Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE007 EA2.05	
	Importance Rating	3.4	

Ability to determine or interpret the following as they apply to a reactor trip: Reactor trip first-out indication

RO Question #1

Given the following initial plant conditions:

- The operating crew is performing O-1.2, Plant Startup from Hot Shutdown to Full Load
- Reactor power is 5% and stable

Subsequently, at the following times:

- 1201: Intermediate Range Instrument N-35 fails HIGH
- 1202: Bus 12A – Bus 11A Tie Breaker trips OPEN
- 1203: A LOCA occurs and RCS pressure rapidly lowers to 1000 psig
- 1204: CNMT pressure is 4 psig and rising

Which ONE of the following Annunciators will be flashing RED at time 1205?

A. D-15, RCS LOOP A LOW FLOW

B. D-18, INTERMEDIATE RANGE REACTOR TRIP

C. D-19, PRESSURIZER LO PRESS SI

D. D-28, CONTAINMENT PRESSURE 4 PSI

Answer: **B**

Explanation:

A. INCORRECT. Plausible since loss of power to Bus 11A will result in trip of ‘A’ RCP. Incorrect since reactor power is below the P-8 Permissive setpoint of 25%; therefore, the reactor will not automatically trip.

B. CORRECT. In accordance with P-1, Attachment 1, a single Intermediate Range instrument failing to greater than 25% current equivalent power would result in an automatic reactor trip. This trip can be bypassed once reactor power is greater than P-10 (8%) and the Operator manually blocks the trips by depressing BOTH Intermediate Range Trip Defeat pushbuttons in accordance with O-1.2, Step 6.8.4. Since the initial conditions given indicate reactor power is 5%, the Intermediate Range Reactor Trip has NOT been defeated. Since this would generate the FIRST reactor trip signal, the associated Annunciator (D-18) would be flashing RED.

C. INCORRECT. Plausible since a Safety Injection (SI) actuation, whether manual or automatic, will result in a reactor trip. Incorrect since the reactor would have already received a trip signal from N-35 failing high at time 1201, which would become the first out (RED) Annunciator.

D. INCORRECT. Plausible since high CNMT pressure (4 psig) will result in a Safety Injection signal and the reactor tripping. Incorrect since the reactor would have already received a trip signal from N-35 failing high at time 1201, which would become the first out (RED) Annunciator.

Technical Reference(s): P-1, Reactor Control and Protection System (p40-42, 45; Rev 078)

O-1.2, Plant Startup from Hot Shutdown to Full Load (p45-46; Rev 220)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP00C 1.02; R3501C 1.04

Question Source:	Bank #	X
	Modified Bank #	
	New	

Question History: Last NRC Exam 2014 Braidwood ILT NRC

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

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE008 AK1.01	
	Importance Rating	3.2	

Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Thermodynamics and flow characteristics of open or leaking valves

RO Question #2

Given the following plant conditions:

- Plant tripped from 100% power due to PCV-430, PRZR PORV, failing OPEN
- MOV-516, PRZR PORV Block Valve, can NOT be closed
- Pressurizer pressure is 1165 psig
- PRT pressure is 35 psig

What is the condition of the fluid and the approximate indicated temperature downstream of PCV-430?

A. Saturated steam at 260°F

B. Saturated steam at 281°F

C. Superheated steam at 285°F

D. Superheated steam at 302°F

Answer: D

Explanation:

A. INCORRECT. Plausible if the Applicant uses an enthalpy of 1165 and PRT pressure at 35 psia. Incorrect since isenthalpic throttling of a PORV at 1165 psig (1180 psia) to the PRT at 35 psig (50 psia) results in superheated steam at approximately 302°F.

B. INCORRECT. Plausible if the Applicant uses an enthalpy of 1165 and PRT pressure at 50 psia. Incorrect since isenthalpic throttling of a PORV at 1165 psig (1180 psia) to the PRT at 35 psig (50 psia) results in superheated steam at approximately 302°F.

C. INCORRECT. Plausible if the Applicant fails to convert PRT pressure to 50 psia. Incorrect since isenthalpic throttling of a PORV at 1165 psig (1180 psia) to the PRT at 35 psig (50 psia) results in superheated steam at approximately 302°F.

D. CORRECT. Pressurizer pressure at 1165 psig has an enthalpy of 1185.1. Isenthalpic throttling to 50 psia puts the steam in the superheat region between 280°F and 320°F. Interpolating the curve puts the temperature at approximately 302°F

Technical Reference(s): Steam Tables

Proposed references to be provided to applicants during examination: Steam Tables

Learning Objective: Generic Fundamentals 193004, Thermodynamic Processes, ELO 1.5

Question Source: Bank # X
Modified Bank # ____
New ____

Question History: Last NRC Exam 2016 Byron ILT NRC

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

10 CFR Part 55 Content: 55.41 .5
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE009 G2.1.19	
	Importance Rating	3.9	

Ability to use plant computers to evaluate system or component status. (Small Break LOCA)

RO Question #3

Given the following plant conditions:

- Plant was operating at 100% power
- A LOCA occurred in CNMT
- ‘C’ SI Pump was only SI Pump that initially started
- ECCS equipment that failed upon initial actuation was started by the HCO while performing ATT-27.0, Attachment Automatic Action Verification
- The Operating crew is transitioning out of E-0, Reactor Trip or Safety Injection

Based on the provided PPCS displays:

1) Classify the LOCA size
   AND
2) The earliest time RCP trip criteria was met is at time ___________.

NOTE: **Reference(s) attached**

A. 1) Small Break
   2) 0743

B. 1) Small Break
   2) 0748

C. 1) Large Break
   2) 0743

D. 1) Large Break
   2) 0748

Answer: **B**

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since there is SI flow to both loops. Incorrect since the indicated flow is indicative of only a single SI Pump in operation. Therefore, RCP trip criteria is not met with only one SI Pump running.

B. CORRECT. According to E-1 Background Document, break sizes from 3/8" up to one foot in diameter are considered a small break LOCA. Sizes greater than one foot are classified as a large break LOCA and "RCS rapidly depressurizes to values close to the containment atmospheric pressure." Additionally, FR-P.1 Background Document states "For transients where RCS pressure is less than the RHR pump shutoff head (350 psig) and flow from the RHR pumps has been verified, the operator should return to the procedure and step in effect since these symptoms are indicative of a large-break LOCA." Therefore, the given information indicates a small break LOCA is in progress. E-0 Foldout Page states RCP Trip Criteria as: "SI pumps – AT LEAST TWO RUNNING; and RCS pressure minus maximum S/G pressure – LESS THAN 210 PSI [240 psi adverse CNMT]". At time 0743, the indicated SI flow is that of only one SI Pump running; therefore, RCP Trip Criteria is not met until approximately 0748 when the HCO restarts the tripped SI Pumps resulting in higher SI flow indicated and RCP Trip Criteria being met.

C. INCORRECT. The first part is plausible since RCS pressure lowered rapidly to less than S/G pressures and the Applicant may misinterpret this as meeting the criteria for a large break LOCA. Incorrect since RCS pressure is above RHR Pump shutoff head. The second part is plausible since there is SI flow to both loops. Incorrect since the indicated flow is indicative of only a single SI Pump in operation. Therefore, RCP trip criteria is not met with only one SI Pump running.

D. INCORRECT. The first part is plausible since RCS pressure lowered rapidly to less than S/G pressures and the Applicant may misinterpret this as meeting the criteria for a large break LOCA. Incorrect since RCS pressure is above RHR Pump shutoff head. The second part is correct.

Technical Reference(s): E-0, Reactor Trip or Safety Injection (Foldout Page p1; Rev 049)

E-1 Background Document (p3; Rev 024)

FR-P.1 Background Document (p14; Rev 007)

Proposed references to be provided to applicants during examination: PPCS Displays

Learning Objective: R2701C 1.07b; REP00C 1.04

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge ____________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .10

55.43 ____________

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE011 EA1.01	
	Importance Rating	3.7*	

Ability to operate and monitor the following as they apply to a Large Break LOCA: Control of RCS pressure and temperature to avoid violating PTS limits

RO Question #4

Given the following plant conditions:

- Plant was operating at 100% power
- A Design Basis LOCA occurred
- The operating crew entered the appropriate EOP
- A RED condition on the Integrity Critical Safety Function Status Tree developed
- The operating crew entered FR-P.1, Response to Imminent Pressurized Thermal Shock Condition

Which ONE of the following correctly completes the statement below?

After checking RCS pressure and ____(1)____ flow, the US will ____(2)____ FR-P.1.

A. (1) RHR
(2) exit

B. (1) RHR
(2) continue in

C. (1) SI
(2) exit

D. (1) SI
(2) continue in

Answer: A

Explanation:

A. CORRECT. In accordance with FR-P.1, Step 1 RNO, **IF** RCS pressure is not greater than 300 psig [350 psig adverse CNMT] the Operator is directed to check RHR flow greater than 475 gpm and then return to procedure and step in effect. According to FR-P.1 Background Document "For transients where RCS pressure is less than the RHR pump shutoff head and flow from the RHR pumps has been verified, the operator should return to the procedure and step in effect since these symptoms are indicative of a large-break LOCA. In this instance, the actions of FR-P.1 should not be performed since pressurized thermal shock is not a serious concern for a large-break LOCA."

B. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret that once FR-P.1 is entered, the procedure must be completed prior to transition out of FR-P.1. Incorrect since FR-P.1 is exited once it is determined that the RCS cannot be repressurized.

C. INCORRECT. The first part is plausible since the Operator will check SI and RHR Pump flows during the performance of ATT-27.0 and the Operator may misinterpret the step in FR-P.1 as referring to SI Pump flow. Incorrect since RHR flow is checked to determine whether or not RCS pressure can become a PTS concern. The second part is correct.

D. INCORRECT. The first part is plausible since the Operator will check SI and RHR Pump flows during the performance of ATT-27.0 and the Operator may misinterpret the step in FR-P.1 as referring to SI Pump flow. Incorrect since RHR flow is checked to determine whether or not RCS pressure can become a PTS concern. The second part is plausible since the Applicant may misinterpret that once FR-P.1 is entered, the procedure must be completed prior to transition out of FR-P.1. Incorrect since FR-P.1 is exited once it is determined that the RCS cannot be repressurized.

Technical Reference(s): FR-P.1, Response to Imminent Pressurized Thermal Shock Condition (p3; Rev 034)

FR-P.1 Background Document (p14; Rev 007)

Proposed references to be provided to applicants during examination: None

Learning Objective: RFRP1C 2.01

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE022 AA2.04	
	Importance Rating	2.9	

Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: How long PZR level can be maintained within limits

RO Question #5

Given the following plant conditions:

- Reactor power is 75% and stable
- A loss of all Charging occurs
- HCO has immediately isolated Letdown
- Pressurizer level is on program
- Pressurizer level is trending down at approximately 1% every 10 minutes

With **NO** further action by the operating crew, how much time will elapse before the procedurally directed reactor trip setpoint is reached?

A. 270 minutes

B. 360 minutes

C. 420 minutes

D. 510 minutes

Answer: **C**

Explanation:

Pressurizer level is programmed as a linear function from the No-Load value of 20% to 56% at full power. The following calculation is performed to determine the programmed PRZR level at 75% reactor power:

PRZR Level = 20% + (P_{Rx} x 36%/100%) = 20% + (75% x 0.36) = 20% + 27% = **47%**

To determine time to reach the reactor trip setpoint in AP-CVCS.3, the Operator must perform the following calculation to calculate the time it will take from an initial PRZR level of 47% to the procedurally directed reactor trip setpoint of 5%:

TIME = (47% - 5%) x 10 min/% = 42% x 10 min/% = **420 minutes**

A. INCORRECT. Plausible since the Applicant may correctly determine that the initial PRZR level is 47%; however, misinterpret that the procedurally directed reactor trip setpoint is 20%. Incorrect since AP-CVCS.3, Step 6 RNO directs the Operator to trip the reactor at 5% PRZR level.

B. INCORRECT. Plausible since the Applicant may incorrectly determine that the initial PRZR level is 56% (normal 100% power level) and misinterpret that the procedurally directed reactor trip setpoint is 20%. Incorrect since initial PRZR level would be programmed at 47% and AP-CVCS.3, Step 6 RNO directs the operator to trip the reactor at 5% PRZR level.

C. CORRECT. According to P-10, Section 5.3 “The reference level is programmed as a function of Average T_{AVG} (between 20% and 56% level) as T_{AVG} changes from 547°F to 574°F.” (0% to 100% reactor power) “The controlling pressurizer level channel will attempt to regulate charging flow to maintain measured level equal to programmed level.” Therefore, at 75% reactor power the PRZR level would be approximately 47%. AP-CVCS.3, Step 6 RNO directs the operator to trip the reactor and go to E-0 IF PRZR level is NOT GREATER THAN 5%. At a level reduction rate of 1%/10 minutes (given), it will take approximately 420 minutes for PRZR level to lower from 47% to 5%.

D. INCORRECT. Plausible since the Applicant may incorrectly determine that the initial PRZR level is 56% (normal 100% power level) and correctly know the correct trip setpoint (5%). Incorrect since at 75% reactor power, initial PRZR level would be 47%.

Technical Reference(s): AP-CVCS.3, Loss of All Charging Flow (p4; Rev 016)
P-10, Instrument Failure Reference Manual (p22; Rev 022)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP31C 2.01; R1401C 1.09

Question Source:	
Bank #	
Modified Bank #	
New	X

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE025 AK1.01	
	Importance Rating	3.9	

Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation

RO Question #6

Given the following plant conditions:

- Plant is in MODE 5
- RCS is at reduced inventory
- Both RHR Pumps trip and can NOT be started
- The operating crew enters the appropriate Abnormal Operating Procedure (AP)

Which ONE of the following correctly completes the statements below?

1) In accordance with Technical Specification LCO 3.4.8, RCS Loops – MODE 5, Loops Not Filled, the Operators must initiate action to restore one RHR loop to OPERABLE status and in operation __________________.

   AND

2) All open Containment penetrations must be closed within two hours or _________, whichever is less.

A. 1) immediately
   2) time to boil

B. 1) immediately
   2) time to core uncovery

C. 1) within 15 minutes
   2) time to boil

D. 1) within 15 minutes
   2) time to core uncovery

Answer: **A**

Explanation:

A. CORRECT. Technical Specification 3.4.8, Condition B.2 Required Action “Initiate action to restore one RHR loop to OPERABLE status and operation” with a Completion Time of “Immediately”. AP-RHR.2, Step 4 directs the Operator to “Initiate CNMT Closure (Refer to O-2.3.1A, CONTAINMENT CLOSURE CAPABILITY IN TWO HOURS DURING RCS REDUCED INVENTORY OPERATION)” O-2.3.1.A, Step 1.1 states “This procedure provides instructions necessary to perform the following: Establish containment closure within Time-To-Boil **OR** 120.0 minutes (whichever is less) should RHR cooling be lost.”

B. INCORRECT. The first part is correct. The second part is plausible since time to boil and time to core uncovery are tracked by the Operators while the RCS is in reduced inventory and the Applicant may misinterpret that the required time to establish CNMT closure is based on two hours or time to core uncovery, whichever is less. Incorrect since O-2.3.1A requires CNMT closure within two hours or Time-to-Boil, whichever is less.

C. INCORRECT. The first part is plausible since Technical Specification LCO 3.4.8 contains a NOTE “All RHR pumps may be de-energized for ≤ 15 minutes when switching from one loop to another provided” and the Applicant may misinterpret this time with the Completion Time for Condition B. Incorrect since Technical Specification LCO 3.4.8, Condition B has a Completion Time of “Immediately”. The second part is correct.

D. INCORRECT. The first part is plausible since Technical Specification LCO 3.4.8 contains a NOTE “All RHR pumps may be de-energized for ≤ 15 minutes when switching from one loop to another provided” and the Applicant may misinterpret this time with the Completion Time for Condition B. Incorrect since Technical Specification LCO 3.4.8, Condition B has a Completion Time of “Immediately”. The second part is plausible since time to boil and time to core uncovery are tracked by the Operators while the RCS is in reduced inventory and the Applicant may misinterpret that the required time to establish CNMT closure is based on two hours or time to core uncovery, whichever is less. Incorrect since O-2.3.1A requires CNMT closure within two hours or Time-to-Boil, whichever is less.

Technical Reference(s):	
	Technical Specification LCO 3.4.8, RCS Loops – MODE 5, Loops Not Filled (Amendment 139)
	AP-RHR.2, Loss of RHR While Operating at RCS Reduced Inventory Conditions (p4; Rev 01800)
	O-2.3.1A, Containment Closure Capability Within Two Hours During RCS Reduced Inventory Operation (p4; Rev 027)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP25C 2.01; ROP14C 1.01; R2501C 1.12

Question Source:	Bank #		
	Modified Bank #	X	
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	.10	
	55.43		

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE027 AK2.03	
	Importance Rating	2.6	

Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners

RO Question #7

Given the following plant conditions:

- Plant is operating at 100% power
- P/429A, PRESSURIZER PRESSURE DEFEAT switch is in NORMAL
- PRZR PRESS CONTROLLER, 431K, fails and its output slowly lowers to 0%

Assuming **NO** Operator action is taken, which ONE of the following identifies the system response to this failure?

PRZR Spray Valves will . . .

A. close, PRZR Heaters will energize, one PORV will lift

B. close, PRZR Heaters will energize, two PORVs will lift

C. open, Reactor will trip, RCS pressure will stabilize above SI Pump shutoff head

D. open, Reactor will trip, RCS pressure will stabilize below SI Pump shutoff head

Answer: **A**

Explanation:

A. CORRECT. PRZR Pressure Controller, 431K, failing low has the same effect as the controlling PRZR pressure channel failing low. According to P-10, Section 5.2.3.3.c the following plant response results: Control and Backup heaters ON, PORV-431C is prevented from opening, PORV-430 will still operate. Additionally, “A sensed low pressure would turn heaters ON and drive the actual pressure abnormally high. PORV actuation from alternate sensors ensures against high pressure damage from this transient.”

B. INCORRECT. Plausible since the PRZR Spray Valve and Heater response is correct and the Applicant may misinterpret that since only a controller failure is given, that the PORV inputs from the four PRZR Pressure detectors will still function to open both PORVs. Incorrect since PT-449 provides the PCV-431C input via the PRZR Pressure Controller, 431K, which has failed low; therefore, PCV-431C is prevented from opening.

C. INCORRECT. Plausible since this would be the plant response if the PRZR Pressure Controller, 431K, output had failed high and the Applicant may misinterpret the operation of PRZR Pressure Controller, 431K. Additionally, once RCS pressure lowers to the automatic SI setpoint (1750 psig), a CNMT Isolation will occur resulting in the PRZR Spray Valves closing and RCS depressurization stops. This results in RCS pressure stabilizing between 1700 – 1750 psig, above the SI Pump shutoff head. Incorrect since PRZR Pressure Controller, 431K, output failing low provides plant response similar to that for a low-pressure condition.

D. INCORRECT. Plausible since this would be the plant response if the PRZR Pressure Controller, 431K, output had failed high and the Applicant may misinterpret the operation of PRZR Pressure Controller, 431K. Additionally, the Operator may misinterpret that upon automatic SI actuation the PRZR Spray Valves remain open resulting in continued RCS depressurization until the SI Pump flow rate makes up for the pressure reduction caused by the failed open PRZR Spray Valves. This misinterpreted condition would result in RCS pressure stabilizing below the shutoff head of the SI Pumps. Incorrect since PRZR Pressure Controller, 431K, output failing low provides plant response similar to that for a low-pressure condition and PRZR spray valves would close upon actuation of SI/CI.

Technical Reference(s): P-10 Instrument Failure Reference Manual (p15-21; Rev 022)

Proposed references to be provided to applicants during examination: None

Learning Objective: RIC02C 1.06

Question Source: Bank # ______
Modified Bank # X
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE029 EK3.08	
	Importance Rating	3.6*	

Knowledge of the reasons for the following responses as they apply to the ATWS: Closing the main steam isolation valve

RO Question #8

Given the following plant conditions:

- The operating crew is responding to a loss of feedwater ATWS
- The Operating crew is performing Immediate Actions of FR-S.1, Response to Reactor Restart/ATWS
  - HCO has started inserting control rods
  - CO was unable to manually trip the turbine and is closing the MSIVs

What is the basis for closing both Main Steam Isolation Valves (MSIVs) during Immediate Actions for this event in accordance with FR-S.1 Background Document?

A. To prevent Main Turbine overspeed and damage

B. To force RCS heatup and thereby reactor shutdown

C. To maintain S/G inventory available as a secondary heat sink

D. To prevent a S/G tube creep rupture from occurring

Answer: C

Explanation:

A. INCORRECT. Plausible since the Applicant may misinterpret that the RCS heatup due to the ATWS will cause an increase in S/G pressure and result in turbine overspeed. Incorrect since the basis for closing MSIVs during a loss of feedwater ATWS is to maintain S/G inventory.

B. INCORRECT. Plausible since tripping the turbine or closing the MSIVs will result in RCS heatup and add negative reactivity. Incorrect since the basis for closing MSIVs during a loss of feedwater ATWS is to maintain S/G inventory.

C. CORRECT. According to FR-S.1 Background Document "For an ATWS event where a loss of normal feedwater has occurred, analyses have shown that a turbine trip is necessary (within 30 seconds) to maintain S/G inventory."

D. INCORRECT. Plausible since this would be the failure mechanism for a sustained hot dry S/G which would occur if the MSIVs remained OPEN without a source of makeup in accordance with FR-H.1 Background Document. Incorrect since the basis for closing MSIVs during a loss of feedwater ATWS is to maintain S/G inventory.

Technical Reference(s): FR-S.1, Response to Reactor Restart/ATWS (p3; Rev 025)

FR-S.1 Background Document (p33; Rev 011)

FR-H.1 Background Document (p61-65; Rev 009)

Proposed references to be provided to applicants during examination: None

Learning Objective: RFRS1C 1.07

Question Source: Bank # ____

Modified Bank # ____

New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 .10

55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE038 EK3.06	
	Importance Rating	4.2	

Knowledge of the reasons for the following responses as they apply to the SGTR: Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures

RO Question #9

Given the following plant conditions:

- The plant was operating at 100% power when a SGTR on ‘B’ S/G occurred
- The crew has transitioned to E-3, Steam Generator Tube Rupture
- Charging and Normal Letdown have been established
- RCS cooldown and depressurization steps are complete
- CNMT pressure is 0.2 psig and stable
- ‘A’ S/G level is 52% and stable
- ‘B’ S/G level is 32% and slowly lowering
- PRZR level is 40% and slowly rising

Based on the given plant conditions, which ONE of the following describes:

(1) the required Operator action in accordance with E-3;

AND

(2) the reason ES-3.1, Post SGTR Cooldown Using Backfill, would be the preferred procedure?

A. (1) raise Charging flow
(2) minimizes radiological releases

B. (1) energize PRZR Heaters
(2) minimizes radiological releases

C. (1) raise Charging flow
(2) minimizes adverse effects on primary system components

D. (1) energize PRZR Heaters

(2) minimizes adverse effects on primary system components

Answer: **B**

Explanation:

A. INCORRECT. The first part is plausible since raising Charging flow is an action used to control PRZR and ruptured S/G levels; however, with PRZR level greater than 20%, raising Charging flow would not be an option. Incorrect since with the given PRZR level, energizing PRZR Heaters is the required action. The second part is correct.

B. CORRECT. In accordance with E-3, Step 38, with PRZR level between 50% and 75% and Ruptured S/G narrow range level lowering, the Operator is directed to Energize PRZR Heaters. Additionally, according to E-3 Background Document, "In general, post-SGTR cooldown using backfill is the preferred method since it minimizes radiological releases and facilitates processing of contaminated primary coolant."

C. INCORRECT. The first part is plausible since raising Charging flow is an action used to control PRZR and ruptured S/G levels; however, with PRZR level greater than 20%, raising Charging flow would not be an option. Incorrect since with the given PRZR level, energizing PRZR Heaters is the required action. The second part is plausible since this is an advantage for using ES-3.2, Post SGTR Cooldown Using Blowdown. Incorrect since ES-3.1 allows S/G water to backfill into the RCS which minimizes radiological releases.

D. INCORRECT. The first part is correct. The second part is plausible since this is an advantage for using ES-3.2, Post SGTR Cooldown Using Blowdown. Incorrect since ES-3.1 allows S/G water to backfill into the RCS which minimizes radiological releases.

Technical Reference(s): E-3, Steam Generator Tube Rupture (p37; Rev 051)
E-3 Background Document (P164; Rev 022)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP03C 2.01; RES31C 1.01

Question Source: Bank # X
Modified Bank # ____
New ____

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE040 AK2.02	
	Importance Rating	2.6*	

Knowledge of the interrelations between the Steam Line Rupture and the following: Sensors and detectors

RO Question #10

Given the following MCB indications immediately after a large steam break:

- PI-468, PI-469, and PI-482A STEAM GEN A PRESSURE: 200 psig and lowering
- PI-478, PI-479, and PI-483A, STEAM GEN B PRESSURE: 1050 psig and lowering
- PI-484, STEAM HEADER PRESSURE: 100 psig and lowering
- FI-464, FI-465, and FI-498, STEAM GEN A STEAM FLOW: 4.6 x 10^6 lbm/hr and stable
- FI-474, FI-475, and FI-499, STEAM GEN B STEAM FLOW: 0.1 x 10^6 lbm/hr and stable
- PI-485 and PI-486, TURBINE 1ST STG PRESS: 75 psig and lowering

Which ONE of the following is the location of the steam break?

A. Between ‘A’ S/G and ‘A’ S/G flow detectors

B. Between ‘A’ S/G flow detectors and ‘A’ MSIV

C. Between the MSIVs and PT-484

D. Between the Turbine Stop Valves and the HP Turbine

Answer: B

Explanation:

A. INCORRECT. Plausible since ‘A’ S/G pressure is low. Incorrect since if the break was located here, ‘A’ S/G steam flow would be approximately zero.

B. CORRECT. ‘A’ S/G pressure low with ‘B’ S/G pressure high indicates separation between S/Gs, and an automatic SI condition on low S/G pressure (< 514 psig in ‘A’ S/G). It can be assumed that ‘A’ MSIV is closed due to SI + High-High Steam Flow (4.40x10^6 lbm/hr). With high steam flow indicated and the ‘A’ MSIV closed, the break must be upstream of the ‘A’

MSIV.

C. INCORRECT. Plausible since PI-484 indicates a low pressure. Incorrect since if the break was located here, both S/G pressures would be equal.

D. INCORRECT. Plausible since this is the lowest indicated pressure. Incorrect since if the break was located here, both S/G pressures would be equal.

Technical Reference(s): 33013-1231, Sheet 1, Main Steam (MS) (Safety Related) P&ID (Rev 4)

33013-1232, Main Steam Non-Safety Related (MS) P&ID (Rev 43)

P-1, Reactor Control and Protection System (p45; Rev 078)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4001C 1.04 & 1.06

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .7

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE054 AK3.03	
	Importance Rating	3.8	

Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): Manual control of AFW flow control valves

RO Question #11

Given the following initial plant conditions:

- Plant is operating at 65% reactor power
- 'B' Main Feedwater (MFW) Pump tripped
- The operating crew entered AP-FW.1, Abnormal MFW Pump Flow or NPSH
- CO started all 3 AFW Pumps and verified flow
- The operating crew commenced load reduction in accordance with AP-TURB.5, Rapid Load Reduction

Given the following current plant conditions:

- The load reduction has been stopped and reactor power stabilized
- Both S/G levels: 59% and slowly rising
- Both Feedwater Regulating Valves (FRVs) at 30% OPEN in AUTO
- All 3 AFW Pumps running and providing flow
- The operating crew is performing AP-FW.1, Step 12, Establish Stable Plant Conditions

Which ONE of the following correctly states the reason the US directs the CO to manually throttle AFW flow in accordance with AP-FW.1, Step 12?

A. To prevent MFW isolation

B. To allow restoration of S/G blowdown flow

C. To allow FRVs to restore S/G water level

D. To restore AFW System to normal alignment

Answer: **C**

Explanation:

A. INCORRECT. Plausible since MFW Isolation occurs at 85% S/G water level. Incorrect since AP-FW.1 Background Document states the reason for throttling AFW flow is to allow the FRVs to restore S/G water level.

B. INCORRECT. Plausible since S/G blowdowns were isolated when the AFW Pumps were started and are to be restored in Step 15 and the Applicant may misinterpret that this is the reason for securing AFW flow in Step 12 in preparation for restoring S/G blowdown flow. Incorrect since the AFW flow is secured to allow FRVs to restore S/G water level in Step 12.

C. CORRECT. According to AP-FW.1 Background Document Step 12 "directs throttling AFW if S/G levels are high. This allows the MFW Reg Valves to restore S/G levels."

D. INCORRECT. Plausible since it is desirable to have all safety systems in their normal alignment. Incorrect since this step is performed later in AP-FW.1 after stable plant conditions have been verified.

Technical Reference(s): AP-FW.1, Abnormal MFW Pump Flow or NPSH (p10-11; Rev 021)

AP-FW.1 Background Document (p7; Rev 004)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP09C 1.03, 2.01

Question Source: Bank # ____

Modified Bank # ____

New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .10

55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE055 EA1.07	
	Importance Rating	4.3	

Ability to operate and monitor the following as they apply to a Station Blackout: Restoration of power from offsite

RO Question #12

Given the following plant conditions:

- A Station Blackout has occurred
- The operating crew has implemented ECA-0.0, Loss of All AC Power, and are at Step 23, Monitor RCS Integrity
- The CO has just restored power to Buses 14 and 18 in accordance with ER-ELEC.1, Restoration of Offsite Power

Which ONE of the following describes how the operating crew will proceed?

A. Return to E-0, Reactor Trip or Safety Injection

B. Go to Step 34 and manually start individual loads

C. Go to Step 34 and monitor automatic loading of ECCS loads

D. Continue with ECA-0.0, Step 24, Initiate Depressurization of Intact S/Gs to 360 psig

Answer: **B**

Explanation:

The Operating crew will enter ECA-0.0 and progress through Steps 1 – 6, performing actions as directed. At Step 7, if an EDG is started and power is restored to Safeguards Buses, the Operator exits ECA-0.0 and returns to procedure in effect (E-0). However, if an EDG cannot be started and Safeguards Bus power restored, the Operator proceeds to Step 8. A CAUTION prior to Step 8 states "WHEN power is restored to Bus 14 AND/OR Bus 16, recovery actions continue starting at Step 34." The Operator continues performing actions at Step 8 and the associated RNO actions, as applicable until reaching Step 23 (given plant conditions). At this point the plant conditions state that the CO has restored Buses 14 and 18 using Offsite Power (last stem bullet). Based on the Step 8 CAUTION, the US will continue with ECA-0.0 at Step 34 to establish stable plant conditions (Steps 34 through 37) and exit ECA-0.0 at Step 38, instead of continuing with Step 24.

A. INCORRECT. Plausible since this would be correct if the ED/Gs had been started in Step 7 prior to the CAUTIONs of Step 8. Incorrect since ECA-0.0 requires that the operating crew now go to Step 34 and continue to a recovery procedure, not return to E-0.
B. CORRECT. According to ECA-0.0, Step 8 CAUTIONs "WHEN power is restored to Bus 14 AND/OR Bus 16, recovery actions should continue starting with Step 34. IF an SI signal exists OR IF an SI signal is actuated during this procedure, it should be reset to permit manual loading of equipment on an AC emergency bus." Step 8 directs the operating crew to place AC emergency bus loads in PULL STOP so that the loads cannot automatically start upon power restoration.
C. INCORRECT. Plausible since going to Step 34 is correct. Incorrect since Step 8 directs the operating crew to place AC emergency bus loads in PULL STOP so that the loads cannot automatically start upon power restoration.
D. INCORRECT. Plausible since this would be the progression if power was not restored to either Bus 14 or Bus 16. Incorrect since the Step 8 CAUTION directs the operating crew to continue with Step 34 once power is restored to either Bus 14 or Bus 16.

Technical Reference(s): ECA-0.0, Loss of All AC Power (p11; Rev 044)
ECA-0.0 Background Document (p65-66; Rev 021)

Proposed references to be provided to applicants during examination: None

Learning Objective: REC00C 1.02

Question Source: Bank # X
Modified Bank #
New

Question History: Last NRC Exam 2019 Wolf Creek ILT NRC

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

10 CFR Part 55 Content: 55.41 .10
55.43

Comments: Ginna's Facility Representative has determined that this is RO knowledge based on the answer testing a CAUTION in ECA-0.0, which is tied to a RO knowledge-based learning objective. RO's are required to know procedural NOTEs and CAUTIONs and the basis for them.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE056 G2.4.6	
	Importance Rating	3.7	

Knowledge of EOP mitigation strategies. (Loss of Offsite Power)

RO Question #13

Given the following plant conditions:

- Plant is operating at 100% power
- Electric plant is in a 50/50 Normal lineup
- Offsite Power Circuit 767 is lost
- ‘B’ EDG automatically starts and re-energizes its associated Safeguards Buses
- The operating crew enters AP-ELEC.1, Loss of 12A and/or 12B Busses

Which ONE of the following correctly completes the statements below?

1) The HCO will Monitor T_{AVG} with Rod Control in ________________.

AND

2) When restarting equipment for recovery, it is preferable that the crew starts ___________________ equipment.

A. 1) MANUAL
2) ‘A’ Train

B. 1) AUTOMATIC
2) ‘A’ Train

C. 1) MANUAL
2) ‘B’ Train

D. 1) AUTOMATIC
2) ‘B’ Train

Answer: A

Explanation:

A. CORRECT. In accordance with AP-ELEC.1, Step 2 the Operator is directed to place rods in

MANUAL and manually move control rods to control T_{AVG}. According to AP-ELEC.1, Step 14 NOTE “When restarting equipment for recovery, it is preferable to start equipment on busses being supplied from offsite power.”

B. INCORRECT. The first part is plausible since this is the normal mode of operation for Rod Control and the Applicant may misinterpret that since all Instrument Buses have been re-energized, AP-ELEC.1 will allow this operation. Incorrect since AP-ELEC.1, Step 2 directs the Operator to place Rod Control in MANUAL to control T_{AVG}. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since ‘B’ Train loads would have tripped on undervoltage upon the loss of Offsite Power Circuit 767 and the Applicant may misinterpret that these loads will be restarted. Incorrect since AP-ELEC.1, Step 14 NOTE “When restarting equipment for recovery, it is preferable to start equipment on busses being supplied from offsite power.”

D. INCORRECT. The first part is plausible since this is the normal mode of operation for Rod Control and the Applicant may misinterpret that since all Instrument Buses have been re-energized, AP-ELEC.1 will allow this operation. Incorrect since AP-ELEC.1, Step 2 directs the Operator to place Rod Control in MANUAL to control T_{AVG}. The second part is plausible since ‘B’ Train loads would have tripped on undervoltage upon the loss of Offsite Power Circuit 767 and the Applicant may misinterpret that these loads will be restarted. Incorrect since AP-ELEC.1, Step 14 NOTE “When restarting equipment for recovery, it is preferable to start equipment on busses being supplied from offsite power.”

Technical Reference(s): AP-ELEC.1, Loss of 12A and/or 12B Busses (p3 &10; Rev 033)

AP-ELEC.1 Background Document (p4 &10; Rev 003)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP07C 2.01

Question Source: Bank # ______

Modified Bank # ______

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE057 AA2.18	
	Importance Rating	3.1	

Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: The indicator, valve, breaker, or damper position which will occur on a loss of power

RO Question #14

Given the following plant conditions:

- Plant is operating at 100% power
- Annunciator E-14, LOSS B INSTR. BUS, alarms
- HCO reports INSTRUMENT BUS B AC VOLTMETER indicates 0 VAC

Which ONE of the following correctly answers the questions below when Instrument Bus ‘B’ de-energizes?

1) What is the expected condition of Channel 2 (White) Bistable Status Lights?

AND

2) What is the effect on plant systems?

A. 1) LIT
2) Steam Dumps will ARM

B. 1) LIT
2) Normal Letdown will isolate

C. 1) Extinguished
2) Steam Dumps will ARM

D. 1) Extinguished
2) Normal Letdown will isolate

Answer: **D**

Explanation:

A. INCORRECT. The first part is plausible since loss of either Instrument Bus ‘A’ or ‘C’ would result in the associated bistable status lights being LIT. Incorrect since Instrument Bus ‘B’ supplies power to the Channel 1 and 2 Bistable Status Lights; therefore, these Bistable Status Lights would be out (off). The second part is plausible since loss of Instrument Bus ‘C’ would result in PT-486 failing low causing Steam Dumps to ARM and Instrument Bus ‘B’ supplies power to PT-485, which is the other Turbine First Stage Pressure instrument. Incorrect since loss of PI-485 has NO effect on Steam Dumps.

B. INCORRECT. The first part is plausible since loss of either Instrument Bus ‘A’ or ‘C’ would result in the associated bistable status lights being LIT. Incorrect since Instrument Bus ‘B’ supplies power to the Channel 1 and 2 Bistable Status Lights; therefore, these Bistable Status Lights would be out (off). The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since loss of Instrument Bus ‘C’ would result in PT-486 failing low causing Steam Dumps to ARM and Instrument Bus ‘B’ supplies power to PT-485, which is the other Turbine First Stage Pressure instrument. Incorrect since loss of PI-485 has NO effect on Steam Dumps.

D. CORRECT. According to P-10, Section 5.12.1.10.c “Failure of B Instrument Bus will cause all red and white bistable status lights to go out since they are powered from bus B.” Additionally, Section 5.12.3.3.b states “Pressurizer level channel LT-427 failed low will cause letdown isolation and trip the Pressurizer heaters.”

Technical Reference(s): P-10, Instrument Failure Reference Manual (p63-69; Rev 022)

Proposed references to be provided to applicants during examination: None

Learning Objective: RIC12C 1.06

Question Source: Bank # ____
Modified Bank # ____
New X

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 .7
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	APE058 G2.4.45	
	Importance Rating	4.1	

Ability to prioritize and interpret the significance of each annunciator or alarm. (Loss of DC Power)

RO Question #15

Given the following plant conditions:

- The plant is operating at 100% power
- Multiple faults have occurred affecting the DC Electrical System
- The following Annunciators are alarming:
    - J-21, 1A OR 1B BATTERY UNDERVOLTAGE
    - J-23, BATTERY BANK GROUND
    - J-24, EMERGENCY DIESEL GEN 1A PANEL
    - J-32, EMERGENCY DIESEL GEN 1B PANEL
- The CO reports the following DC voltage indications:
    - DC BUS A VOLTMETER: 133 VDC
    - DC BUS B VOLTMETER: 0 VDC

Which ONE of the following correctly answers the statements below?

(1) The Operating crew will FIRST enter ____________ to mitigate the event:
AND
(2) If a subsequent Loss of Offsite Power were to occur, ___________ will automatically start.

A. (1) ER-ELEC.2, Recovery from Loss of A or B DC Train
(2) ONLY ‘A’ EDG

B. (1) ER-ELEC.2, Recovery from Loss of A or B DC Train
(2) BOTH EDGs

C. (1) E-0, Reactor Trip or Safety Injection
(2) ONLY ‘A’ EDG

D. (1) E-0, Reactor Trip or Safety Injection
   (2) BOTH EDGs

Answer: **D**

Explanation:

A. INCORRECT. The first part is plausible since the entry conditions for ER-ELEC.2 are satisfied and the Applicant may misinterpret the plant conditions and determine that the reactor will not trip on a loss of DV Bus. Incorrect since the RTB UV Coils will de-energize when the associated DC Train is de-energized. The second part is plausible since each EDG is supplied control power from both DC Trains and the Applicant may misinterpret the operation of the EDG control power circuitry. Incorrect since the EDG control power circuitry will automatically swap to the other DC Train upon loss of its primary DC control power.

B. INCORRECT. The first part is plausible since the entry conditions for ER-ELEC.2 are satisfied and the Applicant may misinterpret the plant conditions and determine that the reactor will not trip on a loss of DV Bus. Incorrect since the RTB UV Coils will de-energize when the associated DC Train is de-energized. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since each EDG is supplied control power from both DC Trains and the Applicant may misinterpret the operation of the EDG control power circuitry. Incorrect since the EDG control power circuitry will automatically swap to the other DC Train upon loss of its primary DC control power.

D. CORRECT. The Applicant has to recognize that Annunciator J-21, in conjunction with the given DC Bus values would result in the ‘B’ Train Reactor Trip Breakers’ UV coils de-energizing causing a reactor trip to occur. The Operators would respond by entering E-0 to mitigate the event. Given that Annunciator J-21 has been received, the ‘B’ EDG DC control power will automatically swap to the ‘A’ DC Train. Both EDGs would start upon a subsequent loss of Offsite Power.

Technical Reference(s): ER-ELEC.2, Recovery from Loss of A or B DC Train (p4 & 16; Rev 016)

UFSAR Section 8.3.2, Direct Current Power Systems (Rev 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0901C 1.06a, 1.11a

Question Source: Bank # _____

Modified Bank # _____

New X

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge ____________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .10

55.43 ____________

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE W/E04 EK1.3	
	Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as they apply to the (LOCA Outside Containment): Annunciators and conditions indicating signals, and remedial actions associated with the (LOCA Outside Containment).

RO Question #16

Given the following plant conditions:

- The following Annunciators are LIT:
  - D-19, PRESSURIZER LO PRESS SI 1750 PSIG
  - D-20, PRESSURIZER LO PRESS TRIP 1873 PSIG
- The following Radiation Monitors are in ALARM:
  - R-10B, Plant Vent Iodine
  - R-14, Plant Vent Gas
- ECA-1.2, LOCA Outside Containment, has been entered
- All isolation actions in ECA-1.2 have been completed
- The HCO reports the following MCB indications now exist:
  - RCS pressure is 1100 psig and STABLE
  - ECCS flow remains STABLE at 400 gpm

Which ONE of the following describes the status of the LOCA;
AND
the required procedural transition?

A. The LOCA is isolated;
transition to ES-1.1, SI Termination

B. The LOCA is isolated;
transition to E-1, Loss of Reactor or Secondary Coolant

C. The LOCA is NOT isolated;
transition to E-1, Loss of Reactor or Secondary Coolant

D. The LOCA is NOT isolated;
transition to ECA-1.1, Loss of Emergency Coolant Recirculation

Answer: **D**

Explanation:

A. INCORRECT. The first part is plausible since RCS pressure and ECCS flow are both stable, the Applicant may misinterpret these indications as (additional) leakage has been isolated. Incorrect since ECA-1.2, Step 7 requires INCREASING RCS pressure to identify leak isolation. The second part is plausible since ES-1.1 would be the final transition if the Applicant misinterprets that the RCS leak is isolated. Incorrect since the RCS leak is NOT isolated and this procedure transition would not be made.

B. INCORRECT. The first part is plausible since RCS pressure and ECCS flow are both stable, the Applicant may misinterpret these indications as (additional) leakage has been isolated. Incorrect since ECA-1.2, Step 7 requires INCREASING RCS pressure to identify leak isolation. The second part is plausible since if the Applicant misinterprets that the RCS leak is isolated, ECA-1.2, Step 8 directs a transition to E-1 IF the leakage is isolated. Incorrect since the RCS leak is NOT isolated and this procedure transition would not be made.

C. INCORRECT. The first part is correct. The second part is plausible ECA-1.2, Step 8 directs a transition to E-1 IF the leakage is isolated and the Applicant may misinterpret this transition requirement. Since ECA-1.2 can be entered from E-0 or E-1, a return to the procedure dealing with a LOCA might be interpreted to be the required action as no further actions are available in ECA-1.2. Incorrect since ECA-1.2, Step 7 requires INCREASING RCS pressure to identify leak isolation and since the RCS leak is NOT isolated and this procedure transition would not be made.

D. CORRECT. With RCS pressure NOT rising, the leak has not been isolated and ECA-1.2, Step 7 RNO directs the Operator to “Go to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1”.

Technical Reference(s): ECA-1.2, LOCA Outside Containment (Rev 00800)

Proposed references to be provided to applicants during examination: None

Learning Objective: REC12C 2.01

Question Source: Bank # X
Modified Bank # ______
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments: RO knowledge since the first part is testing the overall mitigative strategy (Major Action Categories) of ECA-1.2 and the second part is testing Entry Conditions for the applicable Emergency Procedures. Both of these are RO knowledge.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE W/E11 EK2.1	
	Importance Rating	3.6	

Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

RO Question #17

Given the following plant conditions:

- A LOCA has occurred
- The operating crew is preparing to align for High Head Recirculation of Containment Sump 'B' in accordance with ES-1.3, Transfer to Cold Leg Recirculation
- RCS pressure is 350 psig and stable

Which ONE of the following would prevent High Head Recirculation flow? (**Consider each condition separately**)

A. MOV-896B, RWST Outlet to CNMT Spray & SI Pumps, is OPEN

B. MOV-851A, RHR Pump Suction from CNMT Sump B, is CLOSED

C. BOTH MOV-897 and MOV-898, SI Recirc to RWST, are OPEN

D. BOTH MOV-857A and MOV-857C, RHR Pump Discharge to SI Pump Suction, are CLOSED

Answer: **C**

Explanation:

A. INCORRECT. Plausible since ES-1.3, Step 11.a directs the Operator "Close RWST outlet valves to SI and CNMT spray pumps; MOV-896A and MOV-896B". Incorrect since these valves are in series; therefore, only one is required to be closed to establish Sump Recirculation alignment.

B. INCORRECT. Plausible since ES-1.3, Step 7.c directs the Operator "Verify the following valves – OPEN; MOV-851A and MOV-851B". Incorrect since ES-1.3, Step 7.c RNO states

“Ensure at least one valve in each set open.” This would only prevent ‘A’ RHR Pump from supplying Sump Recirculation flow.

C. CORRECT. Given that RCS pressure is greater than 250 psig [300 psig adverse CNMT], ES-1.3, Step 12.a RNO directs the Operator to check RCS subcooling and PRZR level. With a LOCA in progress, either or both of these conditions will not be satisfied, so the Operator is directed to Step 12d. ES-1.3, Step 12.d directs the Operator to “Align operating RHR pump flow path(s) to SI and CNMT spray pump suction” by opening MOV-857A and MOV-857C OR MOV-857B. According to UFSAR, Table 6.3-8 MOV-857A, B, C are interlocked. “If 850A or 850B is open, then 857A, B, and C cannot be opened unless either 896A or B is closed and either 897 or 898 is closed. This prevents putting water from sump B into the RWST.” Since neither of these flow paths is available, ES-1.3 Foldout Page directs the Operator to transition to ECA-1.1, Loss of Emergency Coolant Recirculation.

D. INCORRECT. Plausible since ES-1.3, Step 12.d directs the Operator “Align operating RHR pump flow path(s) to SI and CNMT spray pump suction.” Incorrect since the flow path from ‘B’ RHR Pump is available to supply Sump Recirculation flow.

Technical Reference(s): ES-1.3, Transfer to Cold Leg Recirculation (p8, 11-13, & 29; Rev 047)

UFSAR Table 6.3-8, Safety Injection Valve Operation and Interlocks (Rev 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2701C 1.03d, 1.07d

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2011 Ginna ILT NRC SRO Retake

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .7

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	EPE W/E05 EA1.1	
	Importance Rating	4.1	

Ability to operate and/or monitor the following as they apply to the (Loss of Secondary Heat Sink): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

RO Question #18

Given the following plant conditions:

- Plant was operating at 100% power
- A steam line break in the Intermediate Building occurred
- The reactor tripped and SI actuated
- Both MDAFW and the TDAFW Pumps have tripped
- The operating crew has entered FR-H.1, Response to Loss of Secondary Heat Sink

Which ONE of the following completes the statement below?

The SI signal will cause ______________________.

A. the SAFW Pumps to not start.

B. the SAFW Pump Discharge Valves to CLOSE.

C. the MDAFW Pump start signal interlock preventing the start of the SAFW Pumps.

D. a Service Water isolation which eliminates the suction source for the SAFW Pumps.

Answer: A

Explanation:

A. CORRECT. According to AR-AA-5, SI will give a trip signal to the SAFW Pumps. The Operator must reset SI in FR-H.1 in order to allow operation of the SAFW system to restore S/G water levels.

B. INCORRECT. Plausible since an SI signal closes many non-Safeguards valves to prevent inadvertent operation and potential overloading of the EDGs. Incorrect since SI does not send a CLOSE signal to the SAFW Pump Discharge Valves.

C. INCORRECT. Plausible since there is an interlock that prevents the SAFW Pumps from being started when the associated MDAFW Pump is running. Incorrect since the interlock is provided by the MDAFW Pump breakers and not the START signal from SI. Since the MDAFW Pumps are tripped, the interlock is not functioning.

D. INCORRECT. Plausible since an SI signal with the EDGs supplying their respective safeguards buses would result in SW isolation and SW is the preferred source of water for the SAFW System. Incorrect since the given plant conditions would not result in a SW isolation.

Technical Reference(s): AR-AA-5, STDBY AUX FW PMP C OR D TRIP (Rev 00800)

10905-0445, Standby Aux Feedwtr Pump C Elementary Wiring Diagram (Rev 9)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4201C 1.04, 1.07

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 .7

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE005 AK3.02	
	Importance Rating	3.6	

Knowledge of the reasons for the following responses as they apply to the Inoperable/Stuck Control Rod: Rod insertion limits

RO Question #19

Given the following initial plant conditions:

- Reactor power is 65% following a load reduction
- A Bank ‘D’ Control Rod is misaligned above its associated Rod Bank
- ER-RCC.2, Restoring a Misaligned RCC, is being performed to recover the Control Rod

Subsequently:

- The misaligned Control Rod is aligned with its associated Rod Bank
- Reactor power is 65%

1) The misaligned Control Rod was inserted by placing the ROD CONTROL BANK SELECTOR switch in the ______ position.

   AND

2) What is the reason that calculated Rod Insertion Limits have not changed following completion of the realignment?

A. 1) M (MANUAL)
   2) Reactor power is stable

B. 1) CB D
   2) Reactor power is stable

C. 1) M (MANUAL)
   2) P/A Converter is reset

D. 1) CB D
   2) P/A Converter is reset

Answer: **B**

Explanation:

A. INCORRECT. The first part is plausible since this would be correct for manually moving Control Rods. Incorrect since ER-RCC.2 directs the Operator to place the ROD CONTROL BANK SELECTOR switch to the affected bank. The second part is correct.

B. CORRECT. ER-RCC.2, Step 6.2.1 directs the Operator to "**PLACE** control rod bank selector switch to the affected bank." This would be CB D for Control Bank 'D' which is the normal control bank when the reactor is at power. UFSAR, Section 7.7.1.2.9 states "The control rod insertion limits are calculated as a linear function of power and reactor coolant temperature." COLR Figure COLR-3 shows that for a given reactor power, Rod Insertion Limits would remain the same.

C. INCORRECT. The first part is plausible since this would be correct for manually moving Control Rods. Incorrect since ER-RCC.2 directs the Operator to place the ROD CONTROL BANK SELECTOR switch to the affected bank. The second part is plausible since the P/A Converter provides an input into the Rod Insertion Limit alarms and is reset following Control Rod recovery. Incorrect since the Rod Insertion Limit is calculated as a linear function of power and reactor coolant temperature.

D. INCORRECT. The first part is correct. The second part is plausible since the P/A Converter provides an input into the Rod Insertion Limit alarms and is reset following Control Rod recovery. Incorrect since the Rod Insertion Limit is calculated as a linear function of power and reactor coolant temperature.

Technical Reference(s):
- ER-RCC.2, Restoring a Misaligned RCC (Rev 018)
- UFSAR, Section 7.7.1.2.9 Rod Insertion Limit Circuit (Rev 28)
- COLR Figure COLR-3, Control Bank Insertion Limits (Rev 0)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2901C 1.02c; RER12C 2.01

Question Source:
- Bank # ____
- Modified Bank # X
- New ____

Question History: Last NRC Exam 2005 Kewaunee ILT NRC

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE028 AK2.03	
	Importance Rating	2.6	

Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: Controllers and positioners

RO Question #20

Given the following plant conditions:

- 100% Reactor power
- The Charging Pump in AUTO controller output fails to 100%
- PRZR level DEFEAT switch L/428A is in NORMAL

Which ONE of the following describes expected plant response with no Operator actions?

1) The Charging Pump in AUTO will ________ Charging line flow;
   AND
2) 5 minutes later, Pressurizer Backup Heaters will be ____________.

A. 1) raise
   2) de-energized

B. 1) lower
   2) de-energized

C. 1) raise
   2) energized

D. 1) lower
   2) energized

Answer: **C**

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since the Backup Heaters are normally de-energized when the plant is operating at steady state power.

Incorrect since the rising PRZR level will cause the Backup Heaters to energize due to insurge.

B. INCORRECT. The first part is plausible if the Applicant believes that the controller failing to 100% will result in the Charging Pump slowing down as in a reverse acting controller. Incorrect since the Charging Pump controller is a direct acting where 100% equals maximum speed of the Charging Pump. The second part is plausible since the Backup Heaters are normally de-energized when the plant is operating at steady state power. Incorrect since the rising PRZR level will cause the Backup Heaters to energize due to insurge.

C. CORRECT. As Charging Pump controller output increases, Charging Pump speed increases resulting in higher flow. This causes Charging line flow to rise. According to AR-F-4, Step 4.2, the Operator is directed to verify that Backup Heaters are on at a PRZR level of + 5% above program level.

D. INCORRECT. The first part is plausible if the Applicant believes that the controller failing to 100% will result in the Charging Pump slowing down as in a reverse acting controller. Incorrect since the Charging Pump controller is a direct acting where 100% equals maximum speed of the Charging Pump. The second part is correct.

Technical Reference(s): AR-F-4, PRESSURIZER LEVEL DEVIATION -5 NORMAL +5 (Rev 006)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1601C 1.06

Question Source: Bank # X

Modified Bank # ____

New ____

Question History: Last NRC Exam 2019 Ginna ILT

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .7

55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE032 G2.2.39	
	Importance Rating	3.9	

Knowledge of less than or equal to one hour Technical Specification action statements for systems. (Loss of Source Range Nuclear Instrumentation)

RO Question #21

Given the following plant conditions:

- Reactor startup is in progress
- Control Bank 'D' Control Rods are withdrawn to 216 steps
- RCS dilution is in progress
- N-31, SR Channel, indicates $1x10^3$ cps and slowly rising
- HCO reports that N-32, SR Channel, has failed LOW
- N-35 and N-36, IR Channels, indicate < 1E-11 amps

Which ONE of the following correctly states the required action in accordance with Technical Specification LCO 3.3.1, Reactor Trip System (RTS) Instrumentation?

A. Suspend the RCS dilution immediately

B. Open Reactor Trip Breakers immediately

C. Restore N-32 to OPERABLE within one hour

D. Raise Thermal Power > 5E-11 amps within one hour

Answer: A

Explanation:

A. CORRECT. With the plant in MODE 2, LCO 3.3.1, Table 3.3.1-1 requires 2 channels for Function 4, Source Range Neutron Flux. LCO Conditions F and G are applicable. Condition F has the following Required Actions and Completion Times: F.1 Open RTBs and RTBBs upon discovery of two inoperable channels immediately; F.2 Suspend operations involving positive reactivity additions immediately; and F.3 Restore channel to OPERABLE status within 48 hours.

B. INCORRECT. Plausible since this would be the action if both Source Range channels were INOPERABLE. Incorrect since N-31 is still OPERABLE; therefore, Required Action F.1 does not apply.

C. INCORRECT. Plausible since this action would be required if the plant was in MODE 3 and both Source Range channels were discovered to be INOPERABLE. Incorrect since the given plant conditions have the plant in MODE 2 and N-31 is still OPERABLE.

D. INCORRECT. Plausible since LCO 3.3.1, Condition F is only applicable if both Intermediate Range channels are < 5E-11 amps and the Applicant may misinterpret this as a required action to exit the Condition, similar to the Required Action for an Intermediate Range channel out of service. Incorrect since Required Action F.2 applies with the plant in MODE 2 and one Source Range channel INOPERABLE.

Technical Reference(s): Technical Specification 3.3.1, Reactor Trip System (RTS) Instrumentation (Amendment 132)

Proposed references to be provided to applicants during examination: None

Learning Objective: RIC08C 1.12a

Question Source:	Bank #	X
	Modified Bank #	
	New	

Question History: Last NRC Exam __________

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE060 AA1.01	
	Importance Rating	2.8	

Ability to operate and/or monitor the following as they apply to the Accidental Gaseous Radwaste: Area radiation monitors

RO Question #22

Given the following plant conditions:

- Gas Decay Tank 'A' rupture disc bursts as designed
- RV-1621, GAS DECAY TANK A RELIEF VALVE, lifts to lower tank pressure

Assuming NO Operator actions, which Radiation Monitor will provide the first indication of RV-1621 lifting?

A. R-4, Charging Pump Room

B. R-9, Letdown Line

C. R-14, Plant Vent Noble Gas

D. R-42, Auxiliary Building Basement Air Monitor

Answer: **C**

Explanation:

A. INCORRECT. Plausible since R-4 is located in the Charging Pump Room and the Applicant may misinterpret that the relief valve relieves to the Charging Pump Room floor drains causing the radiation monitor to alarm. Also, R-4 is checked in E-0, Step 26 to determine if an RCS leak in the Auxiliary Building exists. Incorrect since RV-1621 relieves to the plant vent and will NOT cause R-4 to alarm.

B. INCORRECT. Plausible since R-9 is located outside the Charging Pump Room and the Applicant may misinterpret that the relief valve relieves to the Auxiliary Building floor drains near the NaOH Room causing the radiation monitor to alarm. Also, R-9 is checked in E-0, Step 26 to determine if an RCS leak in the Auxiliary Building exists. Incorrect since RV-1621 relieves to the plant vent and will NOT cause R-9 to alarm.

C. CORRECT. According to 33013-1273, Sheet 2, RV-1621 relieves to the Plant Vent. In accordance with P-9 "R-14, Plant Vent Noble Gas Monitor, measures low concentrations of

Xe-133 from reactor coolant leaks in the Auxiliary Building, gas decay tank releases, or from taking primary system samples in the nuclear sample room."

D. INCORRECT. Plausible since R-42 is located in the Auxiliary Building Basement and the Applicant may misinterpret that the relief valve relieves to the Auxiliary Building basement floor drains causing the radiation monitor to alarm. Incorrect since RV-1621 relieves to the plant vent and will NOT cause R-42 to alarm.

Technical Reference(s): P-9, Radiation Monitoring System (p15; Rev 107)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3901C 1.03

Question Source: Bank # X

Modified Bank # ______

New ______

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 .11

55.43 ______

Comments: K/A is met since there are no Area Radiation Monitors at Ginna that would detect an accidental gaseous radwaste release. The radwaste system gas goes through the plant vent either through normal release, rupture disk failure, or relief valve lifting. Therefore, the only Radiation Monitors that would detect this release would be one of the plant vent process Radiation Monitors.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE068 AA2.06	
	Importance Rating	4.1	

Ability to determine and interpret the following as they apply to the Control Room Evacuation: RCS pressure

RO Question #23

Given the following plant conditions:

- The Control Room has been evacuated due to toxic fumes
- The US has entered AP-CR.1, Control Room Inaccessibility
- PRZR level is 20% and stable
- PRZR pressure is 2200 psig and lowering slowly

Which ONE of the following describes how Pressurizer pressure is controlled in accordance with AP-CR.1?

Pressurizer pressure is monitored by the HCO in the ____(1)____ Local Operating Station and will be controlled using local operation of ____(2)____.

A. (1) Charging Pump area
   (2) PRZR Backup Heaters, ONLY

B. (1) Charging Pump area
   (2) PRZR Proportional and Backup Heaters

C. (1) Auxiliary Feedwater (AFW) Pump area
   (2) PRZR Backup Heaters, ONLY

D. (1) Auxiliary Feedwater (AFW) Pump area
   (2) PRZR Proportional and Backup Heaters

Answer: C

Explanation:

A. INCORRECT. The first part is plausible since PRZR level is monitored at the Charging

Pump area in Step 10 and the Applicant may misinterpret that this is also where PRZR pressure is monitored. Incorrect since the Charging Pump area does not contain PRZR pressure indication. The second part is correct.

B. INCORRECT. The first part is plausible since PRZR level is monitored at the Charging Pump area in Step 10 and the Applicant may misinterpret that this is also where PRZR pressure is monitored. Incorrect since the Charging Pump area does not contain PRZR pressure indication. The second part is plausible since local operation of PRZR Backup Heaters is correct and local operation of PRZR Proportional Heaters is performed in ER-PRZR.1 and the Operator may misinterpret that this local operation is also performed in AP-CR.1. Incorrect since PRZR pressure is lowering and AP-CR.1, Step 11 RNO directs the Operator to locally control PRZR Backup Heaters to raise pressure.

C. CORRECT. AP-CR.1, Step 11 directs the Operator to "Locally Monitor PRZR Pressure – PRESSURE STABLE (AFW pump area, west wall)". AP-CR.1, Step 11 RNO directs the Operator "IF pressure lowering, THEN perform the following: a. Transfer PRZR heater backup group to local control; b. Verify PRZR level greater than 13%; c. Energize PRZR heater backup group."

D. INCORRECT. The first part is correct. The second part is plausible since local operation of PRZR Backup Heaters is correct and local operation of PRZR Proportional Heaters is performed in ER-PRZR.1 and the Operator may misinterpret that this local operation is also performed in AP-CR.1. Incorrect since PRZR pressure is lowering and AP-CR.1, Step 11 RNO directs the Operator to locally control PRZR Backup Heaters to raise pressure.

Technical Reference(s): AP-CR.1, Control Room Inaccessibility (p9; Rev 025)
AP-CR.1 Background Document (p8; Rev 004)

Proposed references to be provided to applicants during examination: None

Learning Objective: R5401C 1.02; RAP04C 2.01

Question Source:
Bank # ___
Modified Bank # X
New ___

Question History: Last NRC Exam ___

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	APE076 AK2.01	
	Importance Rating	2.6	

Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors

RO Question #24

Given the following plant conditions:

- The plant is operating at 100% power
- Annunciator E-24, RMS AREA MONITOR HIGH ACTIVITY, is alarming
- HCO reports that R-9, Letdown Line Monitor, is in HIGH alarm

Which ONE of the following correctly completes the statement below?

The US will implement ________(1)________ **AND** the R-9 alarm will be verified by ________(2)________.

A. (1) AP-RCS.1, Reactor Coolant Leak
(2) RCS sample for activity

B. (1) AP-RCS.3, High Reactor Coolant Activity
(2) RCS sample for activity

C. (1) AP-RCS.1, Reactor Coolant Leak
(2) comparison with other Auxiliary Building radiation monitors

D. (1) AP-RCS.3, High Reactor Coolant Activity
(2) comparison with other Auxiliary Building radiation monitors

Answer: B

Explanation:

A. INCORRECT. The first part is plausible since many Auxiliary Building radiation monitor alarms are entry conditions for AP-RCS.1. Incorrect since AR-RMS-9 directs the Operator to enter AP-RCS.3. The second part is correct.

B. CORRECT. In accordance with AR-RMS-9, the Operator is directed to "GO TO AP-RCS.3." AP-RCS.3, Step 1 directs the Operator to "Verify RCS Activity: Direct Chemistry to sample RCS for activity". AP-RCS.3 Background Document for Step 1 states "A radiation monitor may be erroneously indicating high, when activity has not changed. Therefore, samples should be obtained and analyzed to determine if RCS activity is actually higher than normal."

C. INCORRECT. The first part is plausible since many Auxiliary Building radiation monitor alarms are entry conditions for AP-RCS.1. Incorrect since AR-RMS-9 directs the Operator to enter AP-RCS.3. The second part is plausible since AP-RCS.3, Step 5 directs the Operator to "Evaluate AUX BLDG Radiation Levels" by checking AUX BLDG radiation monitors. Incorrect since the purpose of this step is to determine if a local radiation emergency should be declared.

D. INCORRECT. The first part is correct. The second part is plausible since AP-RCS.3, Step 5 directs the Operator to "Evaluate AUX BLDG Radiation Levels" by checking AUX BLDG radiation monitors. Incorrect since the purpose of this step is to determine if a local radiation emergency should be declared.

Technical Reference(s): AR-RMS-9, R9 LETDOWN LINE MONITOR (Rev 4)
AP-RCS.3, High Reactor Coolant Activity (p3; Rev 014)
AP-RCS.3 Background Document (p2; Rev 1)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3901C 1.11; RAP17C 1.02 & 2.01

Question Source: Bank # X
Modified Bank # ____
New ____

Question History: Last NRC Exam 2012 Ginna ILT Retake

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

10 CFR Part 55 Content: 55.41 .10
55.43 ____

Comments: K/A is matched since R-9 is an area radiation monitor that is used at Ginna as a process radiation monitor.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	EPE W/E15 EA1.1	
	Importance Rating	2.9	

Ability to operate and/or monitor the following as they apply to the (Containment Flooding): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

RO Question #25

Given the following plant conditions:

- The plant has experienced a large break LOCA
- The crew is responding with the appropriate procedures
- 3 hours after the transfer to CNMT sump recirculation, the STA reports an ORANGE path for the CONTAINMENT CSFST
- CNMT SUMP LEVEL 180 INCHES lights are both LIT
- Annunciator C-20, CONTAINMENT SUMP B HI LEVEL, is illuminated

Which ONE of the following indications would the operating crew use to identify the unexpected source of water to Containment Sump ‘B’?

NOTE: RWST, Refueling Water Storage Tank
FWST, Fire Water Storage Tank

A. RWST level is 13%.
Annunciator B-16, RWST LO-LO LEVEL, is illuminated.

B. FWST level is 5.75 feet.
The Fire Service Water Booster pump is running. Annunciator K-31, FIRE SYSTEM ALARM PANEL, is illuminated.

C. Service Water pressure is 80 psig with 3 SW Pumps running.
Annunciator I-2, CW PUMP SEAL WTR FLTR HI DIFF PRESS is illuminated.

D. Four (4) CNMT Recirculation Fans are running.
Annunciator C-10, CONTAINMENT RECIRC CLRS WATER OUTLET LO FLOW 1050

GPM, is illuminated.

Answer: **D**

Explanation:

A. INCORRECT. Plausible since this is an expected condition for sump recirculation. Incorrect since the RWST contents have been used, but only accounts for a fraction of the sump level indication.

B. INCORRECT. Plausible since FWST level is expected to be 5.75 feet prior to the Booster Pump stopping. Incorrect since Firewater is normally isolated to CNMT.

C. INCORRECT. Plausible since a third SW Pump may be started to accomplish sump recirculation. Incorrect since the filter D/P alarm may be expected but is not related to the 'B' Sump level.

D. CORRECT. According to FR-Z.2 Background Document "The maximum level of water in the containment following a major accident generally is based upon the entire water contents of the reactor coolant system, refueling water storage tank, and SI accumulators. An indicated water level in the containment greater than the maximum expected volume (design basis flood level) is an indication that water volumes other than those represented by the above noted volumes have been in traduced into the containment." Four CRFs are expected to be running, but SW low flow alarm is not expected, and would indicate a SW pipe break in CNMT which would account for the C-10 alarm.

Technical Reference(s): FR-Z.2, Response to Containment Flooding (p3; Rev 008)
FR-Z.2 Background Document (p2 & 8; Rev 003)

Proposed references to be provided to applicants during examination: None

Learning Objective: RFRZ2C 1.03

Question Source: Bank # X
Modified Bank #
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	.4
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	EPE W/E10 EA2.1	
	Importance Rating	3.2	

Ability to determine and interpret the following as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

RO Question #26

Given the following plant conditions:

- The reactor was initially at 100% power
- An arc fault occurs in Bus 11A resulting in damage to both Buses 11A and 11B
- The operating crew has decided to NOT energize Buses 13 and 15 from the Safeguards Buses Crosstie Breakers due to operational risk
- The operating crew is performing ES-0.1, Reactor Trip Response
- RCS temperature is 545°F and stable
- SM has determined that the plant must be cooled down to Cold Shutdown conditions within four (4) hours

Which ONE of the following correctly states:

1) the procedure the operating crew will directly transition to from ES-0.1,
   AND
2) the final procedure entered to establish the desired final plant conditions?

A. 1) O-2.1, Normal Shutdown to Hot Shutdown
   2) ES-0.2, Natural Circulation Cooldown

B. 1) ES-0.2, Natural Circulation Cooldown
   2) ES-0.2, Natural Circulation Cooldown

C. 1) O-2.1, Normal Shutdown to Hot Shutdown
   2) ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel

D. 1) ES-0.2, Natural Circulation Cooldown
   2) ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel

Answer: D

Explanation:

A. INCORRECT. The first part ES-0.1, Step 21 directs the Operator to transition to O-2.1 if at least one RCP is running. Incorrect since there are no RCPs available for the given plant conditions. The second part is plausible since the Applicant may determine that ES-0.2 will allow the plant to be cooled down at a rate fast enough to reach Cold Shutdown within four hours. Incorrect since ES-0.2 limits RCS cooldown rate to a maximum 25°F/HR; therefore, the crew must transition to ES-0.3 to reach Cold Shutdown conditions within four hours. ES-0.2 is a plausible transition from O-2.1 since the third entry condition of ES-0.2 is “Other normal operating procedures when a natural circulation cooldown is required.”

B. INCORRECT. The first part is correct. The second part is plausible since the Applicant may determine that ES-0.2 will allow the plant to be cooled down at a rate fast enough to reach Cold Shutdown within four hours. Incorrect since ES-0.2 limits RCS cooldown rate to a maximum 25°F/HR; therefore, the crew must transition to ES-0.3 to reach Cold Shutdown conditions within four hours.

C. INCORRECT. The first part ES-0.1, Step 21 directs the Operator to transition to O-2.1 if at least one RCP is running. Incorrect since there are no RCPs available for the given plant conditions. The second part is correct. ES-0.3 is a plausible transition from O-2.1 since the third entry condition of ES-0.2 is “Other normal operating procedures when a natural circulation cooldown is required.” The Operator can then transition to ES-0.3 from ES-0.2.

D. CORRECT. In accordance with ES-0.1, Step 21 RNO, the Operator is directed to “Go to ES-0.2, NATURAL CIRCULATION COOLDOWN, Step 1” since there are no RCPs running. ES-0.2, Step 17 NOTE states “If at any time it is determined that a natural circulation cooldown and depressurization must be performed at a rate that may form a steam void in the vessel, then procedure ES-0.3, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL, should be used.” ES-0.3 Entry Condition A is “ES-0.2 after completing the first 16 steps, if rapid cooldown or depressurization is required.” Additionally, ES-0.3, Step 7 directs the Operator to “Maintain cooldown rate in RCS cold legs – LESS THAN 100°F/HR.”

Technical Reference(s):
ES-0.1, Reactor Trip Response (p23; Rev 032)
ES-0.2, Natural Circulation Cooldown (p2&16; Rev 018)
ES-0.3, Natural Circulation Cooldown with Steam Void in Vessel (p2&8; Rev 015)

Proposed references to be provided to applicants during examination: None

Learning Objective: RES02C 2.01; RES03C 2.01

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History:	Last NRC Exam	

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments: Ginna's Facility Representative has determined that the question requires knowledge of the overall mitigative strategy (Major Action Categories) and Entry Conditions for Emergency procedure which is RO knowledge.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	EPE W/E08 EK1.1	
	Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as they apply to the (Pressurized Thermal Shock): Components, capacity, and function of emergency systems.

RO Question #27

Given the following plant conditions:

- Plant is operating at 100% power
- 'B' S/G experiences a fault concurrent with loss of Offsite Power
- A RED path on Integrity CSFST is validated
- The US enters FR-P.1, Response to Imminent Pressurized Thermal Shock
- 'A' S/G water level is 20% and slowly rising
- 'B' S/G water level is 200 inches and lowering
- CNMT pressure is 0.3 psig and stable

Which ONE of the correctly completes the statements below?

1) AFW flow to 'A' S/G will be throttled to ________________.
AND
2) The RCS will be depressurized utilizing ________________.

A. 1) 0 gpm
2) PRZR PORVs

B. 1) 0 gpm
2) Auxiliary Spray

C. 1) 200 gpm
2) PRZR PORVs

D. 1) 200 gpm
2) Auxiliary Spray

Answer: A

Explanation:

A. CORRECT. According to FR-P.1, Step 2 RNO, if RCS cold leg temperatures are NOT stable or rising, the Operator is directed to “Control total feed flow to non-faulted S/G(s) greater than 200 gpm until narrow range level greater than 7% [25% adverse CNMT] in at least one non-faulted S/G. WHEN S/G level greater than 7% [25% adverse CNMT] in one non-faulted S/G, THEN limit feed flow to stop RCS cooldown.” According to FR-P.1, Step 17.a RNO since normal PRZR spray is NOT available, the Operator is directed to “Use one PRZR PORV” to depressurize the RCS. IF no PRZR PORV available, THEN use auxiliary spray valve (AOV-296).” Since the given plant conditions allow the use of a PRZR PORV, this would be the preferred method to depressurize the RCS.

B. INCORRECT. The first part is correct. The second part is plausible since in other procedures, Auxiliary Spray is preferred instead of using a PRZR PORV for lowering RCS pressure. Incorrect since in FR-P.1, this action is only performed if a PRZR PORV is not available, but the given plant conditions have the PRZR PORVs available.

C. INCORRECT. The first part is plausible since AFW flow to the non-faulted S/G would be maintained at least 200 gpm until S/G level was greater than 25% if adverse CNMT conditions existed. Incorrect since with the given plant conditions, adverse CNMT conditions do NOT exist. The second part is correct.

D. INCORRECT. The first part is plausible since AFW flow to the non-faulted S/G would be maintained at least 200 gpm until S/G level was greater than 25% if adverse CNMT conditions existed. Incorrect since with the given plant conditions, adverse CNMT conditions do NOT exist. The second part is plausible since in other procedures, Auxiliary Spray is preferred instead of using a PRZR PORV for lowering RCS pressure. Incorrect since in FR-P.1, this action is only performed if a PRZR PORV is not available, but the given plant conditions have the PRZR PORVs available.

Technical Reference(s): FR-P.1, Response to Imminent Pressurized Thermal Shock Condition (p4 & 14; Rev 034)

FR-P.1 Background Document (p40; Rev 007)

Proposed references to be provided to applicants during examination: None

Learning Objective: RFRP1C 1.03, 2.01

Question Source:

Bank #	
Modified Bank #	X
New	

Question History: Last NRC Exam 2018 Turkey Point ILT

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .8

55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 K5.03	
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP shutdown on T-ave., including the reason for the unreliability of T-ave. in the shutdown loop

RO Question #28

Given the following plant conditions:

- Plant startup is in progress
- Reactor power is 20% and stable
- 'B' RCP trips

Which ONE of the following choices completes the statements regarding the immediate impact on reactor operations **AND** idle loop T_{AVG} one minute following the trip of 'B' RCP?
(Assume NO Operator actions taken)

1) The reactor ______________.
AND
2) 'B' Loop T_{AVG} is unreliable since it is indicative of 'A' Loop _____________.

A. 1) remains at power
2) T_{HOT}

B. 1) remains at power
2) T_{COLD}

C. 1) automatically trips
2) T_{HOT}

D. 1) automatically trips
2) T_{COLD}

Answer: B

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since there will be reduced heat transfer to the idle loop S/G and the Applicant may misinterpret this impact as causing idle loop T_{AVG} to rise to idle loop T_{HOT}. Incorrect since idle loop T_{AVG} will lower to T_{COLD} of the operating loop due to reverse flow.

B. CORRECT. In accordance with P-1, the reactor will remain at power since reactor power is less than the P-8 setpoint of 25%. T_{AVG} in the loop with the tripped RCP will lower due to reverse flow from the operating loop. This will cause the idle loop T_{AVG} to lower to the operating loop T_{COLD}.

C. INCORRECT. The first part is plausible if the Applicant misinterprets the automatic reactor protection setpoint with the procedural requirement, in accordance with AP-RCS.2, to trip the reactor if both RCPs are not in operation. Incorrect since the automatic reactor trip setpoint for a single RCP operating is 25%. The second part is plausible since there will be reduced heat transfer to the idle loop S/G and the Applicant may misinterpret this impact as causing idle loop T_{AVG} to rise to idle loop T_{HOT}. Incorrect since idle loop T_{AVG} will lower to T_{COLD} of the operating loop due to reverse flow.

D. INCORRECT. The first part is plausible if the Applicant misinterprets the automatic reactor protection setpoint with the procedural requirement, in accordance with AP-RCS.2, to trip the reactor if both RCPs are not in operation. Incorrect since the automatic reactor trip setpoint for a single RCP operating is 25%. The second part is correct.

Technical Reference(s):
- P-1, Reactor Control and Protection System (p40 & 42; Rev 078)
- AP-RCS.2, Loss of Reactor Coolant Flow (p3; Rev 013)
- UFSAR, Section 15.4.3.1 (p129 of 276; Rev 28)
- ROC01S, Partial Loss of Reactor Coolant Flow (slides 26 & 29; Rev 13)

Proposed references to be provided to applicants during examination: None

Learning Objective: ROC01S 1.01, 1.02

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2008 Harris ILT

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

10 CFR Part 55 Content:	55.41	.3
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 A4.06	
	Importance Rating	3.6	

Ability to manually operate and/or monitor in the control room:
Letdown isolation and flow control valves (CVCS)

RO Question #29

Given the following plant conditions:

- Plant is operating at 100% power
- CCW Surge Tank level is 55% and rising
- R-17, CCW SYS Process Monitor, counts are rising, but below the warning setpoint
- VCT level is 20% and lowering
- Letdown line flow is 60gpm and stable
- PCV-135, LOW PRESS LTDN PRESS CONTROL VLV, demand is 25% OPEN
- Crew is performing actions in AP-CCW.1, Leakage into the Component Cooling Loop

Based on the given plant conditions, what is the NEXT required Operator action in accordance with AP-CCW.1?

A. Close Normal Letdown isolation valves

B. Close RCV-017, CCW SURGE TK VENT

C. Adjust charging flow to maintain PRZR level at program

D. Perform a manual makeup to the VCT to maintain level greater than 20%

Answer: **A**

Explanation:

A. CORRECT. Parameters given in the stem indicate leakage into the Non-Regenerative Heat Exchanger (NRHX). In accordance with AP-CCW.1, Step 4b, PCV-135 demand should be 40% open with letdown flow at 60gpm. PCV-135 demand at 30% open indicates it is closing due to the leak, to maintain upstream letdown pressure at the normal 250 psig setpoint. In accordance with AP-CCW.1, Step 4b RNO, with PCV-135 demand less than 40%, Normal Letdown is isolated by closing letdown isolation valves AOV-427, letdown orifice valves, and AOV-371.

B. INCORRECT. Plausible since the Applicant may misinterpret that RCV-017 should be manually closed due to rising CCW Surge Tank level and rising counts on R-17. A CAUTION in AP-CCW.1 states “DURING THE PERFORMANCE OF THIS PROCEDURE, RCV-017 SHOULD BE MONITORED TO ENSURE CLOSURE ON CCW SYSTEM RADIATION MONITOR ALARM.” The purpose of the CAUTION is to ensure CCW vent closes if R-17 alarms. Incorrect since the stem states R-17 is below the alarm setpoint. RCV-017 would be manually closed if it failed to automatically close on R-17 alarm.

C. INCORRECT. Plausible since PRZR level lowers for most RCS leaks and the Applicant may misinterpret the given plant conditions and determine that PRZR level would be lowering due to the leak. Incorrect however, since prior to isolation, PRZR level would be stable for the leak location (NRHX) and only VCT level would be lowering. Additionally, adjusting charging flow to restore PRZR level is an action in AP-CCW.1, but is performed after letdown is isolated or subsequently restored on an alternate path such as Excess Letdown.

D. INCORRECT. Plausible since 20% is the setpoint for an automatic makeup to the VCT and VCT level would be lowering with the leak location in the NRHX. Manual makeup to the VCT, if automatic makeup fails, is an action in many APs and EOPs and the Applicant may misinterpret this is an action directed by AP-CCW.1. Incorrect since automatic makeup control would be armed and AP-CCW.1 does not contain actions to perform manual makeup to the VCT.

Technical Reference(s):	AP-CCW.1, Leakage into the Component Cooling Loop (p3 & 5; Rev 01901)
	AP-CCW.1 Background Document (p2 & 4; Rev 2)
	R1601C, Chemical and Volume Control System (slide 66; Rev 30)
	AR-A-2, VCT LEVEL 14% 86 (p6; Rev 01100)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1601C 1.10b

Question Source:	Bank #	
	Modified Bank #	X
	New	

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments: Question is RO knowledge since it can be answered solely by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure.

Question meets the K/A since Ginna does not have a specific Letdown flow control valve. Letdown flow is controlled by the differential pressure across the Letdown Orifice Valves which is controlled by PCV-135, Low Pressure Letdown Pressure Control Valve. The question requires the Applicant to monitor and determine that the demand on PCV-135 controller is not correct for the given plant conditions, which is an indication that a leak exists in the Letdown line.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 K5.02	
	Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as they apply to the CVCS: Explosion hazard associated with hydrogen containing systems

RO Question #30

Given the following plant conditions:

- Plant shutdown for a Refueling Outage is in progress
- The operating crew is performing O-2.2, Plant Shutdown from Hot Shutdown to Cold Conditions

Which ONE of the following correctly completes the statements below?

(1) RCS hydrogen concentration is reduced by burping the VCT prior to the addition of ___________________.

AND

(2) RCS hydrogen concentration is reduced during plant cooldown because _________________________.

A. (1) Hydrazine
(2) oxygen pitting corrosion is not a concern at low RCS temperatures

B. (1) Hydrogen Peroxide
(2) oxygen pitting corrosion is not a concern at low RCS temperatures

C. (1) Hydrazine
(2) hydrogen presents an explosion hazard when the RCS is opened for maintenance

D. (1) Hydrogen Peroxide
(2) hydrogen presents an explosion hazard when the RCS is opened for maintenance

Answer: **D**

Explanation:

A. INCORRECT. The first part is plausible since Hydrazine is added to the primary when starting up following an Outage in order to scavenge oxygen from the RCS and the Applicant may misinterpret that Hydrazine is added during plant shutdown to Cold Shutdown conditions. Incorrect since hydrogen peroxide addition is performed during shutdown to facilitate a crud burst in the RCS. The second part is plausible since the rate of oxygen pitting corrosion does decrease at lower temperatures. Incorrect since it is still a concern at lower temperatures.

B. INCORRECT. The first part is correct. The second part is plausible since the rate of oxygen pitting corrosion does decrease at lower temperatures. Incorrect since it is still a concern at lower temperatures.

C. INCORRECT. The first part is plausible since Hydrazine is added to the primary when starting up following an Outage in order to scavenge oxygen from the RCS and the Applicant may misinterpret that Hydrazine is added during plant shutdown to Cold Shutdown conditions. Incorrect since hydrogen peroxide addition is performed during shutdown to facilitate a crud burst in the RCS. The second part is correct.

D. CORRECT. In accordance with O-2.2, Step 6.1.7 NOTE "Hydrogen concentration does **NOT** have to be reduced at Hot Shutdown conditions. However, it should be less than 8 cc/kg **PRIOR** to addition of hydrogen peroxide at 140°F to 180°F. Therefore, the cooldown may proceed before hydrogen concentration has been reduced." Hydrogen peroxide is added to the RCS in O-2.2, Step 6.9.5. Hydrogen concentration is reduced to minimize explosion hazards if the RCS is to be opened.

Technical Reference(s): O-2.2, Plant Shutdown from Hot Shutdown to Cold Conditions (p15 & 60; Rev 165)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1601C 1.02j & 1.03a, f

Question Source: Bank # ____

Modified Bank # ____

New X

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ____

10 CFR Part 55 Content:	55.41	.5
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005 K4.01	
	Importance Rating	3.0	

Knowledge of RHRS design feature(s) and/or interlock(s) which provide for the following: Overpressure mitigation system

RO Question #31

Which ONE of the following is the <u>permissive interlock value</u> that must be satisfied to open MOV-700, RHR Pump Suction from Loop A Hot Leg, and MOV-721, RHR Pump Discharge to Loop B Cold Leg?

A. RCS Temperature setpoint is 280°F

B. RCS Pressure setpoint is 410 psig

C. RCS Pressure setpoint is 600 psig

D. RCS Temperature setpoint is 350°F

Answer: B

Explanation:

A. INCORRECT. Plausible since O-2.2, Step 6.5.1 NOTE "It is desirable that RHR **NOT** be placed in service **UNLESS** cooldown is to continue to less than 280°F RCS Temperature." Incorrect since this is a procedure requirement and not an actual interlock.

B. CORRECT. According to UFSAR, Section 5.4.5.3.1.1 "Permissive interlocks required to open the four residual heat removal system isolation valves are listed below. MOV-700 and MOV-721: Reactor coolant system pressure must be less than 410 psig."

C. INCORRECT. Plausible since this is the design pressure of the RHR System piping and there is an RCS pressure interlock for these valves. Incorrect since this is not the correct pressure setpoint for the interlock.

D. INCORRECT. Plausible since O-2.2 requires that the RCS temperature be between 330°F and 350°F to align RHR System for shutdown operations. Incorrect since this is a procedure requirement and not an actual interlock.

Technical Reference(s): UFSAR, Section 5.4.5.3.1.1 & 5.4.5.3.2.1 (Rev 28)

O-2.2, Plant Shutdown from Hot Shutdown to Cold Conditions (p40 & 47; Rev 165)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2501C 1.07

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 .7

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 K2.02	
	Importance Rating	2.5*	

Knowledge of bus power supplies to the following: Valve operators for accumulators (ECCS)

RO Question #32

Given the following plant conditions:

- Plant is operating at 100% power
- A LOCA occurs
- Bus 14 Normal Feed Breaker has tripped
- Annunciator L-5, SAFEGUARDS BUS MAIN BREAKER OVERCURRENT TRIP, is LIT
- US has transitioned to E-1, Loss of Reactor or Secondary Coolant

NOTE: **MOV-841, SI ACCUMULATOR A DISCH TO LOOP B**
MOV-865, SI ACCUMULATOR B DISCH TO LOOP A

Which ONE of the following correctly completes the statements below?

MOV-841 and MOV-865 breakers are normally ______(1)__________ during plant operation.
AND
For the given plant conditions, power is unavailable to manually operate ____(2)____ if SI Accumulator isolation is required.

A. (1) closed
(2) MOV-841

B. (1) closed
(2) MOV-865

C. (1) open
(2) MOV-841

D. (1) open

(2) MOV-865

Answer: **C**

Explanation:

A. INCORRECT. The first part is plausible since the Accumulator Discharge Valves perform a critical safety function and some SI valves receive an open signal on SI actuation. Incorrect since the Accumulator Discharge Valves are normally open and de-energized at power. The second part is correct.

B. INCORRECT. The first part is plausible since the Accumulator Discharge Valves perform a critical safety function and some SI valves receive an open signal on SI actuation. Incorrect since the Accumulator Discharge Valves are normally open and de-energized at power. The second part is plausible since Accumulator ‘B’ discharges to ‘A’ Loop and the Applicant may misinterpret that MOV-865 is powered from Bus 14 (‘A’ Train). Incorrect since MOV-841 receives power from MCC ‘C’, which is powered from Bus 14 and is unavailable.

C. CORRECT. According to UFSAR, Table 6.3-8 MOV-841 and MOV-865 “Accumulator discharge to loop cold legs. They are normally open. Alternating current power is removed during operation.” According to P-12, Attachment 24, MOV-841 receives power from MCC ‘C’, which is powered from Bus 14. With the given plant conditions that Bus 14 normal feed breaker as tripped and Annunciator L-5 is lit, Bus 14 is de-energized. Therefore, MOV-841 is not available to be manually closed from the MCB.

D. INCORRECT. The first part is correct. The second part is plausible since Accumulator ‘B’ discharges to ‘A’ Loop and the Applicant may misinterpret that MOV-865 is powered from Bus 14 (‘A’ Train). Incorrect since MOV-841 receives power from MCC ‘C’, which is powered from Bus 14 and is unavailable.

Technical Reference(s): UFSAR, Table 6.3-8 (Rev 28)

P-12, Electrical Systems Precautions, Limitations, and Setpoints (p47 & 50; Rev 029)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2701C 1.05c

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2013 VC Summer ILT NRC

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

10 CFR Part 55 Content:	55.41	.8
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 K4.01	
	Importance Rating	2.6	

Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: Quench tank cooling

RO Question #33

Which ONE of the following correctly completes the statements below?

(1) Following a discharge from the PORVs, the PRT is cooled by RMW supplied to the ____________ of the PRT.
AND
(2) The warm PRT water is drained directly to the ________________.

A. (1) top spray nozzles
(2) Reactor Coolant Drain Tank (RCDT)

B. (1) top spray nozzles
(2) Waste Holdup Tank (WHUT)

C. (1) bottom perforated distribution header
(2) Reactor Coolant Drain Tank (RCDT)

D. (1) bottom perforated distribution header
(2) Waste Holdup Tank (WHUT)

Answer: A

Explanation:

A. CORRECT. According to UFSAR, Section 5.4.8.1 “If the temperature in the tank rises above 120°F during plant operation, the tank is cooled by spraying in cool makeup water and draining out the warm mixture to the reactor coolant drain tank.” Additionally, UFSAR Figure 5.4-9 shows that the demineralized spray water (RMW) is connected to the top of the PRT.

B. INCORRECT. The first part is correct. The second part is plausible the RCDT is drained to the WHUT and the Applicant may misinterpret that the PRT is also drained directly to the

WHUT. Incorrect since the PRT is drained to the RCDT.

C. INCORRECT. The first part is plausible since this is the location of the perforated distribution header that the PORVs discharge into and the Applicant may misinterpret that this is the location of the RMW inlet. Incorrect since RMW is connected to the spray nozzles in the top of the PRT. The second part is correct.

D. INCORRECT. The first part is plausible since this is the location of the perforated distribution header that the PORVs discharge into and the Applicant may misinterpret that this is the location of the RMW inlet. Incorrect since RMW is connected to the spray nozzles in the top of the PRT. The second part is plausible the RCDT is drained to the WHUT and the Applicant may misinterpret that the PRT is also drained directly to the WHUT. Incorrect since the PRT is drained to the RCDT.

Technical Reference(s): UFSAR, Section 5.4.8.1, Figure 5.4-9 (Rev 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1401C 1.03d, 1.07b

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008 A1.01	
	Importance Rating	2.8	

Ability to predict and/or monitor changes to parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW flow rate

RO Question #34

Given the following plant conditions:

- Plant is shutting down for a forced outage
- The operating crew is preparing to align RHR for Shutdown Cooling operation
- 'A' CCW Pump is running

Which ONE of the following correctly completes the statements below?

(1) Minimum CCW System flow rate with one CCW Pump in continuous operation is ____________.

AND

(2) After starting the second CCW Pump to support RHR Cooling, CCW flow must be throttled to less than or equal to ________________.

A. (1) 330 gpm
(2) 2400 gpm

B. (1) 330 gpm
(2) 4900 gpm

C. (1) 435 gpm
(2) 2400 gpm

D. (1) 435 gpm
(2) 4900 gpm

Answer: **D**

Explanation:

A. INCORRECT. The first part is plausible since the minimum flow of 165 gpm to each RCP will result in a MCB alarm and the Applicant may determine that this is the minimum required CCW System flow rate. Incorrect since the minimum CCW flow rate is 435 gpm. The second part is plausible since this is the maximum flow for a single CCW Pump and Heat Exchanger and the Applicant may misinterpret the requirement for CCW System alignment to support RHR Cooling. Incorrect since both CCW Pumps and Heat Exchangers must be in service to support RHR Cooling.

B. INCORRECT. The first part is plausible since the minimum flow of 165 gpm to each RCP will result in a MCB alarm and the Applicant may determine that this is the minimum required CCW System flow rate. Incorrect since the minimum CCW flow rate is 435 gpm. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since this is the maximum flow for a single CCW Pump and Heat Exchanger and the Applicant may misinterpret the requirement for CCW System alignment to support RHR Cooling. Incorrect since both CCW Pumps and Heat Exchangers must be in service to support RHR Cooling.

D. CORRECT. According to S-8A, Step 6.12.10 CAUTION “The minimum flow for one CCW Pump during continuous operation **SHOULD NOT** go below 435 gpm. When starting a pump always verify that sufficient CCW flow paths are in service to satisfy the minimum flow requirements.” According to O-2.2, Step 6.3.13.4, the Operator is directed to establish two CCW Pump and Heat Exchangers in service prior to establishing RHR Cooling. S-8A, Step 6.21.11 NOTE “**MAXIMUM** CCW flow is defined as less than 2400 gpm, with **ONE** CCW Heat Exchanger (1 CCW Pump) and less than 4900 gpm, with **TWO** (2) CCW Heat Exchangers (2 CCW Pumps) in service.”

Technical Reference(s):
- S-8A, Component Cooling Water System Startup and Normal Operation Valve Alignment (p22&42; Rev 059)
- O-2.2, Plant Shutdown from Hot Shutdown to Cold Conditions (p31; Rev 165)
- AR-A-7, RCP A CCW RETURN HI TEMP OR LO FLOW 165 GPM 125°F (Rev 00801)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2801C 1.07, 1.09

Question Source:
- Bank #
- Modified Bank #
- New X

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	.5
	55.43	

Comments:

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

Knowledge of bus power supplies for the following: Indicator for PORV position

RO Question #35

Which ONE of the following correctly completes the statement below?

Pressurizer PORV PCV-431C control power is powered from the ___(1)___ DC Bus. If the DC Bus to this valve is de-energized, PCV-431C will indicate___(2)___.

A. (1) ‘A’
(2) GREEN light LIT, RED light OFF

B. (1) ‘A’
(2) Both lights OFF

C. (1) ‘B’
(2) GREEN light LIT, RED light OFF

D. (1) ‘B’
(2) Both lights OFF

Answer: D

Explanation:

A. INCORRECT. The first part is plausible since PORV PCV-430 control power is powered from ‘A’ DC Bus and the Applicant may misinterpret that PCV-431C is also powered from ‘A’ DC Bus. Incorrect since PCV-431C control power is powered from ‘B’ DC Bus. The second part is plausible since this condition would be correct with a loss of ‘A’ DC Bus. Incorrect since both lights would be extinguished on a loss of ‘B’ DC Bus.

B. INCORRECT. The first part is plausible since PORV PCV-430 control power is powered from ‘A’ DC Bus and the Applicant may misinterpret that PCV-431C is also powered from ‘A’ DC Bus. Incorrect since PCV-431C control power is powered from ‘B’ DC Bus. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since this condition would be correct with a loss of ‘A’ DC Bus. Incorrect since both lights would be extinguished on a loss of ‘B’ DC Bus.

D. CORRECT. According to ER-ELEC.2, Attachment 1, page 3 of 3, for a Loss of DC Train B, the table shows that PORV-431C will not work on N2 (8619B) nor Instrument Air (8620B). These are the solenoids that operate allowing PORV-431C to OPEN when required. According to P-10, Step 5.13.3.2 “Breaker status and valve position indication will be lost for equipment supplied by the failed DC bus.” Therefore, the RED and GREEN indicator lights for PORV-431C will be extinguished.

Technical Reference(s):
- ER-ELEC.2, Recovery from Loss of A or B DC Train (p 16-18; Rev 016)
- P-10, Instrument Failure Reference Manual (p73; Rev 022)
- 21946-0751B, SOV-8619B N2 Arming Vlv Control Schematic (Rev 1)
- 21946-0755, PCV-431C PRZR Power Operated Relief Vlv Control Schematic (Rev 1)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0901C 1.06

Question Source:
- Bank #: X
- Modified Bank #:
- New:

Question History: Last NRC Exam 2014 Ginna ILT NRC

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

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

Comments:

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

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

RO Question #36

Given the following plant conditions:

- Reactor power is 100%
- Control Rods are in AUTO
- PI-430, Pressurizer Pressure, failed low and has been defeated in accordance with ER-INST.1, Reactor Protection Bistable Defeat After Instrumentation Loop Failure

Subsequently:

- Channel 4 T_{HOT} experiences an OPEN RTD failure

Which **ONE** of the following describes the effect of these conditions on the Rod Control System?

A. Control rods drive IN due to Average T_{AVG} failed high

B. Control rods drive IN due to a single Loop T_{AVG} failed high

C. Reactor trip breakers OPEN due to two channels OTΔT bistables tripped

D. Reactor trip breakers OPEN due to two channels OPΔT bistables tripped

Answer: C

Explanation:

A. INCORRECT. Plausible since P-10, Section 5.1.3 shows that for T_{HOT} RTD open, Average T_{AVG} will fail to ≥576.66°F. Incorrect since the Pressurizer Pressure channel defeat (for the channel failure) includes a defeat of OTΔT bistables which would result in a reactor trip.

B. INCORRECT. Plausible since P-10, Section 5.1.3 shows that for T_{HOT} RTD open, Loop T_{AVG} will fail to ≥ 584.625°F and Loop T_{AVG} is an input into the Average T_{AVG} computer. Incorrect since the Pressurizer Pressure channel defeat (for the channel failure) includes a defeat of

OTΔT bistables which would result in a reactor trip.

C. CORRECT. Channel 2 Pressurizer pressure defeat of PI-430 for channel failure, as given in the stem, includes a defeat of the Over Temperature ΔT trip in accordance with ER-INST.1, Attachment 6, Step 5.0. This defeat, combined with P-10, Section 5.1.3, indicates that a T_{HOT} open would result in Loop ΔT to exceed 85°F. This exceeds the trip setpoint for OTΔT and results in 2 out of 4 OTΔT channels exceeding the setpoint. According to P-1, Attachment 1, OTΔT reactor trip Coincidence is 2 of 4 channels. Additionally, ER-INST.1, Step 6.3.4 NOTE states "Completing the attachment changes ΔT runback and rod stop logic to 2/3, and changes ΔT reactor trip logic to 1/3 on the remaining channels."

D. INCORRECT. Plausible since (without reference) the Applicant may interpret that a defeat of Pressurizer Pressure PI-430 would include a defeat of Over Power ΔT. Incorrect since ER-INST.1, Attachment 6, Step 5.0 shows that a defeat of PI-430 includes a defeat of Over Temperature ΔT channel, not Over Power ΔT.

Technical Reference(s):
- P-10, Instrument Failure Reference Manual (p12; Rev 022)
- P-1, Reactor Control and Protection System (p40; Rev 078)
- ER-INST.1, Reactor Protection Bistable Defeat After Instrumentation Loop Failure (p14 & 54; Rev 040)
- Technical Specification LCO 3.3.1 Basis (p B3.3.1-3 & B3.3.1-48; Rev 61)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3501C 1.09

Question Source:
- Bank # X
- Modified Bank #
- New

Question History: Last NRC Exam 2015 South Texas ILT

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

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

Comments: K/A is matched since according to Technical Specification LCO 3.3.1 Basis “The RTS instrumentation is segmented into three distinct but interconnected modules as described in UFSAR, Chapter 7 (Ref. 4):

a. Field transmitters or process sensors;
b. Signal process control and protection equipment; and
c. Reactor trip switchgear

These modules are shown in Figure B3.3.1-1 and discussed in more detail below.”

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 A3.06	
	Importance Rating	3.7	

Ability to monitor automatic operation of the RPS, including: Trip logic

RO Question #37

Given the following plant conditions:

- Reactor power is 75% and stable
- Power Range Channel N-41 fails LOW
- The crew is preparing to defeat N-41 in accordance with ER-NIS.3, PR Malfunction

Which ONE of the following combinations of OPERABLE Power Range Channels indicates the status of the trip logic coincidence that would result in a Power Range High Flux Trip?

1) PRIOR to Power Range Channel N-41 being defeated
AND
2) AFTER Power Range Channel N-41 is defeated

	PRIOR to Channel Defeat	**AFTER Channel Defeat**
A.	1/3	1/3
B.	2/3	1/3
C.	1/3	2/3
D.	2/3	2/3

Answer: **B**

Explanation:

A. INCORRECT. The first part is plausible since the Applicant may misinterpret the trip logic before N-41 is removed from service and determine that the logic has changed to 1/3, which would be the same as post-defeat. Incorrect since N-41 has failed low, it is not providing an input signal into the Power Range High Flux reactor trip logic. Therefore, the trip logic would still require two valid signals. The second part is correct.

B. CORRECT. Since N-41 has failed low, the Power Range High Flux Trip Logic will never receive a trip signal from N-41. Since two high flux signals are required to initiate a trip signal and there are only three channels remaining which could potentially generate a Power Range High Flux Trip signal, the logic for a Power Range High Flux trip is 2/3 channels. In accordance with ER-NIS.3, Step 11.0 NOTE “Performing the next step will remove the ΔI signal to the ΔT channel and place the 108% bistable in the tripped condition.” This provides one of the two required Power Range High Flux Trip signals requiring only one additional trip signal from the three remaining channels.

C. INCORRECT. The first part is plausible since the Applicant may misinterpret the trip logic before N-41 is removed from service and determine that the logic has changed to 1/3, which would be the same as post-defeat. Incorrect since N-41 has failed low, it is not providing an input signal into the Power Range High Flux reactor trip logic. Therefore, the trip logic would still require two valid signals. The second part is plausible since ER-NIS.3 places the affected Power Range DROPPED ROD MODE switch to BYPASS which causes Annunciator E-7, NIS TRIP BYPASS, to alarm and the Applicant may misinterpret that this means that the N-41 trip signals are all bypassed. Incorrect since removing the INSTR POWER fuses in Step 11.0 of the defeat attachment trips the Power Range High Flux Trip bistable inserting a trip signal, requiring only one additional signal from the three remaining channels.

D. INCORRECT. The first part is correct. The second part is plausible since ER-NIS.3 places the affected Power Range DROPPED ROD MODE switch to BYPASS which causes Annunciator E-7, NIS TRIP BYPASS, to alarm and the Applicant may misinterpret that this means that the N-41 trip signals are all bypassed. Incorrect since removing the INSTR POWER fuses in Step 11.0 of the defeat attachment trips the Power Range High Flux Trip bistable inserting a trip signal, requiring only one additional signal from the three remaining channels.

Technical Reference(s): P-1, Reactor Control and Protection System (p40; Rev 078)

ER-NIS.3, PR Malfunction (p8-11; Rev 027)

Proposed references to be provided to applicants during examination: None

Learning Objective: RIC10C 1.06

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2009 McGuire ILT RO Retake

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

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013 K6.01	
	Importance Rating	2.7*	

Knowledge of the effect of a loss or malfunction of the following will have on the ESFAS: Sensors and detectors

RO Question #38

Given the following plant conditions:

- Plant is operating at 100% power
- PT-945, CNMT CH 1A PRESSURE, fails to 0 psig and is DEFEATED
- All bistable status lights with the associated CNMT Pressure channel are LIT

Subsequently:

- PT-949, CNMT CH 3A PRESSURE, fails to 60 psig

What is the plant response, if any, to the second CNMT Pressure channel failure?

A. ONLY Safety Injection will occur

B. ONLY Steamline Isolation Actuation will occur

C. Safety Injection, Steamline Isolation, and CNMT Spray Actuations will ALL occur

D. NO Engineered Safety Feature Actuation will occur

Answer: A

Explanation:

A. CORRECT. According to ER-INST.1, Attachment 18, defeating PT-945 will insert a trip signal for CNMT Hi Pressure SI and CNMT Spray. The failure of the second CNMT Pressure transmitter PT-949 provides the required coincidence for Safety Injection actuation (2/3 on PT-945/PT-947/PT-949). Steamline Isolation will not occur since the 2/3 coincidence on PT-946/PT-948/PT-950 is NOT met. Containment Spray actuation will NOT occur since the 2/3 on 2/2 sets (PT-945/PT-947/PT-949 + PT/946/PT-948/PT-950) is NOT met.

B. INCORRECT. Plausible since Steamline Isolation actuation is met with a 2/3 coincidence on CNMT Pressure transmitters and the Applicant may misinterpret which set has tripped. Incorrect since Steamline Isolation will not occur due to the 2/3 coincidence on PT-946/PT-948/PT-950 is NOT met. Incorrect since the CNMT pressure transmitters that have failed do NOT provide input into the Steamline Isolation circuitry.

C. INCORRECT. Plausible since the logic is met for Safety Injection actuation, and the Applicant may misinterpret which CNMT Pressure transmitters provide input into the Steamline Isolation and CNMT Spray actuation circuits. The given plant conditions place the first set in 2/3 coincidence and the Applicant may misinterpret this as meeting the required actuation coincidence. Incorrect since the CNMT pressure transmitters that have failed do NOT provide input into the Steamline Isolation circuitry and since CNMT Spray actuation will NOT occur due to the 2/3 on 2/2 sets (PT-945/PT-947/PT-949 + PT/946/PT-948/PT-950) is NOT met.

D. INCORRECT. Plausible since the Applicant may misinterpret which CNMT Pressure transmitters belong to each set and conclude that the required coincidence for any ESFAS Actuation has not been met. Incorrect since the failure of the second CNMT Pressure transmitter PT-949 provides the required coincidence for Safety Injection actuation (2/3 on PT-945/PT-947/PT-949).

Technical Reference(s): P-1, Reactor Control and Protection System (p16-17, 45-46; Rev 078)

ER-INST.1, Reactor Protection Bistable Defeat after Instrumentation Loop Failure (p83-84; Rev 040)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3501C 1.06, 1.07c, 1.09d

Question Source: Bank # X

Modified Bank # ____

New ____

Question History: Last NRC Exam ____

Question Cognitive Level: Memory or Fundamental Knowledge ____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .7

55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 A1.02	
	Importance Rating	3.6	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment pressure

RO Question #39

Given the following initial plant conditions:

- Plant is operating at 100% power
- ‘A’ CNMT Recirc Fan is tagged for repairs

Subsequently:

- A Design Basis LOCA occurs inside CNMT
- Both Offsite Power circuits are lost
- ‘A’ Emergency Diesel Generator (EDG) fails to start and cannot be started
- CNMT Pressure is 45 psig and rising

Based on the given plant conditions, CNMT maximum design pressure of ______(1)____ ______(2)____ be exceeded.

A. (1) 55 psig
(2) WILL

B. (1) 55 psig
(2) WILL NOT

C. (1) 60 psig
(2) WILL

D. (1) 60 psig
(2) WILL NOT

Answer: D

Explanation:

A. INCORRECT. The first part is plausible since this is the peak pressure expected following a double ended pump suction break with minimum safeguards and the Applicant may misinterpret this to be the CNMT maximum design pressure. Incorrect since the CNMT design maximum pressure is 60 psig. The second part is plausible if the Applicant misinterprets the given plant conditions and determines that the minimum required CNMT Recirc Fan Coolers will not be available. Incorrect since the given plant conditions will result in 'B' and 'C' CNMT Recirc Fans being available and operating.

B. INCORRECT. The first part is plausible since this is the peak pressure expected following a double ended pump suction break with minimum safeguards and the Applicant may misinterpret this to be the CNMT maximum design pressure. Incorrect since the CNMT design maximum pressure is 60 psig. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible if the Applicant misinterprets the given plant conditions and determines that the minimum required CNMT Recirc Fan Coolers will not be available. Incorrect since the given plant conditions will result in 'B' and 'C' CNMT Recirc Fans being available and operating.

D. CORRECT. According to Technical Specification LCO 3.6.4 Basis "The maximum containment pressure resulting from the worst case SLB, 59.7 psig, does not exceed the containment design pressure, 60 psig." Technical Specification LCO 3.6.6 Basis states "During a DBA, a minimum of 2 CRFC units and one CS train are required to maintain the containment peak pressure and temperature below the design limits." According to P-12, Bus 14 powers CNMT Spray Pump 1A, CNMT Recirc Fans 1A and 1D, and Bus 16 powers CNMT Spray Pump 1B, CNMT Recirc Fans 1B and 1C. Therefore, with the loss of Offsite Power, 'B' EDG will start and load onto Buses 16 and 17 and supply power to CNMT Recirc Fans 'B' and 'C' and CNMT Spray Pump 'B', which is the minimum required to prevent exceeding CNMT design pressure and temperature. These components will automatically start upon SI and CNMT Spray Actuations.

Technical Reference(s):	Technical Specification LCO 3.6.4 Basis (p3.6.4-1; Rev 72)
	Technical Specification LCO 3.6.6 Basis (p3.6.6-6; Rev 76)
	P-12, Electrical Systems Precautions, Limitations, and Setpoints (p35 & 37; Rev 029)
	UFSAR Section 6.2.1.2.2.8 (Rev 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2101C 1.07; R2201C 1.12b; R2401C 1.12b

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	.5
	55.43	

Comments: Ginna's Facility Representative has determined that this is RO knowledge since the question is testing systems knowledge of CNMT design features and UFSAR accident analysis worst-case failure scenario knowledge.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 A2.05	
	Importance Rating	3.1	

Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Major leak in CCS

RO Question #40

Given the following plant conditions:

- Plant is operating at 100% power
- The Operating crew is performing actions in AP-SW.1, Service Water Leak
- The SW leak is in Containment

(1) In accordance with AP-SW.1, what indications will be used to determine if a CNMT Recirc Fan Cooler (CRFC) is leaking;

AND

(2) What isolation valves will be closed to isolate the leaking CRFC in accordance with ATT-2.3, Attachment SW Loads in CNMT?

A. (1) CRFC SW outlet flows
(2) SW Inlet Isolation Valve, ONLY

B. (1) CRFC SW outlet pressures
(2) SW Inlet Isolation Valve, ONLY

C. (1) CRFC SW outlet flows
(2) SW Inlet and Outlet Isolation Valves

D. (1) CRFC SW outlet pressures
(2) SW Inlet and Outlet Isolation Valves

Answer: **C**

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since some SW supplied coolers have separate inlet and outlet lines with check valves installed on the outlet of the cooler requiring only the inlet isolation valve to be closed to stop a leak. Incorrect since the CRFCs have common discharge lines with NO check valves installed requiring both the inlet and outlet isolation valves to be closed to stop the leak.

B. INCORRECT. The first part is plausible since SW outlet pressures are used to determine if a Reactor Compartment cooler is leaking. Incorrect since AP-SW.1 directs checking CRFC SW outlet flows to identify the leaking CRFC. The second part is plausible since some SW supplied coolers have separate inlet and outlet lines with check valves installed on the outlet of the cooler requiring only the inlet isolation valve to be closed to stop a leak. Incorrect since the CRFCs have common discharge lines with NO check valves installed requiring both the inlet and outlet isolation valves to be closed to stop the leak.

C. CORRECT. In accordance with AP-SW.1, Step 5 directs the Operator to check "CNMT recirc fan SW flows – APPROXIMATELY EQUAL." ATT-2.3 directs the Operator to isolate CNMT recirc fan coolers one at a time by closing the SW inlet and SW outlet valves to each cooler.

D. INCORRECT. The first part is plausible since SW outlet pressures are used to determine if a Reactor Compartment cooler is leaking. Incorrect since AP-SW.1 directs checking CRFC SW outlet flows to identify the leaking CRFC. The second part is correct.

Technical Reference(s): AP-SW.1, Service Water Leak (p7; Rev 02300)
ATT-2.3, Attachment SW Loads in CNMT (Rev 4)

Proposed references to be provided to applicants during examination: None

Learning Objective: R5101C 1.06

Question Source:
Bank # ____
Modified Bank # ____
New X

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:	55.41	.5
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026 A4.05	
	Importance Rating	3.5	

Ability to manually operate and/or monitor in the control room: Containment spray reset switches (Containment Spray System)

RO Question #41

Given the following initial plant conditions:

- The plant tripped and SI occurred due to a Design-Basis LOCA
- The HCO had to manually actuate CNMT Spray

Currently:

- CNMT pressure has lowered to 24 psig and slowly lowering

Which ONE of the following correctly completes the statements below?

1) To manually actuate CNMT Spray, the HCO depressed _________ Manual CNMT Spray actuation pushbuttons.

   AND

2) For the current CNMT conditions, if the HCO depresses the CNMT Spray RESET pushbutton, the CNMT Spray actuation signal ___________ reset.

A. 1) 1 of 2
   2) will

B. 1) 1 of 2
   2) will NOT

C. 1) 2 of 2
   2) will

D. 1) 2 of 2
   2) will NOT

Answer: **C**

Explanation:

A. INCORRECT. The first part is plausible since the Applicant may misinterpret that the manual CNMT Spray signal is operates similar to the manual reactor trip pushbuttons in that only one is required to be depressed to actuate the signal. Incorrect since both Manual CNMT Spray pushbuttons must be depressed at the same time to actuate the CNMT Spray signal. The second part is correct.

B. INCORRECT. The first part is plausible since the Applicant may misinterpret that the manual CNMT Spray signal is operates similar to the manual reactor trip pushbuttons in that only one is required to be depressed to actuate the signal. Incorrect since both Manual CNMT Spray pushbuttons must be depressed at the same time to actuate the CNMT Spray signal. The second part is plausible since the Applicant may misinterpret that the setpoint is 18 psig for CNMT Spray. Incorrect since 18 psig is the Steam Line Isolation setpoint for CNMT pressure and the CNMT Spray resets at 25 psig.

C. CORRECT. According to 33013-1353, Sheet 7, both Manual CNMT Spray pushbuttons must be depressed to actuate CNMT Spray. Additionally, AR-A-27 states that an alarm setpoint is 2/2 Manual Spray. AR-A-27 Comments states that the “Alarm resets at ≤ 25 psig.”

D. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret that the setpoint is 18 psig for CNMT Spray. Incorrect since 18 psig is the Steam Line Isolation setpoint for CNMT pressure and the CNMT Spray resets at 25 psig.

Technical Reference(s): 33013-1353, Sheet 7 Logic Diagram Safeguards Actuation Signals (Rev 6)

AR-A-27, CONTAINMENT SPRAY 2/3 + 2/3 > 28 PSI (Rev 9)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2401C 1.02c

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2015 Vogtle ILT

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 A3.02	
	Importance Rating	3.1	

Ability to monitor automatic operation of the MRSS, including:
Isolation of the MRSS

RO Question #42

To protect the Main Turbine from overspeed following a loss of generator load, the Reheater ______(1)____ Valves will CLOSE first at __(2)____ of rated speed (1800 rpm).

A. (1) Intercept
(2) 103%

B. (1) Intercept
(2) 109.3%

C. (1) Stop
(2) 103%

D. (1) Stop
(2) 109.3%

Answer: A

Explanation:

A. CORRECT. According to UFSAR, Section 10.2.3.4.2 “If the auxiliary governor senses an overspeed condition at 103% of 1800 rpm, high-pressure fluid from the top of the dump valve pistons of the reheater intercept valves will be dumped through the stop valve emergency trip line in the drain header. This action will close the reheater intercept valves.”

B. INCORRECT. The first part is correct. The second part is plausible since this is the setpoint which causes the Reheater Stop Valves to close. Incorrect since the Reheater Intercept Valves close at 103% of turbine rated speed.

C. INCORRECT. The first part is plausible since the Reheater Stop Valves provide backup overspeed protection for the Reheater Intercept Valves. Incorrect since the Reheater Intercept Valves close first. The second part is correct.

D. INCORRECT. The first part is plausible since the Reheater Stop Valves provide backup overspeed protection for the Reheater Intercept Valves. Incorrect since the Reheater Intercept Valves close first. The second part is plausible since this is the setpoint which causes the Reheater Stop Valves to close. Incorrect since the Reheater Intercept Valves close at 103% of turbine rated speed.

Technical Reference(s): UFSAR Sections 10.2.3.4.2 & 10.3.2.11 (Rev 28)
R4001C, Main Steam System Presentation (slide 87)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4001C 1.02k, 1.09

Question Source:
Bank # X
Modified Bank #
New

Question History: Last NRC Exam 2013Catawba ILT

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

10 CFR Part 55 Content:
55.41 .7
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059 A4.11	
	Importance Rating	3.1	

Ability to manually operate and monitor in the control room: Recovery from automatic feedwater isolation (Main Feedwater (MFW) System)

RO Question #43

Given the following plant conditions:

- Plant is operating at 100% reactor power

Which ONE of the following correctly completes the statements below?

1) The setpoint for automatic feedwater isolation is a S/G water level of __________.

   AND

2) When S/G water level lowers below the setpoint, the associated Feed Regulating Valve __________________ to control S/G water level.

A. 1) 80%
   2) will automatically re-open

B. 1) 80%
   2) must be manually re-opened

C. 1) 85%
   2) will automatically re-open

D. 1) 85%
   2) must be manually re-opened

Answer: C

Explanation:

A. INCORRECT. The first part is plausible since A-503.1, Step 5.3.B.1.d.(1) directs the Operator to secure/throttle AFW flow to an intact S/G at this S/G water level during adverse CNMT conditions to prevent the S/G becoming unavailable for RCS cooldown. Incorrect since the FRVs will not receive a close signal until 85% is reached in the associated S/G.

The second part is correct.

B. INCORRECT. The first part is plausible since A-503.1, Step 5.3.B.1.d.(1) directs the Operator to secure/throttle AFW flow to an intact S/G at this S/G water level during adverse CNMT conditions to prevent the S/G becoming unavailable for RCS cooldown. Incorrect since the FRVs will not receive a close signal until 85% is reached in the associated S/G. The second part is plausible since many components that automatically isolate must be manually realigned. Incorrect since the FRV will automatically re-open once the high S/G water level condition has cleared.

C. CORRECT. AR-G-12 states "85% Feedwater Isolation will occur" and directs the Operator to verify the FRV and bypass valves are closed. Technical Specification LCO 3.7.3 Basis states "The MFRVs and bypass valves close on receipt of a safety injection signal, a SG high level signal, or on a reactor trip with T_{avg} < 554°F with the associated MFRV in auto." UFSAR, Section 15.1.2.1.1states "When the water level drops below the high steam generator water level setpoint, the closure signal clears and the valves will reopen, potentially causing the steam generator water level to increase."

D. INCORRECT. The first part is correct. The second part is plausible since many components that automatically isolate must be manually realigned. Incorrect since the FRV will automatically re-open once the high S/G water level condition has cleared.

Technical Reference(s):	AR-G-12, S/G A HI LEVEL 85% (Rev 8)
	Technical Specification LCO 3.7.3 Basis (p B3.7.3-1; Rev 58)
	UFSAR Section 15.1.2.1.1 (Rev 28)
	10905-0733, Main FW Control AOV to S/G A 4269 Elementary Wiring Diagram (Rev 4)
	A-503.1, Emergency and Abnormal Operating Procedures Users Guide (p42; Rev 053)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4301C 1.07v

Question Source: Bank # ______

Modified Bank # ______

New X

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments: K/A is matched since at Ginna an automatic feedwater isolation occurs at 85% S/G water level. The associated Main Feedwater regulating and Bypass Valves CLOSE and will re-open once S/G water level lowers below 85%. The Feedwater Isolation signal removes the ability for the Operator to manually operate the associated Feedwater Regulating Valve(s).

Additionally, NUREG-1122, Section 1.9.2 states that the Ability Stem Statement for Plant Systems A4 is “Ability to manually operate and/or monitor in the control room”; whereas for System 059, Main Feedwater (MFW) System A4 states “Ability to manually operate and monitor in the control room”.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059 G2.1.28	
	Importance Rating	4.1	

Knowledge of the purpose and function of major system components and controls. (Main Feedwater (MFW) System)

RO Question #44

Given the following plant conditions:

- The Operating crew is performing AP-TURB.5, Rapid Load Reduction, to take the plant off-line
- The Operating crew is preparing to stop a Main Feedwater (MFW) Pump

Which ONE of the following completes the statements below?

(1) The highest Main Feedwater flow which will result in the MFW Pump Recirc Valves opening is approximately _______________ and lowering;

AND

(2) The CO will CLOSE the MFW Pump Recirc Valve for the MFW Pump being stopped by placing the MFW Pump control switch in _____________________.

A. (1) 500 gpm
(2) PULL STOP

B. (1) 500 gpm
(2) STOP and releasing to mid-position

C. (1) 2000 gpm
(2) PULL STOP

D. (1) 2000 gpm
(2) STOP and releasing to mid-position

Answer: **C**

Explanation:

A. INCORRECT. The first part is plausible since this is the setpoint for the HDT Pump Recirc Valve and the Applicant may misinterpret this for the MFW Pump Recirc Valve Setpoint. Incorrect since the MFW Pump Recirc Valve opens at approximately 2000 gpm. The second part is correct.

B. INCORRECT. The first part is plausible since this is the setpoint for the HDT Pump Recirc Valve and the Applicant may misinterpret this for the MFW Pump Recirc Valve Setpoint. Incorrect since the MFW Pump Recirc Valve opens at approximately 2000 gpm. The second part is plausible since the Applicant may misinterpret that the Recirc Valve closes once the associated MFW Pump is stopped. Incorrect since the MFW Pump control switch must be placed in PULL STOP in order to close the Recirc Valve.

C. CORRECT. According to AR-H-2, the setpoint for the MFW Pump Recirc Valve opening is 0.95 X 10^6 lbm/HR which is approximately 2000 gpm. In accordance with AP-TURB.5, Step 11.f directs the Operator to “Close the secured MFW pump recirc valve by placing the control switch in pull stop.”

D. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret that the Recirc Valve closes once the associated MFW Pump is stopped. Incorrect since the MFW Pump control switch must be placed in PULL STOP in order to close the Recirc Valve.

Technical Reference(s): AR-H-2, FEED WATER PUMP LIGHT LOAD (Rev 9)
AP-TURB.5, Rapid Load Reduction (p8; Rev 020)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4301C 1.07r

Question Source: Bank # ____
Modified Bank # ____
New X

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 .4
55.43 ____

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 K5.02	
	Importance Rating	3.2	

Knowledge of the operational implications of the following concepts as they apply to the AFW: Decay heat sources and magnitude

RO Question #45

In accordance with the UFSAR:

A MINIMUM of ___(1)____ MDAFW Pump(s) can supply 100% of the required feedwater for removal of decay heat from the plant for a MINIMUM of ______(2)____ after reactor trip from full power given the MINIMUM amount of water in the Condensate Storage Tanks.

A. (1) ONE
   (2) 2 hours

B. (1) ONE
   (2) 4 hours

C. (1) TWO
   (2) 2 hours

D. (1) TWO
   (2) 4 hours

Answer: A

Explanation:

A. CORRECT. In accordance with UFSAR, Section 10.5.2.2 "The turbine-driven auxiliary feedwater pump (TDAFW) can supply 200% of the required feedwater and one motor-driven auxiliary feedwater pump (MDAFW) can supply 100% of the required feedwater for removal of decay heat from the plant. The minimum amount of water in the condensate storage tanks (CST) (24,350 gal) is the amount needed to remove decay heat for 2 hr after reactor scram from full power.".

B. INCORRECT. The first part is correct. The second part is plausible since this is the completion time to "Verify by administrative means OPERABILITY of backup water supply" in accordance with Technical Specifications LCO 3.7.6, Condition A. Incorrect since the CST minimum water level is based on 2 hours.

C. INCORRECT. The first part is plausible since both MDAFW Pumps start on a trip from full power and the Applicant may misinterpret that both MDAFW Pumps or one MDAFW Pump and the TDAFW Pump are needed to supply 100% feedwater requirements. Incorrect since one MDAFW Pump will supply 100% of the required feedwater flow. The second part is correct.

D. INCORRECT. The first part is plausible since both MDAFW Pumps start on a trip from full power and the Applicant may misinterpret that both MDAFW Pumps or one MDAFW Pump and the TDAFW Pump are needed to supply 100% feedwater requirements. Incorrect since one MDAFW Pump will supply 100% of the required feedwater flow. The second part is plausible since this is the completion time to “Verify by administrative means OPERABILITY of backup water supply” in accordance with Technical Specifications LCO 3.7.6, Condition A. Incorrect since the CST minimum water level is based on 2 hours.

Technical Reference(s): UFSAR, Section 10.5.2.2 (Rev 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4201C 1.09

Question Source: Bank # X
Modified Bank # ______
New ______

Question History: Last NRC Exam 2016 Callaway ILT

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

10 CFR Part 55 Content: 55.41 .5
55.43 ______

Comments: K/A is matched since the question is asking the AFW requirements in accordance with the UFSAR regarding minimum AFW for decay heat removal and how long the minimum CST volume will last.

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

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

RO Question #46

Given the following plant conditions:

- Plant is operating at 99% power
- STP-O-16.3A, AFW Pump A Discharge MOV Test, is in progress
- MOV-4007, 1A AFW Pump Discharge MOV, is CLOSED

Subsequently:

- 'A' MDAFW Pump is started and trips after two (2) seconds

Which ONE of the following correctly describes the response of MOV-4007?

MOV-4007 will ____________________.

A. remain CLOSED

B. OPEN and remain fully OPEN

C. OPEN and throttle back to mid-position

D. OPEN and then fully CLOSE

Answer: **B**

Explanation:

A. INCORRECT. Plausible since the Applicant may misinterpret that MOV-4007 does not begin to open until the MDAFW Pump is supplying flow. Incorrect since MOV-4007 gets an open signal from the 'A' MDAFW Pump start signal and begins to immediately open.

B. CORRECT. According to 10905-0659, MOV-4007 OPEN contactor energizes when the 'A' MDAFW Pump breaker closes (52/MAFP1A contact). The OPEN contactor will remain energized until the valve reaches the fully open position (LS4 contact). Since there is no flow due to 'A' MDAFW Pump tripping, the throttle back circuit (FC-2001) will not close to energize the CLOSE contactor resulting in MOV-4007 remaining fully OPEN.

C. INCORRECT. Plausible since this is the normal operation of MOV-4007 if it was initially closed when 'A' MDAFW Pump starts and continues to run. Incorrect since the throttle back circuit (FC-2001) will not close to energize the CLOSE contactor resulting in MOV-4007 remaining fully OPEN.

D. INCORRECT. Plausible since this would be the operation if MOV-4007 control switch was taken to CLOSE after it was opened and the Applicant may misinterpret the function of the throttle back circuit. Incorrect since the throttle back circuit (FC-2001) will not close to energize the CLOSE contactor resulting in MOV-4007 remaining fully OPEN

Technical Reference(s): 10905-0659, MDAFW Pump A Discharge VLV MOV-4007 Elementary Wiring Diagram (Rev 6)

STP-O-16.3A, AFW Pump A Discharge MOV Test (p12-17; Rev 009)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4201C 1.07b

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2018 Turkey Point ILT

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .7

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062 A1.01	
	Importance Rating	3.4	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Significance of D/G load limits

RO Question #47

Given the following plant conditions:

- Plant is operating at 100% power
- STP-O-12.1, Emergency Diesel Generator A, is in progress
- 'A' EDG is paralleled with the grid, carrying 2050 kW load
- A grid disturbance causes grid frequency to lower very slightly
- Grid voltage remains constant

Which ONE of the following correctly completes the statements below?

(1) 'A' EDG response is ____________________;

AND

(2) The significance of operating the EDG above the two-hour load rating is ________.

A. (1) kW output RISES
(2) mechanical stress on the EDG engine

B. (1) kW output LOWERS
(2) mechanical stress on the EDG engine

C. (1) kW output RISES
(2) accumulation of combustion and lubricating products in the exhaust system

D. (1) kW output LOWERS
(2) accumulation of combustion and lubricating products in the exhaust system

Answer: **A**

Explanation:

A. CORRECT. As grid frequency lowers, the EDG will assume more real load (kW) since EDG speed is constant. Technical Specification Basis for Surveillance SR 3.8.1.3 states "The upper load band limit of < 2250 kW is the DG two-hour rating and is provided to avoid routine overloading of the DG which may result in more frequent inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY."

B. INCORRECT. The first part is plausible if the Applicant misinterprets the relationship between EDG and grid frequency relationships and the effect on load shifting. Incorrect since as grid frequency lowers, the EDG will assume more real load. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since this is the concern with light loading of the EDG and the Applicant may misinterpret the reasons why light and over-loading the EDG is a concern. Incorrect since overloading of the EDG puts additional stress on the diesel engine driving the generator.

D. INCORRECT. The first part is plausible if the Applicant misinterprets the relationship between EDG and grid frequency relationships and the effect on load shifting. Incorrect since as grid frequency lowers, the EDG will assume more real load. The second part is plausible since this is the concern with light loading of the EDG and the Applicant may misinterpret the reasons why light and over-loading the EDG is a concern. Incorrect since overloading of the EDG puts additional stress on the diesel engine driving the generator.

Technical Reference(s): Technical Specification Basis Surveillance SR 3.8.1.3 (Rev 92)

INPO GFE 191005, Motors and Generators (slides 102-107;Rev 3.1)

Proposed references to be provided to applicants during examination: None

Learning Objective: INPO GFE 191005 ELO 4.3

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2017 Beaver Valley ILT

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	.8
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063 G2.2.12	
	Importance Rating	3.7	

Knowledge of surveillance procedures. (DC Electrical Distribution)

RO Question #48

Given the following plant conditions:

- Plant is operating at 100% power
- The CO is performing Surveillance Requirement 3.8.4.1, Verify Battery Terminal Voltage in accordance with O-6.11, Surveillance Requirement/Routine Operations Check Sheet

Which ONE of the following correctly completes the statements below?

In accordance with Surveillance Requirement 3.8.4.1:

1) Battery terminal voltage is required to be a minimum of ________________.
   AND
2) Maintaining required terminal voltage ensures the Batteries are capable of carrying expected shutdown loads following a plant trip and loss of all AC power for a period of ____________.

A. 1) ≥ 135 VDC
   2) 4 hours

B. 1) ≥ 135 VDC
   2) 8 hours

C. 1) ≥ 129 VDC
   2) 4 hours

D. 1) ≥ 129 VDC
   2) 8 hours

Answer: C

Explanation:

A. INCORRECT. The first part is plausible since terminal voltage will be approximately 135 VDC when the Battery is initially taken off the Equalizing Charge according to O-6.11, Surveillance 3.8.4.1. Incorrect since this Surveillance directs the Operator to wait five minutes and then verify terminal voltage is ≥ 129 VDC. The second part is correct.

B. INCORRECT. The first part is plausible since terminal voltage will be approximately 135 VDC when the Battery is initially taken off the Equalizing Charge according to O-6.11, Surveillance 3.8.4.1. Incorrect since this Surveillance directs the Operator to wait five minutes and then verify terminal voltage is ≥ 129 VDC. The second part is plausible since this would be the coping time for the Main Batteries once DC loads are shed in accordance with ATT-8.0. Incorrect since the Station Batteries are designed for 4 hours without coping strategies.

C. CORRECT. O-6.11, Attachment 7, Surveillance 3.8.4.1 NOTE states "SR 3.8.4.1 **SHALL** be performed with the battery on "float" charge. **IF** an "equalize charge is in progress the applicable charger must be adjusted to "float" voltage prior to recording voltage." Additionally, Step 2.2 directs the Operator to "**TURN** the Equalize Timer to zero hours. Battery terminal voltage should lower to approximately 135 volts." Step 2.3 states "**WAIT** approximately five minutes **THEN VERIFY** the battery terminal voltage is greater than or equal to 129 volts". According to UFSAR, Section 8.3.2.2 "Each of the two station batteries is capable of carrying its expected shutdown loads following a plant trip and a loss of all ac power for a period of 4 hours without battery terminal voltage falling below 108.6 V."

D. INCORRECT. The first part is correct. The second part is plausible since this would be the coping time for the Main Batteries once DC loads are shed in accordance with ATT-8.0. Incorrect since the Station Batteries are designed for 4 hours without coping strategies.

Technical Reference(s):	
	O-6.11, Surveillance Requirement/Routine Operations Check Sheet (p38-40; Rev 210)
	Technical Specification LCO 3.8.4, DC Sources – MODES 1, 2, 3, and 4 (p 3.8.4-2; Amendment 130)
	UFSAR Section 8.3.2.2 (Rev 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0901C 1.13c

Question Source:		
	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063 K3.02	
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: Components using DC control power

RO Question #49

Which ONE of the following correctly describes how the 480 VAC Distribution System will be affected by the loss of a DC Distribution Train?

NOTE: MCC – Motor Control Center

Breakers on the associated 480 Volt Buses WILL/WILL NOT have the ability to be operated from the MCB?

Associated MCC loads WILL/WILL NOT have the ability to be operated from the MCB?

	480 Volt Buses	MCC Loads
A.	WILL	WILL
B.	WILL	WILL NOT
C.	WILL NOT	WILL
D.	WILL NOT	WILL NOT

Answer: B

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since the 480 VAC Buses have a throwover relay that will automatically transfer control power to the unaffected DC Train. The Applicant may misinterpret that this is also true for the MCCs. Incorrect since the MCCs do not have a throwover relay for control power.

B. CORRECT. In accordance with UFSAR Section 8.3.2, Direct Current Power Systems, "Automatic transfer of 125-V dc load groups from train A to train B (or vice versa) occurs in fifteen locations. Six throwover relays provide control power for the six 480-V buses, …" The throwover relays automatically transfer load to the redundant train on loss of power from the normal source." In accordance with P-10, Instrument Failure Reference Manual, Section 5.13, DC Bus Failure, "Expected plant response for loss vital DC Bus A/B: Busses and MCCs – Loss of Control Power for all MCCs associated with the DC Bus".

C. INCORRECT. The first part is plausible since the Applicant may misinterpret that the control power circuitry for 480 VAC Bus control power does not automatically transfer. Incorrect since 480 VAC Buses have a throwover relay to automatically transfer control power to the opposite train. The second part is plausible since the 480 VAC Buses have a throwover relay that will automatically transfer control power to the unaffected DC Train. The Applicant may misinterpret that this is also true for the MCCs. Incorrect since the MCCs do not have a throwover relay for control power.

D. INCORRECT. The first part is plausible since the Applicant may misinterpret that the control power circuitry for 480 VAC Bus control power does not automatically transfer. Incorrect since 480 VAC Buses have a throwover relay to automatically transfer control power to the opposite train. The second part is correct.

Technical Reference(s): UFSAR Section 8.3.2 (Rev 28)

P-10, Instrument Failure Reference Manual (p70-73; Rev 022)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0901C 1.06a

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2019 Ginna ILT

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 .7

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064 K1.02	
	Importance Rating	3.1	

Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems: D/G cooling water system

RO Question #50

Which of the following Diesel Generator components are cooled by the Jacket Water subsystem?

1. Crankcase Exhauster
2. Turbocharger
3. Lubricating Oil
4. Engine Block (cylinder liners)

A. 1 and 2

B. 1 and 3

C. 2 and 4

D. 3 and 4

Answer: **C**

Explanation:

A. INCORRECT. Plausible since the EDG Turbocharger is cooled by the Jacket Water System and since the Crankcase Exhauster is started by the Jacket Water pressure switch and the Applicant may misinterpret that the Crankcase Exhauster also has cooling provided by the Jacket Water. Incorrect since the Crankcase Exhauster is air cooled.

B. INCORRECT. Plausible since the EDG Lubricating Oil Coolers are provided external cooling and since the Crankcase Exhauster is started by the Jacket Water pressure switch and the Applicant may misinterpret that the EDG Lubricating Oil Coolers and Crankcase Exhauster also have cooling provided by the Jacket Water. Incorrect since the Crankcase Exhauster is air cooled and the EDG Lubricating Oil Cooler is supplied by the SW System.

C. CORRECT. 33013-1239 shows that the EDG Jacket Water System provides cooling to the

EDG crankcase (cylinder liners, the EDG Turbocharger, and the EDG Turbocharger After Cooler.

D. INCORRECT. Plausible since the Engine Crankcase (cylinder liners) are cooled by the Jacket Water System and since the EDG Lubricating Oil Coolers are provided external cooling and the Applicant may misinterpret that these are cooled by the Jacket Water System. Incorrect since the EDG Lubricating Oil Cooler is supplied by the SW System.

Technical Reference(s): 33013-1239, Sheet 1, Diesel Generator – A P&ID (Rev 30)
R0801C, EDG Presentation (slide 74; Rev 35)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0801C 1.02j, 1.03c, 1.07j

Question Source: Bank # X
Modified Bank #
New

Question History: Last NRC Exam 2015 Fort Calhoun ILT

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

10 CFR Part 55 Content: 55.41 .8
55.43

Comments:

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

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

RO Question #51

Given the following initial plant conditions:

- Plant is operating at 100% power
- RMS channel R-17, Component Cooling Water, drawer indicates 2.1E03 cpm and stable

Subsequently:

- R-17 rises rapidly to >1E06 cpm, and now indicates "EEEEE"
- R-17 drawer WARN and HIGH lights are LIT
- 40 gpm letdown orifice valve AOV-200B is in service
- PCV-135, Letdown Pressure Control Valve, is 35% open and stable
- Both RCP labyrinth seal D/Ps are 40" and stable

Which ONE of the following:

(1) indicates the reason for these indications;
AND
(2) describes the procedure that would be entered to respond to the event?

A. (1) Detector failure
(2) STP-O-17.2, Process Radiation Monitors R-11 thru R-18 Source Check, Alarm Setpoint Verification, and Functional Test

B. (1) Detector failure
(2) AR-E-16, RMS PROCESS MONITOR HIGH ACTIVITY

C. (1) RCS in-leakage to CCW system
(2) AR-E-16, RMS PROCESS MONITOR HIGH ACTIVITY

D. (1) RCS in-leakage to CCW system
   (2) AP-CCW.1, Leakage into the Component Cooling Loop

Answer: **B**

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since the Operators would utilize STP-O-17.2 to perform a functionality test to determine failure and the Applicant may misinterpret that this contains actions that would help mitigate the event. Incorrect since the STP-O-17.2 does NOT contain any actions that would assist in mitigating the event.

B. CORRECT. The given indications indicate a detector failure <u>high</u> which over-ranged the circuit and activated the WARN and HIGH range alarm circuits in the drawer. The HIGH alarm will close RCV-017, the CCW Vent Valve (but that information is not provided) and provide an input into the E-16 Annunciator. The E-16 Alarm Response procedure will provide further guidance (e.g., verify that automatic actions have occurred).

C. INCORRECT. The first part is plausible since Warning or High alarm on R-17 is the primary means of detecting in-leakage into the CCW system and R-17 would also indicate "EEEEE" if the detector is over-ranged due to high activity. Incorrect since the plant parameters provided in the initial conditions indicate that neither the NRHX nor Thermal Barrier HX is leaking. The second part is correct.

D. INCORRECT. The first part is plausible since Warning or High alarm on R-17 is the primary means of detecting in-leakage into the CCW system and R-17 would also indicate "EEEEE" if the detector is over-ranged due to high activity. Incorrect since the plant parameters provided in the initial conditions indicate that neither the NRHX nor Thermal Barrier HX is leaking. The second part is plausible since it's the procedure which E-16 will direct transition to, but given the lack of supporting plant information to confirm a leak into the CCW system, is not the correct procedure to be entered initially.

Technical Reference(s):	AR-E-16, RMS Process Monitor High Activity (Rev 01102)
	STP-O-17.2, Process Radiation Monitors R-11 thru R-18, R-20 thru R-22 and Iodine Monitors R-10A and R-10B Source Check, Alarm Setpoint Verification, and Functional Test (p75-78; Rev 011)
	AP-CCW.1, Leakage into the Component Cooling Loop (p2; Rev 01901)
	VTD-V0115-4045, Instruction Manual Universal Digital ratemeter Models 942A-100 and 942A-200 (p 3-3; Rev 6/89)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3901C 1.07d

Question Source: Bank # X
Modified Bank #
New

Question History: Last NRC Exam 2012 Ginna ILT

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

10 CFR Part 55 Content: 55.41 .11
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073 K4.01	
	Importance Rating	4.0	

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

RO Question #52

Which ONE of the following Radiation Monitors will automatically terminate a release in progress?

A. R-10B, Plant Vent Iodine

B. R-16, CNMT Fan Cooling

C. R-19, Steam Generator Blowdown

D. R-20B, Spent Fuel Pool Heat Exchanger Service Water

Answer: **C**

Explanation:

A. INCORRECT. Plausible since R-13 and R-14 reaching their associated alarm setpoint will actuate a plant ventilation isolation terminating a release in progress and the Applicant may misinterpret the function of R-10B to be the same as R-13 and R-14. Incorrect since R-10B will NOT actuate a plant ventilation isolation.

B. INCORRECT. Plausible since AR-RMS-16 Alarm Response (Containment Fan Cooling) would direct the Operator to sample and isolate the source of the release. Incorrect since R-16 does NOT have any automatic actions associated with it.

C. CORRECT. According to AR-RMS-19, an alarm on R-19 will result in the following automatic action “Steam Generator Blowdown and Sample Valves CLOSE”.

D. INCORRECT. Plausible since R-20B is a process radiation monitor and the Applicant may misinterpret that this radiation monitor has automatic actions associated with it similar to most of the process radiation monitors. Incorrect since R-20B does NOT have any automatic actions associated with it.

Technical Reference(s):
- AR-RMS-10B, R10B VENT IODINE (Rev 00701)
- AR-RMS-16, CONTAINMENT FAN COOLING (Rev 00600)
- AR-RMS-19, STEAM GENERATOR BLOWDOWN (Rev 00900)
- AR-RMS-20B, R-20B SPENT FUEL POOL HX-B SERV WTR (Rev 00601)
- P-9, Radiation Monitoring System (p11, 16, 18-19; Rev 107)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3901C 1.04a, 1.07f

Question Source:
Bank # X
Modified Bank #
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content:
55.41 .11
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 K3.07	
	Importance Rating	3.7	

Knowledge of the effect that a loss or malfunction of the SWS will have on the following: ESF loads

RO Question #53

In accordance with Technical Specification LCO 3.7.8, SW System:

1) the MAXIMUM screenhouse water temperature that SW Pumps remain OPERABLE is ____________.

AND

2) high SW temperature jeopardizes the ability ________________________.

A. 1) 82°F
   2) to achieve Cold Shutdown during a normal plant cooldown.

B. 1) 82°F
   2) of ESF systems to reduce CNMT pressure during accident conditions.

C. 1) 85°F
   2) to achieve Cold Shutdown during a normal plant cooldown.

D. 1) 85°F
   2) of ESF systems to reduce CNMT pressure during accident conditions.

Answer: D

Explanation:

A. INCORRECT. The first part is plausible since this is the limit established in O-6.13 that requires increased monitoring of Screenhouse water temperature, and the Applicant may misinterpret this to be the Technical Specification limit. Incorrect since Technical Specification LCO 3.7.8 limit on Screenhouse water temperature is 85°F. The second part is plausible since high SW temperature would result in a longer time required to cooldown the plant. Incorrect since the ability to cooldown the plant to Cold Shutdown is more a function of AFW supply and RHR cooling, both of which would still be adequate to provide for RCS cooling.

B. INCORRECT. The first part is plausible since this is the limit established in O-6.13 that requires increased monitoring of Screenhouse water temperature, and the Applicant may misinterpret this to be the Technical Specification limit. Incorrect since Technical Specification LCO 3.7.8 limit on Screenhouse water temperature is 85°F. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since high SW temperature would result in a longer time required to cooldown the plant. Incorrect since the ability to cooldown the plant to Cold Shutdown is more a function of AFW supply and RHR cooling, both of which would still be adequate to provide for RCS cooling.

D. CORRECT. According to Technical Specification LCO 3.7.8 Basis "A SW pump is considered OPERABLE when it is capable of taking suction from the screenhouse and providing cooling water to the loop header as assumed in the accident analyses. This includes consideration of available net positive suction head (NPSH) to the SW pumps and the temperature of the suction source. The following are the minimum requirements of the screenhouse bay with respect to OPERABILITY of the SW system: Level ≥ 14 feet; and Temperature ≤ 85°F." Additionally, "If the cooling water supply to the containment recirculation fan coolers and CCW heat exchangers is too warm, the accident analyses with respect to containment pressure response following a SLB and the containment sump fluid temperature following a LOCA may no longer be bounding. The bounding maximum SW water temperature assumed for long-term containment response and SLB analysis is 85°F."

Technical Reference(s): Technical Specification LCO 3.7.8 Basis (p 3.7.8-3, 3.7.8-4 & 3.7.8-7; Rev 68 & 77)

O-6.13, Daily Surveillance Log (p22 & 39; Rev 202)

Proposed references to be provided to applicants during examination: None

Learning Objective: R5101C 1.04h, 1.12b

Question Source: Bank # ____
Modified Bank # X
New ____

Question History: Last NRC Exam 2013 VC Summer ILT

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

10 CFR Part 55 Content:	55.41	.4
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078 K1.05	
	Importance Rating	3.4*	

Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: MSIV air

RO Question #54

Given the following plant conditions:

- Plant is operating at 100% power

Which ONE of the following correctly describes the effect on the Main Steam Isolation Valve(s) (MSIV(s)) for a:

1) complete loss of Instrument Air;
   OR
2) loss of DC Train?

A. 1) both MSIVs will CLOSE
   2) associated MSIV remains OPEN

B. 1) both MSIVs will CLOSE
   2) associated MSIV will CLOSE

C. 1) both MSIVs remain OPEN
   2) associated MSIV remains OPEN

D. 1) both MSIVs remain OPEN
   2) associated MSIV will CLOSE

Answer: **A**

Explanation:

A. CORRECT. According to UFSAR, Section 10.3.2.6 “The open position of the disk is full horizontal, held open against the flow of steam by an air cylinder.” According to 33013-1231, the MSIVs are FAIL CLOSE valves on loss of Instrument Air. The Instrument Air solenoid valves are pilot assisted, latching (dual coil) solenoid valves that are energized to either OPEN or CLOSE the associated MSIV. In accordance with P-10, Section 5.13 a loss of DC Train the associated MSIV fails As-Is and will **NOT** close on an automatic isolation signal.

B. INCORRECT. The first part is correct. The second part is plausible since the Applicant may have the misconception that the solenoid operated air valve is normally energized to maintain air pressure to the associated MSIV. Incorrect since the solenoid operated air valves are only energized to reposition the MSIV and loss of DC power results in the associated MSIV failing As-Is (OPEN for the given plant conditions).

C. INCORRECT. The first part is plausible since the Applicant may have a misconception of the operation of the MSIV actuator and believe that Instrument Air provides the motive force to CLOSE the MSIV. Incorrect since Instrument Air maintains the MSIV OPEN against steam flow. The second part is correct.

D. INCORRECT. The first part is plausible since the Applicant may have a misconception of the operation of the MSIV actuator and believe that Instrument Air provides the motive force to CLOSE the MSIV. Incorrect since Instrument Air maintains the MSIV OPEN against steam flow. The second part is plausible since the Applicant may have the misconception that the solenoid operated air valve is normally energized to maintain air pressure to the associated MSIV. Incorrect since the solenoid operated air valves are only energized to reposition the MSIV and loss of DC power results in the associated MSIV failing As-Is (OPEN for the given plant conditions).

Technical Reference(s):	UFSAR, Section 10.3.2.6 (Rev 28)
	33013-1231, Main Steam (MS) (Safety Related) P&ID (Rev 4)
	P-10, Instrument failure Reference Manual (p70-73; Rev 022)
	10905-0732, AOV-3517 Main Steam Isolation VLV A Elementary Wiring Diagram (Rev 2)
	Operation and Maintenance Manual for Pilot Assisted, Latching, 2 Way Valve (p4-5; Rev G)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4001C 1.03a, 1.10e

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103 A2.04	
	Importance Rating	3.5*	

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

RO Question #55

Given the following initial plant conditions:

- Plant is in MODE 6
- Core reload was started last shift and 10 fuel assemblies were loaded
- Source Range Nuclear Instrument baselines have been established and are:
  - N-31 is 35 cps
  - N-32 is 40 cps

Subsequently, following the fifteenth (15) fuel assembly being loaded in the core:

- Annunciator E-29, SOURCE RANGE HI FLUX AT SHUTDOWN, alarms
- Source Range Nuclear Instrument readings:
  - N-31 indicates 105 cps
  - N-32 indicates 115 cps

Which ONE of the following identifies:

(1) whether Annunciator E-29 is valid for the given plant conditions;
AND
(2) the action(s) required to be taken.

A. (1) NOT VALID
(2) Notify the Refueling SRO to closely monitor conditions while continuing with core reload.

B. (1) NOT VALID
(2) Make an announcement that the alarm was NOT valid and all personnel may remain in CNMT.

C. (1) VALID
   (2) Direct Refueling SRO to immediately stop refueling activities and evacuate all personnel from CNMT.

D. (1) VALID
   (2) Direct Refueling SRO to remove the last fuel assembly from the core and return it to the Spent Fuel Pool prior to evacuating.

Answer: C

Explanation:

A. INCORRECT. The first part is plausible if the Applicant does not know the setpoint for the alarm. Incorrect since E-29 is set to alarm when baseline SR counts double. The second part is plausible since the Applicant may interpret that this critical path work would continue with closer monitoring in place. Incorrect since the alarm is valid and core reload must be stopped.

B. INCORRECT. The first part is plausible if the Applicant does not know the setpoint for the alarm. Incorrect since E-29 is set to alarm when baseline SR counts double. The second part is plausible since the Applicant may interpret that this is an acceptable action due to the invalid alarm. Incorrect since the alarm is valid and CNMT must be evacuated.

C. CORRECT. In accordance with AR-E-29 the Alarm Setpoint is “Rise in shutdown count rate by a factor of 2.” According to AR-E-29, Step 4.3 “**IF** count rate is actually rising, **THEN PERFORM** the following: **EVACUATE** containment and **STOP** any evolutions which could be causing positive reactivity addition.”

D. INCORRECT. The first part is correct. The second part is plausible since the Applicant may interpret this as a necessary action to remove the positive reactivity just added to the core. Incorrect since AR-E-29 requires the CNMT be evacuated and core reload stopped.

Technical Reference(s): AR-E-29, SOURCE RANGE HI FLUX AT SHUTDOWN (Rev 008)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3301C 1.11a

Question Source:
- Bank #: X
- Modified Bank #:
- New:

Question History: Last NRC Exam 2017 Point Beach ILT

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .5

55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	001 A4.11	
	Importance Rating	3.5	

Ability to manually operate and/or monitor in the control room: Determination of SDM (Control Rod Drive System)

RO Question #56

Given the following:

- Plant is operating at 100% power when 'B' Heater Drain Tank Pump trips
- The crew performs a rapid load reduction and reactor power is stable at 68%
- The following Annunciators are currently LIT:
  - C-31, INSERTION LIMIT BANK D LO
  - C-32, INSERTION LIMIT BANK D LO-LO

(1) Which ONE of the following is correct concerning Shutdown Margin (SDM)?

AND

(2) What action should the HCO take in accordance with the appropriate plant procedure(s)?

A. (1) adequate SDM is maintained
(2) perform normal boration

B. (1) adequate SDM is NOT maintained
(2) perform normal boration

C. (1) adequate SDM is maintained
(2) perform emergency boration

D. (1) adequate SDM is NOT maintained
(2) perform emergency boration

Answer: B

Explanation:

A. INCORRECT. The first part is plausible since the Applicant may misinterpret that

Annunciator C-32, like C-31, will alarm prior to the actual Rod Insertion Limit being violated and therefore, that SDM is maintained. Incorrect since Annunciator C-32 alarms at the Rod Insertion Limit and Control Banks below the associated RIL LO-LO limit does NOT ensure adequate SDM. The second part is correct.

B. CORRECT. According to P-1, Section 6.1.1.6 "The Control Banks must be maintained above their respective insertion limits (LO-LO Alarm) to ensure the following: Adequate shutdown margin in the event of a Reactor trip; Maximum possible ejected Rod reactivity limits are maintained; Acceptable core power distribution/peaking factors." Also, "Boration of the RCS should be initiated immediately if the insertion limit LO-LO alarm is actuated, using S-3.1, Boron Concentration Control, **AND** appropriate Alarm Response procedure." AR-C-32, Step 2 provides the direction for the Operator to perform a normal boration.

C. INCORRECT. The first part is plausible since the Applicant may misinterpret that Annunciator C-32, like C-31, will alarm prior to the actual Rod Insertion Limit being violated and therefore, that SDM is maintained. Incorrect since Annunciator C-32 alarms at the Rod Insertion Limit and Control Banks below the associated RIL LO-LO limit does NOT ensure adequate SDM. The second part is plausible since the Applicant may misinterpret the procedural direction and conclude that the quickest means available must be used to clear the alarm. Incorrect since AR-C-32 directs the Operator to perform a normal boration method.

D. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret the procedural direction and conclude that the quickest means available must be used to clear the alarm. Incorrect since AR-C-32 directs the Operator to perform a normal boration method.

Technical Reference(s): AR-C-31, INSERTION LIMIT BANK D LO (Rev 7)
AR-C-32, INSERTION LIMIT BANK D LO-LO (Rev 7)
P-1, Reactor Control and Protection System (p7; Rev 078)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2901C 1.07, 1.11a

Question Source:
Bank # ______
Modified Bank # X
New ______

Question History: Last NRC Exam 2015 Comanche Peak ILT

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

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	011 K2.01	
	Importance Rating	3.1	

Knowledge of bus power supplies to the following: Charging pumps

RO Question #57

Given the following plant conditions:

- Plant is operating at 100% power
- ‘A’ and ‘C’ Charging Pumps are running
- Bus 14 trips on overcurrent
- The Operating crew enters AP-ELEC.14/16, Loss of Safeguards Bus 14/16
- Instrument Bus ‘B’ has been transferred to the Maintenance CVT

Which ONE of the following correctly completes the statements below?

1) The Charging Pump that remains running is ____________.
AND
2) The HCO will restore Letdown in accordance with AP-ELEC.14/16, by establishing __________________.

A. 1) ‘A’ Charging Pump
2) Excess Letdown

B. 1) ‘A’ Charging Pump
2) Normal Letdown

C. 1) ‘C’ Charging Pump
2) Excess Letdown

D. 1) ‘C’ Charging Pump
2) Normal Letdown

Answer: **D**

Explanation:

A. INCORRECT. The first part is plausible the Applicant may understand that two Charging Pumps are powered from the Bus 16, but incorrectly assumes they are ‘A’ and ‘B’, vice ‘B’ and ‘C’ Charging Pumps. The second part is plausible since the Applicant may misinterpret the direction in AP-ELEC.14/16 and incorrectly conclude that with only a single Charging Pump running that Excess Letdown must be placed in service. Incorrect since AP-ELEC.14/16 will direct the Operator to start a second Charging Pump.

B. INCORRECT. The first part is plausible the Applicant may understand that two Charging Pumps are powered from the Bus 16, but incorrectly assumes they are ‘A’ and ‘B’, vice ‘B’ and ‘C’ Charging Pumps. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret the direction in AP-ELEC.14/16 and incorrectly conclude that with only a single Charging Pump running that Excess Letdown must be placed in service. Incorrect since AP-ELEC.14/16 will direct the Operator to start a second Charging Pump.

D. CORRECT. In accordance with P-12, 480V Bus 14 is the power supply for Charging Pump 1A and 480V Bus 16 is the power supply for Charging Pumps 1B and 1C. Since Bus 14 trips, only ‘C’ Charging Pump remains running. In accordance with AP-ELEC.14/16, Step 12, the Operator is directed to “Establish Normal Letdown (Refer to ATT-9.0, ATTACHMENT LETDOWN)”.

Technical Reference(s): P-12, Electrical Systems Precautions, Limitations, and Setpoints (p35 & 37; Rev 029)

AP-ELEC.14/16, Loss of Safeguards Bus 14/16 (p12; Rev 01203)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1601C 1.05a; RAP66C 2.01

Question Source: Bank # ______

Modified Bank # ______

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	017 K4.02	
	Importance Rating	3.1	

Knowledge of ITM system design feature(s) and/or interlock(s) which provide for the following: Sensing and determination of location core hot spots

RO Question #58

When evaluating F-0.2, Core Cooling CSFST, a minimum of ___________ CETs reading greater than 1200°F indicates Inadequate Core Cooling.

A. Two (2)

B. Four (4)

C. Five (5)

D. Eight (8)

Answer: C

Explanation:

A. INCORRECT. Plausible since the Applicant may misinterpret the requirement to be one channel for a total of two CETs. Incorrect since F-0.2 requires at least 5 CETs reading greater than 1200°F.

B. INCORRECT. Plausible since the Applicant may misinterpret the required number of CETs for F-0.2 determination as the same for Technical Specifications operability for each quadrant (four per quadrant). Incorrect since F-0.2 requires at least 5 CETs reading greater than 1200°F.

C. CORRECT. According to F-0.2 Background Document, “at least 5 thermocouples should be reading greater than 1200°F. Five has been chosen to allow for thermocouples failing high.”

D. INCORRECT. Plausible since the Applicant may misapply the NOTE in Technical Specifications LCO 3.3.3, Table 3.3.3-1, and misinterpret the required number of CETs per channel to be a single CET and therefore, the total required number of CETs for the entire core to be eight (one per channel/2 channels per quadrant). Incorrect since F-0.2 requires at least 5 CETs reading greater than 1200°F.

Technical Reference(s): F-0.2, Core Cooling CSFST, Background Document (p4; Rev 002)

Technical Specification LCO 3.3.3 LCO Basis (p3.3.3-8 thru 3.3.3-11; Rev 73)

Proposed references to be provided to applicants during examination: None

Learning Objective: RRC1C 1.02

Question Source: Bank # ____
Modified Bank # ____
New X

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 .7
55.43 ____

Comments: Ginna's Facility Representative has determined that the question is testing RO knowledge since it is assessing conditions required for Core Cooling Red Path entry, which is RO knowledge.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	028 K6.01	
	Importance Rating	2.6	

Knowledge of the effect of a loss or malfunction of the following will have on the HRPS: Hydrogen recombiners

RO Question #59

Given the following plant conditions:

- A Design Basis LOCA occurred
- Both Hydrogen Recombiners are unavailable
- CNMT hydrogen concentration is 4% and rising

Which ONE of the following correctly completes the statements below?

1) In accordance with the UFSAR, if the hydrogen gas in CNMT detonates, ________________________;

   AND

2) The preferred method for venting CNMT is via the ____________________ to mitigate this concern.

A. 1) over-pressurization of CNMT can occur;
   2) Mini-Purge System

B. 1) over-pressurization of CNMT can occur;
   2) Equipment Air Lock

C. 1) equipment in CNMT can be damaged due to excessive heat
   2) Mini-Purge System

D. 1) equipment in CNMT can be damaged due to excessive heat
   2) Equipment Air Lock

Answer: **A**

Explanation:

A. CORRECT. According to UFSAR, Section 6.2.5.2.2 “Venting must be accomplished before 6% by volume hydrogen is accumulated. A higher concentration, if accidentally ignited, could result in dynamic overpressures capable of damaging the containment.” Also, “It is concluded that proper protection of the health and safety of the public is served by providing the recombiner system, thus avoiding the necessity of venting at any specific time. However, the alternative means exists of avoiding a serious hazard by controlled venting if for any reason the recombiners were not operable.” According to SAG-7, Attachment A, Venting through Mini-Purge System would be the first choice to vent the CNMT.

B. INCORRECT. The first part is correct. The second part is plausible since using the Personnel Air Lock is an alternative method listed in SAG-7, Attachment A for venting the CNMT and the Applicant may misinterpret that using the Equipment Air Lock to vent CNMT is the preferred option in SAG-7. Incorrect since using the Equipment Air Lock could potentially create an uncontrolled release to the environment from CNMT.

C. INCORRECT. The first part is plausible since the Applicant may misinterpret that ignition of the hydrogen in CNMT would potentially result in equipment damage from the excessive heat, and this being the primary concern. Incorrect since the primary concern is potentially over-pressurizing the CNMT vessel. The second part is correct.

D. INCORRECT. The first part is plausible since the Applicant may misinterpret that ignition of the hydrogen in CNMT would potentially result in equipment damage from the excessive heat, and this being the primary concern. Incorrect since the primary concern is potentially over-pressurizing the CNMT vessel. The second part is plausible since using the Personnel Air Lock is an alternative method listed in SAG-7, Attachment A for venting the CNMT and the Applicant may misinterpret that using the Equipment Air Lock to vent CNMT is the preferred option. Incorrect since using the Equipment Air Lock could potentially create an uncontrolled release to the environment from CNMT.

Technical Reference(s): UFSAR, Section 6.2.5.2.2 (Rev 28)
SAG-7, Reduce Containment Hydrogen (p38-42; Rev 003)

Proposed references to be provided to applicants during examination: None

Learning Objective: R6501C 1.06; R2201C 1.01d

Question Source: Bank # X
Modified Bank #
New

Question History: Last NRC Exam 2007 Diablo Canyon ILT Retake

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

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments: Ginna's Facility Representative has determined that the question is testing RO knowledge even though the Technical Reference for the second part of the question is SAG-7, the second part of the question is testing system knowledge and control of unmonitored potentially contaminated release to the environment. The Mini-Purge System is a monitored release path whereas venting through the Equipment Air Lock would NOT be a monitored release path to the environment.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	035 G2.2.44	
	Importance Rating	4.2	

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. (Steam Generator System)

RO Question #60

Given the following plant conditions:

- The unit automatically tripped from rated power
- The Operating crew has entered E-0, Reactor Trip or Safety Injection
- RCS temperature: 525°F and lowering
- RCS pressure: 1700 psig and lowering
- Pressurizer Level: 2% and lowering
- Containment pressure: 0.1 psig and stable
- A S/G Pressure: 600 psig and lowering quickly
- B S/G Pressure: 450 psig and lowering quickly
- A S/G Water Level: 275 inches (WR) and lowering
- B S/G Water Level: 260 inches (WR) and lowering

In accordance with A-503.1, Emergency and Abnormal Operating Procedures Users Guide:

Which ONE of the following actions is the Operating crew allowed to perform upon completion of E-0 Immediate Actions based on the given plant conditions?

A. Secure AFW flow to ‘B’ S/G, ONLY

B. Secure AFW flow to both S/Gs

C. Reduce AFW flow to each S/G to 50 gpm

D. Reduce AFW flow to each S/G to 100 gpm

Answer: **C**

Explanation:

A. INCORRECT. Plausible since the Applicant may correctly determine that 'B' S/G is faulted and misinterpret that the 'A' S/G conditions are due to RCS cooldown. A-503.1 contains guidance for securing AFW flow to a single faulted S/G. Incorrect since 'A' S/G water level would be expected to be rising even with the RCS cooldown and 'A' S/G pressure is less than the saturation pressure for the given RCS temperature.

B. INCORRECT. Plausible since the Applicant may correctly determine that both S/Gs are faulted and misinterpret the A-503.1 guidance for securing AFW flow to faulted S/Gs. Incorrect since for the given plant conditions A-503.1 directs the Operator to reduce AFW flow to each S/G to 50 gpm.

C. CORRECT. The given plant conditions are indicative of both S/Gs being faulted: lowering pressure and level with the MDAFW Pumps and TDAFW Pump supplying flow. A-503.1, Section 5.3.B.1.d(5) states "Throttling AFW to both S/Gs to 50 gpm each S/G when both S/Gs are faulted to mitigate an uncontrolled cooldown of the RCS which is resulting from faulted S/Gs."

D. INCORRECT. Plausible since the Applicant may misinterpret that with both S/G water levels less than 7% that a minimum of 200 gpm total AFW flow is required to preclude a Red Path entry on the Heat Sink CSFST. Incorrect since the given plant conditions indicate that both S/Gs are faulted and A-503.1 directs the Operator to reduce AFW flow to each S/G to 50 gpm.

Technical Reference(s): A-503.1, Emergency and Abnormal Operating Procedures Users Guide (p42-43; Rev 053)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP50C 1.08

Question Source: Bank # _____

Modified Bank # _____

New X

Question History: Last NRC Exam _____

Question Cognitive Level: Memory or Fundamental Knowledge _____

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .10
55.43

Comments: K/A is matched since the Applicant must interpret the given plant conditions, determine the actions necessary in accordance with A-503.1, and understand how those actions affect Steam Generator conditions and the plant overall.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	041 A1.02	
	Importance Rating	3.1	

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam pressure

RO Question #61

Given the following plant conditions:

- Reactor power is 18% and stable
- The Operating crew is preparing to synchronize the Main Generator to the grid
- Steam Dump Mode Selector switch is in MANUAL
- HCV-484, S/G COND STM DUMP, is in AUTO and the controller output fails to 100%

Which ONE of the following describes the effect of the controller failure on S/G pressure with **NO Operator actions**?

S/G pressure will…

A. lower continually until S/Gs are completely depressurized

B. lower and then rise toward Atmospheric Relief Valve (ARV) setpoint

C. rise and cycle at the S/G Safety Valve setpoint

D. rise and cycle at the Atmospheric Relief Valve (ARV) setpoint

Answer: B

Explanation:

A. INCORRECT. Plausible since the Steam Dumps will OPEN resulting in lowering S/G pressures. Incorrect since the reactor will trip once S/G pressure reaches 514 psig (or earlier dependent upon PRZR pressure), causing a Main Steam Isolation signal on low T_{AVG} (545°F) and high steam flow. Once the MSIVs close, S/G pressure will begin to rise.

B. CORRECT. At this point in the plant startup, Steam Dump are in pressure control MODE (MAN-AUTO) with HCV-484 set to control S/G pressure. HCV-484 controller output failing to 100% will result in the Steam Dump valves failing OPEN causing S/G pressures to lower. S/G pressure will continue to lower until the reactor trips on either low PRZR pressure or low S/G pressure SI. Once the SI signal occurs, a Main Steam Isolation signal on low T_{AVG} (545°F) and high steam flow will occur causing the MSIVs to CLOSE. Once the MSIVs close, S/G pressure will begin to rise.

C. INCORRECT. Plausible since this would be the effect if HCV-484 controller was reverse acting (similar to some MCB controllers) and the ARVs failed to operate and the Applicant may misinterpret the operation of HCV-484 controller and conclude that the output failing to 100% will result in the Steam Dump valves closing. Incorrect since HCV-484 output failing to 100% will cause the Steam Dump valves to OPEN and S/G pressure to lower.

D. INCORRECT. Plausible since this would be the effect if HCV-484 controller was reverse acting (similar to some MCB controllers) and the Applicant may misinterpret the operation of HCV-484 controller and conclude that the output failing to 100% will result in the Steam Dump valves closing. Incorrect since HCV-484 output failing to 100% will cause the Steam Dump valves to OPEN and S/G pressure to lower.

Technical Reference(s): P-1, Reactor Control and Protection System (p41 & 45; Rev 078)

P-10, Instrument Failure Reference Manual (p100-102; Rev 022)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4501C 1.06, 1.09

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .5

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	045 K5.01	
	Importance Rating	2.8*	

Knowledge of the operational implications of the following concepts as they apply to the MT/G System: Possible presence of explosive mixture in generator if hydrogen purity deteriorates

RO Question #62

Given the following plant conditions:

- The plant is operating at 100% power
- Main Generator hydrogen purge is secured to perform a hydrogen consumption calculation
- Main Generator hydrogen purity is 97% and slowly lowering

Which ONE of the following correctly describes:

(1) the primary concern if Generator hydrogen purity continues to lower;

AND

(2) the first actions taken by the Operating crew in response to the current plant conditions?

A. (1) reduced cooling of the Main Generator Exciter windings
(2) purge the Main Generator with hydrogen to raise purity

B. (1) reduced cooling of the Main Generator Exciter windings
(2) initiate a rapid load reduction to take the Main Generator off-line

C. (1) potential explosive mixture in the Main Generator
(2) purge the Main Generator with hydrogen to raise purity

D. (1) potential explosive mixture in the Main Generator
(2) initiate a rapid load reduction to take the Main Generator off-line

Answer: **C**

Explanation:

A. INCORRECT. The first part is plausible since hydrogen is used to cool the Main Generator stator windings and the Applicant may misinterpret that the exciter windings are also cooled using hydrogen. Incorrect since the exciter windings are cooled via an attached fan within the exciter enclosure. The second part is correct.

B. INCORRECT. The first part is plausible since hydrogen is used to cool the Main Generator stator windings and the Applicant may misinterpret that the exciter windings are also cooled using hydrogen. Incorrect since the exciter windings are cooled via an attached fan within the exciter enclosure. The second part is plausible since a rapid load reduction would not be performed until a secondary effect from the low purity is seen in the Control Room (i.e. vibrations) and a different AR or AP was entered. Incorrect since low hydrogen purity can be corrected with the Generator on-line by purging with fresh hydrogen.

C. CORRECT. Reduced hydrogen purity (≤ 75%) can result in an explosion given an ignition source. Any arcing inside of the generator could produce the combination required to cause an explosion. According to AR-H2 PANEL-1 "IF purity is low, THEN perform the following: Check operation of Pressure Equalizing valves AND Hydrogen Drainside Regulator; Initiate purge of generator per T-31.6."

D. INCORRECT. The first part is correct. The second part is plausible since a rapid load reduction would not be performed until a secondary effect from the low purity is seen in the Control Room (i.e. vibrations) and a different AR or AP was entered. Incorrect since low hydrogen purity can be corrected with the Generator on-line by purging with fresh hydrogen.

Technical Reference(s):	AR-H_2 PANEL-1, HYDROGEN PURITY HIGH OR LOW (Rev 00301)
	UFSAR Sections 10.7.6 and 10.7.6.8 (Rev 28)
	T-31.6, Purging Turbine Generator to Increase Hydrogen Purity (p4-5; Rev 01503)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0501C 1.09

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	.5
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	056 K1.03	
	Importance Rating	2.6*	

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

RO Question #63

Given the following plant conditions:

- Plant is operating at 100% power
- Annunciator H-30, CONDENSATE BYPASS VALVE OPEN, alarms
- CO reports AOV-3959, Condensate Bypass Valve, is OPEN

Which ONE of the following describes the effect on plant operation?

A. Main Feed Pump suction flow will RISE

B. Condensate Storage Tank level will RISE

C. Main Feed Pump suction pressure will LOWER

D. Feedwater inlet temperature to S/Gs will LOWER

Answer: D

Explanation:

A. INCORRECT. Plausible since MFW Pump suction parameters will be affected when the Condensate Bypass Valve opens, and the Applicant may misinterpret that one of the effects is a change in suction flow. Incorrect since MFW Pump flow (suction and discharge) is controlled by the Main Feed regulating Valve position, which has not changed.

B. INCORRECT. Plausible since this would be correct if the Condensate Reject AOV Bypass Valve failed open and the Applicant may misinterpret the functions of these two valves. Incorrect since the opening of the Condensate Bypass Valve will no effect on CST level.

C. INCORRECT. Plausible since MFW Pump suction pressure will be affected when the Condensate Bypass Valve opens, and the Applicant may misinterpret the effect of this event. Incorrect since the Condensate Bypass Valve opens to raise MFW Pump suction pressure.

D. CORRECT. According to UFSAR, Section 15.1.1.1 "An extreme example of excess heat removal by the feedwater system is the transient associated with the accidental opening of the condensate bypass valve which diverts flow around the low-pressure feedwater heaters. In the event of an accidental opening, there is a sudden reduction in inlet feedwater temperature to the steam generators."

Technical Reference(s): UFSAR, Section 15.1.1.1 (Rev 28)

AR-H-30, CONDENSATE BYPASS VALVE OPEN (Rev 01201)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4301C 1.06b, 1.10i

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2011 Ginna ILT SRO Retake

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .14

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	068 A2.04	
	Importance Rating	3.3	

Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation

RO Question #64

Given the following plant conditions:

- Plant is in MODE 6
- Lake temperature is 31.5°F
- Recirc Gate is fully closed
- Circ Water Pump trip interlock for R-18, Liquid Waste Disposal, is bypassed
- ‘A’ Circ Water Pump is started
- IMD has adjusted R-18 setpoint
- ‘A’ Monitor Tank release is started

Subsequently:

- Operating crew enters ER-SC.3, Low Screenhouse Water Level
- The Recirc Gate is being cycled to mitigate Frazil Ice effects

Which ONE of the following correctly completes the statements below?

(1) When the Recirc Gate is opened, the R-18 alarm setpoint will be _______ conservative.
AND
(2) If ‘A’ Circ Water Pump and ALL Service Water Pumps were tripped due to loss of suction, the Primary EO will stop the Monitor Tank release by closing _______________.

A. (1) less
(2) RCV-018, LIQUID WASTE RELEASE OUTLET AOV TO DISCHARGE CANAL

B. (1) more
(2) RCV-018, LIQUID WASTE RELEASE OUTLET AOV TO DISCHARGE CANAL

C. (1) less
(2) V-1249, MONITOR TANK PUMP DISCHARGE ISOLATION VALVE TO WASTE DISCHARGE LINE

D. (1) more
(2) V-1249, MONITOR TANK PUMP DISCHARGE ISOLATION VALVE TO WASTE DISCHARGE LINE

Answer: **A**

Explanation:

A. CORRECT. In accordance with P-9, Radiation Monitoring System, Section 6.2.12.2 "The Circ water Pump interlock for AOV-18 is bypassed during refueling outages to allow effluent releases to the discharge canal using service water pumps." Additionally, "AOV-18 will not close if dilution flow is lost, and the alarm will also be non-conservative due to lower dilution flow rate from the Service Water pumps." As the Recirc Gate is opened, some discharge canal flow is recirculated back to the Screenhouse bay resulting in lower discharge canal flow to Lake Ontario. According to S-3.4K, Step 6.2.2 CAUTION "**WHEN** Circ Water Pump interlock to RCV-18 is bypassed, **THEN** RCV-18 **SHALL** be closed if all Service Water Pump dilution flow is lost."

B. INCORRECT. The first part is plausible if the Applicant misinterprets the relationship between discharge canal flow, R-18 alarm setpoint, and that the automatic interlock to secure the discharge is bypassed in MODE 6 concluding that cycling of the discharge gate will increase flow vice lower it through the canal The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret the function of the R-18 Circ Water Pump bypass and determine that RCV-18 can not be closed from the Waste Panel and that the release must be terminated by closing the manual valve in the waste discharge line. Incorrect since RCV-18 can still be closed by the EO at the Waste Panel as directed by the operating procedure.

D. INCORRECT. The first part is plausible if the Applicant misinterprets the relationship between discharge canal flow, R-18 alarm setpoint, and that the automatic interlock to secure the discharge is bypassed in MODE 6 concluding that cycling of the discharge gate will increase flow vice lower it through the canal The second part is plausible since the Applicant may misinterpret the function of the R-18 Circ Water Pump bypass and determine that RCV-18 can not be closed from the Waste Panel and that the release must be terminated by closing the manual valve in the waste discharge line. Incorrect since RCV-18 can still be closed by the EO at the Waste Panel as directed by the operating procedure.

Technical Reference(s): P-9, Radiation Monitoring System (p17; Rev 107)
S-3.4K, Releasing Monitor Tank A or B to Discharge Canal (p9 & 14; Rev 033)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3801C 4.01i; R3901C 1.07e

Question Source: Bank # X
Modified Bank #
New

Question History: Last NRC Exam 2019 Ginna ILT

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

10 CFR Part 55 Content: 55.41 .7
55.43

Comments: The K/A is matched even though the question does not directly state that an interlock failure has occurred; however, bypassing the Circ Water Pump interlock for RCV-018 is an operation that results in failure of automatic isolation capability when both Circ Water Pumps trip.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	072 K3.01	
	Importance Rating	3.2*	

Knowledge of the effect that a loss or malfunction of the ARM system will have on the following: Containment ventilation isolation

RO Question #65

Given the following plant conditions:

- Core reload is in progress
- Containment Purge System is operating
- R-29, CONTAINMENT HIGH RANGE, Radiation Monitor fails high

Which ONE of the following describes:

(1) the status of Containment Purge;

AND

(2) the immediate action(s) (if any) required by Technical Specifications?

A. (1) Containment Purge is automatically terminated
(2) CORE ALTERATIONS may continue

B. (1) Containment Purge is automatically terminated
(2) Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies within Containment

C. (1) Containment Purge remains in operation
(2) CORE ALTERATIONS may continue

D. (1) Containment Purge remains in operation
(2) Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies within Containment

Answer: C

Explanation:

A. INCORRECT. The first part is plausible since the failure of either R-11 or R-12 radiation monitors would result in a Containment Ventilation Isolation signal which automatically TRIPS the Containment Purge Supply and Exhaust Fans and CLOSES the Containment Purge Supply and Exhaust valves. Incorrect since R-29 does NOT have any automatic actions associated. The second part is correct.

B. INCORRECT. The first part is plausible since the failure of either R-11 or R-12 radiation monitors would result in a Containment Ventilation Isolation signal which automatically TRIPS the Containment Purge Supply and Exhaust Fans and CLOSES the Containment Purge Supply and Exhaust valves. Incorrect since R-29 does NOT have any automatic actions associated. The second part is plausible since the Technical Specification actions for loss of equipment required for refueling activities is to immediately suspend core alterations and the movement of irradiated fuel assemblies. Incorrect since R-29 is not required for refueling operations by Technical Specifications.

C. CORRECT. In accordance with AR-A-25, failure of R-29 does not result in a Containment Ventilation Isolation signal and S-23.2.2 does not require that radiation monitor R-29 be OPERABLE during Purge System operations. Technical Specifications LCO 3.3.5 does not require that R-29 be OPERABLE during Refueling activities.

D. INCORRECT. The first part is correct. The second part is plausible since the Technical Specification actions for loss of equipment required for refueling activities is to immediately suspend core alterations and the movement of irradiated fuel assemblies. Incorrect since R-29 is not required for refueling operations by Technical Specifications.

Technical Reference(s):	AR-A-25, CONTAINMENT VENTILATION ISOLATION (Rev 01202)
	Technical Specification LCO 3.3.5 Basis (p B3.3.5-4; Rev 42)
	S-23.2.2, Containment Purge Procedure (p10; Rev 057)
	P-9, Radiation Monitoring System (p8; Rev 107)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3901C 1.06

Question Source:	Bank #	X
	Modified Bank #	
	New	

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	.7
	55.43	

Comments: K/A is matched since R-29 is an Area Radiation Monitor and the Applicant must determine the effect of R-29 failure on CNMT Purge and Core alterations (i.e. has a CVI occurred) and also distinguish between failure of R-29 as compared to R-11 or R-12. Ginna does not have any Area Radiation Monitors that cause a CNMT Ventilation Isolation.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.1.1	
	Importance Rating	3.8	

Knowledge of conduct of operations requirements.

RO Question #66

What is the frequency of Main Control Board walkdowns in accordance with OP-AA-103-102, Watch-Standing Practices?

1) The Unit Reactor Operator is to perform a panel walk down a minimum of once ____________.

AND

2) The Unit Supervisor is to perform a panel walk down a minimum of once _______.

A. 1) per hour
2) every four (4) hours

B. 1) per hour
2) per shift

C. 1) every four (4) hours
2) every four (4) hours

D. 1) every four (4) hours
2) per shift

Answer: A

Explanation:

A. CORRECT. According to OP-AA-103-102, Section 4.4 “The Unit Reactor Operator shall **PERFORM** an hourly walk-down of the unit (including front and back panels, nuclear instrumentation drawers, radiation monitoring, computer displays, PPC alarms, PPC indications, etc.). The Assist Reactor Operator is expected to assist and share this responsibility.” Also, “The Unit Supervisor shall **PERFORM** a MCR panel/PPC walk-down as described above at least every four hours.” OPG-OPERATIONS-EXPECTATIONS, Section 4.14 contains the same requirements.

B. INCORRECT. The first part is correct. The second part is plausible since the US is required to perform a walk-down with the RO once per shift. Incorrect since the US is required to perform a panel walk down every four hours.

C. INCORRECT. The first part is plausible since the US is required to perform a panel walk down every four hours and the Applicant may misinterpret this requirement to apply to the RO as well. Incorrect since the RO is required to perform a panel walk down once per hour. The second part is correct.

D. INCORRECT. The first part is plausible since the US is required to perform a panel walk down every four hours and the Applicant may misinterpret this requirement to apply to the RO as well. Incorrect since the RO is required to perform a panel walk down once per hour. The second part is plausible since the US is required to perform a walk-down with the RO once per shift. Incorrect since the US is required to perform a panel walk down every four hours.

Technical Reference(s): OP-AA-103-102, Watch-Standing Practices (p8; Rev 19)

OPG-OPERATIONS-EXPECTATIONS, Operations Department Expectations (p11; Rev 053)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAD03C 1.03

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2016 Browns Ferry ILT

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 .10

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.1.26	
	Importance Rating	3.4	

Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen).

RO Question #67

In accordance with OP-AA-109-101, Personnel and Equipment Tagout Process, which ONE of the following includes ALL the situations listed below which would require double isolation boundary protection (two closed valves in series)?

1. System fluid temperature greater than 200°F
2. System pressure greater than 200 psig
3. System contains noxious chemicals as determined by Chemistry personnel

A. 1 only

B. 1 and 3 only

C. 2 and 3 only

D. 1, 2, and 3

Answer: B

Explanation:

A. INCORRECT. Plausible since System temperature > 200°F requires double isolation boundary protection. Incorrect since Systems containing noxious chemicals also requires double isolation boundary protection.

B. CORRECT. In accordance with OP-AA-109-101, Step 7.3.1.1 “For fluid or gas systems with temperature greater than 200 degrees Fahrenheit (93 degrees Celsius) or pressure greater than 500 psig (35 bar), or involving noxious chemicals (as determined by the material safety data sheet or chemistry personnel), the work area should be isolated using double isolation boundary protection (two closed valves in series), with a tell-tale vent or drain valve between the two closed valves opened.”

C. INCORRECT. Plausible since Systems containing noxious chemicals also requires double isolation boundary protection. Incorrect since System temperature > 200°F also requires double isolation boundary protection.

D. INCORRECT. Plausible since System temperature > 200°F and Systems containing noxious chemicals requires double isolation boundary protection. Incorrect since System pressure greater than 200 psig does NOT require double isolation boundary protection.

Technical Reference(s): OP-AA-109-101, Personnel and Equipment Tagout Process (p25; Rev 016)

Proposed references to be provided to applicants during examination: None

Learning Objective: N-CE-SA-CT-CIN 1.2

Question Source:

	Bank #	
	Modified Bank #	
	New	X

Question History: Last NRC Exam

Question Cognitive Level:

	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:

	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.1.29	
	Importance Rating	4.1	

Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.

RO Question #68

Given the following plant conditions:

- You are conducting a valve lineup
- You are verifying the position of a normally LOCKED OPEN valve
- The locking device is preventing any movement of the handwheel

In accordance with A-52.2, Control of Locked Valve and Breaker Operation, which ONE of the following describes the method used to verify the OPEN position of this locked valve?

A. Verify position by local stem indicator, local light indication if available, or limit switch position if available. Positioning of the valve is NOT required

B. Verify position from the last completed alignment documentation (tagout clearance, system alignment procedure, etc.). Positioning of the valve is NOT required.

C. Remove the locking device, attempt to move the valve in the OPEN direction, return valve to original position, and reinstall locking device. Second person verifies the locking device installed.

D. Remove the locking device, attempt to move the valve in the CLOSED direction, return valve to original position, and reinstall locking device. Second person verifies the locking device installed.

Answer: D

Explanation:

A. INCORRECT. Plausible since the Applicant may incorrectly assume that the valve position can be verified solely by valve stem position, indicator lights, or limit switch operation. Incorrect since A-52.2 is very specific in how to verify a LOCKED OPEN valve.

B. INCORRECT. Plausible since the Applicant may misinterpret the requirements in A-52.2, and incorrectly determine that the last completed valve position documentation is adequate enough to verify the position. Incorrect since A-52.2 is very specific in how to verify a LOCKED OPEN valve.

C. INCORRECT. Plausible since this distractor is correct EXCEPT that A-52.2 specifies that the valve is to be checked in the CLOSED position.

D. CORRECT. In accordance with A-52.2, Step 6.3.4.2 “To verify the position of a locked valve: Locked open – attempt to move handwheel or operator in the closed direction only enough to verify valve indicating the valve is open. Return valve to original position. If unable to move the operator due to the locking device, remove the locking device and attempt to move the operator, or handwheel in the closed direction only enough to verify valve movement. Return valve to original position. Reinstall the locking device and verify that it is securely locked in the required position. If the locking device was unlocked, a second verification of the locking device installation is required.”

Technical Reference(s): A-52.2, Control of Locked Valve and Breaker Operation (p8; Rev 180)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAD08C 1.05

Question Source: Bank # X
Modified Bank #
New

Question History: Last NRC Exam 2012 Ginna RO Retake ILT

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

10 CFR Part 55 Content: 55.41 .10
55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.2.13	
	Importance Rating	4.1	

Knowledge of tagging and clearance procedures.

RO Question #69

Which ONE of the following describes the correct order for hanging isolation tags when tagging out a Safety Injection (SI) Pump in accordance with OP-AA-109-101, Personnel and Equipment Tagout Process?

1) The _______________ isolation tags are hung first;
AND
2) The first tag will be placed on the ______________________.

A. 1) Electrical
2) power supply breaker

B. 1) Electrical
2) Pump control switch

C. 1) Mechanical
2) suction valve

D. 1) Mechanical
2) discharge valve

Answer: B

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret that the procedure requires the breaker to be tagged prior to the control switch. Incorrect since OP-AA-109-101 requires that the control switch be tagged prior to the breaker.

B. CORRECT. According to OP-AA-109-101, Section 7.2.7 “Sequence of Step Performance; Application sequencing should use the following techniques as applicable: Control switches; Power supplies – breakers, disconnects, fuses, etc.; Mechanical isolation points: Isolate discharge valves prior to suction valves." According to P-15.7, Section 6.1, Step 6.1.1.1 directs the Operator to "**VERIFY** control switch is in **PULL STOP**" and then the breaker is removed in Steps 6.1.1.4 through 6.1.1.18.

C. INCORRECT. The first part is plausible since the Applicant may misinterpret the requirements in OP-AA-109-101 that the mechanical isolation should be placed first. Incorrect since the electrical isolation is placed first. The second part is plausible since the Applicant may misinterpret that the suction valve should be tagged prior to the discharge valve for the mechanical isolation. Incorrect since the discharge valve would be tagged prior to the suction valve.

D. INCORRECT. The first part is plausible since the Applicant may misinterpret the requirements in OP-AA-109-101 that the mechanical isolation should be placed first. Incorrect since the electrical isolation is placed first. The second part is plausible since the discharge valve would be tagged prior to the suction valve. Incorrect since the electrical isolation is tagged prior to the mechanical isolation.

Technical Reference(s): OP-AA-109-101, Personnel and Equipment Tagout Process (p24 – 25; Rev 016)

Proposed references to be provided to applicants during examination: None

Learning Objective: N-CE-SA-C-CIN 1.12

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2010 Byron ILT

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 .10

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.2.14	
	Importance Rating	3.9	

Knowledge of the process for controlling equipment configuration or status.

RO Question #70

In accordance with OP-AA-108-117, Protected Equipment program, which ONE of the following correctly states a condition: (**Consider each condition separately**)

(1) requiring equipment to be protected;

AND

(2) in which access on or near protected equipment would be allowed, without SM approval?

A. (1) RHR Pump Surveillance Test with a scheduled duration of 4 hours
(2) Racking out a breaker on a protected 480V Bus for tagging a pump that is NOT protected

B. (1) RHR Pump Surveillance Test with a scheduled duration of 4 hours
(2) Operators are performing actions in accordance with Emergency Operating Procedures

C. (1) Time to reach 200°F in the Spent Fuel Pool (SFP) on loss of normal cooling is less than 72 hours
(2) Racking out a breaker on a protected 480V Bus for tagging a pump that is NOT protected

D. (1) Time to reach 200°F in the Spent Fuel Pool (SFP) on loss of normal cooling is less than 72 hours
(2) Operators are performing actions in accordance with Emergency Operating Procedures

Answer: **D**

Explanation:

A. INCORRECT. The first part is plausible since a Surveillance Test with a scheduled duration greater than one shift would require equipment to be protected. Incorrect since for short duration surveillance testing, posting is not required. The second part is plausible since clearance application that requires only manipulation of a breaker handle would be allowed. Incorrect since racking out a breaker is more intrusive and the potential for accidental mis-positioning of protected equipment exists.

B. INCORRECT. The first part is plausible since a Surveillance Test with a scheduled duration greater than one shift would require equipment to be protected. Incorrect since for short duration surveillance testing, posting is not required. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since clearance application that requires only manipulation of a breaker handle would be allowed. Incorrect since racking out a breaker is more intrusive and the potential for accidental mis-positioning of protected equipment exists.

D. CORRECT. According to OP-AA-108-117, Step 4.2.5 "**When** the time for the SFP to reach 200 degrees Fahrenheit upon loss of normal cooling is less than 72 hours, **then PROTECT** the active loops (pumps, breakers, cooling water, make-up, and other associated components if manipulation or inadvertent bumping of those components could affect operation of the system) of SFP system." Additionally, Step 4.4.1 states "Generally, access or work on or near protected equipment will not be allowed. Exceptions to this rule are as follows: Personnel performing Abnormal Operating Procedures / Emergency Operating Procedures / Emergency Plan actions."

Technical Reference(s): OP-AA-108-117, Protected Equipment Program (p5-6 & 10; Rev 5)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAD03C 1.05

Question Source:
- Bank #: ____
- Modified Bank #: ____
- New: X

Question History: Last NRC Exam ____

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

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.2.43	
	Importance Rating	3.0	

Knowledge of the process used to track inoperable alarms.

RO Question #71

Given the following plant conditions:

- Annunciator H-24, STATION SERVICE AIR LO PRESS 85 PSI, has alarmed repeatedly over the past day
- IMD has installed a recorder to continuously monitor Service Air pressure
- The recorder for Service Air pressure indicates 118 psig and stable
- EOs have been dispatched several times and report:
  - **NO** abnormal indications on Service Air System
  - Service Air pressure reads 120 psig locally

Based on the given conditions, which ONE of the following completes the below statements?

(1) In accordance with OPG-ANNUNCIATOR-FLAGGING, Annunciator Flagging, Annunciator H-24 _________ the definition of a Nuisance Alarm.

AND

(2) An Annunciator that is determined to be a Nuisance Alarm will have a ___ dot placed on the associated annunciator window and be tracked on attachment 1, Alarm Annunciator Out of Service Log.

A. (1) meets
(2) blue

B. (1) does NOT meet
(2) blue

C. (1) meets
(2) black

D. (1) does NOT meet
(2) black

Answer: **C**

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible the Applicant may misinterpret that this constitutes the Annunciator having been taken out of service and warrants placing a blue dot on the Annunciator window. Incorrect since OPG-ANNUNCIATOR-FLAGGING, Step 3.5 states "an Annunciator is considered out of service when the alarm circuitry has been disabled **OR** has all inputs removed. **ENSURE** guidance of CC-AA-112, TEMPORARY CONFIGURATION CHANGES is followed."

B. INCORRECT. The first part is plausible since the Applicant may be unsure of the definition of a Nuisance Alarm and conclude that Maintenance must determine that the Annunciator is malfunctioning prior to being called Nuisance. Incorrect since this is a Nuisance Alarm. The second part is plausible the Applicant may misinterpret that this constitutes the Annunciator having been taken out of service and warrants placing a blue dot on the Annunciator window. Incorrect since OPG-ANNUNCIATOR-FLAGGING, Step 3.5 states "an Annunciator is considered out of service when the alarm circuitry has been disabled **OR** has all inputs removed. **ENSURE** guidance of CC-AA-112, TEMPORARY CONFIGURATION CHANGES is followed."

C. CORRECT. In accordance with OPG-ANNUNCIATOR-FLAGGING, "Alarms may be considered a nuisance **IF** the alarm is: **NOT** valid for existing station, system, **OR** equipment conditions; A result of a loop, circuit, **OR** equipment failure; Although valid, repeated actuation of an alarm that distracts operators." Additionally, BLACK DOT "The sticker is placed on an annunciator window **OR** flag to indicate the following: For placement on alarm windows of nuisance alarms with the approval of the Control Room Senior Reactor Operator." Additionally, OP-AA-103-102 states "Nuisance alarms are an operator distraction. Efforts should be made to **ELIMINATE** the source of the alarm signal/input by operational or maintenance actions. **ANNOTATE** alarm windows disabled in this manner."

D. INCORRECT. The first part is plausible since the Applicant may be unsure of the definition of a Nuisance Alarm and conclude that Maintenance must determine that the Annunciator is malfunctioning prior to being called Nuisance. Incorrect since this is a Nuisance Alarm. The second part is correct.

Technical Reference(s): OPG-ANNUNCIATOR-FLAGGING, Annunciator, Flagging (p3; Rev 003)

OP-AA-103-102, Watch-Standing Practices (p11; Rev 19)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAD74C 1.02

Question Source:	Bank #	X
	Modified Bank #	
	New	

Question History:	Last NRC Exam	

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.3.14	
	Importance Rating	3.4	

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

RO Question #72

Given the following plant conditions:

- A Steam Generator Tube Rupture has occurred on 'B' S/G
- The crew has performed all actions of E-3, Steam Generator Tube Rupture, up to the step to commence depressurization of the RCS

Which ONE of the following describes the action taken with 'B' S/G Atmospheric Relief Valve (ARV), **AND** the reason for this action?

A. CLOSED with controller in MANUAL; minimize radioactive release to atmosphere

B. CLOSED with controller in MANUAL; ensures minimum RCS subcooling will be maintained when RCS depressurization is initiated

C. Set at 1050 psig with controller in AUTO; minimize radioactive release to atmosphere

D. Set at 1050 psig with controller in AUTO; ensures minimum RCS subcooling will be maintained when RCS depressurization is initiated

Answer: C

Explanation:

A. INCORRECT. The first part is plausible since E-3, Step 4.b RNO states "WHEN ruptured S/G pressure less than 1050 psig, THEN verify S/G ARV closed. IF NOT closed, THEN place controller in MANUAL and close S/G ARV." Incorrect since this action is only taken for an ARV controller failure. The second part is correct.

B. INCORRECT. The first part is plausible since E-3, Step 4.b RNO states "WHEN ruptured S/G pressure less than 1050 psig, THEN verify S/G ARV closed. IF NOT closed, THEN place controller in MANUAL and close S/G ARV." Incorrect since this action is only taken for an ARV controller failure. The second part is plausible since this is the reason for the initial RCS cooldown which is based on ruptured S/G pressure. Incorrect since setting the ruptured S/G ARV controller to 1050 psig only minimizes radiological releases and does not affect RCS subcooling.

C. CORRECT. In accordance with E-3, Step 4.a “Adjust ruptured S/G ARV controller to 1050 psig in AUTO”. In accordance with E-3 Background Document, the purpose for Step 4 actions is “To isolate flow from the ruptured steam generator to minimize radiological releases; and To maintain pressure in the ruptured steam generator greater than the pressure in at least one intact steam generator following cooldown of the RCS in subsequent steps.” Setting the ruptured S/G ARV controller to 1050 psig only minimizes radiological releases.

D. INCORRECT. The first part is correct. The second part is plausible since this is the reason for the initial RCS cooldown which is based on ruptured S/G pressure. Incorrect since setting the ruptured S/G ARV controller to 1050 psig only minimizes radiological releases and does not affect RCS subcooling.

Technical Reference(s): E-3, Steam Generator Tube Rupture (p5; Rev 051)
E-3 Background Document (p61; Rev 022)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP03C 2.01

Question Source: Bank # X
Modified Bank #
New

Question History: Last NRC Exam 2019 Ginna ILT

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

10 CFR Part 55 Content: 55.41 .10
55.43

Comments: The KA is matched since the Operator must demonstrate knowledge of the actions taken in E-3. Such accidents provide a direct release path for contaminated primary coolant to the environment via the secondary side relief valves.

The question is at the Comprehension/Analysis cognitive level because the Operator must evaluate the current and/or postulated conditions.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.3.15	
	Importance Rating	2.9	

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

RO Question #73

Given the following indications for Victoreen Model 942 Radiation Monitors (R-1 through R-9):

- RANGE alarm is LIT
- Display indicates 0.00
- Monitor Bar Graph display is extinguished

Which ONE of the following describes the cause for these indications; **AND** the condition required to reset the RANGE alarm?

A. Detector power loss; alarm must be manually reset once power is restored

B. Detector power loss; alarm will automatically reset once power is restored

C. Radiation field is below the instrument range; alarm must be manually reset once the radiation field increases into the range of the detector

D. Radiation field is below the instrument range; alarm will automatically reset once the radiation field increases into the range of the detector

Answer: D

Explanation:

A. INCORRECT. The cause is plausible since a loss of detector power will cause the bar graph display to extinguish. Incorrect since a detector power loss will cause a FAIL alarm. The reset condition is plausible since the radiation monitor would have to be manually reset following a loss of detector power. Incorrect since this cause would automatically reset.

B. INCORRECT. The cause is plausible since a loss of detector power will cause the bar graph display to extinguish. Incorrect since a detector power loss will cause a FAIL alarm. The reset condition is correct.

C. INCORRECT. The cause is correct. The reset condition is plausible since the Applicant may misinterpret the operation of the radiation monitor and think that a manual reset is required. Incorrect since this cause would automatically reset.

D. CORRECT. According to STP-O-17.1, Precaution and Limitation 4.3 “For Victoreen Model 946 monitors (R-1 through R-9), if the measured radiation field is below 0.1 mR/hr, the front panel display will indicate 0.00 mR/hr, the bargraph will be extinguished, and the range alarm indicator will be illuminated in RED. When the measured radiation field increases into the range of the detector, the range alarm indicator will extinguish and normal operation will begin.”

Technical Reference(s): STP-O-17.1, Performance Test of Area, High Range Area, AVT Area and Wide Range Area Radiation Monitors (p7; Rev 010)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3901C 1.09

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 .11

55.43

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.4.14	
	Importance Rating	3.8	

Knowledge of general guidelines for EOP usage.

RO Question #74

Which ONE of the following correctly answers the statements below regarding the use of NOTEs and CAUTIONs in accordance with A-503.1, Emergency and Abnormal Operating Procedures Users Guide?

(1) All NOTEs and CAUTIONs must be read aloud.

AND

(2) An applicable CAUTION encountered a second time in the same procedure may be paraphrased.

	(1)	(2)
A.	NO	NO
B.	NO	YES
C.	YES	NO
D.	YES	YES

Answer: **B**

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret the requirements in A-503.1 and applicable CAUTIONs must be read as written, similar to when the CAUTION first encountered. Incorrect since a repeat CAUTION may be paraphrased.

B. CORRECT. According to A-503.1, Section 2.5.G "If the procedure reader (Unit Supervisor US)) determines that a caution is not applicable to current plant conditions, he may decide not to read the caution aloud. The procedure reader (US) is not required to read aloud EOP/AOP notes. If an applicable caution is encountered more than once by a shift while performing a procedure or series of procedures, it shall be read aloud and acknowledged appropriately the first time. Subsequently, a repeat caution may be paraphrased to the extent necessary to reemphasize the concern."

C. INCORRECT. The first part is plausible since the Applicant may misinterpret the requirement for NOTEs and CAUTIONs to be the same as that for procedure steps. Incorrect since the US need not read NOTEs and CAUTIONs aloud if determined to be not applicable to current plant conditions. The second part is plausible since the Applicant may misinterpret the requirements in A-503.1 and applicable CAUTIONs must be read as written, similar to when the CAUTION first encountered. Incorrect since a repeat CAUTION may be paraphrased.

D. INCORRECT. The first part is plausible since the Applicant may misinterpret the requirement for NOTEs and CAUTIONs to be the same as that for procedure steps. Incorrect since the US need not read NOTEs and CAUTIONs aloud if determined to be not applicable to current plant conditions. The second part is correct.

Technical Reference(s): A-503.1, Emergency and Abnormal Operating Procedures Users Guide (p25; Rev 053)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP50C 1.15

Question Source:	Bank #	X
	Modified Bank #	
	New	

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	.10
	55.43	

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.4.43	
	Importance Rating	3.2	

Knowledge of emergency communications systems and techniques.

RO Question #75

Given the following plant conditions:

- An Emergency Event has been declared
- The Shift Communicator is preparing to notify the State and County agencies following event classification from the Control Room

Which ONE of the following emergency communications systems will the Shift Communicator use FIRST to notify the State and County agencies?

A. Normal plant phone

B. Control Room Blue phone

C. Emergency Notification System (ENS) Red phone

D. Radiological Emergency Communication System (RECS) phone

Answer: D

Explanation:

A. INCORRECT. Plausible since the normal plant phones would be used to communicate emergency data if the RECS phone was NOT available. Incorrect since the RECS phone is the primary means to communicate with the State and County agencies.

B. INCORRECT. Plausible since the Control Room Blue phones would be used to communicate emergency data if the RECS phone and normal plant phones were NOT available. Incorrect since the Control Room Blue phones are used if the RECS phone and normal plant phones do not function.

C. INCORRECT. Plausible since the Applicant may misinterpret that the ENS phone would be used to communicate emergency data with the State and County agencies, as well as, the NRC. Incorrect since the ENS phone would only be used to contact State and County agencies if all other means of emergency communication were NOT available.

D. CORRECT. In accordance with EP-CE-114-100, Attachment 1, Step 2.A lists the following phone systems are to be used to transmit information or form to State and Counties from the Control Room (in order of preference): RECS phone, Normal plant phones, Control Room Blue phones, Control Room Cellular phone, and Control Room Hardwired Satellite phone. Additionally, Attachment 4 directs the Operator for "RECS line notification to State and Counties".

Technical Reference(s): EP-CE-114-100, Emergency Notifications (p6-12 & 21; Rev 9)

Proposed references to be provided to applicants during examination: None

Learning Objective: RSC02C 8.00

Question Source: Bank # ______
Modified Bank # ______
New X

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 .10
55.43 ______

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	APE015 AA2.10	
	Importance Rating	3.7	

Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on loss of cooling or seal injection

SRO Question #76

Given the following plant conditions:

- Plant is operating at 100% power
- All operating Charging Pumps trip
- The Operating crew enters AP-CVCS.3, Loss of All Charging Flow
- Attempts to restore Charging have been unsuccessful

Which ONE of the following correctly completes the statements below?

In accordance with AP-CVCS.3, the Operating crew will:

1) commence a load reduction at a rate of __________;
AND
2) the reactor will be immediately tripped, and RCPs tripped if Annunciator ________ alarms.

A. 1) 3 %/min
2) A-7, RCP A CCW RETURN HI TEMP OR LO FLOW

B. 1) 3 %/min
2) A-21, COMP COOLING HX OUT HI TEMP

C. 1) 5 %/min
2) A-7, RCP A CCW RETURN HI TEMP OR LO FLOW

D. 1) 5 %/min
2) A-21, COMP COOLING HX OUT HI TEMP

Answer: **C**

Explanation:

A. INCORRECT. The first part is plausible since 3 %/min is the load reduction rate in AP-SG.1 and the Applicant may misinterpret the rates between the procedures. Incorrect since AP-CVCS.3 directs the Operator to initiate a load reduction at a rate of 5 %/min. The second part is correct.

B. INCORRECT. The first part is plausible since 3 %/min is the load reduction rate in AP-SG.1 and the Applicant may misinterpret the rates between the procedures. Incorrect since AP-CVCS.3 directs the Operator to initiate a load reduction at a rate of 5 %/min. The second part is plausible since AR-A-21 directs the Operator to refer to the applicable AP-CCW procedure which have criteria to trip the reactor and RCPs. Incorrect since AP-CVCS.3, Step 5 RNO directs the Operator to trip the reactor and RCP if Annunciator A-7(15) is NOT extinguished.

C. CORRECT. In accordance with AP-CVCS.3, Step 17, the Operator is directed to "Initiate Load Reduction: Select rate of 5%/min on thumbwheel". Additionally, AP-CVCS.3, Step 5 directs the Operator to "Check CCW To RCP Thermal Barriers: Annunciator A-7(15), RCP A(B) CCW RETURN HI TEMP OR LOW FLOW – EXTINGUISHED". With the Annunciator in alarm, Step 5 RNO directs the Operator "IF CCW lost to RCP(s), THEN perform the following: Trip the Reactor; Trip affected RCP(s)."

D. INCORRECT. The first part is correct. The second part is plausible since AR-A-21 directs the Operator to refer to the applicable AP-CCW procedure which have criteria to trip the reactor and RCPs. Incorrect since AP-CVCS.3, Step 5 RNO directs the Operator to trip the reactor and RCP if Annunciator A-7(15) is NOT extinguished.

Technical Reference(s): AP-CVCS.3, Loss of All Charging Flow (p4 & 11; Rev 016)
AR-A-7, RCP A CCW RETURN HI TEMP OR LO FLOW (Rev 00801)
AR-A-21, COMP COOLING HX OUT HI TEMP (Rev 01000)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP31C 2.01

Question Source: Bank # ______
Modified Bank # ______
New X

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments: The first part of the question tests specific knowledge of AP-CVCS.3, Step 17 which would not be required of a RO Applicant. The second part of the question is RO since a RO Applicant would be expected to use integrated plant knowledge to answer this part of the question. Therefore, this is an SRO question.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	APE026 G2.1.20	
	Importance Rating	4.6	

Ability to interpret and execute procedure steps. (Loss of CCW)

SRO Question #77

Given the following initial plant conditions:

- The Operating crew is performing O-2.2, Plant Shutdown from Hot Shutdown to Cold Conditions
- RHR Cooling has just been placed in service
- RCS cooldown rate is < 20°F/hr

Subsequently, 5 minutes later:

- Both CCW Pumps trip and cannot be started
- The US enters AP-CCW.3, Loss of CCW – Plant Shutdown
- RCS temperature is rising

Which ONE of the following correctly answers the questions below?

1) What is the preferred method for stabilizing RCS temperature in accordance with AP-CCW.3 for the given plant conditions?
   AND
2) In accordance with Technical Specification LCO 3.7.7, Component Cooling Water (CCW) System, is LCO 3.0.3 applicable if CCW Pumps can NOT be restored?

A. 1) Initiate S/G blowdown from both S/Gs
   2) NO

B. 1) Control S/G ARVs
   2) NO

C. 1) Initiate S/G blowdown from both S/Gs
   2) YES

D. 1) Control S/G ARVs

2) YES

Answer: **B**

Explanation:

A. INCORRECT. The first part is plausible since AP-CCW.3, Step 7 RNO has the option to Initiate S/G blowdown to stabilize RCS temperature. Incorrect since this method would only be used if S/G cooling via the ARVs did NOT provide adequate cooling. The second part is correct.

B. CORRECT. According to AP-CCW.3, Step 7 RNO, since the given plant conditions have RCS temperature rising, the Operator is directed "IF RHR cooling available, THEN adjust RHR cooling to stabilize RCS temperature AND go to Step 8. IF S/G cooling available, THEN control S/G ARVs to stabilize RCS temperature." With the given plant conditions that RHR cooling just being placed in service, the Applicant must determine that S/Gs are still available. With the loss of CCW, adjusting RHR cooling will have minimal to no effect on RCS temperature. According to Technical Specification LCO 3.7.7, Condition 'D' NOTE "LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one CCW train, one CCW heat exchanger, and the loop header are restored to OPERABLE status." Additionally, LCO 3.7.7 Basis states "Required Actions D.1, D.2, and D.3 are modified by a Note indicating that all required MODE changes or power reductions required by other LCOs are suspended until one CCW train, one CCW heat exchanger, and the loop header are restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition."

C. INCORRECT. The first part is plausible since AP-CCW.3, Step 7 RNO has the option to Initiate S/G blowdown to stabilize RCS temperature. Incorrect since this method would only be used if S/G cooling via the ARVs did NOT provide adequate cooling. The second part is plausible since most Technical Specification LCOs direct the Operator to enter LCO 3.0.3 when all trains are INOPERABLE. Incorrect since Technical Specification LCO 3.7.7 directs the Operator that LCO 3.0.3 is not applicable for the given plant conditions.

D. INCORRECT. The first part is correct. The second part is plausible since most Technical Specification LCOs direct the Operator to enter LCO 3.0.3 when all trains are INOPERABLE. Incorrect since Technical Specification LCO 3.7.7 directs the Operator that LCO 3.0.3 is not applicable for the given plant conditions.

Technical Reference(s):	AP-CCW.3, Loss of CCW – Plant Shutdown (p7; Rev 01902)
	Technical Specification LCO 3.7.7, Component Cooling Water (CCW) System (Amendment 80)
	Technical Specification LCO 3.7.7 Basis (Rev 58)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP03C 2.01; R2501C 1.13

Question Source:	Bank #	
	Modified Bank #	
	New	X

Question History:	Last NRC Exam	

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

10 CFR Part 55 Content:	55.41	
	55.43	.2

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	EPE038 EA2.08	
	Importance Rating	4.4	

Ability to determine or interpret the following as they apply to a SGTR: Viable alternatives for placing plant in safe condition when condenser is not available

SRO Question #78

Given the following initial plant conditions:

- A SGTR has occurred on 'B' S/G
- ‘B’ S/G level is 30% and slowly rising
- Concurrent with the SGTR, a loss of Offsite Power occurred
- Both MDAFW Pumps tripped upon starting
- ‘A’ MSIV is OPEN
- Initial RCS cooldown and depressurization has been completed

Subsequently:

- The TDAFW Pump trips on overspeed
- Attempts to reset the TDAFW Pump have been unsuccessful
- ‘A’ S/G water level is 5% and slowly lowering
- The US is at the procedure step to check if SI flow should be terminated

Which ONE of the following completes the following statements in accordance with E-3, Steam Generator Tube Rupture?

1) The initial RCS cooldown was completed by ________________________.
AND
2) The US will transition to ________________ to complete plant stabilization.

A. 1) dumping steam to condenser from ‘A’ S/G
2) ES-3.1, Post-SGTR Cooldown Using Backfill

B. 1) opening ‘A’ S/G Atmospheric relief Valve (ARV)
2) ES-3.1, Post-SGTR Cooldown Using Backfill

C. 1) dumping steam to condenser from ‘A’ S/G
   2) ECA-3.1, SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired

D. 1) opening ‘A’ S/G Atmospheric relief Valve (ARV)
   2) ECA-3.1, SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired

Answer: D

Explanation:

A. INCORRECT. The first part is plausible since the Applicant may misinterpret the plant conditions and determine that the condenser is available, and this would be the normal method for cooling down the RCS in E-3. Incorrect since the loss of Offsite Power has resulted in loss of Circ Water Pumps rendering the condenser unavailable. The second part is plausible since the Applicant may determine that sufficient inventory is available in the intact S/G to make the normal transition to ES-3.1. Incorrect since the intact S/G water level is not sufficient with no AFW flow available; therefore, transition to ECA-3.1 must be made.

B. INCORRECT. The first part is correct. The second part is plausible since the Applicant may determine that sufficient inventory is available in the intact S/G to make the normal transition to ES-3.1. Incorrect since the intact S/G water level is not sufficient with no AFW flow available; therefore, transition to ECA-3.1 must be made.

C. INCORRECT. The first part is plausible since the Applicant may misinterpret the plant conditions and determine that the condenser is available, and this would be the normal method for cooling down the RCS in E-3. Incorrect since the loss of Offsite Power has resulted in loss of Circ Water Pumps rendering the condenser unavailable. The second part is correct.

D. CORRECT. The loss of Offsite Power has resulted in Buses 11A and 11B being de-energized, which results in a loss of both Circ Water Pumps. With no Circ Water Pumps running, the condenser is unavailable even though the intact S/G (‘A’ S/G) MSIV is OPEN. Accordingly, E-3 Step 10.b RNO directs the Operator to “Manually or locally initiate steam dump from intact S/G at maximum rate using S/G ARV.” If unable to initiate dumping steam to condenser. Additionally, E-3, Step 21.b directs the Operator to verify “Secondary heat sink: Total feed flow to S/G(s) – GREATER THAN 200 GPM AVAILABLE; OR Narrow range level in at least one intact S/G – GREATER THAN 7% [25% adverse CNMT]”. Since neither condition is satisfied, Step 21.b RNO states “IF neither condition satisfied, THEN do NOT stop SI pumps. Go to ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT – SUBCOOLED RECOVERY DESIRED, Step 1.”

Technical Reference(s): E-3, Steam Generator Tube Rupture (p11 & 20; Rev 051)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP03C 2.01

Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	APE062 G2.4.8	
	Importance Rating	4.5	

Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (Loss of SW)

SRO Question #79

Given the following plant conditions:
- Plant tripped from 100% power due to an inadvertent SI signal
- The Operating crew has transitioned to ES-1.1, SI Termination
- ‘A’ SW Pump is running
- NO other SW Pump can be started
- SI and RHR Pumps have been stopped and placed in AUTO
- PRZR level is 9% and rising slowly

Which ONE of the following correctly completes the statements below?

In accordance with A-503.1, Emergency and Abnormal Operating Procedures Users Guide, the US…

1) ____________ perform AP-SW.2, Loss of Service Water, in parallel with ES-1.1;

   AND

2) in response to PRZR level, ________ required to transition to FR-I.2, Response to Low Pressurizer Level.

A. 1) should
   2) is

B. 1) should
   2) is NOT

C. 1) should NOT
   2) is

D. 1) should NOT

2) is NOT

Answer: **B**

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret that transition to a Functional Restoration (FR) procedure is always required when CSFSTs are being monitored. Incorrect since this would be a YELLOW path FR which is not required to be transitioned to from the EOP network.

B. CORRECT. In accordance with A-503.1, Section 5.2.M.4 “The words “refer to” followed by a procedure designator and title, are used to denote a procedure which may provide necessary or useful information during the execution of an EOP/AP. In general, those procedures referenced cover low probability occurrences, or plant evaluations with their own procedures whose inclusion in the EOP/AP would cause excessive complication of and reduced effectiveness of the EOP/AP. Referenced procedures should be performed in parallel with the primary procedure.” Also, A-503.1, Section 5.2.A.6 states “When performing EOPs, various plant conditions may occur which would normally be addressed by AOPs (such as ARs or APs). Actions may be taken per AOPs that DO NOT conflict with the actions of the EOPs if adequate resources are available. The AOP should be entered and procedure steps followed.” Additionally, ES-1.1, Step 3.a RNO directs the Operator if “at least two SW pumps are NOT running” then “Refer to AP-SW.2, LOSS OF SERVICE WATER”. According to F-0.6, Inventory CSFST, with SI Pumps not running and PRZR level < 13%, a YELLOW path terminus is present. According to A-503.1, Section 5.2.P.4.c.(7) “YELLOW path terminus (3) is an indication of an off-normal condition. These do not take priority over other procedures.” Also, ES-1.1 contains steps that will mitigate the low PRZR level.

C. INCORRECT. The first part is plausible since the Applicant may misinterpret the requirements of A-503.1 in that EOP actions take priority over AOP actions and not realize that the mitigating actions contained in AP-SW.2 will assist in mitigating the plant event. Incorrect since the actions contained in AP-SW.2 will NOT conflict with the actions contained in ES-1.1. The second part is plausible since the Applicant may misinterpret that transition to a Functional Restoration (FR) procedure is always required when CSFSTs are being monitored. Incorrect since this would be a YELLOW path FR which is not required to be transitioned to from the EOP network.

D. INCORRECT. The first part is plausible since the Applicant may misinterpret the requirements of A-503.1 in that EOP actions take priority over AOP actions and not realize that the mitigating actions contained in AP-SW.2 will assist in mitigating the plant event. Incorrect since the actions contained in AP-SW.2 will NOT conflict with the actions contained in ES-1.1. The second part is correct.

Technical Reference(s):	A-503.1, Emergency and Abnormal Operating Procedures Users Guide (p22, 28 &35; Rev 053)
	ES-1.1, SI Termination (p4; Rev 036)
	F-0.6, Inventory CSFST (p2; Rev 005)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP50C 1.08, 1.13; RES11C 2.01

Question Source:
- Bank #: X
- Modified Bank #:
- New:

Question History: Last NRC Exam 2014 Vogtle ILT

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

10 CFR Part 55 Content:
- 55.41:
- 55.43: .5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	APE065 G2.4.31	
	Importance Rating	4.1	

Knowledge of annunciator alarms, indications, or response procedures. (Loss of Instrument Air)

SRO Question #80

Given the following plant conditions:

- Plant is operating at 100% power
- AOV-5392, Instrument Air to CNMT, fails closed
- The US enters AP-IA.1, Loss of Instrument Air
- The HCO can NOT open AOV-5392

Which ONE of the following correctly answers the statements below in accordance with AP-IA.1?

(1) Instrument Air to CNMT will be restored by ________________________.

AND

(2) Entry into Technical Specification 3.6.3, Containment Isolation Boundaries _____ required.

A. (1) locally opening AOV-5392
(2) is

B. (1) locally opening AOV-5392
(2) is NOT

C. (1) aligning Service Air to Instrument Air Crossties for CNMT
(2) is

D. (1) aligning Service Air to Instrument Air Crossties for CNMT
(2) is NOT

Answer: **A**

Explanation:

A. CORRECT. In accordance with AP-IA.1, Step 4.a RNO if AOV-5392 is NOT OPEN, the Operator is directed to “Locally open AOV-5392.” AP-IA.1, Step 4 NOTE states “EO will need to be stationed at AOV-5392 if locally opened and ITS LCO 3.6.3.A, Containment Isolation Boundaries is applicable.”

B. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret that with AOV-5392 locally opened and an EO stationed to immediately close the valve upon receipt of a CI signal, that the CNMT Isolation Boundary is still OPERABLE. Incorrect since AOV-5392 will NOT automatically CLOSE if locally opened.

C. INCORRECT. The first part is plausible since this would be the method of restoring Air pressure to CNMT in ER-FIRE.1. Incorrect since AP-IA.1 directs the Operator to locally open AOV-5392. The second part is correct.

D. INCORRECT. The first part is plausible since this would be the method of restoring Air pressure to CNMT in ER-FIRE.1. Incorrect since AP-IA.1 directs the Operator to locally open AOV-5392. The second part is plausible since Service air to CNMT is a closed system and remains intact and the Applicant may misinterpret that the CNMT Isolation Boundary is still OPERABLE. Incorrect since unisolating Service Air to CNMT requires entry into Technical Specification LCO 3.6.3.

Technical Reference(s): AP-IA.1, Loss of Instrument Air (Rev 024)

ER-FIRE.1, Alternate Shutdown for Control Room Abandonment (p73-75; Rev 046)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAP10C 2.01

Question Source:
- Bank # ____
- Modified Bank # ____
- New X

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
- 55.41 ____
- 55.43 .5

Comments: SRO-only since the first part requires the Applicant to have knowledge of the specific procedure step and is NOT solely systems knowledge, Immediate Operator Actions, entry conditions for AOPs, nor the overall mitigative strategy of the AOP. Additionally, the second part requires the Applicant to have knowledge of what constitutes an OPERABLE Containment Isolation Boundary which is contained in the Technical Specification Basis and is NOT “above-the-line” information.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	APE077 AA2.05	
	Importance Rating	3.8	

Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Operational status of offsite circuit

SRO Question #81

Given the following initial plant conditions:

- Plant is operating at 100% power
- Severe thunderstorms are being experienced
- RG&E Energy Control Center (ECC) has informed the Control Room that the Post Contingency Low Voltage Alarm (PCLVA) is in alarm
- The Operating crew is monitoring O-6.9, Ginna Station operating Limits for Station 13A Transmission

Subsequently, the Main Generator trips

Which ONE of the following correctly completes the following statements in accordance with O-6.9?

1) With the PCLVA in alarm, 480V Safeguard Buses WILL/WILL NOT experience an undervoltage condition, concurrent with worst-case accident loading;

   AND

2) The OPERABILITY status of the Offsite Power circuits is ______________.

A. 1) WILL
   2) INOPERABLE

B. 1) WILL NOT
   2) INOPERABLE

C. 1) WILL
   2) OPERABLE

D. 1) WILL NOT

2) OPERABLE

Answer: **A**

Explanation:

A. CORRECT. According to O-6.9, Precaution 4.8 “The absence of a PCLVA on the RG&E SECAS **OR** compliance with the Offsite Power Operability limits indicated in Attachment 2 through Attachment 9 for the current circuit configuration will ensure that the subsequent offsite 115kV system voltage transient will **NOT** result in Ginna Station experiencing an undervoltage condition on the 480V Safeguard Busses **IF** the main generator should trip.” Additionally, Precaution 4.4 states “Offsite power is inoperable if any of the following conditions exist: Circuit 7T **AND** 767 voltage regulators are inoperable; The PCLVA is in alarm; or When using curves **AND** STA 13A voltage is below the curve.” Therefore, if the PCLVA is in alarm and the Main Generator trips (given plant conditions), the Safeguards Buses can be expected to experience an undervoltage condition with worst-case accident loading.

B. INCORRECT. The first part is plausible since the Applicant may misinterpret that the O-6.9 requirements are concerned with Offsite Power loss vice a loss of the Main Generator and therefore, the Safeguard Buses will NOT experience an undervoltage condition for the given plant conditions. Incorrect since the main generator trip is the contingent event of concern and with worst-case accident loading, the Safeguard Buses will experience an undervoltage condition. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret the given plant conditions and conclude that the Post Contingency Low Voltage Early Warning Alarm is in therefore; the Offsite Power circuits would be OPERABLE. Incorrect since the PCLVA being in alarm makes the Offsite Power circuits INOPERABLE.

D. INCORRECT. The first part is plausible since the Applicant may misinterpret that the O-6.9 requirements are concerned with Offsite Power loss vice a loss of the Main Generator and therefore, the Safeguard Buses will NOT experience an undervoltage condition for the given plant conditions. Incorrect since the main generator trip is the contingent event of concern and with worst-case accident loading, the Safeguard Buses will experience an undervoltage condition. The second part is plausible since the Applicant may misinterpret the given plant conditions and conclude that the Post Contingency Low Voltage Early Warning Alarm is in therefore; the Offsite Power circuits would be OPERABLE. Incorrect since the PCLVA being in alarm makes the Offsite Power circuits INOPERABLE.

Technical Reference(s): O-6.9, Ginna Station Operating Limits for Station 13A Transmission (p6 – 7; Rev 041)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0503C 1.06, 1.13

Question Source: Bank # X
Modified Bank # ______
New ______

Question History: Last NRC Exam 2012 Ginna ILT

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

10 CFR Part 55 Content: 55.41 ______
55.43 .2

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	APE001 AA2.03	
	Importance Rating	4.8	

Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Proper actions to be taken if automatic safety functions have not taken place

SRO Question #82

Given the following plant conditions:

- Crew is performing O-1.2, Plant Startup from Hot Shutdown to Full Load
- Power is being raised from 50% to 100% power at 10%/hr using boron dilution and rod withdrawal
- Power is currently at 65%
- Control Bank D Rods are at 160 steps
- ROD CONTROL BANK SELECTOR switch is in MANUAL
- When the ROD IN/OUT switch is released, control rods continued withdrawing

Which ONE of the following identifies:

(1) ATWS Mitigation protection OPERABILITY;

AND

(2) the actions required by the appropriate plant procedure?

A. (1) OPERABLE
(2) trip the reactor and enter E-0, Reactor Trip or Safety Injection

B. (1) INOPERABLE
(2) trip the reactor and enter E-0, Reactor Trip or Safety Injection

C. (1) OPERABLE
(2) place ROD CONTROL BANK SELECTOR switch to AUTO and verify Control Rod motion stops

D. (1) INOPERABLE
(2) place ROD CONTROL BANK SELECTOR switch to AUTO and verify Control Rod motion stops

Answer: **B**

Explanation:

A. INCORRECT. The first part is plausible since the Applicant may misinterpret that operability of ATWS mitigation capability is only concerned with PORVs and AMSAC being operational. Incorrect since the ROD IN/OUT switch has malfunctioned, the HCO no longer has manual control rod capability; therefore, ATWS mitigation capability is NOT functional. The second part is correct.

B. CORRECT. TRM TR 3.4.3 Basis states “There are three functions that are required to provide ATWS protection for these events: 1) Both Pressurizer PORVs shall be operable with their respective block valves open; 2) The Operator shall have manual control rod insertion capability; 3) The ATWS Mitigation System Actuation Circuitry (AMSAC) shall be operable.” Additionally, “If any of these three functions which are required to provide ATWS protection are inoperable, then ATWS mitigation protection shall be declared inoperable immediately.” Therefore, since control rods continued moving with Rod Control Bank Selector switch in MANUAL, there is a malfunction in the ROD IN/OUT switch making manual rod insertion unavailable and ATWS mitigation capability NON-FUNCTIONAL. In accordance with AP-RCC.1, Step 1 the Operator is directed to “Evaluate Rod Control System Operability” by checking turbine load stable, placing rods in MANUAL, and verifying control rod motion stops. Since control rods were initially in MANUAL and there was continuous rod withdrawal, the RNO directs the Operator to “Manually trip the reactor and go to E-0, REACTOR TRIP or SAFETY INJECTION.”

C. INCORRECT. The first part is plausible since the Applicant may misinterpret that operability of ATWS mitigation capability is only concerned with PORVs and AMSAC being operational. Incorrect since the ROD IN/OUT switch has malfunctioned, the HCO no longer has manual control rod capability; therefore, ATWS mitigation capability is NOT functional. The second part is plausible since AP-RCC.1, Step 1 directs the Operator to place Rods to MANUAL and verify control rod motion stops and the Applicant may misinterpret that AP-RCC.1 also directs placing rods to AUTO and verifying rod motion stops.” Incorrect since AP-RCC.1, Step 1.c RNO directs the Operator to manually trip the reactor and enter E-0 if rod motion does NOT stop with rods in MANUAL.

D. INCORRECT. The first part is correct. The second part is plausible since AP-RCC.1, Step 1 directs the Operator to place Rods to MANUAL and verify control rod motion stops and the Applicant may misinterpret that AP-RCC.1 also directs placing rods to AUTO and verifying rod motion stops.” Incorrect since AP-RCC.1, Step 1.c RNO directs the Operator to manually trip the reactor and enter E-0 if rod motion does NOT stop with rods in MANUAL.

Technical Reference(s):	Technical Requirements TR 3.4.3 Basis (p TRB3.4.3-1 – 3.4.3-2; Rev 61)
	AP-RCC.1, Continuous Control Rod Withdrawal/Insertion (p3; Rev 012)

Proposed references to be provided to applicants during examination: None

Learning Objective: R4201C 1.13; RAP12C 2.01

Question Source: Bank # ______

Modified Bank # ______

New X

Question History: Last NRC Exam ______

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 .2

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	APE033 AA2.08	
	Importance Rating	3.4	

Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Intermediate range channel operability

SRO Question #83

Given the following plant conditions:

- The plant is stabilized at 2.5% power with O-1.2, Plant Startup from Hot Shutdown to Full Load, in progress
- The HCO reports the following Intermediate Range Channel indications:
  - N-35 indicates 8.5×10^{-7} amps
  - N-36 indicates 1.1×10^{-5} amps

Which ONE of the following correctly completes the statements below?

(1) The US will determine that Intermediate Range Channel ________ has failed the Channel Check and must be declared INOPERABLE.

AND

(2) In accordance with Technical Specifications, the US ___________ allowed to enter MODE 1.

A. (1) N-35
(2) is

B. (1) N-36
(2) is

C. (1) N-35
(2) is NOT

D. (1) N-36
(2) is NOT

Answer: **A**

Explanation:

A. CORRECT. According to P-6, Attachment 6, with the reactor operating at approximately 2.5% power (given), the IR Channels should indicate approximately 1×10^{-5} amps. Since N-35 is indicating a decade below that value, it fails the Channel Check requirement and should be declared INOPERABLE. According to Technical Specification LCO 3.3.1, Table 3.3.1-1, Function 3 applies which directs the Operator to Condition E, with the Required Actions to "Reduce THERMAL POWER to < 5E-11 amps **OR** Increase THERMAL POWER to ≥ 8% RTP" with a Completion Time of "2 hours".

B. INCORRECT. The first part is plausible since the Applicant may misinterpret the nuclear instrumentation range overlaps and determine that N-36 has failed its Channel Check. Incorrect since at approximately 2.5% power, the IR channels should indicate approximately 1E-5 amps. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since LCO 3.3.1, Condition E contains a NOTE that prevents raising reactor power greater than or equal to 8% power if both IR Channels are INOPERABLE **OR** power is initially ≤ 5E-11 amps. Incorrect since only one IR channel is IN OPERABLE and reactor power is approximately 1E-5 amps.

D. INCORRECT. The first part is plausible since the Applicant may misinterpret the nuclear instrumentation range overlaps and determine that N-36 has failed its Channel Check. Incorrect since at approximately 2.5% power, the IR channels should indicate approximately 1E-5 amps. The second part is plausible since LCO 3.3.1, Condition E contains a NOTE that prevents raising reactor power greater than or equal to 8% power if both IR Channels are INOPERABLE **OR** power is initially ≤ 5E-11 amps. Incorrect since only one IR channel is IN OPERABLE and reactor power is approximately 1E-5 amps.

Technical Reference(s): P-6, Precautions, Limitations and Setpoints Nuclear Instrumentation System (p31; Rev 02200)

Technical Specification LCO 3.3.1, Reactor Trip System (RTS) Instrumentation (Amendment 112)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3301C 1.13a, 1.13c

Question Source: Bank #

Modified Bank # X

New

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	
	55.43	.2

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	APE067 AA2.13	
	Importance Rating	4.4	

Ability to determine and interpret the following as they apply to the Plant Fire on Site: Need for emergency plant shutdown

SRO Question #84

Given the following plant conditions:

- Plant is operating at 17% power
- Main Generator has just been synchronized to the grid
- Bus 11A and 11B Normal Feed Breakers are OPEN
- 'A' MFW Pump is running
- Fire Detection System Z41, TURBINE BUILDING 271-0 SWITCHGEAR AREA, alarms
- The US enters ER-FIRE.0, Control Room Response to Fire Alarms and Reports
- Fire Brigade Captain reports an active fire in Bus 11A
- Bus 11A-12A Bus Tie Breaker trips

Which ONE of the following correctly states the procedure the US should enter FIRST to mitigate the event?

A. AP-CW.1, Loss of a Circ Water Pump

B. AP-RCS.2, Loss of Reactor Coolant Flow

C. AP-TURB.1, Turbine Trip Without Reactor Trip Required

D. E-0, Reactor Trip or Safety Injection

Answer: B

Explanation:

A. INCORRECT. Plausible since 'A' Circ Water Pump has lost power and an entry condition for AP-CW.1 is met. Incorrect since 'B' Circ Water Pump is still operating, and AP-CW.1 would not provide any actions to stabilize the plant by an emergency shutdown.

B. CORRECT. Trip of the Bus 11A-12A Bus Tie Breaker results in loss of power to Bus 11A which results in loss of power to 'A' RCP. AP-RCS.2, Step 1 RNO directs the Operator "IF reactor trip breakers closed, THEN trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION" if both RCPs are NOT running.

C. INCORRECT. Plausible since the operating MFW Pump has lost power and this will result in a turbine trip. Since the given reactor power is less than the P-9 setpoint (50%), the reactor will not automatically trip and an entry condition for AP-TURB.1 is met. Incorrect since it is expected that the Turbine Stop Valves are closed and a manual reactor trip would not be directed in AP-TURB.1. Therefore, AP-TURB.1 would not provide any actions to stabilize the plant by an emergency shutdown.

D. INCORRECT. Plausible since the Applicant may determine that an automatic reactor trip has occurred due to either the loss of 'A' RCP and/or loss of 'A' Circ Water Pump. Incorrect since an automatic reactor trip has not occurred, the reactor must be manually tripped as directed in AP-RCS.2 and then E-0 will be entered.

Technical Reference(s):
- P-12, Electrical Systems Precautions, Limitations and Setpoints (p33; Rev 029)
- AP-CW.1, Loss of a Circ Water Pump (p2-3; Rev 01400)
- AP-RCS.2, Loss of Reactor Coolant Flow (Rev 13)
- AP-TURB.1, Turbine Trip Without Rx Trip Required (p2 & 4; Rev 020)
- P-1, Reactor Control and Protection System (p40-42; Rev 078)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0601C 1.04 & 1.06

Question Source: Bank # ____
Modified Bank # ____
New X

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 .5

Comments: K/A matched as Applicant must determine the conditions require an emergency plant shutdown and select the appropriate procedure that will direct the shutdown.

SRO-only knowledge since the Applicant must “Assess plant conditions (normal, abnormal, or emergency) and then select a procedure with which to proceed”.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	EPE 074 G2.4.21	
	Importance Rating	4.6	

Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (Inadequate Core Cooling)

SRO Question #85

Given the following plant conditions:

- A LOCA has occurred
- ALL SI Pumps have failed
- Both RCPs are secured
- The Operating crew is performing E-0, Reactor Trip or Safety Injection
- RCS pressure is 700 psig and stable
- Average Core Exit Thermocouples (CETs) are 710°F and rising slowly
- Both trains of RVLIS indicates 53% and lowering slowly
- CNMT pressure is 5 psig and stable

Which ONE of the following describes the procedure the US will transition to from E-0;
AND
the first major action that is required?

A. FR-C.1, Response to Inadequate Core Cooling;
start one RCP to establish forced circulation cooling flow

B. FR-C.1, Response to Inadequate Core Cooling;
dump steam from intact S/Gs to cooldown and depressurize the RCS

C. FR-C.2, Response to Degraded Core Cooling;
start one RCP to establish forced circulation cooling flow

D. FR-C.2, Response to Degraded Core Cooling;
dump steam from intact S/Gs to cooldown and depressurize the RCS

Answer: B

Explanation:

A. INCORRECT. The procedure is correct. The action is plausible since FR-C.1, Step 5 directs the Operator to "Check RCP Support Conditions" and the Applicant may Misinterpret this step to imply to start an RCP if support conditions are available. Incorrect since RCPs are not started in FR-C.1 until after S/Gs are depressurized and CETs are > 1200°F.

B. CORRECT. In accordance with F-0.2, Core Cooling CSFST with CETs > 700°F, subcooling will NOT be greater than the requirements of FIG-1.0, no RCPs are running, and RVLIS less than 55% (adverse CNMT exists in given plant conditions); the Operator is directed to "GO TO FR-C.1". FR-C.1, Step 16 directs the Operator to "Depressurize All Intact S/Gs to 260 PSIG". FR-C.1 Background Document states that this is performed to depressurize the RCS to allow SI Accumulator injection. Additionally, dumping steam will result in lowering RCS temperature.

C. INCORRECT. The procedure is plausible since the Applicant may misinterpret the F-0.2 requirements for RVLIS during adverse CNMT conditions and determine that FR-C.2 entry is required. Incorrect since the given RVLIS level is less than that required for FR-C.1 entry. The action is plausible since FR-C.1, Step 5 directs the Operator to "Check RCP Support Conditions" and the Applicant may Misinterpret this step to imply to start an RCP if support conditions are available. Incorrect since RCPs are not started in FR-C.1 until after S/Gs are depressurized and CETs are > 1200°F.

D. INCORRECT. The procedure is plausible since the Applicant may misinterpret the F-0.2 requirements for RVLIS during adverse CNMT conditions and determine that FR-C.2 entry is required. Incorrect since the given RVLIS level is less than that required for FR-C.1 entry. The action is correct.

Technical Reference(s):
- F-0.2, Core Cooling CSFST (p2; Rev 00600)
- FIG-1.0, Figure MIN Subcooling (Rev 00200)
- FR-C.1, Response to Inadequate Core Cooling (p6 & 10; Rev 029)

Proposed references to be provided to applicants during examination: None

Learning Objective: RFRC1C 1.02, 2.01

Question Source:
- Bank # X
- Modified Bank #
- New

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	003 G2.1.32	
	Importance Rating	4.0	

Ability to explain and apply system limits and precautions.
(RCPS)

SRO Question #86

Given the following plant conditions:

- Plant is in MODE 5, Loops Filled
- 'A' RCP is being started to commence RCS heatup
- The following conditions exist:
    - RCS T_{COLD}: 145°F
    - S/G Secondary Handholes: 155°F

1) In accordance with O-1.1, Plant Heatup from Cold Shutdown to Hot Shutdown, can the RCP be started;

   AND

2) For the given plant conditions, what are the Technical Specification bases for the criteria to start an RCP?

A. 1) YES
   2) To prevent RCS overpressurization due to energy addition to the RCS.

B. 1) YES
   2) To prevent challenging the opening of the PORVs which are NOT designed to relieve a water solid condition.

C. 1) NO
   2) To prevent RCS overpressurization due to energy addition to the RCS.

D. 1) NO
   2) To prevent challenging the opening of the PORVs which are NOT designed to relieve a water solid condition.

Answer: **C**

Explanation:

A. INCORRECT. The first part is plausible since Technical Specification LCO 3.4.7 would allow starting the RCP with the secondary side of each S/G ≤ 50°F above each RCS cold leg temperature before the start of the RCP. Incorrect since O-1.1, Step 6.2.15 requires RCS temperature greater than or equal to 0°F AND less than or equal to 5°F above S/G Secondary Handhole temperature. The second part is correct.

B. INCORRECT. The first part is plausible since Technical Specification LCO 3.4.7 would allow starting the RCP with the secondary side of each S/G ≤ 50°F above each RCS cold leg temperature before the start of the RCP. Incorrect since O-1.1, Step 6.2.15 requires RCS temperature greater than or equal to 0°F AND less than or equal to 5°F above S/G Secondary Handhole temperature. The second part is plausible since the PORVs are NOT designed for relieving water when the plant is at full power. Incorrect since the PORVs are set for LTOP and are designed to relieve water solid conditions at low temperatures.

C. CORRECT. In accordance with O-1.1, Step 6.2.15 directs the Operator "**PERFORM** RCP Pre-Start RCS/Steam Generator Temperature Profile Check as follows: **IF** either S/G Secondary Handhole temperature is greater than or equal to 0°F above either Loop Cold Leg temperature, **THEN ALLOW** reactor coolant temperature to rise. **MAINTAIN** RCS temperature greater than **OR** equal to 0°F **AND** less than or equal to 5°F above S/G Secondary Handhole temperature." According to Technical Specification LCO 3.4.7 Basis "Restraints on the pressurizer water volume and SG secondary side water temperature are to prevent a low temperature overpressure event due to a thermal transient when an RCP is started and the colder RCS water enters the warmer SG and expands."

D. INCORRECT. The first part is correct. The second part is plausible since the PORVs are NOT designed for relieving water when the plant is at full power. Incorrect since the PORVs are set for LTOP and are designed to relieve water solid conditions at low temperatures.

Technical Reference(s):	O-1.1, Plant Heatup from Cold Shutdown to Hot Shutdown (p27, 33-34; Rev 176)
	Technical Specification LCO 3.4.7 Basis (p 3.4.7-1 through 3.4.7-3; Rev 76)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1301C 1.09

Question Source:	Bank #	
	Modified Bank #	X
	New	

Question History: Last NRC Exam 2010 St Lucie NRC ILT

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

10 CFR Part 55 Content:	55.41	
	55.43	.2

Comments: SRO level since the Applicant must assess the plant conditions during a normal evolution and then select the appropriate section of the procedure and associated attachment with which to proceed.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	006 G2.4.30	
	Importance Rating	4.1	

Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator. (ECCS)

SRO Question #87

Which ONE of the following correctly states a condition and the required notification that must be made by the Shift Manager?

A. The NRC must be notified within eight (8) hours of a valid Safety Injection (SI) actuation.

B. State and County agencies must be notified within 30 minutes of a change in Protective Action Recommendations (PARs).

C. The Public Service Commission (PSC) must be notified within eight (8) hours of an Emergency Action Level (EAL) declaration.

D. RG&E Energy Control Center (ECC) must be notified within one hour of the Main Generator Voltage Regulator shifting to MANUAL.

Answer: A

Explanation:

A. CORRECT. In accordance with LS-AA-1110, SAF 1.7 “The licensee shall notify the NRC as soon as practical and in all cases, within eight hours of the occurrence of … any event or condition that results valid actuation of any of the systems listed in 50.72(b)(3)(iv)(B) except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.” “ECCS for PWRs including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.”

B. INCORRECT. Plausible since the State and County agencies would be notified of a change in PARs from the station. Incorrect since the change in PARs notification is required within 15 minutes, not 30 minutes according to EP-CE-114-100.

C. INCORRECT. Plausible since Public Service Commission must be notified of an EAL declaration. Incorrect since the notification of EAL declaration is required within one hour, not eight hours according to OPG-NOTIFICATION.

D. INCORRECT. Plausible since RG&E ECC must be notified if the Main Generator Voltage Regulator is NOT in automatic voltage control mode. Incorrect since the notification for the Main Generator Voltage Regulator NOT in automatic mode must be made within 30 minutes according to O-6.9.6, not one hour.

Technical Reference(s):
- LS-AA-1110, Safety, SAF-1.7, System Actuation not Including RPS (Rev 30)
- EP-CE-114-100, Emergency Notifications (p8; Rev 9)
- OPG-NOTIFICATION, Required Notifications to the PSC, Senior Management, Operations Management (p5 & 8; Rev 031)
- O-6.9.6, Transmission System Operations (p15; Rev 003)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAD04C 1.05

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41
55.43 .5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	007 A2.02	
	Importance Rating	3.2	

Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal pressure in the PRT

SRO Question #88

Given the following plant conditions:

- A malfunction occurs resulting in a reactor trip
- The operating crew enters the appropriate procedure

2 minutes later, the following conditions exist:

- RCS pressure is 2235 psig and stable
- PRT pressure is 12 psig and stable
- PRT level is 84% and stable
- PRZR PORV Outlet Temperature is 250°F and slowly lowering

Which ONE of the following procedures would be used to directly restore the above PRT conditions to normal?

A. E-0, Reactor Trip or Safety Injection

B. AP-PRZR.1, Abnormal Pressurizer Pressure

C. AR-F-1, PRT LIQUID HI TEMP 220°F

D. AR-F-9, PRT HI PRESS 5 PSI

Answer: **B**

Explanation:

A. INCORRECT. Plausible since the US will enter E-0 upon the reactor trip and E-0, Step 27 directs the operator to evaluate PRT conditions. Incorrect since E-0, Step 27 RNO does not contain guidance to correct PRT conditions that are not normal.

B. CORRECT. AP-PRZR.1, Step 16 directs the operator to “Check PRT Conditions” and since PRT pressure is NOT “APPROXIMATELY 1.5 PSIG AND STABLE”, Step 16.b RNO states “Open PRT vent, AOV-527 until PRT pressure is approximately 1.5 psig. IF PRT pressure will NOT lower, THEN open PRT drain valve, AOV-526, to reduce pressure until PRT pressure is approximately 1.5 psig.” Additionally, AP-PRZR.1, Step 16.a and 16.b RNO contains guidance to correct abnormal PRT level and temperature. According to UFSAR, Section 5.4.8.1 “The tank size is based on the requirement to condense and cool a discharge equivalent to 110% of the full power pressurizer steam volume. Assuming an initial tank water temperature of 125°F, the tank is capable of absorbing an amount of heat such that the final water temperature is no greater than 200°F.”

C. INCORRECT. Plausible since AR-F-1 contains guidance to add reactor makeup water and drain the PRT to reduce PRT temperature to normal and the Applicant may incorrectly determine that for the given plant conditions that the PRT temperature is greater than 220°F. Incorrect since AR-F-1 only contains guidance to restore PRT temperature and NOT pressure and the given plant conditions would not result in PRT temperature being greater than 200°F. Additionally, incorrect since according to UFSAR, Section 5.4.8.1 “The tank size is based on the requirement to condense and cool a discharge equivalent to 110% of the full power pressurizer steam volume. Assuming an initial tank water temperature of 125°F, the tank is capable of absorbing an amount of heat such that the final water temperature is no greater than 200°F.” Therefore, AR-F-1 entry conditions would not be met.

D. INCORRECT. Plausible since AR-F-9 would be applicable for the given plant conditions and Step 4.3 contains guidance for the Operator to lower PRT level to lower PRT pressure. Incorrect since AR-F-9 does NOT have guidance to restore PRT level or temperature, which would be elevated.

Technical Reference(s):	AP-PRZR.1, Abnormal Pressurizer Pressure (p10; Rev 01700)
	E-0, Reactor Trip or Safety Injection (p5 & 20; Rev 049)
	AR-F-1, PRT LIQUID HI TEMP 220°F (Rev 00601)
	AR-F-9, PRT HI PRESS 5 PSI (Rev 00800)

Proposed references to be provided to applicants during examination: None

Learning Objective: R1401C 1.07b, 1.09, 1.11

Question Source:	Bank #	X
	Modified Bank #	
	New	

Question History: Last NRC Exam ____________

Question Cognitive Level: Memory or Fundamental Knowledge ____________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____________

55.43 .5

Comments: SRO level since the Applicant is recalling what strategy or action is written into a plant procedure, including when the strategy or action is required.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	013 A2.06	
	Importance Rating	4.0	

Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent ESFAS actuation

SRO Question #89

Given the following plant conditions:

- Plant is performing a startup from a Refueling Outage
- Reactor power is 15% and stable
- The Operating crew is preparing to synchronize the Main Generator to the Grid
- An inadvertent Main Feedwater Isolation signal occurred
- Both S/G narrow range levels are 20% and lowering slowly

Which ONE of the following correctly completes the statements below?

1) In accordance with Technical Specification LCO 3.7.3, Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulating Valves (MFRVs), and Associated Bypass Valves, the MFRVs are ___________.

   AND

2) The US will direct the Board Operator(s) to ___________________ in accordance with AP-FW.1, Abnormal MFW Pump Flow or NPSH, in order to mitigate the event.

A. 1) OPERABLE
   2) trip the reactor

B. 1) OPERABLE
   2) place MFRVs in MANUAL and restore S/G water level to 52%

C. 1) INOPERABLE
   2) trip the reactor

D. 1) INOPERABLE

2) place MFRVs in MANUAL and restore S/G water level to 52%

Answer: **A**

Explanation:

A. CORRECT. In accordance with Technical Specification LCO 3.7.3 Basis "The MFRVs, associated bypass valves, and MFIVs are considered OPERABLE when isolation times are within limits and they can close on an isolation actuation signal. The isolation signal from reactor trip with T_{avg} < 554°F with the associated MFRV in auto is not a requirement for OPERABILITY." According to AP-FW.1, Step 3 RNO if MFW flow is NOT greater than steam flow OR if S/G levels are NOT stabilizing at or returning to program, the Operator is directed "IF MFW regulating valves NOT controlling in AUTO, THEN place affected S/G(s) MFW regulating valve and bypass valve in MANUAL and restore S/G level to 52%." This will be unsuccessful since the Feedwater Isolation signal causes both MFRVs and Bypass Valves to CLOSE and they can NOT be opened manually. Further Step 3 RNO directs the Operator "IF S/G level is less than 20% AND MFW flow is less than steam flow, THEN trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION."

B. INCORRECT. The first part is correct. The second part is plausible since this action is directed by AP-FW.1, Step 3 RNO and the Applicant may misinterpret the operation of the MFRV with a Feedwater Isolation signal present. Incorrect since the Feedwater Isolation signal will prevent manual operation of the MRFVs and Bypass Valves.

C. INCORRECT. The first part is plausible since the Applicant may misinterpret the OPERABILITY requirements for the MFRV and determine that since the Automatic Feedwater Isolation Actuation Logic and Actuation Relays is INOPERABLE that the associated MFRVs are INOPERABLE as well. Incorrect since the only OPERABILITY concern is that the MFRVs and bypass valves close on an isolation signal, which they did. The second part is correct.

D. INCORRECT. The first part is plausible since the Applicant may misinterpret the OPERABILITY requirements for the MFRV and determine that since the Automatic Feedwater Isolation Actuation Logic and Actuation Relays is INOPERABLE that the associated MFRVs are INOPERABLE as well. Incorrect since the only OPERABILITY concern is that the MFRVs and bypass valves close on an isolation signal, which they did. The second part is plausible since this action is directed by AP-FW.1, Step 3 RNO and the Applicant may misinterpret the operation of the MFRV with a Feedwater Isolation signal present. Incorrect since the Feedwater Isolation signal will prevent manual operation of the MRFVs and Bypass Valves.

Technical Reference(s):	Technical Specification LCO 3.7.3 Basis (p B3.7.3-3; Rev 58)
	AP-FW.1, Abnormal MFW Pump Flow or NPSH (p6; Rev 021)
	Technical Specification LCO 3.3.2 Basis (p B3.3.2-20; Rev 42)

Proposed references to be provided to applicants during examination: None

Learning Objective:	R4301C 1.07v; 1.12b	
Question Source:	Bank #	
	Modified Bank #	
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	064 G2.2.37	
	Importance Rating	4.6	

Ability to determine operability and/or availability of safety related equipment. (ED/G System)

SRO Question #90

In accordance with Technical Specification Section 3.8 Bases, which ONE of the following is a condition that requires immediately declaring 'B' EDG INOPERABLE?
(Other than those stated, no additional Operator actions have been taken)

A. 'B EDG starts and energizes Buses 16 and 17 in 15 seconds.

B. 'B' EDG Room 1B1 Supply Fan breaker trips and will NOT reset.

C. 'B' EDG Fuel Oil Day Tank sight glass level is discovered to be 9.9 inches

D. SW AOV-4599G and AOV-4599H, EDG B SW ISOLATION valves, will **NOT** open and Operators have opened V-4599J, EDG B SW ISOL BYPASS VLV.

Answer: A

Explanation:

A. CORRECT. According to Technical Specification LCO 3.8.1 Basis lists conditions under which a DG is considered OPERABLE. Item 'a' in that list states "The DG is capable of starting, accelerating to rated speed and voltage, and connecting to its respective 480 V safeguards buses on actuation of Loss of Power (LOP) instrumentation within 10 seconds."

B. INCORRECT. Plausible since EDG ROOM ventilation trains are part of EDG operability and the Applicant may misinterpret that the EDG must be declared INOPERABLE if any fans are not operable. Incorrect since Technical Specification LCO 3.8.1 Basis states "A ventilation train consisting of at least one of two fans and the associated ductwork and dampers is OPERABLE."

C. INCORRECT. Plausible since O-6.11, Attachment 7, SR 3.8.1.4 requires that each EDG Day Tank level be ≥ 10 inches by sight glass indication. Incorrect since Technical Specification LCO 3.8.1 Basis for SR 3.8.1.4 states "A level of 8.75 inches, as read on the local sight glass, achieves these requirements."

D. INCORRECT. Plausible since Technical Specification LCO 3.8.1 Basis lists conditions under which a DG is considered OPERABLE. Item ‘h’ in that list states “*Two service water AOVs to the diesel generator heat exchangers are OPERABLE (capable of opening) or, either one AOV is open or the manual bypass valve is open*.” The Applicant may misinterpret this requirement to be that either solenoid operated valve must be capable of automatically opening upon EDG start. Incorrect since the Technical Specification Basis allows for the manual bypass to be open to meet this requirement.

Technical Reference(s): Technical Specification LCO 3.8.1 Basis (Rev 74)

O-6.11, Surveillance Requirement/Routine Operation Check Sheet (p37; Rev 210)

Proposed references to be provided to applicants during examination: None

Learning Objective: R0801C 1.12b

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 .2

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	014 A2.04	
	Importance Rating	3.9	

Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Misaligned rod

SRO Question #91

Given the following plant conditions following a transient from 100% power:

- Reactor power is 90% and stable
- T_{AVG} = 569°F and stable
- Group counter Bank D = 205 steps
- MRPI Rod C7 Bank D = 188 steps
- MRPI Rod K7 Bank D = 188 steps
- MRPI Rods G3 and G11 Bank D = 200 steps
- Annunciator C-5, PPCS ROD SEQUENCE OR ROD DEVIATION, alarm lit
- Annunciator F-29, PPCS AXIAL OR QUADRANT POWER TILT, alarm lit
- Rods are believed to be trippable
- The crew has entered AP-RCC.2, RCC/RPI Malfunction

Which ONE of the following describes the next required action in accordance with AP-RCC.2?

A. Withdraw control rods to restore T_{AVG} to program

B. Realign rods C7 and K7 in accordance with ER-RCC.2, Restoring a Misaligned RCC

C. Shutdown per O-2.1, Plant Shutdown to Hot Shutdown

D. Perform applicable portions of STP-O-1, Rod Control System

Answer: **C**

Explanation:

A. INCORRECT. Plausible since control rod insertion is allowed for temperature control and the Applicant may misinterpret the procedural requirements and conclude that rod withdrawal is also allowed. Incorrect since with T_{AVG} low, turbine load would be adjusted (lowered) to raise T_{AVG} to match the T_{REF} value since rod withdrawal is NOT allowed.

B. INCORRECT. Plausible since this would be the action directed by AP-RCC.2 for a single misaligned control rod. Incorrect since AP-RCC.2, Step 10 RNO directs the Operator to initiate plant shutdown.

C. CORRECT. In accordance with Technical Specification LCO 3.1.4 Basis “Shutdown and control rod OPERABILITY is defined as being trippable such that the necessary negative reactivity assumed in the accident analysis is available. If a control rod(s) is discovered to be immovable but remains trippable and aligned, the control rod is considered to be OPERABLE. They are INOPERABLE since they don't meet the Surveillance Requirement for alignment. AP-RCC.2, Step 10 RNO directs the Operator “IF two or more rods are misaligned, THEN initiate plant shutdown (Refer to O-2.1, NORMAL SHUTDOWN TO HOT SHUTDOWN).”

D. INCORRECT. Plausible since AP-RCC.2, Step 13 directs verification of control rod operability per STP-O-1 during post rod recovery. Incorrect since this action would be valid only for a single misaligned rod. With 2 misaligned rods, a shutdown per O-2.1 is required.

Technical Reference(s): AP-RCC.2, RCC/RPI Malfunction (Rev 01400)

Technical Specification LCO 3.1.4 Basis (p B3.1.4-4; Rev 60)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3001C 1.06, 1.11b, 1.12b; RAP13C 2.01

Question Source: Bank # X

Modified Bank #

New

Question History: Last NRC Exam 2018 Ginna ILT

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 .5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	029 G2.1.23	
	Importance Rating	4.4	

Ability to perform specific system and integrated plant procedures during all modes of plant operation. (Containment Purge System)

SRO Question #92

Given the following plant conditions:

- The plant is in a Refueling Outage
- Reactor Head lift will occur in 30 minutes
- The US is directed to establish the required CNMT Purge alignment in accordance with S-23.2.2, Containment Purge Procedure

Which ONE of the following describes:

(1) the preferred CNMT Purge system lineup for the upcoming Reactor Head lift;

AND

(2) the reason for the alignment, under these conditions?

A. (1) ONE Purge Exhaust Fan in operation, NO Purge Supply Fans in operation
(2) provide for adequate Containment cooling

B. (1) ONE Purge Exhaust Fan in operation, NO Purge Supply Fans in operation
(2) minimize radioactive release to the environment

C. (1) TWO Purge Exhaust Fans in operation, ONE Purge Supply Fan in operation
(2) provide for adequate Containment cooling

D. (1) TWO Purge Exhaust Fans in operation, ONE Purge Supply Fan in operation
(2) minimize radioactive release to the environment

Answer: **D**

Explanation:

A. INCORRECT. The first part is plausible since this is an acceptable Purge system alignment and the Applicant may misinterpret that this alignment will be satisfactory during the Reactor Head lift. Incorrect since S-23.2.2 specifically states the preferred alignment is 2 Exhaust and 1 Supply fan in operation. The second part is plausible since a secondary purpose of the CNMT Purge System is to circulate fresh air into CNMT for cooling. Incorrect since the primary purpose for this evolution would be to ensure that CNMT pressure remains negative to minimize the potential for radioactive release to the environment.

B. INCORRECT. The first part is plausible since this is an acceptable Purge system alignment and the Applicant may misinterpret that this alignment will be satisfactory during the Reactor Head lift. Incorrect since S-23.2.2 specifically states the preferred alignment is 2 Exhaust and 1 Supply fan in operation. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since a secondary purpose of the CNMT Purge System is to circulate fresh air into CNMT for cooling. Incorrect since the primary purpose for this evolution would be to ensure that CNMT pressure remains negative to minimize the potential for radioactive release to the environment.

D. CORRECT. In accordance with S-23.2.2, Precaution 4.4 “During potential high airborne evolutions such as Reactor Head lift **OR** Reactor Cavity decontamination, the preferred Containment Purge system lineup is as follows: <u>Two</u> Purge Exhaust Fans in operation / <u>ONE</u> Purge Supply Fan in operation.” Additionally, Precaution 4.3 states “Combinations of Purge Supply and Exhaust fans should be in service to ensure containment is at a negative pressure.” This ensures that any potential radioactivity that becomes airborne will NOT be released into the environment. UFSAR, Section 3.5.1.1.2 states “Systems whose failure may result in release of unacceptable amounts of radioactivity are as follows: Spent fuel pool cooling and cleanup system, Sampling system, Waste disposal system, Containment purge system, and Instrument and service air systems.”

Technical Reference(s): S-23.2.2, Containment Purge Procedure (p6; Rev 057)
UFSAR, Section 3.5.1.1.2 (Rev 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: R2201C 1.09

Question Source: Bank # X
Modified Bank # ______
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:	55.41	
	55.43	.4

Comments: The first part of the question is testing specific CNMT Purge alignment required during Reactor Head lift and can NOT be answered solely on system knowledge as it additionally requires knowledge of refueling limitations which is SRO knowledge. Additionally, the second part of the question is testing UFSAR basis for the required system alignment. Therefore, the question meets the requirements for an SRO question.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	034 K4.01	
	Importance Rating	3.4	

Knowledge of design feature(s) and/or interlock(s) which provide for the following: Fuel protection from binding and dropping (Fuel Handling Equipment System)

SRO Question #93

Given the following plant conditions:

- Plant is in MODE 6
- Core offload is in progress
- A fuel assembly becomes stuck on the lower core support plate
- The overload circuit on the Fuel Hoist actuates at 2250 pounds

Which ONE of the following correctly completes the statements below?

1) The Refueling SRO will enter __________________ to mitigate the fuel assembly becoming stuck during withdrawal from the core.

AND

2) The next action taken by the Refueling crew is ______________________.

A. 1) RF-301, Refueling Operations (Offload, Shuffle, Reload)
2) raise the Fuel Hoist overload setting to 2500 pounds

B. 1) RF-301, Refueling Operations (Offload, Shuffle, Reload)
2) remove Fuel Assemblies adjacent to the affected (stuck) Fuel Assembly

C. 1) RF-601, Fuel Handling Accident Instructions
2) raise the Fuel Hoist overload setting to 2500 pounds

D. 1) RF-601, Fuel Handling Accident Instructions
2) remove Fuel Assemblies adjacent to the affected (stuck) Fuel Assembly

Answer: **D**

Explanation:

A. INCORRECT. The first part is plausible since RF-301, Attachment 1 contains guidance for the Refueling crew when a fuel assembly becomes stuck during insertion. Incorrect since RF-601 contains the required actions for a stuck fuel assembly during offload. The second part is plausible since RF-601, Attachment 2 contains a step to raise the overload setting and attempt one more removal of the stuck assembly. Incorrect since RF-601, Attachment 2 CAUTION states "**WHEN** attempting to remove fuel assembly, exert a **MAXIMUM** force of 2250 pounds **OR** the preset overload limit, as indicated on the fuel hoist load cell readout."

B. INCORRECT. The first part is plausible since RF-301, Attachment 1 contains guidance for the Refueling crew when a fuel assembly becomes stuck during insertion. Incorrect since RF-601 contains the required actions for a stuck fuel assembly during offload. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since RF-601, Attachment 2 contains a step to raise the overload setting and attempt one more removal of the stuck assembly. Incorrect since RF-601, Attachment 2 CAUTION states "**WHEN** attempting to remove fuel assembly, exert a **MAXIMUM** force of 2250 pounds **OR** the preset overload limit, as indicated on the fuel hoist load cell readout."

D. CORRECT. According to RF-601, Step 1.1.2 "These instructions describe three (3) separate accidents involving fuel handling: A Fuel Assembly becomes stuck inside the reactor vessel or inside the transfer basket." RF-601, Attachment 2, Step 3.2.5 states "**IF** the Fuel Assembly remains stuck on the lower core support plate, **THEN PERFORM** the following: **REMOVE AND TRANSFER** Fuel Assemblies adjacent to the affected Fuel Assembly."

Technical Reference(s): RF-601, Fuel Handling Accident Instructions (p4, 10-11; Rev 003)

RF-301, Refueling Operations (Offload, Shuffle, Reload) (p29 – 36; Rev 015)

Proposed references to be provided to applicants during examination: None

Learning Objective: R3701C 1.11

Question Source:
Bank # ____
Modified Bank # ____
New X

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:	55.41	
	55.43	.7

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	G2.1.37	
	Importance Rating	4.6	

Knowledge of procedures, guidelines, or limitations associated with reactivity management.

SRO Question #94

Given the following plant conditions:

- Plant is starting up in accordance with O-1.2, Plant Startup from Hot Shutdown to Full Load
- Reactor power is stable at 1%
- A Reactivity Management SRO is stationed

Which ONE of the following correctly completes the statements below?

(1) During low power operations, the HCO can NOT exceed ________ continuous steps of outward rod motion without stopping to check all available indications of power.

AND

(2) The _________ authorizes the reactivity management plan.

A. (1) five (5)
(2) Reactor Engineering Manager

B. (1) twelve (12)
(2) Reactor Engineering Manager

C. (1) five (5)
(2) Senior Reactor Operator (SRO)

D. (1) twelve (12)
(2) Senior Reactor Operator (SRO)

Answer: **C**

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since the Reactor Engineering Manager is responsible for providing oversight, direction and approval of the reactivity management plan. Incorrect since the SRO authorizes the reactivity management plan.

B. INCORRECT. The first part is plausible since twelve steps is the MRPI transition points for rod position indication and the Applicant may misinterpret that this would be the required stopping point for monitoring power indications. Incorrect since O-1.2, Attachment 9 directs that only five continuous steps are allowed without stopping and checking power indications. The second part is plausible since the Reactor Engineering Manager is responsible for providing oversight, direction and approval of the reactivity management plan. Incorrect since the SRO authorizes the reactivity management plan.

C. CORRECT. In accordance with O-1.2, Attachment 9, Step 2.0 "**WHEN** as Low Power Operations above the Point of Adding Heat (POAH), **THEN DO NOT** exceed five continuous steps of outward motion without stopping to check all available indications of power." Additionally, OP-AP-300-1003, Step 3.7 states "A Senior Reactor Operator (SRO) provides Operations review and authorization of the ReMA Package, activates the ReMA for performance, and terminates the ReMA, as required."

D. INCORRECT. The first part is plausible since twelve steps is the MRPI transition points for rod position indication and the Applicant may misinterpret that this would be the required stopping point for monitoring power indications. Incorrect since O-1.2, Attachment 9 directs that only five continuous steps are allowed without stopping and checking power indications. The second part is correct.

Technical Reference(s): O-1.2, Plant Startup from Hot Shutdown to Full Load (p148; Rev 220)

OP-AP-300-1003, PWR Reactivity Maneuver (p3, 10 & 11; Rev 11)

Proposed references to be provided to applicants during examination: None

Learning Objective: ROP01C 1.01; RAD29C 1.03

Question Source:		
	Bank #	X
	Modified Bank #	
	New	

Question History: Last NRC Exam 2013 Farley ILT

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

10 CFR Part 55 Content:	55.41	
	55.43	.6

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	G2.1.41	
	Importance Rating	3.7	

Knowledge of the refueling process.

SRO Question #95

Which ONE of the following would be the FIRST core alteration activity requiring direct supervision by the Refueling SRO?

A. Reactor Vessel Head lift

B. Control Rod Drive Shaft unlatch checks

C. Upper Internals lift

D. Moving the first fuel assembly

Answer: A

Explanation:

A. CORRECT. According to O-15.1, Precaution 4.11 “All core alterations **SHALL** be directly supervised by either a Licensed Senior Reactor Operator OR Senior Reactor Operator limited to fuel handling who has **NO** other concurrent responsibilities during this operation.” The Reactor Vessel Head lift is the first core alteration to be performed according to RF-100. Additionally, RF-405, Precaution 4.1 states “Reactor Vessel Head lift is considered a core alteration due to the possibility to lift a Control Rod Assembly.”

B. INCORRECT. Plausible since according to RF-406, Precaution 4.1 states “Control Rod Drive Shaft unlatching **AND** Upper Internals removal is considered a core alteration due to the possibility to lift a Control Rod Assembly.” The Applicant may misinterpret that lifting the Reactor Vessel Head does NOT constitute a core alteration. Incorrect since lifting the Reactor Vessel head is a core alteration and is performed first in the sequence of refueling.

C. INCORRECT. Plausible since according to RF-406, Precaution 4.1 states “Control Rod Drive Shaft unlatching **AND** Upper Internals removal is considered a core alteration due to the possibility to lift a Control Rod Assembly.” The Applicant may misinterpret that lifting the Reactor Vessel Head and unlatching Control Rod Shafts do NOT constitute a core alteration. Incorrect since lifting the Reactor Vessel head is a core alteration and is performed first in the sequence of refueling.

D. INCORRECT. Plausible since the Applicant may misinterpret that the Refueling SRO is only required to directly supervise the actual movement of fuel assemblies. Incorrect since the Refueling SRO is required to directly supervise ALL core alterations and lifting the Reactor Vessel head is a core alteration and is performed first in the sequence of refueling.

Technical Reference(s):
- O-15.1, Administrative Requirement Checklist for Entry to MODE 6 and Refueling Conditions (p7; Rev 037)
- RF-405, Reactor Vessel Head Removal (p5; Rev 006)
- RF-406, Control Rod Drive Shaft Unlatching and Upper Internals Removal (p5; Rev 008)

Proposed references to be provided to applicants during examination: None

Learning Objective: RRF02C 4.00

Question Source:
- Bank #: X
- Modified Bank #:
- New:

Question History: Last NRC Exam 2016 Diablo Canyon ILT

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

10 CFR Part 55 Content:
- 55.41:
- 55.43: .6

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	G2.2.5	
	Importance Rating	3.2	

Knowledge of the process for making design or operating changes to the facility.

SRO Question #96

Given the following:

- The plant is at 100% power
- A temporary sump pump is being installed in the Screenhouse in direct support of normally scheduled maintenance replacement of the Screenhouse Circ Water Bay Sump Pumps
- The replacement will be performed under a Work Order and a Temporary Configuration Change Package (TCCP), and is expected to take less than 60 days

For the given situation:

(1) Whose approval is required for this temporary change installation in the plant;

AND

(2) Will a 10CFR50.59 screening be required for this activity?

A. (1) Unit Supervisor (US) ONLY
(2) 10CFR50.59 screening IS required

B. (1) Unit Supervisor (US) ONLY
(2) 10CFR50.59 screening is NOT required

C. (1) Unit Supervisor (US) AND Maintenance Manager
(2) 10CFR50.59 screening IS required

D. (1) Unit Supervisor (US) AND Maintenance Manager
(2) 10CFR50.59 screening is NOT required

Answer: **B**

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since LS-AA-104-1000 requires that “10CFR50.59 should be applied to temporary changes proposed as compensatory measures to address degraded or non-conforming conditions”. Incorrect since temporary changes installed in support of maintenance activities are subject to 10CFR50.65 if removed within 90 days.

B. CORRECT. According to OP-AA-108-101, the Operations US has the responsibility for “authorizing installation and removal of TCCs, controlling TCC tag numbers and tags, and entering required information into the TCC Tracking Log.” According to LS-AA-104-1000 “Temporary changes to the facility are subject to 10CFR50.59 in the same manner as permanent changes to the facility. The exception to this is temporary changes that are installed in support of maintenance activities. These temporary changes are subject to 10CFR50.65 (Maintenance Rule) rather than 10CFR50.59 if the temporary change is removed within 90 days. However, if the temporary change in support of maintenance is to be in effect during at-power operations for more than 90 days, then the temporary change is subject to 10CFR50.59.”

C. INCORRECT. The first part is plausible since according to CC-AA-112 “the Maintenance Manager is responsible for ensuring that maintenance/contract personnel are familiar with CC-AA-112 and that the TCC is installed and removed per the approved TCC requirements.” The second part is plausible since LS-AA-104-1000 requires that “10CFR50.59 should be applied to temporary changes proposed as compensatory measures to address degraded or non-conforming conditions”. Incorrect since temporary changes installed in support of maintenance activities are subject to 10CFR50.65 if removed within 90 days.

D. INCORRECT. The first part is plausible since according to CC-AA-112 “the Maintenance Manager is responsible for ensuring that maintenance/contract personnel are familiar with CC-AA-112 and that the TCC is installed and removed per the approved TCC requirements.” The second part is correct.

Technical Reference(s):	OP-AA-108-101, Control of Equipment and System Status (p3; Rev 15)
	LS-AA-104-1000, Exelon 50.59 Resource Manual (p5-7; Rev 14)
	CC-AA-112, Temporary Configuration Changes (p3; Rev 29)

Proposed references to be provided to applicants during examination: None

Learning Objective: RAD32C 1.06

Question Source:	Bank #	X
	Modified Bank #	
	New	

Question History: Last NRC Exam 2012 Ginna ILT

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

10 CFR Part 55 Content:	55.41	
	55.43	.3

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	G2.2.21	
	Importance Rating	4.1	

Knowledge of pre- and post-maintenance operability requirements.

SRO Question #97

Which ONE of the following correctly completes the statements below in accordance with Technical Specifications?

1) A Reactor Protection System (RPS) relay that has been declared INOPERABLE and placed in a tripped condition to comply with Technical Specification Required Actions may be returned to service ______________ to demonstrate its OPERABILITY.
2) This allowance ____________ applicable to restoring equipment to service to demonstrate the operability of OTHER equipment.

A. 1) under administrative control
   2) is NOT

B. 1) after performance of a Risk Assessment
   2) is NOT

C. 1) under administrative control
   2) is also

D. 1) after performance of a Risk Assessment
   2) is also

Answer: C

Explanation:

A. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret that the requirement for returning the relay to service only applies to that specific relay. Incorrect since LCO 3.0.5 allows for the testing of other equipment.

B. INCORRECT. The first part is plausible since Technical Specification LCO 3.0.4.b allows entry into a MODE when an LCO is not met after performance of a risk assessment addressing inoperable systems and components and the Applicant may misinterpret that this applies to returning INOPERABLE equipment to service. Incorrect since LCO 3.0.5 contains the requirements. The second part is plausible since the Applicant may misinterpret that the requirement for returning the relay to service only applies to that specific relay. Incorrect since LCO 3.0.5 allows for the testing of other equipment.

C. CORRECT. In accordance with Technical Specification LCO 3.0.5 “Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to determine OPERABILITY.”

D. INCORRECT. The first part is plausible since Technical Specification LCO 3.0.4.b allows entry into a MODE when an LCO is not met after performance of a risk assessment addressing inoperable systems and components and the Applicant may misinterpret that this applies to returning INOPERABLE equipment to service. Incorrect since LCO 3.0.5 contains the requirements. The second part is correct.

Technical Reference(s): Technical Specification LCO 3.0.5 (Amendment 126)
Technical Specification LCO 3.0.5 Basis (Rev 82)

Proposed references to be provided to applicants during examination: None

Learning Objective: RTS03C 1.03

Question Source:	Bank #	X
	Modified Bank #	
	New	

Question History: Last NRC Exam 2015 Browns Ferry ILT

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

10 CFR Part 55 Content:	55.41	
	55.43	.2

Comments: K/A is matched since LCO 3.0.5 provides allowances to return equipment to service for the purpose of performing operability testing.

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	G2.3.13	
	Importance Rating	3.8	

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

SRO Question #98

Given the following plant conditions:

- An Operator has been directed by the US to enter the Auxiliary Building to isolate a safety-related system during implementation of AP-RCS.1, Reactor Coolant Leak
- Radiological surveys taken on piping and valves are:
  - Valve 1: 100 mrem/hr at 30 cm
  - Valve 2: 1100 mrem/hr at 30 cm
- The Shift Manager (SM) is monitoring EALs, but none have been declared
- A Radiation Protection (RP) brief for entry into the area has been completed

Based on the given plant conditions, which ONE of the following correctly states:

(1) the radiological posting required for this area;
AND
(2) the additional approval required, if any, prior to entry by the Operator?

A. (1) High Radiation Area
(2) Shift Manager

B. (1) High Radiation Area
(2) NO additional approval required

C. (1) Locked High Radiation Area
(2) Shift Manager

D. (1) Locked High Radiation Area
(2) NO additional approval required

Answer: **D**

Explanation:

A. INCORRECT. The first part is plausible since one of the two radiological readings are less than 1000 mrem/hr and the Applicant may misinterpret the requirements for posting the area. Incorrect since one reading is greater than 1000 mrem/hr; therefore, the area is required to be posted as a Locked HRA. The second part is plausible since the SM becomes the Shift ED during emergency events and the Applicant may misinterpret the plant conditions and conclude that SM permission is required for the exposure during the emergency. Incorrect since dose rates are less than 500 Rem/hr, no additional approvals are required.

B. INCORRECT. The first part is plausible since one of the two radiological readings are less than 1000 mrem/hr and the Applicant may misinterpret the requirements for posting the area. Incorrect since one reading is greater than 1000 mrem/hr; therefore, the area is required to be posted as a Locked HRA. The second part is correct.

C. INCORRECT. The first part is correct. The second part is plausible since the SM becomes the Shift ED during emergency events and the Applicant may misinterpret the plant conditions and conclude that SM permission is required for the exposure during the emergency. Incorrect since dose rates are less than 500 Rem/hr, no additional approvals are required.

D. CORRECT. Technical Specification, Section 5.7.2 requires that a "High Radiation Area with radiation levels > 1000 mrem/hr at a distance of 30 cm shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Supervisor on duty or radiation protection supervision." RP-AA-460-003, Section 5.9.1 states "For entry into a Locked High Radiation Area, where deep dose equivalent rates are greater than or equal to 1,000 mrem per hour at 30 centimeters but less than 500 rads in 1 hour at 1 meter from the source of radiation or from any surface that the radiation penetrates, no additional approvals are required prior to entry." The only required approvals would be from the RP brief and RWP.

Technical Reference(s):	Technical Specification, Section 5.7 (Amendment 80)
	RP-AA-460-003, Access to HRAs/LHRAs and Contaminated Areas in Response to a Potential or Actual Emergency (Rev 10)
	EP-AA-113, Personnel Protective Actions (p6-7; Rev 15)

Proposed references to be provided to applicants during examination: None

Learning Objective: RTS00C, 4.01

Question Source:	Bank #	
	Modified Bank #	X
	New	
Question History:	Last NRC Exam	2017 Point Beach ILT
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	.4

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	G2.4.22	
	Importance Rating	4.4	

Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

SRO Question #99

In accordance with A-503.1, Emergency and Abnormal Operating Procedures Users Guide, in which ONE of the following situations will the crew remain in a “Lower Priority” Emergency Operating Procedure (EOP) when Functional restoration (FR) Procedure entry conditions are met?

A. ONLY after declaring a 10CFR50.54(x) to document the Departure from the License Condition.

B. When a NOTE or CAUTION within the Lower Priority EOP specifies precedence over the Higher Priority FR procedure.

C. ONLY when the Higher Priority FR Procedure cannot be performed as written and the Shift Manager authorizes a Procedure Deviation to remain in a Lower Priority EOP.

D. While performing a RED Path FR Procedure, a higher priority RED Path is diagnosed. The crew should remain in the current procedure until complete and THEN transition ONLY if the RED Path condition still exists.

Answer: **B**

Explanation:

A. INCORRECT. Plausible since a Departure from License Condition allows the Operating crew to take action contrary to procedural guidance to protect the health and safety of the public. Incorrect since this only allows for deviations from License Conditions and Technical Specifications and NOT to perform a procedure of lower priority.

B. CORRECT. In accordance with A-503.1, Section 5.2.P.4.a “Red or Orange challenges to a Critical Safety function take priority over all E, ES, ER, and ECA procedures except as follows: (1) When performing E-0, Reactor Trip or Safety Injection, monitoring of CSFSTs (and therefore transition to FR procedures) should not be initiated before either reaching the E-0 step which directs monitoring CSFSTs (Step 19), or before being transitioned out of E-0 (Steps 15 – 17); (2) When performing ECA-0, Loss of All AC, series of procedures, transition to FR procedures should not occur before explicitly allowed by the ECA-0 series procedure; (3) When performing ES-1.3, Transfer to Cold Leg Recirculation, transition to FR procedures should not occur until explicitly allowed by ES-1.3; (5) When performing ECA-1.3, Response to Sump B Blockage." The associated procedures contain a NOTE that "FR procedures should not be implemented."

C. INCORRECT. Plausible since A-503.1, Section 5.3.A.1 states "A procedural deviation should only be considered when all three of the following are met: a. As-written procedural guidance is deficient due to current plant or equipment conditions; b. Insufficient time exists to implement the normal procedure change policy; c. An immediate need exists to prevent or to minimize one or more of the following: Injury to personnel, Damage to plant equipment, Threat to health and safety of the public (e.g. radioactive release)." Incorrect since a Procedure Deviation only applies to the written guidance within the current procedure and would not allow the Operating crew to perform a procedure of lower priority.

D. INCORRECT. Plausible since A-503.1 states that once implemented a FR procedure is to be completed and the Applicant may misinterpret this requirement to mean that the current FR procedure must be completed even though a higher priority FR procedure is diagnosed and required. Incorrect since A-503.1, Section 5.2.P.4.c.(2) states "If during the performance of an FR procedure from a RED path condition, a higher priority RED path is determined, the FR procedure in use is suspended, and the higher priority condition procedure is addressed. Once the higher priority procedure is complete, the suspended procedure for the lower priority condition is to be completed."

Technical Reference(s):	A-503.1, Emergency and Abnormal operating Procedures Users Guide (p34, 40-41; Rev 053)
	ECA-0.0, Loss of All AC Power (p3; Rev 044)
	ES-1.3, Transfer to Cold Leg Recirculation (p3; Rev 047)
	ECA-1.3, Response to Sump B Blockage (p3; Rev 004)

Proposed references to be provided to applicants during examination: None

Learning Objective: REP50C 1.28, 1.12, 1.26

Question Source:	Bank #	X
	Modified Bank #	
	New	

Question History: Last NRC Exam 2019 Wolf Creek ILT

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

10 CFR Part 55 Content:	55.41	
	55.43	.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	G2.4.40	
	Importance Rating	4.5	

Knowledge of SRO responsibilities in emergency plan implementation.

SRO Question #100

Which ONE of the following correctly completes the statements below?

1) The accountability process of personnel within the Protected Area must be completed within a MAXIMUM of ____________ from the time a Site Area Emergency is declared.
2) The Shift Emergency Director May/May NOT conduct Accountability at the ALERT level prior to TSC activation.

A. 1) 60 minutes
   2) may

B. 1) 60 minutes
   2) may NOT

C. 1) 30 minutes
   2) may

D. 1) 30 minutes
   2) may NOT

Answer: **C**

Explanation:

A. INCORRECT. The first part is plausible since the Emergency Director (Station or Corporate) is required to declare an event within 15 minutes and notify the NRC within 60 minutes of declaration. Incorrect since accountability of personnel is required within 30 minutes of declaring a SAE or GE. The second part is correct.

B. INCORRECT. The first part is plausible since the Emergency Director (Station or Corporate) is required to declare an event within 15 minutes and notify the NRC within 60 minutes of declaration. Incorrect since accountability of personnel is required within 30 minutes of declaring a SAE or GE. The second part is plausible since the Applicant may misinterpret that the requirement for Site Accountability can only be performed at the SAE or GE levels. Incorrect since Site Accountability can be performed at the ALERT level at the discretion of the Emergency Director.

C. CORRECT. According to EP-AA-113, Section 4.1.1.C "Once initiated, accountability is required to be performed (i.e., the names of missing persons identified by security and the number of missing provided to the Station Emergency Director) within 30 minutes of the declaration of a Site Area Emergency or General Emergency." Additionally, EP-AA-113, Step 4.1.1.1 states "Accountability may be conducted at the ALERT level at the discretion of the Station Emergency Director, or Shift Manager (Shift Emergency Director) prior to TSC activation."

D. INCORRECT. The first part is correct. The second part is plausible since the Applicant may misinterpret that the requirement for Site Accountability can only be performed at the SAE or GE levels. Incorrect since Site Accountability can be performed at the ALERT level at the discretion of the Emergency Director.

Technical Reference(s): EP-AA-113, Personnel Protective Actions (p3-4; Rev 15)

Proposed references to be provided to applicants during examination: None

Learning Objective: RSC02C 13.00 & 14.00

Question Source: Bank # ____
Modified Bank # X
New ____

Question History: Last NRC Exam 2014 Robinson ILT

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

10 CFR Part 55 Content: 55.41 ____
55.43 .5

Comments: