Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	003 A1.03		
	Importance Rating	2.6		

Reactor Coolant Pump: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including: RCP motor stator winding temperatures

Question # 1

- Unit 2 at 15% power
- Multiple computer points are in alarm on Reactor Coolant Pump (RCP) 2-03
- RCP 2-03 parameters are as follows:

SEAL WTR IN TEMP (T0183A)	225°F and slowly rising
MOT UP RDL BRG TEMP (T0453A)	187°F and slowly rising
MOT LOW RDL BRG TEMP (T0455A)	190°F and slowly rising
LOW SEAL WATER BRG TEMP (T0457A)	220°F and slowly rising
MOT STAT WNDG TEMP (T0452A)	304°F and slowly rising

Which of the following lists the required actions in accordance with ABN-101, Reactor Coolant Pump Trip/Malfunction?

A. ENSURE at least 35 gpm of Component Cooling Water flow to RCP 2-03 Thermal Barrier Cooler

B. ENSURE at least 8 gpm of Seal Injection flow to RCP 2-03 #1 Seal

C. TRIP the Reactor, TRIP RCP 2-03, due to high stator winding temperature

D. TRIP the Reactor, TRIP RCP 2-03, due to high seal water inlet temperature

Answer: C

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge of and predict required actions on a High RCP Motor Stator Winding Temperature.

Explanation:

A. Incorrect. Plausible because Step 3 of ABN-101, Section 8.0, is to verify CCW thermal barrier flow to all RCPs greater than or equal to 35 gpm per pump, however, the Reactor and the RCP should have been tripped on Step 2. If the candidate believed RCP 2-03 did not need to be tripped this would be the correct answer.

B. Incorrect. Plausible because ABN-101, Section 8.0, Step 2 RNO would require a reactor trip followed by a trip of RCP 2-03, THEN Seal Injection flow must be increased, as necessary. Ensuring a Seal Injection flow of greater than 8 gpm without having conducted a reactor and RCP 2-03 trip is wrong.

C. Correct. With RCP 2-03 Motor Stator Winding Temperature above 300°F a reactor trip followed by a trip of RCP 2-03 is required. ABN-101, Attachment 1, RCP Parameters, lists the upper operating limits for all RCPs, above these limits the RCP must be tripped in accordance with ABN-101, Section 8.0, Step 2 RNO.

D. Incorrect. Plausible because the temperature limit at which an RCP trip is required on High Seal Water Inlet Temperature is 235°F. This is a number often confused with the high temperature limit on the Lower Seal Water Bearing Temperature of 225°F. If a candidate confused these numbers they would believe a reactor trip/RCP 2-03 trip would be required on High Seal Water Inlet Temperature.

Technical Reference(s)	ABN-101	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21ABN101OB105 Analyze the response to an RCP High Temperature or Loss of CCW to any RCP in accordance with ABN 1-01, Reactor Coolant Pump Trip/Malfunction.

Question Source: Bank # ILOT8392
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

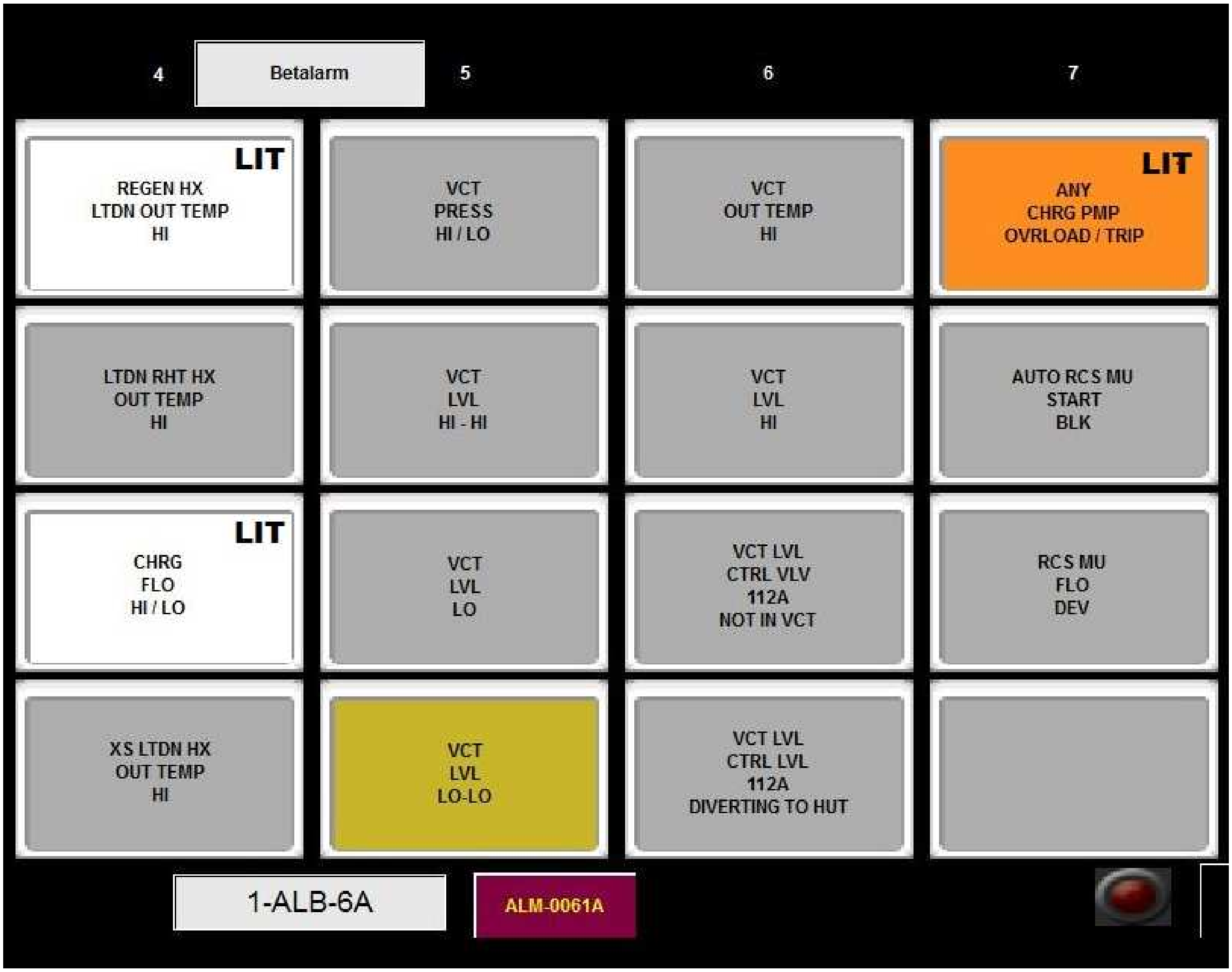
Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 3
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	004 G.2.4.45		
	Importance Rating	4.1		

Chemical and Volume Control: Ability to prioritize and interpret the significance of each annunciator or alarm.

Question # 2



Based on the LIT alarms above, what action has priority?

A. Isolate letdown

B. Start standby CCP

C. Place PDP in service

D. Adjust charging flow control valve

Answer: B	

K/A Match:

The question is a match for the K/A as it requires the applicant to assess Chemical and Volume Control System alarms which have annunciated together and interpret their significance. The applicant must then assign priority to the alarms by stating the priority action.

Explanation:

A. Incorrect. Plausible because isolating letdown would be performed if the standby CCP would not start per ABN-105, CVCS Malfunctions.

B. Correct. Starting a CCP is an initial operator action performed per ABN-105, CVCS Malfunctions.

C. Incorrect. Plausible because PDP would be started if the standby CCP would not start, after isolating letdown and verifying RCP thermal barrier CCW flow > 35 gpm per ABN-105, CVCS Malfunctions.

D. Incorrect. Plausible because the CHRG FLO HI/LO response includes the charging flow control valve malfunction as a cause.

Technical Reference(s)	ABN-105	Attached w/ Revision # See Comments / Reference
	ALM-0061A	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSCS1OB103, Describe the components of the Chemical and Volume Control system including interrelations with other systems to include interlocks and control loops.

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 6
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	005 K1.06		
	Importance Rating	3.5		

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

Question # 3

- Unit 1 Large Break LOCA
- RHR aligned for Cold Leg Recirculation per EOS-1.3A, Transfer to Cold Leg Recirculation

In Cold Leg Recirculation, Train A RHR is aligned to _________.

A. CCP suctions ONLY

B. SIP suctions ONLY

C. BOTH CCP and SIP suctions

D. CCP or SIP suctions with manual valve manipulation

Answer: C

K/A Match:

This matches the K/A by requiring the operator to know the physical connection between the RHRS from the pump discharge to the CCP and SIP suctions.

Explanation:

A. Incorrect. Plausible because 1/1-8804A, RHRP 1 TO CCP SUCT VLV would appear to supply only the CCPs due to the nomenclature however, once 1/1-8807A&B (SI < - - > CHRG SUCT XTIE VLVS) are open both/either RHRP is aligned to supply the suctions of the CCPs and SIPs.

B. Incorrect. Plausible because 1/1-8804B, RHRP 2 TO SIP SUCT VLV would appear to supply only the SIPs due to the nomenclature however, once 1/1-8807A&B (SI < - - > CHRG SUCT XTIE VLVS) are open both/either RHRP is aligned to supply the suctions of the CCPs and SIPs.

C. Correct. Once 1/1-8807A&B (SI < - - > CHRG SUCT XTIE VLVS) and 1/1-8804A&B (RHRPs TO CCP/SIP VLVs) are opened in EOS-1.3A both trains of RHR are aligned to supply the suction the CCPs and SIPs.

D. Incorrect. Plausible because opening 1/1-8807A&B (SI < - - > CHRG SUCT XTIE VLVS) is required to cross-tie the CCP and SIP suction and it could be thought they could be manipulated to select either pump however, the cross-tie valves are not closed by either the cold-leg or hot-leg recirculation procedure.

Technical Reference(s)	RHR Study Guide	Attached w/ Revision # See Comments / Reference
	EOS-1.3A	
	SOP-102A	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSRH1OB105 Explain the normal, abnormal, and emergency operation of the Residual Heat Removal 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
	Level of Difficulty	3

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	006 K5.11		
	Importance Rating	2.5		

Emergency Core Cooling: Knowledge of the operational implications of the following concepts as they apply to ECCS: Basic heat transfer equation

Question # 4

- Unit 1 Small Break LOCA
- RCS pressure is 685 psig
- RWST temperature is 72°F
- All RCPs are tripped
- Core Decay Heat is 1.5%
- RVLIS 11" light is the ONLY RVLIS light LIT

What is the MINIMUM ECCS flow rate required to maintain a MAXIMUM of 430°F ΔT from ECCS inlet to core exit?

(Assume a specific heat for water = 1.0 Btu/lbm-°F and specific volume for water = 72°F)

A. 900 gpm

B. 800 gpm

C. 400 gpm

D. 300 gpm

Answer: A

K/A Match:

The question is a K/A match as it requires the applicant to use the Basic Heat Transfer Equation to calculate a minimum ECCS flow for a given situation.

Explanation:

A. Correct. The 430°F ΔT corresponds to the maximum temperature rise which would maintain the core exit in a subcooled liquid.

In accordance with the equation sheet:

$\dot{Q} = \dot{m}c_p\Delta T \quad \dot{Q} = 3612\ MW\ x\ 0.015 = 54.18\ MW$

$\Delta T = 430°\text{F}$

1MW = 3.41 x 10^6 Btu/hr

1 gal = 8.35 lbm

$\dot{m} = 858\ gpm$

Thus 900 gpm would be in excess of the 858 and would maintain the ΔT less than 430°F.

B. Incorrect. As described in ‘A’ 800 gpm is less than 858 but is plausible if choosing a lower amount based on a lack of understanding that the ECCS flow rate must be greater than 858 gpm.

C. Incorrect. This value is more than the value calculated in ‘D’ below and is plausible if the assumption is made when calculating a value of 300 gpm that this is not allowing for any conservatism and 400 gpm should be selected.

D. Incorrect. This value is calculated the same as the value described in ‘A’ above but with the inherent mistake of using MWe instead of MWth, thus 1265 vs. 3612. This mistake yields 300 gpm.

Technical Reference(s)	GFE Equation and Conversion Sheet	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: GFE Equation and Conversion Sheet

Learning Objective: LO21MCOTAAOB103, Analyze the core cooling mechanisms of a LOCA

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 3

Level of Difficulty 4

10 CFR Part 55 Content: 55.41 14
55.43

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	007 A2.06		
	Importance Rating	2.6		

Pressurizer Relief/Quench Tank: Ability to (a) predict the impacts of the following malfunctions or operations on the PS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Bubble formation in PZR

Question # 5

- Unit 1 in MODE 5 performing RCS vacuum fill to establish a Pressurizer Bubble
- 1-LI-452, PRZR LVL COLD CAL is 25%
- CCP 1-01 and the PDP are both unavailable
- CCP 1-02 is supplying RCP seal injection and RCS fill from the RWST
- PRT Level is 5%
- PRT pressure is 23" Hg vacuum

Subsequently:

CCP 1-02 trips and restoration of a charging pump is expected to take several hours.

What action is taken in response to CCP 1-02 trip?

A. Close both PORVs to isolate vacuum fill manifold.

B. Break vacuum by closing containment service air isolation valve.

C. Establish 20 psig nitrogen pressure in PRT.

D. Align PRT vent to in service gas decay tank.

Answer: B

K/A Match:

The question is a K/A match as a malfunction occurs during Pressurizer Bubble formation and the applicant is required to delineate what procedural actions are necessary for the Pressurizer Relief Tank system.

Explanation:

A. Incorrect. Plausible because closing the PORVs would isolate the vacuum manifold on the PRT from the RCS but vacuum would not be broken on the PRT.

B. Correct. Per SOP-101A the PRT should not be subjected to a vacuum environment longer than necessary. Closing the Containment service air isolation valve secures the eductor which breaks the vacuum on the PRT.

C. Incorrect. Plausible because establishing a 20 psig nitrogen overpressure would remove the vacuum from the PRT but PRT pressure is limited to 15 psig with the PORVs open.

D. Incorrect. The PRT is aligned to the in service gas decay tank for fission gas processing not to change PRT pressure.

Technical Reference(s)	SOP-101A	Attached w/ Revision # See Comments / Reference
	SOP-109A	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSRC1OB105, Explain the normal, abnormal and emergency operation of the Reactor Coolant 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 ___
Level of Difficulty 4

10 CFR Part 55 Content: 55.41 10
55.43 ___

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	007 A4.04		
	Importance Rating	2.6		

Pressurizer Relief/Quench Tank: Ability to manually operate and/or monitor in the control room: PZR vent valve

Question # 6

- FRC-0.1A, Response to Inadequate Core Cooling is in progress
- Pressurizer Vent Valves 1-HV-3609 and 1-HV-3610 are OPEN
- Procedure directs Pressurizer Vent Valves be manually CLOSED

After operating BOTH hand switches, the following indications are observed:

- 1-HS-3609, PRZR VENT VLV – GREEN light LIT and RED light DARK
- 1-HS-3610, PRZR VENT VLV – GREEN light DARK and RED light LIT

During operation a KEY ________ required to CLOSE each valve.

Based on indications, flow from pressurizer vent valves to containment ________ isolated.

A. is NOT
is

B. is NOT
is NOT

C. is
is

D. is
is NOT

Answer: C

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate the ability to manually operate and monitor the Pressurizer Vent Valves from the control room. Valves do not discharge to the PRT at CPNPP.

Explanation:

A. Incorrect. Part 1 is incorrect but plausible if the applicant does not realize that these valves must be operated with a key and that the key must remain in the switch thus each valve requires its own key. Part 2 is correct as described in ‘C’ below.

B. Incorrect. Part 1 is incorrect but plausible as described in ‘A’ above. Part 2 is incorrect but plausible as described in ‘D’ below.

C. Correct. Part 1 is correct as a Key is required as these valves do not have a handswitch per se but a key switch which must be turned with the key in the switch and each valve must be operated with its own key as the key cannot be removed in the OPEN position. Part 2 is correct in that the pressurizer vent valves are in series and therefore either valve indicating closed would isolate the flow path.

D. Incorrect. Part 1 is correct as described in ‘C’ above. Part 2 is incorrect but plausible if believed that the pressurizer vent valves were parallel flow paths as opposed to in series on a single vent path.

Technical Reference(s)	FRC-0.1A	Attached w/ Revision # See Comments / Reference
	M1-0251	
	E1-0064 Sht. 042	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSRC1OB103, Describe the components of the Reactor Coolant system including interrelations with other systems to include interlocks and control loops

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
Level of Difficulty 4

10 CFR Part 55 Content: 55.41 3
55.43

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	008 K4.09		
	Importance Rating	2.7		

Component Cooling Water: Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: The "standby" feature for the CCW pumps

Question # 7

Which of the following identifies the condition in which Train B Component Cooling Water Pump will automatically start?

A. AUTO start signal of Train A Station Service Water Pump on low flow in Train B Station Service Water header.

B. Component Cooling Water low flow on Train A Component Cooling Water Heat Exchanger outlet.

C. AUTO start signal of Train A Station Service Water Pump on low pressure in Train B Station Service Water header.

D. Component Cooling Water low pressure on Train A Component Cooling Water Heat Exchanger outlet.

Answer: D

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge of the “Standby Auto Start” feature for CCW pumps.

Explanation:

A. Incorrect. Plausible because the Train B Station Service Water Pump will start on low pressure in the Train A Station Service Water header which, in turn, will provide a start signal to the Train B Component Cooling Water Pump. However, the distractor is written for the wrong trains and for “low flow” which is often confused with “low pressure.”

B. Incorrect. Plausible because the Component Cooling Water Pump will auto start on low pressure in the alternate Component Cooling Water Train. “Low pressure” and “low flow” are often confused when determining Component Cooling Water pump start signals.

C. Incorrect. Plausible because the Train B Station Service Water Pump will start on low pressure in the Train A Station Service Water header which, in turn, will provide a start signal to the Train B Component Cooling Water Pump. This distractor is written for the wrong trains.

D. Correct. This condition will auto start the Train B Component Cooling Water Pump.

Technical Reference(s)	SOP-502A Precautions and Limitations	Attached w/ Revision # See Comments / Reference
	CCW Study Guide	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSCC1OB105 Explain the normal, abnormal and emergency operation of the Component Cooling Water System.

Question Source: Bank # ILOT8157
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 7
55.43

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	010 K1.05		
	Importance Rating	3.4		

Pressurizer Pressure Control: Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: PRTS

Question # 8

- IPO-005A, Plant Cooldown from Hot Standby to Cold Shutdown is in progress
- Preparing for Solid Plant Operations
- 1-TI-454, PRZR Vapor Temp is 430°F
- 1-PI-469 PRT PRESS is 5 psig
- Each PORV is opened for a short duration

Temperature of steam entering PRT is ________. (ignore ambient heat losses)

Discharge from each PORV enters PRT through ________.

A. 430°F
different discharge lines

B. 430°F
same discharge line

C. <430°F
different discharge lines

D. <430°F
same discharge line

Answer: D

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge of both the physical connections between the PRZ PCS and the PRT but also the relationship between the fluid properties in the two systems.

Explanation:

A. Incorrect. Part 1 is incorrect but plausible because it could be thought that the temperature of the fluid passing through the valve doesn't change only the pressure does. Part 2 is incorrect but plausible as described in 'C' below.

B. Incorrect. Part 1 is incorrect but plausible as described in 'A' above. Part 2 is correct as described in 'D' below.

C. Incorrect. Part 1 is correct as described in 'D' below. Part 2 is incorrect but plausible because multiple items discharge to the PRT but they all combine into a common header before they enter the PRT.

D. Correct. Part 1 is correct in accordance with the steam tables, for saturated steam at 430°F an enthalpy of 1203.85 Btu/lbm exists. When passing through the isenthalpic expansion process to an absolute pressure of 20 psia (5 psig) the resultant fluid temperature is superheated at 326°F. Part 2 is correct in that all of the valves discharge into the PRT through a common header.

Technical Reference(s)	Steam Tables	Attached w/ Revision # See Comments / Reference
	M1-0251	

Proposed references to be provided during examination: Steam Tables

Learning Objective: LO21SYSRC1OB103, Describe the components of the Reactor Coolant system including interrelations with other systems to include interlocks and control loops

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 3
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	010 K2.03		
	Importance Rating	2.8		

Pressurizer Pressure Control: Knowledge of bus power supplies to the following: Indicator for PORV position

Question # 9

- Unit 1 is responding to Red Path on Heat Sink CSFST
- Both PORVs are OPEN for Bleed and Feed

Subsequently:

- A loss of 125V DC Distribution Panel 1ED2-1 occurs

1-PCV-____, PRZR PORV, will be CLOSED with GREEN handswitch light ______.

A. 455A
DARK

B. 455A
LIT

C. 456
DARK

D. 456
LIT

Answer: C

K/A Match:

The question matches the K/A as it requires the operator to demonstrate knowledge of the power supply to the PORV indications in the control room and the valve response to the loss of power.

Explanation:

A. Incorrect. First part is incorrect but plausible if the applicant believes that 1ED2-1 is the power supply for PORV-455A. There are multiple instances of valves that one would think are powered from a particular train based on valve numbering and powered from the opposite train. For instance, 1-HS-3609, PRZR Vent Valve would appear by component numbering to be a Train A valve, however, it is powered from 1ED2-1. Second part is correct (See C below).

B. Incorrect. First part is incorrect but plausible (See A above). Second part is incorrect but plausible if the applicant does not recall that the control power supplies the AOV solenoids and indication power. It could be thought that indication power is separate and a closed PORV would normally have the GREEN light LIT.

C. Correct. First part is correct because control power to PORV-456 is supplied from 1ED2-1and the loss of control power to the AOV solenoids will cause the valve to close. Second part is correct because control power to PORV-456 is supplied from 1ED2-1 and when control power is lost the handswitch light indications for the PORV on CB-05 will extinguish.

D. Incorrect. First part is correct (See C above). Second part is incorrect but plausible (See B above).

Technical Reference(s)	E1-0020 Sheet F	Attached w/ Revision # See Comments / Reference
	E1-0064 Sheet 12	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSPP1OB107 Explain the normal, abnormal and emergency operation of the Pressurizer Pressure and Level Control 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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 3
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	012 A1.01		
	Importance Rating	2.9		

Reactor Protection: Ability to predict and/or monitor Changes in parameters (to prevent exceeding design limits) associated with operating the RPS controls including: Trip setpoint adjustment

Question # 10

Unit 1 at 80% power during Unit startup.

Which of the following describes how Overtemperature and Overpower N-16 trip setpoints will change, if any?

(Assume RCS and flux distribution parameters remain on program/target, as power is raised to 100%)

	Overtemperature N-16 Setpoint	**Overpower N-16 Setpoint**
A.	Increase	remain the same
B.	decrease	decrease
C.	decrease	remain the same
D.	remain the same	increase

Answer: C

K/A Match:

The question matches the K/A as it requires the applicant to predict the changes in RPS parameters and in particular the variable OT N-16 setpoint.

Explanation:

A. Incorrect. Plausible because the Overpower trip setpoint does not change, and the applicant could readily confuse the fact that the Overtemperature setpoint decreases, with the term "increase"; i.e., meaning the actual value is closer to the setpoint.
B. Incorrect. First part is correct. The Overpower setpoint does not increase from its nominal value. There are however, some effects of temperature shielding due to Tcold changes, as explained in the Study Guide material for this topic. This could be confused and misinterpreted by the applicant as a decrease in the setpoint.
C. Correct. Since Tavg at 80% power is less than at 100% power, the Overpower setpoint will be at

its nominal full power value and thus, will not change from 80% to 100% power, assuming Tavg stay on program. The Overtemperature setpoint, on the other hand, CAN increase or decrease from its nominal value. Since program Tavg will increase several more degrees during the power escalation, the trip setpoint will become more limiting, decreasing to its nominal full power value.

D. Incorrect. Overtemperature setpoint change could be confused and reversed with Overpower, which does not change for the conditions given. Plausibility of second part previously described in "A" above.

Technical Reference(s)	Reactor Protection and ESFAS Study Guide	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21SYSES1OB103, Describe the Reactor Protection and Engineered Safeguard Actuation Systems including interrelations with other systems to include interlocks and control loops.

Question Source: Bank # NRC 2015
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam 2015 NRC Question 24

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 5
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	012 A3.07		
	Importance Rating	4.0		

Reactor Protection: Ability to monitor automatic operation of the RPS, including: Trip breakers

Question # 11

Unit 1 is at 100%

When the following indications are observed:

What is the status two minutes after above indications first appeared, regarding Steam Dump operation and valves which receive a Feedwater Isolation signal?

Steam dumps are operating on the __________ Controller and Feedwater _______ isolated.

A. Load Reject
Is NOT

B. Load Reject
is

C. Plant Trip

Is NOT

D. Plant Trip
is

Answer: D	

K/A Match:

The question matches the K/A as it requires the operator to monitor the positions of the Reactor Trip Breakers following automatic operation of the Reactor Protection System and based on the indication determines plant response.

Explanation:

A. Incorrect. First part is incorrect see B below. Second part is incorrect see C below.

B. Incorrect. First part is incorrect but plausible if the student does not have a solid understanding of the relationship of each train of P-4 associated with the reactor trip breakers. Second part is correct see D below.

C. Incorrect. First part is correct see D below. Second part is incorrect but plausible if the operator does not realize that a single P-4 is all that is required for the Feedwater Isolation to be complete and that a single train P-4 signal does exist.

D. Correct. B train P-4 reactor trip permissive determines which controller the steam dumps function on. For this case the B train RTB opened as designed which places the steam dumps on the plant trip controller. The second part is correct based on the fact that P-4 also inputs to the Feedwater isolation signal. Since 2 minutes have passed from the time of the trip Tave has lowered to less than 564°F which is a setpoint, if a P-4 signal is present and Tave is less than 564°F a Feedwater isolation signal will be generated.

Technical Reference(s)	RPS and ESFAS Study Guide	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21SYSES1OB104, Explain the instrumentation and controls of the Reactor Protection System and Engineered Safety Features Actuation System and Predict the system response.

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
	Level of Difficulty	3

10 CFR Part 55 Content:	55.41	6
	55.43	

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	013 K1.18		
	Importance Rating	3.7		

<u>Engineered Safety Features Actuation</u>: Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: Premature reset of ESF actuation

Question # 12

- Unit 1 Small Break LOCA
- Safety Injection actuated on Low Pressurizer Pressure
- Containment pressure is 2.5 psig and slowly rising
- ESF actuation signals have been reset per EOP-1.0A, Loss of Reactor or Secondary Coolant
- RCS Cooldown to Cold Shutdown initiated per EOS-1.2A, Post LOCA Cooldown and Depressurization

<u>Subsequently:</u>

- Large Break LOCA occurs
- Containment pressure is 25 psig and rising

Which of the following completes statements below regarding status of RHR system?

RHR Pumps ________.

When RWST level lowers to 33%, 1/1-8811A/B, CNTMT SMP TO RHRP 1 AND 2 SUCT ISOL VLVS ________.

A. must be MANUALLY started
must be MANUALLY opened

B. must be MANUALLY started
AUTOMATICALLY open

C. AUTOMATICALLY start
must be MANUALLY opened

D. AUTOMATICALLY start
AUTOMATICALLY open

Answer: A

K/A Match:

This question matches the K/A by requiring knowledge of an ESFAS System (RHR) and how resetting the Safety Injection and RWST Auto Swapover Signals effects that system when subsequent conditions meet the criteria for a Safety Injection.

Explanation:

A. Correct. A SBLOCA has occurred and the crew has transitioned through the ERG network from EOP-0.0A to EOP-1.0A to EOS-1.2A. The ESF Actuation Signals, including RHR Auto Switchover, are required to be reset in EOP-1.0A prior to transition to EOS-1.2A and again in EOS-1.2A prior to conducting a cooldown to Cold Shutdown. When the SI is reset manual action must be taken to restart the RHR pumps if RCS pressure drops in an uncontrolled manner to less than 325 psig (425 psig for adverse containment conditions). Also, when RHR Auto Switchover is reset, as RWST level reaches 33% the 1/1-8811A/B valves will no longer automatically open, therefore, requiring manual action to open these valves by meeting the necessary interlocks as delineated in EOS-1.3A, Step 3.a. RNO.

B. Incorrect. First part is correct, see ‘A’ above. Second part is incorrect but plausible because the RWST Auto Swapover Signal is an energized to actuate signal. This signal along with the Containment Spray signal are exceptions to the rule for ESFAS, as the others are de-energized to actuate. This leads many to believe that although the Signal was reset after the SI that it will provide the signal again when RWST level reaches the 33%.

C. Incorrect. First part is incorrect but plausible because there is a subsequent condition (Containment Pressure HI-1) that meets the requirements for Safety Injection. A common misconception is that a subsequent Safety Injection will occur and that the ECCS pumps will automatically start. However, when the SI signal has been reset and the P-4 signal is present then an Auto SI Block will occur as long as 60 seconds elapsed from time of SI initiation to when the signal was reset. Second part is correct, see ‘A’ above.

D. Incorrect. First part is incorrect, see ‘C’ above. Second part is incorrect, see ‘B’ above.

Technical Reference(s)	Rx Protection and ESFAS Study Guide	Attached w/ Revision # See Comments / Reference
	1-PCIP, Window 2.8	
	EOP-1.0A	
	EOS-1.2A	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSRH1OB104 Explain the instrumentation and controls of the Residual Heat Removal System and predict the system response.

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
	Level of Difficulty	3

10 CFR Part 55 Content:	55.41	7
	55.43	

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	073 K3.01		
	Importance Rating	3.6		

Process Radiation Monitoring: Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent release

Question # 13

- Turbine Building Sumps are being released per STA-603-16, Secondary Waste Release Data Sheet
- 1-RE-5100 (TBD172), TURBINE BUILDING DRAINS RADIATION MONITOR has lost power

1-RV-5100A, TURB BLDG SMP 1-02 DISCH DRN HDR TO LVW/EVAP POND ISOL VLV ________.

1-RV-5100B, TURB BLDG SMP 1-02 DISCH HDR TO WWHT ISOL VLV ________.

A. CLOSES
CLOSES

B. OPENS
CLOSES

C. CLOSES
OPENS

D. OPENS
OPENS

Answer: C

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate the effect that a loss of the process radiation monitor has on a radioactive effluent release

Explanation:

A. Incorrect. Plausible if thought that all discharge will isolate on a loss of power to the radiation monitor, however the loss of power to the monitor has the same effect as a high radiation signal which would cause 1-RV-5100A to close isolating the Low Volume Waste flow path and 1-RV-5100B to open aligning the Co-current Waste flowpath.

B. Incorrect. Plausible if thought normal release flowpath is to Co-current Waste, however 1-RV-5100A will close isolating the Low Volume Waste flowpath and 1-RV-5100B will open aligning the Co-current Waste flowpath.

C. Correct. The normal release flowpath is to Low Volume Waste, so 1-RV-5100A will close isolating the Low Volume Waste flowpath and 1-RV-5100B will open aligning the Co-current Waste flowpath.

D. Incorrect. Plausible if thought that 1-RV-5100A and 1-RV-5100B open in response to the loss of power to the radiation monitor which controls valve positions.

Technical Reference(s)	ALM-3200	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21SYSRM1OB104, Explain the instrumentation and controls of the Digital Radiation Monitoring System and Predict the system response.

Question Source: Bank # ILOT6520
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 13
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	022 K4.04		
	Importance Rating	2.8		

Containment Cooling: Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following: Cooling of control rod drive motors

Question # 14

The Control Rod Drive Mechanism (CRDM) ventilation fans and air handling units operate as follows;

The CRDM ventilation fans automatically ____________ due to Blackout signal.

The CRDM air handling units cool the air ____________ the CRDM shroud.

A. start
leaving

B. start
entering

C. stop
leaving

D. stop
entering

Answer: A

K/A Match:

This question matches the K/A by requiring demonstration of knowledge of Containment Cooling System design and Safety Injection and Blackout stop/start functions.

Explanation:

A. Correct. CRDM fans automatically are started by the Blackout sequencer and the design is to cool the air leaving the CRDM shroud lowering Containment air temperature. This lowers average Containment air temperature which is drawn into the CRDM shroud.

B. Incorrect. Plausible because the fans automatically start during a Blackout however system design cools the air leaving the CRDM shroud not entering.

C. Incorrect. Plausible because the fans automatically stop on a Safety Injection and the air is cooled leaving the CRDM shroud.

D. Incorrect. Plausible because the fans automatically stop on a Safety Injection however the air is cooled leaving the CRDM shroud.

Technical Reference(s)	Containment Ventilation Study Guide	Attached w/ Revision # See Comments / Reference
	ALM-0031A	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSCL1OB103 Explain the normal, abnormal and emergency operation of the Containment Ventilation 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 ____
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 7
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	026 K2.02		
	Importance Rating	2.7		

Containment Spray: Knowledge of bus power supplies to the following: MOVs

Question # 15

Containment Spray has actuated on Unit 2

Which energized power supplies are required for Chemical Additive Tank contents to be supplied to BOTH trains of Containment Spray?

A. 2ED1-1 and 2ED2-1

B. 2EB1-1 and 2EB2-1

C. 2ED1-1 and 2EB2-1

D. 2EB1-1 and 2ED2-1

Answer: B

K/A Match:

The question matches the K/A as it requires a demonstration of knowledge of the bus power supplies of important MOVs in the Containment Spray System.

Explanation:

A. Incorrect. Incorrect but plausible because 2ED1-1 and 2ED2-1 are the power supplies to 2-LV-4752 and 2-LV-4753 which are the AOVs in series with the MOVs as described in 'B' below. This answer is plausible if thought that the MOVs were the normally open valves and that power was necessary to the AOVs which must open. This answer is additionally incorrect in that a loss of power to the AOVs results in a failure open of the valves thus having power to the AOVs is not necessary to satisfy the given conditions in the stem.

B. Correct. The Chemical Additive Tank to eductor valves are in series valves powered by opposite train power. The normally open valves in series are AOVs and the normally closed valves which must open upon a Containment Spray signal are MOVs. The stated power supplies are for the MOVs on Train A and B respectively for valves 2-LV-4754 and 2-LV-4755 which must each have power an open in order to supply Chemical addition to the eductors of the Containment Spray Pumps.

C. Incorrect. Incorrect but plausible if believed that the in series valves were different between Train A and Train B to ensure diversity exists in supplying the Chemical Additive Tank contents to containment via one train of Containment Spray. However, this is not the design of the system as the MOVs and AOVs perform the same function on both trains.

D. Incorrect. Incorrect but plausible as described in 'C' above with the opposite configuration.

Technical Reference(s)	E2-0007, Sh. - & Sh. B	Attached w/ Revision # See Comments / Reference
	E2-0020, Sh. B & Sh. G	
	Containment Spray Big Book	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSCT1OB102 Describe the basic design and flowpath of the 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 X
Comprehension or Analysis ______
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 7
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	039 K5.08		
	Importance Rating	3.6		

Main and Reheat Steam: Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity

Question # 16

Unit 1 is at 100% when a steam generator Atmospheric Relief Valve (ARV) fails OPEN.

The ARV opening adds __________ reactivity to the core.

The net reactivity effect is greater at _________ of Core Life.

A. negative
End

B. negative
Beginning

C. positive
End

D. positive
Beginning

Answer: C

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate an understanding of how increased steam demand by the main steam system will affect reactivity and how it changes with core life.

Explanation:

A. Incorrect. Plausible if thought that steam demand adds negative reactivity to the core. Reactivity effects at EOL are greater due to a larger MTC.

B. Incorrect. Plausible if thought that steam demand adds negative reactivity to the core. Also plausible if thought that reactivity effects at BOL are greater.

C. Correct. Steam demand adds positive reactivity to the core and reactivity effects at EOL are greater due to larger MTC.

D. Incorrect. Plausible because steam demand adds positive reactivity to the core. Also plausible if thought that reactivity effects at BOL are greater.

Technical Reference(s)	Increased Heat Removal Accidents Study Guide	Attached w/ Revision # See Comments / Reference
	LO21GFRCOF	

Proposed references to be provided during examination: None

Learning Objective: LO21MCOTA8OB102, Discuss the excessive increase in secondary steam flow transient

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

Level of Difficulty 2

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	059 A3.02		
	Importance Rating	2.9		

Main Feedwater: Ability to monitor automatic operation of the MFW, including: Programmed levels of the S/G

Question # 17

- Unit 2 Reactor power = 60% and stable
- SG NR levels = 65% increasing slowly

Based on above conditions, SG levels are __________________.

To return steam generator levels to setpoint feedwater regulating valve controller demand will automatically _______.

A. moving closer to setpoint
decrease

B. moving closer to setpoint
increase

C. moving away from setpoint
decrease

D. moving away from setpoint
increase

Answer: C

K/A Match:

This question matches the KA by requiring knowledge of the feedwater control, SG program levels and the differences between Unit 1 and 2.

Explanation:

A. Incorrect. 1st part is incorrect because the program level setpoint for Unit 2 is 64%. It is plausible because if it were Unit 1 with a program level setpoint of 67%, it would be correct. 2nd part is correct because with steam generator level above program level and the FRVs controller demand will decrease.

B. Incorrect. 1st part is incorrect but plausible as described in 'A' above. 2nd part is incorrect because steam generator level above program level the FRVs controller demand will decrease. See D below.

C. Correct. The first part is correct as the program level setpoint for Unit 2 is 64%, the current level and trend is moving away from setpoint. 2nd part is correct because with steam generator level above program level the FRVs controller demand will decrease to allow the valve to close and return level back to program.

D. Incorrect. The first part is correct as the program level setpoint for Unit 2 is 64%, the current level and trend is moving farther away from setpoint. 2nd part is incorrect but plausible because steam generator level is above program level and the FRVs controller demand would need to decrease to lower level not increase.

Technical Reference(s)	Main Feedwater Study Guide	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21SYSMR1OB105 Explain the normal, abnormal and emergency operation of the Main Steam System.

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

Question History: Last NRC Exam ______

Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis X
- Level of Difficulty 3

10 CFR Part 55 Content:
- 55.41 7
- 55.43 ______

Comments / Reference: Original Question	Revision:

Unit 2 plant conditions:

- Reactor power = 60%
- SG NR levels = 65% increasing slowly

Which of the following correctly completes the statements?

1) Based on the above conditions, SG levels are ____(1)____.

2) A Steam line break at this power level would result in a ____(2)____ cool down than the same break at 100% power.

A. (1) moving closer to their setpoint
(2) larger

B. (1) moving closer to their setpoint
(2) smaller

C. (1) moving farther away from their setpoint
(2) larger

D. (1) moving farther away from their setpoint
(2) smaller

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	061 A2.06		
	Importance Rating	2.7		

Auxiliary/Emergency Feedwater: Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Back leakage of MFW

Question # 18

- Unit 1 Reactor Power is 18% with power escalation in progress per IPO-003A, Power Operations
- 1-ALB-8A, Window 3.5, SG 3 AFW NZL TEMP HI is in alarm
- 1-TI-2473A, SG 3 MDAFW TEMP is reading 265°F and slowly rising

Per ABN-305, Auxiliary Feedwater System Malfunction …

If 1-TI-2473A, SG 3 MDAFW TEMP cannot be lowered to less than 250°F then ________ must be started and throttled to 25-35 gpm to lower temperature.

While the associated MDAFWP is running and flow throttled, it ________ OPERABLE

A. MDAFWP 1-01
is

B. MDAFWP 1-02
is

C. MDAFWP 1-01
is NOT

D. MDAFWP 1-02
is NOT

Answer: D

K/A Match:

This question matches the K/A as it requires demonstration of knowledge of the impact of AFW system back leakage and how to mitigate the consequences of back leakage using procedures.

Explanation:

A. Incorrect. First part is incorrect but plausible because the back leakage is occurring on Steam Generator 1-03, which is a Train A component that is commonly mistaken to be supplied by the Train A MDAFWP (1-01). However, Steam Generator 1-03 is fed by the Train B MDAFWP (1-02). The second part is incorrect but plausible because the MDAFWP is currently available and capable of feeding the Steam Generators, however, when Reactor Power is >10% the MDAFWP Flow Control Valves are required to full open in order for the pump to be OPERABLE.

B. Incorrect. First part is correct, Steam Generator 1-03 is supplied by MDAFWP 1-02. Second part is incorrect but plausible, see ‘A’ above.

C. Incorrect. First part is incorrect but plausible, see ‘A’ above. Second part is correct, when Reactor Power is >10% the MDAFWP Flow Control Valves are required to be full open for the MDAFWP to be OPERABLE. In this case the valves are throttled to achieve a flow rate of 25-35 gpm to cool the associated piping to ambient conditions with water from the CST.

D. Correct. First part is correct, see ‘B’ above. Second part is correct, see ‘C’ above.

Technical Reference(s)	Auxiliary Feedwater Study Guide	Attached w/ Revision # See Comments / Reference
	Auxiliary Feedwater System Fig 1	
	ABN-305	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSAF1OB105 Explain the normal, abnormal and emergency operation of the Auxiliary Feedwater 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
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 7
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	062 A4.03		
	Importance Rating	2.8		

AC Electrical Distribution: Ability to manually operate and/or monitor in the control room: Synchroscope, including an understanding of running and incoming voltages

Question # 19

- Transformer XST2 is supplying 1EA1 6900 VAC Bus
- Emergency Diesel Generator 1-01 has been started per SOP-609A, Diesel Generator System

Which of the following describes the actions required to close Emergency Diesel Generator 1-01 output breaker 1EG1?

The synchroscope is selected to ON, Diesel Generator voltage is adjusted to ________ 1EA1 Voltage. Diesel Generator speed is adjusted to rotate slowly in the ________ direction.

A. slightly higher than
fast

B. slightly higher than
slow

C. match
fast

D. match
slow

Answer: A

K/A Match:

The question matches the K/A as it requires the operator to demonstrate knowledge of the ability to manipulate the AC Electrical Distribution System including how Diesel Generator voltage and speed must be adjusted when the Diesel is paralleled to the grid.

Explanation:

A. Correct. The Diesel Generator is being paralleled to the grid so diesel voltage is adjusted to slightly higher (1 to 2 volts) than grid voltage and Diesel Generator frequency is adjusted so that the synchroscope is moving slowly (2 to 4 RPM) in the Fast direction to ensure the diesel generator synchronizes to the grid.

B. Incorrect. Plausible because the Diesel Generator is being paralleled to the grid. Second part is plausible because if transferring from the diesel generator to the grid the synchroscope is adjusted to move in the Slow direction.

C. Incorrect. Plausible because if restoring the bus to the grid the voltages are matched. To parallel the diesel generator to the grid the synchroscope is adjusted to move in the Fast direction.

D. Incorrect. Plausible because if restoring the bus to the grid the voltages are matched. Second part is plausible because if transferring from the diesel generator to the grid the synchroscope is adjusted to move in the Slow direction.

Technical Reference(s)	SOP-609A, Section 5.2	Attached w/ Revision # See Comments / Reference
	SOP-609A, Section 5.7	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSED1OB123 Explain the normal, abnormal and emergency operation of the Emergency Diesel Generator 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
- Level of Difficulty 3

10 CFR Part 55 Content:
- 55.41 8
- 55.43 _____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	063 G.2.2.44		
	Importance Rating	4.2		

DC Electrical Distribution: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question # 20

- Unit 1 at 100%
- 1-ALB-10B, Window 1.14, BATT CHRG BC1ED1-1/BC1ED1-2 TRBL is LIT with the following indications:

Battery 1ED1 ________ supplying DC loads.

Per ALM-0102A, Alarm Procedure 1-ALB-10B________ should be placed in service.

A. is
Switchboard 1ED1

B. is
Battery 1ED1 standby battery charger

C. is NOT
Switchboard 1ED1

D. is NOT
Battery 1ED1 standby battery charger

Answer: B	

K/A Match:

The question matches the K/A as it demonstrates the ability of the operator to interpret DC indications and select the proper action in response to those indications.

Explanation:

A. Incorrect. First part is correct see B below. Second part is incorrect but plausible since it could be thought that the switchboard should be placed in service based on indicated current and voltage.

B. Correct. Based on battery current and voltage indications battery 1ED1 is supplying DC loads. With the current meter showing discharge and not charge this indicates the battery is supplying the switchboard. If the charger was in service current would show a charge not discharge of the battery. Switchboard 1ED1 is in service or there would be no indicated current, so the standby battery charger should be placed in service per the ALM to restore the battery and carry bus loads.

C. Incorrect. First part is incorrect but plausible because if the operator does not understand how the charge / discharge meter works they could mistake the reading for the charger supplying the battery. Second part is incorrect see A above.

D. Incorrect. First part is incorrect see C above. Second part is correct see B above.

Technical Reference(s)	SOP-605A	Attached w/ Revision # See Comments / Reference
	ALM-0102A	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSDC1OB008 Comprehend the normal, abnormal and emergency operation of the DC Electrical Distribution 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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 7
55.43 ___

Comments / Reference: SOP-605A, Section 5.1.2	Revision: 12

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	064 K3.01		
	Importance Rating	3.8		

Emergency Diesel Generator: Knowledge of the effect that a loss or malfunction of the ED/G System will have on the following: Systems controlled by automatic loader.

Question # 21

- OPT-214A, Diesel Generator Operability Test Section 8.1, Train A Diesel Generator Monthly Operability Test is being performed.
- Emergency Diesel Generator (EDG) 1-01 is Slow Started by a Remote Normal Start.
- EDG 1-01 trips on Diesel Generator Lube Oil Header Pressure Low.
- A Loss of All Off-Site Power occurs.

EDG 1-01 _______________.

Loads are ____________________ onto Bus 1EA1.

A. requires normal start
automatically sequenced

B. immediately starts
manually loaded

C. immediately starts
automatically sequenced

D. requires normal start
manually loaded

Answer: C

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate knowledge of the EDG following a malfunction and the subsequent effect upon the automatic loader.

Explanation:

A. Incorrect. First part is incorrect but plausible in that if the applicant believed that the Lube Oil Header Pressure Low trip or the Fast/Slow Start Switch in the Slow position would prevent the EDG from starting as a result of the Blackout Signal. However, the Lube Oil Header Pressure Low trip is bypassed by an Emergency Start and the EDG will start at a slower rate in the Slow position. Second part is correct see C below.

B. Incorrect. First part is correct see C below. Second part is incorrect but plausible if thought the EDG will respond normally to an Emergency Start. However, with the Fast/Slow switch in the Slow position the EDG will not reach rated speed and voltage in less than 10 seconds and the sequencer would not respond normally based on that.

C. Correct. In accordance with OPT-214A, with the EDG in the slow position the EDG will respond to a Blackout signal but not reach rated speed and voltage within 10 seconds. Further the Lube Oil Header Pressure Low trip is bypassed with an Emergency Start signal which a Blackout start signal is so the EDG will start and load per the Blackout Sequencer only at a delayed time as the EDG will not obtain rated voltage within the 10 second time required for full Operability.

D. Incorrect. First part is incorrect see A above. Second part is incorrect see B above.

Technical Reference(s)	OPT-214A	Attached w/ Revision # See Comments / Reference
	Emergency Diesel Generators Study Guide	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSED1OB123, Explain the normal, abnormal and emergency operation of the Emergency Diesel Generator 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
Level of Difficulty 4

10 CFR Part 55 Content: 55.41 7
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	064 K6.07		
	Importance Rating	2.7		

Emergency Diesel Generator: Knowledge of the effect of a loss or malfunction of the following will have on the ED/G system: Air receivers

Question # 22

Which of the following describes the effect of allowing BOTH the Diesel Generator Starting Air Receivers to drop to 145 psig?

The Emergency Diesel Generator Engine Start Circuit will accept a ________ Start signal.

A. Local Emergency

B. Safety Injection

C. Bus Undervoltage

D. Manual Normal

Answer: D

K/A Match:

The question matches the K/A as it requires the operator to demonstrate knowledge of a loss of air in the Diesel Generator Starting Air Receivers and how that affects the Diesel Generator start capability.

Explanation:

A. Incorrect. Plausible because it would seem that a Local Emergency Start Signal should attempt to start the EDG without regard to starting air receiver pressure as the concern for multiple automatic start attempts is eliminated. When receiver pressure is below 150 psig, only a Manual Normal Start signal will be accepted.

B. Incorrect. Plausible because it would seem that an Emergency Start Signal should attempt to start the EDG without regard to starting air receiver pressure, however, automatic emergency starts would continue to lower the starting air receiver pressure so this is not the case. When receiver pressure is below 150 psig, only a Manual Normal Start signal will be accepted.

C. Incorrect. Plausible because it would seem that a Bus Undervoltage Start Signal would be the highest priority start of the EDG and should occur without regard to starting air receiver pressure, however, automatic emergency starts would continue to lower the starting air receiver pressure so this is not the case. When receiver pressure is below 150 psig, only a Manual Normal Start signal will be accepted.

D. Correct. When receiver pressure is below 150 psig, only a Manual Normal Start signal will be accepted.

Technical Reference(s)	SOP-609A, Section 4.2 & 5.2	Attached w/ Revision # See Comments / Reference
	Emergency Diesel Generator Study Guide	
	Pgs. 34 & 75 of 109	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSED1OB111 Explain the normal, abnormal and emergency operation of the EDG Starting Air system.

Question Source: Bank # ILOT8093
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 8
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	073 A2.02		
	Importance Rating	2.7		

Process Radiation Monitoring: Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure

Question # 23

- A Gaseous Decay Tank release is in progress.
- An OPERATE FAILURE - CHANNEL OPERATE FAILURE Digital Radiation Monitor System alarm is received for the Plant Vent Stack WRGM Channel RE-5570A during the release.

______ alarm color indicates an OPERATE FAILURE.

What action is required?

A. Blue
Verify X-HCV-0014, GWPS DISCH PLT EXH PLNM ISOL VLV, automatically closes

B. Blue
Auto function of X-HCV-0014, GWPS DISCH PLT EXH PLNM ISOL VLV, is disabled and the valve must be manually closed

C. Red
Verify X-HCV-0014, GWPS DISCH PLT EXH PLNM ISOL VLV, automatically closes

D. Red
Auto function of X-HCV-0014, GWPS DISCH PLT EXH PLNM ISOL VLV, is disabled and the valve must be manually closed

Answer: A

K/A Match:

The question matches the K/A as it requires the operator to demonstrate knowledge of the impact of a detector failure in the Process Radiation Monitoring System and determine the correct action for the failure.

Explanation:

A. Correct. An operate failure indicates blue, and the valve will automatically close on an operate failure.

B. Incorrect. Plausible as the color indication is correct, but the valve will automatically close on an operate failure.

C. Incorrect. Plausible since this is a valid valve response, but the red alarm color is for high radiation level.

D. Incorrect. Plausible since the red color is used in the scheme, but it is for high radiation level.

Technical Reference(s)	ALM-3200, Attachment 3	Attached w/ Revision # See Comments / Reference
	SOP-706, Attachment 1	
	DRMS Study Guide Pg. 32 of 56	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSRM1OB104 Explain the instrumentation and controls of the Digital Radiation Monitoring System and predict the system response.

Question Source: Bank # ILOT8094
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 11
55.43 ________

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	061 K6.02		
	Importance Rating	2.6		

Auxiliary/Emergency Feedwater: Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps

Question # 24

- Unit 1 Reactor power is 45%
- The running Main Feedwater pump trips
- Motor Driven Auxiliary Feedwater Pump 1-01 fails to start

Turbine Driven Auxiliary Feedwater Pump will automatically start ________.

A single MDAFWP flow shall be limited to a MAXIMUM of ________ gpm when feeding all four SGs with both trains cross-connected.

A. when ONLY ONE Steam Generator NR level reaches its LOW-LOW setpoint
700

B. when ONLY ONE Steam Generator NR level reaches its LOW-LOW setpoint
800

C. when a MINIMUM of TWO Steam Generator NR levels reach their LOW-LOW setpoint
700

D. when a MINIMUM of TWO Steam Generator NR levels reach their LOW-LOW setpoint
800

Answer: D

K/A Match:

This question matches the KA by requiring knowledge of how a failed AFW pump will impact operation of the remaining components.

Explanation:

A. Incorrect. First part is incorrect because for the Turbine Driven AFW pump, the start setpoint is 2/4 SGs at the LOW-LOW setpoint. It is plausible because the auto start for the Motor Driven AFW pumps occur when 1 SG is at the LOW-LOW setpoint. 2nd part is incorrect because flow is limited to 800 gpm. It is plausible because the orifice installed downstream of each Feed Regulating Valve is designed to limit flow to 700 gpm to preclude run-out conditions.

B. Incorrect. First part is incorrect but plausible (see A). Second part is correct. When cross-connected, flow is limited to 800 gpm to prevent a run-out condition.

C. Incorrect. First part is correct. The Auto-Start setpoint for the Turbine Driven AFW pump is LOW-LOW on 2/4 SGs. Second part is incorrect but plausible (see A).

D. Correct. First part is correct (see C). Second part is correct (see B).

Technical Reference(s)	ABN-305	Attached w/ Revision # See Comments / Reference
	Auxiliary Feedwater Study Guide	
	AFW System Figures 1 & 2	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSAF1OB105, Explain the normal, abnormal and emergency operation of the Auxiliary Feedwater System.

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

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X
Level of Difficulty 3

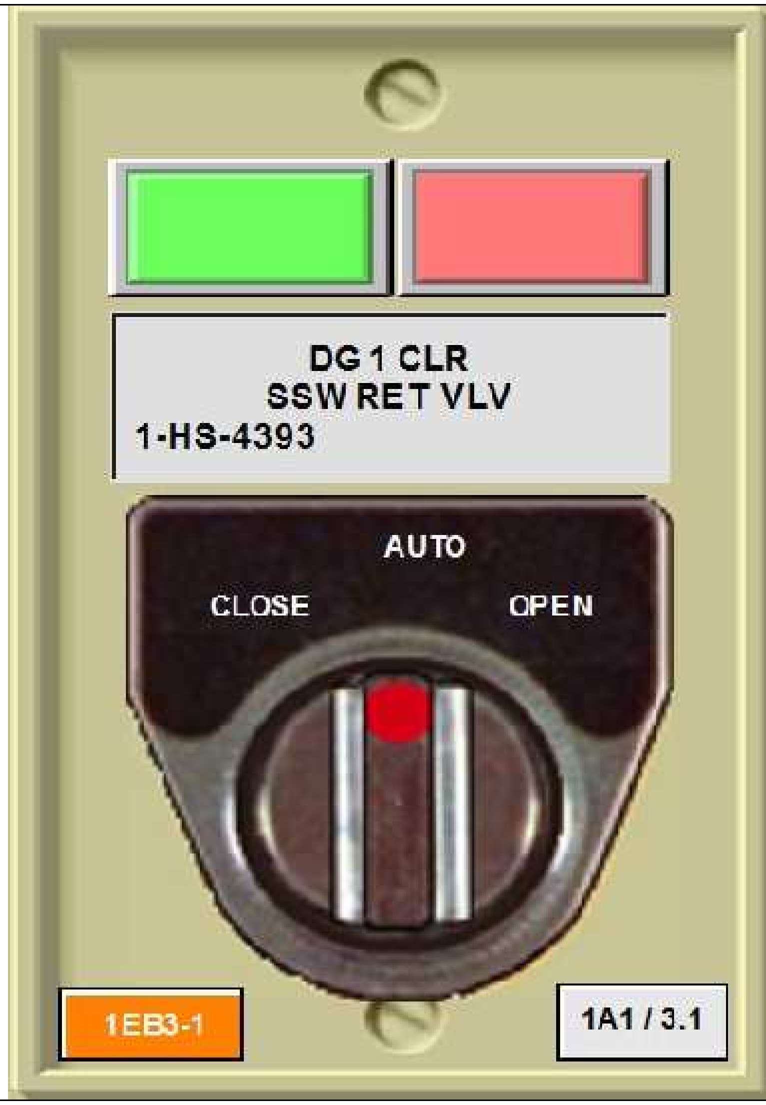
10 CFR Part 55 Content: 55.41 7
55.43

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	076 A4.02		
	Importance Rating	2.6		

Service Water: Ability to manually operate and/or monitor in the control room: SWS valves

Question # 25

- Unit 1 was at 100%
- SSWP 1-01 in AUTO shutdown
- Reactor Trip and Safety Injection occurred
- 1-HS-4393, DG 1 CLR SSW RET VLV indicated as shown during SI actuation



Before the SI was actuated, 1-HS-4393, DG 1 CLR SSW RET VLV was indicating ________.
After SI sequencer times out, 1-HS-4393, DG 1 CLR SSW RET VLV will indicate _________.

A. GREEN light ON, RED light OFF
GREEN light OFF, RED light ON

B. GREEN light OFF, RED light ON
GREEN light ON, RED light OFF

C. GREEN light ON, RED light OFF
GREEN light ON, RED light OFF

D. GREEN light OFF, RED light ON
GREEN light OFF, RED light ON

Answer: A	

K/A Match:

This question matches the K/A as it requires the applicant to demonstrate knowledge of SSW system valve indications in the control room during SI (DG emergency Start).

Explanation:

A. Correct. First part is correct because the valve had to be closed prior to the SI or it would not be stroking. Second part is correct because the SI initiated an emergency start of the EDG which sends a signal to open the valve to provide EDG cooling.

B. Incorrect. First part is incorrect but plausible because the normal position for the valve would be GREEN light OFF and RED light ON. Second part is incorrect but plausible if the applicant believes that because SSWP 1-01 is shutdown that it would inhibit 1-HV-4393 from opening.

C. Incorrect. First part is correct (See A above). Second part is incorrect but plausible (See B above).

D. Incorrect. First part is incorrect but plausible (See B above). Second part is correct (See A above).

Technical Reference(s)	SSW Study Guide	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21SYSSW1OB103 Explain the components of the Station Service Water system including interrelations with other systems to include interlocks and control loops.

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 10
55.43

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	078 K2.01		
	Importance Rating	2.7		

Instrument Air: Knowledge of bus power supplies to the following instrument air compressor

Question # 26

________ is the power supply to Instrument Air Compressor 1-02.

A. 1EB3-1

B. 1EB4-1

C. 1EB3

D. 1EB4

Answer: D

K/A Match:

This question matches the K/A as it requires demonstration of knowledge of the power supply to IAC 1-02.

Explanation:

A. Incorrect. Plausible because this is the control power supply to 1-01 Instrument Air Dryer, however, the power supply to Instrument Air Compressor 1-02 is 1EB4.

B. Incorrect. Plausible because this is the control power supply to 1-02 Instrument Air Dryer, however, the power supply to Instrument Air Compressor 1-02 is 1EB4.

C. Incorrect. Plausible because this is the power supply to Instrument Air Compressor 1-01.

D. Correct. This is the power supply to Instrument Air Compressor 1-01.

Technical Reference(s)	SOP-509A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21SYSIA1OB103 Describe the components of the Instrument Air System including interrelations with other systems to include interlocks and control loops.

Question Source:	Bank #		
	Modified Bank #	X	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	2013	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
	Level of Difficulty	2	
10 CFR Part 55 Content:	55.41	7	
	55.43		

Original Question:
What is the power supply for Instrument Air Compressor 1-01? A. 1EB3-1 B. 1EB4-1 C. 1EB3 D. 1EB4 Proposed Answer: C

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	078 A3.01		
	Importance Rating	3.1		

Instrument Air: Ability to monitor automatic operation of the IAS, including: Air pressure

Question # 27

- Per OWI-409, Equipment Rotation Program, Instrument Air Compressor (IAC) 2-01 was placed in LEAD
- Unit 2 BOP reports that 2-PI-3488, INST AIR AFTFILT OUT PRESS has been cycling between 105 psig and 115 psig approximately every 12 minutes
- The BOP states that 2-PI-3490, CNTMT INSTR AIR HDR PRESS has tracked consistently with 2-PI-3488

Per SOP-509B…

Instrument Air System, IAC 2-01 ________ loading and unloading in the proper pressure range.

When IAC 2-01 was placed in LEAD the 'number of starts per day' was set to ________.

A. is
72

B. is NOT
72

C. is
0

D. is NOT
0

Answer: C

K/A Match:

The question matches the K/A as it requires the operator to demonstrate knowledge of the automatic operation of the Instrument Air System including air pressure at which the Lead IAC would cycle on and off if its controller was not properly set during equipment rotation and further requires the applicant to demonstrate knowledge of proper loading and unloading of the compressors to control air pressure.

Explanation:

A. Incorrect. Part 1 is correct as described in 'C' below. Part 2 is incorrect but plausible in that the Standby IAC will be set at 72 starts per day to conserve starting duty. As the compressors were just swapped from STANDBY to LEAD not have the proper setting on the number of starts per day is plausible.

B. Incorrect. Part 1 is incorrect but plausible as the STANDBY compressor would cycle pressure between 100 and 115 if required. Part 2 is incorrect but plausible as described in 'A' above.

C. Correct. Part 1 is correct as the LEAD compressor should load and unload between 105 psig and 115 psig. However, the Unit 2 IACs are normally set to not start and stop the compressor motors but to run continuously. Part 2 is correct in that when the number of starts per day is set to '0' the compressor will not cycle on and off but load and unload as necessary to control pressure.

D. Incorrect. Part 1 is incorrect but plausible as described in 'B' above. Part 2 is correct as described in 'C' above.

Technical Reference(s)	SOP-509B	Attached w/ Revision # See Comments / Reference
	OWI-409	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSIA1OB105 Explain the normal, abnormal and emergency operation of the Instrument Air 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
- Level of Difficulty 3

10 CFR Part 55 Content:
- 55.41 7
- 55.43 ________

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	1		
	K/A	103 K3.01		
	Importance Rating	3.3		

Containment: Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under shutdown conditions

Question # 28

- Unit 2 is draining the RCS in MODE 5
- Per IPO-010B, Reactor Coolant System Reduced Inventory Operations, entry is allowed into Reduced Inventory with containment penetrations impaired and not sealed.

A MAXIMUM of ______ penetrations are allowed to be impaired and not sealed when reducing RCS level below a MINIMUM of ______ inches above core plate.

A. 5
80

B. 5
120

C. 10
80

D. 10
120

Answer: C

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge of the effect that a containment malfunction will have on shutdown containment integrity which would restrict the plant from reduced inventory operations.

Explanation:

A. Incorrect. Plausible as 5 penetrations is a more conservative allowance than 10. The reduced inventory level is correct.

B. Incorrect. Plausible as 5 penetrations is a more conservative allowance than 10. The inventory level of 120 inches is used in the procedure for a drain down plateau which can either be established or passed through if continuing to reduced inventory operations.

C. Correct. In accordance with IPO-010B, a maximum of 10 containment penetrations can be impaired and not sealed. Reduced inventory is defined as less than 80 inches above core plate.

D. Incorrect. Plausible as the number of penetrations is correct as discussed in ‘C’ above. The level is incorrect but plausible as discussed in ‘B’ above.

Technical Reference(s)	IPO-010B	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21IPO010OB104 Discuss the Precautions, Limitations and Attachments of IPO-010, “RCS Reduced Inventory Operations.”

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:
- Level of Difficulty: 3

10 CFR Part 55 Content:
- 55.41: 10
- 55.43:

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	2		
	K/A	011 G.2.1.23		
	Importance Rating	4.3		

Pressurizer Level Control: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Question # 29

IPO-005B, Plant Cooldown from Hot Standby to Cold Shutdown is in progress.

Pressurizer level must be maintained less than _________ until SI is blocked.

Cold calibrated Pressurizer level instrument should be used for indication after ___________ temperature is below 450°F.

A. 30%
RCS Hot Leg

B. 30%
Pressurizer

C. 70%
RCS Hot Leg

D. 70%
Pressurizer

Answer: B

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate the Pressurizer Level operational control band and instrument used in performing a cooldown of the RCS.

Explanation:

A. Incorrect. The first part is correct as described in 'B' below. The second part is incorrect but plausible as MODE changes in IPO-005B are based on Hot Leg temperatures but the use of the cold calibrated pressurizer level instrument is based on pressurizer temperature.

B. Correct. In accordance with IPO-005B, pressurizer level must be maintained less than 30% until Safety Injection is blocked. Use of the Cold Calibrated Pressurizer Level instrument is based on pressurizer temperature.

C. Incorrect. The first part is incorrect but plausible as the control band rises to 50% to 70% after Safety Injection is blocked but must be maintained less than 30% until this point. The second part as described in 'A' above.

D. Incorrect. The first part as described in 'C' above. The second part is correct as described in 'B' above.

Technical Reference(s)	IPO-005B	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21IPO005OB102, Discuss the actions for conducting a cooldown from MODE 3 to MODE 5 in accordance with IPO-005, Plant Cooldown for Hot Standby to Cold Shutdown.

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 ____
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 10
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	2		
	K/A	015 A1.02		
	Importance Rating	3.5		

Nuclear Instrumentation: Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the NIS controls including: SUR

Question # 30

- A Reactor startup is in progress on Unit 1
- The reactor is critical at 10^{-8} amps
- The Unit Supervisor has directed raising reactor power to approximately 2%

Per IPO-002A, Plant Start From Hot Standby a startup rate of ______ is established using control rods to raise power to approximately 2%.

A. 0.15 dpm

B. 0.2 dpm

C. 0.5 dpm

D. 1.0 dpm

Answer: C

K/A Match:

The question is a K/A match as it requires the applicant to process known procedural requirements that are followed to prevent an inadvertent SR reactor trip when raising power to approximately 2%.

Explanation:

A. Incorrect. Plausible because if diluting to criticality the following applies; The Source Range and Intermediate Range overlap region should be passed through quickly with a steady positive startup rate. Experience has shown that all Control Rods at FOP and dilution secured will result in an approximate 0.15 DPM startup rate. Expeditiously proceeding through this region should prevent inadvertent Source Range Reactor Trip caused by Intermediate Range perturbations around the P-6 setpoint.

B. Incorrect. Plausible because as power approaches $3X10^{-6}$ amps startup rate is reduced to approximately 0.2 DPM.

C. Correct. During a control rod startup the following applies; The Source Range and Intermediate Range overlap region should be passed through quickly with a steady startup rate of approximately 0.5 DPM. Expeditiously proceeding through this region should prevent inadvertent Source Range Reactor Trip caused by Intermediate Range perturbations around the P-6 setpoint.

D. Incorrect.

Technical Reference(s)	IPO-002A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: ______

Learning Objective: LO21IP002OB103, Discuss the actions to perform a reactor startup per IPO-002, Plant Startup from Hot Standby.

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 ______
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 1
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	2		
	K/A	017 K5.02		
	Importance Rating	3.7		

In-Core Temperature Monitor: Knowledge of the operational implications of the following concepts as they apply to the ITM system: Saturation and subcooling of water

Question # 31

- A Small Break LOCA has occurred
- RCPs have been tripped
- 11 and 22 inch above core plate RVLIS lights are LIT
- ECCS is injecting
- RCS pressure is 1385 psig
- All Hot Leg temperatures are 587ºF
- All CETs are between 570ºF and 575ºF

The condition of the fluid at the core exit is ________.

A. saturated liquid

B. superheated steam

C. saturated steam

D. subcooled liquid

Answer: D

K/A Match:

The question matches the K/A as it requires the operator to demonstrate knowledge of the Core Cooling Monitor System and determine the state of the core coolant based on the conditions provided.

Explanation:

A. Incorrect. Plausible if calculation is performed using hot leg temperature of 587°F and quality at 0.

B. Incorrect. Plausible if thought that the saturated conditions in the hot leg were indicative of superheated conditions at the hotter core outlet.

C. Incorrect. Plausible if thought that the saturated condition in the hot legs were indicative of saturated conditions at the core outlet; however, as the fluid level is below the hot legs, only steam is being carried over to the hot leg RTDs.

D. Correct. The Core Exit Thermocouples indicate 12°F to 17°F subcooled.

Technical Reference(s)	Steam Tables	Attached w/ Revision # See Comments / Reference
	LO21.SYS.RC3 Study Guide	

Proposed references to be provided during examination: Steam Tables

Learning Objective: LO21MCOTAAOB100 Upon completion of this lesson the student shall be able to discuss how various loss of coolant accidents can affect core cooling mechanisms and reduce the margin to thermal limits and, in extreme cases, can result in the release of radioactivity to the environment.

Question Source: Bank # ILOT1634
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 8
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	2		
	K/A	028 A4.01		
	Importance Rating	4.0		

Hydrogen Recombiner and Purge Control: Ability to manually operate and/or monitor in the control room: HRPS controls

Question # 32

- A LOCA has occurred on Unit 2
- Containment Pressure is 2.8 psig
- The Emergency Coordinator has authorized use of the Hydrogen Purge Supply and Exhaust System for performance of a Hydrogen dilution per SOP-205, Hydrogen Purge Supply and Exhaust System

Start Hydrogen Purge Exhaust Supply Fans at _________.

If a Unit 1 Safety Injection were to occur during the Unit 2 Hydrogen dilution, the Hydrogen Purge Exhaust and Supply Fans would ________.

A. 1-CB-03
continue to run

B. 2-CB-03
continue to run

C. 1-CB-03
trip

D. 2-CB-03
trip

Answer: C

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate the manual operations and expected response during hydrogen purge supply and exhaust system operation. The applicant is further required to demonstrate an understanding of the operational implications during a Hydrogen Purge if a Safety Injection were to occur on the opposite unit.

Explanation:

A. Incorrect. The first part is correct as described in ‘C’ below. The second part is incorrect but plausible if believed that a Safety Injection did not generate a Containment Ventilation Isolation for the opposite unit. However, a Containment Ventilation Isolation signal on either unit trips the Hydrogen Purge Supply and Exhaust Fans.

B. Incorrect. The first part is incorrect but plausible as the Unit 2 dampers and valves for the Hydrogen Purge System are operated from Unit 2 control boards, however, the Supply and Exhaust Fans which are common are only operated from the Unit 1 Control Board. The second part is incorrect but plausible as described in ‘A’ above.

C. Correct. The first part is correct as the controls for the Hydrogen Purge Supply and Exhaust Fans are only on the Unit 1 Control Board. The second part is correct as a Safety Injection signal on the opposite unit initiates a Containment Ventilation Signal which trips the Hydrogen Purge Supply and Exhaust Fans.

D. Incorrect. The first part is incorrect but plausible as described in ‘B’ above. The second part is correct as described in ‘C’ above.

Technical Reference(s)	SOP-205	Attached w/ Revision # See Comments / Reference
	Containment Ventilation Study Guide	
	E1-0059 Sht. 023	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSOB103, Explain the normal, abnormal and emergency operation of the containment ventilation 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 ______
Level of Difficulty 4

10 CFR Part 55 Content: 55.41 10
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	2		
	K/A	035 K6.01		
	Importance Rating	3.2		

Steam Generator: Knowledge of the effect of a loss or malfunction on the following will have on the S/GS: MSIVs

Question # 33

- Unit 1 is at 100% power
- **SG 1-01** MSIV closes spuriously
- Automatic Reactor Trip fails

Prior to control systems attempting to mitigate the transient...

SG 1-02 Narrow Range level will ___________________.

RCS T_{AVE} will _______________.

A. swell
remain the same

B. swell
rise

C. shrink
remain the same

D. shrink
rise

Answer: B

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge of the effect a spurious MSIV closure will have on the associated Steam Generator and RCS Loop as well as what action is required.

Explanation:

A. Incorrect. First part is correct, see B below. Second part is incorrect Tave will not remain the same since this is a heat rejection back to the RCS. Plausible as the student could confuse this loss of heat removal from one loop and the gaining of heat removal from the other loops as a balance of heat removal.

B. Correct. First part initially when the MSIV closes the AFFECTED SG pressure rises and experiences shrink while the UNAFFECTED SGs experience swell due to a rapid increase in steam demand (as well as pressure lowering). Second part with the loss of heat sink from 1 of 4 loops the Tave in the affected loop will rise which inputs into overall Tave causing it to rise as well.

C. Incorrect. First part is incorrect but plausible since the rise in steam demand on the remaining 3 SGs could be thought as supplying more steam so water level would initially lower or shrink. Second part is incorrect see A above.

D. Incorrect. First part is incorrect see C above. Second part is correct, see B above.

Technical Reference(s)	LO21SYSMR1	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21SYSMR1OB104, EXPLAIN the instrumentation and controls of the Main Steam system and PREDICT the system response.

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 5
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	2		
	K/A	041 K3.02		
	Importance Rating	3.8		

Steam Dump/Turbine Bypass Control: Knowledge of the effect that a loss or malfunction of the SDS will have on the following: RCS

Question # 34

- The reactor is critical at 10^{-8} amps during a startup near the end of cycle life
- Steam Dumps are in Steam Pressure Mode in AUTOMATIC
- Main Steam Header pressure channel PT-507 fails to 0 psig.

Which of the following accurately describes the resulting RCS temperature and reactor power response?

RCS T_{AVE} stabilizes at _____.

Reactor power stabilizes _______________.

A. 562ºF
in the Source Range

B. 562ºF
at 10^{-8} amps

C. 557ºF
in the Source Range

D. 557ºF
at 10^{-8} amps

Answer: B

K/A Match:

The question matches the K/A as the applicant must demonstrate knowledge of how a malfunction of the steam dump system will affect the RCS system. A failure low of the main steam header pressure instrument (steam dump input) caused the RCS to heat up.

Explanation:

A. Incorrect. First part is correct since steam dumps will not react to the rising RCS temperature the ARVs will control the RCS Tave at 562ºF. The second part is wrong because at this power level (10^{-8}) the reactor is below the point of adding heat so there is no effect of the higher temperature on overall power. Plausible (common mistake) since the rise in temperature normally would have the effect of negative reactivity and lower power IF the reactor was above the point of adding heat.

B. Correct. SD1- Failure of AB PT-507 to a low condition has the effect of closing the Steam Dump Valves provided the Steam Dump System is operating in its Steam Pressure Mode (at power less than 15% by procedure the steam dumps are in this mode). The adverse effect of this action is to allow heat up of the RCS. ABN-709 requires the operator to take manual control of u-PK-507 and adjust Steam Dump Valve position as necessary to maintain the desired SG pressure. MR1 - Each atmospheric relief valve has open/close indicating lights on the vertical section of CB-07. Normal setpoint pressure is 1125 psig. Without operator action, temperature will increase to 562ºF. Since the reactor is below the point of adding heat the temperature rise will have no effect on overall reactor power.

C. Incorrect. First part is wrong since steam dumps will not be able to control SG pressure at 1092 psig (557ºF RCS) because the pressure that controls this is reading zero. The second part is wrong since the reactor is below the point of adding heat so there will be no feedback from the negative effects of temperature, power will remain constant. Plausible if the student believes the steam dumps are in Tave mode and that temperature would affect them.

D. Incorrect. First part is wrong since steam dumps will not be able to control SG pressure at 1092 psig (557ºF RCS) because the pressure that controls this is reading zero. Second part is correct. Plausible if the student believes that the steam dumps will function normally because they could be in Tave mode.

Technical Reference(s)	ABN-709	Attached w/ Revision # See Comments / Reference
	LO21SSD1	
	Steam Tables	

Proposed references to be provided during examination: Steam Tables

Learning Objective: LO21SYSSD1OB104, Explain the instrumentation and controls of the Steam Dump System and predict the system response.

Question Source: Bank # ILOT6474
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
	Level of Difficulty	3

10 CFR Part 55 Content:	55.41	7
	55.43	

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	2		
	K/A	045 K1.20		
	Importance Rating	3.4		

Main Turbine Generator: Knowledge of the physical connections and/or cause effect relationships between the MT/G system and the following systems: Protection system

Question # 35

Which of the following completes the statement below regarding the Main Turbine-Generator Protection System?

A Main Turbine-Generator trip will occur if Primary Water...

temperature rises above the setpoint of________

OR

head tank level drops below the setpoint of ________

A. 140°F
85%

B. 140°F
78%

C. 131°F
85%

D. 131°F
78%

Answer: B

K/A Match:

This question matches the K/A as it requires demonstration of knowledge on the cause effect relationship between the M/TG system and its protection system regarding Primary Water trip values.

Explanation:

A. Incorrect. First part is correct, a Main Turbine-Generator trip will occur if Primary Water supply temperature exceeds 140°F. Second part is incorrec t but plausible because the Primary Water Head Tank Low Level alarm comes in at 85% level.

B. Correct. First part is correct, see 'A' above. The second part is correct a Main Turbine-Generator trip will occur if Primary Water Head Tank level drops below 78%.

C. Incorrect. First part is incorrect, but plausible because the Generator Primary Water Temperature High alarm is received if temperature exceeds 131°F . Second part is incorrect, but plausible, see 'A' above.

D. Incorrect. First part is incorrect, but plausible, see 'C' above. Second part is correct, see 'B' above.

Technical Reference(s)	ALB-9B, Window 3.8	Attached w/ Revision # See Comments / Reference
	ALB-10A, Window 2.12	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSMG1OB119 State the function of the Generator Monitoring and 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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 4
55.43

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	2		
	K/A	075 A2.02		
	Importance Rating	2.5		

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

Question # 36

- Unit 1 is stable at 44%
- All Circulating Water Pumps trip

TRIP the _______ and ensure _______ Instrument Air Compressors shutdown.

A. Turbine
   Unit

B. Turbine
   Common

C. Reactor
   Unit

D. Reactor
   Common

Answer: D

K/A Match:

The question matches the K/A as it requires the applicant to predict the impact of losing the Circulating Water Pumps and take action in accordance with procedural guidance to mitigate the consequences.

Explanation:

A. Incorrect. The first part is incorrect but plausible because several procedures do not require a reactor trip below 50%, however a loss of all CWPs above 10% requires a reactor trip. Second part is incorrect see C below.

B. Incorrect. The first part is incorrect see ‘A’ above. The second part is correct as described in ‘D’ below.

C. Incorrect. The first part is correct as described in ‘D’ below. The second part is incorrect but plausible if thought that the Unit compressors were cooled by CW and they needed to be shutdown to prevent damage.

D. Correct. In accordance with ABN-304, the reactor is tripped due to power being above 10% with all CWPs tripped. The common instrument air compressors lose cooling from TPCW when all CWPs trip and must be shutdown, the unit instrument air compressors are cooled by CCW which is unaffected by a loss of CW.

Technical Reference(s)	ABN-304	Attached w/ Revision # See Comments / Reference
	ABN-306	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN304OB101, Analyze the response to a Circulating Water Pump Trip in accordance with ABN-304, Main Condenser and Circulating Water System 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 ______
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 10
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	2		
	K/A	014 K4.06		
	Importance Rating	3.4		

Rod Position Indication System (RPIS): Knowledge of RPIS design feature(s) and/or interlock(s) which provide for the following: Individual and Group Misalignment

Question # 37

- Shutdown Bank A is being exercised
- Current Full Out Position (FOP) is 228 steps

When Shutdown Bank A reaches ________ steps the __________ alarm will annunciate indicating rod misalignment.

A. 216
DRPI URGENT FAIL

B. 216
DRPI ROD DEV

C. 210
DRPI URGENT FAIL

D. 210
DRPI ROD DEV

Answer: D

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate the knowledge of design features for individual and group rod misalignments.

Explanation:

A. Incorrect. Plausible because a control bank rod 12 steps away from the other rods in the bank generates a DRPI ROD DEV alarm and 216 steps is 12 steps from 228 steps. Plausibility is given in C below..

B. Incorrect. Plausibility is given in A and C.

C. Incorrect. Plausible because 210 is the correct step level, however the DRPI URGENT FAIL are is not generated. The DRPI UNRGENT FAIL alarm is plausible because it generates a DRPI ROD DEV alarm.

D. Correct. Any shutdown bank less than or equal to 210 steps will generate a DRPI ROD DEV alarm.

Technical Reference(s)	ALM-0064A	Attached w/ Revision # See Comments / Reference
	DRPI Study Guide	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSRI1OB104, Explain the instrumentation and controls of the Rod Control Indication System and Predict the system response.

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 6
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	2		
	Group	2		
	K/A	001 A3.07		
	Importance Rating	4.1		

Control Rod Drive: Ability to monitor automatic operation of the CRDS, including: Boration/dilution.

Question # 38

During an RCS boration with control rods in automatic

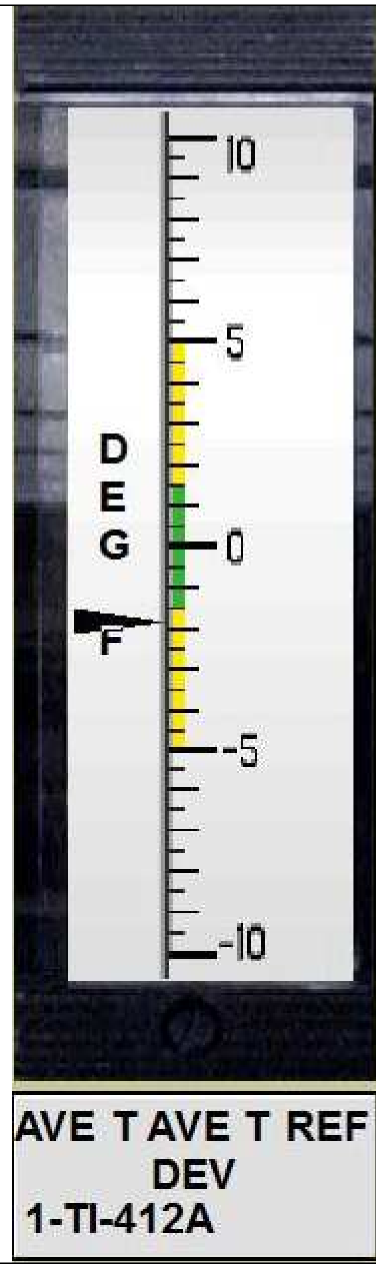
Control rods are _____________ at ___ steps per minute.

A. withdrawing
8

B. withdrawing
16

C. inserting
8

D. inserting
16



Answer: A

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge how control rods respond in automatic when a boration has taken place.

Explanation:

A. Correct. With AVE $T_{AVE} - T_{REF}$ deviation 1.7°F low due to the boration CBD rods w ithdraw at 8 steps per minute.

B. Incorrect. Plausible because the first part is correct. Second part is plausible based on the thought that rod speed is in the sloped portion of the speed logic.

C. Incorrect. Plausible based on a misconception that a negative temperature deviation would cause rods to insert. 8 spm is correct rod speed.

D. Incorrect. Plausible based on a misconception that a negative temperature deviation would cause rods to insert. Second part is plausible based on the thought that rod speed is in the sloped portion of the speed logic

Technical Reference(s)	Rod Control Study Guide	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21SYSCR1OB105, Explain the normal, abnormal and emergency operation of the Rod Control 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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 6
55.43

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	007 EA1.10		
	Importance Rating	3.7		

Reactor Trip – Stabilization - Recovery: Ability to operate and monitor the following as they apply to a reactor trip: S/G pressure

Question # 39

- EOS-0.1A, Reactor Trip Response is in progress
- A complete Loss of Offsite Power has occurred
- All Steam Generator pressures are 1130 psig and rising

What actions are required?

A. Place each Steam Generator Atmospheric Relief Valve Controller in Manual, adjust to control at 1092 psig and place back into Auto

B. Place each Steam Generator Atmospheric Relief Valve Controller in Manual and manually control each Steam Generator pressure to 1125 psig

C. Place Steam Dump Controller in Steam Pressure Mode, adjust to control at 1092 psig and place back into Auto

D. Place Steam Dump Controller in Manual and adjust to control at 1092 psig and manually control Steam Header pressure to 1092 psig

Answer: A

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate the ability to operate and monitor the Steam Generator pressures during a Reactor Trip stabilization.

Explanation:

A. Correct. As temperature would be above 557 ºF with Steam Generator pressures at 1130 psig, EOS-0.1A, Step 1 RNO would need to be performed. In accordance with the procedure and operational guidance this would require taking the controller to 1092 psig to control at 557 ºF.

B. Incorrect. Incorrect but plausible if believed that the Atmospheric Relief Valves were not functioning in Auto and manual operation was required. However, manual operation at 1125 psig which is their Auto setpoint would not mitigate the high pressure and temperature conditions which exist.

C. Incorrect. Incorrect but plausible if believed that the Steam Dumps were not working properly in the TAVE Mode and required operation in the Steam Pressure Mode. However, with a Loss of Offsite Power the condenser is not available and the Steam Dumps will not work.

D. Incorrect. Incorrect but plausible if believed that the Steam Dumps were not working properly in the TAVE Mode and required manual operation. However, with a Loss of Offsite Power the condenser is not available and the Steam Dumps will not work.

Technical Reference(s)	EOS-0.1A	Attached w/ Revision # See Comments / Reference
	OPGD-3 Attachment 5	
	ARV Operator Aid	

Proposed references to be provided during examination: None

Learning Objective: ______

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
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 10
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	009 G.2.4.6		
	Importance Rating	3.7		

Small Break LOCA: Knowledge of EOP mitigation strategies.

Question # 40

- Unit 2 is performing EOS-1.2B, Post LOCA Cooldown and Depressurization
- Containment pressure is 6 psig

Per Attachment 1.A, Foldout for EOS-1.2B, SI reinitiation is required if Pressurizer Level cannot be maintained GREATER THAN a setpoint level of ______%.

A. 13

B. 15

C. 32

D. 34

Answer: B

K/A Match:

The question tests the applicant's knowledge of EOS-1.2B Foldout Page actions required to mitigate the effect of increased RCS inventory loss on core cooling while in the EOPs.

Explanation:

A. Incorrect. Plausible because this is the non-adverse containment value for both Units.

B. Correct. Per EOS-1.2B, if pressurizer level cannot be maintained greater than 15% with adverse containment, SI reinitiation is required.

C. Incorrect. Plausible because pressurizer level of 32% is an adverse containment value used in EOS-1.2B, Step 18, but is used for the reason of determining if normal charging can be established.

D. Incorrect. Plausible because this is the corresponding pressurizer level value for adverse containment on Unit 1.

Technical Reference(s)	EOS-1.2B	Attached w/ Revision # See Comments / Reference
	EOS-1.2A	

Proposed references to be provided during examination: None

Learning Objective: LO21ERGE12OB107, Identify the items on EOS-1.2 Foldout Page including any equipment, parameter, setpoint or condition

Question Source: Bank #
Modified Bank # ILOT0867 (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 10
55.43

Original question ILOT 0867	

During performance of EOS-1.2A, Post LOCA Cooldown and Depressurization, which of the following would require SI REINITIATION with Containment Building Pressure at 6 psig?

A. RCS subcooling margin 68°F and pzr level 36%

B. RCS subcooling margin 63°F and pzr level 26%

C. RCS subcooling margin 58°F and pzr level 36%

D. RCS subcooling margin 56°F and pzr level 35%

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	011 EK2.02		
	Importance Rating	2.6		

Large Break LOCA: Knowledge of the interrelations between the Large Break LOCA and the following: Pumps

Question # 41

- Unit 1 Large Break LOCA
- Containment pressure is 35 psig and slowly decreasing
- CET temperatures are 300℉ and slowly increasing
- RCS pressure is approximately the same as Containment pressure

Reactor Coolant Pumps should be tripped …

A. to minimize heat input into RCS.

B. to minimize inventory loss from RCS.

C. due to a loss of CCW to RCP thermal barriers.

D. due to a loss of CCW to RCP motor and bearing coolers.

Answer: D

K/A Match:

The question matches the K/A as it requires the operator to demonstrate knowledge of the interrelationship between a Large Break LOCA and why RCPs should be tripped on a LBLOCA.

Explanation:

A. Incorrect. Plausible because heat input from the reactor coolant pumps is a concern for some accidents, however, not for Large Break LOCAs.

B. Incorrect. Plausible because inventory loss is a concern for a Small Break LOCA, however, not for a Large Break LOCA since adequate cooling flow enters the core from ECCS.

C. Incorrect. Plausible because loss of CCW flow to the RCP thermal barrier might be a reason to trip the pump, however, not for these circumstances.

D. Correct. Given the conditions listed in the Stem, the RCPs must be secured for the reason listed.

Technical Reference(s)	FRZ-0.1A, Step 4	Attached w/ Revision # See Comments / Reference
	FRZ-0.1A, Attachment 5 & 6	

Proposed references to be provided during examination: None

Learning Objective: DISCUSS the operator actions, including all cautions, notes, RNOs and bases associated with EOP-1.0. (LO21.ERG.E1A.OB04)

Question Source: Bank # ____
Modified Bank # ILOT5786 (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level:
Memory or Fundamental Knowledge ____
Comprehension or Analysis X
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 10
55.43 ____

Original question ILOT5786	

1	ID: ILOT5786	Points: 1.00

Given the following conditions:

- Unit 1 has experienced a large break loss of coolant accident (LOCA).
- Containment pressure is 35 psig and slowly decreasing.
- Core Exit Thermocouple temperatures are 400ºF and slowly increasing.
- RCS pressure is approximately the same as Containment pressure.
- Centrifugal Charging Pump (CCP) 1-02 tripped after automatically starting.
- All other Emergency Core Cooling System equipment is operating properly.

Which of the following describes:

1) The status of the Reactor Coolant Pumps (RCPs); and,

2) The reason for making this decision?

A. 1) RCPs should continue to run
2) Even with the inventory loss from the RCS, the rising CETs requires RCPs be run.

B. 1) RCPs should continue to run
2) Even with the loss of CCW flow to the RCP Motor and Bearing Coolers., the rising CETs requires RCPs be run.

C. 1) RCPs should be tripped
2) To minimize the inventory loss from the Reactor Coolant System.

D. 1) RCPs should be tripped
2) Due to a loss of CCW flow to the RCP Motor and Bearing Coolers.

Answer: D

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	022 AK3.03		
	Importance Rating	3.1		

Loss of Rx Coolant Makeup: Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: Performance of lineup to establish excess letdown after determining need.

Question # 42

Charging has been lowered to supply RCP seals only and Excess Letdown is being placed in service.

Which of the following describes the reason for a Caution taken while placing Excess Letdown in service?

Direct Excess Letdown to _______ for 10 minutes to avoid ________________________.

A. RCDT
unplanned boration or dilution

B. RCDT
CVCS piping thermal shock

C. PRT
unplanned boration or dilution

D. PRT
CVCS piping thermal shock

Answer: A

K/A Match:

The question is a K/A match as it requires the applicant to know the reason for flushing Excess Letdown to the RCDT when placing in service.

Explanation:

A. Correct. Excess letdown is directed to the RCDT and flushed for 10 minutes to avoid an unplanned boration or dilution.

B. Incorrect. First part is correct (See A above). Second part is plausible if thought that due to the limited flow through excess letdown that pre-warming is need prior to aligning to charging pump suction.

C. Incorrect. Plausible if thought that excess letdown is aligned to the PRT vice RCDT. Second part is correct (See A above)

D. Incorrect. First part is correct (See A above). Second part is plausible if thought that due to the limited flow through excess letdown that pre-warming is need prior to aligning to charging pump suction.

E.

Technical Reference(s)	SOP-103A	Attached w/ Revision # See Comments / Reference
	CVCS Study Guide	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN105OB105, Analyze the response to a Reactor Makeup System Malfunction in accordance with ABN-105, CVCS System Malfunctions

Question Source: Bank # ILOT8030
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

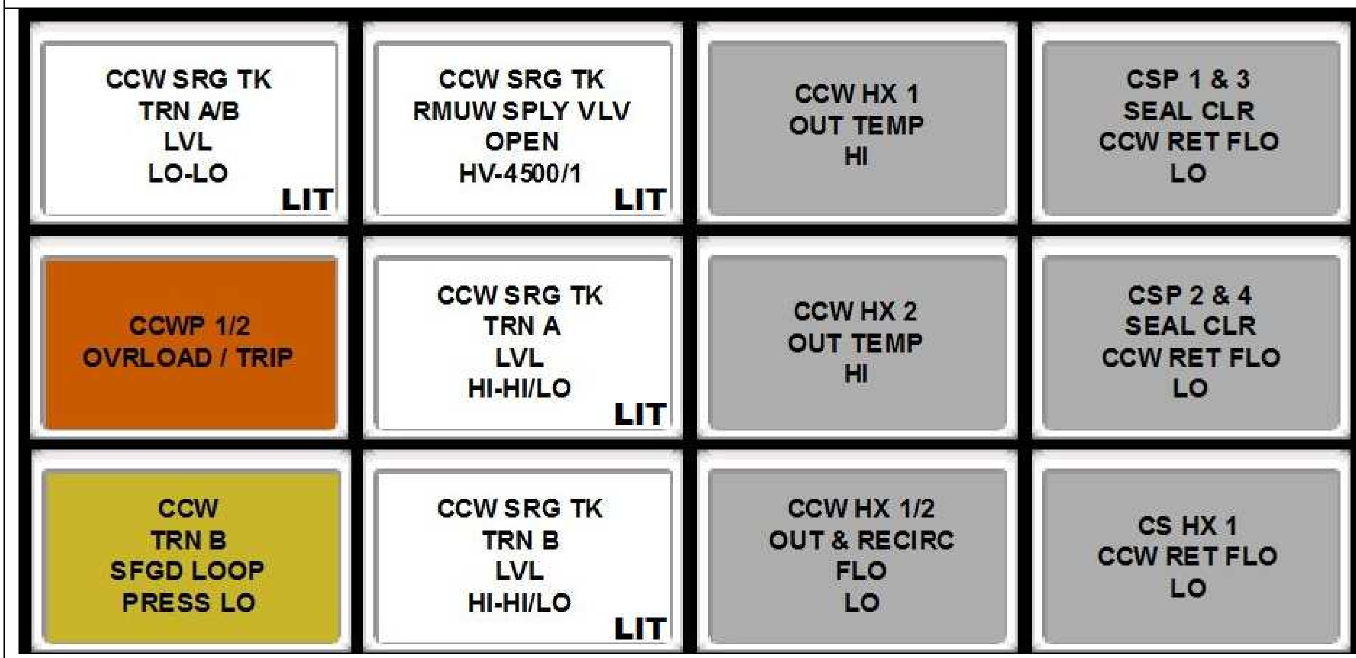
Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 6
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	026 AA2.03		
	Importance Rating	2.6		

Loss of Component Cooling Water: Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The valve lineups necessary to restart the CCWS while bypassing the portions of the system causing the abnormal condition

Question # 43



The above Unit 1, LIT alarms were received and subsequently

- 1-HS-4512, SFGD LOOP CCW RET VLV and 1-HS-4514, SGFD LOOP CCW SPLY VLV were closed on affected train
- The leakage was NOT stopped

What additional valve must be closed to isolate affected train and restore normal operation in unaffected train?

A. 1CC-0021, CCW SRG TK 1-01 TRN A OUT VLV

B. 1CC-0023, CCW PUMP 1-01 SUCT ISOL VLV

C. 1CC-0071, CCW SRG TK 1-01 TRN B OUT VLV

D. 1CC-0067, CCW PUMP 1-02 SUCT ISOL VLV

Answer: A

K/A Match:

The question matches the K/A as it requires the operator to demonstrate procedural knowledge on how to position valves to isolate the leaking train from the non-leaking train so that operation can continue with the unaffected train.

Explanation:

A. Correct. In accordance with ABN-502, the affected train surge tank outlet valve must also be closed in order to allow the surge tank to be refilled for the operating train.

B. Incorrect. Incorrect but plausible as this is the manual pump suction valve. This valve is plausible if thought that the Train is isolated with the exception of the small portion of the system from the surge tank to the suction of the pump. However, this isolation would not isolate the section of piping from the surge tank to the suction piping which may be the location of the leak. Further, the procedure does not call out for this valve to be isolated as it does the surge tank isolation valve.

C. Incorrect. Incorrect but plausible as this is the same valve as the correct Train A valve for Train B. The applicant must recognize the configuration and determine the train which must be isolated.

D. Incorrect. Incorrect but plausible for Train B as described in 'B' above for the affected Train A.

Technical Reference(s)	ABN-502,	Attached w/ Revision # See Comments / Reference
	M1-0229-A	
	M1-0229-B	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN501OB106 ANALYZE the response to Leakage Out of the CCW System in accordance with ABN-502, Component Cooling Water System 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 ______
Comprehension or Analysis X
Level of Difficulty 4

10 CFR Part 55 Content: 55.41 4
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	027 AK1.02		
	Importance Rating	2.8		

Pressurizer Pressure Control System (PZR PCS) Malfunction: Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Expansion of liquids as temperature increases.

Question # 44

- Unit 1 is at 100%
- Pressurizer Pressure channel 1-PT-455 fails low

(Assume no operator action)

Actual pressurizer pressure _________ and pressurizer liquid will ___________.

A. rises
outsurge

B. rises
insurge

C. lowers
outsurge

D. lowers
insurge

Answer: A

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate an understanding of how a Pressurizer Pressure Control System malfunction can cause thermal expansion of pressurizer liquid.

Explanation:

A. Correct. With no operator action PRZR controlling pressure channel failing low will not allow spray valves to open and will energize all BU heaters. This leads to an increase in pressurizer liquid temperature which causes pressure to rise and liquid to outsurge from the pressurizer.

B. Incorrect. First part is correct (See A above). Second part is plausible if applicant thinks the heaters coming on will raise pressure enough to lower specific volume causing an insurge, however temperature effect is greater than the pressure effect and specific volume will rise leading to the outsurge

C. Incorrect. First part is plausible if the applicant thinks that a pressure channel failing lower opens the spray valves vice turning on the heaters. Second part is correct (See A above).

D. Incorrect. First part is plausible (See C above). Second part is plausible (See B above).

Technical Reference(s)	Pressurizer Pressure and Level Control System Study Guide	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21SYSPP1OB104, Explain the instrumentation and controls of the Pressurizer Pressure Control System and predict the system response

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 7
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	029 EK2.06		
	Importance Rating	2.9		

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

Question # 45

- Reactor Trip Breaker testing is in progress on Unit 1 Train "A"
- Train “A” Reactor Trip Breaker (RTA) is OPEN
- Train “A” Reactor Trip Bypass Breaker (BYA) is CLOSED
- Train “B” Reactor Trip Breaker (RTB) is CLOSED
- Reactor failed to trip from AUTOMATIC signal

Which failure prevented AUTOMATIC Reactor Trip?

A. RTB Undervoltage Trip coil failed to energize

B. BYA Undervoltage Trip coil failed to de-energize

C. BYA Shunt Trip coil failed to energize

D. RTB Shunt Trip coil failed to de-energize

Answer: B

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge of Reactor Trip and Bypass Breakers and how failures associated with components of these breakers can cause an ATWT.

Explanation:

A. Incorrect. Plausible because RTB is equipped with an undervoltage trip coil, however, trip coils are normally energized and de-energize on a trip signal.

B. Correct. Given the conditions listed, the BYA Undervoltage Trip coil failed to de-energize.

C. Incorrect. Plausible because the Shunt Trip coil is designed to energize and trip open the breaker, however, BYA is not equipped with a Shunt Trip.

D. Incorrect. Plausible because RTB is equipped with a Shunt Trip coil, however, it energizes to trip.

Technical Reference(s)	Reactor Protection and ESFAS Study Guide Pg. 29 of 94	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21SYSES2OB103 Describe the components of the Solid State Protection System including interrelations with other systems to include interlocks and control loops.

Question Source: Bank # ILOT8120
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 6
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	038 G.2.4.4		
	Importance Rating	4.5		

Steam Gen. Tube Rupture: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Question # 46

- Main Steam Line 1-04 Radiation Monitor (MSL181) alarmed RED just prior to an AUTOMATIC Reactor Trip and Safety Injection
- AFW flow SG 1-04 is secured
- SG 1-04 level is rising faster than the other three steam generators
- SG 1-04 pressure is 1080 psig and rising and the other three SG pressures are 1040psig and slowly lowering
- Containment, Safeguards, and Auxiliary Building radiation monitors are NORMAL
- RCS pressure is 1620 psig and slowly lowering

Transition to...

A. EOP-3.0A, Steam Generator Tube Rupture

B. EOP-2.0A, Faulted Steam Generator Isolation

C. EOP-1.0A, Loss of Reactor or Secondary Coolant

D. EOS-1.1A, Safety Injection Termination

Answer: A

K/A Match:

The question matches the K/A as it requires the applicant to recognize abnormal indications for a SGTR which are entry level conditions for EOP-3.0A, Steam Generator Tube Rupture.

Explanation:

A. Correct. Indications require transition to EOP-3.0A, SGTR.

B. Incorrect. Plausible because EOP-2.0A is a transition from EOP-0.0A but pressures in SG 1-01, 1-02, & 1-03 are NOT lowering uncontrollably.

C. Incorrect. Plausible because EOP-1.0A is a transition from EOP-0.0A but containment radiation is NORMAL.

D. Incorrect. Plausible because EOS-1.1A is a transition from EOP-0.0A but RCS pressure is lowering NOT stable.

Technical Reference(s)	EOP-0.0A Steps 12, 13, 14, & 15	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21ERGE0AOB105, Analyze the diagnostic steps of EOP-0.0, Reactor Trip or Safety Injection

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 ___
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 10
55.43 ___

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	054 AK3.05		
	Importance Rating	4.6		

Loss of Main Feedwater: Knowledge of the reason for the following responses as they apply to the Loss of Main Feedwater (MFW): HPI/PORV cycling upon total feedwater loss.

Question # 47

A total loss of feedwater has occurred to all Steam Generators and no source of feedwater can be established.

What would be the result, if Safety Injection was initiated to provide core cooling without opening an adequate RCS bleed path?

RCS pressure would cycle with Pressurizer__________________ and Safety Injection flowrate _________ be adequate for core cooling.

A. PORVs
would

B. Spray valves
would

C. PORVs
would NOT

D. Spray valves
would NOT

Answer: C

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate an understanding of the complete loss of feedwater and the resulting plant response of primary pressure cycling on the PORVs. This response would be indicative of inappropriate operator response to the event, but is pertinent to understanding the recovery strategies which are employed.

Explanation:

A. Incorrect. Incorrect but plausible as described in 'C' below this is what occurs. However, the SI flow is not adequate with the RCS maintained at the PORV pressure.

B. Incorrect. Incorrect but plausible as described in 'D' below. The SI flow rate would not be adequate as stated in this distractor but remains plausible if thought that CCP flow was sufficient at normal operating pressure to remove the decay heat. However, the RCS pressure is actually higher cycling on the PORVs.

C. Correct. In accordance with the analyses that have been developed for complete loss of heat sink events. The bleed path must be opened and remain open for adequate cooling to occur until a secondary

D. Incorrect. Incorrect but plausible if believed the concern was only RCS pressure. However, prior to initiating Safety Injection, the RCPs are tripped. As such, the pressurizer spray is not available and letdown is isolated at the same time so Auxiliary Spray is not available. Thus the PORVs would cycle. The portion of the distractor that says that SI flow rate would not be adequate is correct as pressure would remain too high.

Technical Reference(s)	LO21MCOMI4	Attached w/ Revision # See Comments / Reference
	FRH-0.1A Bases	

Proposed references to be provided during examination: None

Learning Objective: LO21MCOMI4OB102, Analyze the acceptability of the three postulated recovery techniques for a Loss of Heat Sink

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 ___
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 8
55.43 ___

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	055 EA2.04		
	Importance Rating	3.7		

Station Blackout: Ability to determine or interpret the following as they apply to a Station Blackout: Instruments and controls operable with only dc battery power available.

Question # 48

During a Station Blackout additional load shedding is performed when safeguards battery voltage is less than 110 volts to allow for ____________________ and _________________.

A. battery charger restoration with portable generator
plant monitoring and control until AC power restored

B. battery charger restoration with portable generator
Safeguards Bus supply breaker closure

C. Diesel Generator field flashing
plant monitoring and control until AC power restored

D. Diesel Generator field flashing
Safeguards Bus supply breaker closure

Answer: D

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge of actions necessary regarding DC power supplies in order to operate controls to reestablish safeguards AC power.

Explanation:

A. Incorrect. Plausible because Attachments 2.A and 2.B when performed ensure sufficient time to restore battery chargers using a portable generator, however this is not what is accomplished by Attachment 2.C. The second part is incorrect but plausible (See C below).

B. Incorrect. First part is incorrect but plausible (See A above). Second part is correct (See D below).

C. Incorrect. First part is correct (See D below). The second part is incorrect but plausible because load shedding does provide for plant monitoring and control until AC power is restored during initial load shedding not the load shedding performed per Attachment 2.C.

D. Correct. If battery voltage lowers to less than 110 volts the associated bus is further load shed to ensure adequate voltage remains for flashing the diesel generator field or closing safeguards bus supply breakers for power restoration.

Technical Reference(s)	ECA-0.0A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21ERGC00OB105 Given a procedural step, or sequence of steps from ECA-0.0, state the purpose/basis for the step(s).

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 10
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	056 AK1.03		
	Importance Rating	3.1		

Loss of Off-site Power: Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: Definition of subcooling: use of the steam tables to determine it.

Question # 49

- Unit 1 Loss of All AC Power
- Power has been restored to Bus 1EA1
- Limited instrumentation has remained energized in the Control Room
- The following indications are available:
  - 1-TI-413A, HL 1 TEMP (WR) 450°F
  - 1-TI-423A, HL 2 TEMP (WR) 461°F
  - 1-TI-3611-2, CORE EXIT TEMP 472°F
  - 1-PI-405, HL 1 PRESS (WR) 535 psig
  - 1-PI-3616, RCS PRESS (WR) 615 psig

Calculated RCS subcooling is _____.

A. 40°F

B. 27°F

C. 16°F

D. 5°F

Answer: D

K/A Match:

The question is a match to the K/A as it requires the applicant to determine the subcooling following a loss of offsite and onsite power. The applicant must determine which instruments to use to obtain for a determination which is required in order for the procedure transition and recovery.

Explanation:

A. Incorrect. Subcooling for Highest RCS Pressure of 615 psig (630 psia) and Lowest RCS temperature 450°F.

B. Incorrect. Subcooling for Lowest RCS Pressure of 535 psig (550 psia) and Lowest RCS temperature 450°F.

C. Incorrect. Subcooling for Lowest RCS Pressure of 535 psig (550 psia) and Average RCS temperature 461°F.

D. Correct. Subcooling for Lowest RCS Pressure of 535 psig (550 psia) and Highest RCS temperature 472°F.

Technical Reference(s)	ECA-0.0A	Attached w/ Revision # See Comments / Reference
	Steam Tables	
	Core Cooling Monitor/RVLIS Study Guide	

Proposed references to be provided during examination: Steam Tables

Learning Objective: LO21ERGC00OB107, Identify the proper transitions out of ECA-0.0.

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 14
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	057 AA1.03		
	Importance Rating	3.6		

Loss of Vital AC Inst.Bus: Ability to operate and/or monitor the following as they apply to the Loss of Vital AC Instrument Bus: Feedwater pump speed to control pressure and level in S/G.

Question # 50

- Unit 1 is at 80% RTP and stable
- A loss of 1PC1 has just occurred

What actions are required?

Place 1-SK-509A, FWPT MASTER SPD CTRL in Manual and _______ demand.

Place ________________________________ in Manual and CONTROL level at program.

A. RAISE
1-FK-510, SG 1 FW FLO CTRL and 1-FK-540, SG 4 FW FLO CTRL.

B. RAISE
1-FK-520, SG 2 FW FLO CTRL and 1-FK-530, SG 3 FW FLO CTRL

C. LOWER
1-FK-510, SG 1 FW FLO CTRL and 1-FK-540, SG 4 FW FLO CTRL

D. LOWER
1-FK-520, SG 2 FW FLO CTRL and 1-FK-530, SG 3 FW FLO CTRL

Answer: A

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge of what controls require operation with respect to feedwater and steam generator level following the loss of a vital AC instrument bus.

Explanation:

A. Correct. In accordance with ABN-603, the loss of 1PC1 will reduce the speed of the Main Feedwater Pumps, thus requiring demand to be raised and all of the inputs to the Steam Generator Water Level Control program. Second part is the required actions for a failure of 1PC1. LT-551 and LT-554 are Channel 1 instruments which are the normal channels selected for control therefore alternate channels must be selected for control.

B. Incorrect. First part is correct see A above. Second part is incorrect as these instruments are not lost for the protection bus stated in the stem. Plausible if the student confuses the power supplies.

C. Incorrect. First part is incorrect but plausible if the student doesn't understand the inputs to the MFP speed controller and believes that a loss of power to the input channels would cause speed to rise. Second part is correct see A above.

D. Incorrect. First part is incorrect see C above. Second part is incorrect see B above.

Technical Reference(s)	ABN-603A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21ABN603OB101, Analyze the response to Loss of a Protection Bus in accordance with ABN-603, Loss of Protection or Instrument Bus

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
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 7
55.43 ___

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	058 G.2.1.20		
	Importance Rating	4.6		

Loss of DC Power: Ability to interpret and execute procedure steps.

Question # 51

1-ALB-10B Window 1.13, 125 VDC SWITCH PNL 1ED1 TRBL annunciates

Per ALM-0102A, 1-ALB-10B...

Place GROUND TEST switch in TEST; if NEGATIVE-GND white light is dimly LIT and the POSITIVE-GND white light is normally LIT; a ground condition exists for the ________ terminal.

Verify 1ED1 DC VOLTS are greater than or equal to ___ VDC.

A. NEGATIVE
128

B. NEGATIVE
120

C. POSITIVE
128

D. POSITIVE
120

Answer: A

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate the ability to interpret and execute procedure steps when responding to a loss of DC power.

Explanation:

A. Correct. Part 1 is correct in accordance with ALM-0102A. Part 2 is correct in accordance with ALM-0102A.

B. Incorrect. Part 1 is correct as described in 'A' above. Part 2 is incorrect but plausible in that the Alarm Setpoint 120 VDC but the procedure requires that Voltage be verified above 128 VDC.

C. Incorrect. Part 1 is incorrect in that the indications are for a ground on the Negative Terminal, not the Positive terminal. Part 2 is correct as described in 'A' above.

D. Incorrect. Part 1 is incorrect as described in 'C' above. Part 2 is incorrect but plausible as described in 'B' above.

Technical Reference(s)	ALM-0102A	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21SYSDC1OB104, Explain the normal, abnormal and emergency operation of the DC Electrical Distribution system

Question Source: Bank #
Modified Bank # ILOT8123 (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 10
55.43

Original Question: ILOT8123

1	ID: ILOT8123	Points: 1.00

Given the following conditions:

- Unit 1 is at 100% power with all systems in their normal alignment.
- ALB-10B, 1.13, 125 VDC SWITCH PNL 1ED1 TRBL, is alarming.
- V-1ED1, 125 VDC SWITCH PNL 1ED1 VOLT, is indicating 135 volts.
- When the GROUND TEST switch at Panel 1ED1 is placed in "TEST", both the NEGATIVE-GND and POSITIVE-GND white lights are lit.
- The POSITIVE-GND white light is brighter and the NEGATIVE-GND white light is dimmer.

Which of the following is the cause of the trouble alarm on 125 VDC Panel 1ED1, and the component supplying power to Panel 1ED1?

A. A POSITIVE ground exists and the BATTERY is supplying the bus.

B. A POSITIVE ground exists and the BATTERY CHARGER is supplying the bus.

C. A NEGATIVE ground exists and the BATTERY is supplying the bus.

D. A NEGATIVE ground exists and the BATTERY CHARGER is supplying the bus.

Answer: D

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	062 AK3.04		
	Importance Rating	3.5		

Loss of Nuclear Svc Water: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: Effect on the nuclear service water discharge flow header on a loss of CCW

Question # 52

- Train B CCW heat exchanger was removed from service for fouling

To return Train B CCW heat exchanger to service following tube cleaning …

The ________ side of Train B CCW heat exchanger should be filled, vented and pressurized first to prevent ________.

A. shell
release of hydrazine into safe shutdown impoundment

B. shell
chloride infusion from tube leak

C. tube
release of hydrazine into safe shutdown impoundment

D. tube
chloride infusion from tube leak

Answer: B

K/A Match:

This question matches the KA by requiring knowledge of the effect on SSW system when recovering from a loss of the CCW heat exchanger.

Explanation:

A. Incorrect. 1st part is correct. The shell side of the CCW heat exchanger is the side in which CCW flows. 2nd part is incorrect because CCW is at a higher pressure than SSW which could lead to a release of hydrazine to the SSI.

B. Correct. Filling, venting and pressurizing the CCW side of the heat exchanger prior to starting the Train B SSWP will prevent chloride infusion into the CCW side of the CCW heat exchanger due to CCW pressure being greater than SSW pressure.

C. Incorrect. 1st part is incorrect because the tube side of the CCW heat exchanger is the side SSW flows through. 2nd part is incorrect but plausible (See A).

D. Incorrect. 1st part is incorrect but plausible (see C). 2nd part is correct (See B).

Technical Reference(s)	Station Service Water Study Guide Pgs. 26 & 27 of 42	Attached w/ Revision # See Comments / Reference
	SOP-502A Precautions	
	OP51.SYS.SW1.FIG1	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSSW1OB103 Describe the components of the Station Service Water System including interrelations with other systems to include interlocks and control loops.

Question Source: Bank # ILOT8634
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 4
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	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.

Question # 53

- Unit 1 is at 100%
- Instrument Air Compressor 1-01 is tagged out-of-service for a maintenance inspection
- Instrument Air Compressor X-01 is running in LEAD and aligned to Unit 1 through Air Dryer X-01
- Instrument Air Compressor 1-02 is in BACKUP
- Instrument Air header pressure is 110 psig

Subsequently:
- Annunciator 1-ALB-1, Window 2.2, COMM INSTR AIR COMPR 1/2 TRIP is received.
- Instrument air header pressure begins to lower steadily.
- The crew has entered ABN-301, Instrument Air System Malfunction

The HIGHEST pressure that Instrument Air Compressor 1-02 will automatically start on lowering Instrument Air header pressure is ____ psig.

If Instrument Air Header pressure continues to lower, the operators are required to manually trip the reactor when pressure lowers to a procedurally specified MAXIMUM value of _____ psig.

A. 95
35

B. 95
45

C. 100
35

D. 100
45

Answer: C

K/A Match:

The question is a K/A match as a loss of instrument air has occurred and the applicant is to distinguish the automatic actions which should occur from the backup/emergency air compressor for the given situation. Further, the applicant demonstrates the ability to monitor the air pressure at which a Unit trip would be required in accordance with procedural direction.

Explanation:

A. Incorrect. Plausible as the 95 psig value would be correct if the instrument air compressors were aligned in the opposite order (i.e. 1-02 in LEAD and X-01 in BACKUP). This configuration is allowed but is not the preferred alignment in accordance with SOP-509A and was the operating configuration for the original question. The second part of the answer is correct in that the operator must trip the reactor at 35 psig header pressure in accordance with ABN-301.

B. Incorrect. Plausible as the first part of the answer is explained in ‘A’ above. The second part of the answer is incorrect but plausible in that Step 2.3.5 uses 45 psig as the value to distinguish that instrument air header pressure is being restored and is thus a familiar number for the operators responding to a loss of instrument air.

C. Correct. When the 1-02 instrument air compressor is in BACKUP it will auto start at 100 psig on lowering pressure as stated in ABN-301. The second part is correct as described in ‘A’ above.

D. Incorrect. The first part of the answer is correct as described in ‘C’ above. The second part is incorrect but plausible as described in ‘B’ above.

Technical Reference(s)	SOP-509A	Attached w/ Revision # See Comments / Reference
	ALM-0011A	
	ABN-301	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN301OB103, Analyze the response to Instrument Air Compressor Trip or Header Pressure Low in accordance with ABN-301 Instrument Air System Malfunction

Question Source: Bank # ____
Modified Bank # ILOT8667 (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 7
55.43 ____

Original Question: ILOT8667

1	ID: ILOT8667	Points: 1.00

Given the following Unit 1 conditions:

- The Unit is at 100% power.
- Instrument Air Compressor 1-01 is tagged out for inspection.
- Instrument Air Compressor 1-02 is running in LEAD.
- Instrument Air Compressor X-01 is in STANDBY and aligned to Unit 1 through Air Dryer X-01.
- Instrument Air header pressure is 110 psig.

Subsequently:

- Annunciator, 1-ALB-1, Window 2.1, INSTR AIR COMPR 1/2 TRIP is received.
- Instrument Air header pressure begins to lower steadily.
- The crew has entered ABN-301, Instrument Air System Malfunction.

Which of the following completes the statements below:

1. The HIGHEST value that Instrument Air Compressor X-01 will start on lowering Instrument Air header pressure is _____(1)_____.
2. If Instrument Air Header pressure continues to lower, the operators are FIRST required to manually trip the reactor when pressure lowers to a procedurally specified value of _____(2)_____.

A. (1) 95 psig
(2) 35 psig

B. (1) 95 psig
(2) 48 psig

C. (1) 100 psig
(2) 35 psig

D. (1) 100 psig
(2) 48 psig

Answer: A

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	W/E12 AK3.04		
	Importance Rating	4.5		

Steam Line Rupture: Knowledge of the reasons for the following responses as they apply to the Steam Line Rupture: Actions contained in EOPs for steam line rupture

Question # 54

- EOP-2.0A, Faulted Steam Generator Isolation, in progress
- 125 VDC Battery BT1D2 verified aligned from either 125 VDC Battery Chargers BC1D2 or BC1D24

BT1D2 is aligned to either BC1D2 or BC1D24 to prevent _______.

A. a potential loss of Unit Auxiliary Transformer 1UT

B. a potential loss of the Main Turbine DC Emergency Oil Pump

C. inadvertent opening of Main Steam Isolation Valves

D. inadvertent opening of Atmospheric Relief Valves

Answer: C

K/A Match:

The question matches the K/A as it requires the applicant to know the reason for the actions performed in response to a steam line rupture outside of containment.

Explanation:

A. Incorrect. Plausible because power for Unit Auxiliary Transformer 1UT comes from Bus 1D2-2.

B. Incorrect. Plausible because the Main Turbine Emergency DC Oil Pump is powered from Bus 1D2, however, not Distribution Panel 1D2.

C. Correct. Because the power supply to Battery Charger BC1D2 is load shed on a Safety Injection Signal (SIS), EOP-2.0A requires an alignment to Battery Charger BC1D24. If Battery Charger BC1D24 is not available, the SIS is reset, and Battery Charger BC1D2 is placed in service. Either of these actions is performed to ensure that the Main Steam Isolation Valves remain closed.

D. Incorrect. Plausible because the emergency overrides for Atmospheric Release Valves #2 and #4 are powered from Bus 1ED2.

Technical Reference(s)	EOP-2.0A	Attached w/ Revision # See Comments / Reference
	DC Electrical Study Guide	
	E1-0019 Sht. C	

Proposed references to be provided during examination: None

Learning Objective: LO21ERGE2AOB104, Given a procedural step, or sequence of steps from EOP-2.0, Faulted Steam Generator Isolation, **STATE** the purpose/basis for the step(s).

Question Source:	Bank #	ILOT0905	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam 2012

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
	Level of Difficulty	3

10 CFR Part 55 Content: 55.41 10
55.43

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	W/E05 EK2.2		
	Importance Rating	3.9		

Inadequate Heat Transfer – Loss of Secondary Heat Sink: Knowledge of the interrelations between the Loss of Secondary Heat Sink 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.

Question # 55

- ALM-0052B, 2-ALB-5B is in progress re-establishing the loop seal for 1/2-PCV-456, PRZR PORV.
- FRH-0.1B, Response to Loss of Secondary Heat Sink, is in progress.
- While establishing RCS Bleed Path, power to 1/2-8000B, PRZR PORV BLK VLV is LOST and cannot be restored.

With existing conditions, and NO additional operator action, core cooling will be _________.

Bleed and Feed is initiated when indicated wide range level in at LEAST ______ SGs is less than 21% (25% adverse).

A. adequate
two

B. inadequate
two

C. adequate
three

D. inadequate
three

Answer: D

K/A Match:

This question meets the K/A because it involves a Loss of Secondary Heat Sink, and tests knowledge of heat removal (S/Gs), and requires the applicant to recognize that with no additional operator action, there will be INADEQUATE core cooling (emergency coolant bleed path is undersized and adequate feed is inhibited).

Explanation:

A. Incorrect. First part is incorrect but plausible if applicant recognizes there is a flowpath for Bleed and Feed, but with one PORV isolated, and no additional operator action to open reactor head and pressurizer vent valves core cooling is inadequate. Second part is incorrect but plausible as two SG levels being less than the specified level is used in the AFW system for initiation of an automatic Turbine Driven AFW Pump start.

B. Incorrect. First part is correct (See D below). Second part incorrect but plausible (See A above).

C. Incorrect. First part is incorrect but plausible (See A above). Second part is correct (See D below).

D. Correct. First part is correct because a single PORV will not pass enough flow for adequate core cooling. Second part is correct per FRH-0.1B, Step 12; WR level in at least 3 SGs < 21% requires initiation of Bleed and Feed.

Technical Reference(s)	FRH-0.1B, Response to Loss of Secondary Heat Sink, Steps 12 and Step 22	Attached w/ Revision # See Comments / Reference
	ALM-0052B, Window 3.1	

Proposed references to be provided during examination: None

Learning Objective: LO24ERGFH1OB104, Given a procedure, step, note or caution, Discuss the reason or basis for the step, note or caution in FRH-0.1.

Question Source: Bank # ILOT8575
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 10
55.43

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	1		
	K/A	077 AA2.04		
	Importance Rating	3.6		

Generator Voltage and Electric Grid Disturbances: Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: VARs outside the capability curve.

Question # 56

- Unit 1 is responding to a grid disturbance
- Main Generator is operating at 1200 MWe and +800 MVARs

What action is required to restore Main Generator within Generator Capability Curve?

REFERENCE PROVIDED

A. Raise Exciter Current Target

B. Lower Exciter Current Target

C. Raise Voltage Target

D. Lower Voltage Target

Answer: D

K/A Match:

The question matches the K/A as it requires the applicant to detail actions to restore the main generator within the generator capability curve following a grid voltage disturbance.

Explanation:

A. Incorrect. Plausible if the voltage regulator was in manual and the applicant had a misunderstanding of the relationship between exciter current target and MVARs.

B. Incorrect. Plausible because if the voltage regulator was in manual this would be the correct method to adjust MVARs.

C. Incorrect. Plausible if the applicant had a misunderstanding of the relationship between voltage target and MVARs.

D. Correct. With the voltage regulator in auto the operator will lower the voltage target to bring the generator back within the acceptable range of the capability curve.

Technical Reference(s)	ABN-402	Attached w/ Revision # See Comments / Reference
	IPO-003A	
	TDM-401	

Proposed references to be provided during examination: TDM-401A, Reactive Capability Curve

Learning Objective: LO21SYSMG1OB126, Explain the normal, abnormal and emergency operation of the Main Generator

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
Level of Difficulty 4

10 CFR Part 55 Content: 55.41 5
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	2		
	K/A	001 AK1.21		
	Importance Rating	2.9		

Continuous Rod Withdrawal: Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal: Integral rod worth.

Question # 57

- Unit 1 is performing a reactor startup
- Control Bank D rods are at 125 steps being withdrawn
- 1/1-FLRM CONTROL ROD MOTION CTRL is released
- Control Bank D continues to withdraw

The RO is required to ________.

As Control Bank D continues to withdraw integral rod worth available for insertion ________.

A. trip the reactor
decreases

B. trip the reactor
increases

C. insert all control rods CBO
decreases

D. insert all control rods to CBO
increases

Answer: B

K/A Match:

The question matches the K/A as it requires the applicant to assert the affect that Integral Rod Worth has during a Continuous Rod Withdrawal event.

Explanation:

A. Incorrect. Part 1 is correct as described in 'B' below. Part 2 is incorrect but plausible if the applicant confuses the integral worth of the control rods with the differential worth as with Control Bank D past the core mid plane the Differential Rod Worth should be decreasing.

B. Correct. Part 1 is correct in accordance with ABN-712 if the control rods continue to move after being placed in manual (they started in manual), the operator is instructed to trip the reactor. Part 2 is correct as the control rods continue to withdrawal the integral worth of the control rods to be inserted into the core increases, which is primary to terminating a continuous rod withdrawal event.

C. Incorrect. Part 1 is incorrect but plausible as ABN-712 requires that Control Rods be inserted to CBO during a Reactor Startup, however the previous sub step would trip the reactor. Part 2 incorrect but plausible as described in 'A' above.

D. Incorrect. Part 1 is incorrect but plausible as described in 'C' above. Part 2 is correct as described in 'B' above.

Technical Reference(s)	ABN-712	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21ABN712OB101, Analyze response to an abnormal rod control response in MODE 1 or 2 in accordance with ABN-712, Rod Control System 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 ____
Comprehension or Analysis X
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 1
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	2		
	K/A	028 EK3.05		
	Importance Rating	3.7		

Pressurizer Level Malfunction: Knowledge of the reasons for the following responses as they apply to the Pressurizer level Control malfunctions: Actions contained in EOP for PZR level malfunctions.

Question # 58

- Unit 2 at 100% power
- ABN-706, Pressurizer Level Instrument Malfunction in progress
- PROMPT Team has been contacted to place SW # 7 switch on NMT2 card for channel 0461 in CLOSED position

Channel 0461 will be in a ______ condition.

This functionally results in Pressurizer Water Level – High Reactor Trip having ______ coincidence logic to process a Reactor Trip for Pressurizer Water Level – High.

A. tripped
1 out of 2

B. tripped
1 out of 3

C. bypassed
1 out of 2

D. bypassed
1 out of 3

Answer: A

K/A Match:

The question is a K/A match as it specifically addresses a situation where a malfunction has occurred in the Pressurizer Level system and requires the applicant to detail the reason for taking the action delineated in ABN-706 for this malfunction.

Explanation:

A. Correct. First part is correct because placing the SW # 7 switch on NMT2 card in the closed position “trips the bistable.” Second part is correct as Technical Specification 3.3.1-1, the Pressurizer Water Level – High has 3 Required Channels and is a 2 out of 3 normal coincidence. With one channel in Trip the remaining 2 channels are in a 1 of 2 coincidence which prevents a single channel failure from preventing or causing a reactor trip.

B. Incorrect. First part is correct (See A above). Second part is incorrect but plausible if the applicant believes the normal coincidence for Pressurizer Water Level – High has 4 Required Channels and thus a 2 out of 4 normal logic. With this assumption and with one channel in Trip the remaining 3 channels are in a 1 out of 3 coincidence to process a Reactor Trip.

C. Incorrect. First part is incorrect but plausible as one channel may be placed in bypass for surveillance testing. If the reason for having PROMPT Team manipulate the switch was not understood, the applicant could believe that the channel is being bypassed for trouble shooting, repair or testing. Second part is correct (See A above).

D. Incorrect. First part is incorrect but plausible (See B above). Second is incorrect but plausible (See C above).

Technical Reference(s)	LO21SYSIC3	Attached w/ Revision # See Comments / Reference
	RPS & ESFAS Study Guide	
	TS Table 3.3.1-2	
	ABN-706	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN705 OB102, Analyze the Response to a Pressurizer Level Malfunction in accordance with ABN-706, Pressurizer Level Instrument Malfunction

Question Source: Bank # ____
Modified Bank # ILOT0009 (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 10

Original Bank Question: ILOT0009
Given the following conditions: • Unit 2 is operating at 95% power. • The crew is responding to a failure of pressurizer level instrument channel 0461. • The operators have contacted the PROMPT team to place SW7 on the NMT card for channel 0461 in the CLOSED position. Which of the following describes the reason for placing SW7 on the NMT card for channel 0461 in the CLOSED position? A. This blocks a trip input from the failed channel so that a spurious Reactor trip is prevented. B. This blocks output from the control board demultiplexer to the meter of the failed channel. C. This ensures that if the operator selects the failed channel with 1/2-LS-459D, PRZR LVL CTRL CHAN SELECT switch that control will not be transferred from an operable alternate control channel. D. This ensure that the failed channel provides a trip input to SSPS to allow a Reactor trip. Answer: D

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	2		
	K/A	032 G.2.4.20		
	Importance Rating	3.8		

Loss of Source Range NI: Knowledge of the operational implications of EOP warnings, cautions, and notes.

Question # 59

- EOP-0.0A, Reactor Trip Or Safety Injection in progress
- 1-NI-32, Source Range detector inoperable
- Automatic Safety Injection is reinstated per EOP-0.0A, Attachment 9, Post Event System Realignment which requires operation of the reactor trip breakers

____ Source Range channel(s) must be operable to reinstate Automatic Safety Injection.

Proper operation of each reactor trip breaker to reinstate an automatic safety injection signal is __________.

A. Two
CLOSED only

B. Two
CLOSED then OPENED

C. One
CLOSED only

D. One
CLOSED then OPENED

Answer: B

K/A Match:

Asks the applicant to recall a note during the performance of an EOP which discusses the loss of a source range detector (or the minimum required number of) while performing the procedure step.

Explanation:

A. Incorrect. First part is correct (See B below). Second part is incorrect but plausible if applicant believes that closing the Reactor Trip Breaker would reinstate the automatic safety injection, but proper operation is to cycle the breaker closed and then open.

B. Correct. The note prior to EOP-0.0A, Attachment 9, Step 11b requires two source range channels operable prior to closing reactor trip breakers to reinstate automatic safety injection. Based on one channel being inoperable automatic safety injection may NOT be reinstated. The second part is correct to restore this functionality both breakers would need to be cycled closed and open.

C. Incorrect. First part is incorrect but plausible as if the operator believes that with only one detector you are allowed to reset the breakers to reset auto SI. Second part is incorrect but plausible (See A above).

D. Incorrect. First part is incorrect but plausible (See C above). Second part is correct (See B above).

Technical Reference(s)	EOP-0.0A, Attachment 9	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: DISCUSS EOP-0.0, Reactor Trip or Safety Injection including the Purpose, Applicability, Symptoms/Entry Conditions, Operator Actions, Bases, Foldout Pages and Attachments. (LO21.ERG.E0A.OB07)

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 ______
Level of Difficulty 3

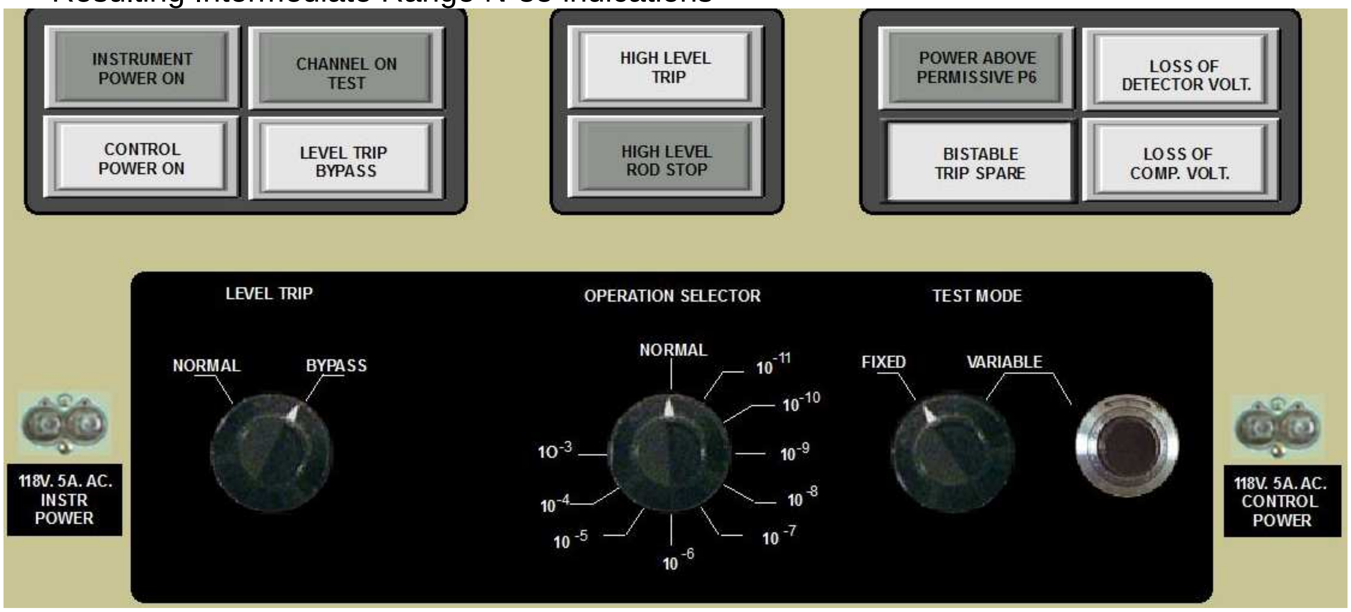
10 CFR Part 55 Content: 55.41 10
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	2		
	K/A	033 AA1.01		
	Importance Rating	2.9		

Loss of Intermediate Range NI: Ability to operate and / or monitor the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Power-available indicators in cabinets or equipment drawers

Question # 60

- Unit 1 is stable at 3% power
- An I&C technician is troubleshooting Intermediate Range N-35
- Resulting Intermediate Range N-35 indications



I&C technician removed the ________ power fuses.

Unit 1 reactor is ________.

A. Control
tripped

B. Control
stable at 3%

C. Instrument
tripped

D. Instrument
stable at 3%

Answer: D	

K/A Match:

The question matches the K/A as it requires the applicant to recognize the light indications on the IR NI detector drawer and determine the status of the instrument and plant.

Explanation:

A. Incorrect. Plausible because there are two choices from which to choose and the applicant must know that the power on light is lit when power is available and dark when it is NOT available. If the Control Power fuse were removed the only lights lit would be the INSTRUMENT POWER ON and BISTABLE TRIP SPARE lights. Even with the Level Trip switch in Bypass if the Control Power fuses are removed the reactor will trip.

B. Incorrect. First part is plausible (See A above). Second part is correct (See D below).

C. Incorrect. First part is correct (See D below). Second part is correct (See A above).

D. Correct. With the Level Trip switch in Bypass removing the Instrument Power fuses provides the given indication. With Level Trip in Bypass removing the Instrument Power fuses does NOT trip the reactor.

Technical Reference(s)	ABN-702	Attached w/ Revision # See Comments / Reference
	LO21SYSEC1	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSEC1OB04 - **EXPLAIN** the instrumentation and controls of the Excore Instrumentation system and **PREDICT** the system response.

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 6
55.43

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	2		
	K/A	037 AA2.11		
	Importance Rating	3.8		

Steam Generator Tube Leak: Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: When to isolate one or more S/Gs

Question # 61

- Unit 2 at 85% power
- ABN-106, High Secondary Activity, in progress due to tube leak on Steam Generator 2-01
- Chemistry and N-16 Radiation monitors confirm leak rate of 105 gpd

Per ABN-106...

2-HS-2452-2, AFWPT STM SPLY VLV MSL 1 ____ placed in Pull Out prior to MODE 3 entry.

Steam Generator 2-01 MSIV ____ closed prior to MODE 3 entry.

A. is NOT
   is

B. is
   is

C. is NOT
   is NOT

D. is
   is NOT

Answer: D

K/A Match:

The question requires the applicant to determine using the given current plant conditions and procedural knowledge when isolation of the S/G tube leak is acceptable.

Explanation:

A. Incorrect. Part 1 is incorrect but plausible (See C below). Part 2 is incorrect but plausible (See B below).

B. Incorrect. Part 1 is correct (See D below). Part 2 is incorrect but plausible if the applicant believes that at low power prior to MODE 3 entry the 2-01 MSIV can be closed to limit radiological release.

C. Incorrect. Part 1 is incorrect but plausible in that the remainder of the SG isolation occurs in Step 16, which in accordance with ABN-106, the operators cannot proceed past Step 14 until the unit is in MODE 3. The plausibility of MODE 3 is further enhanced by the required voluntary entrance into TS 3.7.5 for closing the steam admission valve. Part 2 is correct (See D below).

D. Correct. In accordance with ABN-106 at step 8 the Turbine Driven Steam admission valve on SG 2-01 would be placed in Pull Out. Step 8 can be performed in MODE 1 as the procedure only restricts that the operators cannot proceed past Step 14 until the unit is in MODE 3. If RCS activity and tube leak size warrants cooling down with SG 2-01 isolated that isolation is not performed until after MODE 3 entry in ABN-106.

Technical Reference(s)	ABN-106	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21ABN106 – Analyze the response to a S/G tube leakage greater than or equal to 75 GPD in accordance with ABN-106, High Secondary Activity.

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 ____
Level of Difficulty 4

10 CFR Part 55 Content: 55.41 10
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	2		
	K/A	051 AK3.01		
	Importance Rating	2.8		

Loss of Condenser Vacuum: Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capacity upon loss of condenser vacuum

Question # 62

- Unit 2 is in a forced outage
- In Hot Standby using IPO-007B, Maintaining Hot Standby
- RCS temperature is 557°F and stable on Steam Dumps

Subsequently:

- Main Condenser vacuum begins rapidly lowering

When Main Condenser vacuum drops below the setpoint of ______ the Steam Dump valves will close to prevent _________ in the Main Condenser.

A. 12.0”Hg
overheating

B. 12.3”Hg
overheating

C. 12.0”Hg
overpressure

D. 12.3”Hg
overpressure

Answer: D

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge of plant response with respect to the capacity of the steam dump system to prevent damage to the main condenser upon loss of condenser vacuum.

Explanation:

A. Incorrect. First part is incorrect but plausible (See C below). Second part is incorrect but plausible (See B below).

B. Incorrect. First part is correct (See D below). Second part is incorrect but plausible because the applicant could think that stopping steam flow into the main condenser will stop the heat input and protect the condenser from over heat not overpressure.

C. Incorrect. First part is incorrect but plausible in that the Unit 2 setpoint was recently 12.0"Hg. Second part is correct (See D below).

D. Correct. First part is correct in that C-9 is lost when condenser vacuum drops below 12.3"Hg. Second part is correct in that C-9 will be lost and the steam dump valves close to protect the condenser from rupture due to overpressure.

Technical Reference(s)	ALM-0065A	Attached w/ Revision # See Comments / Reference
	ABN-304	
	Steam Dump Study Guide	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN304OB02, **ANALYZE** the response to Main or Auxiliary Condenser Vacuum Decreasing in accordance with ABN-304, Main Condenser and Circulating Water System 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 ______
- Comprehension or Analysis X
- Level of Difficulty 2

10 CFR Part 55 Content:
- 55.41 7
- 55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	2		
	K/A	W/E10 EK1.2		
	Importance Rating	3.4		

Natural Circulation with Steam Void in Vessel with/without RVLIS: Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS): Normal, abnormal and emergency operating procedures associated with (Natural Circulation with Steam Void in Vessel with/without RVLIS).

Question # 63

- Unit 1 is responding to loss of Off-site power
- EOS-0.3A, Natural Circulation Cooldown with Steam Void in Vessel (With RVLIS) is in progress
- RCS depressurization and cooldown are in progress
- Letdown is NOT in service

RCS will be depressurized with _____________.

RCS cooldown rate is limited to ______ °F/hr.

A. PRZR PORV
   50

B. PRZR PORV
   100

C. Auxiliary Spray
   50

D. Auxiliary Spray
   100

Answer: B

K/A Match:

The question matches the K/A as the applicant is required to demonstrate knowledge of the method used for depressurization and the limits on cooldown rate while performing a cooldown under natural circulation with a void in the vessel.

Explanation:

A. Incorrect. First part is correct (See B below). Second part is incorrect but plausible as the cooldown rate limit is 50°F/hr in EOS-0.4A .

B. Correct. First part is correct per EOS-0.3A without letdown in service a PRZR PORV is used to depressurize the RCS. Second part is correct per EOS-0.3A the cooldown rate limit is 100°F/hr .

C. Incorrect. First part is incorrect but plausible because if letdown were in service auxiliary spray would be the preferred method for depressurization. Second part is incorrect but plausible (See A above).

D. Incorrect. First part is incorrect but plausible (See C above). Second part is correct (See B above).

Technical Reference(s)	EOS-0.4A	Attached w/ Revision # See Comments / Reference
	EOS-0.3A	

Proposed references to be provided during examination: None

Learning Objective: LO21ERGE02OB03, Discuss the ERG background for performing Natural Circulation Cooldown with and without RVLIS indications.

Question Source: Bank # ILOT8647
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X
Level of Difficulty 4

10 CFR Part 55 Content: 55.41 10
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	2		
	K/A	076 AA2.02		
	Importance Rating	2.8		

High Reactor Coolant Activity: Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Corrective actions required for high fission product activity in RCS

Question # 64

- Unit 1 at 100%
- Letdown flow is 75 gpm
- 1-RE-0406 (FFL160), GROSS FAILED FUEL MONITOR, has alarmed
- Chemistry reports that Reactor Coolant System specific activity has increased steadily over the past several days

Per ABN-102, High Reactor Coolant Activity letdown flow should be __________ to minimize personnel radiation exposure.

A. lowered to 0 gpm

B. lowered to 45 gpm

C. raised to 120 gpm

D. raised to 195 gpm

Answer: C

K/A Match:

The question is a K/A match as it requires the applicant to understand the ABN procedure and know what corrective action is taken based on evaluation of current plant conditions (RCS high activity).

Explanation:

A. Incorrect. Plausible since isolating letdown will prevent the activity from circulating in the Auxiliary and Safeguards Buildings, but it will not reduce RCS activity impacting future dose.

B. Incorrect. Plausible since reducing letdown will minimize the activity circulating in the Auxiliary and Safeguards Buildings while still allowing some cleanup of the RCS, but it will not maximize the reduction in RCS activity impacting future dose.

C. Correct. Letdown flow should be increased to a maximum value, but less than 140 gpm, to allow mechanical filtration of the letdown flow via the mixed bed demineralizers, minimizing future dose.

D. Incorrect. Plausible as all letdown valves open would give this value but flow is limited to 140 gpm when RCS temp is ≥ 500 degrees.

Technical Reference(s)	ABN-102	Attached w/ Revision # See Comments / Reference
	SOP-103A	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN103OB01, ANALYZE the response to High Reactor Coolant Activity in accordance with ABN-102, High Reactor Coolant Activity.

Question Source: Bank # ILOT0081
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ________
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 10
55.43 ________

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	1		
	Group	2		
	K/A	W/E15 EK2.1		
	Importance Rating	2.8		

Containment Flooding: Knowledge of the interrelations between the (Containment Flooding) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question # 65

- At 0800, Unit 1Large Break LOCA
- EOS-1.3A, Transfer to Cold Leg Recirculation is complete
- EOP-1.0A, Loss of Reactor or Secondary Coolant in progress
- At 1020, CNTMT RECIRC SMP LVL indicates 815’ 0” on all channels

MINIMUM CNTMT RECIRC SMP LVL of _____ would require entry into FRZ-0.2A, Response to Containment Flooding.

What is the operational concern for high Containment water level?

A. 816’ 0”
Damaging critical plant components necessary for recovery

B. 817’ 0”
Damaging critical plant components necessary for recovery

C. 816’ 0”
Flooding Motor Operated Valves to be operated in EOS-1.4A

D. 817’ 0”
Flooding Motor Operated Valves to be operated in EOS-1.4A

Answer: A

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate knowledge of the entry conditions to FRZ-0.2 for Containment Flooding and the interrelations with components potentially affected by the flooding.

Explanation:

A. Correct. Part 1 is correct in accordance with the Containment Status Tree when Containment Sump Level is no longer less than 816 ft. FRZ-0.2A must be entered. As the Containment Sump Level indication moves in 1 ft. increments the next indication after 815' 0" is 816' 0" and FRZ-0.2A must be performed. Part 2 is correct in accordance with the lesson plan that covers FRZ-0.2 excessive flooding has the potential for damaging critical components needed for plant recovery.

B. Incorrect. Plausible as the part 1 answer would be correct if the applicant believed that the Containment Status Tree required the transition when Containment Sump Level was greater than 816' 0" as described in 'A' above, the next level that would be greater than 816' 0" would be 817' 0". The answer to Part 2 is correct as described in 'A' above.

C. Incorrect. Plausible as the part 1 answer is correct as described in 'A' above. Part 2 is incorrect in that the motor operated valves needed for EOS-1.4A performance are all located in the Safeguards Building and thus not subject to Containment Flooding concerns. In particular these valves are 8809A/B, 8716A/B, 8840, 8821A/B, 8802A/B and 8835. This answer is plausible as the next major action in providing long term cooling is required to be performed at 3 hours after the initiating event in accordance with EOP-1.0A. It is plausible that the applicant could believe that these valves are subject to the flooding but as mentioned above, these valves are all outside of containment.

D. Incorrect. Plausible for Part 1 as described in 'B' above. Plausible for Part 2 as described in 'C' above.

Technical Reference(s)	Containment Status Tree	Attached w/ Revision # See Comments / Reference
	LO21ERGFZ2	
	EOS-1.4A	
	EOP-1.0A	

Proposed references to be provided during examination: None

Learning Objective: LO21ERGFZ2OB01, State the purpose of FRZ-0.2A/B.

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	
	Level of Difficulty	3

10 CFR Part 55 Content:	55.41	10
	55.43	

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	3		
	Group			
	K/A	G.2.1.15		
	Importance Rating	2.7		

Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

Question # 66

Per ODA-106, Review of Documents and Operational Experience Feedback, Operations Standing Orders...

superseded Operations Standing Orders __________ vaulted as station records.

Operations Standing Orders __________ temporarily contradict procedures until a procedure change is processed.

A. are
may

B. are
may NOT

C. are NOT
may

D. are NOT
may NOT

Answer: B

K/A Match:

The question is a K/A match as it requires the applicant to know how shift orders are handled per the Operations Administrative procedure.

Explanation:

A. Incorrect. First part is correct as described in 'B' below. Second part is incorrect but plausible as the applicant may think that operations standing orders may be used to correct an erroneous procedure until a procedure change can be processed.

B. Correct. First part is correct per ODA-106 which states that superseded operations standing orders are vaulted as station records. Second part is correct per ODA-106 which states that whoever submits a shift order should ensure that it does not contradict Technical Specifications or procedures.

C. Incorrect. First part is incorrect but plausible as the applicant must recall that superseded standing orders are station records that are vaulted and shift orders are not vaulted. Second part is incorrect but plausible as described in 'A' above.

D. Incorrect. First part is incorrect but plausible as described in 'C' above. Second part is correct per ODA-106 which states that whoever submits a standing order should ensure that it does not contradict Technical Specifications or procedures.

Technical Reference(s)	ODA-106	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO22ADMXA1OB01, Administrative Workbook

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 ___
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 10
55.43 ___

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	3		
	Group			
	K/A	G.2.1.38		
	Importance Rating	3.7		

Knowledge of the station's requirements for verbal communications when implementing procedures.

Question # 67

- Unit 2 Reactor tripped from 100% power
- EOS-0.1B, Reactor Trip Response in progress
- The Continuous Action at Step 6 is being directed

Identify the expected verbal communication based on CPNPP verbal requirements during implementation of Emergency Response Guidelines within the Control Room?

A. US: "Joe, this is a Continuous Action step. Verify All AC Busses - Energized by Offsite Power"
RO: "Dave, this is a Continuous Action step"
US: "Joe, that is correct"
RO: "Dave all AC busses are energized by offsite power"
US: "Understand all AC busses are energized by offsite power"

B. US: "Joe, this next step is a Continuous Action step."
RO: "Dave, the next step is a Continuous Action."
US: "Joe, that is correct, Verify All AC Busses - Energized by Offsite Power"
RO: "Dave, that is correct. They are."

C. US: "Continuous Action step. Joe Verify All AC Busses - Energized by Offsite Power"
RO: "Dave all AC busses are energized by offsite power"
US: "Joe all AC busses are energized by offsite power"
RO: "That is correct"

D. US: "Attention in the Control Room. Continuous Action step. End of attention."
RO: "This is a Continuous Action step."
US: "Joe, that is correct, Verify All AC Busses - Energized by Offsite Power"
RO: "Dave all AC busses are energized by offsite power."
US: "Understand they are"

Answer: C

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate the knowledge of proper verbal communication during the performance of an operations procedure.

Explanation:

A. Incorrect. By calling Joe's name, the Unit Supervisor requires a repeat back or action response such as the information requested in the verification. With the Reactor Operator's response of only repeating the first part back this would be incorrect. Further when the Unit Supervisor acknowledges that the busses are all energized, the Reactor Operator does not provide the necessary confirmation.

B. Incorrect. The Unit Supervisor starts incorrectly by requiring Joe to repeat back that the step is a continuous action step. Dave's verification is inappropriate as it is not concise and could be misinterpreted when he states 'they are'. The Unit Supervisor does not provide the necessary confirmation that the information was accurately received.

C. Correct. In accordance with the Guidance, stating the Continuous Action Step without stating a name first is appropriate as a repeat back is not necessary. Then the operator's name is called with the required request. As the request is a verification the operator can provide the requested information as a physical plant change is not required. The Unit Supervisor repeats the information and the Reactor Operator confirms the accuracy. This is a proper communication.

D. Incorrect. The Unit Supervisor starts with a broadcast announcement which is excessive but the Reactor Operator provides a direct response which is outside the communication guidelines. The Unit Supervisor's confirmation is ambiguous in 'understand they are'.

Technical Reference(s)	NMG-114	Attached w/ Revision # See Comments / Reference
	OPGD-3 Attachment 6	

Proposed references to be provided during examination: None

Learning Objective: LO21ADMXA1OB01, Conduct of Operations

Question Source: Bank #
Modified Bank # ILOT0882 (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
	Level of Difficulty	2

10 CFR Part 55 Content:	55.41	10
	55.43	

Original Question: ILOT0882

1	ID: ILOT0882	Points: 1.00

Given the following conditions:

- Unit 2 Reactor tripped from 100% power.
- The crew transitioned to EOS-0.1B, Reactor Trip Response.
- The transition brief was completed.
- The Continuous Action at Step 1 is being directed.

Identify the expected verbal communication based on CPNPP verbal requirements during implementation of Emergency Response Guidelines within the Control Room?

A. US: "Continuous Action, Joe. Check RCS temperature stable at or trending to five five seven degrees."
RO: "Dave, RCS temperature is five six one degrees and slowly lowering."
US: "Joe, RCS temperature is five six one degrees and slowly lowering."
RO: "That is correct."

B. US: "Joe, this next step is a Continuous Action."
RO: "Dave, the next step is a Continuous Action."
US: "Joe, that is correct. Check RCS temperature stable at or trending to five five seven degrees."
RO: "Dave, RCS temperature is five six one degrees and slowly decreasing."

C. US: "Joe, Check RCS temperature stable at or trending to five five seven degrees."
RO: "RCS temperature is five six one degrees and slowly lowering."
US: "Five sixty one and slowly lowering."
RO: "That's right."

D. US: "Attention in the Control Room, Continuous Action. End of announcement."
US: "Joe, Check RCS temperature stable at or trending to five five seven degrees."
RO: "I understand that this step is a Continuous Action."
RO: "Dave, RCS temperature is five six one degrees and slowly decreasing."

Answer: A

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	3		
	Group			
	K/A	G.2.1.41		
	Importance Rating	2.8		

Knowledge of the refueling process.

Question # 68

- The Unit 1 just completed a full core offload outage
- The core contains NO fuel assemblies

Per RFO-102, Refueling Operation, Section 5.5, Reloading the Core, MODE 6 entry will be declared when ______________.

A. containment closure is established in preparation for core reload

B. new fuel assembly is placed in the containment storage racks

C. fuel assembly is engaged by Refueling Machine for core reload

D. irradiated fuel assembly is placed in its designated core position

Answer: C

K/A Match:

The question is a match for the K/A as it requires the applicant to demonstrate knowledge of the refueling process and in particular at what point MODE 6 (refueling) is entered.

Explanation:

A. Incorrect. Plausible because containment closure is established per OPT-408A prior to MODE 6 entry in RFO-102. An applicant could misapply this information and believe that MODE 6 begins when closure is established.

B. Incorrect. Plausible because it is specifically excluded as a MODE 6 entry criteria by RFO-102.

C. Correct. Per Step 5.5.4.A of RFO-102, when the first assembly is engaged by the Refueling Machine MODE 6 is declared and logged.

D. Incorrect. Plausible if thought that MODE 6 is not entered until an assembly is placed in the core. An applicant could reason that the "full core offload" condition exists until there is at least one assembly in the core, and MODE 6 is entered only after the core offload condition no longer exists.

Technical Reference(s)	RFO-102	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21RFOFH2OB02, Apply the administrative requirements of the Fuel Handling system including TS, TRM, and ODCM.

Question Source: Bank # ILOT8071
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 10
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	3		
	Group			
	K/A	G.2.2.13		
	Importance Rating	4.1		

Knowledge of tagging and clearance procedures

Question # 69

- As part of a clearance several normally sealed throttled valves were closed

During restoration, a ________ seal should be reapplied to these valves.

A. Red

B. Green

C. Blue

D. Yellow

Answer: C

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate knowledge the color of seal that would be applied when restoring a clearance.

Explanation:

A. Incorrect. Plausible as this is the color used to identify normally sealed open or off positions.

B. Incorrect. Plausible as this is the color used to identify normally sealed closed positions.

C. Correct. This is the color used to identify normally sealed throttled positions.

D. Incorrect. Plausible as this is the color used for personal safety

Technical Reference(s)	ODA-403	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:

Question Source: Bank #
Modified Bank # ILOT5922 (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 10
55.43

Original Question: ILOT5922

1	ID: ILOT5922	Points: 1.00

While at power, CCW pump 1-01 was tagged out for maintenance. As part of the clearance, several normally sealed open valves were closed.

During restoration what type of seal should be reapplied to these valves?

A. BLUE seals.

B. GREEN seals.

C. YELLOW seals.

D. RED seals.

Answer: D

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	3		
	Group			
	K/A	G.2.2.38		
	Importance Rating	3.6		

Knowledge of conditions and limitations in the facility license.

Question # 70

- 0900 on 4/1/16 – Unit 1 is operating at Rated Thermal Power
- 1200 on 4/1/16 – Unit 1 LEFM is declared unavailable
- 2100 on 4/1/16 – Unit 1 Calorimetric is due

MAXIMUM allowable Unit 1 Thermal Power is ________once LEFM is declared unavailable at 1200.

MAXIMUM allowable Unit 1 Thermal Power is ________ during the performance of the Unit Calorimetric at 2100.

A. 3612 MW_{th}
3612 MW_{th}

B. 3612 MW_{th}
3562 MW_{th}

C. 3575 MW_{th}
3575 MW_{th}

D. 3575 MW_{th}
3562 MW_{th}

Answer: B

K/A Match:

The question matches the K/A as it requires the applicant to know the license conditions restrictions on operation of the plant with the LEFM unavailable.

Explanation:

A. Incorrect. Part 1 is correct as described in 'B' below. Part 2 is incorrect but plausible as the procedure clearly states that the next calorimetric must be performed at the restricted thermal power operation limit of 3562 MW_{th}. Thus the 3612 MW_{th} is plausible if the applicant either did not realize that the restriction existed or believed that the restricted power limit was only AFTER performing the next calorimetric.

B. Correct. Part 1 is correct in accordance with the CPNPP facility license and procedural direction, operation can continue at the Rated Thermal Power of 3612 MW_{th} until the next calorimetric is required. Part 2 is correct in accordance with OPT-309, the next calorimetric must be performed at the restricted thermal power operation limit of 3562 MW_{th}.

C. Incorrect. Part 1 is incorrect but plausible in that the same major section of IPO-003A which details the requirements concerning Power Operations with the LEFM unavailable has restrictions for operating above 3575 MW_{th}, thus the potential for confusion with respect to the operational limits is plausible. Part 2 is incorrect but also plausible based on the assumption contained in Part 1 above and the logic for Part 2 as described in 'A' above.

D. Incorrect. Part 1 as described in 'C' above. Part 2 is correct as described in 'B' above.

Technical Reference(s)	IPO-003A	Attached w/ Revision # See Comments / Reference
	OPT-309	
	Technical Specification 1.1	
	Technical Requirement Manual 13.3.34	

Proposed references to be provided during examination: None

Learning Objective: LO21IPO003OB102, Discuss the actions for operating at constant turbine load in accordance with IPO-003, Power Operations

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 ____
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 5
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	3		
	Group			
	K/A	G.2.3.4		
	Importance Rating	3.2		

Knowledge of radiation exposure limits under normal or emergency conditions.

Question # 71

- An operator has been assigned to escort a Radiation worker
- A TLD has been issued for the Escorted Radiation Worker
- Escorted Radiation Worker has been Authorized to receive maximum dose allowable for an Escorted Radiation Worker at CPNPP

MAXIMUM annual Deep Dose Equivalent administrative exposure levels that can be received by each individual are...

Escorted Radiation Worker can receive ____ mRem

Operator can receive ____ mRem

A. 4000
2000

B. 2000
4000

C. 2000
2000

D. 4000
4000

Answer: C

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge of radiation exposure limits under normal conditions for an escort and escorted radiation worker.

Explanation:

A. Incorrect. Part 1 is plausible since 4000 mrem was the administrative limit until 2008 at CPNPP for Escorted Radiation Worker. Part 2 is correct for the operator. It is plausible that the applicant knows their administrative limit and believes that the limits for CPNPP full time employees are lower than a temporary assignee of a vendor based on the site's strict adherence to ALARA principles. Prior limits were 4000 mrem for both the escort and the escorted radiation worker.

B. Incorrect. Part 1 is correct per current revision of STA-655. Part 2 is incorrect but plausible as 4000 mrem was the previous administrative limit at CPNPP until 2008. It is plausible to believe that the escorted radiation worker would be allowed a dose of one-half of that of a full time radiation worker.

C. Correct. Per STA-655 ATT 8.A these are the administrative limits set for CPNPP.

D. Incorrect. Part 1 is plausible as described in 'A' above. Part 2 is plausible as described in 'B' above.

Technical Reference(s)	STA-655 Current Revision	Attached w/ Revision # See Comments / Reference
	STA-655 Retired Revsion	

Proposed references to be provided during examination: None

Learning Objective: ADMXA1OB103, Radiation Control

Question Source: Bank # ______
Modified Bank # ILOT7247 (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ______
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 12
55.43 ______

Original Question: ILOT7247

1 ID: ILOT7247 Points: 1.00

Given the following conditions:

- Spent Fuel Pool re-racking is in progress.
- An operator has been assigned to escort a Radiation worker in the Spent Fuel Pool area.
- The re-racking is expected to last for one month.
- A TLD and appropriate authorization has been issued for the Escorted Radiation Worker.

Which of the following identifies the Annual Deep Dose Equivalent Administrative Exposure Levels that can be received for both individuals?

A. 5000 mrem for the Escorted Radiation Worker and 2000 mrem for the Operator.

B. 2000 mrem for the Escorted Radiation Worker and 5000 mrem for the Operator.

C. 2000 mrem for the Escorted Radiation Worker and 2000 mrem for the Operator.

D. 5000 mrem for the Escorted Radiation Worker and 5000 mrem for the Operator.

Answer: C

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	3		
	Group			
	K/A	G.2.3.7		
:	Importance Rating	3.5		

Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Question # 72

- Unit 2 is at 30% raising power post outage
- You will be entering Unit 2 containment for Boric Acid Leak inspection
- RWP 20160203 Rev. 00 will be used for inspection
- RP pre-job briefs are complete

Which of the following actions require contacting RP prior to performance?

REFERENCE PROVIDED

A. Tightening a pipe cap not exhibiting full thread engagement

B. Kneeling in an area posted at 150 kdpm/100cm^2

C. Entering an area posted at 75 mRem/hr

D. Inspecting from a scaffold platform at 8' off of the 808' elevation floor

Answer: D

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate the ability to comply with the requirements of an RWP during normal operations.

Explanation:

A. Incorrect. In accordance with the RWP, RP would need to be contacted if a contaminated system breach was necessary as would be expected with indicated Boric Acid leaking. However, tightening a pipe cap without full thread engagement would not be a system breach and would therefore be acceptable without prior contacting RP.

B. Incorrect. In accordance with the RWP, knee pads should be worn when kneeling in a posted high contamination area. High contamination areas are posted at 50 kdpm/100cm^2 thus this area would qualify. However, in accordance with the RWP expected contamination levels are anticipated up to 150 kdpm/100cm^2 and Radiation Worker Stop Instructions are at > 200 kdpm/100cm^2.

C. Incorrect. In accordance with the RWP, entering a High Radiation Area requires contacting RP prior to performance. 75 mRem/hr is less than 100 mRem/hr when the area would be designated as a High Radiation Area. Administrative posting is allowed as low as 80 mRem/hr, thus 75 mRem/hr is used to avoid confusion for the applicants.

D. Correct. In accordance with the RWP, accessing normally inaccessible areas > 7 feet off floor requires contacting RP prior to performance. For the operator to be working on a 8' scaffold, the individual would be accessing areas > 7' off the floor.

Technical Reference(s)	RWP 20160203 Rev. 00	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: RWP 20160203 Rev. 00

Learning Objective: LO21ADMXA1OB02, Radiation Control

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 ______
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 12
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	3		
	Group			
	K/A	G.2.4.39		
	Importance Rating	3.9		

Knowledge of RO responsibilities in emergency plan implementation.

Question # 73

- Crew composition:
  - Shift Manager
  - 3 Senior Reactor Operators
  - Field Support Supervisor position is NOT filled
  - 5 Reactor Operators
  - 8 Nuclear Equipment Operators
  - Non Operations Staffing is at MINIMUM Shift Crew Composition
- An ALERT has been declared on Unit 1
- After initial notifications were completed, NRC requested ENS line be manned by a dedicated individual

Per ODA-102, Conduct of Operations the ______________________ should be assigned to the ENS line, and that individual may also __________________________.

A. Unit 2 Balance of Plant Operator
operate common system equipment

B. Relief Reactor Operator
operate common system equipment

C. Unit 2 Balance of Plant Operator
keep OSC Manager informed of NEO activities

D. Relief Reactor Operator
keep OSC Manager informed of NEO activities

Answer: B

K/A Match:

The question is a K/A match as it requires the applicant to determine which Reactor Operator has specific duties during implementation of the Emergency Plan.

Explanation:

A. Incorrect. The first part is incorrect but plausible as ODA-102, states that the BOP on the unaffected unit is to perform the duties of the Relief Reactor Operator if the Relief Reactor Operator is unavailable. Knowledge of minimum shift staffing shows that the Relief Reactor Operator position is available. The second part is correct as described in ‘B’ below.

B. Correct. In accordance with ODA-102, the Relief Reactor Operator should be assigned these duties as the SROs are at minimum crew staffing and cannot be assigned to the ENS. ODA-102, lists other responsibilities that the Relief Reactor Operator can be asked to do during emergencies and states that they may operate the common equipment as directed.

C. Incorrect. The first part is incorrect but plausible as described in ‘A’ above. The second part is incorrect but plausible as described in ‘D’ below.

D. Incorrect. The first part is correct as described in ‘B’ above. The second part is incorrect but plausible if thought that since the Field Support Supervisor position is unmanned, the Relief Reactor Operator should assume the duty of keeping the OSC Manager informed of NEO activities in lieu of the Field Support Supervisor. This is not one of the duties assigned to the Relief Reactor Operator.

Technical Reference(s)	ODA-102	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21ADMXA1OB04, Emergency Procedures/Plan

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 ____
Level of Difficulty 3

10 CFR Part 55 Content:
55.41 10
55.43 ____

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier	3		
	Group			
	K/A	G.2.4.46		
	Importance Rating	4.2		

Ability to verify that the alarms are consistent with the plant conditions.

Question # 74

A WHITE annunciator alarms in the Control Room.

Responding to the ALARM, an operator would normally expect to identify _______________ in the _______________ band.

A. a single parameter
YELLOW

B. multiple parameters
YELLOW

C. a single parameter
ORANGE

D. multiple parameters
ORANGE

Answer: A

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate the ability to verify that alarm indication is consistent with control room indication.

Explanation:

A. Correct. In accordance with OWI-109, the Yellow Band is used to identify system or equipment values where conditions are out of the normal operating bands. Normally, the bands will be set to correspond to alarm setpoints. Thus, the transition from the Green Band to the Yellow Band should closely correspond to the Alarm setpoint. As only one Alarm is annunciating, this would normally correspond to a single parameter.

B. Incorrect. In accordance with OWI-109, a Yellow or Orange Alarm window would correspond to an event in which multiple parameters would be out of the normal operating band. This distractor is plausible as numerous annunciators would result in these indications but these annunciators should be Yellow or Orange.

C. Incorrect. As the Orange Band is more significant than the Yellow Band, this answer is incorrect but plausible as an individual alarm could result in a meter being in the Orange Band. However, as the Orange Band is to identify values that correspond to conditions requiring operator action or expected system automatic response it is unlikely that only one White annunciator would alarm.

D. Incorrect. As the Orange Band is more significant than the Yellow Band, this answer is incorrect but plausible as numerous annunciators would result in these indications but these annunciators should be Orange or Red. However, as the Orange Band is to identify values that correspond to conditions requiring operator action or expected system automatic response it is unlikely that only one White annunciator would alarm.

Technical Reference(s)	OWI-109	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21ADMXA1, Emergency Procedures/Plan

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 10

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 3	Tier	3		
	Group			
	K/A	G.2.4.50		
	Importance Rating	4.2		

Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Question # 75

An Unexpected Alarm annunciates in the Control Room for the first time during the shift.

Operator <u>first</u> refers to _____________ section of Alarm Procedure (ALM) for corresponding annunciator to check if input has actually exceeded setpoint.

Operator communicates alarm to Unit Supervisor and the ________________________ OPERATOR ACTIONS per corresponding ALM.

A. LOGIC
Unit Supervisor directs

B. PLANT COMPUTER
Unit Supervisor directs

C. LOGIC
Operator performs

D. PLANT COMPUTER
Operator performs

Answer: C

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate the ability to verify the alarm setpoints and the requirements to operate controls identified in the ALM.

Explanation:

A. Incorrect. The first part is correct (See C below). The second part is incorrect but plausible (See B below).

B. Incorrect. The first part is incorrect but plausible (See C below). The second part is incorrect but plausible as any subsequent occurrence of the unexpected alarm would be communicated to the Unit Supervisor and the Unit Supervisor normally directs plant operations but the Unit Supervisor would not direct the operator actions of the ALM as the ALM is an in hand procedure for use by the operator.

C. Correct. In accordance with ODA-205, the LOGIC portion of each annunciator response shows the setpoints and logic which are necessary to annunciate the alarm. The PLANT COMPUTER section could be utilized to determine appropriate computer alarm setpoints but would NOT be the expected first method for verifying the alarm setpoints. In accordance with OPGD-3, Attachment 4 for the first occurrence of an unexpected alarm, the operator should communicate the alarm to the Unit Supervisor and perform the operator actions of the ALM.

D. Incorrect. The first part is incorrect but plausible (See C above). The second part is correct (See C above).

Technical Reference(s)	OPGD-3 Attachment 4	Attached w/ Revision # See Comments / Reference
	ODA-205	

Proposed references to be provided during examination: None

Learning Objective: LO21ADMXA1OB04, Emergency Procedures/Plan

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 ______
Level of Difficulty 2

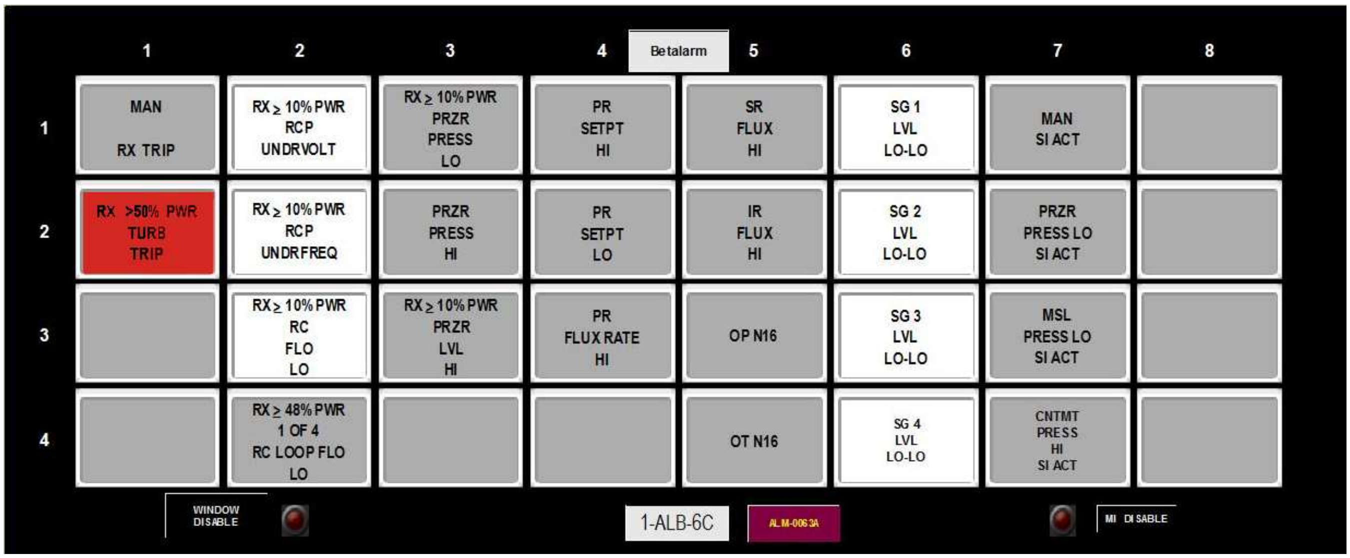
10 CFR Part 55 Content: 55.41 10
55.43 ______

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			1
	Group			1
	K/A	007 EA2.05		
	Importance Rating			3.9

Reactor Trip – Stabilization - Recovery: Ability to determine or interpret the following as they apply to a reactor trip: Reactor trip first-out indication

Question # 76

- Unit 1 Reactor Trip occurred



- The plant is stable in EOS-0.1A, Reactor Trip Response
- Shift Manager has directed a plant cooldown

The initiating event for the Reactor Trip is _______________.

The plant cool down will be performed per ________________________.

A. Loss of Offsite Power
IPO-005A, Plant Cooldown from Hot Standby to Cold Shutdown

B. Reactor Coolant Pump trip
IPO-005A, Plant Cooldown from Hot Standby to Cold Shutdown

C. Loss of Offsite Power
EOS-0.2A, Natural Circulation Cooldown

D. Reactor Coolant Pump trip
EOS-0.2A, Natural Circulation Cooldown

Answer: C

K/A Match:

The question is a K/A match as it requires the applicant interpret from the first out panel indications what caused the reactor trip.

SRO Only:
The question is SRO only level as it requires knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures.

Explanation:

A. Incorrect. First part is correct (See C below). Second part is incorrect but plausible as the applicant may not comprehend that an IPO-005A cooldown cannot be performed without RCPs running.

B. Incorrect. First part is incorrect but plausible as there are three first out annunciators that indicate RCPs are tripped, however the RED annunciator RX>50% TURB TRIP is LIT due to the loss of offsite power and is the cause of the reactor trip. Second part is incorrect but plausible (See A above).

C. Correct. First part is correct because the loss of offsite power caused a main generator lockout which causes the turbine trip that led to the RX>50% TURB TRIP annunciator being lit RED. Second part is correct because with no reactor coolant pumps available and a plant cooldown directed EOS-0.2A must be used.

D. Incorrect. First part is incorrect but plausible (See C above). Second part is correct (See C above).

Technical Reference(s)	ALM-4000A	Attached w/ Revision # See Comments / Reference
	EOS-0.1A	

Proposed references to be provided during examination: None

Learning Objective: ANALYZE the recovery technique used and the procedure steps of EOS-0.1, Reactor Trip Response. (LO21.ERG.E01.OB02)

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 ______
55.43 5

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			1
	Group			1
	K/A	008 G.2.1.7		
	Importance Rating			4.7

Pressurizer Vapor Space Accident: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and interpretation.

Question # 77

- Unit 2 has been operating with leaking PRZR PORV 2-PCV-456
- 2-8000B, PRZR PORV BLK VLV is CLOSED
- Steam Generator fault occurred
- EOP-1.0B, Loss of Reactor or Secondary Coolant in progress
- Following parameters observed:
  - RCS pressure 1200 psig and lowering
  - WR RCS temperature 330℉ and rising
  - 2-PCV-455A, PRZR PORV is OPEN

What action is required in accordance with EOP-1.0B?

A. OPEN 2-8000B, PRZR PORV BLK VLV

B. CLOSE 2-PCV-455A, PRZR PORV

C. CLOSE 2-8000A, PRZR PORV BLK VLV

D. ENSURE 2-PCV-455A, PRZR PORV Closes

Answer: D

K/A Match:

The question is a match to the K/A as it requires the applicant to assess the plant conditions and make an operational judgment in conjunction with procedural knowledge as the appropriate course of action during a pressurizer vapor space accident.

SRO Only:

The question is SRO knowledge as it requires the applicant to assess plant conditions (normal, abnormal, or emergency) and then select a procedure or section of a procedure to mitigate, recover, or with which to proceed.

Explanation:

A. Incorrect. Opening 2-8000B is plausible but incorrect. If the applicant believed that the step required BOTH block valves to be open. However Step 5c clearly states At Least One Open. As 2-8000A is open, the procedure does NOT require 2-8000B be opened as this would introduce an additional RCS leakage path.

B. Incorrect. Closing 2-PCV-455A is plausible but incorrect. If the applicant was to misunderstand the step to only apply to the normal pressurizer PORV setpoint of 2335 and not realize that the pressure/temperature relationship for LTOP was satisfied, the applicant could believe that closing the PORV to stop the RCS leak was appropriate as directed in Step 5b RNO.

C. Incorrect. Closing 2-8000A is plausible but incorrect. As described in ‘B’ above it is plausible to believe that the PORV should not be open at this time. If this was the case the applicant could believe that 2-PCV-455A was the broken component and believe that the appropriate mitigation was to close the block valve as directed in Step 5b RNO.

D. Correct. In accordance with EOP-1.0B Step 5 and Bases, under the given plant conditions 2-PCV-455A should be open and should remain open until the RCS pressure/temperature relationship changes to clear the LTOP setpoints. Once operation is established outside of the LTOP setpoints the valve should automatically close and Step 5 should be performed again as a continuous action step with the result being verification that 2-PCV-455A is closed.

Technical Reference(s)	EOP-1.0B	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21ERGE1AOB104, Discuss the operator actions, including all cautions, notes, RNOs and bases associated with EOP-1.0.

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
	Level of Difficulty	4

10 CFR Part 55 Content:	55.41	
	55.43	5

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			1
	Group			1
	K/A	029 G.2.4.9		
	Importance Rating			4.2

ATWS: Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	
Question # 78	

- Unit 1 performing Beginning of Life startup
- Reactor critical at 10^{-8} amps
- Main Steamline Break (MSLB) inside containment SG 1-03
- Reactor CANNOT be tripped from control room
- FRS-0.1A, Response to Nuclear Power Generation/ATWT in progress
- At FRS-0.1A, Step 10, Check SG Levels

Based on SG levels, throttle AFW flow to minimum of …

A. 0 gpm total flow

B. 150 gpm per SG

C. 460 gpm total flow

D. 860 gpm total flow

Answer: A	

K/A Match:

The question is a K/A match as it requires the applicant to assess an ATWS at low power conditions for the implication on mitigation strategies.

SRO Only:

The question is SRO knowledge as it requires the applicant to assess plant conditions (normal, abnormal, or emergency) and then select a procedure or section of a procedure to mitigate, recover, or with which to proceed.

Explanation:

A. Correct. For the low power situation at BOL the amount of AFW flow which is required to maintain temperature is very low. Therefore the requirement for substantially more AFW flow than is necessary under these conditions would require AFW reduction. Step 10 would address that SG levels which would have initially been at 67% are actually higher now and should be controlled between 50 and 60%, thus reducing AFW flow to 0 gpm would be appropriate within the recovery strategies of Steps 10 and 12.

B. Incorrect. Incorrect but plausible as Attachment 1A for EOP-0.0A and EOS-0.1A have an AFW control philosophy which is to normally throttle AFW flow to 150 to 200 gpm per SG. However, in this low power situation in which 860 gpm has been required, this normal amount greatly exceeds the requirements. The same guidance would require AFW be stopped to SG 1-03.

C. Incorrect. Incorrect but plausible as the normal heat sink requirements are that either SG level is maintained or 460 gpm is maintained. In Step 10 the requirement is either SG level or 860 gpm. For this low power situation, the SG levels would be above the control band and therefore no AFW flow is required.

D. Incorrect. Incorrect but plausible as this step requires either SG level or 860 gpm. For this low power situation, the SG levels would be above the control band and therefore no AFW flow is required.

Technical Reference(s)	FRS-0.1A	Attached w/ Revision # See Comments / Reference
	EOP-0.0A	

Proposed references to be provided during examination: None

Learning Objective: LO21ERGFS1OB104, Given a procedural Step, Note or Caution, Discuss the reason or basis for the Step, Note or Caution in FRS-0.1A/B, Response to Nuclear Power Generation/ATWT

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
	Level of Difficulty	4
10 CFR Part 55 Content:	55.41	
	55.43	5

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			1
	Group			1
	K/A	040 AA2.05		
	Importance Rating			4.5

Steam Line Rupture: Ability to determine and interpret the following as they apply to the Steam Line Rupture: When ESFAS systems may be secured

Question # 79

- Unit 2 has experienced a Steam Line Break outside containment
- A fault occurred on 2ST, STATION SERVICE TRANSFORMER 2ST
- 2EA2, Safeguards 6.9KV received an 86-1 lockout when the Reactor Tripped
- EOP-2.0B, Faulted Steam Generator Isolation, Step 8 in progress
- SG 2-01 NR level is 0%
- All other SG NR levels are 5% to 8%
- AFW total flow is 470 gpm and stable
- RCS subcooling is 52℉
- RCS pressure is 2240 psig and decreasing
- Pressurizer Level is 70% and increasing

TDAFWP ________ be placed in Pull Out.

Unit Supervisor has announced transition to ________________________.

A. may
EOP-1.0B, Loss of Reactor or Secondary Coolant

B. may
EOS-1.1B, Safety Injection Termination

C. may NOT
EOP-1.0B, Loss of Reactor or Secondary Coolant

D. may NOT
EOS-1.1B, Safety Injection Termination

Answer: D

K/A Match:

The question matches the K/A as it requires the applicant to determine and interpret the plant indications concerning whether ESFAS systems (ECCS) can be secured.

SRO Only:

The question is SRO only in that it requires the applicant to exhibit SRO level knowledge of the ERG beyond the overall mitigative strategies and requires the SRO to make a procedure choice beyond the major EOPs.

Explanation:

A. Incorrect. First part is incorrect but plausible (See B below). Second part is incorrect but plausible (See C below).

B. Incorrect. First part is incorrect but plausible if the applicant does not recognize that 2 of the 3 intact SGs are being fed by the TDAFWP even though a single MDAFWP can supply greater than the minimum 460 gpm for heat sink maintenance. Second part is correct (See D below).

C. Incorrect. First part is correct (See D below). Second part is incorrect but plausible if the applicant does not identify RCS pressure decreasing is due to PORV cycling and transition to EOP-1.0B is not appropriate because transition to EOS-1.1B is needed to stop ECCS injection which is causing the PORV to cycle and PRZR level to rise.

D. Correct. First part is correct as the TDAFWP is feeding 2 of the 3 intact SGs and should not be secured. Second part is correct as SI termination criteria per EOP-2.0B are met even though RCS pressure is decreasing the applicant must determine that is due to a PORV cycling due to RCS pressure and PRZR level rising.

Technical Reference(s)	EOP-2.0B	Attached w/ Revision # See Comments / Reference
	EOP-0.0B	

Proposed references to be provided during examination: None

Learning Objective: LO21ERGE2AOB106, Identify the proper transitions out of EOP-2.0.

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
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 ______

55.43 5

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			1
	Group			1
	K/A	054 G.2.4.11		
	Importance Rating			4.1

Loss of Main Feedwater: Knowledge of abnormal condition procedures.	
Question # 80	

- Unit 2 is at 800 MWe
- All 4 Feedwater Isolation Valves CLOSE
- EOS-0.1B, Reactor Trip Response is in progress

Unit Supervisor directs parallel performance of ABN appropriate for initiating event.

Unit Supervisor can assign parallel procedure to ________.

Proper ABN for initiating event is ________.

A. an RO or SRO
ABN-603, Loss of Protection or Instrument Bus

B. ONLY an SRO
ABN-603, Loss of Protection or Instrument Bus

C. an RO or SRO
ABN-604, Loss of Non-1E Instrument Bus

D. ONLY an SRO
ABN-604, Loss of Non-1E Instrument Bus

Answer: A	

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate knowledge of abnormal condition procedures which contain the guidance for the event which generated a loss of main feedwater by closing the Main Feedwater Isolation Valves.

SRO Only:

The question is SRO only because it requires the applicant to demonstrate knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of abnormal and emergency procedures.

Explanation:

A. Correct. Part 1 is correct in accordance with ODA-407; any licensed operator may be assigned the parallel procedure. Part 2 is correct as ABN-603 has guidance that a loss of 2EC1 or 2EC2 will cause the FWIVs to close on Unit 2 (this is a unit difference).

B. Incorrect. Part 1 is incorrect but plausible if believed that ONLY an SRO can be directed to perform an ABN in parallel. However, in accordance with ODA-207 this is not the case. Part 2 is correct as described in ‘A’ above.

C. Incorrect. Part 1 is correct as described in ‘A’ above. Part 2 is incorrect but plausible as ABN-604 has guidance that a loss of 2C2 or 2C3 will cause a loss of forward flow from the Heater Drain Pumps which could appear as a loss of main feedwater as the main feed pumps would lose suction pressure but this does not equate to the instantaneous failure of the FWIVs closing.

D. Incorrect. Part 1 is incorrect but plausible as described in ‘B’ above. Part 2 is incorrect but plausible as described in ‘C’ above.

Technical Reference(s)	ODA-407	Attached w/ Revision # See Comments / Reference
	ABN-603	
	ABN-604	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN603OB102, Analyze the response to a Loss of Instrument Bus in accordance with ABN-603, Loss of Protection or Instrument Bus

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
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 __________

55.43 5

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			1
	Group			1
	K/A	W/E05 EA2.01		
	Importance Rating			4.4

Inadequate Heat Transfer – Loss of Secondary Heat Sink: Facility conditions and selection of appropriate procedures during abnormal and emergency operations	
Question # 81	

- Unit 2 Reactor Trip
- While evaluating plant status in EOS-0.1B, Reactor Trip Response, a RED path occurred on the Heat Sink Critical Safety Function Status Tree
- FRH-0.1B, Loss of Heat Sink is in progress

Subsequently:

- At Step 8, Main Feedwater is established to Steam Generator 2-02
- SG 2-02 wide range level is rising
- CETs are lowering
- SG 2-02 narrow range level is 11% and slowly rising

Per FRH-0.1B, operator actions to establish a heat sink ________ complete.

Transition back to EOS-0.1B, ________ allowed.

A. are
is

B. are
is NOT

C. are NOT
is

D. are NOT
is NOT

Answer: A

K/A Match:

The question matches the K/A as it requires the operator to identify the conditions necessary for selection of the appropriate procedure based on the correct plant indications.

SRO Only:

The question satisfies the criteria for SRO only in accordance with 10 CFR 55.43(b)(5) as knowledge of administrative procedures for proper ERG rules of usage are demonstrated in conjunction with decision points in the EOP contingency procedure on when leaving the procedure should occur.

Explanation:

A. Correct. First part is correct as the actions to establish a heat sink are complete. The operator is only instructed to maintain the flow to the SG, which does not require further operator action. The second part is correct as in accordance with the FRH-0.1B Bases as long as flow is verified and CET temperatures are lowering the transition back to the procedure and step in effect is the correct SRO action.

B. Incorrect. First part is correct (See A above). Second part is incorrect but plausible if thought that the SRO must wait until SG Narrow Range level meets the minimum level for secondary heat sink (Unit difference is employed here as the minimum level for Unit 1 is 43% vice the 10% for Unit 2). The combination is also plausible if believed that the current SG level is adequate but further actions must be taken within the procedure as ERG rules of usage do not allow the SRO to leave an FRG until a defined point of transition is reached. However, these indications meet a defined point of transition.

C. Incorrect. First part is incorrect but plausible if believed that the operator must take further action to establish a secondary heat sink level (Unit difference is employed here as the minimum level for Unit 1 is 43% vice the 10% for Unit 2. Second part is correct (See A above).

D. Incorrect. First part is incorrect but plausible (See C above). Second part is incorrect but plausible (See B above).

Technical Reference(s)	ODA-407 ERG Rules of Usage	Attached w/ Revision # See Comments / Reference
	FRH-0.1B Step 8 and Bases	

Proposed references to be provided during examination: None

Learning Objective: LO21ERGFH1OB104 Given a procedural step, note, or caution, discuss the reason or basis for the step, note, or caution in FRH-0.1

Question Source: Bank #
Modified Bank # NRC 2015 #77 (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
	Level of Difficulty	4

10 CFR Part 55 Content:	55.41	
	55.43	5

Original Question

Given the following Unit 2 conditions:

- A Reactor Trip has occurred.
- While evaluating plant status in EOS-0.1B, Reactor Trip Response, a RED path occurred on the Heat Sink Critical Safety Function Status Tree.
- FRH-0.1B, Loss of Heat Sink is in progress.

Subsequently:

- During performance of Step 8, Main Feedwater is established to Steam Generator 2-02.
- Steam Generator 2-02 wide range level is rising.
- Core Exit Thermocouples are lowering.
- Steam Generator 2-02 Narrow Range Level is 1% and slowly rising.

Which of the following correctly completes the statements below?

1. In accordance with FRH-0.1B, operator actions to establish a heat sink ___(1)___ complete.

2. Transition back to EOS-0.1B, ___(2)___ allowed.

A. (1) are
(2) is

B. (1) are
(2) is NOT

C. (1) are NOT
(2) is

D. (1) are NOT
(2) is NOT

Answer: A

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			1
	Group			2
	K/A	003 AA2.04		
	Importance Rating			3.6

Dropped Control Rod: Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod motion stops due to dropped rod

Question # 82

- Unit 1 at 100% RTP and 1265 MWe near End of Life
- Control Rod Bank D at 215 steps
- 1/1-RBSS, CONTROL ROD BANK SELECT in AUTO
- Control Rod D4 drops into core

Subsequently

- 1-TI-412A, AVE TAVE-TREF DEV. Is -3°F and stable
- 1-NI-41B, PR POWER CHAN I reads 102%
- 1-NI-42B, PR POWER CHAN II reads 102%
- 1-NI-43B, PR POWER CHAN III reads 68%
- 1-NI-44B, PR POWER CHAN IV reads 103%
- Turbine load remains at 1265 MWe

Remaining Control Rod Bank D rods are ____ steps withdrawn.

Per ABN-712, Rod Control System Malfunction ...

Reduce turbine load to MAXIMUM of _____ MW prior to placing 1/1-RBSS in MANUAL.

A. 223
1100

B. 231
1100

C. 223
1228

D. 231
1228

Answer: A

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate the ability to determine where the C-11 control rod stop would stop rods following a dropped control rod.

SRO Only:

The question is SRO only knowledge level as it requires demonstration of the knowledge of diagnostic steps including the action required to be performed based on not obtaining the Action/Expected Response and thus implementing a Response Not Obtained action.

Explanation:

A. Correct. Part 1 is correct as a response to the negative reactivity insertion from the dropped control rod, the RCS average temperature will lower resulting in a demand for rod withdrawal. Control Rod Bank D will withdraw until the C-11 rod stop is reached at 223 steps withdrawn. Part 2 is correct in accordance with ABN-712, when the operator identifies that power is not less than or equal to 100% on the highest reading NI the RNO must be implemented. The RNO requires a load reduction to 1100 MW. Following the load reduction, 1/1-RBSS is placed in MANUAL.

B. Incorrect. Part 1 is incorrect but plausible if the applicant did not account for the C-11 rod stop and in such case the Control Rod Bank D rods would withdraw to the full out position of 231 steps withdrawn as RCS temperature remains well below TREF. Part 2 is correct as described in ‘A’ above.

C. Incorrect. Part 1 is correct as described in ‘A’ above. Part 2 is incorrect but plausible if the Action/Expected Response in combination with the Response Not Obtained of ABN-712 is misunderstood to require the highest reading NI to be less than 100%. (1265 MW/103% = 12.28 MW/%) as the highest reading NI is 3% above 100% this yields (3% x 12.28 MW/% = 37 MW) and finally subtracting this amount from the current load (1265 MW – 37 MW = 1228 MW). Therefore, the applicant could calculate that the necessary load reduction to reduce power below 100% on the highest reading NI would be a turbine load of 1228 MW or less.

D. Incorrect. Part 1 is incorrect but plausible as described in ‘B’ above. Part 2 is incorrect but plausible as described in ‘C’ above.

Technical Reference(s)	ABN-712	Attached w/ Revision # See Comments / Reference
	Rod Control Study Guide	
	Rod Position Indication Study Guide	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN712OB102, Analyze the response to a Dropped or Misaligned Rod in Mode 1 or 2 in accordance with ABN-712, Rod Control System 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	
	Level of Difficulty	3
10 CFR Part 55 Content:	55.41	
	55.43	5

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			1
	Group			2
	K/A	005 G.2.1.32		
	Importance Rating			4.0

Inoperable/Stuck Control Rod: Ability to explain and apply system limits and precautions.

Question # 83

- OPT-106A, Control Rod Exercise is in progress to satisfy SR 3.1.4.2
- Shutdown Bank A (SBA) was inserted to 217 steps
- SBA was withdrawn to 228 steps
- DRPI indication for SBA control rod D14 did not change during withdrawal

LCO 3.1.5, Shutdown Bank Insertion Limits is ___________________________.

When SR 3.1.4.2 is completed a MAXIMUM of ___ hours are allowed to restore SBA rods above insertion limits.

A. NOT applicable during performance of SR 3.1.4.2
two

B. NOT applicable during performance of SR 3.1.4.2
six

C. applicable due to rod control malfunction
two

D. applicable due to rod control malfunction
six

Answer: C

K/A Match:

This question is a K/A match as the question delineates a situation in which a Shutdown Bank rod has become stuck below the insertion limit and it requires the applicant to determine that LCO Note which allows the applicability of the LCO to be suspended during the surveillance performance cannot continue to be used as directed by the Limitations OPT-106A.

SRO Only:

The question is SRO Only as it requires Application of Required Actions with > 1 hour Completion Times.

Explanation:

A. Incorrect. Plausible as with SBA below the Shutdown Bank RIL of 218 Steps withdrawn, the LCO is NOT met. However, a NOTE to the Applicability section of LCO 3.1.5 states the LCO is not applicable while performing SR 3.1.4.2. If the applicant is not aware of the OPT-106A, Limitation 5.2.3 which states that the NOTE is no longer applicable if a malfunction occurs, this answer would be correct. In accordance with Required Action A.2, a maximum of 2 hours is allowed to restore the shutdown banks to within limits.

B. Incorrect. Plausible as with SBA below the Shutdown Bank RIL of 218 Steps withdrawn, the LCO is NOT met. However, a NOTE to the Applicability section of LCO 3.1.5 states the LCO is not applicable while performing SR 3.1.4.2. If the applicant is not aware of the OPT-106A, Limitation 5.2.3 which states that the NOTE is no longer applicable if a malfunction occurs, this answer would be correct with the exception of 6 hours versus the 2 hours actually allowed. The 6 hour time is plausible in that the allowed time to be in MODE 3 is 6 hours.

C. Correct. With SBA below the Shutdown Bank RIL of 218 Steps withdrawn, the LCO is NOT met. A NOTE to the Applicability section of LCO 3.1.5 states the LCO is not applicable while performing SR 3.1.4.2. However, OPT-106A, Limitation 5.2.3 says the NOTE is no longer applicable if a malfunction occurs. The indications of a stuck control rod would be a malfunction thus requiring the suspension of the Applicability Note and LCO 3.1.5 would need to be entered at the current time. In accordance with Required Action A.2, a maximum of 2 hours is allowed to restore the shutdown banks to within limits.

D. Incorrect. Plausible as stated in ‘C’ the first part is correct. The second part of the answer is incorrect in that a maximum of 2 hours versus 6 hours is allowed to restore the shutdown banks to within limits. The 6 hour time is plausible in that the allowed time to be in MODE 3 is 6 hours.

Technical Reference(s)	OPT-106A	Attached w/ Revision # See Comments / Reference
	TS 3.1.4	
	TS 3.1.5	
	Unit 1 COLR	

Proposed references to be provided during examination: None

Learning Objective:	LO21SYSRI1OB108, Apply the administrative requirements of the Rod Control Indication and Rod Insertion Limit (RIL) Monitor Systems, including Technical Specifications, TRM and ODCM.

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
	Level of Difficulty	4

10 CFR Part 55 Content:	55.41	
	55.43	2

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			1
	Group			2
	K/A	067 G.2.2.38		
	Importance Rating			4.5

Plant Fire On-site: Knowledge of conditions and limitations in the facility license.

Question # 84

- Both units are in MODE 1
- A fire is reported in Unit 1 Cable Spreading Room
- Shift Manager ordered Control Room Evacuation
- ABN-803A, Response to a Fire in the Control Room or Cable Spreading Room in progress

Once decision to leave Control Room is made, reactor trip response is per _______________.

Per facility license a MINIMUM of _________ are required to achieve Hot Shutdown on Unit 1.

A. ABN-803A, Response to a Fire in the Control Room or Cable Spreading
1 RO and 1 NEO

B. ABN-803A, Response to a Fire in the Control Room or Cable Spreading
2 ROs and 2 NEOs

C. EOS-0.1A, Reactor Trip Response
1 RO and 1 NEO

D. EOS-0.1A, Reactor Trip Response
2 ROs and 2 NEOs

Answer: B

K/A Match:

The question is a match for the K/A as a fire exists on-site and the applicant is required to demonstrate an understanding of the facility license conditions contained in the Fire Protection Report for having 2 ROs and 2 NEOs for the stated function.

SRO Only:

The question is SRO only in that it specifically addresses conditions and limitations in the facility license and in particular shift staffing requirements. The question additionally has a procedure selection element in that the applicant must know that EGs do not apply to reactor trip response once the decision to leave the control room is made.

Explanation:

A. Incorrect. First part is correct (See B below). Second part is incorrect but plausible if the applicant believes that 2 ROs and 2 NEOs are required for both units to achieve hot shutdown.

B. Correct. First part is correct because once the decision to leave the Control Room has been made, Emergency Response Guidelines (ERGs) DO NOT apply. ERGs may be referred to, but should not be used for Reactor Trip Response. Second part is correct because the FPR specifies a minimum of 2 ROs and 2 NEOs to achieve hot shutdown.

C. Incorrect. First part is incorrect but plausible if the applicant does not recall the note from ABN-803A that states that the ERGs do not apply once the decision leave the control room is made. Second part is incorrect but plausible (See A above). Second part is incorrect but plausible (See A above).

D. Incorrect. First part is incorrect but plausible (See C above). Second part is correct (See B above).

Technical Reference(s)	ABN-803B	Attached w/ Revision # See Comments / Reference
	Fire Protection Report	

Proposed references to be provided during examination: None

Learning Objective: LO21ADMFP1OB104, Explain the administrative requirements of abnormal operations of the Fire 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	
	Level of Difficulty	3
10 CFR Part 55 Content:	55.41	
	55.43	1

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			1
	Group			2
	K/A	W/E 08 EA2.1		
	Importance Rating			4.2

Pressurized Thermal Shock: Ability to determine and interpret the following as they apply to the (Pressurized Thermal Shock). Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Question # 85

- Unit 2 in EOP-3.0B, Steam Generator Tube Rupture due to rupture on SG 2-03
- Loss of Offsite Power occurred with Reactor Trip and Safety Injection
- RCS cooldown in progress
- RCS cold leg 2-01, 2-02 and 2-04 temperatures are 475℉ and slowly lowering
- RCS cold leg 2-03 temperature is 225℉ and slowly lowering
- RCS WR pressure is 1250 psig and lowering rapidly
- Both PRZR PORVs are OPEN

Which of the following should be performed?

A. Remain in EOP-3.0B, Steam Generator Tube Rupture until Safety Injection is terminated

B. Transition to FRP-0.1B, Response to Pressurized Thermal Shock and STOP RCS cooldown

C. Remain in EOP-3.0B, Steam Generator Tube Rupture and CLOSE both PORVs

D. Transition to FRP-0.1B, Response to Pressurized Thermal Shock and CLOSE both PORVs

Answer: A

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate the ability to select the correct procedure and action with regard to PTS during response to a SGTR.

SRO Only:

The question is SRO only level as the applicant must assess plant conditions and then select a procedure to path to mitigate, recover, or with which to proceed.

Explanation:

A. Correct. The procedure and action are appropriate based on the Caution prior to Step 6 in EOP-3.0B.

B. Incorrect. The procedure and action are incorrect but plausible as the applicant may not recall the not prior to Step 6 in EOP-3.0B regarding the loop Tc with the SGTR and transition to FRP-0.1B and stop the cooldown.

C. Incorrect. The procedure is correct but the action is incorrect but plausible because the applicant may not realize the PORVs should be open due to LTOP being armed by the loop with the SGTR and should not be closed.

D. Incorrect. The procedure and action are incorrect but plausible (See B and C above).

Technical Reference(s)	EOP-3.0B	Attached w/ Revision # See Comments / Reference
	FRP-0.1B	

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the symptoms, or Entry Conditions for FRP-0.1 A/B. (LO21.ERG.FP1.OB03)

Question Source: Bank # ILOT8441
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X
Level of Difficulty 4

10 CFR Part 55 Content: 55.41 ______
55.43 1

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			2
	Group			1
	K/A	006 A2.13		
	Importance Rating			4.2

Emergency Core Cooling: Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent SIS actuation

Question # 86

- Unit 1 in MODE 3 waiting to enter MODE 2
- An Inadvertent Safety Injection is actuated

Minimum flow protection for any running Centrifugal Charging Pump is provided by recirculation flow to the ________________________.

Unit Supervisor ________ expected to exercise Procedure Expediency as described in Operations Guideline 3, Attachment 6, Strategies for Successful Transient Mitigation.

A. Refueling Water Storage Tank
is NOT

B. Refueling Water Storage Tank
is

C. Charging Pump suction header
is NOT

D. Charging Pump suction header
is

Answer: B

K/A Match:

The question matches the K/A as it requires the applicant to determine that an Inadvertent Safety Injection has occurred and to predict the impact to the High Head ECCS pumps. The question further requires the applicant to demonstrate knowledge of the procedural expectations for the Unit Supervisor to exhibit procedure expediency in responding to the event in order to satisfy the requirements of the safety analysis.

SRO Only:

The question is SRO only as it requires the applicant to demonstrate knowledge of administrative procedures that specify implementation of emergency procedures and specifically those dealing with a Time Critical Action.

Explanation:

A. Incorrect. First part is correct (See B below). Second part is incorrect but plausible if believed that TCA validations encompass a normal pace and flowpath through the ERG network to satisfy the inadvertent SI event.

B. Correct. First part is correct in that with a Safety Injection actuation the CCP discharge aligns such that normal miniflow and charging lines are isolated and safety injection line and an alternate miniflow back to the RWST are opened. As the RCS pressure remains high (2235 psig) and will increase as fluid is injected from the inadvertent SI, the CCPs are protected by the alternate miniflow lines back to the RWST. Second part is correct per OPGD 3, Att. 6 and STI-214.01, procedure expediency is expected to be used by the Unit Supervisor to prevent the pressurizer from going solid.

C. Incorrect. First part is incorrect but plausible because normal miniflow from the CCPs is directed back to the Charging Pump suction header. Second part is incorrect but plausible (See A above).

D. Incorrect. First part is incorrect but plausible (See C above). Second part is correct (See B above).

Technical Reference(s)	STI-214.01	Attached w/ Revision # See Comments / Reference
	OPGD-3, Att. 6	
	CVCS Study Guide	

Proposed references to be provided during examination: None

Learning Objective: LO21ERGE11OB103, Identify the symptoms/entry conditions for EOS-1.1.

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
	Level of Difficulty	2
10 CFR Part 55 Content:	55.41	
	55.43	2

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			2
	Group			1
	K/A	008 G.2.2.3		
	Importance Rating			3.9

Component Cooling Water: (multi-unit license) Knowledge of the design, procedural, and operational differences between units.	
Question # 87	

- Unit 2 in MODE 4
- IPO-005B, Plant Shutdown from Hot Standby to Cold Shutdown in progress
- All RCPs have been secured
- All SGs are drained to 5% Narrow Range
- CCW Pumps 2-01 and 2-02 running
- CCW Surge Tank Level is lowering with following Annunciators in alarm:
  - 2-ALB-3B, Window 2.4 - CCW SRG TK TRN A LVL HI-HI/LO
  - 2-ALB-3B, Window 1.3 - CCW SRG TK TRN A/B LVL LO-LO
  - CCW Surge Tank levels are slowly lowering on each compartment

Per ABN-502, Component Cooling Water System Malfunctions, when CCW Surge Tank falls below a MAXIMUM level of ________ the affected safeguards loop will isolate.

When affected train CCW pump is placed in PULL OUT, ______________________________ Limiting Condition(s) for Operation must be entered.

A. 33%
ONLY 3.7.7, Component Cooling Water System

B. 57%
ONLY 3.7.7, Component Cooling Water System

C. 33%
3.7.7, Component Cooling Water System, and
3.4.6, RCS Loops – MODE 4

D. 57%
3.7.7, Component Cooling Water System, and
3.4.6, RCS Loops – MODE 4

Answer: C	

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate knowledge of the design and operational differences between the units with respect to the CCW systems.

SRO Only:

The question is SRO only as it requires the applicant to demonstrate knowledge of information contained within a Condition of greater than 1 hour (72 hours) and knowledge of the specifics with respect to the Operability of RCS Loops (SR 3.4.6.2).

Explanation:

A. Incorrect. Part 1 is correct as described in 'C' below. Part 2 is incorrect but plausible as the applicant must determine that both RHR Loops are the required OPERABLE Loops and therefore in accordance with the NOTE in TS 3.7.7 Condition A, TS 3.4.6 must also be entered.

B. Incorrect. Part 1 is incorrect but plausible as 57% is the Unit 1 number in accordance with ABN-502. Part 2 is incorrect but plausible as described in 'A' above.

C. Correct. The tank is common above 37% on Unit 2 and the leak cannot be identified using this methodology until level reaches 33% as specified in ABN-502. Part 2 is correct in that the NOTE in Condition A of TS 3.7.7 requires that the applicable Conditions of LCO 3.4.6 be entered. As all RCPs have been secured and the Steam Generator Levels are below the 10% required by SR 3.4.6.2, the LCO must be entered as only a single RCS Loop is OPERABLE.

D. Incorrect. Part 1 is incorrect but plausible as 57% is the Unit 1 number in accordance with ABN-502. Part 2 is correct as described in 'C' above.

Technical Reference(s)	ABN-502	Attached w/ Revision # See Comments / Reference
	TS 3.7.7	
	TS 3.4.6	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN501OB106, Analyze the response to Leakage Out of the CCW System in accordance with ABN-502, Component Cooling Water System Malfunctions

Question Source: Bank # ____
Modified Bank # ILOT8304 (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
	Level of Difficulty	2

10 CFR Part 55 Content:	55.41	
	55.43	2

Original Question: ILOT8304

1	ID: ILOT8304	Points: 1.00

Given the following conditions:

- Component Cooling Water Pumps 2-01 and 2-02 are in service.
- Component Cooling Water Surge Tank Level is lowering with the following Annunciators in alarm:
- 2-ALB-3B, Window 2.4 - CCW SRG TK TRN A LVL HI-HI/LO
- 2-ALB-3B, Window 1.3 - CCW SRG TK TRN A/B LVL LO-LO
- Component Cooling Water Surge Tank levels are slowly lowering on each compartment.

In accordance with ABN-502, Component Cooling Water System Malfunctions, to identify the source of the leak:

1) Which valves will be closed; and,

2) What level will the side with the leak lower below, while the non-leaking side of the tank remains stable?

A. 1) Non-Safeguards Loop Isolation Valves
   2) 33%

B. 1) Non-Safeguards Loop Isolation Valves
   2) 57%

C. 1) Safeguards Loop Supply and Return Isolation Valves, one Train at a time
   2) 33%

D. 1) Safeguards Loop Supply and Return Isolation Valves, one Train at a time
   2) 57%

Answer: C

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			2
	Group			1
	K/A	013 A2.05		
	Importance Rating			4.2

Engineered Safety Features Actuation: 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: Loss of dc control power

Question # 88

- Unit 1 is at 100%
- A Bus fault results in the loss of 1ED2
- 1-ALB-8B, Window 4.5 TD AFWP STM SPLY VLV LEAKING HV-2452-1/2 annunciates

Action to______________________________ will successfully mitigate the impact on AFW system due to the loss of 1ED2.

Proper procedure selection is ______________________________.

A. place 1-HS-2452-2, AFWPT STM SPLY VLV MSL 1 in PULL-OUT
ALM-0082A, Alarm Procedure 1-ALB-8B

B. place1-HS-2452-2, AFWPT STM SPLY VLV MSL 1 in PULL-OUT
ABN-305, Auxiliary Feedwater System Malfunction

C. reduce turbine load 50 MWe
ALM-0082A, Alarm Procedure 1-ALB-8B

D. reduce turbine load 50 MWe
ABN-305, Auxiliary Feedwater System Malfunction

Answer: D

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate the ability to predict the impact of a loss of DC control power on the ESFAS, and in particular the TDAFW Pump. It further meets the K/A as it requires determination of what procedure and what action is required to be taken in order to control the consequences of the malfunction.

SRO Only:

The question is SRO only as it requires detailed knowledge of the content and specific actions to be taken and requires the applicant to choose which procedure provides the appropriate procedural instruction.

Explanation:

A. Incorrect. This distractor is incorrect but plausible in that with DC control power placing the Steam Admission Valve handswitch in Pull-Out would control the event. However, the loss of DC prevents this action from working and thus does not CLOSE the valve as stated in the distractor. This guidance is contained in BOTH the ALM and ABN and is therefore plausible for both procedures.

B. Incorrect. Incorrect but plausible as described in 'A' above.

C. Incorrect. This distractor is incorrect but plausible as the ALM does provide instruction for closing the valve by placing the handswitch in Pull-Out which will NOT work without control power. However, the ALM does NOT have the necessary guidance about performing the load reduction which is required.

D. Correct. The loss of 1ED2 will cause the TDAFWP Steam Admission Valve to fail open. As such, the TDAFWP will start and the steam demand will increase reactor power to greater than 100%. As local action will be required to isolate the open Steam Admission Valve, ABN-305 instruction is to perform a 50 MWe load reduction.

Technical Reference(s)	ALM-0082A	Attached w/ Revision # See Comments / Reference
	ABN-305	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN305OB106, Analyze the response to an inadvertent Turbine Driven AFW Pump start in accordance with ABN-305, Auxiliary Feedwater System 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	____
	Comprehension or Analysis	X
	Level of Difficulty	2
10 CFR Part 55 Content:	55.41	____________
	55.43	5

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			2
	Group			1
	K/A	039 A2.04		
	Importance Rating			3.7

Main and Reheat Steam: Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunctioning steam dump

Question # 89

- Steam Generator (SG) 1-02 has experienced a Steam Generator Tube Rupture (SGTR)
- 1EC1 is de-energized
- EOP-3.0A, Steam Generator Tube Rupture Step 6, Initiate RCS Cooldown is in progress
- RCS temperature lowers to 553°F but PCIP Window 3. 6, TAVE LO-LO P-12 remains DARK

The following EOP-3.0A action satisfies Unit 1 SGTR safety analysis ___________________.

A MINIMUM of _________ ARV(s) required for a Post SGTR Cooldown.

A. SG 1-01 and SG 1-03 ARVs are OPENED using their CONTROL OVERRIDE and SG 1-04 ARV is fully OPENED using its controller in manual

TWO

B. SG 1-01, SG 1-03 and SG 1-04 ARVs are fully OPENED using their controllers in manual

TWO

C. SG 1-01 and SG 1-03 ARVs are OPENED using their CONTROL OVERRIDE and SG 1-04 ARV is fully OPENED using its controller in manual

ONE

D. SG 1-01, SG 1-03 and SG 1-04 ARVs are fully OPENED using their controllers in manual

ONE

Answer: A

K/A Match:

The question is a K/A match as it requires the applicant to predict the impact of the failure of steam dump valves to open in plant cooldown mode. In conjunction with the SGTR and 1EC1 failure, EOP-3.0A directs that the Control Overrides be used to operate parts of the Main Steam system, namely the ARVs.

SRO Only:

The question is SRO only as it requires Technical Specification Bases knowledge on the ARV requirement for Post SGTR Cooldown to be answered correctly.

Explanation:

A. Correct. Part 1 is correct per EOP-3.0A, with a loss of 1EC1 the ARV controllers for SGs 1 & 3 will not function and the Control Overrides must be used for those two ARVs (Unit difference as these do not exist on Unit 2 and thus the procedure does not have these requirements). Part 2 is correct per Technical Specification Bases as Unit 1 requires two ARVs for Post SGTR Cooldown (Unit difference as Unit 2 only requires one).

B. Incorrect. Part 1 is incorrect but plausible if the applicant did not understand that the failure of 1EC1 has rendered SGs 1 & 3 controllers unavailable for ARV operation. Part 2 is correct as described in ‘A’ above.

C. Incorrect. Part 1 is correct as described in ‘A’ above. Part 2 is incorrect but plausible as this is the Technical Specification Bases for Unit 2.

D. Incorrect. Part 1 is incorrect but plausible as described in ‘B’ above. Part 2 is incorrect but plausible as described in ‘C’ above.

Technical Reference(s)	EOP-3.0A	Attached w/ Revision # See Comments / Reference
	Technical Specification 3.7.4 Bases	
	ABN-709	

Proposed references to be provided during examination: None

Learning Objective: LO21ERGE3A, State the bases for operator actions, notes and cautions for EOP-3.0

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
	Level of Difficulty	3

10 CFR Part 55 Content:	55.41	
	55.43	2

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			2
	Group			1
	K/A	059 G.2.2.44		
	Importance Rating			4.4

Main Feedwater: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question # 90

- Unit 1 is at 100%
- 1/1-RBSS, Control Rod Bank Select is in MAN for troubleshooting

Subsequently the following alarms annunciate:

- 1-ALB-7B Window 1.12, FWPT A TRIP
- 1-ALB-6D Window 1.9, ANY TURB RUNBACK EFFECTIVE
- 1-ALB-6D Window 1.7, ANY CONTROL ROD BANK AT LO LMT
- 1-ALB-6D Window 2.7, ANY CONTROL ROD BANK AT LO-LO LMT

During Turbine Runback, Control Rods should be manually inserted until AVE $T_{AVE} – T_{REF}$ deviation is _______.

Restoring control rods above insertion limit within completion time allows for ____________.

A. 0°F $\pm$ 1.0°F
correctly aligning and starting components

B. 0°F $\pm$ 1.0°F
evaluating and repairing minor problems

C. + 5.0°F
correctly aligning and starting components

D. + 5.0°F
evaluating and repairing minor problems

Answer: D

K/A Match:

The question matches the K/A as it takes a Main Feedwater event and requires the applicant to interpret the control room indications and verify status and proper operator manual response.

SRO Only:

The question is SRO only as it requires knowledge of the Control Rod Insertion Limit Technical Specification Bases to answer. This Action is greater than 1 hour and does not fall in RO knowledge.

Explanation:

A. Incorrect. Part 1 is incorrect but plausible as described in 'B' below. Part 2 is incorrect but plausible as described in 'C' below.

B. Incorrect. Part 1 is incorrect but plausible because 0℉ + 1.0℉ is the normal value expected for AVE $T_{AVE} - T_{REF}$ deviation. However, if the rods are inserted for this time, a significant temperature overshoot occurs which is why the guidance has been placed in OPGD-3. Part 2 is correct as described in 'D' below.

C. Incorrect. Part 1 is correct as described in 'D' below. Part 2 is incorrect but plausible because if SDM is not within limits LCO 3.1.1 Bases states 15 minute completion time is to provide adequate time for the operator to correctly align and start the required systems and components.

D. Correct. Part 1 is correct in accordance with OPGD-3, Attachment 6. Part 2 is correct in accordance with Technical Specification 3.1.6 Bases.

Technical Reference(s)	OPGD-3, Attachment 6	Attached w/ Revision # See Comments / Reference
	ABN-302	
	Technical Specification 3.1.1.Bases	
	Technical Specification 3.1.6.Bases	

Proposed references to be provided during examination: None

Learning Objective: LO21ABN302OB101, Analyze the response to a Feedwater Pump Trip in accordance with ABN-302, Feedwater, Condensate, Heater Drains System Malfunctions

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	
	Level of Difficulty	3

10 CFR Part 55 Content:	55.41	
	55.43	2

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			2
	Group			2
	K/A	027 G.2.4.50		
	Importance Rating			4.0

Containment Iodine Removal: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Question # 91

- Unit 1 in MODE 1
- Both Containment Preaccess Filtration Fans in service for a planned containment entry per STA-620, Containment Entry
- 1-ALB-3A Window 3.4, CNTMT PREACC FILT 17 ΔP HI annunciates

Per 1-ALB-3A, ALM-0031A, STOP the Preaccess Filtration Fan ___ per SOP-801A.

Following Containment Entry, positions of containment personnel air lock hydraulic sump pump discharge isolation valves 1BS-0016 and 1BS-0017 are tracked against LCO ____________________.

A. 11
   3.6.2 Containment Air Lock

B. 11
   3.6.3 Containment Isolation Valves

C. 12
   3.6.2 Containment Air Lock

D. 12
   3.6.3 Containment Isolation Valves

Answer: A

K/A Match:

The question is a K/A match as it requires the applicant to operate the proper controls based on the alarm received.

SRO Only:

The question is SRO only knowledge as the applicant must apply the LCO Tracking Program to determine which T/S LCO containment isolation valve positions will be tracked against.

Explanation:

A. Correct. First part is correct because 1-ALB-3A Window 3.4, CNTMT PREACC FILT 17 ΔP HI response is to stop Preaccess Filtration Fan 11 per SOP-801A. Second part is correct as ODA-308 standard LCOAR for T/S 3.6.2 tracks the listed valve positions until the air lock is tested.

B. Incorrect. First part is correct (See A above). Second part is incorrect but plausible because the listed valves are containment isolation valves but they are not tracked against LCO 3.6.3 per ODA-308 which requires tracking against LCO 3.6.2.

C. Incorrect. First part is incorrect but plausible because if the alarm were 1-ALB-3A Window 4.4, CNTMT PREACC FILT 18 ΔP HI response is to stop Preaccess Filtration Fan 12 per SOP-801A. Second part is correct (See A above).

D. Incorrect. First part is incorrect but plausible (See C above). Second part is incorrect but plausible (See B above).

Technical Reference(s)	ALM-0031A	Attached w/ Revision # See Comments / Reference
	ODA-308 Form 3.6.2	
	SOP-907A and B	

Proposed references to be provided during examination: None

Learning Objective: LO21SYSCL1OB103, Explain the normal, abnormal and emergency operations of the Containment Ventilation 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	
	Level of Difficulty	4

10 CFR Part 55 Content:	55.41	
	55.43	5

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			2
	Group			2
	K/A	035 A2.01		
	Importance Rating			4.6

Steam Generator System: Ability to (a) predict the impacts of the following malfunctions or operations on the S/GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulted or ruptured S/Gs

Question # 92

- Unit 2 Main Steam Line Break inside Containment
- In EOP-0.0B, Reactor Trip or Safety Injection, Containment pressure reached 19 psig
- Transition made to EOP-2.0B, Faulted Steam Generator Isolation with Containment pressure at 22 psig
- EOP-0.0B, Attachment 2 is complete

Critical Safety Function CONTAINMENT Status Tree is __________.

Per 'ERG rules of usage' the Unit Supervisor will ________________________________.

A. RED
complete EOP-2.0B, Faulted Steam Generator Isolation

B. RED
transition to FRZ-0.1B, Response to Containment High Pressure

C. ORANGE
complete EOP-2.0B, Faulted Steam Generator Isolation

D. ORANGE
transition to FRZ-0.1B, Response to Containment High Pressure

Answer: D

K/A Match:

The question matches the K/A as it requires the applicant to predict the impact of the faulted SG on Containment pressure.

SRO Only:

The question is SRO only knowledge as it requires knowledge of application of ‘ERG rules of usage’ to determine which procedure to implement in response to the faulted SG.

Explanation:

A. Incorrect. First part is incorrect but plausible (See B below). Second part is incorrect but plausible (See C below).

B. Incorrect. First part is incorrect but plausible because the applicant may believe that entry into FRZ-0.1B is only done due to a RED path condition on the Containment CSFST. Second part is correct (See D below).

C. Incorrect. First part is correct (See D below). Second part is incorrect but plausible because ODA-407 ‘ERG rules of usage’ would not require transition to FRZ-0.1B if containment pressure had lowered and the Containment CSFST was GREEN. The applicant may think that because the continuous action step (EOP-0.0B, Step 7) was performed that FRZ-0.1B entry is not required.

D. Correct. First part is correct between 18 & 50 psig in containment the ORANGE path is in effect. Second part is correct because ODA-407, ‘ERG rules of usage’ requires entering FRZ-0.1B even after Containment Spray alignment being verified in EOP-0.0B, Step 7 because the ORANGE path still exists.

Technical Reference(s)	ODA-407	Attached w/ Revision # See Comments / Reference
	FRZ-0.1B	
	EOP-0.0B	

Proposed references to be provided during examination: None

Learning Objective:

Question Source: Bank # ILOT5978
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level:
Memory or Fundamental Knowledge X
Comprehension or Analysis
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 ________
55.43 5

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			2
	Group			2
	K/A	034 A2.01		
	Importance Rating			4.4

Fuel Handling Equipment: Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Dropped fuel element

Question # 93

- Unit 2 in MODE 6 reloading core
- Control Room notified an irradiated fuel assembly has been dropped into the core
- ABN-908, Fuel Handling Accident is in progress

Per ABN-908 ...

Fuel Handling Supervisor should ensure the transfer cart is in the ___________ Building with Fuel Transfer Tube gate valve closed.

Containment Purge may have to be stopped to enable ____________________.

A. Fuel
closing the Fuel Transfer Tube gate valve

B. Fuel
installation of the equipment hatch

C. Containment
closing the Fuel Transfer Tube gate valve

D. Containment
installation of the equipment hatch

Answer: B

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate the ability to take action to control the event in accordance with ABN-908.

SRO Only:

The question is SRO only in that it requires the applicant to demonstrate knowledge of tasks performed by the Fuel Handling Supervisor which is an SRO position. It also requires knowledge of the content of the procedure and not just the overall mitigation strategy.

Explanation:

A. Incorrect. First part is correct (See B below). Second part is incorrect but plausible as the applicant may believe that the ΔP between the Containment and Fuel buildings would prevent closing the fuel transfer tube gate valve.

B. Correct. First part is correct per ABN-908, Step 2.3.8; the Fuel Handling Supervisor has this specific responsibility. Second part is correct as a note in ABN-908 informs the user that it may be necessary to secure Containment Purge to enable installation of the equipment hatch.

C. Incorrect. First part is incorrect but plausible as the applicant may believe that the transfer car should be left in Containment in preparation for putting the damaged assembly in the transfer car. Second part is part is incorrect but plausible (See A above).

D. Incorrect. First part is incorrect but plausible (See C above). Second part is correct (See B above).

Technical Reference(s)	ABN-908	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the normal, abnormal and emergency operation of conducting Fuel Handling. (LO21.RFO.FH2.OB01)

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 ______
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 ______
55.43 7

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			3
	Group			
	K/A	G.2.1.2		
	Importance Rating			4.4

Knowledge of operator responsibilities during all modes of plant operation.

Question # 94

Preparations in progress to change from MODE 4 to MODE 3 following Unit 1 refueling outage.

Per IPO-001A, Plant Heatup from Cold Shutdown to Hot Standby, the ___________________ must grant permission to change MODE and the _________________ must approve the MODE change.

A. Plant Manager
   Shift Manager

B. Plant Manager
   Shift Operations Manager

C. Operations Director
   Shift Manager

D. Operations Director
   Shift Operations Manager

Answer: C

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate knowledge of SRO responsibilities with respect to the approval process for making MODE changes.

SRO Only:
The question is SRO only as it details SRO task knowledge of the administrative procedures that specify implementation and coordination of plant normal procedures.

Explanation:

A. Incorrect. First part is incorrect but plausible because if the MODE change is following an RPS/ESF event in which certain criteria are not met the Plant Manager must grant permission for the MODE change. Second part is correct (See C below).

B. Incorrect. First part is incorrect but plausible (See A above). Second part is incorrect but plausible if the applicant believes the SOM approves MODE changes vice the SM.

C. Correct. First part is correct per IPO-001A, Operations Director or Shift Operations Manager has a signoff to grant permission for MODE change. Second part is correct per IPO-001A, the Shift Manager has the approval authority for the MODE change.

D. Incorrect. First part is correct (See C above). Second part is incorrect but plausible (See B above).

Technical Reference(s)	IPO-001A	Attached w/ Revision # See Comments / Reference
	ODA-108	
	ODA-108-1	

Proposed references to be provided during examination: None

Learning Objective: LO21ADMXA1, Conduct of Operations

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 ______
Level of Difficulty 3

10 CFR Part 55 Content: 55.41 ______
55.43 5

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			3
	Group			
	K/A	G.2.1.4		
	Importance Rating			3.8

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.

Question # 95

- An on shift SRO is completing Shift Operations watch bill.
- Four staff SRO work histories are reviewed to determine if they have maintained active license status.

On shift SRO determines only one staff SRO has met the MINIMUM requirements for maintaining active license status per ODA-315, Licensed Operator Maintenance Tracking.

The staff SRO completed five 12-hour shifts during the previous quarter including turnovers, ____.

A. with one four hour absence for makeup of a missed simulator training scenario

B. each shift the SRO had to utilize Short Term Relief to attend a daily meeting

C. with the fifth shift beginning at 1800 on the last day the calendar quarter

D. with the exception of one end of shift turnover due to a family emergency

Answer: B

K/A Match:

The question matches the K/A as it requires the applicant to have knowledge of active license maintenance requirements.

SRO Only:

The question is SRO knowledge in that the applicant must have knowledge of conditions and limitations in the facility license with regard to not meeting administrative controls listed in Technical Specifications Section 5 such as shift staffing requirements.

Explanation:

A. Incorrect. Incorrect but plausible as this is 5 12-hour shifts including turnover. However, the four hour absence is in excess of the allowed Short Term Relief as allowed per OWI-107 and thus that shift would not count.

B. Correct. Per ODA-315, 5 12-hour shifts including turnover are the minimum to maintain an Active License status. Allowances for Short Term Relief per OWI-107 are allowed when completing the shifts per ODA-315.

C. Incorrect. Incorrect but plausible as this is 5 12-hour shifts. However, in accordance with ODA-315, they must all be completed in the previous quarter.

D. Incorrect. Incorrect but plausible as this is 5 12-hour shifts. However, both turnovers must be included per ODA-315 in order to be counted.

Technical Reference(s)	ODA-315	Attached w/ Revision # See Comments / Reference
	OWI-107	

Proposed references to be provided during examination: None

Learning Objective:

Question Source: Bank # ILOT1673
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis
Level of Difficulty 3

10 CFR Part 55 Content: 55.41
55.43 2

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			3
	Group			
	K/A	G.2.2.42		
	Importance Rating			4.6

Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Question # 96

Which situation requires Technical Specification LCO entry per STI-422.01, Operability Determination and Functionality Assessment Program?

A. Completed Immediate Operability Determination for Technical Specification SSC operability is inconclusive.

B. SM requests engineering perform Prompt Operability Determination on Technical Specification SSC degraded condition.

C. Non-conforming condition on Technical Specification SSC is identified and reported to SM.

D. Work Control Center SRO determines Technical Specification SSC will be rendered inoperable during Impact Assessment.

Answer: A

K/A Match:

The question matches the K/A as it requires the applicant to demonstrate an understanding of when in the Operability Determination process entry-level conditions for the Technical Specifications is met.

SRO Only:

The question is SRO only as only SROs may make Operability Determinations and the entire operability determination process is collectively application of Section 3.0 of the Technical Specifications.

Explanation:

A. Correct. In accordance with the definition of Reasonable Expectation in STI-422.01, "A reasonable expectation is a high standard. There is no such thing as an indeterminate operability state; an SSC is either Operable or Not Operable".

B. Incorrect. The answer is plausible as the SM may request a POD when the conclusions reached on the IOD need further support or confirmation. However, a POD is not needed when the component has been determined to be inoperable per the IOD and sufficient information was available during the IOD for that conclusion.

C. Incorrect. The answer is plausible as declaring the equipment inoperable when a non-conforming condition is identified would be considered conservative in some respects; an IOD is required by an on-shift SRO prior to the inoperable declaration.

D. Incorrect. The answer is plausible as the Work Control Center SRO normally would perform Operability Assessments, which are for planned work impact. These assessments are documented in the clearance or work package but the TS entry is not made until such time as necessary during the actual work performance.

Technical Reference(s)	STI-422.01	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: LO21ADMXA2, Equipment Control

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 ____
Level of Difficulty 4

10 CFR Part 55 Content: 55.41 ________

55.43 2

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			3
	Group			
	K/A	G.2.3.11		
	Importance Rating			4.3

Ability to control radiation releases.

Question # 97

- Batch Liquid Radioactive Effluent Release is planned
- XRE-5253 is inoperable

Per ODCM 3.3.3.4, Radioactive Liquid Effluent Monitoring Instrumentation and STA-603, Control of Station Radioactive Effluents ...

Shift Manager may approve release; provided that prior to release two independent samples, collected a MINIMUM of ________ minutes apart are analyzed, and two technically qualified individuals independently verify the release rate calculations and discharge lineup.

Analyses and required verifications are documented on ________.

A. 60
STA-603-13, Batch Radioactive Effluent Release Verification Sheet

B. 15
STA-603-13, Batch Radioactive Effluent Release Verification Sheet

C. 60
ODA-308-ODCM-3.3.3.4.1, Standard LCOAR for ODCM 3.3.3.4.1 Radioactive Liquid Effluent Monitoring Instrumentation – Liquid Radwaste Effluents

D. 15
ODA-308-ODCM-3.3.3.4.1, Standard LCOAR for ODCM 3.3.3.4.1 Radioactive Liquid Effluent Monitoring Instrumentation – Liquid Radwaste Effluents

Answer: B

K/A Match:

The question is a match to the K/A as it requires the applicant to demonstrate the ability to control radiation releases and specifically while operating within the confines of an LCO.

SRO Only:

The question is SRO only as it requires specific knowledge of the administrative controls of implementing specific Required Actions in the ODCM.

Explanation:

A. Incorrect. Part 1 is incorrect but plausible as the minimum recirc time for sampling the tanks are 60 minutes, however, once recirced for the appropriate time the samples need only be separated by 15 minutes. Part 2 is correct as described in ‘B’ below.

B. Correct. Part 1 is correct in that the independent samples are required to be collected a minimum of 15 minutes apart per STA-603-10 and STA-603-13 forms. Part 2 is correct in that although the documentation and tracking of the LCOAR condition is required in accordance with the Standard LCOAR, proper documentation of the analyses and verifications are done in accordance with STA-603-13.

C. Incorrect. Part 1 is incorrect as described in ‘A’ above. Part 2 is incorrect as STA-603-13 is the proper documentation. The Standard LCOAR form is a plausible distractor as the LCO and Required Actions are tracked via the Standard LCOAR, but as can be seen, documentation of the independent analyses and verifications are not delineated within the Standard LCOAR.

D. Incorrect. Part 1 is correct as described in ‘B’ above. Part 2 is incorrect as described in ‘C’ above.

Technical Reference(s)	STA-603	Attached w/ Revision # See Comments / Reference
	ODCM 3.3.3.4	
	ODA-308-ODCM-3.3.3.4.1	

Proposed references to be provided during examination: None

Learning Objective: LO21ADMXA1OB03, Radiation Control

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	
	Level of Difficulty	2

10 CFR Part 55 Content:	55.41	
	55.43	2

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			3
	Group			
	K/A	G.2.3.12		
	Importance Rating			3.7

Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Question # 98

Upper Internals Lift in progress with Fuel Handling SRO on station.

All non-essential personnel are required to leave the Containment elevations of ________.

Fuel Handling Supervisor ________ observe, until the evolution is complete.

A. 832' through 860'
should

B. 860' and above
should

C. 832' through 860'
shall

D. 860' and above
shall

Answer: D

K/A Match:

The question is a K/A match as it requires the applicant to be knowledgeable of radiation worker practices during the Refueling Operation of upper internals lift.

SRO Only:

The question is SRO only as it requires the applicant to demonstrate knowledge of the refueling floor SRO responsibilities.

Explanation:

A. Incorrect. Part 1 is incorrect as described in ‘C’ below. Part 2 is incorrect as described in ‘B’ below.

B. Incorrect. Part 1 is correct as described in ‘D’ below. Part 2 is incorrect in accordance with RFO-102 and the CPNPP Procedure Writers Guide. ‘Should’ is plausible in that it is used to convey management expectations. However, in this case ‘Shall’ is used as observation of the evolution is a Regulatory requirement and may not be deviated. Plausibility for ‘Should’ is also established by the fact that numerous other activities such as the vessel head lift are not Core Alterations and thus would not require observation throughout the evolution by the SRO.

C. Incorrect. Part 1 is incorrect but plausible as the Refueling Operating floor elevation is 832’. It is plausible to believe that as the upper internals may breach the water that 832’ though 860’ should be cleared of non-essential personnel. However, this is not the procedural requirement as adequate shielding exists for several work areas below 860’ elevation but do not in the elevations at 860’ and above. Part 2 is correct as described in ‘D’ below.

D. Correct. Part 1 is correct in accordance with RFO-102. Part 2 is correct in accordance with RFO-102 and the CPNPP Procedure Writers Guide.

Technical Reference(s)	RFO-101	Attached w/ Revision # See Comments / Reference
	RFO-102	
	STA-202	

Proposed references to be provided during examination: None

Learning Objective: LO21RFOFH2OB101, Explain the normal, abnormal and emergency operation of conducting fuel handling.

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 ____
Level of Difficulty 2

10 CFR Part 55 Content: 55.41 ______
55.43 7

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			3
	Group			
	K/A	G.2.4.18		
	Importance Rating			4.0

Knowledge of the specific bases of EOPs.

Question # 99

- Unit 1 Reactor Trip and Safety Injection
- Centrifugal Charging Pump 1-02 tripped
- 86-1 Lockout on 1EA1
- RO informs US that RCP trip criteria on subcooling were exceeded 8 minutes ago and RCPs are still running
- Bottom 4 RVLIS lights LIT
- All SG NR levels 43% to 60%
- AFW flow 500 gpm
- Operator initiated cooldown NOT started

    - US directs tripping RCPs

What are potential consequences of tripping RCPs and bases for this determination?

Severe core uncovery…

A. may occur as greater than 5 minutes have elapsed since RCP trip criteria were met.

B. may occur as conditions for tripping RCPs on loss of subcooling are NOT met.

C. should NOT occur as adequate Auxiliary Feedwater Flow exists for heat removal.

D. should NOT occur as adequate Steam Generator level exists for heat removal.

Answer: A

K/A Match:

The question is a K/A match as it requires the applicant to demonstrate the knowledge of a specific EOP bases which is repeated in various forms throughout the EOPs.

SRO Only:

The question is SRO only knowledge as it demonstrates not only an understanding of the EOP bases but of the Time Significant Action (TSA) which must be met in order to maintain the validity of the EOP bases. As the Timed Operator Action program is a commitment the contents of the procedure fall within conditions and limitation in the facility license.

Explanation:

A. Correct. In accordance with STI-214.01 TSA 2.11, the RCPs must be tripped within 5 minutes of meeting the selected RCP trip criteria. Failure to trip the RCPs within the prescribed time results in increased inventory loss and potential for severe core damage as is detailed in the EOP bases. EOP-0.0A Attachment 1.A Bases is used for detail reference. The bases appears in various locations throughout the ERGs including EOP-0.0, EOP-1.0, EOP-3.0 and FRS-0.1.

B. Incorrect. The answer is incorrect as the conditions for tripping the RCPs is met. The answer is plausible as only 1 Safety Injection pump is still capable of injecting. The applicant could easily believe that a CCP and SIP are required as the stem does not specifically delineate whether the SIP is injecting at this time. It is a common misconception in RCP trip criteria that the pump must be injecting rather than capable of delivering flow to the RCS as described in the EOP-0.0A Bases.

C. Incorrect. Plausible if believed that the existence of a secondary heat sink is sufficient for not uncovering the core if the RCPs were to trip at a later time. However, this is not the case as a SBLOCA requires heat removal via a heat sink but as inventory depletes the secondary heat sink will eventually be lost as heat transmission from the core to the Steam Generators will deteriorate as mass exits the break. Thus although beneficial in heat removal, auxiliary feedwater flow alone does not provide assurance that core uncovery will not occur.

D. Incorrect. Plausible if believed that the existence of a secondary heat sink is sufficient for not uncovering the core if the RCPs were to trip at a later time. However, this is not the case as a SBLOCA requires heat removal via a heat sink but as inventory depletes the secondary heat sink will eventually be lost as heat transmission from the core to the Steam Generators will deteriorate as mass exits the break. Thus although beneficial in heat removal, Steam Generator level alone does not provide assurance that core uncovery will not occur.

Technical Reference(s)	EOP-0.0A Bases	Attached w/ Revision # See Comments / Reference
	STI-214.01	
	Loss of Coolant Accidents Study Guide	

Proposed references to be provided during examination: None

Learning Objective: LO21ERGXD3OB115, Given a plant condition, determine whether or not RCPs should be tripped in accordance with RCP Trip criteria.

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	
	Level of Difficulty	4	
10 CFR Part 55 Content:	55.41		
	55.43	1	

Examination Outline Cross-reference:	Level	RO		SRO
Revision: 2	Tier			3
	Group			
	K/A	G.2.4.40		
	Importance Rating			4.5

Knowledge of SRO responsibilities in emergency plan implementation.

Question # 100

- 0200 Shift Manager (SM) as the Emergency Coordinator declared an ALERT per EPP-201, Assessment of Emergency Action Levels Emergency Classification and Plan Activation
- 0211 Initial Notifications to Offsite Agencies completed per EPP-203, Notifications
- 0229 Escalation criteria for SITE AREA EMERGENCY (SAE) identified by operating crew
- 0242 SM declared a SAE
- 0259 SM approved SAE Notification Form (EPP-203-8) for dissemination to Offsite Agencies

The SAE declaration ________ timely and SAE notification ________ timely.

A. was
was

B. was NOT
was

C. was
was NOT

D. was NOT
was NOT

Answer: C

K/A Match:

The question is a match for the K/A as it details responsibilities which may be assumed by an SRO during emergency plan implementation.

SRO Only:

The question is SRO only as only SRO licensed individuals can assume the role of the Emergency Coordinator prior to the Emergency Coordinator responsibilities being turned over to properly trained staff. As ROs cannot perform these tasks, the question is SRO only.

Explanation:

A. Incorrect. Part 1 is correct as described in ‘C’ below. Part 2 is incorrect but plausible if believed that the 15 minutes for completing the notifications commences when the Emergency Coordinator approves the notification message form. This is incorrect in that approval and dissemination must occur within 15 minutes per EPP-203.

B. Incorrect. Part 1 is incorrect but plausible if believed that the 15 minute time allowance was only for the initial classification and that a shorter duration of 10 minutes was allowed for escalations. Part 2 is incorrect but plausible as described in ‘A’ above.

C. Correct. In accordance with EPP-201, the declaration must be made within 15 minutes of obtaining criteria which warrants an EAL Classification escalation. In accordance with EPP-203-8, notifications cannot be made transmitted to offsite agencies until EPP-203-8 has been approved by the Emergency Coordinator. As the time between the declaration and the notification approval has been 17 minutes, the allotted time of 15 minutes has been exceeded.

D. Incorrect. Part 1 is incorrect but plausible as described in ‘B’ above. Part 2 is correct as described in ‘C’ above.

Technical Reference(s)	EPP-201	Attached w/ Revision # See Comments / Reference
	EPP-203	
	EPP-203-8	
	Shift Manager PAD Task #435	

Proposed references to be provided during examination: None

Learning Objective: ADMXA1OB04, Emergency Procedures/Plan

Question Source: Bank # ____
Modified Bank # ILOT7272 (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
	Level of Difficulty	2

10 CFR Part 55 Content:	55.41	
	55.43	5

Original Question: ILOT7272

1	ID: ILOT7272	Points: 1.00

Given the following conditions:

- At 0200, the Shift Manager (acting as Emergency Coordinator) declared an ALERT based on RCS leakage of 155 gpm.
- At 0211, initial notifications to Offsite Agencies were completed.
- At 0220, the Unit 1 Reactor was tripped and Safety Injection actuated due to increased RCS leakage.
- At 0229, the RCS became saturated and all RVLIS lights indicated DARK.
- At 0247, the Shift Manager (acting as the Emergency Coordinator) declared an escalation of the event to a SITE AREA EMERGENCY.
- At 0259, notification of Emergency Classification escalation to Offsite Agencies was completed.

Which of the following statements is correct regarding the escalation and escalation notification?

A.
- The reclassification was timely
- The notification was **NOT** timely

B.
- The reclassification was **NOT** timely
- The notification was timely

C.
- The reclassification was timely
- The notification was timely

D.
- The reclassification was **NOT** timely
- The notification was **NOT** timely

Answer: B