Written Examination
SONGS
Senior Reactor Operator
Answer Key

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

GENERIC FUNDAMENTALS EXAMINATION
EQUATIONS AND CONVERSIONS HANDOUT SHEET

EQUATIONS

$\dot{Q} = \dot{m}c_p\Delta T$

$\dot{Q} = \dot{m}\Delta h$

$\dot{Q} = UA\Delta T$

$\dot{Q} \propto \dot{m}^3_{Nat\ Circ}$

$\Delta T \propto \dot{m}^2_{Nat\ Circ}$

$K_{eff} = 1/(1 - \rho)$

$\rho = (K_{eff} - 1)/K_{eff}$

$SUR = 26.06/\tau$

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\ \rho}$$

$$\rho = \frac{\ell^*}{\tau} + \frac{\bar{\beta}_{eff}}{1 + \lambda_{eff}\tau}$$

$\ell^* = 1 \times 10^{-4}$ sec

$\lambda_{eff} = 0.1\ sec^{-1}$ (for small positive ρ)

$DRW \propto \varphi^2_{tip}/\varphi^2_{avg}$

$P = P_o 10^{SUR(t)}$

$P = P_o e^{(t/\tau)}$

$A = A_o e^{-\lambda t}$

$CR_{S/D} = S/(1 - K_{eff})$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$1/M = CR_1/CR_X$

$A = \pi r^2$

$F = PA$

$\dot{m} = \rho A\vec{v}$

$\dot{W}_{Pump} = \dot{m}\Delta P\upsilon$

$E = IR$

Thermal Efficiency = Net Work Out/Energy In

$$\frac{g(z_2 - z_1)}{g_c} + \frac{(\vec{v}_2^{\,2} - \vec{v}_1^{\,2})}{2g_c} + \upsilon(P_2 - P_1) + (u_2 - u_1) + (q - w) = 0$$

$g_c = 32.2$ lbm-ft/lbf-sec^2

Steam Tables / Mollier Diagram

CONVERSIONS

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1 Mw	$= 3.41 \times 10^6$ Btu/hr	1 Curie	$= 3.7 \times 10^{10}$ dps
1 hp	$= 2.54 \times 10^3$ Btu/hr	1 kg	$= 2.21$ lbm
1 Btu	$= 778$ ft-lbf	1 gal_{water}	$= 8.35$ lbm
°C	$= (5/9)(°F - 32)$	1 ft^3_{water}	$= 7.48$ gal
°F	$= (9/5)(°C) + 32$		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 A2.03	
	Importance Rating	2.7	

Reactor Coolant Pump System: Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems

Proposed Question: Common 1

Given the following conditions:

- Unit 2 was operating at 100% power when Annunciator 56C04 – RCP P001 OC, alarmed and was followed by an automatic Reactor Trip.
- Following the Reactor Trip, a Small Break Loss of Coolant Accident occurred.
- Reactor Coolant System (RCS) pressure lowered to 1500 PSIA and stabilized.
- RCS T_{COLD} stabilized at 545ºF.

Which of the following identifies the expected Reactor Coolant Pump (RCP) configuration upon entry into SO23-12-3, Loss of Coolant Accident?

RCP P-001…

A. is secured. RCPs P-002, P-003, and P-004 are running.

B. and RCP P-002 are secured. RCPs P-003 and P-004 are running.

C. and RCP P-003 are secured. RCPs P-002 and P-004 are running.

D. and RCP P-004 are secured. RCPs P-002 and P-003 are running.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that three Reactor Coolant Pumps should remain running since a Small Break LOCA is in progress.
B. Incorrect. Plausible because securing RCP P-002 will retain Pressurizer Spray flow and ensures one RCP is secured in each loop, however, RCPs P-001 and P-002 are not diametrically opposed and this condition is known to cause thermal hydraulic issues with this style of Combustion Engineering plant.
C. Incorrect. Plausible because two RCPs are secured, however, RCPs P-002 and P-004 are in the same loop and securing RCPs P-003 will cause a loss of normal Pressurizer Spray flow.
D. Correct. Prior to exiting SO23-12-1, Standard Post Trip Actions, two Reactor Coolant Pumps are secured per the guidance in Step 11. SO23-14-1, Step 11 Bases, states the preferred alignment of Reactor Coolant Pumps is to secure those that are diametrically opposed to one another. In this case, RCPs 1 and 4 would be secured. Additionally, Pressurizer Spray flow is desired and maintained because the Pressurizer Spray Valves come off of RCS Loops 1 & 3.

Technical Reference(s): SO23-12-1, Step 11; SO23-14-1, Step 11 Bases; SO23-5-1.8.1, Attachment 6; SD-SO23-360, Figure I-1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56698 - DESCRIBE the operation of alarms associated with the Reactor Coolant System, including setpoints, possible causes, and effects on system or overall plant operation.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 3, 10
55.43 ____

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	003 K5.03	
Importance Rating	3.1	

Reactor Coolant Pump System: Knowledge of the operational implications of the following concepts as they apply to the RCPs: Effect of RCP shutdown on Tave, including the reason for unreliability of Tave in the shutdown loop

Proposed Question: Common 2

When stopping the last Reactor Coolant Pump (RCP) with Shutdown Cooling (SDC) in service, which of the following instruments is used when monitoring Reactor Coolant System (RCS) cooldown rates?

A. T-351X, SDC Combined Outlet Temperature because this will avoid exceeding cooldown rates when transitioning from RCPs to SDC.

B. T-115 / T-125, RCS Loop T_{COLD} because pump coast down will continue to provide an accurate RCS temperature as cooldown progresses.

C. Representative Core Exit Thermocouple because they provide the most accurate cooldown rate when transitioning from RCPs to SDC.

D. T-8148 / T-8149, SDC Heat Exchanger inlet temperature because adequate mixing has taken place and the cooldown rate can be monitored where turbulent flow exists.

Proposed Answer: A

Explanation:

A. Correct. Per SO23-5-1.8, Shutdown Operations (MODE 5 & 6) L&S, this is the indication to use during the transition in order to provide an accurate cooldown rate.
B. Incorrect. Plausible because this is the last operating loop T_{COLD} that is used prior to transitioning to SDC.
C. Incorrect. Plausible because the CETs may be the only valid Reactor Core temperature indication, however, this is the case on a loss of SDC flow when the Unit is in Midloop condition.
D. Incorrect. Plausible because flow is secured through the Heat Exchangers at this time and it could be thought that once flow is reintroduced an accurate indication would exist for cooldown.

Technical Reference(s): SO23-5-1.8, Attachment 13, L&S 9.7 & 9.8; SO23-5-1.8, Attachment 13, L&S 9.10; SO23-3-2-6, Attachment 16, L&S 7.5

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55072 - As the RO, CONTROL RCS Pressure / Temperature when stopping the last RCP during a Plant Cooldown per SO23-5-1.5.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	3, 5, 10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 K2.06	
	Importance Rating	2.6	

Chemical and Volume Control System: Knowledge of bus power supplies to the following: Control instrumentation

Proposed Question: Common 3

Given the following conditions:

- Unit 2 is at 100% power.
- All Charging Pumps are in AUTO.
- Charging Pump Selector Switch is in the P-192 / P-190 position.
- A loss of Non-1E Instrument Bus #1, 2Q065, has just occurred.

What is the status of the Unit 2 Charging Pumps?

A. P-190, P-191, and P-192 are STOPPED.

B. P-190, P-191, and P-192 are RUNNING.

C. P-191 is RUNNING; P-190 and P-192 are STOPPED.

D. P-190 and P-192 are RUNNING, P-191 is STOPPED.

Proposed Answer: A

Explanation:

A. Correct. Power is lost to the Pressurizer Level Controller which stops all Charging Pumps in AUTO.
B. Incorrect. Plausible because Annunciators for low Pressurizer level will alarm, however, all Charging Pumps are stopped.
C. Incorrect. Plausible because of the Charging Pump Selector Switch position, however, all Charging Pumps in AUTO will stop.
D. Incorrect. Plausible if thought that the Backup Charging Pumps would start, however, all Charging Pumps in AUTO will stop.

Technical Reference(s)	SO23-13-19, Step 2.h Note	Attached w/ Revision # See Comments / Reference
	SO23-13-19, Attachment 1	

Proposed references to be provided during examination: None

Learning Objective:	56100 - Using SO23-13-19, Loss of Non-1E Instrument Buses, DESCRIBE the basis for each Step, Caution, or Note, and the expected plant response for each step.		
Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 K4.12	
	Importance Rating	3.1	

Chemical and Volume Control System: Knowledge of the CVCS design feature(s) and/or interlock(s) which provide for the following: Minimum level of VCT

Proposed Question: Common 4

Given the following conditions:

- Unit 3 Reactor Power is 80% and stable.
- Volume Control Tank (VCT) level is 50% and stable.
- 3LT-0226, VCT Level Transmitter, has just failed instantaneously to 0%.

What is the selected signal for VCT level indicating and the status of Auto Makeup to the VCT?

VCT selected signal indicates...

A. 25%; Auto Makeup has initiated based on selected signal VCT level.

B. 25%; Auto Makeup has NOT initiated because Auto Makeup signal is generated directly from VCT Level Transmitter 3LT-0227.

C. 50%, Auto Makeup has NOT initiated based on selected signal VCT level.

D. 50%, Auto Makeup has initiated because Auto Makeup signal is generated directly from VCT Level Transmitter 3LT-0226.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that LT-0226 represents one half of the signal for VCT level per the Distributed Control System. If level had dropped to this setpoint, Auto Makeup would initiate.
B. Incorrect. Plausible if thought that LT-0226 represents one half of the signal for VCT level per the Distributed Control System. Prior to implementation of the DCS, LT-0227 and LT-0226 had specific functions for swapping Charging Pump suction and initiating Auto Makeup.
C. Correct. Level continues to indicate 50% because the Distributed Control System throws out the bad signal when the rate of change has exceeded 5% per second.
D. Incorrect. Plausible because level would continue to indicate 50%, however, this was the response prior to implementation of the CVCS Distributed Control System (DCS).

Technical Reference(s) SO23-3-2.2, Attachment 20, L&S 1.23 & 4.7 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 115385 - Given a CVCS Makeup system feature, DESCRIBE the major differences in construction and operation between the original CVCS Makeup feature and the replacement CVCS Makeup feature.

Question Source: Bank # ______

Modified Bank # ______ (Note changes or attach parent)

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6, 7

55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	005 A2.02	
Importance Rating	3.5	

Residual Heat Removal: Ability to (a) predict the impacts of the following malfunctions or operations on the RHR system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure transient protection during cold shutdown

Proposed Question: Common 5

Given the following conditions:

- Unit 2 is in MODE 5.
- Shutdown Cooling System is in service.
- The Reactor Coolant System (RCS) is in a solid plant condition.

Per SO23-3-2.6, Shutdown Cooling Operation, starting or stopping a Charging Pump or Low Pressure Safety Injection Pump under these conditions can lead to:

A. Lifting of the Pressurizer Code Safety Relief Valves.

B. Cavitation due to insufficient Net Positive Suction Head.

C. Lifting of the Low Temperature Overpressure Protection Relief Valve.

D. Vortexing in the operating Charging or Low Pressure Safety Injection Pump.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the pressure transient will occur, however, the affected component in MODE 5 is the LTOP Relief Valve.
B. Incorrect. Plausible if thought that a solid plant condition would create cavitation.
C. Correct. Per the Limits and Specifications identified in SO23-3-2.6.
D. Incorrect. Plausible because vortexing is a concern when starting or stopping the LPSI Pump, however, this occurs during a Reduced Inventory Condition.

Technical Reference(s): SO23-3-2.6, Attachment 16, L&S 4.7; SO23-3-2.6, Attachment 16, L&S 7.2 to 7.4 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 52767 - DESCRIBE the cause/effect relationships associated with the following Shutdown Cooling System conditions/operations: The effect on Shutdown Cooling system flow and temperature of starting a LPSI Pump while on SDC.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10, 14	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	005 K1.01	
Importance Rating	3.2	

Residual Heat Removal System: Knowledge of the physical connections and/or cause-effect relationships between the RHRS and the following systems: CCWS

Proposed Question: Common 6

Given the following conditions:

- Plant shutdown and cooldown from 100% power has just been completed.
- Shutdown Cooling System has been established on Train A.
- The RUNNING Train A Component Cooling Water (CCW) Pump, P-025, trips.

Which of the following describes the Shutdown Cooling System (SDC) response?

A. The SDC Pumps will trip on low CCW flow resulting in a complete loss of SDC flow.

B. SDC heat removal is lost and Annunciator 64A37 – SHUTDOWN HX TRAIN A CCW FLOW LO will alarm.

C. Standby CCW Pump, P-024, will AUTO start to maintain SDC heat removal.

D. SDC flow will be degraded and Annunciator 64A37 – SHUTDOWN HX TRAIN A CCW FLOW LO will alarm.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the SDC Pump is cooled by CCW, however, there is no trip associated with low CCW flow.
B. Correct. When CCW flow through the Shutdown Cooling Heat Exchanger drops below 5400 GPM, annunciator 64A37 – SHUTDOWN HX TRAIN A CCW FLOW LO will alarm. In this condition heat removal is lost but Shutdown Cooling flow continues.
C. Incorrect. Plausible because there are auto starts associated with CCW Pump P-024, however, they are disabled whenever the Swing CCW Pump P-025 is aligned (in this case to Train A).
D. Incorrect. Plausible because this annunciator will alarm, however, it is not flow that is degraded, rather heat removal that is lost.

Technical Reference(s) SO23-15-64.A – 64A37 & 64A56 Attached w/ Revision # See Comments / Reference
SD-SO23-400, Figure 2A & Page 11

Proposed references to be provided during examination: None

Learning Objective: 52764 - STATE the names of systems interfacing with the Shutdown Cooling System and DESCRIBE the flowpath and purpose of each interconnection.

Question Source: Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 8
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 K6.01	
	Importance Rating	3.4	

Emergency Core Cooling System: Knowledge of the effect that a loss or malfunction of the following will have on the ECCS: BIT / borated water sources

Proposed Question: Common 7

Which of the following is the reason for maintaining the Refueling Water Storage Tank 362,800 gallons above the Emergency Core Cooling System suction connection during MODES 1 through 4?

A. Shutdown the Reactor and refill the RCS in the event of an Excess Steam Demand Event.

B. Provide a 20 minute supply for ECCS injection and Containment Spray to ensure Containment pressure will decrease to less than 30 PSIG.

C. Provide a sufficient level in the Containment Emergency Sump to permit recirculation following a LOCA.

D. Shutdown the Reactor following mixing of the RWST and RCS water volumes with the most reactive CEA not inserted.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this event will contract the Reactor Coolant System and shutting down the Reactor is a concern, however, this is not the reason for the volume of water maintained.
B. Incorrect. Plausible because providing a 20 minute supply for ECCS injection and Containment Spray is correct, however, the reason is to ensure a 20 minute injection time before swap over to the Containment Emergency Sump.
C. Correct. The reason the RWST is maintained at this volume is to ensure that there is sufficient water in the Containment Emergency Sump to provide adequate net positive suction head once recirculation begins.
D. Incorrect. Plausible because this answer is correct regarding the boron concentration, however, the question asks about the RWST volume.

Technical Reference(s) SD-SO23-740, Page 22 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55823 - Given an operational condition of the Safety Injection and Containment Spray System addressed by a procedural precaution, limitation, or administrative requirement, STATE the limiting condition, the limit or the basis for that limiting condition as applicable.

Question Source: Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 G 2.4.11	
	Importance Rating	4.0	

Pressurizer Relief / Quench Tank System: Knowledge of abnormal condition procedures.

Proposed Question: Common 8

Given the following plant conditions:

- Reactor Coolant System pressure is 2000 PSIA.
- Pressurizer steam temperature is 636°F.
- A Pressurizer Relief Valve is leaking to the Quench Tank.
- Quench Tank pressure is 20 PSIG.

What is the temperature of the fluid entering the Quench Tank?

A. 212ºF.

B. 228ºF.

C. 259ºF.

D. 636ºF.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if a Constant Pressure Line of 15 PSIA is used vice 35 PSIA.

B. Incorrect. Plausible if a Constant Pressure Line of 20 PSIA is used vice 35 PSIA and the Diagram is followed incorrectly.

C. Correct. Using the Mollier Diagram portion of the Steam Tables, an isenthalpic process from 2000 PSIA to 20 PSIG will yield a fluid temperature of 259ºF.

D. Incorrect. Plausible if the concept of isenthalpic expansion through a nozzle is not part of one's thermodynamic vocabulary.

Technical Reference(s) Steam Tables / Mollier Diagram — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Steam Tables / Mollier Diagram

Learning Objective: 116234 - APPLY saturated and superheated steam tables in solving liquid-vapor problems.

Question Source:	BANK #		
	Modified Bank #	SONGS VISION	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	5, 14	
	55.43		

Comments / Reference: From Steam Tables / Mollier Diagram	Revision # N/A
Using a Mollier Diagram follow the Constant Pressure Line for 2000 PSIA down until it intersects the Saturation Line which yields an enthalpy of approximately 1135 BTU/lbm of Enthalpy. Follow this enthalpy line to where it intersects 35 PSIA (20 PSIG) and then follow this Constant Pressure Line up to where it intersects the Saturation Line and read the Constant Temperature in degrees Fahrenheit.	
Comments / Reference: From SONGS VISION	Revision # 10/12/06

Given the following plant conditions:

- Reactor Coolant System is being maintained at 2000 psia
- Pressurizer steam temperature is 636°F.
- A Pressurizer Relief Valve is leaking to the Quench Tank.
- Quench Tank is being maintained at 20 psia.

What is the temperature of the fluid entering the Quench Tank?

A. 636°F

B. 322°F

C. 228°F

D. 212°F

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	008 K4.07	
Importance Rating	2.6	

Component Cooling Water System: Knowledge of the CCWS design feature(s) and/or interlock(s) that provide for the following: Operation of the CCW swing-bus power supply and its associated breakers and controls

Proposed Question: Common 9

Given the following conditions:

- Component Cooling Water (CCW) Pumps P-024 and P-026 are operating.
- P-025, CCW Pump is aligned to Train A in STANDBY.
- Grid instabilities result in a Loss of Offsite Power and Reactor Trip.
- Emergency Diesel Generators have started normally and loaded onto both 1E Buses.
- Pressurizer level is 15% and lowering.
- Pressurizer pressure is 1580 PSIA and lowering slowly.

Assuming NO operator action, which of the following describes the status of the CCW Pumps?

CCW Pumps...

A. P-024, P-025, and P-026 are all RUNNING.

B. P-025 and P-026 are RUNNING and P-024 is OFF.

C. P-024 and P-026 are RUNNING and P-025 is OFF.

D. P-024 and P-025 are RUNNING and P-026 is OFF.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because an SIAS was generated and P-024 was initially running, however, only P-025 and P-026 CCW Pumps are powered.
B. Correct. Even though P-024 was initially running, when power is lost and an SIAS is generated, whichever Train the Swing Pump (P-025) is aligned is the pump that will start.
C. Incorrect. Plausible because the SIAS would start the Train B CCW Pump P-026, however, the Swing Pump is Kirk Key interlocked to start.
D. Incorrect. Plausible if one failed to recognize that an SIAS signal had been generated and assumed that P-025 started on low pressure (and interlocked remove some time ago).

Technical Reference(s) SD-SO23-400, Pages 8, 11, & 32 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56251 - DESCRIBE the operation of the following Component Cooling Water System components, including function, location, and specific features which include type, capacity, power supplies, and normal operating parameters where applicable: CCW Pumps - P024, P025, P026.

Question Source: Bank # SONGS VISION
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 8
55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010 K6.01	
	Importance Rating	2.7	

Pressurizer Pressure Control System: Knowledge of the effect that a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems

Proposed Question: Common 10

Given the following conditions:

- Unit 2 Reactor power is 100% and stable.
- HS-100A, Pressurizer Pressure Control Selector Switch, is positioned to Channel X.
- PT-0100X, Pressurizer Pressure Transmitter, just failed low.

With NO operator action, how will PIC-0100, Pressurizer Pressure Controller, and the Proportional Heaters respond to this failure?

	PIC-0100 Output	Proportional Heater Power
A.	100%	100%
B.	100%	0%
C.	0%	100%
D.	0%	0%

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the Proportional Heaters will be generating an output equal to 100%, however, PIC-0100 output is 0%.
B. Incorrect. Plausible if thought that PIC-0100 was at full output and therefore, the Proportional Heaters would be off.
C. Correct. With a pressure deviation of greater than -25 PSIA, the Proportional Heaters will be generating an output equal to 100%.
D. Incorrect. Plausible because PIC-0100 output is 0%, however, this will generate a proportional heater output equal to 100%.

Technical Reference(s)	SO23-13-27, Page 17	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56417 - DESCRIBE the operation of Pressurizer Pressure Control System components, instrumentation, controls and alarms including function, location, interlocks, capacity and power supplies where applicable.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 K6.09	
	Importance Rating	3.6	

Reactor Protection System: Knowledge of the effect that a loss or malfunction of the following will have on the RPS: CEAC

Proposed Question: Common 11

Given the following conditions:

- The Unit is at 100% power.
- Control Element Assembly Calculator (CEAC) #2 is failed.
- Core Protection Calculator (CPC) Point ID 062 (CEANOP) has been set to "2" on all CPC Channels.

If CEAC #1 fails and NO additional operator action is taken, how will the CPCs respond?

A. All 4 channels will trip after a 20 second time delay.

B. All 4 channels will trip after a 90 minute time delay.

C. No channel will trip since having the CEANOP flag set at a value of "2" indicates both CEACs are inoperable.

D. No channel will trip since CPCs will use the last available valid set of CEA position information in the determination of any penalty factors.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because some penalty factors are induced after 20 seconds, however, not for the conditions listed in the Stem.
B. Correct. Given the conditions listed and no operator action, a trip will occur after a 90 minute time delay.
C. Incorrect. Plausible because it could be thought that this condition is correct, however, the CEANOP flag does not function this way.
D. Incorrect. Plausible because a snapshot of a CEA position is stored within the Core Protection Calculator, however, this is a misinterpretation of how the CPC works since a penalty factor is generated.

Technical Reference(s) SO23-3-2.13, Attachment 8, L&S 1.7 & 1.10 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54689 - DESCRIBE normal, abnormal and emergency system indications and operations of CPCs and CEACs at power, and during normal surveillance activity.

Question Source: Bank # SONGS VISION
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 6, 7
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 K3.01	
	Importance Rating	3.9	

Reactor Protection System: Knowledge of the effect that a loss or malfunction of the RPS will have on the following: CRDS

Proposed Question: Common 12

Given the following conditions:

- Unit 2 is at 50% power.
- Reactor Trip Path Solid State Relay K-1 (SSR K-1) has failed and the relay has actuated.
- All Vital 120 VAC Instrument Buses remain energized.

Which of the following identifies the effect of the relay failure on the Control Element Drive Mechanism Control System (CEDMCS)?

A. Two (2) Reactor Trip Circuit Breakers will open and one (1) CEDMCS Bus will deenergize.

B. Two (2) Reactor Trip Circuit Breakers will open and both CEDMCS Buses remain energized.

C. Four (4) Reactor Trip Circuit Breakers will open and one (1) CEDMCS Bus remains energized.

D. Four (4) Reactor Trip Circuit Breakers will open and both CEDMCS Buses remain energized.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because two Reactor Trip Circuit Breakers will open, however, the CEDMCS Buses remain energized

B. Correct. Each Solid State Relay controls two Reactor Trip Circuit Breakers; the CEDMCS Buses remain energized because power is available from the other Motor Generator Set.

C. Incorrect. Plausible if thought that one Solid State Relay controlled four Reactor Trip Circuit Breakers.

D. Incorrect. Plausible because both CEDMCS Buses remain energized, however, only two of eight Reactor Trip Circuit Breakers will open.

Technical Reference(s) SD-SO23-710, Figure 2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 81786 - DESCRIBE the configuration and operational characteristics of Control Element Drive Mechanism Control System (CEDMCS) components.

Question Source: Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6
55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	013 G 2.1.28	
Importance Rating	4.1	

Engineered Safety Features Actuation System: Conduct of Operations: Knowledge of the purpose and function of major system components and controls

Proposed Question: Common 13

Given the following conditions:

- Unit 2 is at 100% power.
- All Swing Pumps are aligned to Train A.
- P-025, Component Cooling Water (CCW) Pump is running on Train A.
- CCW Pumps P-024 and P-026 are in STANDBY.
- An Operating Basis Earthquake (OBE) occurred resulting in a Reactor Trip and CCW Pump P-025 overcurrent trip.
- During aftershocks, a Small Break Loss of Coolant Accident occurred.
- Safety Injection Actuation Signal (SIAS), Containment Cooling Actuation Signal (CCAS), Control Room Isolation Signal (CRIS), and Emergency Feedwater Actuation Signal (EFAS) have automatically initiated.

With NO operator action, which of the following lists the running High Pressure Safety Injection (HPSI) Pumps and Containment Emergency Cooling Units (ECUs)?

HPSI...

A. Pump P-019 only, and Train B Containment ECUs only.

B. Pump P-019 only, and Train A and B Containment ECUs.

C. Pumps P-018 and P-019, and Train B Containment ECUs only.

D. Pumps P-018 and P-019, and Train A and B Containment ECUs.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because only the Train B Containment ECUs are running, however, both High Pressure Safety Injection Pumps will be in operation.
B. Incorrect. Plausible if thought that only P-019 would be running since P-026 is the only CCW Pump running, however, both HPSI Pumps would be running but only the Train B Containment ECUs (interlocked).
C. Correct. Only Train B Containment ECUs are running because only Train B CCW is in operation (interlocked). Both Train A and Train B HPSI Pumps are running because there is no CCW interlock (only cooling water flow to the pump via the Critical Loop).
D. Incorrect. Plausible because both HPSI Pumps will be running, however, the Containment ECUs are interlocked with Component Cooling Water flow which is only available from Train B.

Technical Reference(s): SD-SO23-400, Pages 9; SD-SO23-400, Figure 1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 81671 - ANALYZE normal and abnormal operations of the Safety Injection and Containment Spray System.

Question Source:
- Bank # ____
- Modified Bank # ____ (Note changes or attach parent)
- New X

Question History: Last NRC Exam ____

Question Cognitive Level:
- Memory or Fundamental Knowledge ____
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 7, 8
- 55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013 A4.02	
	Importance Rating	4.3	

Engineered Safety Features Actuation System: Ability to manually operate and/or monitor in the control room: Reset of ESFAS channels

Proposed Question: Common 14

Given the following conditions:

- Unit 2 tripped 30 minutes ago from 100% power due to a Loss of Condenser Vacuum.
- SO23-3-2.22, Engineering Safeguards Features Actuation System Operation, Attachment 14, EFAS / DEFAS Reset and Restoration, is being performed to reset EFAS-1 and EFAS-2.
- Steam Generator (SG) E-088 is being fed from Auxiliary Feedwater (AFW) Pump P-504 to raise SG level to 55% narrow range.
- SG E-088 narrow range level is currently 45% and controlled.

When EFAS-2 is RESET, AFW Pump P-504 will…

A. automatically stop, but will automatically restart 30 seconds later.

B. remain running and can be manually stopped from the Control Room.

C. remain running and can only be stopped from the Second Point of Control.

D. automatically stop until a subsequent EFAS-2 signal is generated or P-504 is manually started.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because initiation of a Safety Injection Actuation Signal (SIAS) or Loss of Voltage Signal (LOVS) would trip a running Motor Driven AFW Pump. If an EFAS signal were present, P-504 would automatically restart 30 seconds later.
B. Correct. As identified in SO23-3-2.22, Attachment 14, Step 2.3, after RESET or OVERRIDE, EFAS components remain in their actuated position.
C. Incorrect. Plausible because P-504 can be stopped from the Second Point of Control, however, P-504 can be stopped from the Control Room.
D. Incorrect. Plausible because Steam Generator narrow range level is greater than 27%, however, P-504 will remain running.

Technical Reference(s): SO23-3-2.22, Attachment 14, Step 2.3; SD-SO23-780, Pages 43 & 44 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 95875 - STATE the functions and design bases of the engineered safety features actuation system (ESFAS)

Question Source:
Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 7, 10
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 A3.01	
	Importance Rating	4.1	

Containment Cooling System: Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation

Proposed Question: Common 15

Which of the following Engineered Safety Features Actuation Systems signals is generated by an AUTOMATIC Safety Injection Actuation Signal (SIAS) but NOT by a MANUAL SIAS?

A. Main Steam Isolation Signal.

B. Recirculation Actuation Signal.

C. Main Feedwater Isolation Signal.

D. Containment Cooling Actuation Signal.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought there was an interface between an SIAS and a MSIS, however, these signals operate independently of each other. Some MSIS actuated Valves are closed on a CIAS (MSIVs).

B. Incorrect. Plausible if thought there was an interface between an SIAS and an RAS, however, these signals operate independently of each other.

C. Incorrect. Plausible if thought there was an interface between an SIAS and a MFIS, however, these signals operate independently of each other. Main Feedwater is isolated on an MSIS or CIAS.

D. Correct. The Containment Cooling Actuation Signal (CCAS) is only generated by an automatic SIAS. The instrumentation that AUTO initiates an SIAS also generates a CCAS. When a manual SIAS is initiated and neither PZR nor Containment pressures reach their respective AUTO setpoints (1740 PSIA and 3.4 PSIG) then the CCAS will NOT occur. The CCAS components will AUTO start if CCAS is manually actuated.

Technical Reference(s) SD-SO23-720, Pages 10, 20, &29; SD-SO23-250, Page 16 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56628 - DESCRIBE the inputs to the Plant Protection System, the purpose of each, their trip setpoints and actuation logic.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026 K1.02	
	Importance Rating	4.1	

Containment Spray System: Knowledge of the physical connections and/or cause-effect relationships between the CSS and the following systems: Cooling water

Proposed Question: Common 16

Given the following conditions:

- Unit 3 experienced an Excess Steam Demand Event (ESDE) inside Containment and a Loss of Component Cooling Water (CCW) to the Spent Fuel Pool Cooling Heat Exchanger (SFPCHX) 18 hours ago.
- Safety Injection, Containment Cooling, Control Room Isolation, Containment Isolation, Containment Spray, and Recirculation Actuation Signals have all occurred.

As part of the long term recovery actions, SFP Cooling has been re-established causing the CCW outlet temperature from CCW Heat Exchanger E-001 to rise by 5ºF.

Containment Cooling will…

A. degrade due to the rise in Saltwater Cooling temperature.

B. degrade due to the rise in Containment Spray temperature.

C. improve due to re-establishing cooling to the SFPCHX.

D. improve due to the rise in Component Cooling Water flow.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because CCW flow is now higher In CCW Heat Exchanger E-001, however, Saltwater Cooling inlet temperature is the same and outlet temperature is rising.
B. Correct. Containment Spray is being cooled by the Shutdown Cooling (SDC) Heat Exchanger. Placing SFP Cooling in service will reduce CCW flow through the SDC Heat Exchanger because part of the Non-Critical Loop must be realigned which will act to degrade Containment Cooling.
C. Incorrect. Plausible if thought that re-establishing cooling to the Spent Fuel Pool Cooling Heat Exchanger (SFPCHX) would improve flow to the Shutdown Cooling Heat Exchanger (SDCHX), however, the SFPCHX is part of the CCW Non-Critical Loop and this would result in reduced flow to the SDCHX.
D. Incorrect. Plausible if thought there was no appreciable change in CCW flow to the SDCHX when SFP cooling is placed in service.

Technical Reference(s) SO23-SD-400, Figures 1 and 3 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 81637 - EXPLAIN the interfaces between the Containment Air Handling System and other plant systems.

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5, 7, 8
55.43

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	026 A4.01	
Importance Rating	4.5	

Containment Spray System: Ability to manually operate and/or monitor in the control room: CSS controls

Proposed Question: Common 17

Given the following conditions:

- Unit 3 is operating at 100% power in MODE 1.
- A spurious Containment Spray Actuation Signal has just initiated.

Which of the following occurs as a result of the Containment Spray Actuation Signal?

The Containment Spray Pumps...

A. do NOT receive an Auto Start signal.
Containment Spray Header Isolation Valves open.

B. do NOT receive an Auto Start signal.
Containment Emergency Sump Outlet Valves open.

C. receive an Auto Start signal.
Shutdown Cooling Heat Exchanger CCW Isolation Valves open.

D. receive an Auto Start signal.
Safety Injection Pumps and Containment Spray Pumps Mini-Flow Valves open.

Proposed Answer: A

Explanation:

A. Correct. Without an SIAS present, the Containment Spray Pumps will not start. The Containment Spray Header Isolation Valves will open with a CSAS signal.
B. Incorrect. Plausible because the Containment Spray Pumps will not start, however, it is the Containment Spray Header Isolation Valves that open not the Containment Emergency Sump Outlet Valves.
C. Incorrect. Plausible because the Shutdown Cooling Heat Exchanger CCW Isolation Valves will open, however, an SIAS is required to start the Containment Spray Pumps.
D. Incorrect. Plausible if thought that CSAS started the Containment Spray Pumps. The Safety Injection and Containment Spray Mini-Flow Valves open on an SIAS.

Technical Reference(s) SO23-3-2.22, Page 80 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55522 - DESCRIBE the operation of the following Safety Injection and Containment Spray Systems component, including function, location, and specific features including type, capacity, and power supplies as applicable: Containment Spray Pumps, Containment Spray Header Control Valves.

Question Source: Bank # SONGS VISION

Modified Bank # ____ (Note changes or attach parent)

New ____

Question History: Last NRC Exam SONGS 2009

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 8

55.43 ____

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	039 A3.02	
Importance Rating	3.1	

Main and Reheat Steam System: Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS

Proposed Question: Common 18

Given the following conditions:

- Unit is in MODE 3 following a Reactor Trip from 100% power.
- A Loss of Offsite Power occurred 10 minutes ago.
- Containment pressure is 1.3 PSIG.
- Steam Generator (SG) E-088 pressure is 720 PSIA.
- Steam Generator E-089 pressure is 770 PSIA.
- Reactor Coolant System T_{COLD} is 520ºF.
- NO operator actions have been taken post-trip.

Which of the following is the correct position for the listed valves?

1. HV-8421, SG E-089 Atmospheric Dump Valve.
2. HV-8423, Steam Bypass Control System Dump Valve.

A. 1. Closed.
   2. Closed.

B. 1. Open.
   2. Closed.

C. 1. Closed.
   2. Open.

D. 1. Open.
   2. Open.

Proposed Answer: A

Explanation:

A. Correct. This is the correct configuration given current SG pressure and a Loss of Offsite Power.

B. Incorrect. Plausible if thought that the Atmospheric Dump Valves would remain open with a Main Steam Isolation Signal present on the other Steam Generator.

C. Incorrect. Plausible if thought that the conditions in the Stem did NOT impact the SBCS Valve.

D. Incorrect. Plausible if thought that both the SBCS Valve and Atmospheric Dump Valve could be open given RCS temperature.

Technical Reference(s)	SO23-3-2.22, Attachment 11	Attached w/ Revision # See Comments / Reference
	SD-SO23-175, Page 25	
	SD-SO23-720, Page 20	

Proposed references to be provided during examination: None

Learning Objective: 102465 - DESCRIBE the configuration and operational characteristics of Main Steam System components.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	7
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	059 K1.03	
Importance Rating	3.1	

Main Feedwater System: Knowledge of the physical connections and/or cause-effect relationships between the MFW and the following systems: SGs

Proposed Question: Common 19

What is the INITIAL response of the Feedwater Control System #1 to a High Level Override signal from Steam Generator E-089 during operation at 100% power?

In addition to the Main Feedwater Regulating Valve going fully closed, Main Feedwater Pump Speed will…

A. go to minimum and the Main Feedwater Bypass Valve will fully close.

B. NOT be affected and the Main Feedwater Bypass Valve will fully close.

C. go to minimum and the Main Feedwater Bypass Valve will close to 50%.

D. NOT be affected and the Main Feedwater Bypass Valve will close to 50%.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the Main Feed Bypass Valve position is correct, however, Main Feed Pump speed going to minimum is associated with a Reactor Trip Override.
B. Correct. Given the conditions in the Stem, the High Level Override (HLO) does not affect Main Feed Pump speed (it is a high select signal) but will close both the Main Feed Regulating and Main Feed Bypass Valves.
C. Incorrect. Plausible because these actions occur with a Reactor Trip Override. It could be thought that a Reactor trip occurred, however, the HLO occurs at 85% narrow range (NR) level and the RTO occurs coincident with a Reactor trip at 89% NR level.
D. Incorrect. Plausible because Main Feed Pump Speed will not be affected, however, the other actions are associated with a Reactor Trip Override. It could be thought that a Reactor trip occurred, however, the HLO occurs at 85% level and the RTO occurs at 89% level.

Technical Reference(s): SD-SO23-250, Page 65; SO23-9-6, Precaution 4.7 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54724 - DESCRIBE how the Feedwater Control System functions to control Steam Generator level.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	5, 7	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	061 K5.03	
Importance Rating	2.6	

Auxiliary/Emergency Feedwater System: Knowledge of the operational implications of the following concepts as they apply to the AFW: Pump head effects when control valve is shut

Proposed Question: Common 20

Which of the following describes the limitations on Auxiliary Feedwater Pump Discharge Valves HV-4705, HV-4706, HV-4712, and HV-4713 while feeding with the Auxiliary Feedwater System during a plant cooldown?

As Auxiliary Feedwater Pump head rises, the Auxiliary Feedwater Pump Discharge Valves should NOT be throttled...

A. to less than 26% open at any time.

B. to greater than 10% open with a low Steam Generator pressure condition.

C. to less than 10% open at any time. If an Emergency Feedwater Actuation Signal is present, the valves should NOT be throttled less than 26% open.

D. to less than 10% open at any time. If a low Steam Generator pressure condition is present, the valves should NOT be throttled less than 26% open.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the 26% open limit is correct, however, it is only when a low Steam Generator pressure condition exists.
B. Incorrect. Plausible because a low Steam Generator pressure condition is the concern, however, the valve throttle limit is incorrect.
C. Incorrect. Plausible because the valve positioning information is correct, however, it is during a low Steam Generator pressure condition vice an Emergency Feedwater Actuation Signal.
D. Correct. Per the Limitations and Specifics identified in the Auxiliary Feedwater procedure, these are the valve position limits.

Technical Reference(s) SO23-2-4, Attachment 12, L&S 2.1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55812 - Given an operational condition of the Auxiliary Feedwater System addressed by a procedural precaution, limitation, or administrative requirement, STATE the limiting condition and the basis for that limiting condition.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	5, 10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062 K3.02	
	Importance Rating	4.1	

AC Electrical Distribution System: Knowledge of the effect that a loss or malfunction of the AC distribution system will have on the following: EDG

Proposed Question: Common 21

Given the following condition:

- Unit 2 Train A Emergency Diesel Generator 2G002 was operating in parallel with 2A04 for surveillance testing when 2B24 Feeder Breaker tripped on overcurrent.

Which of the following actions is required per SO23-13-26, Loss of Power to an AC Bus?

A. Restore cooling to 2G002 by aligning Firewater cooling to 2G002.

B. Stop 2G002 by placing in MAINTENANCE LOCKOUT due to the loss of Generator cooling.

C. Restore cooling to 2G002 by installing temporary fans in the Unit 2 Diesel Generator Building.

D. Open 2G002 Output Breaker and operate 2G002 unloaded for 15 minutes to lower lube oil temperature.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because Firewater can be aligned to the Emergency Diesel Generator (EDG) when using SO23-13-2, Shutdown from Outside the Control Room, however, this action is not performed in SO23-13-26.
B. Correct. As outlined in SO23-13-26, Attachment 4, Step 2.2.1.
C. Incorrect. Plausible if thought that installing temporary fans would be sufficient to remove EDG generated heat.
D. Incorrect. Plausible because this action is performed during a normal EDG shutdown per SO23-2-13, Diesel Generator Operation, Attachment 2, however, this action is not performed in SO23-13-26.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
	SO23-13-26, Attachment 4, Step 2.2.1	
	SO23-13-2, Attachment 8, Step 3.3.2	
	SO23-13-2, Attachment 2, Step 2.9.1.5	
	SO23-3-3.23, Attachment 12, L&S 3.3	

Proposed references to be provided during examination: None

Learning Objective: 56660 - Given an operational condition of the Emergency Diesel Generator Electrical System addressed by a procedural precaution, limitation, or administrative requirement, STATE the limit, limiting condition, or basis of the limiting condition.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5, 7
55.43 ____

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	063 A3.01	
Importance Rating	2.7	

DC Electrical Distribution System: Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights

Proposed Question: Common 22

Given the following conditions of Unit 3 DC Bus voltage:

- DC Bus 3D1 is at 132 volts and -1 amp.
- DC Bus 3D2 is at 130 volts and -2 amps.
- DC Bus 3D3 is at 131 volts and -2 amps.
- DC Bus 3D4 is at 132 volts and -1 amp.

After a plant transient the following is observed:

- DC Bus 3D1 is at 129 volts and -2 amps.
- DC Bus 3D2 is at 119 volts and +138 amps.
- DC Bus 3D3 is at 131 volts and -3 amps.
- DC Bus 3D4 is at 120 volts and +115 amps.

Which of the following events has caused the change in DC Bus voltage?

A loss of...

A. Bus 3A04.

B. Bus 3A06.

C. Motor Control Center BS.

D. Motor Control Center BQ.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because Bus 3A04 powers two Battery Chargers, however, Bus 3A04 powers D1 and D3.
B. Correct. The initial conditions in the Stem signify the typical DC bus voltages and amperages. The negative indications associated with Bus amperage are reflective of the trickle charge placed on the batteries. A loss of 4160 VAC Bus 3A06 causes loss of the D2 and D4 Battery Chargers. The resultant draw on the buses is reflected in the final amperage readings.
C. Incorrect. Plausible because MCC BS powers the recently installed Train B Swing Battery Charger for Buses D2 and D4, however, this Battery Charger can only be aligned to one bus at a time.
D. Incorrect. Plausible because MCC BQ powers the recently installed Train A Swing Battery Charger, however, this Battery Charger can only be aligned to D1 or D3.

Technical Reference(s): SD-SO23-130, Figure 1; SD-SO23-120, Figure III-1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 52762 - DISCUSS the operation of the 1E 125 VDC/120 VAC System including controls, instrumentation, interlocks, capacity, power supplies and system response to component failures.

Question Source: Bank # ______
Modified Bank # SONGS VISION (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 8
55.43 ______

Comments / Reference: From SONGS VISION	Revision # 09/20/09

Given the following conditions of Unit 2 DC Bus voltage:

- DC Bus 2D1 is at 132 volts and -2 amps.
- DC Bus 2D2 is at 130 volts and -1 amp.
- DC Bus 2D3 is at 131 volts and -3 amps.
- DC Bus 2D4 is at 132 volts and -2 amps.

After a plant transient the following is observed:

- DC Bus 2D1 is at 119 volts and +165 amps.
- DC Bus 2D2 is at 129 volts and -4 amps.
- DC Bus 2D3 is at 120 volts and +131 amps.
- DC Bus 2D4 is at 131 volts and -2 amps.

Which ONE (1) of the following events has caused the change in DC Bus voltage?

A loss of...

A. Motor Control Center BQ has occurred.

B. <u>Bus 2A04 has occurred.</u>

C. Motor Control Center BS has occurred.

D. Bus 2A06 has occurred.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	064 A1.08	
Importance Rating	3.1	

Emergency Diesel Generator System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the EDG system controls including: Maintaining minimum load on EDG (to prevent reverse power)

Proposed Question: Common 23

Which of the following is the LOWEST load an Emergency Diesel Generator should carry when in parallel with another source to prevent an anti-motoring trip per SO23-2-13, Diesel Generator Operations?

A. 0.01 MWe

B. 0.05 MWe

C. 0.1 MWe

D. 0.5 MWe

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that only a minor amount would prevent reverse power trip.
B. Incorrect. Plausible if thought that only a minor amount would prevent reverse power trip.
C. Correct. As outlined in the Limits and Specifications of SO23-2-13.
D. Incorrect. Plausible if thought that ~8% of the power output of the Emergency Diesel Generator was necessary to prevent a reverse power trip.

Technical Reference(s) SO23-2-13, Attachment 23, L&S 6.11 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56660 - Given an operational condition of the Emergency Diesel Generator Electrical System addressed by a procedural precaution, limitation, or administrative requirement, STATE the limit, limiting condition, or basis of the limiting condition.

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	8, 10
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	064 K2.01	
Importance Rating	2.7	

Emergency Diesel Generator System: Knowledge of bus power supplies to the following: Air compressor

Proposed Question: Common 24

Which of the following is the power supply to Unit 2 Train B Emergency Diesel Generator 2G003 Starting Air Compressors?

Motor Control Center...

A. 2BD

B. 2BH

C. 2BDX

D. 2BHX

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because 2BD does power EDG auxiliaries, however it powersTrain A EDG components.
B. Incorrect. Plausible because 2BH does power Train B EDG auxiliaries, however the starting compressors are powered from 2BHX.
C. Incorrect. Plausible because 2BDX does power EDG starting compressors, however it only powers the Train A starting compressors.
D. Correct. 2BHX is the power supply for the Train B EDG starting compressors.

Technical Reference(s) SO23-2-13.1, pages 15, 17, 34, and 36 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 73245 - DESCRIBE the configuration and operational characteristics of Emergency Diesel Generator Mechanical Systems components.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	073 K4.01	
Importance Rating	4.0	

Process Radiation Monitoring System: Knowledge of the PRM system design feature(s) and/or interlock(s) that provide for the following: Release termination when radiation exceeds setpoint

Proposed Question: Common 25

Which of the following is the effect of a loss of power to RE-7865, Plant Vent Stack Wide Range Radiation Monitor, when it is aligned to the Containment Purge Stack?

A loss of power to RE-7865, Plant Vent Stack Wide Range Radiation Monitor would...

A. initiate a Containment Purge Isolation Signal (CPIS).

B. close the Outside Containment Purge Isolation Valves.

C. close FV-7202, Waste Gas Discharge Flow Control Valve.

D. secure Continuous Exhaust Fans A-310, A-311 and A-312.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this Radiation Monitor will isolate the Outside Containment Purge Valves, however, it does not input into the CPIS circuitry.
B. Correct. With this Radiation Monitor aligned to the Containment Purge Stack it will isolate these valves on a high radiation signal or loss of power.
C. Incorrect. Plausible because this action is performed when RE-7865 is aligned to the Primary Vent Stack.
D. Incorrect. Plausible because this action is related to the closure of the Waste Gas Discharge valve, however, RE-7865 must be aligned to the Primary Vent Stack.

Technical Reference(s): SD-SO23-690, Page 8; SO23-8-15, Attachment 8, L&S 4.6 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 52758 - DESCRIBE the operation of the components associated with the Process and Effluent Radiation Monitoring Subsystem, including function, location and specific features including type, capacity and power supplies where applicable.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	11	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 A4.04	
	Importance Rating	3.5	

Service Water System: Ability to manually operate and/or monitor in the control room: Emergency heat loads

Proposed Question: Common 26

Given the following conditions:

- Spurious Containment Isolation Actuation (CIAS) and Containment Purge Isolation (CPIS) Signals have occurred.
- Operators in the Control Room notice Control Element Drive Mechanism (CEDM) Ventilation temperatures rising.

Why would CEDM Ventilation temperatures rise after this event?

A. CPIS isolates the Nuclear Service Water supplying the CEDM Coolers.

B. CPIS isolates the Component Cooling Water Non-Critical Loop supplying the CEDM Coolers.

C. CIAS isolates the Nuclear Service Water supplying the CEDM Coolers.

D. CIAS isolates the Component Cooling Water Non-Critical Loop supplying the CEDM Coolers.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that CPIS interrupted Nuclear Service Water.
B. Incorrect. Plausible if thought that CPIS interrupted Complement Cooling Water.
C. Incorrect. Plausible because the CEDM coolers are isolated, however, the cooling water source is CCW.
D. Correct. When the Non-Critical Loop isolates, flow is lost to the CEDM Coolers.

Technical Reference(s) SD-SO23-400, Page 8 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 81637 - EXPLAIN the interfaces between the Containment Air Handling System and other plant systems.

Question Source:	Bank #		
	Modified Bank #	SONGS VISION	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7, 9	
	55.43		

Comments / Reference: From SONGS VISION	Revision # 07/21/04

A spurious Containment Isolation Actuation Signal (CIAS) and a spurious Containment Purge Isolation Signal (CPIS) have occurred. Operators in the Fuel Handling building notice pool temperatures rising. Why would Spent Fuel Pool temperatures rise after this event?

A. A CPIS would isolate the Component Cooling Water Non-Critical Loop supplying the Spent Fuel Pool Heat Exchangers.

B. A CPIS would isolate the Nuclear Service Water supplying the Spent Fuel Pool Heat Exchangers.

C. A CIAS would isolate the Component Cooling Water Non-Critical Loop supplying the Spent Fuel Pool Heat Exchangers.

D. A CIAS would isolate the Nuclear Service Water supplying the Spent Fuel Pool Heat Exchangers.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	078 K4.03	
Importance Rating	3.1	

Instrument Air System: Knowledge of the IAS system design feature(s) and/or interlock(s) that provide for the following: Securing of SAS upon loss of cooling water

Proposed Question: Common 27

Given the following conditions:

- Unit 2 tripped last shift due to a Generator fault.
- Unit 3 is in a normal 100% power alignment.
- Load Center B10 is aligned to Unit 2.
- Instrument Air Compressors are aligned as follows:
  - C001 is selected to LAG 2.
  - C002 is selected to LEAD.
  - C003 is selected to LAG 1.
- Turbine Plant Cooling Water to Instrument Air Compressors is aligned to Unit 2.

Subsequently, Reserve Auxiliary Transformer 2XR1 relays on Sudden Pressure.

If NO operator action is taken, what is the status of the Instrument Air System when plant conditions stabilize?

A. Nitrogen will maintain Instrument Air header pressure.

B. Instrument Air Compressor C002 will maintain normal system pressure.

C. Instrument Air Compressor C003 will maintain pressure per the LAG 1 setpoints.

D. Respiratory / Service Air System will maintain Instrument Air header pressure.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that loss of Turbine Plant Cooling Water affected all air compressors, however, the RSAS Compressors are air cooled.
B. Incorrect. Plausible because C002 is powered from Bus 2B07 which is powered from Bus 2A07 via 2XR2, however, loss of Turbine Plant Cooling Water prevents the compressor from operating.
C. Incorrect. Plausible because C003 is powered from Bus 3B07 which is powered from Bus 3A07 via 3XR2, however, loss of Turbine Plant Cooling Water prevents the compressor from operating
D. Correct. The RSAS Compressors are air cooled and powered from 3A03 & 3A07 which was not lost when Reserve Auxiliary Transformer 2XR1 relayed.

Technical Reference(s) SD-SO23-570, Page 9, 29, & 46 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 72725 - EXPLAIN the interfaces between the Turbine Plant Cooling Water System and other plant systems.

Question Source: Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103 A2.05	
	Importance Rating	2.9	

Containment System: Ability to (a) predict the impacts of the following malfunctions or operations on the Containment system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Emergency containment entry

Proposed Question: Common 28

Given the following conditions:

- Containment pressure is 3.2 PSIG.
- Containment average temperature is 103ºF.
- Containment nitrogen concentration is 78%.
- Containment oxygen concentration is 21%.
- An Emergency Containment Entry is being planned.

Which of the following actions is REQUIRED in order for the Emergency Containment Entry to occur?

A. Don a Self Contained Breathing Apparatus per SO23-3-2.34, Containment Access Control.

B. Reduce Containment pressure to less than 3 PSIG per SO23-1-4.2, Containment Purge and Recirculation Filtration System.

C. Raise Containment oxygen concentration to > 22% per SO23-1-4.2, Containment Purge and Recirculation Filtration System.

D. Reduce Containment temperature to < 95ºF by placing Containment Emergency Cooling Units in service per SO23-1-4.1, Containment Emergency Cooling.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that donning self-contained breathing apparatus is required at this oxygen concentration, however, Containment pressure is too high for an emergency entry to be performed.
B. Correct. With Containment pressure greater than 3 psig, Containment entry is not allowed. Placing the Containment Mini-Purge in operation is required because a direct of venting of Containment is only allowed if pressure is less than 1 psig.
C. Incorrect. Plausible because there are limits for oxygen concentration, however, 22% oxygen is not out of specification since the limit is ≤ 19.5% or > 23.5%.
D. Incorrect. Plausible because implementing Containment Emergency Cooling is required when average Containment temperature is greater than 105ºF; however, Containment pressure is too high for entry at this time.

Technical Reference(s): SO23-3-2.34, Precaution 4.2; SO23-3-2.34, Attachment 22, L&S 1.5; SO23-1-4.1, Section 6.1; SO23-1-4.2, Attachment 6

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55076 - EXPLAIN the responsibility of the Operations Department for controlling containment access and integrity including: Requirements for access to containment.

Question Source:
Bank #
Modified Bank # SONGS VISION (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 5, 10
55.43

Comments / Reference: From SONGS VISION	Revision # 09/25/08

Given the following conditions:

- Containment pressure is 3.2 psig.
- Containment average temperature is 125°F.
- Containment nitrogen concentration is 82%.
- Containment oxygen concentration is 17%.

Which ONE (1) of the following identifies the impact on performing an Emergency Containment entry and what action should be taken given the conditions listed?

A. An Emergency Containment entry <u>can</u> be performed. Reduce Containment temperature by placing Containment Emergency Cooling Units in service per SO23-1-4.1, Containment Emergency Cooling.

B. <u>An Emergency Containment entry cannot be performed. Reduce Containment pressure to less than 3 psig per SO23-1-4.2, Containment Purge and Recirculation Filtration System, Attachment 6, Operation of the Containment Mini-Purge System.</u>

C. An Emergency Containment entry <u>can</u> be performed. Don a Self Contained Breathing Apparatus per SO23-3-2-34, Containment Access Control with oxygen level less than or equal to 19.5%.

D. An Emergency Containment entry <u>cannot</u> be performed. Reduce Containment nitrogen concentration to 80% by purging one volume from Containment per SO23-3-2-34, Containment Access Control.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	002 A3.03	
Importance Rating	4.4	

Reactor Coolant System: Ability to monitor automatic operation of the RCS, including: Pressure, temperature, and flows

Proposed Question: Common 29

Given the following conditions:

- All systems are in a normal power alignment.
- Unit 2 has been operating at 100% power for 60 days.
- Ten minutes ago, LV-0110A, the in-service Letdown Flow Control Valve, failed CLOSED.

Assuming NO operator actions, what is the status of Reactor Coolant System (RCS) pressure and Letdown flow?

RCS pressure is approximately...

A. 2250 PSIA, and Letdown Flow is SECURED.

B. 2250 PSIA, and LV-0110B modulates OPEN to maintain normal Letdown flow.

C. 2275 PSIA, and Letdown Flow is SECURED.

D. 2275 PSIA, and LV-0110B modulates OPEN to maintain normal Letdown flow.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because Letdown flow is secured, however, RCS pressure stabilizes at 2275 PSIA because pressure continues to increase until the Pressurizer Spray Valves open.
B. Incorrect. Plausible if thought that the standby Letdown Flow Control Valve was available, however, RCS pressure stabilizes at 2275 PSIA.
C. Correct. When Letdown flow secures Charging continues to fill the Pressurizer. As the Pressurizer bubble gets squeezed, pressure rises and the Spray Valve starts open at 25 PSIA above the normal setpoint and holds pressure at approximately 2275 PSIA.
D. Incorrect. Plausible because the RCS pressure is correct and there is a second Letdown Flow Control Valve, however, it must be manually valved in.

Technical Reference(s): SO23-13-27, Attachment 4; SO23-5-1.3, Step 6.13.8 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56419 - DESCRIBE the cause/effect relationships associated with the Pressurizer Pressure and Level Control System and an increasing or decreasing pressurizer pressure and/or level.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	5, 7, 14	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	011 K3.01	
	Importance Rating	3.2	

Pressurizer Level Control System: Knowledge of the effect that a loss or malfunction of the PZR LCS will have on the following: CVCS

Proposed Question: Common 30

Given the following conditions:

- Unit 2 is at 100% power.
- 2LIC-0110, Pressurizer Level Controller, has Level Setpoint 1 (IN-1) as the selected input.
- The Loop 1 T_{COLD} instrument fails HIGH.

How will Letdown flow and the Charging Pumps respond to this failure?

A. Letdown flow will NOT change.
Both Standby Charging Pumps will AUTO start.

B. Letdown flow will go to MAXIMUM.
Both Standby Charging Pumps remain OFF.

C. Letdown flow will go to MINIMUM.
Both Standby Charging Pumps will AUTO start.

D. Letdown flow will NOT change.
Both Standby Charging Pumps will remain OFF.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because Letdown flow will not change, however, the standby Charging Pumps remain off.

B. Incorrect. Plausible because the standby Charging Pumps remain off, however, Letdown flow will not change.

C. Incorrect. Plausible if thought that the change in temperature would affect the Pressurizer Level Setpoint program, however, the program cuts off at 53% and 38%.

D. Correct. With the Reactor at full power, Pressurizer level is being maintained at 53%. When T_{COLD} fails high it causes T_{AVE} to rise. Pressurizer level is limited to 53% per the Pressurizer Level Program, therefore, there is no differential between actual level and setpoint. Letdown flow will not change and the standby Charging Pumps will remain off.

Technical Reference(s) SO23-13-27, Attachments 2 & 3 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56418 - DESCRIBE the operation of Pressurizer Level Control System components, instrumentation, controls and alarms including function, location, interlocks, capacity and power supplies where applicable.

Question Source: Bank # SONGS VISION
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	015 K5.17	
	Importance Rating	3.5	

Nuclear Instrumentation System: Knowledge of the operational implications of the following concepts as they apply to the NIS: DNB and DNBR definition and effects

Proposed Question: Common 31

Which of the following input failures to a Core Protection Calculator (CPC) will cause the Departure from Nucleate Boiling Ratio (DNBR) calculation to be closer to or exceed the trip setpoint for DNBR? (Consider each one separately).

A. Pressurizer pressure transmitter fails HIGH.

B. Reactor Coolant Pump (RCP) speed sensor fails HIGH.

C. Linear Safety Channel Power Range detector fails HIGH.

D. Reactor Coolant System Cold Leg temperature detector fails LOW.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because an actual rise in Pressurizer pressure would cause DNBR to lower as it affects power level, however, a pressure transmitter failing high causes the value of DNBR to rise.
B. Incorrect. Plausible because the Reactor Coolant Pump speed sensor is an input to the DNBR calculation, however, when it fails high it would cause the value of DNBR to rise.
C. Correct. The Power Range detectors input to the DNBR calculation, when it fails high, it would cause the value of DNBR to lower.
D. Incorrect. Plausible because Cold Leg temperature is an input to the DNBR calculation, however, when it fails low it would cause the value of DNBR to rise.

Technical Reference(s): SO23-3-2.13, Attachment 8, L&S 1.1 & 1.9; SO23-13-27, Step 3 Guideline; SD-SO23-710, Page 10 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54674 - DESCRIBE the inputs to the CPCs and CEACs, the purpose of each input, and the outputs available from the system: LIST the inputs to the DNBR calculation.

Question Source: Bank # SONGS VISION

Modified Bank # (Note changes or attach parent)

	New	
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	6, 14
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	016 K1.12	
Importance Rating	3.5	

Non-Nuclear Instrumentation System: Knowledge of the physical connections and/or cause-effect relationships between the NNIS and the following systems: S/G

Proposed Question: Common 32

Given the following conditions:

- Unit 2 is in MODE 3 at normal operating pressure and temperature.
- Main Steam Isolation Valves are CLOSED.
- Reactor Coolant System temperature is being controlled by operation of the Atmospheric Dump Valves in AUTO.

Which of the following statements describes the source of the control signal and the response of the Atmospheric Dump Valve to the specified instrument failure?

Each Atmospheric Dump Valve Controller receives a pressure signal from...

A. only two (2) Steam Generator Pressure Safety Channels.
High failure of a single channel will have no effect.

B. only two (2) Steam Generator Pressure Safety Channels.
Low failure of a single channel will have no effect.

C. all four (4) Steam Generator Pressure Safety Channels.
High failure of a single channel will have no effect.

D. all four (4) Steam Generator Pressure Safety Channels.
Low failure of a single channel will have no effect.

Proposed Answer: A

Explanation:

A. Correct. Two Steam Generator Pressure Safety Channels are used as inputs into the Atmospheric Dump Valve Controller. The lower of the two signals is the controlling channel.
B. Incorrect. Plausible because the signal comes from two Steam Generator Pressure Safety Channels, however, low failure of a single channel will cause the valve to close.
C. Incorrect. Plausible because high failure of a single channel will have no effect, however, only two Steam Generator Pressure Safety Channels are used.
D. Incorrect. Plausible if thought that four Steam Generator Pressure Safety Channels are used, however, low failure of a single channel will cause the valve to close.

Technical Reference(s) SD-SO23-160, Page 19 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 102466 - INTERPRET instrumentation and controls utilized in the Main Steam System.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7, 8	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	033 A1.01	
	Importance Rating	2.7	

Spent Fuel Pool Cooling System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Spent Fuel Pool Cooling System controls including: Spent fuel pool water level

Proposed Question: Common 33

Given the following conditions:

- REFUELING Operations are in progress.
- The Fuel Transfer Canal Gate Valve is OPEN.
- The Digital Level Monitoring System (DLMS) is in service.

Which of the following lists the indications for a Spent Fuel Pool (SFP) leak?

Low level alarm(s) in the...

A. Refueling Pool only; Fuel Handling Building Sump level rises.

B. Refueling Pool and Spent Fuel Pool; Fuel Handling Building Sump level rises.

C. Spent Fuel Pool only; Containment Sump and Fuel Handling Building Sump levels rise.

D. Refueling Pool and Spent Fuel Pool; Containment Sump and Fuel Handling Building Sump levels rise.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the Fuel Building Sump level increase is correct, however, both the Refueling and Spent Fuel Pools will have low level alarms.
B. Correct. These are the correct alarms and indications for a leak with the conditions listed.
C. Incorrect. Plausible if thought that both Sump levels will rise, however, alarms will occur in both the Refueling and Spent Fuel Pools.
D. Incorrect. Plausible because the low-level alarms are correct, however, the Containment Sump level will not rise.

Technical Reference(s) SD-SO23-430, Figure II-1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 81752 - ANALYZE normal and abnormal operations of the Fuel Storage and Spent Fuel Pool Cooling System.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	8, 9	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	034 K4.02	
Importance Rating	3.1	

Fuel Handling Equipment System: Knowledge of design feature(s) and/or interlock(s) that provide for the following: Fuel movement

Proposed Question: Common 34

Given the following condition:

- The Spent Fuel Handling Machine hoist is NOT at the "UP" Limit.

Which of the following describes Bridge and Trolley movement if a fuel bundle is on the hoist?

A. Unrestricted.

B. Interlocked to slow speed.

C. Interlocked to fast speed.

D. Interlocked to stop or prevent movement.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the hoist is partially protected by the mast, however, the upper limit interlock has not been met.
B. Correct. Bridge and trolley movement is limited to 1.5 inches per minute when the hoist is not at the upper limit.
C. Incorrect. Plausible if thought that this was a desired condition for rapid movement of the fuel assembly.
D. Incorrect. Plausible because the hoist is not fully withdrawn into the mast, however, the slow speed interlock allows movement at 1.5 inches per minute to allow positioning of the bridge and trolley.

Technical Reference(s) SD-SO23-430, Pages 45 & 48 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 81750 - DESCRIBE the configuration and operational characteristics of Fuel Storage and Spent Fuel Pool Cooling System components.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #	______	(Note changes or attach parent)
	New	______	
Question History:	Last NRC Exam	______	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis	______	
10 CFR Part 55 Content:	55.41	7	
	55.43	______	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	056 A2.04	
Importance Rating	2.6	

Condensate System: Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of condensate pumps

Proposed Question: Common 35

Given the following plant conditions:

- Unit 3 is at 100% power.
- Condensate Pumps P-050, P-051 and P-052 are initially running.
- Condensate Pump P-050 trips on overcurrent.

Which of the following describes the impact on the Condensate and Feedwater System and what procedural actions should be performed?

A. Condensate Pump P-053 will AUTO start. Refer to SO23-2-2, Condensate Pump Operation, and close the P-053 Discharge Vent Valve.

B. A Main Feedwater Pump will trip on low suction pressure. Refer to SO23-13-28, Rapid Power Reduction, and reduce power to 70% in five minutes.

C. A Heater Drain Pump will trip on low Heater Drain Tank level. Refer to SO23-2-3, Heater Drain Pump Operation, and verify the Mini-Flow Controller set at 3200 GPM.

D. FV-3294, Condensate Mini-Flow Valve will open. Refer to SO23-9-7, Condensate and Feedwater System Operation, and dispatch operator to locally close the valve.

Proposed Answer: A

Explanation:

A. Correct. This is the required action on a Condensate Pump AUTO start caused by trip of a running pump.
B. Incorrect. Plausible because a Main Feedwater Pump will trip if the Standby Condensate Pump had not AUTO started. This is the correct action for a Main Feedwater Pump trip from 100% power.
C. Incorrect. Plausible because this action is required when restarting a tripped Heater Drain Pump, however, Condensate Pump P-053 will AUTO start and prevent the Heater Drain Tank from being pumped down.
D. Incorrect. Plausible if thought that the Mini-Flow Valve opened because the Condensate Pump had tripped, however, this valve acts to limit Condensate header pressure to 690 PSIG.

Technical Reference(s)	SO23-2-2, Attachment 3, L&S 1.4 & 3.3	Attached w/ Revision # See Comments / Reference
	SO23-2-3, Step 6.5.3	
	SO23-2-2, Step 6.1	

Proposed references to be provided during examination: None

Learning Objective: 60787 - ANALYZE normal and abnormal operations of the Condensate and Feedwater System.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	068 K6.10	
	Importance Rating	2.5	

Liquid Radwaste System: Knowledge of the effect that a loss or malfunction of the following will have on the Liquid Radwaste System: Radiation monitors

Proposed Question: Common 36

Given the following plant conditions:

- A release of T-057, Radwaste Secondary Tank is in progress.
- RE-7813, Liquid Radwaste Effluent Line Radiation Monitor, fails and the green OPERATE LED is extinguished.

Per SO23-8-7, Liquid Radioactive Waste Release Operations, which of the following will be required for the release to recommence?

A. The Plant Superintendent must approve the Release Permit.

B. The sample flowrate must be estimated at least once every four (4) hours.

C. The liquid release cannot recommence until RE-7813 is determined to be OPERABLE.

D. Two independent samples, release rate calculations, and valve lineup verifications must be performed.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because a new release permit must be generated, however, there is no requirement for the Plant Superintendent to approve the release.
B. Incorrect. Plausible because sample flowrate must be estimated if the flow instrument is not available, however this is not a required action when RE-7813 is inoperable.
C. Incorrect. Plausible if thought that a release cannot be performed unless a Radiation Monitor was available, however, a release is allowed as long as the requirements of the ODCM are met.
D. Correct. As required per the Limitations and Specifics and the Offsite Dose Calculation Manual (ODCM).

Technical Reference(s) SO23-8-7, Attachment 17, L&S 1.20.1 Attached w/ Revision # See Comments / Reference
SO23-8-7, Attachment 17, L&S 2.3 to 2.5

Proposed references to be provided during examination: None

Learning Objective: 54021 - Given the ODCM, applicable procedures and plant conditions, DETERMINE the action requirements for inoperable components during an effluent release.

Question Source: Bank # SONGS VISION
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 13
55.43 ________

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	071 G 2.1.27	
	Importance Rating	3.9	

Waste Gas Disposal System: Conduct of Operations: Knowledge of system purpose and/or function

Proposed Question: Common 37

Given the following condition:

- Waste Gas System is aligned for normal operation with both Units in MODE 1.

Which of the following describes the Waste Gas System response to a system leak on the common suction to the Waste Gas Compressors?

A. Any Waste Gas Compressor running will be tripped if Waste Gas Surge Tank pressure reaches minus (-) 0.5 PSIG and will not restart until manually reset by an operator.

B. When pressure in the Waste Gas Surge Tank pressure drops below 5.0 PSIG, HV-7240, Nitrogen Supply Valve, will AUTO open to maintain a positive pressure in the Waste Gas Surge Tank.

C. HV-9209, Volume Control Tank Vent Valve, receives a close signal at 0 PSIG to prevent hydrogen addition into the Waste Gas Surge Tank when oxygen could be present.

D. When Waste Gas Surge Tank pressure drops below 0.5 PSIG, PCV-7200, Waste Gas Surge Tank Pressure Control Valve, bleeds the in-service Waste Gas Decay Tank back to the Waste Gas Surge Tank.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the compressors will trip and it could be thought that a manual reset is required.

B. Incorrect. Plausible if thought that the nitrogen supply would automatically maintain a positive pressure.

C. Incorrect. Plausible if thought that with the Waste Gas System depressurized that an appropriate design feature would be to isolate the major source of hydrogen gas, however, this valve only acts as a VCT vent.

D. Correct. At 0.5 psig, PCV-7200 opens to bleed some gas from the in-service Waste Gas Decay Tank to the surge header in an attempt to maintain a positive pressure.

Technical Reference(s) SD-SO23-660, Pages 11 & 17
SD-SO23-390, Page 39
Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56617 - DESCRIBE the configuration and operational characteristics of Gaseous Radwaste System components.

Question Source: Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 13
55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	072 G 2.4.31	
	Importance Rating	4.2	

Area Radiation Monitoring System: Emergency Procedures/Plan: Knowledge of annunciator alarms, indications, or response procedures

Proposed Question: Common 38

Given the following conditions:

- Unit 3 is in MODE 6.
- Irradiated Fuel movement is in progress.
- A Spent Fuel Assembly is damaged while being transported to the Spent Fuel Racks.
- Annunciator 60A23 – FHB RADIATION HI, is in alarm.
- 3RT-7850, Fuel Pool Area Radiation Monitor, HI setpoint was exceeded.
- 3RT-7822 and 3RT-7823, Fuel Handling Building Airborne Radiation Monitor, HI trip setpoint was exceeded.

Which of the following describes the resulting Fuel Handling Building (FHB) ventilation alignment?

Fuel Handling Building...

A. normal Supply and Exhaust Fans are tripped.
FHB Post Accident Cleanup Units take suction from the FHB atmosphere and discharge back to the FHB atmosphere.

B. normal Supply Fan trips and normal Exhaust Fan remains running.
FHB Post Accident Cleanup Units take suction from the FHB atmosphere and discharge back to the FHB atmosphere.

C. normal Supply Fan trips and normal Exhaust Fan remains running.
FHB Post Accident Cleanup Units take suction from the FHB atmosphere and discharge to the Continuous Exhaust Plenum.

D. normal Supply and Exhaust Fans are tripped.
FHB Post Accident Cleanup Units take suction from the FHB atmosphere and discharge to the Continuous Exhaust Plenum.

Proposed Answer: A

Explanation:

A. Correct. With a Process Radiation Monitor high alarm the normal supply and exhaust fans will trip and PACUs will take suction from the FHB atmosphere and discharge back into the FHB to lower radiation level.

B. Incorrect. Plausible because this condition would create a negative pressure inside the FHB and radiation levels do lower because the PACUs are aligned, however, this alignment would allow a radioactive release.

C. Incorrect. Plausible because this flowpath would suspend the release to atmosphere and adding air to the Fuel Handling Building could dilute the atmosphere, however, both of these fans trip. Discharge to the Continuous Exhaust Plenum is the normal alignment.

D. Incorrect. Plausible because the normal supply and exhaust fans will trip, however, the PACUs are aligned to reduce radiation level and discharge to the Continuous Exhaust Plenum is the normal alignment.

Technical Reference(s): SD-SO23-435, Figure 1; SD-SO23-435, Page 22; SO23-15.60.A1 – 60A23 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 103329 - DESCRIBE the configuration and operational characteristics of Radiation Monitoring System components.

Question Source:
- Bank #
- Modified Bank # (Note changes or attach parent)
- New X

Question History: Last NRC Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 11
- 55.43

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	007 G 2.1.7	
Importance Rating	4.4	

Reactor Trip - Stabilization - Recovery: Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation

Proposed Question: Common 39

Which of the following describes Nuclear Instrumentation response to a Reactor Trip from 100% power to the Source Range, from the time Control Element Assemblies are fully inserted?

Prompt Drop...

A. of approximately 3 decades, followed by a - 1/3 DPM Startup Rate for approximately 20 minutes.

B. to 7% power, followed by a -1/3 DPM Startup Rate for approximately 60 minutes.

C. of approximately 3 decades, followed by a -1/3 DPM Startup Rate for approximately 60 minutes.

D. to 7% power, followed by a -1/3 DPM Startup Rate for approximately 20 minutes.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the SUR portion is correct, however, NI power prompt drops to ~7%.
B. Incorrect. Plausible because the power drop is correct, however, the long-lived Delayed Neutron Fraction continues to produce neutrons for 7 ½ lives which is ~ 20 minutes.
C. Incorrect. Plausible if thought that the Delayed Neutron Fraction continued to produce neutrons out to this time frame and that power dropped into the Source Range.
D. Correct. Based on the short-lived Delayed Neutron Fraction and its effect of neutron population.

Technical Reference(s) Lesson Plan 2TA702, Page 5 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 116183 - EXPLAIN reactor response to a control rod insertion.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	1, 6	
	55.43		

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	009 EK2.03	
Importance Rating	3.0	

Small Break LOCA: Knowledge of the interrelations between small break LOCA and the following: SGs

Proposed Question: Common 40

Given the following conditions

- A Reactor Coolant System (RCS) Small Break Loss of Coolant Accident is in progress.
- Safety Injection Actuation Signal has actuated.
- All systems are operating as expected.

Which of the following is the basis for maintaining a Secondary Heat Sink?

A Secondary Heat Sink is maintained...

A. to minimize boron stratification in the RCS.

B. to provide for Containment Temperature and Pressure Control.

C. to allow Reflux boiling to occur prior to voiding in the RCS Hot Legs.

D. to remove decay heat since cooling from the injection flow alone is inadequate.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because boron stratification is a concern, however, more so during a Large Break LOCA
B. Incorrect. Plausible if thought that if Containment pressure remains below 3.4 psig then a Containment Isolation Actuation Signal would not be initiated. The issue here however, is Containment cooling which is actuated by the SIAS.
C. Incorrect. Plausible because this statement is true if talking about a Large Break LOCA.
D. Correct. Any Reactor Coolant System pressure that remains above the shutoff head of the High Pressure Safety Injection Pumps will result in a lowering of inventory without the attendant makeup. The Steam Generators provide decay heat removal until the system is cooled down and depressurized.

Technical Reference(s) SO23-14-3, Step 11 Bases Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:	54933 - Per the LOCA procedure SO23-12-3, DESCRIBE the basis for each step, caution or note and the CEN-152 basis or reason for these steps.		
Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	SONGS 2010	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	5	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015/17 AA1.03	
	Importance Rating	3.7	

RCP Malfunctions: Ability to operate and/or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Reactor trip alarms, switches, and indicators

Proposed Question: Common 41

Given the following conditions:

- SO23-13-2, Shutdown from Outside the Control Room, is in progress.
- Reactor Coolant Pump 2P-001 will NOT trip from the Control Room.
- 2P-001 START / STOP indication is NOT lit in the Control Room or at the breaker.

Which of the following action(s) is (are) required to trip Reactor Coolant Pump 2P-001 per SO23-13-2, Shutdown from Outside the Control Room?

A. Open the Reactor Coolant Pump DC Control Power Breaker.

B. Depress the manual TRIP pushbutton on the breaker and open the DC Control Power Breaker.

C. Remove the cover to the Aux Trip and Lockout 286 Relay and use the insulated cover to manually actuate the relay.

D. Remove the cover to the Aux Trip and Lockout 186 Relay and use the insulated cover to manually actuate the relay.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this action is appropriate if the RCP Breaker is already open, however, this information is not provided in the Stem.
B. Correct. Lack of position indication implies that DC Control Power is not available; therefore, this is the correct action to open the RCP Breaker.
C. Incorrect. Plausible because this is the required action if DC Control Power is available, however, it is implied in the Stem that it is not.
D. Incorrect. Plausible because this is the required action if DC Control Power is available, however, this relay is used for the 4160 V Buses.

Technical Reference(s) SO23-13-2, Attachment 21 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56665 - As an RO or SRO, STATE the immediate operator actions required by a shutdown from outside the control room, per SO23-13-2.

Question Source: Bank # SONGS VISION
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10
55.43 ____

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	022 AK3.06	
Importance Rating	3.2	

Loss of Reactor Coolant Makeup: Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: RCP thermal barrier cooling

Proposed Question: Common 42

Given the following conditions:

- A Loss of Coolant Accident has occurred on Unit 2.
- All Emergency Core Cooling System equipment is functioning as designed.
- Reactor Coolant System pressure is 1000 PSIA and STABLE.
- Highest Core Exit Thermocouple temperature is 450ºF and slowly LOWERING.
- Containment pressure is 4 PSIG and slowly RISING
- SO23-12-3, Loss of Coolant Accident, is being performed.
- All Reactor Coolant Pumps (RCPs) have been STOPPED.

Which of the following describes the basis for stopping all RCPs in this condition?

A. Prevent damage to RCPs due to the loss of cooling water.

B. Maintain adequate subcooling in the Reactor Coolant System.

C. Minimize additional inventory loss from the Reactor Coolant System.

D. Prevent damage to RCPs due to insufficient Net Positive Suction Head.

Proposed Answer: A

Explanation:

A. Correct. With Containment pressure at 4 PSIG, a Containment Isolation Actuation Signal would have actuated which isolates Component Cooling Water Non-Critical Loop flow to the RCPs.

B. Incorrect. Plausible if the Steam Tables are misread, however, there is more than adequate subcooling at this time.

C. Incorrect. Plausible because if the CIAS did not occur until SO23-12-3 was in progress it could be thought that the RCPs were stopped to minimize inventory loss.

D. Incorrect. Plausible if the Steam Tables are misread or with pressure less than 1430 PSIA it is thought that insufficient Net Positive Suction Head exists.

Technical Reference(s) SO23-12-1, Step 9.a RNO; SO23-12-3, Foldout Page — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55868 - Given an operational condition of the Component Cooling Water System, DESCRIBE the precaution, limitation or administrative requirement applicable to the situation.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	5, 7, 10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	025 AA2.02	
	Importance Rating	3.4	

Loss of RHR System: Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Leakage of reactor coolant from RHR into closed cooling water system or into reactor building atmosphere

Proposed Question: Common 43

Given the following conditions:

- Unit 2 is in MODE 4.
- Shutdown Cooling (SDC) System is in service.
- Reactor Coolant System (RCS) temperature is 280ºF.
- RCS pressure is 300 PSIA.
- Train B Component Cooling Water (CCW) Surge Tank Level has risen from 55% to 58% in 20 minutes.
- Pressurizer level is 38% and stable.
- Volume Control Tank level dropped from 58% to 54% in 20 minutes.

Which of the following components is causing the CCW Surge Tank level rise?

A. Control Element Drive Mechanism Cooler leak.

B. Spent Fuel Pool Heat Exchanger tube leak.

C. Shutdown Cooling Heat Exchanger tube leak.

D. Steam Generator Sample Cooler leak.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that the CEDM Cooler was exposed to Reactor Coolant System pressure, however, it is an air to CCW Heat Exchanger (HX) as opposed to an RCS to CCW HX.

B. Incorrect. Plausible because Spent Fuel Pool cooling is a pressurized system, however, nominal Spent Fuel Pool Cooling Pump discharge pressure is only 35 PSIG.

C. Correct. With the RCS at 300 PSIA and nominal CCW Pump discharge pressure of 120-130 PSIG; this is the source of the leakage.

D. Incorrect. Plausible because the Steam Generator Sample Cooler is a pressurized system, however, saturation pressure for 280ºF is only 50 PSIA.

Technical Reference(s)		
	SD-SO23-400, Page 10	Attached w/ Revision # See Comments / Reference
	SD-SO23-430, Page 87	
	SD-SO23-770, Page 34	

Proposed references to be provided during examination: Steam Tables

Learning Objective: 81030 - ANALYZE normal and abnormal operations of the CCW System.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	5, 7, 8	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	026 AA2.06	
	Importance Rating	2.8	

Loss of Component Cooling Water: Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The length of time after the loss of CCW flow to a component before that component may be damaged

Proposed Question: Common 44

Given the following conditions:

- Unit 2 is at 100% power.
- A complete Loss of Component Cooling Water (CCW) occurred 3 minutes ago.
- Reactor Coolant Pump (RCP) P-004 Upper and Middle Seals have failed.
- RCP P-004 Thrust Bearing temperature is 180ºF and rising at a rate of five (5) ºF per minute.

Assuming the current conditions continue, what is the EARLIEST time SO23-13-7, Loss of Component Cooling Water / Saltwater Cooling, would direct tripping the Reactor and securing the RCP?

A. Immediately.

B. 2 minutes.

C. 4 minutes.

D. 9 minutes.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because two seals have failed, however, three seals would have to fail to meet the immediate trip criteria.
B. Correct. Per the guidance in SO23-13-7, the RCP must be tripped in 5 minutes without CCW flow.
C. Incorrect. Plausible because downward thrust bearing temperature would reach 200ºF which would bring in the alarm, however, the setpoint for tripping is 225ºF.
D. Incorrect. Plausible because in 9 minutes thrust bearing temperature would reach 225ºF which is the trip criteria for RCPs P-001, P-003, and P-004 (P-002 is 240ºF).

Technical Reference(s) SO23-13-7, Step 6.c; SO23-15-56.C – 56C07 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:	55526 - As the RO, RESPOND to a Loss of Component Cooling Water or Saltwater Cooling per SO23-13-7.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	8, 10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027 AA1.03	
	Importance Rating	3.6	

Pressurizer Pressure Control System Malfunction: Ability to operate and/or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: Pressure control when on a steam bubble

Proposed Question: Common 45

Given the following conditions:

- Unit 2 is operating at 50% power during a power ascension.
- Forcing Pressurizer Spray with two Spray Valves is in progress.
- PIC-0100, Pressurizer Pressure Controller setpoint is 2225 PSIA.
- Both Spray Valves are in AUTO.
- Both Proportional Heaters and all Non-1E Backup Heaters are ON.
- Both 1E Backup Heaters are OFF and in AUTO.
- Pressurizer Pressure Control System is selected to Channel X.

Subsequently, Pressurizer Pressure Transmitter PT-0100X fails LOW.

With NO operator action, what is the status of the 1E Backup Heaters and Pressurizer Spray Valves one (1) minute later?

A. Both 1E Backup Heaters are OFF, both Pressurizer Spray Valves are OPEN.

B. Both 1E Backup Heaters are ON, both Pressurizer Spray Valves are CLOSED.

C. Train A 1E Backup Heater is ON, Train B 1E Backup Heater is OFF, both Pressurizer Spray Valves are CLOSED.

D. Both 1E Backup Heaters are OFF, Train A Pressurizer Spray Valve is CLOSED, Train B Pressurizer Spray Valve is OPEN.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because Pressurizer pressure will be rising with all Backup Heaters on, therefore, it could be determined that one minute later pressure was greater than 2250 PSIA which would deenergize the Proportional Heaters and open the Spray Valves. The control signal however, is coming from the failed pressure transmitter which is failed low (1500 PSIA).

B. Correct. Pressurizer Spray is forced by lowering the setpoint of the Pressurizer Pressure Controller from 2250 PSIA to 2225 PSIA. Non-1E Backup Heaters are then energized and when pressure rises 25 PSIA above setpoint (now 2250 PSIA) the Pressurizer Spray Valves start to open. With Channel X failed low, all Heaters will be energized and the Spray Valves will be closed because the pressure input is failed low (1500 PSIA).

C. Incorrect. Plausible because both Spray Valves are closed, however, there is no train differentiation on the 1E Backup Heaters related to the pressure transmitter.

D. Incorrect. Plausible if thought that the Backup Heaters turned off due to high pressure and that train separation existed with the Spray Valves.

Technical Reference(s): SO23-13-27, Attachments 1 & 4; SO23-3-1.10, Section 6.3 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56417 - DESCRIBE the operation of Pressurizer Pressure Control System components, instrumentation, controls and alarms including function, location, interlocks, capacity and power supplies where applicable.

Question Source:
- Bank #
- Modified Bank # (Note changes or attach parent)
- New X

Question History: Last NRC Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 5, 7, 14
- 55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	029 EK2.06	
	Importance Rating	2.9	

ATWS: Knowledge of the interrelations between ATWS and the following: Breakers, relays, and disconnects

Proposed Question: Common 46

Given the following condition:

- The Anticipated Transient Without Scram / Diversified Scram System (ATWS / DSS) was actuated on HI-HI PRESSURIZER PRESSURE.

Which of the following components OPEN as a DIRECT result of this activation?

A. Reactor Trip Breakers.

B. Motor Generator Set Supply Breakers.

C. Motor Generator Set Output Contactors.

D. Control Element Drive Mechanism Power Switch Assemblies.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that a signal separate from the shunt and UV coils is sent to the Reactor Trip Breakers since they are downstream of the MG Set Output Contactors.

B. Incorrect. Plausible because there is a misconception that the MG Set Supply Breakers trip. This is because circuit control power is supplied from the upstream (supply breaker) side of the MG Set. See Figure 3.

C. Correct. When two of four HI-HI Pressurizer Pressure signals are generated, ATWS / DSS will open the Motor Generator Set Output Contactors.

D. Incorrect. Plausible because it could be thought that this is what opens, however, the CEDMCS Power Switch (assemblies) direct power to the CEDM coils. See Figure 1A.

Technical Reference(s)		
	SD-SO23-520, Page 2	Attached w/ Revision # See Comments / Reference
	SD-SO23-520, Figure 1 & 3	
	SD-SO23-510, Figure 1A	

Proposed references to be provided during examination: None

Learning Objective: 55845 - STATE the functions of the ATWS/DSS System.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	6, 7	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	038 EA2.01	
Importance Rating	4.1	

Steam Generator Tube Rupture: Ability to determine or interpret the following as they apply to a SGTR: When to isolate one or more SGs

Proposed Question: Common 47

Which of the following parameters would be used to determine which Steam Generator is to be isolated if BOTH are diagnosed with a ruptured tube?

The Steam Generator with the...

A. highest water level.

B. lowest steam pressure.

C. highest radiation levels.

D. lowest feedwater flow.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because Steam Generator level may be higher in one of the SGs, however, when both are ruptured the one with the highest radiation level is isolated.
B. Incorrect. Plausible because Steam Generator pressure may be different in both SGs, however, when both are ruptured the one with the highest radiation level is isolated.
C. Correct. Per Step 7 of SO23-12-4 and EOI Bases SO23-14-4.
D. Incorrect. Plausible because feedwater flow experiences a decrease in proportion to the size of the rupture while at power, however, when both are ruptured the one with the highest radiation level is isolated.

Technical Reference(s): SO23-12-4, Step 7; SO23-14-4, Step 7 Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53000 - Per the SGTR procedure SO23-12-4, DESCRIBE the basis for each step, caution or note and the CEN-152 basis or reason for these steps.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	040 AK1.01	
Importance Rating	4.1	

Steam Line Rupture - Excessive Heat: Knowledge of the operational implications of the following concepts as they apply to the Steam Line Rupture: Consequences of PTS

Proposed Question: Common 48

Which of the following is a primary operational concern related to a large Excess Steam Demand Event (ESDE) per SO23-14-5, ESDE Bases and Deviations Justification?

A. Dryout of the affected Steam Generator may lead to subsequent repressurization and Pressurized Thermal Shock of the Reactor Coolant System.

B. Pressurized Thermal Shock of the Pressurizer and Surge Line may result from cold Emergency Core Cooling System water upon refill.

C. Depressurization of the Reactor Coolant System can lead to loss of subcooling and require trip of Reactor Coolant Pumps.

D. Dryout of the affected Steam Generator may result in Steam Generator Tube Rupture and radioactive release.

Proposed Answer: A

Explanation:
A. Correct. Dryout can lead to repressurization and PTS.
B. Incorrect. Plausible because the Pressurizer and Surge Line are drained due to the Reactor Coolant System cooldown, however, the primary concern is repressurization.
C. Incorrect. Plausible because Reactor Coolant Pumps may need to be tripped, however, it's not due to a loss of subcooling but rather loss of net positive suction head.
D. Incorrect. Plausible because the steam generator secondary side is dry and reintroduction of feedwater could lead to a tube rupture, however, the primary concern is repressurization.

Technical Reference(s) SO23-14-5, Step 7 Bases Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54790 - Per the ESDE procedure SO23-12-5, DESCRIBE the basis for each step, caution or note and the CEN-152 basis or reason for these steps.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	011 G 2.4.49	
	Importance Rating	4.6	

Large Break LOCA: Emergency Procedures/Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls

Proposed Question: Common 49

Given the following conditions:

- Unit 2 tripped from 100% power due to a Loss of Coolant Accident.
- Verification of Safety Injection Throttle / Stop criteria is in progress.
- The following conditions are noted:
  - Steam Generator E-088 narrow range level is 40% with Auxiliary Feedwater flow available and the Atmospheric Dump Valve is THROTTLED OPEN.
  - Steam Generator E-089 wide range level is 60% with no Auxiliary Feedwater flow available and the Atmospheric Dump Valve CLOSED.
  - Core Exit Thermocouple temperature is 460ºF.
  - Pressurizer pressure is 600 PSIA.
  - Pressurizer level is 35% and STABLE.
  - Qualified Safety Parameter Display System indicates Reactor Vessel Plenum level at 80%.

Which of the following criteria does NOT meet Safety Injection Throttle / Stop Criteria?

A. Pressurizer level.

B. Reactor Plenum level.

C. Steam Generator availability.

D. Reactor Coolant System subcooling.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that Pressurizer level is too high, however, as long as level is greater than 30% and not lowering SI Throttle / Stop Criteria is met.
B. Correct. Reactor Vessel Plenum level must exceed 100% for SI Throttle / Stop Criteria.
C. Incorrect. Plausible because one of two Steam Generators does not have feedwater available, however, only one Steam Generator is required per FS-7.
D. Incorrect. Plausible if thought that RCS subcooling must exceed 30ºF, however, RCS subcooling must only be greater than or equal to the minimum required 20ºF.

Technical Reference(s) SO23-12-11, FS-7, Steps a thru e — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Steam Tables

Learning Objective: 54933 - Per the LOCA procedure SO23-12-3, DESCRIBE the basis for each step, caution or note and the CEN-152 basis or reason for these steps.

Question Source: Bank # ____
Modified Bank # SONGS VISION (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5, 10
55.43 ____

Comments / Reference: From SONGS VISION	Revision # 10/17/10

Given the following conditions:

- Unit 2 tripped from 100% power due to a Loss of Coolant Accident.
- Verification of Safety Injection Throttle/Stop criteria is in progress.
- The following conditions are noted:
  - Steam Generator E-088 level is 63% narrow range with Auxiliary Feedwater flow available and the Atmospheric Dump Valve THROTTLED OPEN.
  - Steam Generator E-089 level is 12% narrow range with no Auxiliary Feedwater flow available and the Atmospheric Dump Valve CLOSED.
  - Core Exit Thermocouple temperature is 460ºF.
  - Pressurizer pressure is 550 psia.
  - Pressurizer level is 98% and stable.
  - Qualified Safety Parameter Display System indicates Reactor Head level at 20%.

Which of the following criteria does NOT meet HPSI Throttle / Stop Criteria?

A. Reactor Coolant System subcooling.

B. Pressurizer level.

C. Steam Generator availability.

D. Reactor Head level.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	055 EK1.02	
	Importance Rating	4.1	

Station Blackout: Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling

Proposed Question: Common 50

Based strictly on the stable temperatures listed below, which one of the following is an indication that Natural Circulation exists following a Station Blackout?

	Subcooled Margin	T_{HOT}	T_{COLD}	REP CET
A.	20ºF	550ºF	525ºF	560ºF
B.	15ºF	560ºF	500ºF	570ºF
C.	30ºF	540ºF	495ºF	558ºF
D.	40ºF	540ºF	470ºF	555ºF

Proposed Answer: A

Explanation:

A. Correct. These conditions meet the requirements for Natural Circulation.
B. Incorrect. Plausible if thought that Core Exit Saturation Margin is adequate, however, must be ≥ 20ºF.
C. Incorrect. Plausible if thought that T_{HOT} to REP CET is adequate, however, must be within 16ºF.
D. Incorrect. Plausible if thought that T_{HOT} to T_{COLD} is adequate, however, must be < 58ºF.

Technical Reference(s): SO23-12-8, Steps 10.d thru g — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Steam Tables

Learning Objective: 55268 - Per the Station Blackout procedure SO23-12-8, DESCRIBE the basis for each step, caution or note and the CEN-152 basis or reason for these steps.

Question Source: Bank # SONGS VISION
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10, 14
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	056 AK3.02	
Importance Rating	4.4	

Loss of Offsite Power: Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Actions contained in EOP for loss of offsite power

Proposed Question: Common 51

Given the following conditions:

- Unit 3 was at 100% power.
- A Loss of Offsite Power occurred 15 minutes ago.
- 1E 4 kV Buses have been reenergized from the Emergency Diesel Generators.

In accordance with SO23-12-7, Loss of Forced Circulation / Loss of Offsite Power, which of the following actions is required and what is the reason for this action?

Operate the Auxiliary Feedwater System to establish...

A. 200 GPM flow rate to each Steam Generator for at least 5 minutes to prevent collapsing the feed ring.

B. at least one intact Steam Generator narrow range level between 40% and 80% to prevent collapsing the feed ring.

C. at least one intact Steam Generator narrow range level between 40% and 80% to promote natural circulation.

D. 200 GPM flow rate to each Steam Generator for at least 5 minutes to promote natural circulation.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this action would be required if Auxiliary Feedwater flow were lost, however, based on the information in the Stem there is no indication that this occurred. If flow were lost, the value would be 130 to 150 for 5 minutes, then raise to 200 GPM.
B. Incorrect. Plausible because this is the correct range for Steam Generator level indication, however, Main Feedwater System is not available given the conditions listed.
C. Correct. This is the required action per Step 9 RNO of SO23-12-7.
D. Incorrect. Plausible if thought that Main Feedwater was available. These values apply to the Auxiliary Feedwater System.

Technical Reference(s)	SO23-12-7, Step 9 RNO SO23-12-6, Step 7	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53005 - Per the LOFC/LOOP procedure SO23-12-7, DESCRIBE the basis for each step, caution or note and the CEN-152 basis or reason for these steps.

Question Source:	Bank #		
	Modified Bank #	SONGS VISION	(Note changes or attach parent)
	New		

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10
	55.43	

Comments / Reference: From SONGS VISION	Revision # 6/21/11

Given the following conditions:

- Unit 3 was at 100% power.
- A Loss of Offsite Power occurred 15 minutes ago.
- 1E 4 kV Buses have been reenergized from the Emergency Diesel Generators.
- There are no Reactor Coolant Pumps running.
- Letdown flow has been restored.
- Pressurizer level is 68% and slowly lowering.
- Pressurizer pressure is 2320 psia and slowly lowering.
- Main Steam Isolation Valves are closed.

In accordance with SO23-12-7, Loss of Forced Circulation / Loss of Offsite Power, which ONE (1) of the following actions is required? Operate the...

A. Auxiliary Feedwater System to establish 200 gpm flow rate to each Steam Generator for at least 5 minutes.

B. Main Feedwater System to establish at least one intact Steam Generator level between 40% and 80% on the narrow range indication.

C. Auxiliary Feedwater System to establish at least one intact Steam Generator level between 40% and 80% on the narrow range indication.

D. Main Feedwater System and use Main Steam Safety Valves to steam the Steam Generators to establish natural circulation and decay heat removal.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	057 AA2.12	
Importance Rating	3.5	

Loss of Vital AC Instrument Bus: Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: PZR level controller, instrumentation, and heater indications

Proposed Question: Common 52

Given the following conditions at 100% power:

- Both backup Charging Pumps have AUTO started.
- Pressurizer Level Setpoint has remained steady.
- All Pressurizer Non-1E Heaters have deenergized.

Which of the following sets of conditions would immediately yield the associated system response?

A. HS-0100C, Non-1E PZR LO-LO Level Heater Cutout Switch is selected to Channel X when Pressurizer Level Channel Y failed low.

B. Pressurizer Level selected to Channel X and Pressurizer Level Indicating Controller (LIC-0110) selected to LS1 when Loss of Vital Bus Y01 occurred.

C. HS-0100C, Non-1E PZR LO-LO Level Heater Cutout Switch is selected to Channel Y when Pressurizer Level Channel X failed low.

D. Pressurizer Level selected to Channel Y and Pressurizer Level Indicating Controller (LIC-0110) selected to LS1 when Loss of Vital Bus Y02 occurred.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because Channel Y has failed low, however, with all Non-1E heaters deenergized, HS-0100C would be in the X & Y position.
B. Incorrect. Plausible because with a loss of Vital Bus Y01 and Pressurizer level Channel X selected, the associated response would be correct with Level Setpoint selected to LS2.
C. Incorrect. Plausible if thought that Channel X failed low and with all Non-1E heaters deenergized HS-0100C would be in the X & Y position.
D. Correct. With a loss of Vital Bus Y02 and Pressurizer level selected to Channel Y, the associated response would be correct with Level Setpoint selected to LS1.

Technical Reference(s) SO23-13-18, Attachments 1 & 2; SO23-13-27, Attachment 2 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55180 - Per the Reactor Protection System Failure procedure, SO23-13-18, DESCRIBE the basis for each major step, caution, or note and the expected plant response for each major step.

Question Source:
Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 7
55.43 ______

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	058 AK3.02	
Importance Rating	4.0	

Loss of DC Power: Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Actions contained in EOP for loss of DC power

Proposed Question: Common 53

Which Auxiliary Feedwater Pump cannot be started from the Control Room following a Loss of 125 VDC Bus D2?

A. P-141 due to loss of Train A control power.

B. P-504 due to loss of Train B control power.

C. P-141 due to loss of Train B control power.

D. P-504 due to loss of Train A control power.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because P-141 cannot be started upon loss of a DC Bus, however, it is DC Bus D1.
B. Correct. When DC Bus D2 is lost, control power to Train B equipment is lost; therefore, the Train B Auxiliary Feedwater Pump cannot be started from the Control Room.
C. Incorrect. Plausible because P-141 cannot be started upon loss of a DC Bus, however, there is sufficient power redundancy on P-140 to operate it from the Control Room.
D. Incorrect. Plausible because P-504 is correct, however, there is sufficient power redundancy on P-140 to operate it from the Control Room.

Technical Reference(s) SO23-13-18, Attachment 6 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 52579 - DESCRIBE the integrated operation of the EFAS actuated components to mitigate the severity of an accident, including consequences of misoperation or malfunction.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	7, 8, 10
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	062 G 2.4.35	
Importance Rating	3.8	

Loss of Nuclear Service Water: Emergency Procedures/Plan: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects

Proposed Question: Common 54

Given the following conditions:

- SO23-13-2, Shutdown from Outside of the Control Room, is in progress due to a fire in the Unit 2 Control Room.
- All applicable Fire Isolation Switches have been placed in LOCAL.
- The Radwaste Operator ensures that the Saltwater Cooling / Component Cooling Water (SWC / CCW) Heat Exchanger Outlet Valve is open.

Per SO23-13-2, Shutdown from Outside of the Control Room, which of the following describes the reason this action is required?

A. A hot short due to fire could have caused Heat Exchanger Outlet Valve automatic closure and the associated SWC Pump is running at shutoff head.

B. A hot short due to fire could have caused Heat Exchanger Outlet Valve closure and SWC Pump trip. Opening Heat Exchanger Outlet Valve restarts the SWC Pump.

C. The CRS will be unable to start the SWC Pump from the Second Point of Control due to the interlock with the Heat Exchanger Outlet Valve being closed.

D. The CRS will be starting the SWC Pump from the Second Point of Control and the automatic opening feature is defeated with the Fire Isolation Switch in LOCAL.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because hot shorts can result in spurious equipment operation and it could be thought that there were no operator actions to secure the SWC Pump.
B. Incorrect. Plausible because hot shorts can result in spurious equipment operation and an interlock does exist between these components but no interlocks exist with the Fire Isolation Switch (FIS) in LOCAL.
C. Incorrect. Plausible because an interlock does exist between these components but no interlocks exist with the Fire Isolation Switch in LOCAL.
D. Correct. As part of the local operator actions the Fire Isolation Switch is placed in LOCAL in order to start the SWC Pump. The automatic opening feature is defeated with the FIS in LOCAL.

Technical Reference(s)	SD-SO23-410, Page 7	Attached w/ Revision # See Comments / Reference
	SO23-13-2, Attachment 2, Section 4.0	
	SO23-13-2, Attachment 6, Step 17.3	
	SO23-13-2, Attachment 10, Section 7.0	

Proposed references to be provided during examination: None

Learning Objective: 56671 - DESCRIBE the purpose and operation of the Fire Isolation Switch for a given component.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam SONGS 2009

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content: 55.41 4, 7, 10
55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	065 AA1.04	
	Importance Rating	3.5	

Loss of Instrument Air: Ability to operate and/or monitor the following as they apply to the Loss of Instrument Air: Emergency air compressor

Proposed Question: Common 55

Given the following conditions:

- Unit 2 is at 100% power.
- Instrument Air System is aligned for normal operations.
- Subsequently, a valid Instrument Air Header Pressure Low alarm is received, due to an air leak.
- Instrument Air header pressure is currently 87 PSIG and steady.

Which of the following describes the current status of the Instrument Air System?

A. Three Instrument Air Compressors are running fully loaded.
Nitrogen Backup is maintaining Instrument Air header via a Pressure Control Valve.

B. Three Instrument Air Compressors are running fully loaded.
Service Air is maintaining Instrument Air header via a Pressure Control Valve.

C. Two Instrument Air Compressors are running fully loaded.
One Instrument Air Compressor is running half loaded.
Nitrogen Backup is maintaining Instrument Air header via a Pressure Control Valve.

D. Two Instrument Air Compressors are running fully loaded.
One Instrument Air Compressor is running half loaded.
Service Air is maintaining Instrument Air header via a Pressure Control Valve.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because three Instrument Air Compressors are running, however, they would NOT all be at full load. The Service Air Pressure Control Valve opens at 88 PSIG and would be in service.
B. Incorrect. Plausible because three Instrument Air Compressors are running, however, they would NOT all be at full load because Instrument Air Header pressure only dropped to 87 PSIG. Service Air would be supporting Instrument Air.
C. Incorrect. Plausible because three Instrument Air Compressors are running and the loading is correct, however, Nitrogen Backup would NOT be in service because Instrument Air Header pressure did not drop below 83 PSIG.
D. Correct. Instrument Air header pressure to would have had to drop to 85 PSIG for all three Instrument Air Compressors to run fully loaded. The Service Air Pressure Control Valve opens at 88 PSIG to support Instrument Air.

Technical Reference(s): SO23-13-5, Attachment 6, L&S 1.2 & 1.9; SO23-1-1, Attachment 22, L&S 2.1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 72865 / 72866

DESCRIBE the configuration and operational characteristics of Instrument and Respiratory & Service Air Systems components.

INTERPRET instrumentation and controls utilized in the Instrument and Respiratory & Service Air Systems.

Question Source:
Bank # ____
Modified Bank # SONGS VISION (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 7
55.43 ____

Comments / Reference: From SONGS VISION	Revision # 10/12/09

Given the following conditions:

- Unit 2 is operating at 100% power with the Instrument Air system aligned for normal operations.
- Subsequently, an air leak upstream of the Instrument Air Dryers occurs.
- Instrument Air header pressure lowered to 85 psig and is currently 87 psig and steady.

Which ONE (1) of the following describes the current status of the Instrument Air System?

A. **Three (3) Instrument Air Compressors are running fully loaded.**
Service Air is maintaining Instrument Air via a pressure control valve.

B. Two (2) Instrument Air Compressors are running fully loaded.
Nitrogen Backup is maintaining Instrument Air via a pressure control valve.

C. One (1) Instrument Air Compressor is running fully loaded.
Two (2) Instrument Air Compressors are running half loaded.
Nitrogen Backup and Service Air is maintaining Instrument Air via pressure control valves.

D. Two (2) Instrument Air Compressors are running fully loaded.
One (1) Instrument Air Compressor is running half loaded.
Nitrogen Backup and Service Air is maintaining Instrument Air via pressure control valves and Instrument Air to Containment has closed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	077 G 2.1.23	
	Importance Rating	4.3	

Generator Voltage and Electric Grid Disturbances: Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation

Proposed Question: Common 56

Given the following conditions:

- Emergency Diesel Generator (EDG) surveillance is in progress.
- Train B EDG 2G003 is fully loaded on Bus 2A06.
- Generation Control Center notifies SONGS that a Degraded Grid condition exists.
- Switchyard voltage is 218 kV and steady.

Operator response per SO23-13-4, Operation During Major System Disturbances, requires unloading the Emergency Diesel Generator and opening the Output Breaker.

Which of the following describes the reason for this action per SO23-13-4, Operation During Major System Disturbances?

A. Enable the Degraded Voltage Protection Circuit.

B. Restore Unit 2 to within the Limiting Condition for Operation (LCO).

C. Raise bus voltage to prevent damaging bus loads.

D. Prevent Emergency Diesel Generator trip on Generator Differential Voltage.

Proposed Answer: A

Explanation:

A. Correct. The EDG must be removed from service to enable the Degraded Voltage Protection Circuit.
B. Incorrect. Plausible because this would be correct if both EDG Units were connected to the grid.
C. Incorrect. Plausible because if thought that this action would raise bus voltage.
D. Incorrect. Plausible because it could be thought that preventing a Generator Differential Trip would preserve the Emergency Diesel Generator.

Technical Reference(s) SO23-13-4, Step 4.c RNO Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55215 - Per the Operation During Major System Disturbances procedure, SO23-13-4, DESCRIBE the basis for each step, caution or note and the expected plant response for each step.

Question Source: Bank # SONGS VISION
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 7, 8
55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	001 AA1.02	
	Importance Rating	3.6	

Continuous Rod Withdrawal: Ability to operate and/or monitor the following as they apply to the Continuous Rod Withdrawal: Rod in-hold-out switch

Proposed Question: Common 57

Given the following conditions:

- A Reactor startup is in progress and current power level is 1 x 10^{-6}%.
- A 20 second continuous Control Element Assembly (CEA) withdrawal event now occurs due to the Control Element Drive Mechanism Control System (CEDMCS) Manual Control Switch contact being inadvertently stuck in the WITHDRAW position.
- The CEA withdrawal is stopped by cycling the Manual Control Switch between WITHDRAW and INSERT.

Assuming NO other action is taken, what light indication was available to determine that CEA withdrawal was occurring and which of the following parameters will stabilize at a value SIGNIFICANTLY different from what it was before the event?

A. White UP arrow was lit; Neutron power.

B. Yellow GROUP light was lit; Core ΔT power.

C. White UP arrow was lit; Core ΔT power.

D. Yellow GROUP light was lit; Neutron power.

Proposed Answer: A

Explanation:

A. Correct. During CEA withdrawal the white UP arrow is illuminated. NI power will see the greatest change because the Point of Adding Heat does not occur until approximately 2E-1% power.
B. Incorrect. Plausible because it could be conceived that ΔT power will immediately change, however, not from this power level.
C. Incorrect. Plausible because the white UP arrow is illuminated, however, ΔT power will not change for some time.
D. Incorrect. Plausible because NI power will see the greatest change because the Point of Adding Heat does not occur until approximately 2E-1% power, however, this CEA indication is not available.

Technical Reference(s) SD-SO23-510, Figure 12 Attached w/ Revision # See Comments / Reference
SO23-5-1.3.1, Attachment 5

Proposed references to be provided during examination: None

Learning Objective: 81789 - ANALYZE normal and abnormal operations of the Control Element Drive Mechanism Control System (CEDMCS).

Question Source: Bank # SONGS VISION
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6
55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	003 AA2.01	
	Importance Rating	3.7	

Dropped Control Rod: Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod position indication to actual rod position

Proposed Question: Common 58

Which of the following indications is used to determine Dropped CEA position?

A. Pulse Counter information from the Plant Computer System and Core Mimic Display (Rod Bottom light indication).

B. Reed Switch and Pulse Counter information from both the Plant Computer System and Core Mimic Display (Rod Bottom light indication).

C. Reed Switch indication from the Secondary Rod Position CRT Display and Core Mimic Display (Rod Bottom light indication).

D. Reed Switch and Pulse Counter information from both the Secondary Rod Position CRT Display and the Plant Computer System.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because Pulse Counter is available from PCS and Reed Switch is available from Core Mimic Display, however, Pulse Counter does not update when a CEA is dropped.
B. Incorrect. Plausible because Reed Switch information is available from PCS and the Core Mimic Display is Reed Switch driven, however, Pulse Counter is only available from PCS.
C. Correct. This information is correct as shown.
D. Incorrect. Plausible because Reed Switch and Pulse Counter information are both available from PCS, however, only Reed Switch is available from Secondary Rod Position CRT Display.

Technical Reference(s) SD-SO23-510, Figures 1A, 11, & 12A; SD-SO23-510, Page 7 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 81788 - INTERPRET instrumentation and controls utilized in the Control Element Drive Mechanism Control System (CEDMCS).

Question Source: Bank # SONGS VISION
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	6
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	024 AK1.02	
Importance Rating	3.6	

Emergency Boration: Knowledge of the operational implications of the following concepts as they apply to Emergency Boration: Relationship between boron addition and reactor power

Proposed Question: Common 59

Given the following conditions:

- A Turbine load rejection has occurred.
- Control Element Assemblies (CEAs) were manually inserted in an attempt to maintain T_{COLD} on program.
- Reactor power is currently 80%.
- Annunciators 50A37 – PRE-POWER DEPENDENT INSERTION LIMIT and 50A36 – POWER DEPENDENT INSERTION LIMIT alarm windows are illuminated.
- Steam Bypass Control Valves (SBCS) are modulating CLOSED.

Which of the following actions is required per Annunciator Response Instruction 50A36 – POWER DEPENDENT INSERTION LIMIT?

A. Perform a Rapid Power Reduction per SO23-5-1.7, Power Operations, due to Power Dependent Insertion Limit violation.

B. Realign Group 6 CEAs in accordance with SO23-13-13, Misaligned or Immovable Control Element Assembly, to avoid CPC generated trip.

C. Raise Turbine load to maintain T_{COLD} on program and minimum Steam Generator pressure requirements per SO23-5-1.7, Power Operations.

D. Initiate Emergency Boration per SO23-13-11, Emergency Boration of the RCS / Inadvertent Dilution or Boration, to move the CEAs beyond the Pre-Power Dependent Insertion Limit.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the PDIL alarms are in, however, a Rapid Power Reduction is not required.
B. Incorrect. Plausible because the PDIL alarms are in, however, CEAs are not misaligned from their Group.
C. Incorrect. Plausible because raising Turbine load may cause power transient after a load rejection, and CEAs below PDIL requires Emergency Boration.
D. Correct. Required by Entry Conditions of SO23-13-11, Emergency Boration of the RCS / Inadvertent Dilution or Boration

Technical Reference(s): SO23-13-11, Entry Conditions; SO23-13-13, Step 1; SO23-15-50.A2 – 50A36 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55510 - Per the Emergency Boration of the RCS or Inadvertent Dilution/Boration procedure, SO23-13-11, DESCRIBE the basis for each step, caution, or note and the expected plant response for each step.

Question Source:
Bank # SONGS VISION
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 10
55.43 ____

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	060 AA2.06	
Importance Rating	3.6	

<u>Accidental Gaseous Radwaste Release</u>: Ability to determine and interpret the following as they apply to the Accidental Gaseous Release: Valve lineup for release of radioactive gases

Proposed Question: Common 60

Per SO23-8-14, Radwaste Gas Collection System Operation, which of the following would minimize the amount of radioactive gases released during a Waste Gas Decay Tank rupture?

A. Manual initiation of a Containment Purge Isolation Signal.

B. Only one (1) Waste Gas Decay Tank is aligned for service at a time.

C. A Nitrogen cover gas is maintained on the Waste Gas Decay Tank.

D. A positive pressure is maintained in the Radwaste Building.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because a Containment Purge Isolation Signal will secure ventilation, however, not via this flowpath (separate stack).
B. Correct. As outlined in SO23-8-14.
C. Incorrect. Plausible because nitrogen is piped to the Waste Gas Decay Tanks to ensure oxygen is less than 1%, however, only one Waste Gas Decay Tank is aligned at a time.
D. Incorrect. Plausible if thought that a positive pressure would keep gas in the Waste Gas Decay Tank.

Technical Reference(s):
SO23-8-14, Attachment 20, L&S 5.1
SO23-8-14, Attachment 17, Objective
SD-SO23-660, Figures 1A & 1C

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56617 - DESCRIBE the configuration and operational characteristics of Gaseous Radwaste System components.

Question Source:
Bank # SONGS VISION
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	13
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	061 G 2.4.6	
	Importance Rating	3.7	

Area Radiation Monitoring System Alarms: Emergency Procedures/Plan: Knowledge of EOP mitigation strategies

Proposed Question: Common 61

Given the following conditions:

- Unit 3 is at 100% power.
- Containment Normal Sump level is 30% and stable.
- Pressurizer level is 51% and lowering.
- Volume Control Tank (VCT) level is 50% and lowering.
- Letdown flow is 65 GPM.
- Charging flow is 44 GPM.
- RE-7854, Radwaste Area Radiation Monitor, is in alarm.
- RE-7808 and RE-7865, Primary Vent Stack Radiation Monitors, have rising trends.
- SO23-13-14, Reactor Coolant System Leak, is in progress.

After starting additional Charging Pumps to maintain Pressurizer level and verifying VCT level is being maintained in the normal program band, what is the next action required per SO23-13-14, Reactor Coolant System Leak?

A. Isolate Letdown.

B. Immediately trip the Reactor.

C. Verify Containment Purge is NOT in service.

D. Initiate a Rapid Power Reduction and trip the Reactor at 35% power.

Proposed Answer: A

Explanation:

A. Correct. Letdown must be isolated because a Chemical and Volume Control System leak is occurring outside Containment.

B. Incorrect. Plausible because Pressurizer and VCT levels are both lowering, however, this action would not be performed until all Charging Pumps were running and level was continuing to lower.

C. Incorrect. Plausible because this action is required for an RCS leak per SO23-13-14, however, with RE-7854 in alarm the leakage source is from outside Containment.

D. Incorrect. Plausible because this action is required for an RCS leak per SO23-13-14, however, only after it is determined that the leak is greater than 25 GPM.

Technical Reference(s)	SO23-13-14, Steps 1, 2, & 3	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54932 - Per the Reactor Coolant Leak procedure, SO23-13-14, DESCRIBE the basis for each step, caution, or note and the expected plant response for each step.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	10, 11
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	067 AK1.02	
	Importance Rating	3.1	

Plant Fire on Site: Knowledge of the operational implications of the following concepts as they apply to Plant Fire on Site: Fire fighting

Proposed Question: Common 62

Which of the following describes the response of 2(3)HV-5686, Fire Protection Water to Containment Isolation Valve, to a valid fire signal from inside Containment and how would a subsequent Containment Isolation Actuation Signal (CIAS) impact Containment firefighting efforts?

2(3)HV-5686, Fire Protection Water to Containment Isolation Valve...

A. will automatically open to initiate fire water spray flow inside Containment. CIAS will NOT close 2(3)HV-5686 when a valid fire signal is present.

B. will automatically open to initiate fire water spray flow inside Containment. CIAS will close 2(3)HV-5686, isolating fire water flow to Containment.

C. must be manually opened to initiate fire water spray flow inside Containment. CIAS will NOT close 2(3)HV-5686 when a valid fire signal is present.

D. must be manually opened to initiate fire water spray flow inside Containment. CIAS will close 2(3)HV-5686, isolating fire water flow to Containment.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the deluge valves inside Containment will open, however, this Containment Isolation Valve must be manually opened and closes upon receipt of a CIAS.
B. Incorrect. Plausible because this valve closes upon receipt of a CIAS, however, the valve must be manually opened.
C. Incorrect. Plausible because the valve must be manually opened to initiate fire water flow, however, this valve closes upon receipt of a CIAS.
D. Correct. The valve must be manually opened upon receipt of a fire signal and closes on a CIAS.

Technical Reference(s)		
	SD-SO23-590, Page 12	Attached w/ Revision # See Comments / Reference
	SO23-13-21, Attachment 2, Step 2.12	
	SO23-13-21, Attachment 2, Step 13.1	

Proposed references to be provided during examination: None

Learning Objective: 53413 - Per the Fire procedure, SO23-13-21, DESCRIBE the basis for each step, Caution, or note and the expected plant response for each step.

Question Source: Bank # ____

Modified Bank # ____ (Note changes or attach parent)

New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 9

55.43 ____

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	069 AK2.03	
	Importance Rating	2.8	

Loss of Containment Integrity: Knowledge of the interrelations between the Loss of Containment Integrity and the following: Personnel access hatch and emergency access hatch

Proposed Question: Common 63

Given the following conditions:

- Unit 3 is in MODE 6 with CORE ALTERATIONS in progress.

Which of the following conditions would NOT meet the requirements for Containment Penetrations per Technical Specifications?

A. An electrical penetration seal is REMOVED for repairs.

B. Equipment Hatch is SEALED with 4 equally spaced bolts.

C. One Containment Personnel Air Lock door is OPEN.

D. A Containment Purge is in progress.

Proposed Answer: A

Explanation:

A. Correct. If a penetration seal is removed CORE ALTERATIONS must be stopped.
B. Incorrect. Plausible if thought that four equally spaced bolts was insufficient, however, this meets the requirement of Technical Specification LCO 3.9.3.
C. Incorrect. Plausible if thought that both Containment Air Lock Doors need to be closed, however, this meets the requirement of Technical Specification LCO 3.9.3.
D. Incorrect. Plausible because there is a direct flowpath to atmosphere from Containment, however, this meets the requirement of Technical Specification LCO 3.9.3.

Technical Reference(s) Technical Specification LCO 3.9.3 Attached w/ Revision # See Comments / Reference
SO23-5-1.8, Attachment 13, L&S 8.2

Proposed references to be provided during examination: None

Learning Objective: 55076 - EXPLAIN the responsibility of the Operations Department for controlling containment access and integrity including Technical specification requirements and basis for containment integrity.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	9, 10	
	55.43		

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	076 AA2.01	
Importance Rating	2.7	

High Reactor Coolant Activity: Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Location or process point that is causing an alarm

Proposed Question: Common 64

Which of the following Radiation Monitors will actuate Annunciator 57C20 – RCS LEAKAGE DETECTION ACTIVITY HI?

A. RE-7820, Containment High Range Radiation Monitor.

B. RE-7848, Containment General Area Radiation Monitor.

C. RE-7804, Containment Airborne Radiation Monitor.

D. RE-7828, Containment Purge Stack Radiation Monitor.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this Radiation Monitor has an associated Annunciator, however, it brings in 57C10 – CONTAINMENT RADIATION HI.
B. Incorrect. Plausible because this Radiation Monitor has an associated Annunciator, however, it brings in 57C10 – CONTAINMENT RADIATION HI.
C. Correct. RE-7804 or RE-7807, Containment Airborne Radiation Monitors, particulate or gaseous will bring in this alarm.
D. Incorrect. Plausible because this Radiation Monitor has an associated Annunciator, however, it brings in 57C10 – CONTAINMENT RADIATION HI.

Technical Reference(s) SD-SO23-690, Page 129 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 103330 - INTERPRET instrumentation and controls utilized in the Radiation Monitoring System.

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	11
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	A11 AK2.2	
	Importance Rating	3.2	

RCS Overcooling - PTS: Knowledge of the interrelations between RCS Overcooling 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

Proposed Question: Common 65

Given the following conditions after Unit 2 tripped due to an Excessive Steam Demand Event (ESDE):

- SIAS, CIAS, and MSIS have actuated.
- SO23-12-5, Excessive Steam Demand Event, is in progress.
- Steam Generator (SG) E-089 has just blown dry.
- SG E-089 T_{HOT} is 467ºF and stable.
- SG E-089 T_{COLD} is 456ºF and slowly rising.
- SG E-088 pressure is 700 PSIA and lowering.
- SG E-088 T_{HOT} is 486ºF and lowering.
- SG E-088 T_{COLD} is 477ºF and lowering.
- Pressurizer pressure is 1800 PSIA and rising.
- Pressurizer level is 25% and rising.

Which of the following indicates the target pressure in SG E-088 required to stabilize Reactor Coolant System temperature per SO23-12-11, FS-30, Establish Stable RCS Temperature During ESDE?

A. 450 PSIA.

B. 500 PSIA.

C. 550 PSIA.

D. 600 PSIA.

Proposed Answer: A

Explanation:

A. Correct. Target pressure of 450 PSIA corresponds to a saturation temperature for T_{COLD} of ~ 456ºF which is what the affected SG is approaching.
B. Incorrect. Plausible because the saturation temperature of 467°F corresponds to a saturation pressure of 500 PSIA, however, it is the lowest T_{COLD} that is used.
C. Incorrect. Plausible because the saturation temperature of 477°F corresponds to a saturation pressure of 550 PSIA, however, it is the lowest T_{COLD} that is used.
D. Incorrect. Plausible because the saturation temperature of 486°F corresponds to a saturation pressure of 600 PSIA, however, it is the lowest T_{COLD} that is used.

Technical Reference(s) SO23-12-11, FS-30, Step e — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Steam Tables

Learning Objective: 116234 - APPLY saturated and superheated steam tables in solving liquid-vapor problems.

Question Source:
Bank #
Modified Bank # SONGS VISION (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 5, 10
55.43

Comments / Reference: From SONGS VISION	Revision # 09/15/10

Given the following conditions after Unit 2 tripped due to an Excessive Steam Demand Event (ESDE):

- SIAS, CIAS, and MSIS have actuated.
- SO23-12-5, Excessive Steam Demand Event, is in progress.
- Steam Generator (SG) E-089 has just blown dry.
- SG E-089 T_{HOT} is 495ºF and STABLE.
- SG E-089 T_{COLD} is 445ºF and TRENDING UP.
- SG E-088 pressure is 700 psia and TRENDING DOWN.
- SG E-088 T_{HOT} is 485ºF and TRENDING DOWN.
- SG E-088 T_{COLD} is 478ºF and TRENDING DOWN.
- Pressurizer pressure is 1800 psia and TRENDING UP.
- Pressurizer level is 25% and TRENDING UP.

Which of the following indicates the target pressure in SG E-088 required to stabilize Reactor Coolant System temperature?

A. 400 psia
B. 450 psia
C. 500 psia
D. 550 psia

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	1	
	K/A #	G 2.1.19	
	Importance Rating	3.9	

Conduct of Operations: Ability to use plant computers to evaluate system or component status

Proposed Question: Common 66

Given the following condition on the Data Acquisition System (DAS) Home Page:

- RE-7822, Fuel Handling Building Radiation Monitor, DAS icon is FLASHING and LIGHT BLUE in color.

What condition is sensed by RE-7822, Fuel Handling Building Radiation Monitor?

A. HI-HI alarm.

B. Rate of Change alarm.

C. ALERT alarm.

D. Instrument failure.

Proposed Answer: D

Explanation:
A. Incorrect. Plausible because there is a color associated with this alarm, however, it is red.
B. Incorrect. Plausible because there is a color associated with this alarm, however, it is red.
C. Incorrect. Plausible because there is a color associated with this alarm, however, it is yellow.
D. Correct. Given the conditions listed, this is the indication that will be seen on DAS.

Technical Reference(s) SO23-3-2.36, Attachment 1, Step 2.1.2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53211 - DESCRIBE the indications available to determine Radiation Monitor Status on RMS DAS.

Question Source: Bank # ___
Modified Bank # ___ (Note changes or attach parent)
New X

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	11
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	1	
	K/A #	G 2.1.32	
	Importance Rating	3.8	

Conduct of Operations: Ability to explain and apply system limits and precautions

Proposed Question: Common 67

In accordance with Technical Specifications, each of the following Reactor Trips is specifically designed to provide protection from exceeding the Reactor Core Safety Limits with the EXCEPTION of…

A. Loss of Load Trip.

B. High Local Power Density.

C. Low Pressurizer Pressure.

D. Steam Generator Low Level.

Proposed Answer: A

Explanation:

A. Correct. No credit is taken in Technical Specifications for the Loss of Load trip.
B. Incorrect. Plausible because a High Local Power Density is not directly mentioned in the Safety Limits, however, it provides protection for the Peak Centerline Temperature Safety Limit by minimizing the Linear Heat Rate (< 21 KW/FT) generated.
C. Incorrect. Plausible because there is a Safety Limit for high Pressurizer pressure, however, the input from this trip protects against a low Departure from Nucleate Boiling Ratio.
D. Incorrect. Plausible because Steam Generator low level is not directly mentioned in the Safety Limits, however, it provides protection for Reactor Coolant System overpressure caused by a Loss of Feedwater.

Technical Reference(s): Technical Specification 2.1; Licensee Controlled Specification 2.0.100; Technical Specification Table 3.3.1-1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56636 - DESCRIBE the Technical Specification OPERABILITY requirements and action statements associated with the Plant Protection System.

Question Source:
Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge __X__
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 5, 10
55.43 ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Category #	2	
K/A #	G 2.2.12	
Importance Rating	3.7	

Equipment Control: Knowledge of surveillance procedures

Proposed Question: Common 68

Which of the following sets of Feedback Loops should be in service during Turbine Valve Testing per SO23-3-3.34, Turbine Overspeed Protection Valve Operability Tests?

A. 1st Stage Pressure and Megawatt loops.

B. Megawatt and Frequency loops.

C. Steam Chest Pressure and Frequency loops.

D. Frequency and 1st Stage Pressure loops.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the first stage pressure loop is correct, however, the megawatt loop is not used.
B. Incorrect. Plausible because the frequency loop is correct, however, the megawatt loop is not used.
C. Incorrect. Plausible because the frequency loop is correct, however, there is no steam chest pressure loop.
D. Correct. During Turbine Valve Testing the frequency and first stage pressure loops are in service.

Technical Reference(s): SO23-3-3.34, Attachment 1, Notes; SO23-3-3.34, Attachment 8, L&S 5.4 & 5.5 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54310 - Given an operational condition of the Turbine Control and Supervisory Systems addressed by a procedural precaution, limitation, or administrative requirement, STATE the limiting condition and the basis for that limiting condition.

Question Source: Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	10
	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	2	
	K/A #	G 2.2.14	
	Importance Rating	3.9	

Equipment Control: Knowledge of the process for controlling equipment configuration or status

Proposed Question: Common 69

Per OSM-14, Operations Department Expectations, which of the following describes the situation in which Peer Checking may be substituted for Independent Verification and how is it documented?

A. Manipulation of switches in the Control Room Area.
Independent Verification signature must be marked N/A.

B. Manipulation of switches in the Control Room Area.
Peer Checker may sign as Independent Verifier.

C. Verification of locked valve position outside the Control Room Area.
Peer Checker may sign as Independent Verifier.

D. Verification of locked valve position outside the Control Room Area.
Independent Verification signature must be marked N/A.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this is when Peer Checking is used, however, the Independent Verification signature is signed by the Peer Checker.
B. Correct. Per the OSM, when switches are manipulated in the Control Room the Peer Checker may sign as Independent Verifier.
C. Incorrect. Plausible because the Peer Checker may sign as Independent Verifier, however, Peer Checking is not allowed for verifying locked valve positions.
D. Incorrect. Plausible because it could be thought that a Peer Checker could verify the position of a locked valve by observation, however, not only must the Independent Verification signature be signed but the individual would not be performing the position of a Peer Checker for this evolution.

Technical Reference(s): OSM-14, Page 58 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55184 - DESCRIBE the requirements for implementing Peer Checking.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	10	
	55.43		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Category #	2	
K/A #	G 2.2.35	
Importance Rating	3.6	

Equipment Control: Ability to determine Technical Specification Mode of Operation

Proposed Question: Common 70

According to Technical Specification Safety Limit 2.1.1.1, Departure from Nucleate Boiling Ratio (DNBR) shall be maintained at ≥ 1.31 during which of the following MODES?

A. 1 only.

B. 1 and 2 only.

C. 1, 2, and 3 only.

D. 1, 2, 3 and 4.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because MODE 1 is the Power Operation condition, however, this Safety Limit must be met in MODES 1 and 2.
B. Correct. Per Technical Specification Safety Limit 2.1.1.1.
C. Incorrect. Plausible because MODES 1 and 2 are correct and MODE 3 conditions include normal operating pressure and temperature, however, this SL must only be met in MODES 1 and 2.
D. Incorrect. Plausible because the Reactor Coolant System Pressure Safety Limit must be adhered to in these MODES.

Technical Reference(s) Technical Specification 2.1.1.1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56437 - DESCRIBE or SUMMARIZE the basic format and terminology used in the Technical Specifications and Licensee Controlled Specifications, including Safety Limits.

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41 10	
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Category #	3	
K/A #	G 2.3.7	
Importance Rating	3.5	

Radiation Control: Ability to comply with radiation work permit requirements during normal or abnormal conditions

Proposed Question: Common 71

Given the following conditions:

- A system valve alignment must be performed in an area where the radiation level is 75 mREM/hour.
- The individuals current annual Total Effective Dose Equivalent (TEDE) is 750 mREM.

Which of the following is the MAXIMUM amount of time that an individual can work in this area without exceeding their Administrative Dose Control Level (ADCL)?

A. 1 hour.

B. 2 hours.

C. 3 hours.

D. 4 hours.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if Administrative Dose Control Level is not known.
B. Incorrect. Plausible if Administrative Dose Control Level is not known.
C. Correct. 3 hours x 75 mREM/hour = 225 mREM + 750 mREM = 975 mREM. This is under the Administrative Dose Control Level of 1 REM.
D. Incorrect. Plausible if Administrative Dose Control Level is not known.

Technical Reference(s) SO123-VII-20, Step 6.5.3 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 52585 - Given a type of Radiation Area, DESCRIBE its radiation limits, as well as the access controls required by 10CFR20 and site procedures to enter this area.

Question Source:	Bank #		
	Modified Bank #	SONGS VISION	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	12	
	55.43		

Comments / Reference: From SONGS VISION	Revision # 07/13/05

Given the following conditions:

- The Radwaste Operator is required to complete a system valve alignment in an area where the radiation level is 150 mRem/hour.
- The operator's current annual Total Effective Dose Equivalent (TEDE) is 399 mRem.

What is the maximum time he can work in this area and not exceed his Administrative Dose Control Level (ADCL)?

A. 1 hour.

B. 2 hours.

C. 4 hours.

D. 10 hours.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	3	
	K/A #	G 2.3.11	
	Importance Rating	3.8	

Radiation Control: Ability to control radiation releases

Proposed Question: Common 72

Given the following conditions:

- A Waste Gas Decay Tank release is in progress.
- Subsequently, one of the two operating Continuous Exhaust Fans trips.

What action applies to this condition per SO23-8-15, Radwaste Gas Discharge?

The release...

A. must be manually terminated.

B. will automatically terminate by closure of FV-7202, Waste Gas Decay Tank Header Vent Valve.

C. may continue uninterrupted since the remaining fan provides adequate dilution flow.

D. may continue uninterrupted provided the release flowrate is reduced to one half of the original value.

Proposed Answer: A

Explanation:

A. Correct. As outlined in the SO23-8-15, Step 2.4.
B. Incorrect. Plausible because there is an input to FV-7202 from the Continuous Exhaust Fans, however, all the fans must trip for automatic termination.
C. Incorrect. Plausible because the release will continue since one Continuous Exhaust Fan continues to run; however, per procedure the release must be manually terminated.
D. Incorrect. Plausible because the release will continue since one Continuous Exhaust Fan continues to run and it could be thought that this action was acceptable; however, per procedure the release must be manually terminated.

Technical Reference(s) SO23-8-15, Attachment 1, Step 2.4.1
SO23-8-15, Attachment 8, L&S 1.1 & 1.3
Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:	56619 - ANALYZE normal and abnormal operations of the Gaseous Radwaste System.
Question Source:	Bank # SONGS VISION
	Modified Bank # ______ (Note changes or attach parent)
	New ______
Question History:	Last NRC Exam ______
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis ______
10 CFR Part 55 Content:	55.41 13
	55.43 ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	4	
	K/A #	G 2.4.2	
	Importance Rating	4.5	

Emergency Procedures / Plan: Knowledge of systems setpoints, interlocks and automatic actions associated with EOP entry conditions

Proposed Question: Common 73

Given the following condition:

- Unit 2 is operating at 100% power.

Which of the following conditions would require a Unit trip?

A. Main Condenser ΔT of 27ºF.

B. Stator Hot Gas temperature of 150ºF.

C. Instrument Air header pressure of 65 PSIG.

D. Steam Generator E-089 narrow range level of 90%.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because a 25ºF ΔT is an NPDES violation, however, it does not require a Unit trip.
B. Incorrect. Plausible if thought that this temperature would require a plant trip, however, the Turbine would trip automatically at 181ºF.
C. Incorrect. Plausible if thought that this pressure would require a plant trip, however, a Unit trip must be initiated at 50 PSIG.
D. Correct. The Reactor should automatically trip at 89% narrow range level which would in turn trip the Turbine.

Technical Reference(s):
- SO23-15-52.A – 52A11
- SO23-15-99.C – 99C01
- SO23-13-5, Step 2b
- SO23-2-5, Attachment 18, L&S 2.4

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56628 - DESCRIBE the inputs to the Plant Protection System, the purpose of each, their trip setpoints and actuation logic.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	SONGS 2009	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	4	
	K/A #	G 2.4.26	
	Importance Rating	3.1	

Emergency Procedures / Plan: Knowledge of facility protection requirements, including fire brigade and portable firefighting equipment usage

Proposed Question: Common 74

Given the following condition:

- A fire is reported inside the Protected Area of San Onofre Unit 2.

Which of the following correctly describes the duties of the Designated Licensed Operator, after obtaining an emergency kit and radio per SO23-13-21, Fire?

A. Remain in the Control Room and coordinate the fire fighting effort.

B. Proceed to the staging area for the pre-fire plan and act as Fire Brigade Leader.

C. Proceed to the staging area for the pre-fire plan and act as Operations Fire Technical Advisor.

D. Remain in the Control Room and act as Operations Fire Technical Advisor.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that this was the staging location.
B. Incorrect. Plausible because the area is correct, however, this is the Operations Fire Technical Advisor (OPS FTA) not the Fire Brigade Leader
C. Correct. Per SO23-13-21, Step 4.10.
D. Incorrect. Plausible if thought that the OPS FTA remained in the Control Room.

Technical Reference(s) SO23-13-21, Step 4.13 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54227 - Given a Fire Scenario in the protected area, DESCRIBE the duties and responsibilities of the Fire Technical Advisor and the Fire Department Incident Commander.

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #	4	
	K/A #	G 2.4.32	
	Importance Rating	3.6	

Emergency Procedures / Plan: Knowledge of operator response to loss of all annunciators

Proposed Question: Common 75

Given the following conditions:

- Unit 2 is operating at 100% power.
- A complete Loss of Control Room Annunciators has just occurred.

Per SO23-13-22, Loss of Control Room Annunciators, what is the MAXIMUM amount of time the Unit can maintain 100% power if power cannot be restored to the Control Room Annunciators?

A. 1 hour.

B. 2 hours.

C. 4 hours.

D. 8 hours.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because after one (1) hour additional operators are called out to assist with an intensified program of system monitoring per Step 3.e RNO.

B. Incorrect. Plausible if thought one (1) is for monitoring and one (1) hour later a shutdown must be in progress.

C. Correct. After four (4) hours the Unit must commence a shutdown per Step 3.f RNO for a complete Loss of Control Room Annunciators.

D. Incorrect. After eight (8) hours the Unit must commence a shutdown per Step 4.9 RNO for a loss of power to one or more Annunciator Panels.

Technical Reference(s)	SO23-13-22, Steps 3.e RNO & 3.f RNO SO23-13-22, Steps 4.g RNO	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55507 – As the RO, RESPOND to a Loss of Control Room Annunciators per SO23-13-22.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41	10
	55.43	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	E05 EA2.1	
Importance Rating		4.0

Excess Steam Demand: Ability to determine and interpret the following as they apply to the Excess Steam Demand: Facility conditions and selection of appropriate procedures during abnormal and emergency conditions

Proposed Question: SRO 76

Given the following conditions:

- Unit 3 has experienced a Steam Line Break inside Containment.
- SO23-12-5, Excess Steam Demand Event, Step 13, Limit RCS Re-pressurization, is being performed.
- Reactor Coolant System subcooling is 163ºF and rising.
- Containment pressure is 6 PSIG.

Which of the following actions is required to reduce Reactor Coolant System subcooling and prevent Pressurize Thermal Shock (PTS)?

A. Initiate FS-32, Establish Manual Auxiliary Spray, then transition to SO23-12-11, EOI Supporting Attachments, Attachment 5, Core Exit Saturation Margin Control, using the value of PTS subcooling in place of Core Exit Saturation Margin.

B. Establish normal Auxiliary Spray flow then transition to SO23-12-11, EOI Supporting Attachments, Attachment 5, Core Exit Saturation Margin Control, using the value of Core Exit Saturation Margin in place of PTS subcooling.

C. Initiate FS-32, Establish Manual Auxiliary Spray, then transition to SO23-12-11, EOI Supporting Attachments, Attachment 5, Core Exit Saturation Margin Control, using the value of Core Exit Saturation Margin in place of PTS subcooling.

D. Establish normal Auxiliary Spray flow then transition to SO23-12-11, EOI Supporting Attachments, Attachment 5, Core Exit Saturation Margin Control, using the value of PTS subcooling in place of Core Exit Saturation Margin.

Proposed Answer: A

Explanation:

A. Correct. Use FS-32, Establish Manual Auxiliary Spray, due to the harsh Containment and Core Exit Saturation Margin Control using the value of PTS subcooling in place of Core Exit Saturation Margin.
B. Incorrect. Plausible because Attachment 5 use is correct, however, the normal Auxiliary Spray Valve should not be used in a harsh Containment.
C. Incorrect. Plausible because using FS-32 is correct, however, the value of PTS subcooling is used in place of Core Exit Saturation Margin.
D. Incorrect. Plausible because PZR Spray flow is required and the transition step is correct, however, the normal Auxiliary Spray Valve should not be used in a harsh Containment.

Technical Reference(s): SO23-12-5, Step 13c; SO23-14-5, Step 13c; SO23-12-11, Attachment 5 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53926 - As the SRO, DIRECT response to and recovery from an excess steam demand event per SO23-12-5.

Question Source: Bank # SONGS VISION
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ________
55.43 5

Examination Outline Cross-reference:

	RO	SRO
Level		
Tier #		1
Group #		1
K/A #	011 G 2.2.37	
Importance Rating		4.6

Large Break LOCA: Equipment Control: Ability to determine operability and/or availability of safety related equipment

Proposed Question: SRO 77

Given the following conditions on Unit 2:

- A Loss of Coolant Accident is in progress and the following conditions exist:
  - Offsite Power was lost.
  - Both Atmospheric Dump Valves are available for cooldown.
  - T_{HOT} is 377ºF in Loop 1 and 379ºF in Loop 2.
  - Representative Core Exit Thermocouple (REP CET) temperature is 380ºF.
  - Pressurizer pressure is 318 PSIA.
  - Containment pressure is 20 PSIG.
  - Vessel Plenum level is 100%.
  - Recirculation Actuation Signal has occurred.
- It is desired to place the Shutdown Cooling System in service.

Per SO23-12-3, Loss of Coolant Accident, which of the following must be performed prior to placing the Shutdown Cooling System in service?

A. Reduce Pressurizer Pressure to less than 305 PSIA using Manual Auxiliary Spray per SO23-12-11, EOI Supporting Attachments, FS-32, Establish Manual Auxiliary Spray.

B. Continue Cooldown until REP CET temperature is less than 375ºF per SO23-5-1.5, Plant Shutdown from Hot Standby to Cold Shutdown.

C. Reduce Pressurizer Pressure to less than 305 PSIA using Normal Auxiliary Spray per SO23-3-1.10, Pressurizer Pressure and Level Control.

D. Continue Cooldown until REP CET temperature is less than 375ºF per SO23-12-11, EOI Supporting Attachments, Attachment 3, Cooldown / Depressurization.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the procedure entry is correct for reducing RCS pressure given the conditions listed. The calculation for required pressure for placing the Shutdown Cooling System in service is determined by subtracting the current gauge pressure in Containment (20 PSIG) from the Shutdown Cooling entry pressure requirement of 340 PSIA per SO23-12-3, Step 22.f RNO.

B. Incorrect. Plausible because this action is correct with both ADVs available, however, SO23-5-1.5 is not entered until Shutdown Cooling conditions have been established in SO23-12-3 using SO23-12-11, Attachment 3.

C. Incorrect. Plausible if thought that pressure would need to be reduced further based on the absolute Containment pressure vice the gauge pressure. The procedure entry is incorrect given the conditions listed.

D. Correct. REP CET must be less than 375ºF when both ADVs are available. SO23-12-11, Attachment 3, Cooldown / Depressurization is used until SDC entry conditions have been verified.

Technical Reference(s) SO23-12-3, Steps 22.a, c, e, f, & g
SO23-12-3, Step 23.c
Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54354 - As an SRO, DIRECT and COORDINATE the activities of shift personnel to mitigate a loss of coolant accident event.

Question Source: Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 5

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	015/017 AA2.01	
Importance Rating		3.5

RCP Malfunctions: Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Cause of RCP failure

Proposed Question: SRO 78

Given the following conditions on Reactor Coolant Pump (RCP) 3P-002 at 100% power:

- Controlled Bleed Off flow indicates zero (0) GPM.
- Vapor Seal Cavity indicates 20 PSIA.
- Upper Seal Cavity indicates 800 PSIA.
- Middle Seal Cavity indicates 1525 PSIA.
- SO23-3-3.37, Reactor Coolant System Water Inventory Balance, was performed and leakage into the Containment Sump is six (6) GPM.

Which of the following describes the required actions for these conditions SO23-13-6, Reactor Coolant Pump Seal Failure?

A. Reduce the leakage to less than one (1) GPM within four (4) hours or initiate SO23-13-28, Rapid Power Reduction, Attachment 1, RPR - 30% in 5 Minutes to be in HOT STANDBY in one (1) hour.

B. Continue performing SO23-3-3.37, Reactor Coolant System Water Inventory Balance, at the current power level, and initiate a Containment Entry to determine source of leak.

C. Initiate a controlled power reduction per SO23-5-1.7, Power Operations. After Reactor is tripped and SO23-12-1, Standard Post Trip Actions, Steps 1 and 2 have been performed, trip RCP 3P-002.

D. Trip the Reactor, perform SO23-12-1, Standard Post Trip Actions, Steps 1 and 2, then trip RCP 3P-002 after SO23-12-1, Standard Post Trip Actions, Steps 1 and 2 have been performed.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because these would be the actions for an unidentified leak of this magnitude per SO23-13-14, Reactor Coolant Leak, however, SO23-13-6, Reactor Coolant Pump Seal Malfunction, requires a controlled plant shutdown if the Vapor Seal has failed and leakage into Containment is ≤ 4 GPM. Additionally, this is an incorrect usage of SO23-13-28.

B. Incorrect. Plausible because continued operation with a single failed seal is allowed in SO23-13-6, Reactor Coolant Pump Seal Malfunction, however, in this case SO23-13-6 requires a controlled plant shutdown then tripping the Reactor and affected RCP.

C. Correct. This is the action for Controlled Bleed Off leak into Containment of ≤ 4 gpm per SO23-13-6, Step 3.c. New procedure guidance requires performing Steps 1 and 2 of SO23-12-1 prior to tripping the Reactor Coolant Pump.

D. Incorrect. Plausible because SO23-13-6, Reactor Coolant Pump Seal Malfunction, requires tripping the Reactor and affected RCP if the Vapor Seal has failed and leakage into Containment is > 10 gpm, however, a controlled plant shutdown is desired for this condition.

Technical Reference(s): SO23-13-6, Step 3; SO23-13-28, Attachment 1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55458 - As the SRO, DIRECT operator response to a Reactor Coolant Pump Seal Failure per SO23-13-6.

Question Source:
Bank #
Modified Bank # SONGS VISION (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41
55.43 2, 5

Comments / Reference: From SONGS VISION	Revision # 10/06/10

Given the following conditions on Reactor Coolant Pump (RCP) 2P002 at 100% power:

- Controlled Bleed Off flow indicates zero (0) gpm.
- Vapor Seal Cavity indicates 10 PSIG.
- Upper Seal Cavity indicates 810 PSIG.
- Middle Seal Cavity indicates 1525 PSIG.
- Leakage into the Containment Sump is 12 gpm and has been confirmed by performing a Reactor Coolant System Inventory Balance.

Which of the following describes the required actions for these conditions?

A. Reduce the leakage to less than one (1) gpm within four (4) hours or initiate SO23-13-28, Rapid Power Reduction, to be in HOT STANDBY in six (6) hours.

B. A single Seal Stage has failed. Pump the Containment Sump to maintain an indicated level and enter Containment to re-seat the Vapor Seal.

C. Initiate a controlled power reduction per SO23-5-1.7, Power Operations, then immediately trip the Reactor and Reactor Coolant Pump.

D. <u>Trip the Reactor, perform SO23-12-1, Standard Post Trip Actions, Step 1 and 2, then trip RCP 2P002 five (5) seconds after the CEAs have been inserted.</u>

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	027 G 2.4.50	
Importance Rating		4.0

Pressurizer Pressure Control Malfunction: Emergency Procedures/Plan: Ability to verify system alarms setpoints and operate controls identified in the alarm response manual

Proposed Question: SRO 79

Given the following conditions with Unit 2 at 100% power MOC:

- A failure of Pressurizer Pressure Controller 2PIC-0100 occurred two minutes ago.
- Pressurizer pressure is 2270 PSIA and slowly rising.
- Annunciator 50A02 - COLSS ALARM has just actuated.
- CV9005AVE, MSBSCAL Average Secondary Calorimetric Power, is indicating 100.15% and slowly rising.

Which of the following actions is required?

A. Refer to SO23-5-1.7, Power Operations, and lower Turbine load 5 MWe at 10 MWe/min to reduce power to less than or equal to 100%.

B. Maintain current power level per SO23-5-1.7, Power Operations, until NI / ΔT power does not exceed 100% power averaged over an 8 hour period.

C. Trip the Reactor and enter SO23-12-1, Standard Post Trip Actions.

D. Enter SO23-13-28, Rapid Power Reduction, and perform Attachment 2, RPR - 20% / Hour, until power level is within limits.

Proposed Answer: A

Explanation:

A. Correct. Power must be reduced per the guidance in SO23-5-1.7 whenever a COLSS ALARM annunciates.

B. Incorrect. Plausible because NI / ΔT power is allowed to exceed 100%, however, power must be reduced whenever a COLSS ALARM annunciates.

C. Incorrect. Plausible because it could be thought that this condition would require a Reactor trip, however, it is a violation of the license power level of 3438 MWth.

D. Incorrect. Plausible because power must be reduced, however, procedure entry is incorrect.

Technical Reference(s) SO23-5-1.7, Attachment 16, L&S 2.1 & 2.2
SO23-5-1.7, Step 6.4.1 Note
SO23-15-50.A1, 50A02

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56289 - As the SRO, DIRECT plant operations during Power Operations per SO23-5-1.7.

Question Source: Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 2, 5

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	029 EA2.01	
Importance Rating		4.7

ATWS: Ability to determine and interpret the following as they apply to the ATWS: Reactor nuclear instrumentation

Proposed Question: SRO 80

Given the following conditions:

- An Anticipated Transient Without Scram is in progress on Unit 3.
- Buses 3B04 and 3B06 are deenergized.
- SO23-12-9, Functional Recovery, was entered and a Reactor Coolant System (RCS) depressurization is being performed to take advantage of available High Pressure Safety Injection Pumps.
- SO23-12-10, Safety Function Status Checks, Attachment SF-9 for Functional Recovery, is being performed.

Which of the following satisfies the Safety Function Acceptance Criteria for Reactivity Control per SO23-12-10, Safety Function Status Checks?

1. Boration at 45 GPM, Reactor power is 1 x 10^{-1}% and stable.
2. Boration at 25 GPM, Reactor power is 1 x 10^{-5}% and stable.
3. Boration at 30 GPM, Reactor power is 1 x 10^{-3}% and lowering.
4. Boration at 50 GPM, Reactor power is 1% and lowering.

A. 1 and 3.

B. 1 and 4.

C. 2 and 3.

D. 2 and 4.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the Boration rate in #1 is adequate, however, the power level listed for #1 and #3 does not meet the bases document requirements.
B. Incorrect. Plausible because #4 is correct and the Boration rate in #1 is adequate, however, power level must be lowering if the value is greater than1 x 10^{-4}%.
C. Incorrect. Plausible because #2 is correct and power level is lowering in #3, however, the Boration rate must be at least 40 GPM.
D. Correct. Per SO23-12-10, Safety Function Status Checks, Attachment SF-9, #2 is correct per RC-1 and #4 is correct per RC-3.

Technical Reference(s): SO23-12-10, Attachment SF-9, Step 1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53976 - As the SRO, EVALUATE safety function status check for functional recovery accident mitigation per SO23-12-9.

Question Source:
Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 ____
55.43 5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	058 G 2.4.4	
	Importance Rating		4.7

Loss of DC Power: Emergency Procedures/Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures

Proposed Question: SRO 81

Given the following condition:

- Unit 2 is at 20% power.
- Annunciator 63A32 – 2D1 125 VDC BUS TROUBLE, has just actuated.
- DC Bus 2D1 voltmeter indicates zero (0) volts.

Which of the following actions is required per SO23-13-18, Reactor Protection System Failure / Loss of Vital Bus or 1E DC Bus?

A. Initiate SO23-13-28, Rapid Power Reduction, at a rate of approximately 15% to 20% per hour to lower power level less than 5%.

B. Ensure the Reactor and Turbine are tripped, and enter SO23-12-1, Standard Post Trip Actions.

C. Maintain Steam Generator narrow range levels 50% to 70% in AUTO or MANUAL per SO23-9-6, Feedwater Control System Operation.

D. Declare Train B Emergency Diesel Generator inoperable and initiate SO23-3-3.33, Diesel Generator Monthly and Semiannual Testing.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that a Rapid Power Reduction was required to lower power level below the point where the Auxiliary Feedwater System can provide sufficient flow since the Main Feedwater Isolation Valves close, however, the Reactor must be tripped.

B. Correct. Independent of the power level, the Reactor must be verified tripped whenever DC Bus 1 is lost.

C. Incorrect. Plausible because Steam Generator levels must be maintained, however, this is the incorrect procedure to be using since the Main Feedwater Isolation Valves close.

D. Incorrect. Plausible because the action listed and procedure referenced is correct, however, it is the Train A Emergency Diesel Generator that is affected.

Technical Reference(s) SO23-13-18, Step 2.b RNO; SO23-13-18, Attachment 5 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 115110 - RESPOND to a loss of a 1E DC bus with or without loss of the associated vital bus per SO23-13-18.

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 ____
55.43 5

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	032 AA2.09	
Importance Rating		2.9

Loss of Source Range NI: Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Effect of improper HV setting

Proposed Question: SRO 82

Given the following conditions:

- A Reactor Startup is in progress.
- An OPERABLE Excore Nuclear Instrumentation (NI) Log Safety Channel has a fission chamber applied voltage that is higher than normal.

Which of the following identifies the impact on the Nuclear Instrumentation and what actions should be taken to mitigate the situation?

Excore Nuclear Instruments will shift from a Count Rate Circuit to a Campbelling Circuit at a lower...

A. indicated power level. Perform SO23-3-3.2, Excore Nuclear Instrumentation Calibration, Attachment 4, Channel Check of Excore Safety Channels.

B. actual power level. Perform SO23-3-2.15, Excore Instrumentation Operation, Section 6.4, Actions for Failure of a Single Startup Channel.

C. actual power level. Perform SO23-3-3.2, Excore Nuclear Instrumentation Calibration, Attachment 4, Channel Check of Excore Safety Channels.

D. indicated power level. Perform SO23-3-2.15, Excore Instrumentation Operation, Section 6.4, Actions for Failure of a Single Startup Channel.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the procedure entry is correct, however, it is the actual power level that will be affected by a higher applied voltage.
B. Incorrect. Plausible because the first part of the answer is correct, however, this is the incorrect instrument for this procedure entry (Startup Channel vs. Safety Channel).
C. Correct. A channel calibration would be performed to determine if the Log Power Safety Channels are reading within one half decade of each other per SO2-3-3.2.
D. Incorrect. Plausible if thought that this was the instrument response and procedure entry was correct.

Technical Reference(s): SO23-3-3.2, Attachment 4, Step 2.1.4; SD-SO23-470, Page 8 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55032 - Given plant conditions during a reactor startup, EXPLAIN the precaution, limitation or administrative requirement applicable to the situation.
54968 - RESPOND to a miscalibrated nuclear instrument.

Question Source:
Bank # ____
Modified Bank # SONGS VISION (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 ____
55.43 2, 5

Comments / Reference: From SONGS VISION	Revision # 09/06/10

Given the following conditions:

- A Reactor Startup is in progress.
- Excore Nuclear Instrumentation (NI) Log Safety Channels have a fission chamber applied voltage that is lower than normal.

Which of the following:

1.) Identifies the impact on the Nuclear Instrumentation?
2.) Actions should be taken to mitigate the situation?

A. 1.) Excore Nuclear Instruments will shift from a count rate circuit to a Campbelling circuit at a higher actual nuclear power level.
2.) Perform SO23-3-3.2, Excore Nuclear Instrumentation Calibration, Attachment for Channel Check of Excore Safety Channels.

B. 1.) Excore Nuclear Instruments will shift from a count rate circuit to a Campbelling circuit at a higher indicated nuclear power level.
2.) Perform SO23-3-3.2, Excore Nuclear Instrumentation Calibration, Attachment for Channel Check of Excore Safety Channels.

C. 1.) Excore Nuclear Instruments will shift from a count rate circuit to a Campbelling circuit at a higher actual nuclear power level.
2.) Perform SO23-3-2.15, Excore Instrumentation Operation, Section for Failure of a Single Startup Channel.

D. 1.) Excore Nuclear Instruments will shift from a count rate circuit to a Campbelling circuit at a higher indicated nuclear power level.
2.) Perform SO23-3-2.15, Excore Instrumentation Operation, Section for Failure of a Single Startup Channel.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	051 G 2.4.18	
Importance Rating		4.0

Loss of Condenser Vacuum: Emergency Procedures/Plan: Knowledge of specific bases for EOPs

Proposed Question: SRO 83

Which of the following is the basis for performing SO23-3-3.21, Common Shiftly Surveillance, Section 6.4, Shiftly Flow Estimates, when the Condenser Vacuum Pump starts during a Loss of Condenser Vacuum?

Shiftly Flow Estimates are required by...

A. SONGS Offsite Dose Calculation Manual.

B. Combustion Engineering Generic Accident Management Guidelines.

C. Technical Specification LCO 3.4.17, Steam Generator Tube Integrity.

D. Technical Specification LCO 3.4.15, RCS Leakage Detection Instrumentation.

Proposed Answer: A

Explanation:

A. Correct. Starting the Condenser Vacuum Pump causes a shift in the flow rate through the detector for the Radiation Monitoring System. This action requires performance of Shiftly Flow Estimates to ensure limits imposed by the ODCM are being met.
B. Incorrect. Plausible if thought that this document directed this action.
C. Incorrect. Plausible because this Technical Specification does address Steam Generator tube integrity, however, it does not address Condenser Off Gas Radiation Monitoring.
D. Incorrect. Plausible because this Technical Specification does involve gaseous and particulate Radiation Monitoring Instrumentation, however, it does not address Condenser Off Gas Radiation Monitoring.

Technical Reference(s):
- SO23-13-10, Step 1.a
- SO2-3-3.21, Section 6.4
- SO2-3-3.21, Attachment 3
- Technical Specification LCO 3.4.15 & 3.4.17
- Offsite Dose Calculation Manual Table 4.3

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:	55214 - DESCRIBE how the Off Site Dose Calculation Manual is used to ensure the limitations of 10CFR20, Appendix B and the Radiological Effluent Technical Specifications are not exceeded.	
Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	2, 5

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	061 AA2.06	
Importance Rating		4.1

Area Radiation Monitoring System Alarms: Ability to determine and interpret the following as they apply to the Area Radiation Monitoring System Alarms: Required actions if the alarm channel is out of service

Proposed Question: SRO 84

Given the following conditions on Unit 2 at 100% power:

- RI-7820-1, Containment High Range Area Monitor, is out of service.
- While performing Shiftly Surveillance, the OPERATE light for RI-7820-2, Containment High Range Area Monitor, is extinguished with the Mode Select Switch in OPERATE.
- RI-7820-2 OPERATE light bulb was verified to be working.

Which of the following actions are required to be implemented?

A. Submit a Special Report to the NRC within 8 hours. Restore one (1) channel to OPERABLE status within 14 days.

B. Immediately initiate actions to restore one (1) channel, and within 72 hours, initiate alternate methods of monitoring Containment radiation levels.

C. Initiate a Unit Shutdown per the requirements of Technical Specification 3.0.3 and perform an NRC Notification within one (1) hour of initiation of the shutdown.

D. Initiate monitoring of Area Radiation Monitors outside Containment and restore an OPERABLE channel within 30 days.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that a 14 day report was required, however, LCS 3.3.102, Radiation Monitoring Instrumentation, requires initiating the pre-planned alternate which SO23-3-3.21, Common Shiftly Surveillances, defines as monitoring the outside Containment walls for radiation. TS 3.3.11, Post Accident Monitoring Instrumentation, specifies that one channel be restored within 7 days.

B. Correct. LCS 3.3.102, Radiation Monitoring Instrumentation, requires initiating actions immediately to restore one channel and within 72 hours initiate the pre-planned alternate which SO23-3-3.21, Common Shiftly Surveillances, defines as monitoring the outside Containment walls for radiation. TS LCO 3.3.11, Post Accident Monitoring Instrumentation, also specifies that one channel be restored within 7 days.

C. Incorrect. Plausible if thought that not having any OPERABLE Containment High Range Radiation Monitors would require a plant shutdown which then would require the one hour notification, however, LCO 3.3.11, Post Accident Monitoring Instrumentation, specifies that one channel be restored within 7 days and LCS 3.3.102, Radiation Monitoring Instrumentation, requires initiating actions immediately to restore one channel and within 72 hours initiate the pre-planned alternate.

D. Incorrect. Plausible if thought that the actions specified would be the appropriate pre-planned alternate described by LCS 3.3.102, Radiation Monitoring Instrumentation and that a 30 day ACTION would apply, however, LCS 3.3.102 requires initiating the pre-planned alternate which SO23-3-3.21, Common Shiftly Surveillances, defines as monitoring the outside Containment walls for radiation.

Technical Reference(s): SO23-3-3.21, Step 6.6.3; SO23-3-3.21, Attachment 1, Step 2.1; Licensee Controlled Specification 3.3.102.E; Technical Specification Table 3.3.11-1; Technical Specification LCO 3.3.11 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54872 - APPLY all Administrative Requirements, Technical Specifications, and Action Statements for the Radiation Monitoring System.

Question Source:
Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam SONGS 2010

Question Cognitive Level:
Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 ______
55.43 2

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	A16 G 2.4.21	
Importance Rating		4.6

Excess RCS Leakage: Emergency Procedures/Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question: SRO 85

Given the following conditions:

- SO23-12-9, Functional Recovery was entered due to a Loss of Coolant Accident and Station Blackout on Unit 3.
- Train B 1E 4160 V Bus 3A06 is cross tied to Train B 1E 4160 V Bus 2A06.
- Unit 2 Train B Emergency Diesel Generator 2G003 is supplying Buses 2A06 and 3A06.
- One (1) Group 6 CEA and one (1) Part Length CEA are NOT inserted in the Reactor.
- Containment pressure is 4 PSIG.
- Reactor Coolant System pressure is 1500 PSIA.
- The Pressurizer is empty.
- Subcooled Margin is 0ºF.
- All Train B Engineered Safety Feature actuations occurred as required.
- Reactor Vessel Plenum Level is 21%.

While performing Safety Function Status Checks you learn that the only available High Pressure Safety Injection Pump has just tripped.

Which of the following identifies the highest Safety Functions that are NOT met?

A. Reactivity Control and Vital Auxiliaries Control.

B. RCS Inventory Control and RCS Pressure Control.

C. RCS Pressure Control and Core Heat Removal.

D. Core Heat Removal and RCS Heat Removal.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because Reactivity Control would be entered if two full-length CEAs were withdrawn, however, even with the 1E Buses cross connected, as long as power is available, Vital Auxiliaries would not be entered.

B. Correct. Given the conditions listed, RCS Inventory Control and RCS Pressure Control are the concerns and implementation of FR-3, Recovery - RCS Inventory Control would be performed.

C. Incorrect. Plausible because RCS Pressure Control would be entered, however, not enough information is provided to determine if Core Heat Removal entry is required.

D. Incorrect. Plausible because both Core Heat Removal and RCS Heat Removal could be entered, however, not enough information is provided on SG levels and Core Exit Temperature.

Technical Reference(s) SO23-12-10, Attachment SF-3, Steps 1 to 5 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Steam Tables

Learning Objective: 53929 - As the SRO, DIRECT accident mitigation by use of the functional recovery EOI per SO23-12-9.

Question Source: Bank #
Modified Bank # SONGS VISION (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 5

Comments / Reference: From SONGS VISION	Revision # 10/19/09

Given the following conditions on Unit 2:

- SO23-12-9, Functional Recovery was entered due to a Small Break Loss of Coolant Accident and Excess Steam Demand Event.
- Offsite power is unavailable and Train B Emergency Diesel Generator failed to start.
- Containment pressure is 17 PSIG.
- Reactor Coolant System pressure is 1600 PSIA and the Pressurizer is empty.
- Subcooled Margin is 2°F.
- All Train A Engineered Safety Feature actuations occurred as required.
- Reactor Vessel Level is 21% in the Plenum.

While performing Safety Function Status Checks you learn that the only available Salt Water Cooling Pump has just tripped.

1.) Which ONE (1) of the following identifies the highest Safety Functions that are NOT met?

2.) Which Functional Recovery Procedure will be entered FIRST?

A. 1.) RCS Inventory Control and RCS Pressure Control.
2.) Implement FR-3, Recovery - RCS Inventory Control.

B. 1.) RCS Pressure Control and Core Heat Removal.
2.) Implement FR-4, Recovery - RCS Pressure Control.

C. 1.) Core Heat Removal and RCS Heat Removal.
2.) Implement FR-5, Recovery - Heat Removal.

D. 1.) Containment Isolation and Containment Temperature and Pressure Control.
2.) Implement FR-6, Recovery - Containment Isolation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	006 G 2.2.37	
	Importance Rating		4.6

Emergency Core Cooling System: Equipment Control: Ability to determine operability and / or availability of safety-related equipment

Proposed Question: SRO 86

Given the following conditions:

- Unit 2 is in MODE 2.
- Safety Injection Tank (SIT) pressures are as follows:
  - SIT T-007 is at 619 PSIA.
  - SIT T-008 is at 605 PSIA.
  - SIT T-009 is at 612 PSIA.
  - SIT T-010 is at 622 PSIA.

Which of the following are Technical Specification CONDITION, REQUIRED ACTION, and COMPLETION TIME?

A. One (1) SIT is inoperable. Restore SIT to OPERABLE status within 24 hours.

B. One (1) SIT is inoperable. Enter Limiting Condition for Operation 3.0.3 immediately.

C. Two (2) SITs are inoperable. Restore at least one (1) SIT to OPERABLE status within 24 hours.

D. Two (2) SITs are inoperable. Enter Limiting Condition for Operation 3.0.3 immediately.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that only one SIT was inoperable; in that case the REQUIRED ACTION listed is correct.

B. Incorrect. Plausible because the Technical Specification action is correct, however, there are two (2) SITs inoperable.

C. Incorrect. Plausible because (2) SITs are inoperable, however, this is the REQUIRED ACTION for one inoperable SIT.

D. Correct. Nitrogen cover pressure in each SIT must be ≥ 615 PSIA and ≤ 655 PSIA. With two (2) SITs inoperable, Limiting Condition for Operation 3.0.3 must be entered immediately.

Technical Reference(s) Technical Specification LCO 3.5.1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56649 - Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 2

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	062 G 2.4.21	
	Importance Rating		4.6

AC Electrical Distribution System: Emergency Procedures/Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question: SRO 87

Given the following conditions:

- A Station Blackout occurred one (1) hour ago on Unit 3.
- SO23-12-8, Station Blackout, is being implemented.
- Qualified Safety Parameter Display System is OPERABLE.
- Pressurizer level is 10% and slowly lowering.
- Pressurizer pressure is 1300 PSIA.
- Core Exit Thermocouple temperature is 550ºF.
- Turbine Driven Auxiliary Feedwater Pump is in operation.
- Steam Generator narrow range levels are 75% and slowly rising.

Based on the conditions listed, which of the following should be performed?

A. Transition to SO23-12-9, Functional Recovery, FR-5, Recovery – Heat Removal, when Reactor Vessel water level is in the Plenum region.

B. Remain in SO23-12-8, Station Blackout, and continue SO23-12-11, EOI Supporting Attachments, Attachment 6, Diesel Generator Failure Follow-up Actions.

C. Refer to SO23-V-5, SONGS Severe Accident Management Guidelines, since core damage is imminent.

D. Implement SO23-12-11, EOI Supporting Attachments, Attachment 5, Core Exit Saturation Margin Control, when Pressurizer level reaches 0%.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that a Pressurizer level of 10% would equate to level in the Plenum region, however, P-140, Auxiliary Feedwater Pump is operating and entry into SO23-12-9 to perform FR-5 would not be required.

B. Correct. SO23-14-8, Station Blackout Bases, lists Diesel Generator Failure Follow-up Actions as one of the restoration tools to accomplish the goal of restoring 1E 4kV electrical power.

C. Incorrect. Plausible because the Severe Accident Management Guidelines are available for use, however, conditions have not degraded to the point where the SAMGs would be implemented.

D. Incorrect. Plausible because SO23-12-8, Steps 17.e and 17.f required this action, however, this would not be performed unless power was restored.

Technical Reference(s)
- SO23-12-8, Foldout Page
- SO23-12-8, Step 10.h & RNO
- SO23-14-8, Attachment 1, Step 3
- SO23-12-8, Step 17.e & 17.f
- SO23-V-5, Section 1.0

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53940 - As the SRO, DIRECT operator response to a Station Blackout per SO23-12-8.

Question Source:
- Bank #
- Modified Bank # (Note changes or attach parent)
- New X

Question History: Last NRC Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41
- 55.43 5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	013 A2.06	
	Importance Rating		4.0

Engineered Safety Features Actuation System: Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent ESFAS actuation

Proposed Question: SRO 88

Given the following conditions:

- Unit 3 is in MODE 1.
- An inadvertent Safety Injection Actuation Signal (SIAS) occurs during Matrix Channel surveillance testing.
- Pressurizer Pressure is 2245 PSIA.
- Containment Pressure is 0.3 PSIG.
- I & C reports that a Matrix Channel power supply has failed.

Which of the following Technical Specifications applies and what action must be taken to mitigate the situation?

A. Technical Specification LCO 3.3.5, ESFAS Instrumentation, is applicable.
Enter SO23-12-1, Standard Post Trip Actions, and trip the Reactor and Turbine.

B. Technical Specification LCO 3.3.5, ESFAS Instrumentation, is applicable.
Enter SO23-3-2.7.2, Safety Injection System Removal/Return to Service Operation, to reset the SIAS to restore Normal Containment Cooling.

C. Technical Specification LCO 3.3.6, ESFAS Logic and Manual Trip, is applicable.
Enter SO23-13-28, Rapid Power Reduction, and place the Unit in MODE 3 within six (6) hours.

D. Technical Specification LCO 3.3.6, ESFAS Logic and Manual Trip, is applicable.
Enter SO23-13-17, Recovery from Inadvertent Safety Injection/Containment Isolation or Containment Spray, and override and stop all Charging Pumps.

Proposed Answer: D

Explanation:

A. Incorrect. The Technical Specification entry is incorrect. A Reactor trip and entry into SO23-12-1 per SO23-13-6, RCP Seal Failure would be considered because Controlled Bleed Off flow to the VCT is lost, however, the CBO relief would lift and the RCPs would be protected.
B. Incorrect. Plausible because the condition to reset SIAS to restore normal Containment Cooling is correct, however, the Technical Specification and procedure entry are both incorrect.
C. Incorrect. Plausible because the Technical Specification entry is correct, however, this would be the wrong procedure to enter because a Rapid Power Reduction is not required.
D. Correct. The SIAS must be recovered from in less than one hour to avoid Technical Specification LCO 3.0.3 entry. This is the correct Tech Spec and procedure entry. Letdown is isolated on an SIAS and the Charging Pumps must be overridden and stopped.

Technical Reference(s): Technical Specification LCO 3.3.5; Technical Specification LCO 3.3.6; SO23-13-17, Step 3 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56649 - Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

Question Source: Bank # SONGS VISION
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam SONGS 2009

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 2, 5

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	061 A2.07	
Importance Rating		3.5

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

Proposed Question: SRO 89

Given the following condition:

- A loss of DC Bus D3 was immediately followed by a Unit Trip and Station Blackout.

Which of the following identifies the impact on P-140, Auxiliary Feedwater Pump, and what procedure should be used to place P-140, Auxiliary Feedwater Pump, in service?

P-140 is disabled due to the inability to open...

A. HV-4716, K-007 Turbine Inlet Steam Supply. Enter SO23-12-6, Loss of Feedwater.

B. both HV-4705 and HV-4706, P-140 Motor Operated Discharge Valves. Enter SO23-12-6, Loss of Feedwater.

C. HV-4716, K-007 Turbine Inlet Steam Supply. Refer to SO23-2-4, Auxiliary Feedwater System Operation, Section 6.9, Local Manual AFW Equipment Operations.

D. both HV-4705 and HV-4706, P-140 Motor Operated Discharge Valves. Refer to SO23-2-4, Auxiliary Feedwater System Operation, Section 6.9, Local Manual AFW Equipment Operations.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because DC Bus D3 controls the K-007 Turbine Inlet Steam Supply and Governor Control Unit. Entry into SO23-12-6 would direct reset of the trip; however, a Functional Recovery entry must be performed (Station Blackout and Loss of Feedwater).

B. Incorrect. These valves are powered from DC Buses D1 and D2. Entry into SO23-12-6 would direct reset of the trip; however, a Functional Recovery entry must be performed (Station Blackout and Loss of Feedwater).

C. Correct. P-140 is supplied with DC power from DC Buses D1, D2, and D3. The Main Steam Inlets and Pump Discharge Valves are powered from D1 and D2. DC Bus D3 controls the K-007 Turbine Inlet Steam Supply and Governor Control Unit. The procedure entry is correct.

D. Incorrect. Plausible because these valves are DC powered, however, these valves are powered from DC Buses D1 and D2. The procedure entry is correct.

Technical Reference(s): SO23-13-18, Attachment 7; SO23-2-4, Section 6.9; SD-SO23-780, Figure 1; SD-SO23-780, Pages 57 & 58; SO23-12-6, Purpose / Entry Conditions

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53940 - As the SRO, DIRECT operator response to a station blackout per SO23-12-8.

Question Source:
Bank # SONGS VISION
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 ____
55.43 5

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	012 G 2.2.22	
Importance Rating		4.7

Reactor Protection System: Emergency Procedures/Plan: Knowledge of limiting conditions for operations and safety limits

Proposed Question: SRO 90

Given the following conditions:

- Unit 2 is in MODE 1.
- SO23-3-3.5, CEA / Reactor Trip Circuit Breaker (RTCB) Operability Testing, Attachment 3, RTCB Monthly Test – Modes 1 and 2, is in progress.
- When HS-9132-1, Reactor Trip 1 pushbutton, was depressed, TCB-1 failed to open.

Which of the following correctly identifies the INOPERABLE RTCB Channel(s) and the Technical Specification REQUIRED ACTION?

A. Only RTCB Channel 1 is INOPERABLE.
ENSURE open RTCBs 1 and 5 within one (1) hour.

B. Only RTCB Channel 1 is INOPERABLE.
ENSURE open RTCBs 1 and 2 within one (1) hour.

C. RTCB Channels 1 and 2 are INOPERABLE.
ENSURE open RTCBs 1 and 5 within one (1) hour.

D. RTCB Channels 1 and 2 are INOPERABLE.
ENSURE open RTCBs 1 and 2 within one (1) hour.

Proposed Answer: A

Explanation:

A. Correct. This is the Technical Specification LCO 3.3.4 REQUIRED ACTION because only RTCB Channel 1 is INOPERABLE.
B. Incorrect. Plausible because only RTCB Channel 1 is INOPERABLE, however, RTCBs 1 and 5 must be opened.
C. Incorrect. Plausible because the REQUIRED ACTION is correct, however, only RTCB Channel 1 is INOPERABLE.
D. Incorrect. Plausible if thought that both channels were INOPERABLE and these were their respective RTCBs..

Technical Reference(s): SO23-3-3.5, Attachment 3, Steps 2.1 & 3.1; Technical Specification LCO 3.3.4.B; Technical Specification LCO 3.3.4 Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56649 - Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

Question Source:
- Bank # ______
- Modified Bank # ______ (Note changes or attach parent)
- New X

Question History: Last NRC Exam ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 2, 5

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #		2
Group #		2
K/A #	015 A2.03	
Importance Rating		3.5

Nuclear Instrumentation System: 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: Xenon oscillations

Proposed Question: SRO 91

Given the following while at 100% power:

- A large xenon oscillation is in progress greater than 0.04 Axial Shape Index (ASI) units.
- ASI is currently more positive than Equilibrium Shape Index (ESI).
- The oscillation is moving towards the top of the Core.

Which of the following identifies the guidance in SO23-5-1.7, Power Operations, to mitigate the xenon oscillation and what is the Technical Specification Basis for this action?

A. Direct inserting PLCEAs in small, smooth frequent movements of less than 3 inches per minute to maintain ASI at ESI ± 0.03 shape index units.
Limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of the transient.

B. Commence a Reactor Coolant System boration to lower temperature and drive ASI towards ESI.
Avoids fuel centerline melt by limiting the local peak linear heat rate during steady state operation.

C. Direct inserting Group 6 CEAs at a rate of 5 inches per minute to maintain ASI at ESI ± 0.03 shape index units.
Limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of the transient.

D. Commence a Reactor Coolant System dilution to raise temperature and drive ASI away from ESI.
Avoids fuel centerline melt by limiting the local peak linear heat rate during steady state operation.

Proposed Answer: A

Explanation:

A. Correct. Given the fact that this is identified as a large oscillation in the Stem, inserting Part Length CEAs is appropriate at the speed listed. The Technical Specification Bases is correct as listed.

B. Incorrect. Plausible because all components of this answer are correct for controlling a xenon oscillation, however, given the size of the oscillation listed this is an inappropriate response. Controlling a small xenon oscillation with temperature is appropriate however, this is a large oscillation. The TS basis is also incorrect.

C. Incorrect. Plausible because all components of this answer are correct for controlling a xenon oscillation, however Group 6 rods are used if the part length CEAs are ineffective. Additionally, the rods speed listed is too fast to avoid pellet clad interaction.

D. Incorrect. Plausible because all components of this answer are correct for controlling a xenon oscillation, however, commencing a dilution to raise temperature and drive ASI away from ESI only works if ASI is more negative than ESI. The condition in the Stem is opposite this. The TS basis is also incorrect.

Technical Reference(s): SO23-5-1.7, Attachment 13; Technical Specification LCO 3.2.5 Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56524 - MONITOR and CONTROL ASI during steady state operation per SO23-5-1.7.

56649 - Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

Question Source:
Bank # ____
Modified Bank # SONGS VISION (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41 ____
55.43 2, 5

Comments / Reference: From SONGS VISION	Revision # 09/24/08

Given the following while at 100% power Middle-Of-Core (MOC) conditions:

- A xenon oscillation is in progress greater than 0.05 Axial Shape Index (ASI) units.
- Axial Shape Index is currently more positive than Equilibrium Shape Index (ESI).
- The oscillation is moving towards the top of the Core.

Which ONE (1) of the following:
1.) Identifies the guidance provided to the Reactor Operator to mitigate the xenon oscillation?
2.) What is the Technical Specification Basis for this action?

A. <u>1.) Direct inserting PLCEAs in small, smooth frequent movements of less than 3 inches per minute to maintain ASI at ESI ± 0.01 shape index units.</u>
<u>2.) Limits damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of the transient</u>.

B. 1.) Commence a Reactor Coolant System boration to lower temperature and drive ASI towards ESI.
2.) Avoids fuel centerline melt by limiting the local peak linear heat rate during steady state operation.

C. 1.) Direct inserting Group 6 CEAs in small, smooth frequent movements of less than 5 inches per minute to maintain ASI at ESI ± 0.01 shape index units.
2.) Limits the amount of damage to the fuel cladding during an accident by ensuring that the plant is operating within acceptable conditions at the onset of the transient.

D. 1.) Commence a Reactor Coolant System dilution to raise temperature and drive ASI away from ESI.
2.) Avoids fuel centerline melt by limiting the local peak linear heat rate during steady state operation.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #	045 A2.08	
Importance Rating		3.1

Main Turbine Generator System: Ability to (a) predict the impacts of the following malfunctions or operations on the MTG System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Steam dumps are not cycling properly at low load, or stick open at higher load (isolate and use atmospheric reliefs when necessary)

Proposed Question: SRO 92

Given the following conditions during a Unit 3 startup:

- Reactor power is 20% and T_{COLD} is 544ºF.
- Main Turbine is ready to be synched to the grid.
- Steam Bypass Control System (SBCS) is in AUTO.
- A slightly negative Isothermal Temperature Coefficient is known to exist.

When the Main Turbine is synched to the grid with a Block Load of 100 MWe applied, the SBCS Valves remain in their pre-Block Load position.

Which of the following identifies the plant response to this condition and what action should be taken to mitigate the situation?

RCS T_{COLD} will…

A. rise, resulting in a negative reactivity addition. Place SBCS in MANUAL and restore temperature to program per SO23-3-2.18, Steam Bypass System Operation, Section 6.3, Operation of the SBCS during Plant Startup.

B. rise, resulting in a negative reactivity addition. Withdraw CEAs to restore temperature to program per SO23-3-2.19, Control Element Drive Mechanism Control System Operation, Section 6.12, Repetitive or Emergent Manual CEA Positioning.

C. lower, resulting in a positive reactivity addition. Place SBCS in MANUAL and restore temperature to program per SO23-3-2.18, Steam Bypass System Operation, Section 6.3, Operation of the SBCS during Plant Startup.

D. lower, resulting in a positive reactivity addition. Withdraw CEAs to restore temperature to program per SO23-3-2.19, Control Element Drive Mechanism Control System Operation, Section 6.12, Repetitive or Emergent Manual CEA Positioning.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because taking control of SBCS to restore temperature is appropriate per SO23-5-1.7, however, with a negative ITC, a positive reactivity addition will occur. This is the procedure and section if SBCS Operation were required.

B. Incorrect. Plausible if thought that this is the response with a –ITC. Additionally, SO23-5-1.7 does not allow CEA withdrawal to restore from a secondary plant perturbation. This is the procedure and section if CEA positioning were required.

C. Correct. Per SO23-5-1.7, a - ITC will insert positive reactivity; therefore, taking control of SBCS to restore temperature is appropriate. This guidance is contained in the SO23-3-2.18, SBCS Operation, Section 6.3.

D. Incorrect. Plausible because a positive reactivity addition is occurring, however, SO23-5-1.7 does not allow CEA withdrawal to restore from a secondary plant perturbation. This is the procedure and section if CEA positioning were required.

Technical Reference(s): SO23-5-1.7, Attachment 16, L&S 2.4; SO23-3-1.1, Attachment 11, L&S 2.5; SO23-10-1, Attachment 3 Guidelines; SO23-3-2.18, Section 6.3; SO23-3-2.19, Section 6.12

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56289 - As the SRO, DIRECT plant operations during Power Operations per SO23-5-1.7.

Question Source: Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 5

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #	035 G 2.4.47	
Importance Rating		4.2

Steam Generator System: Emergency Procedures/Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material

Proposed Question: SRO 93

Given the following post-trip conditions on Unit 2:

- Containment pressure is 30 PSIG and slowly rising.
- Containment Radiation Monitors are not alarming or trending to alarm.
- Reactor Coolant System (RCS) T_{COLD} is 510ºF and slowly lowering.
- Steam Generator E-088 wide range (WR) level is 55% and lowering.
- Steam Generator E-088 pressure is 500 PSIG and lowering.
- Steam Generator E-089 narrow range (NR) level is 35% and rising.
- Steam Generator E-089 pressure is 875 PSIG and lowering.
- Chemistry sample indicates elevated boron and radioactivity on Steam Generator (SG) E-089.

Following SO23-12-1, Standard Post Trip Actions, which EOI should be entered and what actions should be taken to mitigate the event?

A. Enter SO23-12-9, Functional Recovery.
Isolate SG E-089 and use SG E-088 to cooldown the RCS.

B. Enter SO23-12-9, Functional Recovery.
Isolate SG E-088 and use SG E-089 to cooldown the RCS.

C. Enter SO23-12-4, Steam Generator Tube Rupture.
Isolate SG E-088 and use SG E-089 to cooldown the RCS.

D. Enter SO23-12-4, Steam Generator Tube Rupture.
Isolate SG E-089 and use SG E-088 to cooldown the RCS.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the procedure entry is correct, however, when one SG has an ESDE and the other SG has a SGTR, then the SG with the SGTR should be used to cooldown per the Note in SO23-12-11, Attachments 28 and 29.

B. Correct. Per the Diagnostic Attachment in SO23-12-1, when two events are in progress entry into the Functional Recovery procedure is required. SG E-088 is isolated (ESDE) and SG E-089 (SGTR) is used for cooldown.

C. Incorrect. Plausible because isolating SG E-088 and cooling down with SG E-089 are the correct actions, however, entry into SO23-12-4 would be correct if only one event was in progress.

D. Incorrect. Plausible if it was not determined that two events were in progress.

Technical Reference(s): SO23-12-1, Attachment 1; SO23-12-11, Attachment 28, Step 4 Note; SO23-12-11, Attachment 29, Step 2 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53929 - As the SRO, DIRECT accident mitigation by use of the Functional Recovery EOI per SO23-12-9.

Question Source:
- Bank #
- Modified Bank # (Note changes or attach parent)
- New X

Question History: Last NRC Exam

Question Cognitive Level:
- Memory or Fundamental Knowledge
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41
- 55.43 5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Category #		1
	K/A #	G 2.1.5	
	Importance Rating		3.9

Conduct of Operations: Ability to use procedures related to shift staffing, such as minimum crew complement, over time limitations, etc.

Proposed Question: SRO 94

Given the following conditions:

- Unit 2 is in MODE 1.
- Unit 3 is in MODE 6 performing CORE ALTERATIONS.
- Crew A is manned to the minimum composition per SO123-0-A1, Conduct of Operations, Attachment 2, Minimum Operations Shift Crew Composition.
- Crew A has two (2) hours remaining on Shift.
- The Refueling SRO has been diagnosed with heat exhaustion and must leave Containment.

Which of the following describes the required action(s) in this situation while maintaining minimum shift manning?

A. The Shift Manager must suspend CORE ALTERATIONS until after turnover of responsibilities to another qualified Refueling SRO.

B. Responsibilities of the Refueling SRO may be turned over to the Reactor Engineering Supervisor for the remainder of the shift.

C. Responsibilities of the Refueling SRO may be turned over to the Shift Manager for the remainder of the Shift. Unit 2 CRS assumes the Shift Manager duties.

D. The CRS of Unit 3 can relieve the Refueling SRO. The Shift Manager may perform concurrent SM/CRS duties until Shift Relief by a qualified CRS.

Proposed Answer: A

Explanation:

A. Correct. Given the conditions listed, CORE ALTERATIONS must be suspended as described in footnote #3 of SO123-0-A1.
B. Incorrect. Plausible if thought that CORE ALTERATIONS were control solely by Reactor Engineering, however, the Refueling SRO need only leave their position on the Refueling Deck to require termination of CORE ALTERATIONS.
C. Incorrect. Plausible if thought this was allowed, however, Minimum Manning would not be met.
D. Incorrect. Plausible if thought this was allowed, however, Minimum Manning would not be met.

Technical Reference(s) SO123-0-A1, Attachment 2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53394 - Given plant conditions, DETERMINE the shift manning requirements.

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______
55.43 6

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Category #		1
	K/A #	G 2.1.34	
	Importance Rating		3.5

Conduct of Operations: Knowledge of primary and secondary plant chemistry limits

Proposed Question: SRO 95

Given the following conditions:

- Unit 2 has been at 100% power for 2 hours.
- Chemistry has just reported the following Reactor Coolant System (RCS) samples:
  - Chloride concentration is 1.9 ppm.
  - Fluoride concentration is 1.6 ppm.

Which of the following actions should the Control Room Supervisor take per the Licensee Controlled Specifications and why?

Within six (6) hours,...

A. place the Unit in MODE 3 because only the Chloride concentration has exceeded the transient limit.

B. place the Unit in MODE 3 because the Chloride and Fluoride concentrations have exceeded the transient limit.

C. reduce RCS pressure to ≤ 500 PSIA because only the Chloride concentration has exceeded the steady-state limit.

D. reduce RCS pressure to ≤ 500 PSIA because the Chloride and Fluoride concentrations have exceeded the steady-state limit.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the fluoride concentration has exceeded the transient limit, however, the Unit must be placed in MODE 3 within six hours.
B. Correct. Unit must be placed in MODE 3 within 6 hours and be in MODE 5 within 36 hours because the chloride and fluoride concentrations have exceeded the transient limit.
C. Incorrect. Plausible because this is a REQUIRED ACTION per CONDITION D, however, this is only applicable in MODES 5 or 6.
D. Incorrect. Plausible because the steady-state chloride and fluoride limits have been exceeded, however, this is only applicable in MODES 5 or 6.

Technical Reference(s)	LCS Table 3.4.101-1	Attached w/ Revision # See Comments / Reference
	Licensee Controlled Specification 3.4.101.A	
	Licensee Controlled Specification 3.4.101.C	
	Licensee Controlled Specification 3.4.101.D	

Proposed references to be provided during examination: None

Learning Objective: 56649 - Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

Question Source:	Bank #	SONGS VISION	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content:	55.41	
	55.43	1, 2

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Category #		2
	K/A #	G 2.2.17	
	Importance Rating		3.8

Equipment Control: Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator

Proposed Question: SRO 96

Which of the following conditions would allow a High Risk Evolution to be performed per SO23-5-1.8.1, Shutdown Nuclear Safety?

Per SO23-5-1.8.1, Shutdown Nuclear Safety, a High Risk Evolution can be performed...

A. given the evolution or activity is covered by an approved site procedure and a planned Defense-In-Depth is in place.

B. only if dual redundant components are available and have been protected.

C. as long as the Safety Monitor calculates a Core Damage Frequency (CDF) less than 0.0001 (one in 10,000 Reactor years).

D. whenever a Safety Function will not be compromised or challenged.

Proposed Answer: A

Explanation:

A. Correct. The evolution must be covered by an approved site procedure and have Defense In Depth in place.
B. Incorrect. Plausible because it could be thought that a redundant component is always required, however, if it involved a pump that did not have any redundancy, that pump itself must be protected.
C. Incorrect. Plausible because this calculation would be performed before the evolution was performed, however, there is not a calculated value that must be met.
D. Incorrect. Plausible because this is a desired state while performing a High Risk Evolution, however, it is not required.

Technical Reference(s) SO23-5-1.8.1, Section 6.11 Attached w/ Revision # See Comments / Reference
SO23-5-1.8.1, Attachment 1

Proposed references to be provided during examination: None

Learning Objective:	56326 - Given plant and equipment status and the applicable Safety Function Fulfillment Plan (SFFP) or Safety Function Protection Plan (SFPP), DETERMINE if the planned Defense In Depth is in the required status for the given outage configuration.
Question Source:	Bank # SONGS VISION Modified Bank # ______ (Note changes or attach parent) New ______
Question History:	Last NRC Exam ______
Question Cognitive Level:	Memory or Fundamental Knowledge X Comprehension or Analysis ______
10 CFR Part 55 Content:	55.41 ______ 55.43 5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Category #		2
	K/A #	G 2.2.40	
	Importance Rating		4.7

Equipment Control: Ability to apply Technical Specifications for a system

Proposed Question: SRO 97

Given the following condition on Unit 2 at 100% power:

- Both Control Element Assembly Calculators (CEAC) have been declared INOPERABLE due to an improper algorithm discovered during I & C testing.

All the following Technical Specification REQUIRED ACTIONS apply, EXCEPT;

A. Verify the Departure from Nucleate Boiling Ratio requirement of LCO 3.2.4, Departure from Nucleate Boiling Ratio (DNBR), is met.

B. Verify all Full Length and Part Length Control Element Assembly (PLCEA) Groups are fully withdrawn and maintained that way except during Surveillance testing.

C. Verify the RSPT / CEAC INOPERABLE addressable constant in each Core Protection Calculator is set to indicate that the applicable CEAC(s) is (are) INOPERABLE.

D. Maintain the Control Element Assembly Rod Drive System in the PLCEA position in the event rod motion is required.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that this would not be required, however, this is a REQUIRED ACTION per the LCO.
B. Incorrect. Plausible if thought that this would not be required, however, this is a REQUIRED ACTION per the LCO.
C. Incorrect. Plausible if thought that this would not be required, however, this is a REQUIRED ACTION per the LCO.
D. Correct. This is NOT part of the additional requirements when the COMPLETION TIME limit expires and the CEACs have NOT been restored.

Technical Reference(s) Technical Specification LCO 3.3.3.B Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56649 - Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

Question Source: Bank # SONGS VISION
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 ____
55.43 2

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		3
K/A #	G 2.3.11	
Importance Rating		4.3

Radiation Control: Ability to control radiation releases

Proposed Question: SRO 98

Given the following condition:

- Chemistry reports Reactor Coolant System (RCS) DOSE EQUIVALENT I-131 in the Unacceptable Region of Technical Specification LCO 3.4.16, RCS Specific Activity, Figure 3.4.16-1, Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit versus Percent of RATED THERMAL POWER With Reactor Coolant Specific Activity > 1.0 µci/gm DOSE EQUIVALENT I-131.

What is the Technical Specification Basis for reducing RCS temperature to less than 500ºF following plant shutdown?

A. Minimizes the temperature related degradation of Ion Exchangers while RCS cleanup is in progress.

B. Slows the release of noble gas to the Reactor Coolant reducing the activity source term.

C. Prevents the direct release of radioactivity should a Steam Generator Tube Rupture occur.

D. Minimizes the magnitude of iodine spiking phenomenon caused by the plant shutdown.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because temperature degradation of Ion Exchangers is always a concern, however, not for this Technical Specification.
B. Incorrect. Plausible because a lower temperature would slow the release of noble gases, however, the basis is direct release of radioactivity during an SGTR.
C. Correct. As outlined in the bases document for Technical Specification LCO 3.4.16.
D. Incorrect. Plausible because higher temperatures would increase the size of the fuel cladding cracks therefore inducing iodine spiking, however, the basis is related to a SGTR.

Technical Reference(s) Technical Specification LCO 3.4.16 Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56649 - Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

Question Source: Bank # SONGS VISION
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______
55.43 2, 4

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		4
K/A #	G 2.4.27	
Importance Rating		3.9

Emergency Procedures / Plan: Knowledge of “fire in the plant” procedures

Proposed Question: SRO 99

Given the following conditions:

- Unit 2 is in MODE 1.
- A fire is confirmed in the SCE Switchyard Relay House.
- SO23-13-21, Fire, was entered and personnel are at the Relay House fighting the fire.
- 10 minutes after the Fire is confirmed, the following occurs:
  - Both 220 kV Switchyard Tie Breakers 4112 and 6112 open.
  - Both Unit 2 Reserve Auxiliary Transformer Supply Breakers 4042 and 6042 open.
- Two (2) minutes later Santiago Breaker 4052 opens.

Which of the following action is required to be performed per SO23-13-21, Fire?

A. Maintain steady state conditions per SO23-5-1.7, Power Operations.

B. Perform a manual trip and enter SO23-12-1, Standard Post Trip Actions.

C. Commence a controlled shutdown per SO23-5-1.7, Power Operations, to be in MODE 3 in six (6) hours.

D. Commence a Rapid Power Reduction per SO23-13-28, Rapid Power Reduction, to be in MODE 3 in six (6) hours.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the Reactor does NOT trip, however, the Unit must be tripped.

B. Correct. When the Unit 2 Reserve Auxiliary Transformer Supply Breakers open, the Emergency Diesel Generators will start and load onto the 1E Buses. The Reactor does NOT trip. SO23-13-21 requires a manual trip because control of Unit 2 can longer be maintained.

C. Incorrect. Plausible because the Reactor would continue to operate in this condition, however, the Unit must be tripped because the fire has caused abnormal operation of multiple systems or components.

D. Incorrect. Plausible because the Reactor does NOT trip, however, Unit 2 must be tripped per SO23-13-21.

Technical Reference(s): SO23-13-21, Steps 2.0 Note & 4.5.5; SO23-13-21, Attachment 1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53944 - As the SRO, DIRECT the operator response to a fire in the plant per SO23-13-21.

Question Source: Bank # SONGS VISION
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		4
K/A #	G 2.4.45	
Importance Rating		4.3

Emergency Procedures / Plan: Ability to prioritize and interpret the significance of each annunciator or alarm

Proposed Question: SRO 100

Given the following conditions:

- While performing SO23-12-1, Standard Post Trip Actions, a Reactor Trip Override (RTO) malfunction has caused an excessive cooldown.
- Manual control of Feedwater was taken in an attempt to stop the overfeed condition.
- The following annunciators are now in alarm:
  - 56A51 – SG2 E088 PRESS LO PRETRIP.
  - 56A53 – SG1 E089 PRESS LO PRETRIP.
- Steam Generator pressures are lowering.
- Subsequently, the following alarms are received:
  - 56A41 – SG2 E088 PRESS LO CHANNEL TRIP.
  - 56A43 – SG1 E089 PRESS LO CHANNEL TRIP.

Which of the following actions should be taken by the Control Room Supervisor?

A. Declare a cooldown in progress, reset the Main Steam Isolation Signal setpoints, and direct the Shift Technical Advisor to continue evaluating plant status per SO23-12-10, Safety Function Status Checks.

B. Ensure Main Feedwater Block Valves, HV-4047, and HV-4051, are CLOSED. Continue monitoring SG pressures while initiating a two-prong attack per SO23-12-9, Functional Recovery.

C. Ensure initiation of Main Steam Isolation Signal. To prevent RCS repressurization, when cooldown terminates, stabilize RCS temperature for the lowest T_{COLD} per SO23-12-1, Standard Post Trip Actions.

D. Verify the Main Steam Safeties are closed, reset the Main Steam Isolation Signal setpoints, and direct the Reactor Operator to perform the FR-5, Recovery – Heat Removal per SO23-12-9, Functional Recovery.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because resetting the setpoints would prevent an MSIS, however, an MSIS needs to be initiated. The Shift Technical Advisor action is appropriate for these conditions.

B. Incorrect. Plausible because closing the MFBVs is desirable, however, an MSIS should be initiated. A two-pronged attack is used when the Functional Recovery procedure is implemented.

C. Correct. This is the desired action because the cooldown must be stopped or MSIS must be initiated per SO23-12-1.

D. Incorrect. Plausible because this action is desirable to determine the source of the cooldown, however, a MSIS should be initiated.

Technical Reference(s): SO23-12-1, Step 8; SO23-15-56.A – 56A51; OSM-9, Step 23 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54352 - As the SRO, DIRECT standard post trip actions per SO23-12-1.

Question Source:

- Bank #: SONGS VISION
- Modified Bank #: ______ (Note changes or attach parent)
- New: ______

Question History: Last NRC Exam ______

Question Cognitive Level:

- Memory or Fundamental Knowledge: ______
- Comprehension or Analysis: X

10 CFR Part 55 Content:

- 55.41: ______
- 55.43: 5