	KA Statement	
079	Station Air System	Generic	Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	
K/A#	2.4.2	K/A Importance	4.5	Exam Level
References provided to Candidate		None	Technical References:	RO DB-OP-02528 R16, Step 4.1
Question Source:		BANK 37548	Level Of Difficulty: (1-5)	2
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content:	(CFR: 41.7 / 45.7 / 45.8)
Objective:				

65. I&C is performing testing on fire detection for EDG Room 2 and inadvertently sends a fire alarm signal from one detector to the Fire Detection system.

The following conditions are noted:

- Annunciator 9-1-G FIRE OR RADIATION TRBL Alarms
- The Control Room Fire and Radiation CRT indicates FP114A, DIESEL GENERATOR ROOM 2 SPRINKLER PREACTION valve has actuated

What will be the status of the sprinkler system in EDG Room 2 and why?

The sprinkler system in EDG Room 2 will _____(1)_____ because _____(2)_____.

- A. (1) be flowing water
(2) the preaction valve has opened to pressurize the sprinkler header
- B. (1) be flowing water
(2) the supervisory air has been vented
- C. (1) not be flowing water
(2) it takes a second alarm signal to pressurize the sprinkler header
- D. (1) not be flowing water
(2) the sprinklers are held closed by a fusible link

Answer: D

Explanation/Justification:

- A. Incorrect – Plausible because the sprinkler header does charge but is not actuated
- B. Incorrect – Plausible because there is supervisory air which would vent if a fusible link melts
- C. Incorrect – Plausible because most safety systems require a redundant signal to actuate
- D. Correct – The EDG Room sprinkler header is dry with supervisory air pressure (for alarm purpose) held by the preaction valve on one side and the fusible links on each sprinkler on the other. The header will charge when a fire alarm causes the preaction valve to open but water will only flow through a sprinkler that has its fusible link melted

Sys #	System	Category	KA Statement
086	Fire Protection System (FPS)	K6. Knowledge of the effect of a loss or malfunction on the Fire Protection System following will have on the :	Fire, smoke, and heat detectors
K/A#	K6.04	K/A Importance 2.6	Exam Level RO
References provided to Candidate	None	Technical References:	System Description 036A, Page 2-2 step 2.1.2.1.2
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:			

66. The plant is in MODE 6 with core reload in progress.

From 0600 to 0700, the operating DH Train was secured to facilitate fuel handling near the Loop 2 RCS Hot Leg.

At 0900, which of the following conditions would require immediately suspending irradiated fuel movement in accordance with DB-OP-00030, Fuel Handling Operations?

- A. Loss of Communications with the Refueling Outage Containment Coordinator.
- B. One Fan of Control Room Emergency Ventilation System is determined to be inoperable. The remaining Fan is operable.
- C. One Train of the Spent Fuel Pool Emergency Ventilation System (EVS) is determined to be inoperable. The remaining Train is operable.
- D. The operating Decay Heat Removal Train is determined to be Inoperable. The standby Train is operable.

Answer: D

Explanation/Justification:

- A. Incorrect – While loss of communications does require suspending fuel handling operation, this individual is not one of the required locations.
- B. Incorrect – TS 3.7.10 only requires immediately suspending fuel handling operations if the CRE Boundary is inoperable.
- C. Incorrect – TS 3.7.13 only requires immediately suspending fuel handling operations if both trains are lost.
- D. Correct – TS 3.9.4 required one DHR Loop to be operable AND in operation. Condition A.2 requires suspending loading fuel assemblies in the core.

Sys #	System	Category		KA Statement
N/A	N/A	Generic		Knowledge of refueling administrative requirements.
K/A#	2.1.40	K/A Importance	2.8	Exam Level
				RO
References provided to Candidate		None	Technical References:	TS 3.9.4, DHR and Coolant Circulation
Question Source:	New		Level Of Difficulty: (1-5)	4
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:	(CFR: 41.10 / 43.5 / 45.13)
Objective:				

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

Over the last 2 minutes, Makeup Tank Level has lower from 75 inches to 70 inches.

Which of the following conditions will confirm the lowering of Makeup Tank Level is due to a Makeup Tank level indicator malfunction as opposed to some other event?

- A. Makeup Tank Pressure is stable at 30 psig.
- B. Pressurizer Level rises from 219 to 226 inches.
- C. RCS Tave Lowers from 582 °F to 580.5 °F.
- D. Makeup Flow is stable at 60 gpm

Answer: A

Explanation/Justification:

- A. Correct – The makeup Tank uses Hydrogen Gas overpressure to control RCS Oxygen. If real Makeup Tank level is lowering, you would see a corresponding change in Makeup Tank Pressure.
- B. Incorrect – Plausible because without leakage, an increase of 7 inches in the Pressurizer would cause Makeup Tank level to lower approximately 5 inches.
- C. Incorrect – Plausible because without leakage, a decrease in RCS Tave of approximately 1.5 °F would cause Makeup Tank level to lower approximately 5 inches.
- D. Incorrect – Makeup Flow rate is independent of Makeup Tank Level. The fact that Letdown flow is stable does not provide information to determine the status of the Makeup Tank Level indicator.

Sys #	System	Category		KA Statement
N/A	N/A	Generic		Ability to identify and interpret diverse indications to validate the response of another indication.
K/A#	2.1.45	K/A Importance	4.3	Exam Level
References provided to Candidate	None			RO
Question Source:	New			General Physics Equation Sheet 1-14
Question Cognitive Level:	Low - Fundamental			Level Of Difficulty: (1-5)
Objective:				3
				10 CFR Part 55 Content:
				(CFR: 41.7 / 43.5 / 45.4)

68. Initial conditions:

- The plant is at 2135 psig and 525 °F.
- No Tech Spec required equipment is INOPERABLE.

AC101, EDG1 Output Breaker, is racked into the test position to support maintenance.

In accordance with Technical Specification 3.8.1, AC Sources - Operating, which one of the following lists the **MINIMUM required** action(s) that must be performed within one hour?

- A. Test start EDG 2 ONLY
- B. Verify correct breaker alignment and indicated power availability for the offsite circuit supplying A Bus ONLY.
- C. Verify correct breaker alignment and indicated power availability for each offsite circuits.
- D. Test start EDG 2 and verify correct breaker alignment and indicated power availability for each offsite circuits.

Answer: C

Explanation/Justification:

- A. Incorrect –Plausible because the #2 EDG will be started, but starting the opposite train EDG is only required within 24 hours
- B. Incorrect – Verification of breaker status within one hour is required on each operable off-site circuits, not just those supplying A Bus. Plausible because A bus is the normal feed to C1 which is fed by EDG 1.
- C. Correct in accordance with T.S. 3.8.1 Condition B with a completion time of 1 hour.
- D. Incorrect – Verification of breaker status within one hour is required, but starting the opposite train EDG is only required within 24 hours.

Sys #	System	Category	KA Statement		
N/A	N/A	Generic	Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.		
K/A#	2.2.36	K/A Importance	3.1	Exam Level	RO
References provided to Candidate		None	Technical References:	T.S. 3.8.1 Condition B & SR 3.8.1.1	
Question Source:		BANK 92552	Level Of Difficulty: (1-5)	3	
Question Cognitive Level:		Low - Memory	10 CFR Part 55 Content:	(CFR: 41.10 / 43.2 / 45.13)	
Objective:					

69. Per NOBP-OP-0007, Infrequently Performed Tests and Evolutions, which of the following individuals make the final determination as to the whether an evolution will be conducted as an IPTE or not?
- A. Shift Engineer
 - B. Shift Manager
 - C. Director Site Operations
 - D. Site Vice President

Answer: C

Explanation/Justification:

- A. Incorrect per NOBP-OP-0007 R05 Page 7 Step 5.1.4.
- B. Incorrect per NOBP-OP-0007 R05 Page 7 Step 5.1.4.
- C. Correct per NOBP-OP-0007 R05 Page 7 Step 5.1.4.
- D. Incorrect per NOBP-OP-0007 R05 Page 7 Step 5.1.4.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the process for conducting special or infrequent tests.
K/A#	2.2.7	K/A Importance	2.9
References provided to Candidate		None	Exam Level
Question Source:		New	RO
Question Cognitive Level:		Low - Memory	NOBP-OP-0007 R05 Page 7 Step 5.1.4
Objective:			Technical References:
			Level Of Difficulty: (1-5)
			10 CFR Part 55 Content:
			3
			(CFR: 41.10 / 43.3 / 45.13)

70. Which OPERATIONAL MODE does the following set of conditions describe?

- $K_{eff} > 0.99$
- $RTP < 5\%$
- $T_{ave} > 280\text{ }^{\circ}\text{F}$

- A. Hot Shutdown
- B. Hot Standby
- C. Startup
- D. Power Operation

Answer: C

Explanation/Justification:

- A. Incorrect per TS Definition Table 1.1-1.
- B. Incorrect per TS Definition Table 1.1-1.
- C. Correct per TS Definition Table 1.1-1.
- D. Incorrect per TS Definition Table 1.1-1.

Sys #	System	Category	KA Statement	
N/A	N/A	Generic	Ability to determine Technical Specification Mode of Operation.	
K/A#	2.2.35	K/A Importance	3.6	Exam Level
References provided to Candidate		None		RO
Question Source:		BANK 29962		Technical References: TS Table 1.1-1
Question Cognitive Level:		Low - Memory		Level Of Difficulty: (1-5) 2
Objective:				10 CFR Part 55 Content: (CFR: 41.7 / 41.10 / 43.2 / 45.13)

71. The following plant conditions exist:

- A large break LOCA with fuel damage has occurred
- All systems function as designed

It becomes necessary to take action to prevent damage to one of the operating LPI pumps

The LPI pump is in a 60 Rem/hr radiation field

Which of the following is the **MAXIMUM** amount of time that a worker is authorized to remain in the above radiation field without exceeding the TEDE emergency dose limits?

- A. 5 minutes
- B. 10 minutes
- C. 30 minutes
- D. 60 minutes

Answer: B

Explanation/Justification:

- A. Incorrect – 5 minute exposure would result in a 5 REM exposure. While this is the TEDE normal Dose Limit it is not the maximum time permitted in that radiation field per the emergency dose limit..
- B. Correct – 10 minute exposure would result in a 10 REM exposure which is less than the TEDE Emergency Dose Limit.
- C. Incorrect – One hour exposure would result in 30 REM dose which exceeds the TEDE Emergency Dose Limit.
- D. Incorrect – One hour exposure would result in 60 REM dose which exceeds the TEDE Emergency Dose Limit.

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	3.2	Exam Level
References provided to Candidate		None		Technical References:
Question Source:	BANK 38751			Level Of Difficulty: (1-5)
Question Cognitive Level:	Low - Fundamental			10 CFR Part 55 Content:
Objective:				3 (CFR: 41.12 / 43.4 / 45.10)

72. The plant is operating at 100% power with all systems in normal alignment. The Containment Purge System is in service and aligned to the Mechanical Penetrations Rooms.

A planned maintenance evolution will involve venting filters at the Miscellaneous Waste Duratek skid.

Which of the following installed Radiation Monitors would give the **FIRST** indication that the venting is creating a radiological airborne hazard?

- A. RE5405, Radwaste Area Exhaust System
- B. RE5403, Fuel Handling Exhaust System
- C. RE5052, Containment Purge Exhaust System
- D. RE4597, Station Vent

Answer: B

Explanation/Justification:

- A. Incorrect – Although it is logical that radioactive systems would be served by the Radwaste Ventilation System, the Duratek System is located in the Spent Fuel Pool Area that is not served by the Radwaste Ventilation System.
- B. Correct – The Duratek System is located in the Spent Fuel Pool Area. As a result, the radiation monitor on this ventilation system would have the first opportunity to detect elevated radiation levels.
- C. Incorrect - Although it is logical that radioactive systems would be served by Containment Purge System in the Penetration Rooms, but the Duratek System is located in the Spent Fuel Pool Area that is not served by the Containment Purge Ventilation System.
- D. Incorrect – A number of ventilation fans directly discharge to the Station Vent without an intermediate Radiation Element which for some locations, the Station Vent Radiation Monitor may be the first indication of a rising radiation trend. In this case, the Station Vent is in the flowpath for release, the Station Vent Radiation Monitors are downstream from the other Radiation Monitors and therefore would not give first indication.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
K/A#	2.3.5	K/A Importance	2.9
References provided to Candidate	None	Exam Level	RO
Question Source:	New	Technical References:	Ops Schematic OS33 A-E, OS34 Sheets 1-3
Question Cognitive Level:	High - Comprehension	Level Of Difficulty: (1-5)	3
Objective:		10 CFR Part 55 Content:	(CFR: 41.11 / 41.12 / 43.4 / 45.9)

73. A Large Break LOCA has occurred. Borated Water Storage Tank level is 30 feet and lowering.

Step 10.2 of DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture directs performing Attachment 7, Section 1, Actions to Close Breakers for DH7A, DH7B, DH9A, DH9B, and HP31.

A review of local area Radiation Monitors in the vicinity of the Motor Control Centers indicates a peak dose rate of 34 REM/hr along the expected travel route to perform the required actions.

A Radiation Protection Technician is not IMMEDIATELY available to provide RP Coverage for this task.

Based on these conditions, what direction will you give the equipment operator and what is the basis for this direction?

As the Reactor Operator, you will _____ (1) _____ this task to an Equipment Operator because _____ (2) _____.

- A. (1) NOT assign
(2) the dose rate exceeds the Locked High Radiation Area dose rate and Equipment Operators do not carry Locked High Radiation Area Keys.
- B. (1) NOT assign
(2) the dose rate exceeds the Very High Radiation Area criteria and entry is not allowed without Radiation Protection coverage.
- C. (1) assign
(2) the task is required to complete the mitigation strategy for a LOCA and the total dose received will be within allowed limitations for post accident response.
- D. (1) assign
(2) the task is required to complete the mitigation strategy for a LOCA. Since the dose limitations for post accident response will be exceeded, prior approval of the Emergency Director is required.

Answer: C

Explanation/Justification:

- A. Incorrect – Assignment of the task is required to enable establishing Containment Emergency Sump as a suction source for the ECCS Pumps and therefore must be assigned.
- B. Incorrect - Assignment of the task is required to enable establishing Containment Emergency Sump as a suction source for the ECCS Pumps and therefore must be assigned.
- C. Correct – Restoring power as directed by Attachment 7 Section 1 is a required mitigation strategy to enable establishing Containment Emergency Sump as a suction source for the ECCS Pumps. As noted in the procedure warning, the total dose received is expected to be less than 2 Rem and based on time motion studies and worst case dose rates, RP coverage is not required.
- D. Incorrect – While the action is part of the required mitigation strategy, the expected dose will be within the allowed dose and not require pre-approval to exceed exposure limits

Sys #	System	Category	KA Statement		
N/A	N/A	Generic	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.		
K/A#	2.3.13	K/A Importance	3.4	Exam Level	RO
References provided to Candidate		None	Technical References:		DB-OP-02000 Attachment 7 section 1 Warning
Question Source:		New	Level Of Difficulty: (1-5)		3
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content:		(CFR: 41.12 / 43.4 / 45.9 / 45.10)

74. The plant is operating at 50% power with all systems in normal alignment. A power increase to 100% power is in progress.

The following Annunciator Alarms are received:

- 13-4-C, DEAR STRG TK 1 LVL.
- 13-4-D, DEAR STRG TK 2 LVL.

Subsequently, the following Deaerator Storage Tank levels are noted:

- LI202, Deaerator Storage Tank Level 1 is 0.5 feet and LOWERING.
- LI205, Deaerator Storage Tank Level 2 is 0.5 feet and LOWERING.

Which of the following actions are **required** for this plant condition?

- A. Stop the power increased until Deaerator Level rises.
- B. Dispatch an Operator to reset High Pressure Feedwater Heater 4 Drains to the Deaerator to prevent losing FW Heater inventory to the Condenser.
- C. Start BOTH FW Heater Drain Pumps to add inventory to the Deaerator.
- D. Trip the Reactor, Trip BOTH MFP's, Initiate and Isolate SFRCS.

Answer: D

Explanation/Justification:

- A. Incorrect – The candidate may select this action because reducing power would reduce Deaerator Inventory usage that could restore Deaerator Level. Also, the automatic runback on low Deaerator Level stops at 60% power.
- B. Incorrect – The candidate may select this action because if FW Heater 4 drains are going to the Condenser, Deaerator level would lower. This action could restore Deaerator level.
- C. Incorrect – The candidate may select this action because starting Heater Drain Pumps would add inventory to the Deaerator. This action could restore Deaerator level.
- D. Correct – This is the correct action per annunciator alarm response procedure DB-OP-02013 in anticipation of a loss of all Main Feedwater.

Sys #	System	Category	KA Statement	
N/A	N/A	Generic	Knowledge of annunciator alarms, indications, or response procedures.	
K/A#	2.4.31	K/A Importance	4.2	Exam Level
References provided to Candidate		None	Technical References:	RO DB-OP-02013 R10, Condensate and Feedwater Alarm Panel 13 Annunciators. Page 45
Question Source:	New		Level Of Difficulty: (1-5)	3
Question Cognitive Level:	Low - Fundamental		10 CFR Part 55 Content:	(CFR: 41.10 / 45.3)
Objective:				

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75. The plant was in MODE 1 when the Control Room was evacuated per DB-OP-02519, Serious Control Room Fire.

When the Supplementary Actions are complete, which ONE of the following describes how inventory is being supplied to the RCS?

- A. High Pressure Injection Pump #1 is RUNNING with an operator MANUALLY controlling HP2C, HPI Train 1 Injection Valve.
- B. High Pressure Injection Pump #2 is RUNNING with an operator MANUALLY controlling HP2A, HPI Train 2 Injection Valve.
- C. Makeup Pump #1 is RUNNING with an operator MANUALLY controlling MU 6420, NORMAL MAKEUP FLOW CONTROLLER BYPASS.
- D. Makeup Pump #2 is RUNNING with an operator MANUALLY controlling MU 6419, MU INJECTION TRAIN 1.

Answer: C

Explanation/Justification:

- A. Incorrect –The plant maintains Hot Standby Conditions following a Serious Control Room Fire. HPI discharge pressure is insufficient to provide RCS inventory at that pressure.
- B. Incorrect –The plant maintains Hot Standby Conditions following a Serious Control Room Fire. HPI discharge pressure is insufficient to provide RCS inventory at that pressure.
- C. Correct – DB-OP-02519 Attachment 5 directs the Equipment Operator to lineup MUP 1 and isolate, MU32 and MU6419 to allow another Operator to control level with MU6420
- D. Incorrect –Although MU is used, the protection of Train 1 for Serious Control Room Fire dictates the used of MU Train 1 and MU Pump 2 is tripped from the control room if time permits.

Sys #	System	Category	KA Statement	
N/A	N/A	Generic	Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	
K/A#	2.4.35	K/A Importance	3.8	Exam Level
References provided to Candidate		None	Technical References:	RO DB-OP-02519 R17 Attachment 5 and Attachment 3 step 6.a.
Question Source:		Bank 29276	Level Of Difficulty: (1-5)	4
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:	(CFR: 41.10 / 43.5 / 45.13)
Objective:				

(SRO ONLY)
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76. The plant is operating at 100% power with all systems in normal alignment **EXCEPT** EDG-1 is on clearance for lube oil replacement. A reactor trip coincident with a loss of offsite power occurs. The following conditions exist:

- All Control rods have inserted **EXCEPT** one control rod remains full out
- RCS pressure is 2245 psig and stable
- The hottest RCS Thot is 610 °F and stable
- EDG 2 has started
- EDG 2 output breaker AD-101 is OPEN
- Both AFW Pumps are in service
- Both SG levels are being controlled at 49 inches

The crew has entered DB-OP-02000, "RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE"

Which of the following specific rules provides the required actions that will mitigate these plant conditions?

- A. Specific Rule 1, Reactivity Control
- B. Specific Rule 2, Actions For Loss Of Subcooling Margin
- C. Specific Rule 5, Pressurized Thermal Shock Requirements
- D. Specific Rule 6, Power For C1 And D1 Buses OR EDG Start

Answer: D

Explanation/Justification: SRO only white paper item E page 7 first bullet. Constructed like example on page 12.

- A. Incorrect. Plausible since all rods did not insert.
- B. Incorrect. RCS temperature is high but subcooling does exist (candidate must calculate subcooling).
- C. Incorrect. RCS pressure is high but RCS temperature is also high which are not indications of a threat to PTS
- D. Correct. IAW DB-OP-02000 Revision 26 step 4.1 supplemental actions page 16. SRO ONLY since it requires the candidate to assess the given plant conditions and determine when to implement the specific rules (appendices). Although the EDG has started, the SRO must be familiar enough with the content of Specific Rule 6 to know that the procedure also includes steps to address closing EDG output in the RNO column and the specific rule entry is still required.

Sys #	System	Category	KA Statement
000056	Loss of Offsite Power	Generic	Ability to interpret and execute procedure steps.
K/A#	2.1.20	K/A Importance 4.6	Exam Level SRO
References provided to Candidate		Steam Tables	Technical References: DB-OP-02000 R26 step 4 supplemental actions page 16 & step 6.2 RNO on page 246.
Question Source: New		Level Of Difficulty: (1-5) 3	
Question Cognitive Level: High - Analysis		10 CFR Part 55 Content: (CFR: 55.43(b)(5))	

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77. The plant was operating at 100% power with all systems in normal alignment.
- A 460 gpm Steam Generator Tube Rupture (SGTR) occurs on SG1
 - The reactor automatically trips coincident with a Loss of Off Site Power
 - All systems function as designed

The crew is cooling down and depressurizing the RCS by performing the actions of DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture.

During this cooldown and depressurization, which of the following conditions are **required** before the command SRO will implement Attachment 17, Isolation of a SG?

- A. T_{hot} reaches 520 °F AND RCS pressure reaches 1000 psig
- B. T_{hot} reaches 500 °F AND RCS pressure reaches 1000 psig
- C. SG1 indicated level is rising AND reaches 200 inches
- D. LPI system flow is ≥ 1350 gpm AND has been for at least 20 minutes.

Answer: A

Explanation/Justification: SRO only white paper item E page 7 first bullet.

- A. Correct. IAW DB-OP-02000 Rev. 26 step 8.37. These are the conditions necessary for the SRO to implement Attachment 17 for isolating a SG. This is SRO only since it requires the candidate to assess the conditions, including diagnosing Reactor Coolant Pumps tripped due to the loss of offsite power that will require the implementation of the Attachment for isolating a ruptured SG.
- B. Incorrect. These are the conditions necessary to isolate a ruptured SG if the plant is being cooled with forced circulation (RCP running)
- C. Incorrect. These are conditions in the SGTR procedure for increasing the C/D rate to 235 °F/hr.
- D. Incorrect. These are the conditions necessary to stop HPI/MU flow.

Sys #	System	Category	KA Statement	
000038	Steam Generator Tube Rupture (SGTR)	EA2 Ability to determine and interpret the following as they apply to SGTR:	When to isolate one or more S/Gs	
K/A#	EA2.01	K/A Importance	4.7	Exam Level
References provided to Candidate		None	Technical References:	SRO DB-OP-02000 Rev. 26 step 8.37 page 116
Question Source:		New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:	10 CFR: 55.43(b)(5)
Objective:				

(SRO ONLY)
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78. • Plant is in Mode 1
 • 120VAC Instrument Bus Y1 has been transferred to Constant Voltage Transformer XY1.

Subsequently, the following event occurs:

- D1P01, DCMCC1 feeder to bus D1P breaker OPENS and bus D1P is de-energized.
- All systems function as designed.

Based on these Plant conditions what is the Operability status of the following:

- (1) Inverter YV1
(2) Essential 120VAC Instrument Bus Y1

- A. (1) Inverter YV1 is Operable
 (2) Y1 is Operable
- B. (1) Inverter YV1 is Operable
 (2) Y1 is Inoperable
- C. (1) Inverter YV1 is Inoperable
 (2) Y1 is Operable
- D. (1) Inverter YV1 is Inoperable
 (2) Y1 is Inoperable

Answer: C

Explanation/Justification: SRO only white paper item B 3rd bullet on page 3 and similar to example on page 13.

- A. Incorrect. Part 1 is incorrect. Part 2 is correct. See correct answer explanation
- B. Incorrect. Both parts are incorrect. See correct answer explanation
- C. Correct. IAW TS Bases 3.8.7 and 3.8.9 The inverter is inoperable if the battery is unavailable even if the rectifier AC source is available and the bus is operable as long as it is energized. SRO only since it requires knowledge of the TS bases to analyze plant conditions to determine if TS action is required (operability determination). Question cannot be answered with RO plant knowledge alone since both the inverter and the bus are energized yet one is operable and one is not. This requires knowledge of the TS bases to make this determination. Opening of this breaker results in the D1P bus de-energizing which results in loss of battery supply to YV1. However YV1 will remain energized from the rectifier and the Y1 bus will remain energized from the constant voltage transformer.
- D. Incorrect. Part 1 is correct. Part 2 is incorrect. See correct answer explanation

Sys #	System	Category	KA Statement
000058	Loss of DC Power	AA2. Ability to determine and interpret the following as they apply to the Loss of DC Power:	DC loads lost; impact on ability to operate and monitor plant systems
K/A#	AA2.03	K/A Importance 3.9	Exam Level SRO
References provided to Candidate		None	Technical References: TS Bases B.3.8.7 and 3.8.9 pages B 3.8.7-2 and B 3.8.9-2. OS-060 sheet 2
Question Source:		New	Level Of Difficulty: (1-5) 4
Question Cognitive Level:		High - Analysis	10 CFR Part 55 Content: 10 CFR: 55.43(b)(2)
Objective:			

(SRO ONLY)
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79. The Plant is in Mode 6 with the Reactor Coolant System level is being drained to 18 inches above hot leg centerline. DH Train 2 is in service providing core cooling; The Steam Generator Nozzle Dams are installed with lower SG Primary Side Manways removed to allow SG inspections.
- (1) How will the DH removal system alignment, be administratively controlled, to limit the impact of a loss of instrument air?
- (2) The operating DH pump trips, IAW Tech Spec bases, why is the equipment hatch and at least one door in each air lock closed?
- A. (1) Throttle DH1A, DH Pump 2 Discharge to RCS to limit total flow through the decay heat removal system.
(2) To ensure the offsite dose limits are not exceeded.
- B. (1) Throttle DH14A, DH Cooler 2 Outlet Flow Control Valve to limit total flow through the decay heat removal system.
(2) To prevent injuries due to steam/hot water environment.
- C. (1) Throttle DH1A, DH Pump 2 Discharge to RCS to limit total flow through the decay heat removal system.
(2) To prevent injuries due to steam/hot water environment.
- D. (1) Throttle DH14A, DH Cooler 2 Outlet Flow Control Valve to limit total flow through the decay heat removal system.
(2) To ensure the offsite dose limits are not exceeded.

Answer: A

Explanation/Justification: SRO only white paper item B page 3 3rd bullet. Similar to example on page 13.

- A. Correct - At low RCS level, the DH System is vulnerable to cavitation during a loss of instrument air. To prevent this from occurring, a motor operated throttle valve is set in accordance with the system operating procedure to limit flow in the event of a loss of instrument air. Part 2 is SRO ONLY since it requires knowledge of TS bases.
- B. Incorrect – Plausible because throttling DH14A would limit flow, but the valve fails open on a loss of air and would cause excess flow from the RCS. Part 2 is incorrect TS bases but plausible since boiling would occur and access to containment would be hazardous.
- C. Incorrect – Part 1 is correct. Part 2 is incorrect TS bases but plausible since boiling would occur and access to containment would be hazardous.
- D. Incorrect – Plausible because throttling DH14A would limit flow, but the valve fails open on a loss of air and would cause excess flow from the RCS. Part 2 is correct.

Sys #	System	Category	KA Statement	
000065	Loss of Instrument Air	Generic	Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	
K/A#	2.4.9	K/A Importance	4.2	Exam Level
References provided to Candidate		None	Technical References:	
Question Source:		New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:	10 CFR: 55.43(b)(2)
Objective:				

(SRO ONLY)
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80. The plant is operating at 100% power with all systems in normal alignment.

- A reactor trip coincident with a loss of offsite power occurs.
- All plant systems respond normally.
- SG levels are 49 inches and stable

Which of the following DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture specific rules, attachments, or supplemental actions are **required** to be entered **AND** what actions will be taken to mitigate these plant conditions?

- A. Enter Specific Rule 4, Steam Generator Control and raise steam generator level to 124 inches.
- B. Enter Attachment 1, Primary Inventory Control Actions and lineup HPI piggyback operations.
- C. Enter Supplemental Action step 4.6, Check for ICS Power available and Initiate and Isolate SFRCS.
- D. Enter Attachment 2, Steam Generator Inventory and Pressure Control and locally throttle the Atmospheric Vent Valves.

Answer: D

Explanation/Justification: SRO only white paper item E page 6 first paragraph and example question on page 11.

- A. Incorrect – Although the loss of Off-Site power causes all RCPs to be lost, the raised loop design at Davis-Besse does not require raising level to 124 inches to promote Natural Circulation Cooling. This is the correct action to promote boiler condenser SG heat transfer if Subcooling Margin is lost.
- B. Incorrect – The previously running MU Pump will restart when the EDG starts. Although cooling to the running Makeup Pump is lost, the pump can operate for at least 1 hour without cooling and the standby Makeup Pump would be available. This is the correct procedure action if all Makeup Pumps are lost.
- C. Incorrect – Although this is the correct action for a loss of ICS power, ICS power is provided by YAU and YBU and is not lost of a loss of Off-Site power.
- D. Correct – IAW DB-OP-02000 step 4.4 and attachment 2 step 2 RNO. The loss of Off-site power will result in an SFRCS Actuation on Steam to Feed Differential Pressure which will cause the AVVs to close. Main Steam Safety Valves would then relieve to remove heat. Manual control of AVV would allow reseating MSSVs, however the loss of off site power will also cause a loss of instrument air preventing control room operations of the AVVs.

Sys #	System	Category	KA Statement
BW/E10	Post-Trip Stabilization	EA2. Ability to determine and interpret the following as they apply to the (Post-Trip Stabilization)	Facility conditions and selection of appropriate procedures during abnormal and emergency operations.
K/A#	EA2.1	K/A Importance 4.0	Exam Level SRO
References provided to Candidate	None	Technical References:	DB-OP-02000 Rev. 26 Supplemental action step 4.4 and Att. 2 step 2 RNO
Question Source:	New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:	High - Comprehension	10 CFR Part 55 Content:	10 CFR: 55.43(b)(5)
Objective:			

(SRO ONLY)
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81. The plant is operating at 100% power.

The following events occur:

- Reactor Trip
- Main Steam Safety Valves on **BOTH** Steam Generators stick open and can not be reseated.
- All Feedwater Flow is isolated to #1 Steam Generator.
- Trickle Feed is established to #2 Steam Generator
- RCS Pressure is stabilized at 1700 psig.

In accordance with DB-OP-02000, RPS, SFAS, SFRCS Trip or SG Tube Rupture, what Steam Generator Level if any, is **required** to be maintained for these plant conditions?

- A. No specific Level requirement exists for these plant conditions.
- B. Maintain 40 inches using Main Feedwater flow.
- C. Maintain 49 inches using Auxiliary Feedwater flow.
- D. Maintain 124 inches using Auxiliary Feedwater flow.

Answer: A

- Explanation/Justification:** SRO only white paper item E 1st paragraph page 6. This is NOT system knowledge since automatic controls must be overridden to implement the trickle feed strategy. The question is at the SRO level since it requires detailed knowledge of how this is accomplished. This is a significant step in implementing this method because if normal levels are maintained, a significant overcooling event will occur. This goes beyond basic purpose, overall sequence, and mitigating strategy of the procedure which indicates SRO only.
- A. Correct per **note** for step 7.28 RNO. SRO ONLY since it requires the candidate to assess the given plant conditions and then select the appropriate procedure actions contained in that section, in this case a note preceding a supplemental action step.
 - B. Incorrect – Plausible because this the normal method of control and sytem used to maintain SG Level following a Reactor Trip
 - C. Incorrect – Plausible because this the normal method of control and sytem used to maintain SG Level following AFW Actuation that would occur when Feedwater isolated to #1 SG.
 - D. Incorrect – Plausible because this the normal method of control and sytem used to maintain SG Level following AFW Actuation that would occur when Feedwater isolated to #1 SG. and RCS Pressure reduction caused by overcooling resulted in an SFAS actuation.

Sys #	System	Category	KA Statement		
BW/E05	Steam Line Rupture - Excessive Heat Transfer	Generic	Knowledge of the operational implications of EOP warnings, cautions, and notes.		
K/A#	2.4.20	K/A Importance	4.3	Exam Level	SRO
References provided to Candidate		None	Technical References:	DB-OP-02000 Step 7.28 RNO DB-OP-02000 Specific Rule 4	
Question Source:		New	Level Of Difficulty: (1-5)	3.5	
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:	10 CFR: 55.43(b)(5)	
Objective:					

(SRO ONLY)

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82. A plant startup-up is in progress in middle of Core life with the following conditions:

- Reactor power is 10^{-8} amps on the Intermediate Range and stable
- Rod Control is in MANUAL
- ICS Reactor Demand is in MANUAL
- Group 6 Rods are at 50%
- The Rod Control Group Select Switch is selected to Group 6
- All Rods are on their normal power supply.

From these initial conditions, the following occurs:

- The Group 6 rods continuously withdraw outward. NI3 and NI4 SUR is 4 DPM.

In accordance with DB-OP-02516, CRD Malfunctions:

(1) What operator actions are **required**?

In the event that Rod Motion does not stop:

(2) IAW Tech. Spec. bases, what Automatic RPS Trip is credited for terminating the event?

- A. (1) Depress and hold the Rod Stop Push Button
(2) High RCS Temperature.
- B. (1) Depress and hold the Rod Stop Push Button
(2) High RCS Pressure.
- C. (1) Turn the Group Select switch to Group 5 position
(2) High RCS Temperature.
- D. (1) Turn the Group Select switch to Group 5 position
(2) High RCS Pressure.

Answer: B

Explanation/Justification: SRO only white paper item B 3rd bullet page 3 and example question on page 13.

- A. Incorrect – Depressing the Rod Stop Pushbutton is the Operator Action directed to stop continuous Rod withdraw by DB-OP-02516. High RCS Temperature trip is plausible because the RCS will heatup once the point of adding heat is reached.
- B. Correct – Depressing the Rod Stop Pushbutton is the Operator Action directed to stop continuous Rod withdraw by DB-OP-02516. High RCS Pressure is the RPS trip that will automatically terminate the event. SRO ONLY since part 2 of the question requires knowledge of TS bases and accident analysis associated with that bases.
- C. Incorrect – While selecting Group 5 with all group 5 rods already withdrawn may stop all rod motion, this action is not directed by DB-OP-02516, CRD Malfunctions. High RCS Temperature trip is plausible because the RCS will heatup once the point of adding heat is reached.
- D. Incorrect – While selecting Group 5 with all group 5 rods already withdrawn may stop all rod motion, this action is not directed by DB-OP-02516, CRD Malfunctions. High RCS Pressure is the automatic trip that will automatically terminate the event.

Sys #	System	Category	KA Statement		
000001	Continuous Rod Withdrawal	AA2. Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal :	Proper actions to be taken if automatic safety functions have not taken place		
K/A#	AA2.03	K/A Importance	4.8	Exam Level	SRO
References provided to Candidate		None	Technical References:		DB-OP-02516 R13 CRD Malfunctions. USAR 15.2.1 Startup Accident - Uncontrolled Control Rod Assembly Group Withdrawal. TS bases page B 3.3.1-9
Question Source:		New	Level Of Difficulty: (1-5)		3.5
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:		10 CFR: 55.43(b)(2)

(SRO ONLY)
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83. A plant startup-up is in progress with the following conditions:

- Reactor power is 3% and stable
- A single Group 2 rod is found at 95%

(1) What Tech. Spec. action is **required**?

(2) The Tech. Spec. regulating rod and safety rod insertion limits ensures the safety analysis assumptions for which of the following remain valid?

- I. Ejected rod worth
- II. Dropped rod worth
- III. Reactivity limits
- IV. Shutdown Margin
- V. MTC is within the limits of the COLR

- A. (1) Verify SDM is within limit OR Initiate boration to restore SDM to within limit.
(2) II, III, & IV **ONLY**
- B. (1) Verify SDM is within limit OR Initiate boration to restore SDM to within limit.
(2) I, III, & IV **ONLY**
- C. (1) Verify SDM is within limit AND Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group insertion limits.
(2) II, & V **ONLY**
- D. (1) Verify SDM is within limit AND Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group insertion limits.
(2) I, & V **ONLY**

Answer: B

Explanation/Justification: SRO only white paper item B 3rd bullet page 3 and example question on page 13.

- A. Incorrect. Right TS action. Wrong bases dropped rod worth is not a bases and Ejected rod worth is a bases.
- B. Correct. IAW TS 3.1.5-1 Amend 279 TS Bases B 3.1.5-1 Rev. 0. Part 1 is RO knowledge since it requires the candidate to recognize the conditions that require TS actions for boration. At DB the term emergency boration was used in the old TS for this action. In the new TS, this is only referred to as initiate boration. For DB this meets the intent of any TS LCO relative to emergency boration. Part 2 is SRO ONLY in that it requires the candidate to have knowledge of TS bases. Discussed with chief examiner to get concurrence that this approach to E-boration at DB still meets the K/A as written. The IR does have a high SUR rod out inhibit however this is not credited in accident analysis or TS bases.
- C. Incorrect. This is the correct TS action if the rod were a group 5, 6, or 7 rod. Wrong bases.
- D. Incorrect. This is the correct TS action if the rod were a group 5, 6, or 7 rod. Wrong bases.

Sys #	System	Category	KA Statement	
000024	Emergency Boration	Generic	Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	
K/A#	2.2.42	K/A Importance	4.6	Exam Level
References provided to Candidate		None	Technical References:	SRO TS 3.1.5-1 Amend 279 TS Bases B 3.1.5-1 Rev. 0
Question Source:		New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:		Low - Fundamental	10 CFR Part 55 Content:	10 CFR: 55.43(b)(2)
Objective:				

(SRO ONLY)
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84. A plant startup-up is in progress with the following conditions:
- Reactor power, as indicated on NI 3 and NI 4 (intermediate range detectors), is 1×10^{-8} amps
 - All systems are in normal alignment for this condition

A fuse in the power supply to the NI 3 detector blows (detector supply voltage is zero).

(1) How will the NI 1 and NI 2 (source range detectors) respond to this blown fuse?

(2) IAW Technical Specifications, what actions are **required**?

(References provided)

- A. (1) re-energize
(2) within 2 hours, Reduce neutron flux to $\leq 1\text{E-}10$ amp.
- B. (1) re-energize
(2) once per 12 hours, Verify SDM is within the limits specified in the COLR.
- C. (1) remain de-energized
(2) within 2 hours, Reduce neutron flux to $\leq 1\text{E-}10$ amp.
- D. (1) remain de-energized
(2) once per 12 hours, Verify SDM is within the limits specified in the COLR.

Answer: A

Explanation/Justification: SRO only white paper item B 3rd bullet page 3 and example question on page 13.

- A. Correct. IAW DB-OP-02505 Rev. 05 Att. 1 pages 34 & 35; TS Bases page B 3.3.10-1 Rev. 1. Part 1 is RO knowledge. Candidate must know the IR and PR contact alignment for the given power level, then determine the indication/impact of the blown fuse on the source range detectors. Part 2 is SRO ONLY Tech Spec application of greater than 1 actions.
- B. Incorrect. Right impact and indication. Wrong TS action. Plausible if candidate applies TS actions for both SR being inoperable (energized above its top range).
- C. Incorrect. Wrong impact. Plausible if candidate does not know the contact alignment or setpoints for re-energizing the source ranges. Right TS action.
- D. Incorrect. Wrong impact. Plausible if candidate does not know the contact alignment or setpoints for re-energizing the source ranges. Wrong TS action. Plausible if candidate applies TS actions for both SR being inoperable (both de-energized but a TS note allows this).

Sys #	System	Category	KA Statement
000033	Loss of Intermediate Range Nuclear Instrumentation	AA2. Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation:	Indication of blown fuse
K/A#	AA2.03	K/A Importance 3.1	Exam Level SRO
References provided to Candidate		Tech Specs 3.3.9 and 3.3.10	Technical References: DB-OP-02505 Rev. 05 Att. 1 pages 34 & 35; TS Bases page B 3.3.10-1 Rev. 1
Question Source:		New	Level Of Difficulty: (1-5) 3.5
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content: 10 CFR: 55.43(b)(2)
Objective:			

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85. The following plant conditions exist at the times(t) specified:

t = 0	The plant was at 100% power.
t = 30 sec	Toxic fumes have entered the control room.
t = 1 min.	DB-OP-02508, Control Room Evacuation has been implemented.
t = 1.5 min.	The reactor and turbine are tripped.
t = 2 min.	SFRCS has been actuated.
t = 5 min	Letdown is isolated.
t = 6 min	The standby Makeup Pump has been started.
t = 20 min	Local shutdown control from the Aux Shutdown Panel has been established.
t = 25 min	Steam generator pressures are between 980 and 1000 PSIG and stable.
t = 30 min	Steam generator levels are stable at 49 inches.

Based on this sequence of events and these indications, what is the **HIGHEST** Emergency Classification?

(References provided)

- A. Unusual Event
- B. Alert
- C. Site Area Emergency
- D. General Emergency

Answer: C

Explanation/Justification: SRO only white paper item F 1st bullet page 9. The note in DB-OP-02508 page 15 is a reminder that if control has not been established W/I 15 minutes, core conditions may be threatened and SAE is appropriate.

- A. Incorrect. Plausible HU5
- B. Incorrect. Plausible HA2 and HA5
- C. Correct. IAW RA-EP-01500, Emergency Classification Rev. 14. HS2 page 29. The note in DB-OP-02508 page 15 reminds the Shift manager that a site area is warranted if control from the Aux Shutdown Panel has NOT been established within 15 minutes. SRO ONLY since requires the candidate to analyze and interpret plant conditions and sequences to select the appropriate EAL. At Davis Besse this is an SRO ONLY task for the on-shift ERO.
- D. Incorrect. Plausible if the HA6 is inappropriately applied.

Sys #	System	Category	KA Statement		
BW/A06	Shutdown Outside Control Room	Generic	Ability to explain and apply system limits and precautions.		
K/A#	2.1.32	K/A Importance	4.0	Exam Level	SRO
References provided to Candidate		RA-EP-01500, Emergency Classification Rev. 14	Technical References:		DB-OP-02508 Rev. 12 Att. 2 page 15 Note 3; RA-EP-01500, Emergency Classification Rev. 14. HS2 page 29
Question Source:		New	Level Of Difficulty: (1-5)		3
Question Cognitive Level:		High - Application	10 CFR Part 55 Content:		10 CFR: 55.43(b)(6)
Objective:					

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86. The plant is operating at 100% power with all systems in normal alignment with the exception that HPI Train 2 is out of service for planned maintenance.

At 0800, a reactor trip occurs. SFAS Actuates on Low RCS Pressure, Low-Low RCS Pressure and High Containment Pressure.

At 0830, BWST level is 39 feet and lowering and level will reach 9 feet at 1630.

At 0900, LPI Train 1 **AND** 2 indicate 0 gallons per minute.

At 0930, Incore temperatures have stabilized at approximately 480 °F with RCS pressure at 500 psig.

Which ONE (1) of the following DB-OP-02000 Attachments provides the required actions that mitigate these plant events?

- A. Attachment 11, HPI Flow Balancing.
- B. Attachment 12, Establishing Long Term Boron Dilution.
- C. Attachment 14, Establishing HPI Alternate Minimum Recirc Flowpath.
- D. Attachment 22, Cross Connect LPI Pump Discharge.

Answer: A

Explanation/Justification: SRO only white paper item E Page 7 first bullet. Similar to question example from page 12.

- A. Correct – Flow Balancing HPI is required during single train operation to protect against an HPI Line Break to ensure at least one HPI injection line flow is reaching the core. SRO ONLY since it requires the candidate to select the appropriate procedure attachment to mitigate the event.
- B. Incorrect – Long term Boron dilutions is required when RCS temperatures are less than 333 °F. At higher temperatures, the boron in the RCS will not precipitate out of solution. As a result, Long Term Boron Dilution is not required for these plant conditions.
- C. Incorrect – HPI Alternate Minimum Recirc is required when BWST level is being reduce at less than 2 foot per hour. At higher flow rates, the RCS will not repressurize above the shutoff head of the HPI Pump. As a result, HPI Alternate Minimum Recirc Flow is not required for these plant conditions.
- D. Incorrect – LPI Pump Discharge is required when a single LPI train is not available. Although no LPI flow exists in this scenario, LPI flows are consistent with the current Plant conditions. As a result, cross connecting LPI discharge is not required.

Sys #	System	Category	KA Statement
006	Emergency Core Cooling System (ECCS)	A2. Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	System leakage
K/A#	A2.03	K/A Importance 3.7	Exam Level SRO
References provided to Candidate		Steam Tables	Technical References: DB-OP-02000 R26 Attachment 11, page 321
Question Source:		New	Level Of Difficulty: (1-5) 3
Question Cognitive Level:		High - Analysis	10 CFR Part 55 Content: 10 CFR: 55.43(b)(5)
Objective:			

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87. The plant is operating at 100% power.

The following events occur:

- A LOCA inside CTMT occurs
- The reactor automatically TRIPS.
- Reactor Coolant System pressure is 185 psig and LOWERING
- Incore temperature is 380 °F and LOWERING.
- Containment Pressure peaked at 37.5 psia and is now STABLE.
- All systems function as designed

- (1) Prior to any operator actions, what will be the status of the Safety Actuation Monitor (SAM) light for CC1407A, CCW FROM CTMT?
- (2) What is the Technical Specification bases for the Safety Features Actuation System (SFAS) Instrumentation?

- A. (1) DIM
(2) To prevent or limit fission product and energy release from the core, to isolate the containment vessel, and to initiate the operation of ESF equipment.
- B. (1) OFF
(2) To prevent or limit fission product and energy release from the core, to isolate the containment vessel, and to initiate the operation of ESF equipment.
- C. (1) DIM
(2) Ensures the Emergency Core Cooling Systems (ECCS) acceptance criteria are met following a LOCA.
- D. (1) OFF
(2) Ensures the Emergency Core Cooling Systems (ECCS) acceptance criteria are met following a LOCA.

Answer: B

Explanation/Justification: SRO only white paper item B Page 3 third bullet. Similar to question example from page 13.

- A. Incorrect. Part 1 DIM is the correct SAM light indication if a SFAS signal were present and the equipment was in the SFAS required position. Based on plant conditions, SFAS Level 4 has not actuated and therefore DIM light status is not correct since CC1407A has not received and SFAS signal. – Part 2 is correct.
- B. Correct. IAW DB-OP-06405 Rev. 13 Attachment 2 and TS 3.3.5 bases. SFAS Level 4 has not actuated and therefore OFF status is correct. Part 1 is RO knowledge in that the ROs should be capable of determining which ESF functions will actuate for the conditions of the stem and they should also be capable of determining the status of the SAM lights. Part 2 is SRO only since it requires knowledge of the TS bases for SFAS.
- C. Incorrect. Part 1 DIM is the correct SAM light indication if a SFAS signal were present and the equipment was in the SFAS required position. Based on plant conditions, SFAS Level 4 has not actuated and therefore DIM light status is not correct since CC1407A has not received and SFAS signal. Part 2 is the TS bases for ECCS not ESF actuation system.
- D. Incorrect. Part 1 is the correct SAM light indication if NO SFAS signal were present OR an SFAS signal was present and the equipment was NOT in the SFAS required position. Based on plant conditions, CC1407A does not have an SFAS signal. Part 2 is the TS bases for ECCS not ESF actuation system.

Sys #	System	Category	KA Statement		
013	Engineered Safety Features Actuation System (ESFAS)	Generic	Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.		
K/A#	2.4.21	K/A Importance	4.6	Exam Level	SRO
References provided to Candidate		None	Technical References:	DB-OP-06405 Rev. 13 Attachment 2 and TS 3.3.5 bases page B 3.3.5-1 Revision 1	
Question Source:		New	Level Of Difficulty: (1-5)		4
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:		10 CFR: 55.43(b)(2)

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88. A Large Break Loss of Coolant Accident has occurred.

Once ECCS suction is transferred to the Emergency Sump, the following indications are noted:

- LPI Train 1 & 2 Flows – BOTH 3900 gpm and stable
- Containment Spray Train 1 Flow – 2000 gpm and stable
- Containment Spray Train 2 Flow – flow fluctuating between 1000 gpm and top of scale
- LPI Train 1 & 2 motor amps – BOTH 60 amps and stable
- Containment Spray Train 1 motor amps 180 amps and stable
- Containment Spray Train 2 motor amps fluctuating between 80 amps and top of scale.
- CS1531 CTMT Spray Train 2 Disch is full OPEN
- DH1A and DH1B LPI injection valves are full OPEN

Which ONE (1) of the following DB-OP-02000 attachments, **require** implementation and what actions will be taken to mitigate these conditions?

- A. Perform Attachment 7, Transferring LPI Suctions to the Emergency Sump, throttle LPI Injections valves DH1A and DH1B.
- B. Perform Attachment 7, Transferring LPI Suctions to the Emergency Sump, throttle CS1531 CTMT Spray Train 2 Disch.
- C. Perform Attachment 27, Mitigation of CTMT Emergency Sump Degradation, throttle LPI Injections valves DH1A and DH1B.
- D. Perform Attachment 27, Mitigation of CTMT Emergency Sump Degradation, throttle CS1531 CTMT Spray Train 2 Disch.

Answer: B

Explanation/Justification: SRO only white paper item E Page 7 first bullet. Similar to question example from page 13.

- A. Incorrect – Part 1 correct. Part 2 actions are incorrect, these are the correct actions if both trains were impacted.
- B. Correct –Part 1 is SRO since it requires the SRO to select which attachment is to be used. DB-OP-02000 Attachment 7 directs verifying CTMT Spray Discharge Valves are positioned to the Throttle position following transfer of ECCS Pump Suctions to the Emergency Sump. Part 2 actions are correct IAW attachment 7.
- C. Incorrect – Part 1 is incorrect. Since only one train is affected. Part 2 is incorrect but these are the correct actions for both trains being affected.
- D. Incorrect – Part 1 is incorrect. Since only one train is affected. Part 2 is correct

Sys #	System	Category	KA Statement
026	Containment Spray System (CSS)	A2. Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Loss of containment spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature exceeded cavitation, voiding), or sump level below cutoff (interlock) limit
K/A#	A2.07	K/A Importance	3.9
References provided to Candidate	None	Exam Level	SRO
Question Source:	New	Technical References:	DB-OP-02000, Attachment 7 USAR Section 6.2.2.2.2 Containment Spray System
Question Cognitive Level:	High - Comprehension	Level Of Difficulty: (1-5)	3
		10 CFR Part 55 Content:	10 CFR: 55.43(b)(5)

(SRO ONLY)

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89. The plant is operating at 80% power, with all systems in a normal lineup **EXCEPT**:
- Unit Load Demand (ULD) is in MAN

The following events occur:

- Tave begins to lower
- Generator MWs begin to lower
- Control rods begin moving OUT
- Feedwater to SG 1 begins to rises
- PZR level is slowly lowering
- Containment pressure begins to rise at 0.1 psig/minute
- Containment temperature begins to rise at 0.25 °F/minute
- Containment radiation remains constant

Based on these indications, how will Reactor power respond to these conditions?

(1) Reactor power will _____.

(2) What is the Technical Specification bases for maximum Containment Temperature?

- A. (1) lower
(2) Containment temperature must be maintained below its maximum allowable value to ensure RCS Leakage Detection Instrumentation remains operable.
- B. (1) lower
(2) Containment temperature must be maintained below its maximum allowable value to ensure that the containment structural temperature is maintained below its design temperature and that required safety related equipment will continue to perform its function.
- C. (1) rise
(2) Containment temperature must be maintained below its maximum allowable value to ensure RCS Leakage Detection Instrumentation remains operable.
- D. (1) rise
(2) Containment temperature must be maintained below its maximum allowable value to ensure that the containment structural temperature is maintained below its design temperature and that required safety related equipment will continue to perform its function.

Answer: D

Explanation/Justification: Part 1 is RO since it can be answered with system knowledge and plant response to transients. Part 2 is SRO only white paper item B Page 3 third bullet. Similar to question example from page 13

- A. Incorrect. Wrong Rx power response, The Bases is modified RCS Leakage instrumentation, not CTMT Temperature.
- B. Incorrect. Wrong Rx power response, wrong actions. These are the right TS Bases for CTMT Temperature.
- C. Incorrect. Right Rx power response, wrong TS Bases. The Bases is modified RCS Leakage instrumentation, not CTMT Temperature
- D. Correct. Correct power response and correct TS Bases for CTMT Temperature.

Sys #	System	Category	KA Statement
039	Main and Reheat Steam System (MRSS)	A2.05 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 3.6	Exam Level SRO
References provided to Candidate		None	Technical References: DB-OP-02525 Rev.10 steps 2.1 TS 3.6.5 Bases LCO Page 3.6.5-2
Question Source: New		Level Of Difficulty: (1-5)	3
Question Cognitive Level: High - Analysis		10 CFR Part 55 Content:	10 CFR: 55.43(b)(2)

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90. The plant is operating at 100% power.

The following events occur:

- Offsite Power is lost.
- The reactor TRIPS.
- Reactor Coolant System pressure is 2350 psig and RISING
- Incore temperature is 585 °F and RISING at 2 °F /minute.
- **BOTH** Auxiliary Feedwater Pumps have TRIPPED and CANNOT be reset.
- C1 & D1 Electrical Buses are being supplied by their respective EDG.
- The Station Blackout Diesel is in standby.
- **BOTH** Makeup Pumps are running.

Based on these plant conditions, which of the following DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture attachments **require** implementation to establish core cooling?

- Attachment 4, MU/HPI/PORV Cooling
- Attachment 5, Guidelines for Restoring Feedwater
- Attachment 6, Reenergization of Buses D2, F2, and MCC F71

- A. Attachment 4 Only
- B. Attachment 4 and Attachment 6 Only
- C. Attachment 5 Only
- D. Attachment 5 and Attachment 6 Only

Answer: D

Explanation/Justification: SRO only white paper item E Page 7 first bullet. Similar to question example from page 12.

- A. Incorrect. Att. 4 would be appropriate for a loss of all Feedwater. However, the initiation point is 600°F and the plant is only at 585°F
- B. Incorrect. Att. 4 would be appropriate for a loss of all Feedwater. However, the initiation point is 600°F and the plant is only at 585°F and Att. 6 restores power that the candidate may believe is necessary to establish MU/HPI/PORV Cooling.
- C. Incorrect. Att. 5 is correct. However, Att. 6 is also required to establish power for the MDP.
- D. Correct. IAW DB-OP-02000 Rev. 26 Section 6, Steps 6.2, ATT. 5 section A, and Att. 6. SRO only since it requires knowledge of when to implement EOP attachments.

Sys #	System	Category	KA Statement	
061	Auxiliary/Emergency Feedwater (AFW) System	Generic	Knowledge of EOP mitigation strategies.	
K/A#	2.4.6	K/A Importance	4.7	Exam Level
References provided to Candidate		None	Technical References:	SRO DB-OP-02000 Rev. 26 Section 6, Steps 6.2 and ATT. 5 section A
Question Source:		New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:	10 CFR: 55.43(b)(5)
Objective:				

(SRO ONLY)
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91. The plant is operating at 100% power with all systems in normal alignment for this power level.

The following events occur:

- (2-3-A) LETDOWN TEMP HI Annunciator Alarm
- MU 32, PRESSURIZER LEVEL CONTROL, closes
- Pressurizer level 228 inches and slowly rising
- Tave is 582 °F and stable
- Makeup Tank level is slowly lowering.

Based on these indications, what procedure entry is **required** and what actions will be taken to mitigate these conditions?

- A. Enter DB-OP-02512, Makeup System Malfunctions and manually trip the reactor and go to DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture.
- B. Enter DB-OP-02512, Makeup System Malfunctions and reduce seal injection flow to 3 gpm per RCP.
- C. Enter DB-OP-02513, Pressurizer System Abnormal Operations and reduce seal injection flow to 3 gpm per RCP.
- D. Enter DB-OP-02513, Pressurizer System Abnormal Operations and manually trip the reactor and go to DB-OP-02000, RPS, SFAS, SFRCS Trip, or SG Tube Rupture.

Answer: B

Explanation/Justification: SRO only white paper item E page 6 first paragraph and example question on page 11.

- A. Incorrect. Correct procedure entry. Incorrect actions. Supplemental actions in DB-OP-02512 will require a manual reactor trip if PZR level reaches 290 inches. 228 inches is the TS LCO level. Candidates may confuse the actions required not meeting the LCO.
- B. Correct – Required action per DB-OP-02512, Makeup System Malfunctions Step 4.3.1. This is a supplemental action that does not directly address the Loss of Letdown, just minimizes the amount of excess inventory being added to the Reactor Coolant System without Letdown available until the reason for the loss of letdown is determined and corrected and requires SRO knowledge of the procedure content. Part 1 AOP entry conditions is RO knowledge. Part 2 is SRO only since it requires the additional knowledge of the actions contained in that section of the procedure and cannot be arrived at with RO system knowledge since there is a requirement to trip the reactor when PZR level reaches certain value in the AOP (290 inches).
- C. Incorrect. Incorrect procedure entry. Correct actions that will minimizes the amount of excess inventory being added to the Reactor Coolant System without Letdown available until the reason for the loss of letdown is determined and corrected.
- D. Incorrect – Incorrect procedure entry. Incorrect actions. Supplemental actions in DB-OP-02512 will require a manual reactor trip if PZR level reaches 290 inches. 228 inches is the TS LCO level. Candidates may confuse the actions required not meeting the LCO.

Sys #	System	Category	KA Statement
011	Pressurizer Level Control System (PZR LCS)	A2. Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Isolation of letdown
K/A#	A2.07	K/A Importance 3.3	Exam Level SRO
References provided to Candidate		None	Technical References: DB-OP-02512 R14, step 4.3.1
Question Source:		New	Level Of Difficulty: (1-5) 3
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content: 10 CFR: 55.43(b)(5)
Objective:			

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92. The plant is operating at 100% power with all systems in normal alignment for this power level **EXCEPT**:
- NI 5, POWER RANGE PWR (RPS CH 2) failed several days ago and ALL **required** actions of DB-OP-02505, Nuclear Instrumentation Failures have been completed.
 - **NOW** NI 6, POWER RANGE PWR (RPS CH 1) fails low and the reactor does NOT trip.
 - The crew re-enters DB-OP-02505, Nuclear Instrumentation Failures.
- (1) IAW the guidance provided in DB-OP-02505, Nuclear Instrumentation Failures, what additional actions will be **required**?
- (2) IAW the Technical Specification bases for the Reactor Protective System (RPS) Instrumentation, which of the following RPS Trip Function is credited in the Davis Besse Accident Analysis?
- A. (1)Place RPS Channel 1 in Manual Bypass.
 (2) High Flux – High Setpoint.
- B. (1)Place RPS Channel 1 in Manual Bypass.
 (2) High Flux – Low Setpoint (S/D Bypass)
- C. (1) Manually Trip RPS Channel 1.
 (2) High Flux – High Setpoint.
- D. (1) Manually Trip RPS Channel 1.
 (2) High Flux – Low Setpoint (S/D Bypass)

Answer: C

Explanation/Justification: Part 1 is RO. Part 2 is SRO only white paper item B Page 3 third bullet. Similar to question example from page 13.

- A. Incorrect. This is the required action for a single NI failure, and has already been completed for RPS CH 2. The additional failure requires tripping the affected channel since only a single channel can be placed in manual bypass. IAW TS Bases 3.3.1, the High Flux trip is credited in the DB Accident Analysis
- B. Incorrect. This is the required action for a single NI failure, and has already been completed for RPS CH 2. The additional failure requires tripping the affected channel since only a single channel can be placed in manual bypass. IAW TS Bases 3.3.1, the High Flux - low setpoint trip is NOT credited in the DB Accident Analysis
- C. Correct. IAW DB-OP-02505 step 4.1.7 on page 12. The candidate must predict the impact of the second NI failure and then select the appropriate actions as specified in the abnormal procedure. IAW TS Bases 3.3.1, the High Flux trip is credited in the DB Accident Analysis.
- D. Incorrect. This is the additional required action for a single NI failure after the first failure is place in Manual Bypass, but IAW TS Bases 3.3.1, the High Flux - low setpoint trip is NOT credited in the DB Accident Analysis

Sys #	System	Category	KA Statement
015	Nuclear Instrumentation System	A2. Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:	Power supply loss or erratic operation
K/A#	A2.01	K/A Importance 3.9	Exam Level SRO
References provided to Candidate		None	Technical References: DB-OP-02505 Rev. 5 step 4.1.7 on page 12 TS Bases .3.3.1 page 3.3.1-9 Applicable Safety Analysis
Question Source:		New	Level Of Difficulty: (1-5) 4
Question Cognitive Level:		High - Analysis	10 CFR Part 55 Content: 10 CFR: 55.43(b)(2)

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93. Fuel handling operations in the Spent Fuel Pool are in progress.
- A loss of Spent Fuel Pool inventory due to a Spent Fuel Pool Cooling System pipe break occurs.
 - Spent Fuel Pool temperature is 130 °F.

In accordance with DB-OP-02547, Spent Fuel Pool Cooling Malfunctions which of the following actions will be implemented in response to this event?

1. Stop the operating SFP Cooling Pumps.
2. Align the operating DH Removal Pump to provide SFP Cooling.
3. Suspend Fuel Handling operations in the Spent Fuel Pool.
4. Suspend Spent Fuel Pool Crane Operations.
5. Restore the Component Cooling Water Non-Essential Header to service.
6. Manually initiate Emergency Ventilation on the Spent Fuel Pool.

- A. 1, 2, 3, & 5 only
- B. 1, 3, 4, & 6 only
- C. 2, 4, 5, & 6 only
- D. 3, 4, 5 & 6 only

Answer: B

Explanation/Justification: SRO only white paper item E Page 6 first paragraph on content of procedure vs. overall mitigative strategy. This question requires the candidate to select the correct 4 procedure steps (content) vs. overall mitigation strategy or purpose. In addition, actions 3 and 4 (only in the correct response) don't directly mitigate the event.

- A. Incorrect – Plausible because all actions are from DB-OP-02547, Spent Fuel Pool Cooling Malfunctions. Actions 2 and 5 are correct for a loss of cooling, not a loss of inventory.
- B. Correct – These are the supplemental actions from DB-OP-02547, Spent Fuel Pool Cooling Malfunctions, Section 4.2, Loss of SFP Inventory. SRO only since it requires the candidate to assess the plant conditions and have the additional knowledge of the actions contained in that procedure.
- C. Incorrect – Plausible because all actions are from DB-OP-02547, Spent Fuel Pool Cooling Malfunctions. Actions 2 and 5 are correct for a loss of cooling, not a loss of inventory
- D. Incorrect – Plausible because all actions are from DB-OP-02547, Spent Fuel Pool Cooling Malfunctions. Actions 2 and 5 are correct for a loss of cooling, not a loss of inventory

Sys #	System	Category	KA Statement
033	Spent Fuel Pool Cooling System (SFPCS)	Generic	Ability to interpret and execute procedure steps.
K/A#	2.1.20	K/A Importance	4.6
References provided to Candidate	None	Exam Level	SRO
Question Source:	New	Technical References:	DB-OP-2547R02 Sect 4.2.1, 4.2.3.2, 4.3.3.3, and 4.2.13 pages 16 and 22.
Question Cognitive Level:	High - Comprehension	Level Of Difficulty: (1-5)	3
		10 CFR Part 55 Content:	10 CFR: 55.43(b)(5)

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94. The plant is in Mode 6 with Fuel Handling in progress.

Fuel Handling will be suspended for approximately 30 hours.

All Fuel Handling Surveillances will be maintained current.

Which one of the following requirements must be observed during the suspension?

- A. A qualified individual must be assigned to monitor Refueling Canal Level and notify the Control Room of any lowering Refueling Canal Level.
- B. A dedicated Reactor Operator must be assigned to monitor the reactivity of the core (neutron count rate).
- C. At least one Emergency Ventilation System Fan must be in service on the Spent Fuel Pool.
- D. The gate between the Spent Fuel Pool and the Transfer Pool shall be installed and the gate valves on the transfer tubes closed as far as possible without damaging the transfer equipment cable.

Answer: D

Explanation/Justification: SRO only white paper item F Page 9 third bullet – Administrative Requirements associated with Refueling Activities.

- A. Incorrect – Lowering of Refueling Canal level requires suspension of the Fuel Handling activities. Suspending fuel handling activities does not require continuous monitoring of refueling canal level.
- B. Incorrect – A dedicated individual is only required to be assigned to monitor the reactivity of the core (neutron count rate) during fuel handling activities that add positive reactivity to the reactor core.
- C. Incorrect – This action would be required if the SFP Ventilation system was not in service.
- D. Correct – This is a required action when suspending fuel handling operations for greater than 24 hours. SRO ONLY in that it requires knowledge of administrative requirements associated with refueling activities.

Sys #	System	Category	KA Statement		
N/A	N/A	Generic	Knowledge of new and spent fuel movement procedures.		
K/A#	2.1.42	K/A Importance	3.4	Exam Level	SRO
References provided to Candidate		None	Technical References:	DB-OP-00030 R12, Fuel Handling Operations Step 6.3.3.	
Question Source:		New	Level Of Difficulty: (1-5)		3.5
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:		10 CFR: 55.43(b)(6)
Objective:					

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95. A Plant startup is in progress with the following plant conditions:

- Plant is in Mode 3
- RCS temperature is 532 °F and stable
- RCS pressure is 2155 psig and stable
- All systems in normal alignment for this power level.

IAW Technical Specifications, Which of the below listed RPS functions **OR** nuclear instruments are **required** to be operable **PRIOR** to closing the CRD trip breakers, with the CRD system capable of rod withdrawal?

- s. High Flux – High setpoint
- t. High Flux – Low setpoint
- u. RC High Temperature
- v. RC High Pressure
- w. RC Pressure – Temperature
- x. Containment High Pressure
- y. Flux - Δ Flux – Flow
- z. Two intermediate range neutron flux channels

(References provided)

- A. s, t, u, & w **Only**
- B. s, v, x, & z **Only**
- C. t, u, y, & z **Only**
- D. v, w, x, & y **Only**

Answer: B

Explanation/Justification: SRO only since it requires the application of tech spec required actions during plant startup before the CRD trip breakers can be closed. SRO only white paper item B first bullet on page 3. Similar to example on page 14.

- A. Incorrect. 2, 3, and 5 are not required.
- B. Correct. IAW TS 3.3.1 table 3.3.1-1 and TS 3.3.10
- C. Incorrect. 2, 3, and 7 are not required.
- D. Incorrect. 5, and 7 are not required

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.
K/A#	2.2.1	K/A Importance	4.4
References provided to Candidate	Tech Spec Table 3.3.1-1 & TS 3.3.10		Exam Level
Question Source:	New	Technical References:	SRO
Question Cognitive Level:	High - Application	Level Of Difficulty: (1-5)	3
Objective:		10 CFR Part 55 Content:	10 CFR: 55.43(b)(2)

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96. The RCS is being drained to remove the Reactor Vessel Head.

Which of the following procedures establishes the administrative controls, for the BWST outlet valves, to prevent accidental flooding of the Refueling Canal when the Reactor Head is removed?

- A. DB-OP-06002, RCS Draining and Nitrogen Blanketing.
- B. DB-OP-06012, Decay Heat and Low Pressure Injection System Operating Procedure.
- C. DB-OP-06023, Fill, Drain, and Purification of the Refueling Canal.
- D. DB-OP-06904, Shutdown Operations.

Answer: D

Explanation/Justification: SRO only white paper item G Page 9 third bullet. Similar to question example from page 12

- A. Incorrect. Plausible since this procedure establishes the system lineup to actually drain the RCS but the procedure does not contain the administrative requirements to prevent accidental flooding of the Refueling Canal.
- B. Incorrect. Plausible since the BWST outlet valves are part of the decay heat removal system but the procedure does not contain the administrative requirements to prevent accidental flooding of the Refueling Canal.
- C. Incorrect. Plausible since this procedure addresses the actions to control the refueling canal water level but the procedure does not contain the administrative requirements to prevent accidental flooding of the Refueling Canal.
- D. Correct answer per DB-OP-06904, Shutdown Operations Step 3.18. SRO ONLY since this requires knowledge of administrative prerequisites for Reactor Vessel disassembly.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.
K/A#	2.2.18	K/A Importance	3.9
References provided to Candidate	None	Exam Level	SRO
Question Source:	New	Technical References:	DB-OP-06904 R42, Shutdown Operations, step 3.18.
Question Cognitive Level:	Low - Memory	Level Of Difficulty: (1-5)	3.5
Objective:		10 CFR Part 55 Content:	10 CFR: 55.43(b)(7)

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97. The Miscellaneous Waste Monitor Tank (MWMT) has been prepared for batch discharge.

The following radiation monitors and flow elements are out of service and INOPERABLE.

- Miscellaneous RE 1878A
- Miscellaneous RE 1878B
- Clean RE 1770B
- FE 4687 Storm Sewer Flow

All other instrumentation is OPERABLE.

Based on these conditions, what Offsite Dose Calculation Manual (ODCM) actions will be **required** in order to discharge this tank?

(References provided)

- A. The system/process flow rate is estimated at least once per 4 hours during the actual release.
- B. At least two independent samples of the tank's content are analyzed and at least two independent verifications of the release rate calculations and discharge valve lineups are performed AND the system/process flow rate is estimated at least once per 4 hours during the actual release.
- C. Grab samples are collected, at least once per 12 hours, and analyzed, at least once per 12 hours, for gross radioactivity (beta or gamma) at a lower limit of detection no greater than 1.0^{-07} $\mu\text{Ci/ml}$ or a gamma isotopic analysis meeting the LLD Requirement of Table 2-3.
- D. At least two independent samples of the tank's content are analyzed and at least two independent verifications of the release rate calculations and discharge valve lineups are performed.

Answer: D

Explanation/Justification: SRO only white paper item A Page 3 – ODCM is listed in TS Section 5.5 and Page 3 item B 4th bullet.

- A. Incorrect. Plausible if the candidate believes the tank being discharged will pass thru the storm sewer FE and that having Clean RE 1770A operable meets the one RM channel operable requirement.
- B. Incorrect. Storm sewer FE is not required for this discharge flowpath. Independent actions are correct.
- C. Incorrect. These are the correct compensatory actions for the liquid waste flow indicator being out of service.
- D. Correct. IAW ODCM Rev. 26 Table 2-1 pages 19 and 20. SRO ONLY since it requires the SRO to have knowledge of the SRO responsibilities for approving liquid waste releases.

Sys #	System	Category	KA Statement	
N/A	N/A	Generic	Ability to approve release permits.	
K/A#	2.3.6	K/A Importance	3.8	Exam Level
References provided to Candidate		ODCM Rev. 27 Table 2-1 and 2-2 pages 19 thru 22		Technical References:
Question Source:		New	Level Of Difficulty: (1-5)	3
Question Cognitive Level:		High - Application	10 CFR Part 55 Content:	10 CFR: 55.43(b)(1 or 2)
Objective:				

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98. The plant is in MODE 6 with core off-load in progress.
- A spent fuel assembly is dropped in the Spent Fuel pool and gases are observed escaping from the assembly.

The following alarms are now present in the control room:

- (9-1 -G) FIRE OR RADIATION TRBL
- (9-3-A) UNIT VENT RAD HI
- High alarm on RE5403A, FUEL HDLG EXH SYS, PARTICULATE
- High alarm on RE8446, FUEL HDLG EXH SYS, Channel 1
- High alarm on RE8447, FUEL HDLG EXH SYS, Channel 2

The following radiation monitor indications are present:

- Fuel Handling Area (RE 8417 and 8418) 1100 mr/hr and stable
- Control Room Area 10 mr/hr and stable
- Station vent Channel 1 Noble Gas (RE 4598) 3.2 $\mu\text{Ci/cc}$
- Spent Fuel Area (RE 8426 and 8427) 1200 mr/hr and stable

The plant technical staff has confirmed that these indications WILL continue for the next 45 minutes before any of these indications will begin to decrease.

Based on these indications, what is the **HIGHEST** Emergency Classification?

(References provided)

- A. Unusual Event
- B. Alert
- C. Site Area Emergency
- D. General Emergency

Answer: C

Explanation/Justification: SRO only white paper item F Page 9 first bullet. plus unique to SRO position at Davis-Besse from Page 10

- A. Incorrect. RU1 has been exceeded. However, this is not the highest classification.
- B. Incorrect. RA1 and RA2 have both been exceeded. However, this is not the highest classification.
- C. Correct. IAW RA-EP-01500, Emergency Classification Rev. 14 Tab RS1 item 1 on page 27. SRO ONLY since requires the candidate to analyze and interpret fixed radiation monitor readings to select the appropriate EAL. At Davis Besse this is an SRO ONLY task for the on-shift ERO.
- D. Incorrect. RG1 has not been exceeded.

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
References provided to Candidate	RA-EP-01500, Emergency Classification Rev. 14	Exam Level	SRO
Question Source:	New	Technical References:	RA-EP-01500, Emergency Classification Rev. 14 Tab RS1 item 1 on page 27.
Question Cognitive Level:	High - Application	Level Of Difficulty: (1-5)	3
Objective:		10 CFR Part 55 Content:	10 CFR: 55.43(b)(4 or 6)

(SRO ONLY)
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99. The plant is operating at 30% power with all systems in normal alignment for this power level.
- The current wind direction is out of the South.
 - The outside operator contacts the control room and reports a serious fire in DIESEL GEN 2 ROOM that has spread to the upper level.
 - The crew has entered DB-OP-02529, Fire Procedure.

Based on these conditions, what procedure and actions will be **required** to mitigate these conditions?

- A. Transition to DB-OP-02501, Serious Station Fire and Trip the Reactor, Initiate AFW flow AND isolation of BOTH SGs.
- B. Transition to DB-OP-02519, Serious Control Room Fire and Trip the Reactor.
- C. Transition to DB-OP-02000, "RPS, SFAS, SFRCS TRIP, OR SG TUBE RUPTURE and Trip the Reactor.
- D. Continue in DB-OP-02529, Fire Procedure and perform a rapid shutdown to Low Level Limits, then Trip the Reactor.

Answer: A

Explanation/Justification: SRO only white paper item E Page 7 third bullet. Matches question example from page 12

- A. Correct. IAW DB-OP-02501 Revision 17 Att. 1 page 7 of 8 and Att. 9 step 2.1. SRO ONLY since it requires the candidate to know the hierarchy of procedures that will be implemented to address the situation. Also the actions contained in the procedure are NOT immediate actions which would be RO knowledge.
- B. Incorrect. Right initial action, wrong procedure transition. This procedure transition would only be required if the smoke/toxic flames from the EDG room fire were to threaten the habitability of the CR. Since the wind direction is from the south, the smoke/toxic flames are being blown away from the CR.
- C. Incorrect. Right action, wrong procedure transition. Transitioning to DB-OP-02000 is in most cases the right transition anytime a reactor trip occurs. However, in the case of a serious fire, the governing procedure is DB-OP-02501 and the SRO must transition to this procedure. This is an exception to the normal rule that DB-OP-02000 is a higher priority than Abnormal Operating procedures.
- D. Incorrect. Wrong action, correct procedure transition. Since power is only at 30%, it may seem prudent to perform a rapid shutdown to the low level limits before tripping the reactor.

Sys #	System	Category	KA Statement
N/A	N/A	Generic	Knowledge of "fire in the plant" procedures.
K/A#	2.4.27	K/A Importance	3.9
References provided to Candidate		None	Exam Level
			Technical References:
			SRO
			DB-OP-02501 Revision 18 Att. 1 page 7 of 8 and
			Att. 9 step 2.1
			DB-OP-01003, Operations Procedure Use
			Instructions step 6.5.2.a
Question Source:		New	Level Of Difficulty: (1-5)
			3
Question Cognitive Level:		High - Comprehension	10 CFR Part 55 Content:
			10 CFR: 55.43(b)(5)
Objective:			

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100. A Large Break LOCA occurred coincident with some fuel damage.
- C1 bus is energized; D1 bus is de-energized and cannot be energized
 - The Shift Manager/Emergency Director has declared a General Emergency
 - Station Isolation has been declared IAW RA-EP-02245
 - One train of HPI, LPI and AFW are all operating
 - An unisolable gaseous release is in progress, from a failed containment penetration
 - The expected duration of the leakage is ~ 2 hours
 - Wind direction is from 18°
 - Dose projections at 5 miles are 0.5 rem and 1.5 rem CDE thyroid

Based on these conditions, what Protective Action Recommendation (PAR) is **required**?

(References provided)

- A. Shelter 2 mile radius & 10 mile downwind subareas 1, 2, 4, & 5
Evacuate 2 mile radius & 10 mile downwind subarea 12
- B. Shelter 2 mile radius & 5 mile downwind subareas 1, & 2
Evacuate 2 mile radius & 5 mile downwind subarea 12
- C. Evacuate 2 mile radius & 5 mile downwind subareas 1, 2, 4, 5, & 12
- D. Evacuate 2 mile radius & 5 mile downwind subareas 1, 2, & 12

Answer: B

Explanation/Justification: SRO only white paper Item D 2nd bullet Pages 6 plus unique to SRO position at Davis-Besse from Page 10.

- A. Incorrect. Plausible if the candidate mis-applies RA-EP-02245 Rev. 5 Attachment 1 by answering yes to station isolation and yes to the dose projections greater than 1 rem. .
- B. Correct IAW RA-EP-02245 Rev. 5 Attachment 1. SRO only in that it requires the implementation of administrative procedures that specify implementing emergency procedures. Specifically the offsite PAR which at Davis Besse is an SRO task for the on-shift ERO.
- C. Incorrect. Plausible if the candidate mis-applies RA-EP-02245 Rev. 5 Attachment 1 by using column B instead of column A.
- D. Incorrect. Plausible if the candidate mis-applies RA-EP-02245 Rev. 5 Attachment 1 by answering no to station isolation.

Sys #	System	Category	KA Statement		
N/A	N/A	Generic	Knowledge of emergency plan protective action recommendations.		
K/A#	2.4.44	K/A Importance	4.4	Exam Level	SRO
References provided to Candidate		RA-EP-02245 Rev. 5	Technical References:	RA-EP-02245 Rev. 5 Attachment 1	
Question Source:		New	Level Of Difficulty: (1-5)		3
Question Cognitive Level:		High Application	10 CFR Part 55 Content:		10 CFR: 55.43(b)(4)
Objective:					

2013 Davis-Besse Written Examination Answer Key

- Questions 1 through 75 are RO level questions.
- Questions 76 through 100 are SRO level questions.

<u>Q#</u>	<u>Answer</u>	<u>Q#</u>	<u>Answer</u>	<u>Q#</u>	<u>Answer</u>	<u>Q#</u>	<u>Answer</u>
1	B	26	A	51	A	76(1)	D
2	B	27	C	52	C	77(2)	A
3	D	28	C	53	C	78(3)	C
4	C	29	D	54	D	79(4)	A
5	D	30	B	55	A	80(5)	D
6	B	31	C	56	D	81(6)	A
7	D	32	B	57	C	82(7)	B
8	B	33	C	58	B	83(8)	B
9	D	34	B	59	C	84(9)	A
10	C	35	B	60	B	85(10)	C
11	A	36	A	61	D	86(11)	A
12	D	37	D	62	A	87(12)	B
13	A	38	D	63	D`	88(13)	B
14	D	39	C	64	A	89(14)	D
15	D	40	A	65	D	90(15)	D
16	D	41	A	66	D	91(16)	B
17	B	42	C	67	A	92(17)	C
18	A	43	A	68	C	93(18)	B
19	A	44	D	69	C	94(19)	D
20	A	45	B	70	C	95(20)	B
21	D	46	D	71	A	96(21)	D
22	A	47	C	72	A	97(22)	D
23	D	48	D	73	C	98(23)	C
24	B	49	C	74	D	99(24)	A
25	C	50	D	75	C	100(25)	B

REFERENCE INDEX

TAB 1	T.S. Table 3.3.1-1	(2 pages)
TAB 2	T.S. 3.3.9	(2 pages)
TAB 3	T.S. 3.3.10	(2 pages)
TAB 4	ODCM Tables 2-1 and 2-2	(4 pages)
TAB 5	RA-EP-01500	(39 pages)
TAB 6	RA-EP-02245	(14 pages)